SECTION 14

SAFETY ANALYSIS

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LIST OF APPENDICES

SECTION 14 SAFETY ANALYSIS

14.1 SAFETY ANALYSIS

14.1.1 Safety Analysis

This section evaluates the safety aspects of the plant and demonstrates that the plant can be operated safely and that exposures from postulated accidents do not exceed the criteria in 10CFR50.67. In previous sections of the safety analysis report, the structures, system and components important to safety were described and evaluated for their susceptibility to malfunctions and failures. In this section, the effects of anticipated transients and component failures are examined to determine their consequences and to demonstrate the capability built into the plant systems to control or accommodate such failures and transients. 01406407

The safety analysis is divided into three different behavior categories:

a. Core and Coolant Boundary Protection Analysis, Section 14.4

b. Standby Safety Features Analysis, Section 14.5

c. Rupture of a Reactor Coolant Pipe, Sections 14.6 and 14.7

With the exception of the locked rotor accident, the transients described in Section 14.4 are accommodated with, at most, a reactor shutdown with the unit being capable of returning to operation after corrective action. In addition, these transients have no offsite radiation consequences. The specific accidents described in Section 14.4 are:

Section 14.4.1	Uncontrolled RCCA Withdrawal from a Subcritical Condition (Condition II)
Section 14.4.2	Uncontrolled RCCA Withdrawal at Power (Condition II)
Section 14.4.3	RCCA Misalignment (Condition II)
Section 14.4.4	Chemical and Volume Control System Malfunction (Condition II)
Section 14.4.5	Start-Up of an Inactive Reactor Coolant Loop (Condition II)
Section 14.4.6	Excessive Heat Removal Due to Feedwater System Malfunctions (Condition II)
Section 14.4.7	Excessive Load Increase Incident (Condition II)

Section 14.4.8 Loss of Reactor Coolant Flow

Flow Coastdown Accidents
One Pump (Condition II)
Both Pumps (Condition III)

Locked Pump Rotor (Condition IV)

Section 14.4.9 Loss of External Electrical Load (Condition II)

Section 14.4.10 Loss of Normal Feedwater (Condition II)

Section 14.4.11 Loss of All AC Power to the Station Auxiliaries (Condition II)

NOTE:

Conditions as used here are per the ANS 51.1-1973 system of plant condition classification. Refer to Table 14.1-1 for definition and acceptance criteria. Per ANS 51.1, the two pump flow coastdown accident is categorized as a condition III event. However, the more conservative condition II acceptance criteria are applied.

The accidents described in Section 14.5 are more severe and may cause release of radioactive materials to the environment. These accidents are classified as limiting faults and are not expected to occur. Adequate provisions have been included in the design of the plant and its standby engineered safety features to limit potential exposure of the public to well below the limits of 10CFR50.67 for situations which could conceivably involve uncontrolled releases of radioactive materials to the environment. The situations which have been considered are: 01406407

Section 14.5.1 Fuel Handling Accidents (Condition IV)

Section 14.5.2 Accidental Release of Radioactive Liquids (Condition IV)

Section 14.5.3 Accidental Release - Waste Gas (Condition IV)

Section 14.5.4 Steam Generator Tube Rupture (Condition IV)

Section 14.5.5 Rupture of a Steam Pipe (Condition IV)

Section 14.5.6 Rupture of a Control Rod Drive Mechanism Housing (RCCA) Ejection) (Condition IV)

The accidents presented in Sections 14.6 and 14.7, the rupture of a reactor coolant pipe (large break and small break, respectively), is the design basis accident and is the primary basis for the design of engineered safety features. It is shown that the acceptance criteria in 10CFR Part 50, Appendix K are satisfied for these accidents.

The analysis in Section 14.9, Environmental Consequences of a Loss of Coolant Accident, presents the off site dose analysis for the large break loss of coolant accident scenario. This section shows that the consequences of this limiting accident are within the criteria in 10CFR50.67. 01406407

14.1.2 Other Analysis

The transients presented in Section 14.8, Anticipated Transients Without Scram, are not design bases events for Prairie Island. This section was added to the safety analysis report as part of the response to 10 CFR Part 50.62.

The analysis in Section 14.10, Long Term Cooling Following a LOCA, presents information relative to the performance requirements for the ECCS in the post-LOCA recirculation mode of operation.

14.1.3 Replacement Steam Generator Designation

The Unit 1 steam generators were designed by Framatome ANP (Framatome), before the Unit 2 steam generators were replaced the Framatome company name changed to AREVA NP (AREVA), therefore the Unit 2 steam generators are AREVA 56/19 model and the Unit 1 steam generators are Framatome 56/19 model. These are the same model steam generators which are commonly referred to as the Replacement Steam Generators (RSG).

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14.3 TRANSIENT ANALYSIS

14.3.1 Calculation Methods and Input Parameters

This section gives an overview of the calculational methods and input parameters used for Prairie Island safety and accident analysis. The core reload safety evaluation methodology is described in Reference 2. Descriptions of the transient-specific analysis methods are provided in the respective USAR sections.

Operating Parameters

To ensure conservative predictions of system responses with respect to the transient acceptance criteria, conservative assumptions are applied.

For most transients that are analyzed for DNB concerns, the revised thermal design procedure (RTDP) methodology (Reference 4) is employed. With this methodology, nominal values are assumed for the initial conditions of power, temperature, pressure, and flow, and the corresponding uncertainty allowances are accounted for statistically in defining the departure from nucleate boiling ratio (DNBR) design limit. Note that the nominal RCS flow assumed in RTDP transient analyses is the minimum measured flow (MMF) of 183,400 gpm.

The following rated values and conservative steady state errors are assumed in RTDP analyses:

NSSS Power (includes 7 MWt RCP heat) =	1684 MWt ± 0.5% Power measurement error
Vessel Average Temperature	560.0 with 4°F and -0.5°F Bias for deadband and measurement error
Primary Coolant System Pressure	2250 ± 60 psia for steady state fluctuation and measurement errors
Primary Coolant System Flow Uncertainty	3%

For transient analyses that are not DNB limited, or for which RTDP is not employed, the initial conditions are obtained by applying the maximum, steady-state uncertainties to the nominal values in the most conservative direction; this is known as Standard Thermal Design Procedure (STDP) or non-RTDP. In these analyses, the RCS flow is assumed to be equal to the TDF. The following rated values and conservative steady-state errors were considered in the analyses:

NSSS Power	=	1690 MWt includes applicable for calorimetric error.
Vessel Average Temperature	=	560.0 ± 4°F for deadband and measurement error.
Primary Coolant System Pressure	=	2250 -60/+40 psia for steady-state fluctuation and measurement errors.

Tables 14.3-1 and 14.3-2 summarize initial conditions and computer codes used in the non-LOCA accident analyses, and identify which DNB transients were analyzed using the RTDP.

Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of control rods and operating instructions. The radial peaking factor ($F_{\Delta H}$) and the total peaking factor (F_Q) characterize the power distribution. The peaking factor limits are provided in the Technical Specifications.

For transients that may be DNB limited, the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated in Figure 14.3-1. All transients that may be DNB limited are assumed to begin with $F_{\Delta H}$ consistent with the initial power level defined in the Technical Specifications. The radial and axial power distributions are input to the VIPRE code, which is used to perform the DNBR calculations.

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For transients that may be overpower limits, the total peaking factor (F_Q) is of importance. These transients are assumed to begin with plant conditions, including power distributions that are consistent with reactor operation as defined in the Technical Specifications.

For overpower transients that are slow with respect to the fuel rod thermal time constant (for example, the Chemical and Volume Control System malfunction that results in a decrease in the boron concentration of the reactor coolant system, lasting many minutes), fuel rod thermal evaluations are performed. For overpower transients that are fast with respect to the fuel rod thermal time constant (for example, the uncontrolled rod cluster control assembly (RCCA) bank withdrawal from subcritical and the RCCA ejection incidents that result in a large power rise over a few seconds), a detailed fuel heat transfer calculation is performed. The fuel rod thermal time constant is a function of system conditions, fuel burnup, and rod power, a typical value at beginning-of-life for high power rods is approximately five seconds.

Reactivity Coefficients

The transient response of the reactor system is dependent on reactivity feedback effects, in particular the isothermal temperature coefficient and the Doppler power coefficient.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values, whereas in the analysis of other events conservatism requires the use of small reactivity coefficients values. Some analyses, such as loss of coolant from cracks or ruptures in the Reactor Coolant System, do not depend on reactivity feedback effects. The justification for use of conservatively large versus small reactivity coefficient values is treated on an event-by-event basis. In some cases, conservative combinations of parameters are used to bound the effects of core life, although these combinations may represent unrealistic situations.

Reactor Protection System

A reactor trip signal acts to open the two series trip breakers feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanism to release the control rods, which then fall by gravity into the core. There are various instrumentation delays associated with each tripping function including delays in signal actuation, in opening the trip breakers and in the release of the rods by the mechanisms. The total delay to the trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. The time delay and setpoint assumed for each tripping function used in the analysis are as follows:

Reactor Trip Function	Setpoint	Time Delay (sec)
Negative Neutron Flux Rate	N/M*	N/A
Positive Neutron Flux Rate	6.06% / 2 sec	0.50
High Neutron Flux - Low Setting	50%	0.45
High Neutron Flux - High Setting	118%	0.45
Overpower ΔT (OPΔT)	Variable - See Figure 14.3-1	6.0
Overtemperature ΔT (OTΔT)	Variable - See Figure 14.3-1	6.0
Low Reactor Coolant Loop Flow	87%	1.2
High Pressurizer Level	N/M*	N/A
Low Pressurizer Pressure	1850 psia	1.0
High Pressurizer Pressure	2425 psia	1.0
Low-Low Steam Generator Level	0% narrow range span	1.5
RCP Undervoltage	N/M*	N/A
RCP Underfrequency	N/M*	N/A
Turbine Trip	N/A	2.0

N/M* - not explicitly modeled in safety analysis

The difference between the limiting trip setpoint assumed for the analysis and the actual trip setpoint represents a conservative allowance for instrumentation channel and setpoint errors. Results of surveillance tests demonstrate that actual instrument delays are equal to or less than the assumed values.

Reference is made above to the OTΔT and OPΔT variable reactor trip setpoints illustrated in Figure 14.3-1. This figure presents the allowable reactor coolant loop average temperature and ΔT for the design flow and power distribution as a function of primary coolant pressure. The boundaries of operation defined by the OPΔT trip and the OTΔT trip are represented as "Protection Lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions a trip would occur well within the area bounded by these lines. These protection lines are based on the safety analysis limit OTΔT and OPΔT setpoint values, which were calculated using the methodology of Reference 45, and are essentially the Technical Specification allowable values with allowances for the adverse instrumentation behavior, setpoint errors, and acceptable drift between instrument calibrations. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line (ΔT versus T_{avg}). The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit value (1.34 for the thimble cell and 1.34 for the typical cell) (Reference 65). All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit value. The area of permissible operation (power, pressure, and temperature) is bounded by the following combination of reactor trips: high neutron flux (fixed setpoint), high pressurizer pressure (fixed setpoint), low pressurizer pressure (fixed setpoint), OPΔT (variable setpoint) and OTΔT (variable setpoint). The DNBR limit value is used for all accidents analyzed with the RTDP (see Table 14.3-1), and is conservative compared to the actual design limit DNBR value required to meet the DNB design basis.

Reactor trip is defined for analytical purposes as the insertion of all full-length rod control cluster assemblies (RCCAs) except for the most reactive RCCA, which is assumed to remain in the fully withdrawn position. This is to provide shutdown margin against the remote possibility of a stuck RCCA condition existing at a time when shutdown is required.

The negative reactivity insertion following a reactor trip is a function of the acceleration of the control rods and the variation in rod worth as a function of rod position. Control rod positions after trip have been determined experimentally as a function of time using an actual prototype assembly under simulated flow conditions. The resulting rod positions were combined with rod worths to define the negative reactivity insertion as a function of time, as shown in Figure 14.3-2.

Reactor protection is designed to prevent cladding damage in all transients and abnormalities. The most probable modes of failure in each protection channel result in a signal calling for the protective trip. Coincidence of two-out-of-three (or two-out-of-four) signals is required where single channel malfunction could cause spurious trips while at power. A single component or channel failure in the protection system itself coincident with one stuck RCCA is always permissible as a contingent failure and does not cause violation of the protection criteria. The reactor protection systems are designed in accordance with the IEEE 279 "Standard for Nuclear Plant Protection System," August 1968.

Control Systems and Engineered Safeguards Features

Where applicable, control system and engineered safeguards features assumptions are identified in the transient-specific sections later in this chapter. However, note that in general a plant control system is modeled as-designed in a transient analysis only if it makes the results more limiting.

Single Active Failures

For Section 14.4 analyses, single active failure assumptions are made to evaluate the capability to mitigate the transient. Coincident failures such as a stuck open relief valve are not assumed.

Computer Codes Utilized

Summaries of the principal computer codes used in transient analyses are given below. Other codes, such as those used in the analysis of reactor coolant system pipe ruptures (Sections 14.6 and 14.7), are summarized in the respective accident analysis section. Table 14.3-1 provides a list of codes used for each transient analysis. 01365252

<u>FACTRAN</u> (Reference 5)

FACTRAN calculates the transient temperature distribution in a cross-section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the cladding, using as input the nuclear power and the time-dependent coolant parameters of pressure, flow, temperature and density. The code uses a fuel model that simultaneously contains the following features: 01365252

a. A sufficiently large number of radial space increments to handle fast transients such as a rod ejection accident,

b. Material properties that are functions of temperature and a sophisticated fuel-to-cladding gap heat transfer calculation, and

c. The necessary calculations to handle post-DNB transients: film boiling heat transfer correlations, Zircaloy-water reaction, and partial melting of the fuel.

<u>RETRAN</u> (Reference 6)

RETRAN is used for studies of transient response of a pressurized water reactor (PWR) system to specified perturbations in process parameters. This code simulates a multi-loop system by a lumped parameter model containing the reactor vessel, hot and cold leg piping, reactor coolant pumps, steam generators (tube and shell sides), steam lines, and the pressurizer. The pressurizer heaters, spray, relief valves, and safety valves may also be modeled. RETRAN includes a point neutron kinetics model and reactivity effects of the moderator, fuel, boron, and control rods. The secondary side of the steam generator uses a detailed nodalization for the thermal transients. The reactor protection system (RPS) simulated in the code includes reactor trips on high neutron flux, high neutron flux rate, overtemperature and overpower ΔT (OTΔT/OPΔT), low reactor coolant system (RCS) flow, high and low pressurizer pressure, high pressurizer level, and low steam generator water level. Control systems are also simulated including rod control and pressurizer pressure control. Parts of the safety injection system (SIS), including the accumulators, may also be modeled. RETRAN approximates the transient value of departure from nucleate boiling ratio (DNBR) based on input from the core thermal safety limits.

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<u>LOFTRAN</u> (Reference 7)

LOFTRAN is used for studies of transient response of a PWR system to specified perturbations in process parameters. This code simulates a multi-loop system by a model containing the reactor vessel, hot and cold leg piping, steam generators (tube and shell sides), the pressurizer and the pressurizer heaters, spray, relief valves, and safety valves. LOFTRAN also includes a point neutron kinetics model and reactivity effects of the moderator, fuel, boron, and rods. The secondary side of the steam generator uses a homogeneous, saturated mixture for the thermal transients. The code simulates the reactor protection system (RPS) which includes reactor trips on high neutron flux, OTΔT, OPΔT, high and low pressurizer pressure, low reactor coolant system (RCS) flow, low steam generator water level, and high pressurizer level. Control systems are also simulated including rod control, steam dump, and pressurizer pressure control. The safety injection system (SIS), including the accumulators, is also modeled. LOFTRAN can also approximate the transient value of DNBR based on input from the core thermal safety limits.

TWINKLE (Reference 8)

TWINKLE is a multi-dimensional neutron kinetics code. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-cladding-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 8,000 spatial points and performs its own steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. The code provides various output, e.g., channelwise power, axial offset, enthalpy, volumetric surge, pointwise power and fuel temperatures. It also predicts the kinetic behavior of a reactor for transients that cause a major perturbation in the spatial neutron flux distribution.

VIPRE (Reference 9)

The VIPRE computer program performs thermal-hydraulic calculations. This code calculates coolant density, mass velocity, enthalpy, void fractions, static pressure and DNBR distributions along flow channels within a reactor core.

ANC (Reference 44)

ANC is an advanced nodal code capable of two-dimensional and three-dimensional neutronics calculations. ANC is the reference model for certain safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, reactivity coefficients, etc. In addition, three-dimensional ANC validates one-dimensional and two-dimensional results and provides information about radial (x-y) peaking factors as a function of axial position. It can calculate discrete pin powers from radial nodal information as well.

RELAP5/MOD2-B&W (Reference 63)

RELAP5/MOD2-B&W code may be used to generate mass and energy releases during a Main Steam Line Break. The code, which is modularized according to components and functions, has been designed to model the behavior of all major components in the reactor system during accidents ranging from large-break LOCAs to anticipated operational transients involving the plant control and protection system. The primary system, secondary system, feedwater train, system controls, and core neutronics can be simulated. Special component models include pumps, valves, heat structures, turbines and separators and accumulators. The fundamental equations, constitutive models and correlations, and method of solution of RELAP5/MOD2 are described in NUREG/CR-4312 and NUREG/CR-5194. RELAP5/MOD2-B&W preserves the original models of RELAP5/MOD2 and adds features and models required for licensing analysis for both LOCA and Non-LOCA accidents and transients.

GOTHIC (Reference 64)

GOTHIC 7 may be used (1) to evaluate the short term peak pressure and temperature response of the containment atmosphere to large pipe breaks in high design systems – the design basis loss of coolant accident (LOCA) and the design basis main steamline break (MSLB), and (2) evaluate the long term containment response following a design basis LOCA. The GOTHIC 7 code takes mass and energy inputs provided by other analyses codes and models the containment response to calculate the resulting containment temperatures and pressures over the duration of the event. The code conservatively models the plant equipment configuration such as the containment fan coil units, safety injection flow, residual heat removal injection and recirculation, containment spray and containment heat structures.

14.3.2 Design Basis Limits for Fission Product Barriers (DBLFPBs)

The NRC has defined the design basis limit for a fission product barrier as the controlling numerical value for a parameter established during the licensing review as presented in the Updated Safety Analysis Report for any parameter(s) used to determine the integrity of the barrier. The list of DBLFPBs for Prairie Island is contained in Table 14.3-3.

14.3.3 Potential Voiding in the Reactor Coolant System During Anticipated Transients

As a result of the TMI-2 incident (NUREG-0737 Item II.K.2.17) and the St. Lucie Cooldown Event, the Westinghouse Owners Group undertook a study to ascertain the potential for void formation during anticipated transients. The potential for void formation depends upon, among other things, the initial temperature of fluid in the upper head region of the reactor vessel. This area is cooled by cold leg water diverted to the upper head and therefore the fluid temperature of this body of water is between the cold and hot leg temperatures. Voids can be created in the upper reactor vessel by either decreasing the pressure below the saturation pressure or by increasing the fluid temperature above the saturation temperature.

Westinghouse's evaluation included the void formation in the upper head region is accounted for in previously submitted transient analyses. As a result of the evaluation the NRC concluded in a December 30, 1983 letter that steam voids do not result in unacceptable consequences during anticipated transients.

Generic Letter 81-21 identified a concern involving steam formation in the upper head region during natural circulation cooldown. The potential for forming a void in the upper head region is minimized by controlling the cooldown and depressurization rates. During natural circulation, cooldown of the upper head region is dependent on heat losses to ambient. This cooldown rate is aided with the CRDM fans running. Thus, different maximum cooldown rates are specified depending on the availability of the CRDM fans. A lower maximum cooldown rate is specified if CRDM fans are not available. Furthermore, if the CRDM fans are not available, a soak should be conducted prior to completely depressurizing the RCS. A cooldown from 547°F to 350°F requires approximately 93,000 gallons of water with the CRDM fans running and approximately 159,000 gallons of water without the CRDM fans running. At least 200,000 gallons of water are normally available in the Condensate Storage Tanks. If the volume available in the Condensate Storage Tanks is less than the required volume for the cooldown, river water would be available for backup (safeguards source).

14.4 ABNORMAL OPERATIONAL TRANSIENT ANALYSIS

A development in nuclear power plant safety studies has been the categorizing of certain anticipated plant transients as a special class of events requiring special attention. These anticipated events are abnormal operational transients resulting from component failure or operator error which constitute a demand for action by the reactor protection system and which are anticipated to occur sometime in the design life of the plant.

For the following plant abnormalities and transients, the reactor control and protection system is relied upon to protect the core and reactor coolant boundary from damage:

a. Uncontrolled RCC Assembly Withdrawal from a Subcritical Position (Section 14.4.1)

b. Uncontrolled RCC Assembly Withdrawal at Power (Section 14.4.2)

c. RCC Assembly Misalignment (Section 14.4.3)

d. Chemical and Volume Control System Malfunction (Section 14.4.4)

e. Start-Up of an Inactive Reactor Coolant Loop (Section 14.4.5)

f. Excessive Heat Removal Due to Feedwater System Malfunctions (Section 14.4.6)

g. Excessive Load Increase Incident (Section 14.4.7)

h. Loss of Reactor Coolant Flow (Section 14.4.8)

i. Loss of External Electrical Load (Section 14.4.9)

j. Loss of Normal Feedwater (Section 14.4.10)

k. Loss of All AC Power to the Station Auxiliaries (Section 14.4.11)

Trip is defined for analytical purposes as the insertion of all full length RCC assemblies except the most reactive assembly which is assumed to remain in the fully withdrawn position. This is to provide margin in shutdown capability against the remote possibility of a stuck RCC assembly condition existing at a time when shutdown is required.

Instrumentation is provided for monitoring all individual RCC assemblies together with their respective bank position. This is in the form of a deviation alarm system. If the rod should deviate from its intended position, the appropriate actions from the Technical Specifications would be initiated. Such occurrences are expected to be extremely rare based on operation and test experience to date.

In summary, reactor protection is designed to prevent cladding damage in all transients and abnormalities listed above. The most probable modes of failure in each protection channel result in a signal calling for the protective trip. Coincidence of two out of three (or two out of four) signals is required where single channel malfunction could cause spurious trips while at power. A single component or channel failure in the protection system itself coincident with one stuck RCCA is always permissible as a contingent failure and does not cause violation of the protection criteria. The reactor protection systems are designed in accordance with the IEEE 279 “Standard for Nuclear Plant Protection System,” August 1968.

14.4.1 Uncontrolled RCCA Withdrawal From a Subcritical Condition

14.4.1.1 Identification of Cause and Frequency Classification

A Rod Cluster Control Assembly (RCCA) withdrawal incident from subcritical is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of rod cluster control assemblies resulting in a power excursion. A RCCA withdrawal incident has an extremely low probability of occurrence but could be caused by a malfunction of the reactor control or control rod drive system.

This event is classified as a Condition II event (moderate frequency).

14.4.1.2 Expected Plant Response

This subsection describes the actual sequence of events and expected system response to an Uncontrolled RCCA Withdrawal from a Subcritical Condition. It does not represent assumptions, requirements, or equipment used in the analysis.

The control rod drive mechanisms are wired into preselected bank configurations which are not altered during core life. The rod assemblies are therefore physically prevented from withdrawing in other than their respective banks. The power supplied to the rod banks is controlled such that no more than two banks can be withdrawn at any time. This limits the rate at which positive reactivity can be added to the core.

For small positive reactivity insertion rates, the nuclear power will increase until rod motion is stopped by the Intermediate Range Rod Blocks or terminated by the Source Range or lower power NIS trips.

For large positive reactivity insertion rates, the nuclear power response is characterized by a very fast rise in power terminated by the Doppler reactivity feedback. After the initial energy release, the reactor power is reduced by this inherent feedback and the transient is terminated by a reactor source range or lower power NIS trip. For these large reactivity insertion rates, the nuclear power is increasing so fast that the Intermediate Range Rod Blocks will not be able to prevent the reactor trip.

Due to the small amount of energy released to the core coolant during this transient, pressure and temperature excursions are minimal.

14.4.1.3 Analysis of Transient

14.4.1.3.1 Methodology

The analysis models the effects of a constant positive reactivity insertion into a critical reactor at hot zero power conditions. The reactor is initially assumed to be critical below the point of adding heat since this results in the maximum nuclear flux peak.

The computer codes used to analyze this transient are described in section 14.3. The analysis is performed in three stages: first, an average core nuclear power transient calculation, then, an average core heat transfer calculation, and finally the DNBR calculation. The average nuclear power transient calculation is performed using the spatial neutron kinetics code TWINKLE, which includes the various total core feedback effects, i.e., Doppler and moderator reactivity. The FACTRAN code is then used to calculate the thermal heat flux transient, based on the nuclear power transient calculated by TWINKLE. FACTRAN also calculates the fuel and cladding temperatures. The average heat flux is next used in the VIPRE code for DNBR calculations.

14.4.1.3.2 Key Physics Parameter Assumptions

The following physics parameters are reviewed each refueling cycle to ensure that the individual parameter used in the analysis is bounding. If it is not bounded, an evaluation is performed to ensure the analysis would bound a cycle specific analysis or a new analysis is performed.

a. *Isothermal Temperature Coefficient* - The contribution of the moderator and Doppler temperature reactivity feedback is negligible during the initial part of the transient because the heat transfer time constant between the fuel and the moderator is much longer than the nuclear flux response time constant. However, after the initial nuclear flux peak, the succeeding rate of heat flux increase is affected by the isothermal temperature coefficient. A conservative isothermal temperature coefficient of +5 pcm/°F has been used in the analysis to yield the maximum peak core heat flux.

b. *Doppler Power Defect* - As the magnitude of the nuclear power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler reactivity coefficient, a conservatively low (absolute magnitude) value for the Doppler power defect is used (1100 pcm). The least negative Doppler feedback effect increases the nuclear flux peak.

c. *Scam Reactivity Curve* - A conservatively slow scram curve based on a 2.7-second drop time to the dashpot is assumed. The corresponding trip reactivity, which accounts for the most reactive rod fully withdrawn, is 1% Δk.

d. *Effective Delayed Neutron Fraction* - The magnitude and width of the initial power peak is sensitive to the delayed neutron fraction (beta). As beta decreases, the magnitude of the initial nuclear power peak increases, but the width of the peak decreases. The important parameter for the analysis is the total amount of energy deposited in the fuel during the nuclear power peak, which increases with increasing beta. The greater energy deposited in the fuel translates into a greater peak heat flux. Therefore, a maximum beta is conservatively assumed.

e. *RCCA Withdrawal Reactivity Insertion Rate* - A constant, maximum reactivity insertion rate of 75 pcm/second, which is greater than that corresponding to the simultaneous withdrawal at maximum speed of the two RCCA banks having the greatest combined worth, is assumed.

14.4.1.3.3 Key System Parameter Assumptions

The following key system parameter assumptions are made to ensure the overall results of the analysis bound actual operation.

a. The reactor is assumed to be at the hot zero power nominal temperature of 547°F. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields larger fuel to water heat transfer, larger fuel thermal capacity, and a less negative (smaller absolute magnitude) Doppler coefficient. The high nuclear flux peak combined with a high fuel thermal capacity and large thermal conductivity yields a larger peak heat flux. The initial effective multiplication factor is assumed to be 1.0 because this results in the maximum nuclear flux peak.

b. The initial power level is assumed to be below the power level expected for any HZP just-critical condition (10^{-9} fraction of nominal power). The combination of highest reactivity insertion rate and low initial power produces the highest peak heat flux.

c. The power range high neutron flux low setting reactor trip function is credited. The assumed setpoint is 10% above the 40% allowable value, i.e., 50%. The most adverse combination of instrument and setpoint errors, as well as delays for reactor trip signal actuation and rod release, are taken into account. However, it is apparent in Figure 14.4-1 that the rise in nuclear flux is so rapid that the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible.

d. The initial system pressure is conservatively minimized at 2190 psia.

e. One reactor coolant pump is assumed to be in operation, although the plant Technical Specifications require both pumps to be in operation. Also, a core flow reduction of 1.1%, corresponding to reactor coolant loop flow asymmetry associated with a maximum loop-to-loop steam generator tube plugging imbalance of 10 percent, has been applied. These flow assumptions conservatively minimize the resulting DNBR.

f. The most limiting axial and radial power shapes, associated with having the two highest combined worth sequential banks in their highest worth position, are assumed in the DNBR analysis.

14.4.1.3.4 Single Active Failure Assumptions of a Safety Grade Component

Mitigation of an Uncontrolled RCCA Withdrawal from a Subcritical Condition transient is accomplished by a reactor trip.

The worst case single failure for an Uncontrolled RCCA Withdrawal from a Subcritical Condition transient is the failure of a reactor protection train. However, the reactor protection system is designed such that any single failure does not prevent proper operation of the protection system (see USAR section 7.4). Therefore the analysis assumes that the reactor protection system operates as designed.

14.4.1.4 Acceptance Criteria

1. The peak fuel centerline temperature must be less than the minimum temperature that could cause fuel melting.

2. The minimum departure from nucleate boiling ratio (DNBR) must be greater than the applicable limit for the DNBR correlation being used accounting for the penalties and factors described in Section 3.2.2, “Thermal Hydraulic Design Analysis”.

3. The peak reactor coolant system (RCS) pressure must be less than 110% of the design pressure. Based on the fact that the total amount of excess energy deposited in the reactor coolant is relatively small, and there is no prolonged power mismatch between the primary and secondary sides, overpressurization of the RCS is not of significant concern. As the pressure response would be less severe than that associated with a loss of external electrical load, it is not explicitly analyzed for this transient.

14.4.1.5 Results and Radiological Consequences

The sequence of events for the uncontrolled RCCA withdrawal from subcritical transient is presented in Table 14.4-1, Figures 14.4-1 through 14.4-5 show the transient behavior of key parameters for the uncontrolled RCCA bank withdrawal for 400 V + fuel. Figures 14.4-1a through 14.4-5a show the same results for 422 V+ fuel or mixed cores.

As shown in Figure 14.4-1 and 14.4-1a, the nuclear power overshoots the nominal full power value, but only for a very short time period, and thus the energy release and the fuel temperature increases are small. The heat flux response, of interest for DNB consideration, is shown in Figure 14.4.2 and 14.4-2a. The beneficial effect of the inherent thermal lag of the fuel is evidenced by a peak heat flux that is less than the nominal full power heat flux. The limiting calculated minimum DNBR is greater than the applicable value. Figures 14.4-3, 14.4-3a, 14.4-4, 14.4-4a, and 14.4-5, 14.4-5a show the response of the fuel centerline, average fuel and cladding temperatures at the hot spot for both fuel types, respectively. Note that the peak fuel centerline temperature is less than the minimum temperature that could cause fuel melting. With the reactor tripped, the plant returns to a stable condition, and may subsequently be cooled down further by following normal plant shutdown procedures.

Radiological consequences are not evaluated for this transient because no fuel pins are expected to experience centerline melt or other modes of failure including clad failure due to experiencing departure from nucleate boiling.

14.4.2 Uncontrolled RCCA Withdrawal at Power

14.4.2.1 Identification of Cause and Frequency Classification

A Rod Cluster Control Assembly (RCCA) withdrawal incident at power is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of rod cluster control assemblies resulting in a power excursion. This transient could be caused by a malfunction of the reactor control or control rod drive system.

This event is classified as a Condition II event (moderate frequency).

14.4.2.2 Expected Plant Response

This subsection describes the actual sequence of events and expected system response to an Uncontrolled RCCA withdrawal at Power. It does not represent assumptions, requirements, or equipment used in the analysis.

The control rod drive mechanisms are wired into preselected bank configurations and the power supplied to the rod banks is controlled such that no more than two banks can be withdrawn at any time. This limits the rate at which positive reactivity can be added to the core.

An uncontrolled RCCA withdrawal at power results in a gradual increase in core power followed by an increase in core heat flux. The resulting mismatch between core power and steam generator heat load results in an increase in reactor coolant temperature and pressure. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise would eventually result in DNB.

Depending on the initial conditions of the reactor and the reactivity insertion rate, a number of different signals could trip the reactor. These signals include, but are not limited to, NIS high flux, Overpower Delta T, Overtemperature Delta T, and NIS positive rate. See USAR section 7.4 for more information on the reactor trips.

14.4.2.3 Analysis of Transient

14.4.2.3.1 Methodology

The RETRAN computer code used to analyze this transient is described in section 14.3.

The reactivity insertion rate determines which protective system function will initiate termination of the transient; therefore a range of insertion rates must be considered. The fast rate bounds the two highest worth banks moving simultaneously. The minimum (slow) rate is determined by varying the reactivity insertion rate until the minimum departure from nucleate boiling ratio has been determined.

14.4.2.3.2 Key Physics Parameter Assumptions

The following physics parameters are reviewed each refueling cycle to ensure that the individual parameter used in the analysis is bounding. If it is not bounded, an evaluation is performed to ensure the analysis would bound a cycle specific analysis or a new analysis is performed.

a. *Reactivity Coefficients* - The analysis considers both minimum and maximum reactivity feedback conditions.

- *Minimum Reactivity Feedback* - A least-negative isothermal temperature coefficient of 0 pcm/°F is assumed at full power, and a least-negative (most-positive) value of +5 pcm/°F is assumed for power levels less than or equal to 70%. In addition, a least-negative Doppler power coefficient and a maximum effective delayed neutron fraction are assumed.

- *Maximum Reactivity Feedback* - A maximum positive moderator density coefficient, a most-negative Doppler temperature coefficient, a most-negative Doppler power coefficient, and a minimum effective delayed neutron fraction are assumed.

b. *Scram Reactivity Curve* - A conservatively slow scram curve based on a 2.4 second drop time to the dashpot is assumed. The corresponding trip reactivity, which accounts for the most reactive rod fully withdrawn, is 4% Δk.

c. RCCA Withdrawal Reactivity Insertion Rate - A broad range of reactivity insertion rates from 1 pcm/second to 110 pcm/second is examined, with the maximum rate being greater than that corresponding to the simultaneous withdrawal at maximum speed (45 inches/minute) of the two RCCA banks having the greater combined worth.

14.4.2.3.3 Key System Parameter Assumptions

The following key system parameter assumptions are made to ensure the overall results of the analysis bound actual operation.

a. The power supply to the RCCA drive mechanisms is such that no more than two banks may be withdrawn simultaneously.

b. The reactivity reduction due to reactor trip is calculated by considering the most adverse combination of instrument and setpoint errors and time delays.

c. The analysis assumes the reactor trip is caused by the Power Range Hi Neutron Flux High Setpoint, Overtemperature Delta T or the Power Range Positive Neutron Flux Rate trip.

d. The RTDP methodology (see Section 14.3) is employed in the cases analyzed for DNB concerns, and the non-RTDP (STDP) methodology is employed in the cases analyzed for pressure concerns.

e. Initial power levels of 10%, 60% and 100% of full power are examined.

f. The initial vessel average temperature is selected for each case based on the assumed initial power level and whether or not RTDP methodology is applied. For the non-RTDP cases, maximum steady-state errors are applied in the most conservative direction. For the RTDP cases, the nominal vessel average temperature value is assumed.

g. The initial RCS flow is assumed to be the minimum measured flow in the DNB cases and the thermal design flow in the pressure cases.

h. The initial pressurizer pressure is selected for each case based on whether or not RTDP methodology is applied. For the non-RTDP cases, maximum steady-state errors are applied in the most conservative direction. For the RTDP cases, the nominal pressurizer pressure value is assumed.

When evaluating primary-side over pressure:

i. The pressurizer PORVs and pressurizer spray systems are disabled.

When evaluating departure from nucleate boiling and secondary-side over-pressure:

j. The pressurizer PORVs and pressurizer spray systems are enabled.

14.4.2.3.4 Single Active Failure Assumptions of a Safety Grade Component

Mitigation of an Uncontrolled RCCA Withdrawal at Power transient is accomplished by a reactor trip.

The worst case single failure for an Uncontrolled RCCA Withdrawal at Power transient is the failure of a reactor protection train. However, the reactor protection system is designed such that any single failure does not prevent proper operation of the protection system (see USAR section 7.4). Therefore the analysis assumes that the reactor protection system operates as designed.

14.4.2.4 Acceptance Criteria

1. The maximum reactor coolant and main steam system pressure must not exceed 110% of their design values.

2. The minimum departure from nucleate boiling ratio (DNBR) must be greater than the applicable limit for the DNBR correlation being used accounting for the penalties and factors described in Section 3.2.2, “Thermal Hydraulic Design Analysis”.

14.4.2.5 Results and Radiological Consequences

Table 14.4-3 presents sample time sequence of events results for the uncontrolled RCCA withdrawal transient initiated from full power with minimum reactivity feedback conditions. Figures 14.4-6 to 14.4-11 show the transient response for a rapid RCCA withdrawal incident starting from full power. Reactor trip on the positive neutron flux rate function occurs shortly after the start of the transient. As 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 DNBR limit is maintained.

The transient response for a slow RCCA withdrawal from full power is shown in Figures 14.4-12 to 14.4-17. Reactor trip on the Overtemperature ΔT reactor trip function occurs after a longer period of time, and the rise in temperature is consequently larger than that of a rapid RCCA withdrawal. Again, the minimum DNBR is greater than the safety analysis limit.

Figure 14.4-18 shows the minimum DNBR as a function of reactivity insertion rate for cases analyzed from full power operation for both minimum and maximum reactivity feedback. The high neutron flux, positive flux rate trip, and Overtemperature ΔT reactor trip functions provide protection over the entire range of reactivity insertion rates analyzed, as the minimum DNBR is never less than the safety analysis limit.

Figures 14.4-19 and 14.4-20 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 60 and 10-percent power, respectively, for both minimum and maximum reactivity feedback. The results are similar to the 100-percent power case, except as the initial power is decreased, the range over which the Overtemperature ΔT reactor trip is effective is increased. In all cases, the minimum DNBR remains above the safety analysis limit. The shape of the curves of minimum DNBR versus reactivity rate is due to the reactor core and coolant system transient response, and to the initiating reactor trip function.

For transients initiated at 100% power, it is noted that:

1. For reactivity insertion rates above approximately 7 pcm/sec reactor trip is initiated by either a high neutron flux trip or a positive flux rate trip for the minimum reactivity feedback cases. The neutron flux level in the core rises rapidly for these insertion rates while the core heat flux lags behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to a significant increase in the heat flux or the water temperature with resultant high minimum DNBRs. As the reactivity insertion rate decreases, the core heat flux and coolant temperature are closer to an equilibrium condition with the neutron flux. Therefore, the minimum DNBR during the transient decreases with decreasing insertion rate.

2. The Overtemperature ΔT reactor trip function initiates a reactor trip when the measured coolant loop ΔT exceeds the OTΔT setpoint, which is based on the measured Reactor Coolant System average temperature and the pressurizer pressure. Note that the average temperature contribution to the OTΔT reactor trip function is lead/lag compensated to compensate for the effects of the thermal capacity of the RCS in response to power increases.

3. For reactivity insertion rates below 7 pcm/sec the Overtemperature ΔT trip terminates the transient. For those reactivity insertion rates, the effectiveness of the Overtemperature ΔT trip increases (in terms of increased minimum DNBR) as insertion rate decreases due to the fact that with lower insertion rates the power increase is slower, and the average coolant temperature increases slower and the system thermal lags and delays become less significant.

4. As the DNBR remains above the limit value during the RCCA withdrawal at power transient, the ability of the primary coolant to remove heat from the fuel rod is not reduced.

The analysis results also show that the peak pressures in the reactor coolant system and main steam system do not exceed 110 percent of their respective design pressures. Thus, all applicable acceptance criteria are met for the uncontrolled RCCA bank withdrawal at power transient.

Radiological consequences are not evaluated for this transient because no fuel pins are expected to experience centerline melt or other modes of failure including clad failure due to experiencing departure from nucleate boiling.

14.4.3 RCCA Misalignment

Two separate RCCA misalignment conditions are considered. The first is statically misaligned RCCAs and the second is Dropped RCCAs or RCCA bank.

14.4.3.1 Statically Misaligned RCCAs

14.4.3.1.1 Identification of Cause and Frequency Classification

In the misalignment transient, one or more RCCAs is assumed to be statically misplaced from the normal or allowed position. This situation might occur if a rod were left behind when inserting or withdrawing banks, or if a single rod were to be withdrawn.

This event is classified as a Condition II event (moderate frequency).

14.4.3.1.2 Expected Plant Response

This subsection describes the actual sequence of events and expected system response to a Statically Misaligned RCCAs. It does not represent assumptions, requirements, or equipment used in the analysis.

A misaligned rod could result in increased peaking factors and a reduction in the departure from nucleate boiling ratio. There is no automatic reactor protection designed to trip the reactor if a misaligned rod exists. There are several alarms and control room indications to alert the operators to a potential statically misaligned RCCA. These include NIS tilts, thermocouple tilts, Rod position indicators, and rod deviation alarms.

14.4.3.1.3 Analysis of Transient

14.4.3.1.3.1 Methodology

The analysis of statically misaligned RCCA is done by modeling the most limiting configuration. The following cases were examined in the analysis assuming the reactor is initially at full power: the worst rod withdrawn with Bank D inserted at the insertion limit, the worst rod bottomed with Bank D inserted at the insertion limit, and the worst rod bottomed with all other rods out. It is assumed that the incident occurs at the time in the cycle at which the maximum all-rods-out $F_{\Delta H}$ occurs. The limiting value of the Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}$ is input to a steady state full power thermal-hydraulic subchannel calculation to determine the departure from nucleate boiling ratio (DNBR).

14.4.3.1.3.2 Key Physics Parameter Assumptions

The following physics parameters are reviewed each refueling cycle to ensure that the individual parameter used in the analysis is bounding. If it is not bounded, an evaluation is performed to ensure the analysis would bound a cycle specific analysis or a new analysis is performed.

a. Nuclear Enthalpy Rise Hot Channel Factor; A conservatively large value is assumed which bounds full power operation from the statepoints as described in the Methodology section.

14.4.3.1.3.3 Key System Parameter Assumptions

Assumptions on key safety parameters are consistent with the Westinghouse Revised Thermal Design Procedure (RTDP).

14.4.3.1.3.4 Single Active Failure Assumptions of a Safety Grade Component

The analysis of a Statically Misaligned RCCA assumes steady state operation. Therefore, there is no actuation of active safety grade components required for mitigation of the transient. Consequently, no single failure assumption is applied.

14.4.3.1.4 Acceptance Criteria

1. The minimum departure from nucleate boiling ratio (DNBR) must be greater than the applicable limit for the DNBR correlation being used accounting for the penalties and factors described in Section 3.2.2, “Thermal Hydraulic Design Analysis”.

2. The fuel temperature and clad strain limits consistent with the acceptance criteria of the Standard Review plan 4.2 are not exceeded.

14.4.3.1.5 Results and Radiological Consequences

This transient is evaluated each refueling cycle to confirm that the DNB design basis is met.

Radiological consequences are not evaluated for this transient because no fuel pins are expected to experience departure from nucleate boiling and thus experience cladding failure.

14.4.3.2 Dropped RCCA

14.4.3.2.1 Identification of Cause and Frequency Classification

In the dropped rod or assembly transient, one or more full length RCCAs or an RCCA bank is assumed to be released by the stationary gripper coils and falls to a fully inserted position in the core.

This event is classified as a Condition II event (moderate frequency).

14.4.3.2.2 Expected Plant Response

This subsection describes the actual sequence of events and expected system response to a Dropped RCCA transient. It does not represent assumptions, requirements, or equipment used in the analysis.

The drop of a single RCCA, multiple RCCAs within the same group, or an RCCA bank typically results in a negative flux rate reactor trip. The core power distribution is not adversely affected during the short interval prior to the trip. If a trip does not occur, skewed power shape may result. The consequences for this event are dependent upon whether the reactor is being operated in an automatic or manual mode. For operation in the manual mode with no operator actions, the plant returns to full power with an assembly fully inserted and a reduction in core thermal margins may result because of a possible increased hot channel peaking factor. If a rod drop event occurs when the reactor is in the automatic mode, the reactor control system responds to both the reactor power drop, as seen by the excore detectors, (mismatch between turbine power and reactor power) and the decrease in the core average temperature and attempts to restore both quantities to their original values. This restoration of reactor power by the reactor control system may result in some power overshoot. This power overshoot combined with the possible increased hot channel peaking factor (due to the inserted RCCA) will cause a reduction in the core thermal margin.

Dropped RCCAs or Banks are detected by a sudden drop in the core power, rod bottom light(s), and asymmetric power distribution as seen on excore detectors or thermocouples.

14.4.3.2.3 Analysis of Transient

14.4.3.2.3.1 Methodology

The methodology used to analyze the dropped RCCA(s) (dropped rod) event is described in Reference 12. In summary, the LOFTRAN computer code is used to generate dropped RCCA(s) reactor system statepoints for bounding ranges of dropped rod and control bank reactivity worths. For each dropped rod case, the statepoint represents the transient system conditions (temperature, pressure, and power) at the limiting point in the transient. No credit for any direct reactor trip due to the dropped rod(s) was taken in the generation of the statepoints. Next, nuclear models developed with the ANC computer code are used to obtain hot channel peaking factors consistent with the primary system conditions and reactor power. By combining the primary conditions from the transient analysis and the hot channel factor from the nuclear analysis, a post-drop $F_{\Delta H}$ is calculated. This is then compared to the $F_{\Delta H}$ limit value that corresponds to the DNBR safety analysis limit, as determined by the VIPRE computer code using the Revised Thermal Design Procedure (RTDP - Reference 4). By meeting the $F_{\Delta H}$ limit for all dropped rod scenarios, the DNB design basis is shown to be satisfied.

The computer codes used to analyze this transient are described in section 14.3.

14.4.3.2.3.2 Key Physics Parameter Assumptions

The following key physics parameter assumptions are made in analyzing the Dropped Rod event:

a. Moderator Temperature Coefficient: a range of values is modeled.

b. Doppler Temperature Coefficient: a most negative value is assumed.

c. Doppler Power Defect: a value of 1% is assumed.

d. Effective Delayed Neutron Fraction: a maximum value is assumed.

e. Prompt Neutron Lifetime: a minimum value is assumed.

f. Dropped Rod Worth: a bounding range of values is modeled.

g. Control Bank Worth: a bounding range of values is modeled.

14.4.3.2.3.3 Key System Parameter Assumptions

The following key system parameter assumptions are made to ensure the overall results of the analysis bound actual plant operation:

a. Full power conditions are assumed.

b. Initial conditions of core power, RCS coolant temperature and pressurizer pressure are assumed to be at their nominal values.

c. The control banks are assumed to be at the insertion limits.

d. The rod control system is assumed to be in automatic mode.

14.4.3.2.3.4 Single Active Failure Assumptions of a Safety Grade Component

The consequences of a dropped rod are dependent on the excore tilts seen by the excore detectors. These detectors have input to both the negative rate trips and the rod control system. The lowest NIS channel input to the rod control system is assumed when evaluating the excore signal response. This assumption simulates the most extreme core tilting that could occur by a dropped RCCA near the core periphery, and leads to an increased core power overshoot.

14.4.3.2.4 Acceptance Criteria

1. The minimum departure from nucleate boiling ratio (DNBR) must be greater than the applicable limit for the DNBR correlation being used accounting for the penalties and factors described in Section 3.2.2, “Thermal Hydraulic Design Analysis”.

2. The fuel temperature and clad strain limits consistent with the acceptance criteria of the Standard Review Plan 4.2 are not exceeded.

14.4.3.2.5 Results and Radiological Consequences

Figures 14.4-21 through 14.4-24 show a typical transient response to a dropped RCCA event with the reactor in automatic rod control. In all cases, the minimum DNBR remains above the limit value; this is confirmed for each reload cycle. In addition, the applicable fuel temperature and clad strain limits are met.

Radiological consequences are not evaluated for this transient because no fuel pins are expected to experience departure from nucleate boiling and thus experience cladding failure.

14.4.4 Chemical and Volume Control System Malfunction

A malfunction of the Chemical and Volume Control System (CVCS) that causes an inadvertent dilution of the Reactor Coolant System (RCS) could occur at any plant operating mode. For the purposes of this analysis, all operating modes are reviewed. The methods used to analyze an event initiated when the reactor is critical are substantially different than when the reactor is subcritical. The critical and subcritical cases are evaluated in two separate subsections below.

The major hazard associated with an unmitigated CVCS malfunction is a reduction in the DNB ratio and/or a complete loss of shutdown margin. The events, therefore, are analyzed in order to determine the minimum DNB ratio or the response time that exists prior to a complete loss of shutdown margin.

14.4.4.1 Critical Reactor

14.4.4.1.1 Identification of Cause and Frequency Classification

The accident considered here is the malfunction of the CVCS resulting in the injection of non-borated water at the maximum possible flowrate to the RCS under at power conditions. With the reactor in automatic control, the decrease in the boron concentration will cause the power and temperature to increase resulting in the insertion of the RCC assemblies and a decrease in shutdown margin. With the reactor in manual control, the decrease in the boron concentration will cause the power and temperature to increase. This will eventually result in the overtemperature or overpower ΔT reactor trip if the operator does not intervene.

The boric acid from the boric acid tank is blended with the reactor makeup water in the blender and the composition is determined by the preset flow rates of boric acid and reactor makeup water on the Reactor Makeup Control System. Two separate operations are required. First, the operator must switch from the automatic makeup or alternate dilution mode to the dilute mode. Second, a control switch must be operated. Omitting either step would prevent dilution. This makes the possibility of inadvertent dilution very small.

Other mechanisms exist which could cause an inadvertent dilution of the Reactor Coolant System. It has been determined that the limiting condition is with the charging pumps. The following discussion evaluates this dilution mechanism.

This event is classified as a Condition II event (moderate frequency).

14.4.4.1.2 Expected Plant Response

This subsection describes the actual sequence of events and expected system response to a CVCS malfunction with the reactor critical. It does not represent assumptions, requirements, or equipment used in the analysis.

Reactivity can be added to the core with the CVCS by supplying reactor makeup water from the reactor makeup control system. An intentional boron dilution is a manual operation performed under operator surveillance. For blended additions, a boric acid blend system is provided to permit the operator to match the concentration of reactor coolant makeup water to that existing in the coolant at the time. The CVCS is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

The primary source of reactor makeup water for the Reactor Coolant System is the reactor makeup water system. If this is the cause, an inadvertent dilution can be readily terminated by isolating this single source. In order for reactor makeup water to be added to the Reactor Coolant System the charging pumps must be running in addition to the reactor makeup water pumps. There are other potential dilution sources and mechanisms (for example, inadvertent valve lineups of ion exchangers, etc.).

Information on the status of the reactor coolant makeup is available to the operator in the control room. Lights are provided on the control board to indicate the operating condition of pumps in the Chemical and Volume Control System. Alarms are actuated to warn the operator if boric acid or reactor makeup water flow rates deviate from preset values.

With the reactor in automatic control, at full power, the power and temperature increase from the boron dilution results in the insertion of a RCC assembly group and a decrease in shutdown margin. A continuation of the dilution and rod insertion would cause the rods to reach the lower limit of the maneuvering band. Before reaching this point, however, two alarms (LOW and LOW-LOW rod insertion alarms) would be actuated to warn the operator of the accident condition. Both alarms alert the operator to initiate boration. With a continued dilution, the available shutdown margin would be lost. In this event, ample time is available following the alarms for the operator to determine the cause, isolate the reactor water makeup source, and initiate boration before this shutdown margin is lost.

With the reactor in manual control, and assuming the operator takes no action, the power and the temperature will rise to the overtemperature or overpower ΔT reactor trip setpoint. Prior to the overtemperature or overpower ΔT trip, an overtemperature/overpower ΔT alarm and turbine runback would be actuated. With the minimum shutdown margin at the beginning of the dilution there is ample time available for the operator to terminate dilution before the reactor can return to criticality following the trip.

Because of the procedures involved in the dilution process, an erroneous dilution is considered unlikely. Nevertheless, if an unintentional dilution of boron in the reactor coolant does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to the dilution is slow enough to allow the operator to determine the cause of the addition and take corrective action before excessive shutdown margin is lost.

14.4.4.1.3 Analysis of Transient

14.4.4.1.3.1 Methodology

The time available for operator action is calculated using the dilution flow rate, volume of the reactor coolant system, and critical and initial boron concentration.

14.4.4.1.3.2 Key Physics Parameter Assumptions

The analyses performed do not directly model any physics parameters, with the exception of boron worth, where the maximum considered is -16.0 pcm/ppm and the minimum considered is -5.0 pcm/ppm.

14.4.4.1.3.3 Key System Parameter Assumptions

The following key system parameter assumptions are made to ensure the overall results of the analysis bound actual operation.

a. Maximum possible charging flow is assumed based on three charging pumps providing full flow. This is selected to bound normal operating configurations and system capabilities.

b. Minimum active volume of the Reactor Coolant System is assumed.

14.4.4.1.3.4 Single Active Failure Assumptions of Safety Grade Components

As the initiation of this event requires multiple system malfunctions and/or operator errors, no additional operator errors are assumed to occur.

The analysis of a CVCS malfunction with rods in automatic does not result in a reactor trip or safety injection signal. Therefore, there is no actuation of active safety grade components required for mitigation of the transient. Consequently, no single failure assumption is applied.

The analysis of a CVCS malfunction with rods in manual results in a reactor trip. The worst case single failure for this condition is the failure of a reactor protection train. However, the reactor protection system is designed such that any single active failure does not prevent proper operation of the protection system (Section 7.4). Therefore, the analysis assumes that the reactor protection system operates as designed.

14.4.4.1.4 Acceptance Criteria

The acceptance criterion applied for an Uncontrolled Boron Dilution event is that there is adequate time for the operator to assess the situation and take appropriate action to prevent a complete loss of shutdown margin. If shutdown margin is maintained, then all Condition II acceptance criteria are met.

The calculated time, from the time an alarm alerts the operator to a dilution to the complete loss of shutdown margin, must be greater than or equal to the following:

Start-up (Mode 2)	15 minutes
Power (Mode 1)	15 minutes

14.4.4.1.5 Results and Radiological Consequences

<u>Dilution during Full-Power Operation (Mode 1)</u>

With the reactor in automatic rod control, the power and temperature increase from the boron dilution results in insertion of the control rods and a decrease in available shutdown margin. The rod insertion limit alarms (low and low-low settings) alert the operator at least 15 minutes prior to loss of shutdown margin. This is sufficient time to determine the cause of dilution, isolate the reactor makeup source, and initiate boration before the available shutdown margin is lost.

With the reactor in manual control, and assuming the operator takes no action, the power and temperature will rise to the overtemperature or overpower ΔT trip reactor trip setpoint. The boron dilution transient in this case is essentially the equivalent to an uncontrolled RCCA bank withdrawal at power. The maximum reactivity insertion rate for a boron dilution is conservatively estimated to be 3.3 pcm/sec. which is within the range of insertion rates analyzed. Thus, the effects of dilution prior to reactor trip are bounded by the uncontrolled RCCA bank withdrawal at power analysis. Following reactor trip, there are greater than 15 minutes prior to criticality. This is sufficient time for the operator to determine the cause of dilution, isolate the reactor water makeup source, and initiate boration before the available shutdown margin is lost.

Dilution at Startup (Mode 2)

In the event of an unplanned approach to criticality or dilution during power escalation while in the startup mode, the plant status is such that minimal impact will result. The plant will slowly escalate in power until the power range high neutron flux low setpoint is reached and a reactor trip occurs. From the time of reactor trip, a time period greater than 15 minutes is available for operator action prior to return to criticality.

Critical Reactor

Because of the procedures involved in the dilution process, an erroneous dilution is considered unlikely. Nevertheless, if an unintentional dilution of boron in the reactor coolant does occur, numerous alarms and indications are available to alert the operator to the conditions. The maximum reactivity addition due to the dilution is slow enough to allow the operator sufficient time to determine the cause of the addition and take corrective action before shutdown margin is lost.

Radiological consequences are not evaluated for this transient because no fuel pins are expected to experience departure from nucleate boiling and thus experience cladding failure.

14.4.4.2 Subcritical Reactor

14.4.4.2.1 Identification of Cause and Frequency Classification

Non-borated water may be added to the RCS to increase core reactivity. If this happens inadvertently because of operator error or equipment malfunction, there is an unwanted increase in core reactivity and a decrease in shutdown margin. Termination of the event relies on operator action to stop the unplanned dilution before the shutdown margin is eliminated.

Other mechanisms exist which could cause an inadvertent dilution of the Reactor Coolant System. It has been determined that the limiting condition is with the charging pumps. The following discussion evaluates this mode of dilution.

This event is classified as a Condition II event (moderate frequency).

14.4.4.2.2 Expected Plant Response

This subsection describes the actual sequence of events and expected system response to a CVCS malfunction with the reactor subcritical. It does not represent assumptions, requirements or equipment used in the analysis.

Reactivity can be added to the core with the CVCS by supplying reactor makeup water from the reactor makeup control system. An intentional boron dilution is a manual operation performed under operator surveillance. For blended injections, a boric acid blend system is provided to permit the operator to match the concentration of reactor coolant makeup water to that existing in the coolant at the time. The CVCS is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

The primary source of reactor makeup water for the Reactor Coolant System is the reactor makeup water system. If this is the cause, an inadvertent dilution can be readily terminated by isolating this single source. In order for a significant flow rate of reactor makeup water to be added to the Reactor Coolant System the charging pumps must be running in addition to the reactor makeup water pumps. There are other potential dilution sources and mechanisms (for example, inadvertent valve lineups of ion exchangers, etc.).

Information on the status of the reactor coolant makeup is available to the operator in the control room. Lights are provided on the control board to indicate the operating condition of pumps in the Chemical and Volume Control System. Alarms are actuated to warn the operator if boric acid or reactor makeup water flow rates deviate from preset values.

Because of the procedures involved in the dilution process, an erroneous dilution is considered unlikely. Nevertheless, if an unintentional dilution of boron in the reactor coolant does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to the dilution is slow enough to allow the operator to determine the cause of the addition and take corrective action before excessive shutdown margin is lost.

The startup mode of operation is a transitory operational mode in which the operator intentionally dilutes and withdraws control rods to take the plant critical. During this mode, the plant is in manual control with the operator required to maintain a high awareness of the plant status. For a normal approach to criticality, the operator must manually initiate a limited dilution and subsequently manually withdraw the control rods, a process that takes several hours. The technical specifications require that the operator determine the estimated critical position of the control rods prior to approaching criticality, thus ensuring that the reactor does not go critical with the control rods below the insertion limits. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the source range reactor trip after receiving P-6 from the intermediate range. Too fast of a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly, leaving insufficient time to manually block the source range reactor trip, and the reactor would immediately shut down.

14.4.4.2.3 Analysis of Transient

14.4.4.2.3.1 Methodology

The methodology, as described in Reference 13, is to determine the minimum required shutdown margin to ensure that the acceptance criteria are satisfied for the specified plant condition and dilution flow rate. The minimum shutdown margin is determined using a relationship for determining the boron concentration as a function of time for a fixed mass and a given dilution rate.

This is based on the assumptions that:

a. The mass being diluted remains constant; i.e., there is a letdown flow rate equal in magnitude to the dilution flow rate.

b. The boron concentration is uniform throughout the mass being diluted; i.e., perfect and instantaneous mixing.

14.4.4.2.3.2 Key Physics Parameter Assumptions

The methods used in this calculation determine the minimum shutdown margin requirements necessary to satisfy the acceptance criterion. There are no key physics parameters that need to be reviewed each refueling cycle.

14.4.4.2.3.3 Key System Parameter Assumptions

The following key system parameter assumptions are made to ensure the overall results of the analysis bound actual operation.

a. The mass being diluted includes all active portions of the Reactor Coolant System (RCS) and interconnecting systems; i.e., where circulation is occurring. This includes the core, baffle region, downcomer, lower plenum, upper plenum, piping, and pumps. Depending on the system configuration being analyzed, it may also include volumes in the steam generators or in the decay heat removal system. The volumes are determined based on nominal dimensions of the systems.

b. The boron concentrations are determined using approved methodologies and include the appropriate calculational uncertainties. These concentrations are determined for the various plant conditions being analyzed, i.e., core exposure, RCS temperature, Xenon concentration.

c. The boron concentration corresponding to the complete loss of SDM is determined assuming that all available trip reactivity (accounting for the possibility of the most reactive rod being stuck) has been inserted into the core.

d. The dilution flow rate is the maximum flow based on the system configuration, system pressure and number of pumps running (note that the number of pumps running may be less than the number of pumps available). For the purpose of this input, an additional pump started for a brief period of time to accomplish such operations as pump switching does not constitute an additional pump running.

14.4.4.2.3.4 Single Active Failure Assumptions of Safety Grade Component

As the initiation of this event requires multiple system malfunctions and/or operator errors, no additional operator errors are assumed to occur.

The analysis of a CVCS malfunction with the reactor subcritical does not result in a reactor trip or safety injection signal. Therefore, there is no actuation of active safety grade components required for mitigation of the transient. Consequently, no single failure assumption is applied.

14.4.4.2.4 Acceptance Criteria

The acceptance criteria for a dilution accident when the reactor is subcritical is the time between the initiation of the dilution and a complete loss of SDM must be greater than or equal to 24 minutes. This provides ample time for operator recognition and mitigation of the dilution.

14.4.4.2.5 Results and Radiological Consequences

These transients are evaluated each refueling cycle to ensure proper shutdown margin requirements are specified. Table 14.4-4 shows the results of a typical cycle-specific bounding analysis.

Radiological consequences are not evaluated for this transient because a complete loss of shutdown margin and subsequent fuel clad damage are not expected to occur.

14.4.5 Start-Up of an Inactive Reactor Coolant Loop

14.4.5.1 Identification of Cause and Frequency Classification

Since there are no isolation valves or check valves in the reactor coolant system, operation of the plant with an inactive loop causes reversed flow through that loop. If there is a thermal load on the steam generator in the inactive loop, the hot leg coolant in that loop will be at a lower temperature than the core inlet temperature. The startup of the pump in the idle loop results in a core flow increase and the injection of colder water into the core. This could cause a rapid reactivity insertion and power increase.

This event is classified as a Condition II event (moderate frequency).

14.4.5.2 Expected Plant Response

This subsection describes the actual sequence of events and expected system response to a Startup of an Inactive Loop. It does not represent assumptions, requirements or equipment used in the analysis.

If the reactor is operated at power with one inactive loop, there is reverse flow through the inactive loop due to the pressure difference across the reactor vessel. The cold leg temperature in the inactive loop is identical to the cold leg temperature of the active loop and to the reactor core inlet temperature. If there is a temperature drop across the steam generator in the inactive loop the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature. The starting of the idle reactor coolant pump results in the injection of colder water into the core and could cause a rapid reactivity/power increase. However, for power on the order of 10% the hot leg temperature of the inactive loop is close to the core inlet temperature, thus limiting the severity of the resulting transient.

Administrative procedures prohibit continuous operation of the plant with one inactive loop. Should the loss of one reactor coolant pump occur during Mode 1, Power Operation, the Reactor Protection System will automatically trip the reactor if the power level is above the P8 permissive. Below the P8 permissive, the Operating Instructions require an administrative shutdown. This is adequate because of the very large power margin to the core safety limits. Moreover, operation below the P8 permissive can be accommodated with only natural circulation. Power operation below the P8 permissive, with one loop out of service, is only permitted for a few hours to either perform a special test under controlled conditions or to allow the operator to proceed with an orderly shutdown as discussed above. No changes in safety settings are required since there are no planned power operations with a loop out of service.

The actual plant response to starting an inactive reactor coolant pump below the P8 permissive depends on many factors. The most significant factor is the moderator temperature coefficient, the more negative the coefficient the larger the resultant power spike. If the initial power level is near the P8 permissive, it is possible that the power spike could enable the low flow trip before the RCS flow increases above the trip setpoint. This would result in a plant trip. If the conditions are such that the plant does not trip, and there is a negative temperature coefficient, there will be a temporary cooldown causing a power spike. Assuming the turbine controls are set to maintain the initial power, the average RCS temperature will return to its initial value with a higher core inlet temperature.

14.4.5.3 Analysis of Transient

14.4.5.3.1 Methodology

The Prairie Island Nuclear Generating Plant Technical Specifications require that both reactor coolant pumps (RCPs) be operating when the reactor is in Mode 1 or Mode 2. One pump operation is not permitted except for startup and physics tests when the thermal power is less than the P-7 reactor trip interlock. In the event that one RCP trips in Mode 1 or 2, the Technical Specifications require the plant to be in Mode 3 within six hours. If an RCP trips above P-7, an automatic reactor trip will be initiated. As the Technical Specifications require both RCPs to be operating in Modes 1 or 2 when not performing tests, an analysis of this event is not necessary.

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14.4.6 Excessive Heat Removal Due to Feedwater System Malfunction

14.4.6.1 Identification of Cause and Frequency Classification

A change in steam generator feedwater conditions that results in an increase in feedwater flow or a decrease in feedwater temperature could result in excessive heat removal from the plant primary coolant system. Such changes in feedwater flow or feedwater temperature are a result of a failure of a feedwater control valve or feedwater bypass valve, failure in the feedwater control system, or operator error.

The occurrence of these failures that result in an excessive heat removal from the plant primary coolant system cause the primary-side temperature and pressure to decrease significantly. The existence of a negative moderator and fuel temperature reactivity coefficients, and the actions initiated by the reactor rod control system can cause core reactivity to rise, as the primary-side temperature decreases. In the absence of a reactor trip or other protective action, this increase in core power, coupled with the decrease in primary-side pressure, can challenge the core thermal limits.

Feedwater Temperature Reduction

An extreme example of excessive heat removal from the RCS is the transient associated with the accidental opening of the feedwater bypass valve, which diverts flow around the low-pressure feedwater heaters. The function of this valve is to maintain net positive suction head on the main feedwater pump in the event that the heater drain pump flow is lost, such as following a large load reduction. In the event of an accidental opening of the feedwater bypass valve, there is a sudden reduction in feedwater inlet temperature to the steam generators. This increased subcooling would create a greater load demand on the RCS due to the increased heat transfer in the steam generator.

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

The net effect on the RCS due to a reduction in feedwater temperature is similar to the effect of increasing secondary steam flow; that is, the reactor will reach a new equilibrium condition at a power level corresponding to the new steam generator ΔT. If the increase in reactor power is large enough, primary RPS trip functions such as high neutron flux, OTΔT, and OPΔT trips will prevent any power increases that could lead to a DNBR lower than the safety analysis limit value. The RPS trip may not actuate if the increase in power is not large enough.

Feedwater Flow Increase

Another example of excessive heat removal from the RCS is a common-mode failure in the feedwater control system that leads to the accidental opening of the feedwater regulating valves to the steam generators.

Accidental opening of one or two feedwater regulating valves results in an increase of feedwater flow to one or two steam generators, causing excessive heat removal from the RCS. This also causes a decrease in FW enthalpy due to the higher velocity of the fluid when passing through the FW heaters. For Prairie Island, the heaters are located before the pipe split between the two steam generators, hence both loops will be affected by the decrease in FW enthalpy, even if only one feedwater regulating valve fails.

At power, excess feedwater flow causes a greater load demand on the primary side due to increased subcooling in the steam generator. With the plant at zero-power conditions, the addition of relatively cold feedwater may cause a decrease in primary-side temperature, and, therefore, a reactivity insertion due to the effects of the negative moderator temperature coefficient. The resultant decrease in the average temperature of the core causes an increase in core power due to moderator and control system feedback. This transient is attenuated by the thermal capacity of the primary and secondary sides. If the increase in reactor power is large enough, primary RPS trip functions such as high neutron flux, OTΔT, or OPΔT will prevent any power increase that can lead to a DNBR less than the safety analysis limit value. The RPS trip functions may not actuate if the increase in power is not large enough.

Continuous addition of cold feedwater after a reactor trip is prevented since the reduction of RCS temperature, pressure, and pressurizer level leads to the actuation of safety injection on low pressurizer pressure. The safety injection signal trips the main feedwater pumps, closes the feedwater pump discharge valves, and closes the main feedwater valves.

This event is classified as a Condition II event (moderate frequency).

14.4.6.2 Expected Plant Response

This subsection describes the actual sequence of events and expected system response to a Feedwater System Malfunction. It does not represent assumptions, requirements, or equipment used in the analysis.

The feedwater malfunction event leads to an increased feedwater flow and/or a reduced feedwater temperature. This increased sub-cooling would create a greater load demand on the Reactor Coolant System due to the increased heat transfer in the steam generator. The resultant reduction in the RCS average temperature is sensed by the rod control logic which would then step the control rods out in an attempt to return T_{ave} to the programmed value. This rod motion and the reduction in T_{ave}, combined with a negative MTC, causes reactor power to increase. Such transients are attenuated by the thermal capacity of the secondary system and of the Reactor Coolant System. While the increase in reactor power reduces the margin to DNB, the decrease in RCS temperature partially offsets this reduction. Depending on the magnitude of the MTC a reactor trip may or may not occur on NIS high power or overpower ΔT.

Continuous addition of cold feedwater after a reactor trip is prevented since the reduction of Reactor Coolant System temperature and pressure leads to the actuation of safety injection on low pressurizer pressure. The safety injection signal trips the main feedwater pumps (which closes the feedwater pump discharge valves), closes main feedwater flow control valves (Regulating and Bypass), and closes the feedwater containment isolation valves. These automatic actions prevent continuous cold feedwater addition after the reactor trip.

14.4.6.3 Analysis of Transient

14.4.6.3.1 Methodology

The computer codes and methods used to analyze this transient are described in section 14.3.

Analyses discussed in the FSAR showed that the maximum reactivity insertion rate which occurs at no load following excessive feedwater addition is less than the maximum value considered in the analysis of a RCCA withdrawal incident from a subcritical condition. Therefore, the reload analyses do not explicitly analyze excessive feedwater addition at no load.

An evaluation of the accidental full opening of a feedwater control valve at full power has shown that the consequences of this incident are no more severe than those resulting from the opening of the feedwater heater bypass valve. Therefore, only the opening of the high pressure feedwater heater bypass valve is analyzed. This is accomplished by assuming an instantaneous decrease in the feedwater enthalpy entering the steam generators.

The computer codes used to analyze this transient are described in Section 14.3.

Feedwater Temperature Reduction

An evaluation method was applied that demonstrates the decreased enthalpy caused by the feedwater temperature reduction is bounded by an equivalent enthalpy reduction that results from an excessive load increase incident (see USAR Section 14.4.7). No explicit analysis is performed.

Feedwater Flow Increase

The feedwater flow increase analysis is performed to demonstrate that the DNB design basis is satisfied, by showing that the calculated minimum DNBR is greater than the safety analysis limit DNBR. The overall analysis process is described as follows:

The system response to a feedwater flow increase transient is analyzed using the RETRAN computer code, which is described in Section 14.3. The results from the RETRAN computer code are used to determine if the DNBR safety analysis limit is met.

Feedwater system failures including the accidental opening of the feedwater regulating valves have the potential of allowing increased feedwater flow to one or two steam generators that will result in excessive heat removal from the RCS. Therefore, it is assumed that one or two feedwater control valves fail in the fully open position allowing the maximum feedwater flow to one or two steam generators.

Cases with and without automatic rod control initiated at hot full-power (HFP) conditions were considered. Also addressed is the initiation of a feedwater malfunction event from a hot zero-power (HZP) condition.

14.4.6.3.2 Key Physics Parameter Assumptions

The following assumption is made for the analysis of the feedwater malfunction event involving the accidental opening of one or two feedwater regulating valves:

a. Maximum (end of life) reactivity feedback conditions with a minimum Doppler-only power defect is conservatively assumed.

14.4.6.3.3 Key System Parameter Assumptions

The following assumption is made for the analysis of the feedwater malfunction event involving the accidental opening of one or two feedwater regulating valves:

- The plant is operating at full-power conditions (and no-load conditions for the HZP case) with the initial reactor power, pressure, and RCS average temperatures assumed to be at the nominal values.
- Uncertainties in initial conditions are included in the DNBR limit calculated using the RTDP methodology (Reference 4), where applicable (full-power cases).
- The feedwater temperature of 434.9°F for the full-power cases is consistent with normal plant conditions.
- The excessive feedwater flow event assumes accidental opening of the feedwater control valves with the reactor at full power with automatic and manual rod control, and zero power while modeling post reactor trip conditions with minimum shutdown margin. The feedwater flow malfunction results in a step increase to 1540 lbm/sec and 20°F temperature reduction for the full-power cases and step increase to 1950 lbm/sec for the zero-power cases. Both the single- and multi-loop failures are analyzed.
- The heat capacity of the RCS metal and steam generator shell are ignored, thereby maximizing the temperature reduction of the RCS coolant.
- The RPS features including OPΔT and turbine trip on hi-hi steam generator water levels are available to provide mitigation of the feedwater system malfunction transient.

14.4.6.3.4 Single Active Failure Assumptions of a Safety Grade Component

The limiting single failure is 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.

14.4.6.4 Acceptance Criteria

Acceptance criteria is applied in the analysis of the ANS Condition II Feedwater System Malfunction event are as follows:

1. Pressure in the reactor coolant and main steam systems shall be maintained below 110% of the respective design values. As the assumptions made in the analysis are defined to minimize the resulting DNBR, the RCS and MSS overpressurization limits are not challenged. The peak RCS and MSS pressures for this event are bounded by those calculated for the Loss of Load/Turbine Trip event.

2. The minimum departure from nucleate boiling ratio (DNBR) must be greater than the applicable limit for the DNBR correlation being used accounting for the penalties and factors described in Section 3.2.2, “Thermal Hydraulic Design Analysis”.

3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently. This criterion is met by demonstrating that the pressurizer does not become water-solid. As these events result in RCS cooldown, pressurizer filling is not a concern.

4. An incident of moderate frequency in combination with any single active component failure shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. As fuel failure is precluded if the DNBR criterion is satisfied, this criterion is met if criterion 2 above is met.

14.4.6.5 Results and Radiological Consequences

Feedwater Temperature Reduction

The opening of a high-pressure feedwater heater bypass valve causes a reduction in feedwater temperature that increases the thermal load on the primary system. The expected reduction in feedwater temperature resulting from the opening of a high-pressure feedwater heater bypass valve is less than 70°F. It was determined that this is bounded by the temperature reduction associated with the 20% step load increase incident analyzed in Section 14.4.7. Hence, the excessive increase in steam flow transient bounds the feedwater temperature reduction transient. As an explicit analysis was not performed, there are no transient results provided.

Feedwater Flow Increase

The results of the feedwater flow increase analysis demonstrate that both the HFP cases and zero-power case meet the applicable DNBR acceptance criteria.

The most limiting case is the excessive feedwater flow to both loops, from a full-power initial condition with automatic rod control. This case gives the largest reactivity feedback and results in the greatest power increase. A turbine trip, which results in a reactor trip, is actuated when the steam generator water level in either steam generator reaches the hi-hi water level setpoint. Assuming the reactor to be in manual rod control results in a slightly less severe transient. The rod control system is not required to function for this event. However, assuming that the rod control system is operable yields a slightly more limiting transient.

The excessive feedwater flow from a zero-power condition models HZP post-trip condition (that is, HZP stuck rod coefficients, minimum shutdown margin) with maximum reactivity feedback conditions (end of life). The limiting HZP feedwater malfunction conditions are bounded by those generated for the steamline break - core response analysis performed at similar, zero-power conditions. Since the zero-power steamline break - core response analysis, documented in Section 14.5.5, is shown to meet the DNBR acceptance criterion, it is concluded that the DNB design basis is met for the feedwater malfunction event (resulting in an increase in feedwater flow) at zero-power conditions. Therefore, no transient results are presented for this case, as no explicit analysis is performed.

The consequence of a feedwater flow increase transient is a turbine trip. Following turbine trip, the reactor will automatically be tripped, either directly due to the turbine trip or due to one of the reactor trip signals discussed in the section relative to the Loss of External Electrical Load. If the reactor was in automatic rod control, the control rods would be inserted at the maximum rate following the turbine trip, and the resulting transient would not be limiting in terms of peak RCS of MSS pressure.

Tables 14.4-7 and 14.4-8 show the time sequence of events and key results for the various analyzed cases.

Figures 14.4-30 through 14.4-34 show transient responses with the replacement steam generators (RSGs) for various system parameters for the most limiting case, i.e., a feedwater flow increase to both loops, initiated from HFP conditions with automatic rod control.

Conclusions

The feedwater temperature reduction transient (accidental opening of the feedwater heater bypass valve) was determined to be less severe than the excessive load increase incident (see Section 14.4.7); no explicit analysis was performed. Based on results for the excessive load increase incident, the applicable acceptance criteria for the feedwater temperature reduction transient have been met.

Analyses of the accidental opening of the feedwater regulating valve(s) were performed from a full-power initial condition with and without automatic rod control, and from a zero-power initial condition. All analyses considered single- and multi-loop failures. The feedwater malfunction event analyzed for an increase in feedwater flow from zero-power initial conditions was determined to be less severe than the steamline break - core response analysis performed in Section 14.5.5. Based on the results of the steamline break - core response event analysis, the applicable acceptance criteria for the feedwater malfunction event at zero-power resulting in an increase in feedwater flow are met. The feedwater malfunction event analyzed for an increase in feedwater flow from full-power initial conditions has been analyzed to show that the minimum DNBR for all cases meets the safety analysis minimum DNBR limit. Therefore, the DNB design basis is satisfied and no fuel damage is predicted.

Radiological consequences are not evaluated for this transient because no fuel pins are expected to experience departure from nucleate boiling and thus experience cladding failure.

The feedwater malfunction event for an increase in feedwater flow from full power has also been evaluated for operation up to a NSSS power of 1690 MWt with reduced power measurement uncertainty (102% of 1657 MWt). The evaluation confirmed that the results for operation at the increased power conditions are less limiting than the results of the current analysis of record. Therefore, the conclusions of the current analysis of record remain applicable for operation up to a NSSS power of 1690 MWt or less.

14.4.7 Excessive Load Increase Incident

14.4.7.1 Identification of Cause and Frequency Classification

An excessive load increase incident is defined as a rapid increase in steam generator steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. It 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 control system.

This event is classified as a Condition II event (moderate frequency).

14.4.7.2 Expected Plant Response

This subsection describes the actual sequence of events and expected system response to Excessive Load Increase Incident. It does not represent assumptions, requirements, or equipment used in the analysis.

The increase in steam flow due to the excessive load increase causes a power mismatch which causes a decrease in reactor coolant temperature. This decrease in reactor coolant temperature results in a core power increase due to Doppler and moderator feedback and/or control system action.

The Reactor Control System is designed to accommodate a small step load increase or a slow ramp load increase without a reactor trip (see section 7.2). Any loading rate in excess of these values may cause a reactor trip actuated by the Reactor Protection System. If the load increase exceeds the capability of the Reactor Control System, the transient is terminated in sufficient time to prevent the DNBR from being reduced below the design limit.

For excessive loading by the operator or by system demand, the turbine load limiter keeps maximum turbine load from exceeding 100% rated load.

During power operation, steam dump to the condenser is controlled by reactor coolant condition signals; i.e., high reactor coolant temperature indicates a need for steam dump. 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.

14.4.7.3 Analysis of Transient

14.4.7.3.1 Methodology

The excessive load increase transient is analyzed using the RETRAN computer code, which is described in Section 14.3.

Only steam flow increases within the capability of the turbine control valves are considered here; larger flow increases are considered as main steam line rupture accidents which are discussed in USAR section 14.5.5.

The transient is initiated by imposing a rapid increase in steam flow to 120% of rated full power flow. For consistency with the original FSAR analysis, four cases are evaluated: moderator reactivity coefficient at minimum and maximum; manual and automatic reactor control.

14.4.7.3.2 Key Physics Parameter Assumptions

The following key physics parameter assumptions are made in analyzing the Excessive Load Increase event.

When analyzing cases with maximum reactivity feedback conditions:

a. Moderator Density Coefficient: a most positive value is assumed.

b. Doppler Temperature Coefficient: a most negative value is assumed.

c. Doppler Power Defect: a most negative value is assumed.

d. Effective Delay Neutron Fraction: a minimum value is assumed.

When analyzing cases with minimum reactivity feedback conditions:

a. Moderator Density Coefficient: a value of 0.0 Δk/g/cc is assumed.

b. Doppler Temperature Coefficient: a value of 0.0 pcm/°F is assumed.

c. Doppler Power Defect: a least negative value is assumed.

d. Effective Delayed Neutron Fraction: a maximum value is assumed.

14.4.7.3.3 Key System Parameter Assumptions

The following key system parameter assumptions are made to ensure the overall results of the analysis bound actual plant operation:

a. Initial conditions of core power, RCS coolant temperature, pressurizer pressure and steam generator level are assumed to be at their nominal values consistent with steady-state full power operation. Uncertainties in the initial conditions of these parameters are not considered, consistent with the application of the RTDP methodology (Reference 4).

b. Minimum measured flow is assumed, consistent with the RTDP methodology.

c. 0% SGTP level is assumed; this maximizes primary-to-secondary heat transfer and results in a more severe RCS cooldown transient.

d. The pressurizer sprays and PORVs are assumed to be operational.

Cases are analyzed with the rod control system assumed to be in both automatic and manual modes.

No reactor protection function or emergency safety features function is credited in the analysis to demonstrate the inherent transient capability of the plant. Under actual operating conditions, a reactor trip may occur after which the plant would quickly stabilize.

14.4.7.3.4 Single Active Failure Assumptions of a Safety Grade Component

The analysis of an Excessive Load increase transient may or may not result in a reactor trip. If the analysis does result in a reactor trip, the worst case single failure would be the failure of a reactor protection train. However, the reactor protection system is designed such that any single failure does not prevent proper operation of the protection system (see USAR section 7.4). Therefore the analysis assumes that the reactor protection system operates as designed. If the analysis does not result in a reactor trip, there is no actuation of active safety grade components required for mitigation of the transient. Consequently, no single failure assumption would be applied.

14.4.7.4 Acceptance Criteria

1. The maximum reactor coolant and main steam system pressure must not exceed 110% of their design values.

2. The minimum departure from nucleate boiling ratio (DNBR) must be greater than the applicable limit for the DNBR correlation being used accounting for the penalties and factors described in Section 3.2.2, “Thermal Hydraulic Design Analysis”.

14.4.7.5 Results and Radiological Consequences

The calculated sequence of events for the cases of Excessive Load Increase incident is shown in Table 14.4-9. The transient response for the limiting case (minimum reactivity feedback with automatic rod control) is shown in Figures 14.4-35 through 14.4-40. The results demonstrate that the minimum DNBR remains above the safety analysis limit for all cases, and the reactor coolant system overpressure limit is not challenged. Additionally, as the event is initiated by an increase in the main steam system flow rate, which results in an overcooling of the RCS and a decrease in the main steam system pressure, the main steam system overpressure limit is not challenged.

Radiological consequences are not evaluated for this transient because no fuel pins are expected to experience departure from nucleate boiling and thus experience cladding failure.

14.4.8 Loss of Reactor Coolant Flow

14.4.8.1 Flow Coast Down

14.4.8.1.1 Identification of Cause and Frequency Classification

A loss of coolant flow incident can result from a mechanical or electrical failure in one or more reactor coolant pumps (RCPs) or from a fault in the power supply to these pumps.

This event is classified as a Condition III event (infrequent frequency) when analyzing the loss of both pumps, and a Condition II event (moderate frequency) when analyzing the loss of just one pump.

14.4.8.1.2 Expected Plant Response

This subsection describes the actual sequence of events and expected system response to a Flow Coast Down transient. It does not represent assumptions, requirements, or equipment used in the analysis.

Simultaneous loss of electrical power to all reactor coolant pumps at full power is the most severe credible loss-of-coolant flow condition. For this condition, the reactor trip together with flow sustained by the inertia of the coolant and RCPs is sufficient to prevent fuel failures and RCS over pressurization.

As a result of loss of driving head supplied by the RCPs, the coolant flow through the core begins to decrease. The hydraulic inertia of the fluid and the flywheels on the pump motors retard this decrease in flow rate. If the reactor is at power at the time of the incident, the immediate effect of loss of coolant flow is a rapid increase in coolant temperature. This increase along with the decrease in coolant flow could result in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor is not tripped promptly. The reactor trips early in the event on one of the low flow trips described in section 7.4.

14.4.8.1.3 Analysis of Transient

14.4.8.1.3.1 Methodology

Two types of loss of flow accidents were analyzed: complete loss of flow due to the loss of two RCPs and partial loss of flow due to the loss of one RCP. However, since the partial loss of flow event is much less limiting than the complete loss of flow event, only the complete loss of flow analysis is presented herein.

This transient is analyzed with two computer codes, which are described in Section 14.3. First, the RETRAN computer code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE computer code is then used to calculate the heat flux and DNBR transients based on the nuclear power and RCS flow from RETRAN. The DNBR transient presented represents the minimum of the typical or thimble cell for the fuel.

This event is analyzed with the Revised Thermal Design Procedure (RTDP) (Ref. 4).

With respect to the overpressure evaluation, the Loss of Flow events are bounded by the Loss of Load/Turbine Trip events, in which assumptions are made to conservatively calculate the RCS and MSS pressure transients. For the Loss of Flow events, turbine trip occurs following reactor trip, whereas for the Loss of Load/Turbine Trip event, the turbine trip is the initiating fault. Therefore, the primary to secondary power mismatch and resultant RCS and MSS heatup and pressurization transients are always more severe for the Loss of Load/Turbine Trip event. For this reason, no attempt is made to calculate the maximum RCS or MSS pressure for the Loss of Flow events.

14.4.8.1.3.2 Key Physics Parameter Assumptions

The following physics parameters are reviewed each refueling cycle to ensure that the individual parameter used in the analysis is bounding. If it is not bounded, an evaluation is performed to ensure the analysis would bound a cycle specific analysis or a new analysis is performed.

a. Isothermal Temperature Coefficient (ITC). A zero ITC is assumed since this maximizes the heat flux during the initial part of the transient, when the minimum DNBR is reached.

b. Doppler Power Coefficient. The largest negative value of the Doppler Power Coefficient is assumed so as to maximize the core power.

c. Scram Reactivity Core. A conservative slow scram curve is assumed. Also, it is assumed that the most reactive RCCA is stuck in its fully withdrawn position.

d. Nuclear Enthalpy Rise Hot Channel Factor. Is assumed to remain within the limits as defined in the Technical Specification for allowable combinations of axial offset and power level.

e. Delayed Neutron Fraction. A conservative maximum value is assumed.

14.4.8.1.3.3 Key System Parameter Assumptions

The following key system parameter assumptions are made to ensure the overall results of the analysis bound actual operation.

a. Consistent with the RTDP methodology, the initial operating conditions are assumed to be at their nominal values, including the steady-state power level, RCS pressure, and RCS vessel average temperature. Minimum Measured Flow (MMF) is also assumed.

b. A reactor trip is actuated by the low coolant flow reactor trip function. The time from the initiation of the low flow signal to initiation of RCCA motion is 1.2 seconds. The trip signal is assumed to be initiated at 87% of full loop flow.

c. No credit is taken for the reactor trip on reactor coolant pump motor breaker open due to low voltage, or the direct reactor trip on undervoltage.

14.4.8.1.3.4 Single Active Failure Assumptions of a Safety Grade Component

Mitigation of the Loss of Reactor Coolant flow transient is accomplished by a reactor trip on low flow. The reactor trip on RCP under voltage and RCP breaker trip utilize non-safety grade components. Therefore, failure of the undervoltage or RCP breaker trips in the analysis does not constitute the single failure of a safety grade component.

The worst case single failure for the Loss of Reactor Coolant flow transient is the failure of a reactor protection train. However, the reactor protection system is designed such that any single failure does not prevent proper operation of the protection system (see USAR section 7.4). Therefore the analysis assumes that the reactor protection system operates as designed.

14.4.8.1.4 Acceptance Criteria

Based on the expected frequency of occurrence, a complete loss of flow transient is considered to be a Condition III event, an infrequent incident, as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Pressurized Water Reactor Plants," ANSI N18.2-1973. However, more restrictive Condition II criteria are applied to the Complete Loss of Flow event. These events are:

1. The maximum reactor coolant and main steam system pressure must not exceed 110% of their design values. As discussed above, the Loss of Flow event is bounded by the Loss of Load/Turbine Trip event with respect to RCS and MSS overpressure.

2. The minimum departure from nucleate boiling ratio (DNBR) must be greater than the applicable limit for the DNBR correlation being used accounting for the penalties and factors described in Section 3.2.2, "Thermal Hydraulic Design Analysis".

14.4.8.1.5 Results and Radiological Consequences

The reactor coolant flow coastdown for the Complete Loss of Flow event is presented in Figures 14.4-41 and 14.4-42 for 400V+ fuel. Figures 14.4-41a and 14.4-42a show the same results for cores containing 422V+ fuel or mixed cores. Reactor coolant flow is calculated based on a momentum balance in the Reactor Coolant System combined with a pump momentum balance. The nuclear power and core average heat flux transients are presented in Figures 14.4-43 and 14.4-44 for 400V+ fuel. Figures 14.4-43a and 14.4-44a show the same results for cores containing 422V+ fuel or mixed cores. The pressurizer pressure and RCS loop temperature transients are shown in Figures 14.4-45 and 14.4-46 for 400V+ fuel. Figures 14.4-45a and 14.4-46a show the same results for cores containing 422V+ fuel or mixed cores. Finally, the hot channel heat flux and DNBR transients are presented in Figures 14.4-47 and 14.4-48 for 400V+ fuel. Figures 14.4-41a and 14.4-42a show the same results for cores containing 422V+ fuel or mixed cores.

For cores containing all 400V+ fuel, the minimum DNBR is slightly less than the safety analysis limit value as shown in Figure 14.4-48. However, there is sufficient margin available between the DNBR design limit and safety analysis limit to ensure that the DNB design basis is satisfied. For cores containing 422V+ fuel or a mixed core, the analysis demonstrates that the minimum DNBR is greater than the safety analysis limit value as shown in Figure 14.4-48a. Therefore, it can be concluded that the DNB design limit is satisfied for all combinations of either 400V+ fuel or 422V+ fuel.

14.4.8.2 Locked Pump Rotor

14.4.8.2.1 Identification of Cause and Frequency Classification

The accident postulated is the instantaneous seizure of the rotor of a single reactor coolant pump.

This event is classified as a Condition IV event (limiting fault).

14.4.8.2.2 Expected Plant Response

This section describes the actual sequence of events and expected system response to a Locked Pump Rotor transient and does not represent assumptions, requirements, or equipment used in the analysis.

This transient is due to the hypothetical instantaneous seizure of a reactor coolant pump rotor. Flow through the reactor coolant system 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 resulting in a pressure increase throughout the Reactor Coolant System. The surge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves. The sudden decrease in core flow while the reactor is at power could result in a degradation of core heat transfer and departure from nucleate boiling in some of the fuel rods.

14.4.8.2.3 Analysis of Transient

14.4.8.2.3.1 Methodology

The Locked Rotor transient is analyzed with two primary computer codes, which are described in Section 14.3. First, the RETRAN computer code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE code is then used to calculate the DNBR (for determination of the percentage of “Rods in DNB”) and peak cladding temperature using the nuclear power and RCS flow from RETRAN.

At the beginning of the postulated RCP Locked Rotor accident, the plant is assumed to be in operation under the most adverse steady state operating conditions, i.e., a maximum steady state thermal power, maximum steady state pressure, and maximum steady state coolant average temperature. The analysis is performed to bound operation with a maximum uniform steady steam generator tube plugging level of 25%.

The Locked Rotor transient is initiated from full power by abruptly seizing one of the reactor coolant pump shafts. The rotor is assumed to be locked for forward flow and free-spinning for reverse flow. This represents the most limiting condition for the Locked Rotor/Shaft Break accidents. The analysis assumes that the other reactor coolant pump continues to operate throughout the event.

Two separate analyses are performed. The first maximizes the RCS and steam generator pressure response. The seconds maximizes the number of fuel pins that may experience departure from nucleate boiling.

14.4.8.2.3.2 Key Physics Parameter Assumptions

The following physics parameters are reviewed each refueling cycle to ensure that the individual parameter used in the analysis is bounding. If it is not bounded, an evaluation is performed to ensure the analysis would bound a cycle specific analysis or a new analysis is performed.

a. Isothermal Temperature Coefficient (ITC). A zero ITC is assumed since this maximizes the heat flux during the initial part of the transient, when the minimum DNBR is reached.

b. Doppler Power Coefficient. The largest negative value of the Doppler Power Coefficient is assumed so as to maximize the core power.

c. Scram Reactivity Curve. A conservative trip reactivity worth versus rod position was modeled in addition to a rod drop time that is conservative for the reduced core flow at the time of the trip. These assumptions are used to minimize the effect of rod insertion following reactor trip and maximize the heat flux statepoint used in the DNBR evaluation for this event. The assumed trip reactivity is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position.

d. Delayed Neutron Fraction. A conservative maximum value is assumed.

14.4.8.2.3.3 Key System Parameter Assumptions

The following key system parameter assumptions are made to ensure the overall results of the analysis bound actual operation.

a. Consistent with the RTDP methodology, the initial operating conditions are assumed to be at their normal values for the Rods-in-DNB case, including the steady-state power level, RCS pressure, and RCS vessel average temperature. Minimum Measured Flow (MMF) is also assumed. For the peak pressure case, the nominal values plus uncertainties are modeled, and Thermal Design Flow (TDF) is assumed.

b. A reactor trip is actuated by low flow. The time from the initiation of the low flow signal to initiation of RCCA motion is 1.2 seconds. The trip signal is assumed to be initiated at 87% of full loop flow.

c. The pressurizer PORVs and pressurizer spray systems are disabled for both the Rods-in-DNB and peak RCS/MSS pressure cases. No credit is taken for the increase in system pressure for the calculation of the number of Rods-in-DNB. Therefore, disabling the pressurizer PORVs and spray system has no effect.

14.4.8.2.3.4 Single Active Failure Assumptions of a Safety Grade Component

Mitigation of the Locked Pump Rotor transient is accomplished by a reactor trip along with the relief of excess pressure through the safety valves.

The worst case single failure for the Locked Pump Rotor transient is the failure of the reactor protection train. However, the reactor protection system is designed such that any single failure does not prevent proper operation of the protection system (see USAR section 7.4). Therefore the analysis assumes that the reactor protection system operates as designed.

14.4.8.2.4 Acceptance Criteria

The reactor coolant pump rotor seizure (Locked Rotor) or Shaft Break accident is classified as a Condition IV event, a limiting fault, as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973. Condition IV events are limiting faults that are not expected to occur, but are postulated because their consequences would include the potential for release of significant amounts of radioactive material.

The acceptance criteria used for this event are:

1. The maximum reactor coolant and main steam system pressure must not exceed 110% of their design values (see Table 14.3-3). The RCS overpressure limit used in this analysis is 2748.5 psia for 400V+ fuel and 2750 psia for 422V+ fuel or mixed cores. The Locked Rotor/Shaft Break event is bounded by the Loss of Load/Turbine Trip event with respect to main steam system pressure due to the early reactor trip. 01469415

2. To ensure a coolable core geometry is maintained, the maximum clad temperature calculated to occur at the core hot spot must not exceed 2700°F, for Standard ZIRLO fuel cladding and 2375°F for Optimized ZIRLO fuel cladding and the local zirconium-water reaction must remain below 16% by weight.

3. The number of fuel rods calculated to experience a DNBR less than the DNBR correlation limit being used (accounting for penalties and factors described in section 3.2.2 “Thermal Hydraulic Design Analysis”) shall not exceed the number which are expected to fail such that the dose due to released activity will exceed the limits of 10CFR50.67. The limit is 20%.

14.4.8.2.5 Results and Radiological Consequences

Figures 14.4-49 through 14.4-56 illustrate the transient response for the Locked Rotor event for 400V+ fuel. Figures 14.4-49a through 14.4-56a show the same results for cores containing 422V+ fuel or mixed cores. The results shown are for the peak RCS pressure/peak clad temperature (PCT) case. The coolant flow through the core is rapidly reduced to less than fifty percent of its initial value (Figure 14.4-49 and 14.4-49a). As shown in Figure 14.4-54 and 14.4-54a, the peak RCS pressure is less than 110% of the design value. Figure 14.4-56 and 14.4-56a shows that the peak cladding temperature is considerably less than the more restrictive limit of 2375°F (associated with Optimized ZIRLO fuel cladding), and the zirconium-water reaction at the hot spot meets the criterion of less than 16%.

Calculations performed with the VIPRE code demonstrate that the maximum percentage of rod-in-DNB for this event is less than 20%. This calculation is based upon the RTDP methodology and utilizes a generic rod census curve, and is confirmed for each reload cycle. This value is less than that assumed in the dose analysis, 20%.

Radiological Consequences:

In the locked rotor accident analysis in the FSAR, 20% of the fuel rods were predicted to experience a DNBR of less than the limit. 20% is used as input to the dose consequence analysis. It is not necessary to perform specific radiological analysis for the locked rotor accident provided the predicted number of failed fuel rods (DNBR < the design limit) is less than 20%.

The key inputs and assumptions used in the Locked Rotor Accident (LRA) radiological consequence analysis analyzed using the Alternative Source Term (AST) are summarized below and provided in Table 14.4-13.

As a result of the accident, the primary coolant is contaminated by activity released from the gap spaces in failed fuel rods. Primary to secondary coolant leakage transfers activity into the Secondary Coolant side. This makes it available for release into the environment via steaming through the Power Operated Release Valves (PORV) and via the turbine driven auxiliary feedwater (TDAFW) pump steam exhaust.

Consistent with Regulatory Guide 1.183, the LRA dose assessment is analyzed for the following:

Dose Due to Postulated Damaged Fuel

The first case involves assumed clad damage to 20 percent of the fuel in the reactor core. In this scenario, it is assumed that all of the damaged fuel activity is homogeneously mixed in the primary coolant, prior to accident initiation.

Dose Due to Equilibrium Secondary Coolant System Iodine

This dose contribution is that which results from the SG Power Operated Relief Valves (PORV) and the Turbine Driven Auxiliary Feedwater (TDAFW) pump steam exhaust release of secondary coolant activity through the SGs. It has been shown that it is more conservative to assume the release is from the PORVs in lieu of the TDAFW pump steam exhaust. This release of secondary coolant activity, existing prior to the LRA accident, is analyzed, and the dose is added to the other modeled case.

Fuel Damage and Core Source Term

For conservatism, the LRA core source term is that associated with a power level of 1,852 MWth

The instantaneous seizure of the RCP rotor associated with the LRA results in the damage of 20% of the fuel. The design basis of this accident assumes that no fuel melt is postulated to occur. Therefore, for Case 1, the source term available for release is associated with this fraction of damaged fuel and the fraction of core activity existing in the gap, plus the pre-accident reactor coolant iodine and noble gas activity associated with 1% fuel defects. The modeling of the pre-accident reactor coolant iodine and noble gas activity associated with 1% fuel defects represents discretionary conservatism relative to the 0.5 μCi/gm DE I-131 and 580 μCi/gm DE Xe-133 equilibrium primary coolant activity concentration Technical Specification limits.

The additional source activity modeled in Case 2 consists of the 0.1 μCi/gm DE I-131 equilibrium secondary coolant activity concentration Technical Specification limit.

Release Rates and Partitioning Factors

Activity that originates in the primary coolant is released to the secondary coolant by means of the primary-to-secondary coolant leak rate. This design basis leak rate value is 150 gpd into each of the two steam generators (SGs). For input into RADTRAD, this rate was converted to 0.0188 cubic feet per minute into each SG.

The methodology used to model steaming of activity through SG PORVs following the postulated LRA assumes an average cumulative release rate through these paths. The partitioning factors are applied to these release rates. Incremental steam mass releases are given in pounds per time interval. For the time intervals used in this accident scenario, release rates were derived by taking the averages of mass releases over each specified time interval. Then these mass flow rates were converted to volumetric flow rates using the assumption of cooled liquid conditions (i.e., 62.4 lbm/ft^3), as specified by the applicable guidance of Regulatory Guide 1.183.

For all post-accident releases through the SG PORVs, the mechanism for release to the environment is steaming of the coolant in the secondary system. Because of this release dynamic, Regulatory Guide 1.183 allows for a reduction in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water. For Iodine, the partitioning factor of 0.01 was taken directly from the suggested guidance of Regulatory Guide 1.183. Reviewing the specified AST release fractions, it is concluded that the only nuclides to be released from the core source term, other than iodines, are noble gas nuclides, and because of the volatility of noble gases, no partitioning is assumed for any such isotopes.

Control Room χ/Q Calculations (Meteorology)

The limiting control room atmospheric dispersion factors for SG PORVs releases are weighted by their portion of the total mass release to determine mass release weighted average atmospheric dispersion factors that are used to model the steam releases.

Acceptance Criteria

According to Regulatory Guide 1.183, the EAB and LPZ dose acceptance criteria for a locked rotor accident is 2.5 rem TEDE, which is 10% of the 10 CFR 50.67 limit of 25 rem TEDE.

The control room dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67

Dose Results

Radiological doses resulting from a design basis locked rotor accident for a control room operator and a person located at EAB or LPZ are to be less than the regulatory dose limits as given below.

LRA Dose Results		
Location	Acceptance Criteria (rem)	TEDE (rem)
Exclusion Area Boundary	2.5	0.49
Low Population Zone	2.5	0.27
Control Room	5.0	4.33

14.4.9 Loss of External Electrical Load

14.4.9.1 Identification of Cause and Frequency Classification

The loss of external electrical load may result from an abnormal increase in network frequency, opening the main breakers from the generator which causes a rapid large Nuclear Steam Supply System load reduction by action of the turbine control, or from a trip of the turbine-generator.

This event is classified as a Condition II event (moderate frequency).

14.4.9.2 Expected Plant Response

This subsection describes the actual sequence of events and expected system response to a Loss of External Load transient. It does not represent assumptions, requirements, or equipment used in the analysis.

The most likely source of a complete loss of load on the Nuclear Steam Supply System is a trip of the turbine-generator. In this case there is a direct reactor trip signal (unless below P9 setpoint) derived from either the turbine autostop oil pressure or a closure of the turbine stop valves. Reactor coolant temperatures and pressures do not significantly increase if the steam bypass system and pressurizer pressure control systems are functioning properly.

As described in Section 7.2 of the USAR, the plant is designed to accept a large loss of load without actuating a reactor trip utilizing the Steam Dump and rod control systems. The reactor power is reduced to a new equilibrium power level at a rate consistent with the capability of the rod control system. The pressurizer power operated relief valves (PORVs) may actuate, but the pressurizer safety valves and steam generator safety valves do not lift.

In the event that the steam dump valves and steam generator PORVs fail to open following a large loss of load, the steam generator safety valves may lift, and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal, or the overtemperature ΔT signal. The steam generator shell side pressure and reactor coolant temperature will increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the RCS and steam generator against overpressure for all load losses without assuming the operation of the steam dump system, steam generator PORVs, pressurizer spray, pressurizer power-operated relief valves, automatic RCCA control, or direct reactor trip on turbine trip.

14.4.9.3 Analysis of Transient

14.4.9.3.1 Methodology

In this analysis, the behavior of the plant is evaluated for a complete loss of steam load (i.e., turbine trip) from full power without direct reactor trip. This is done 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. The turbine is assumed to trip without actuating all the turbine stop valve limit switches. This assumption delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst-case transient. In addition, 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 (except for long-term recovery) to mitigate the consequences of the transient.

The turbine trip transient is analyzed by employing the detailed digital computer code RETRAN, which is described in Section 14.3.

14.4.9.3.2 Key Physics Parameter Assumptions

The following physics parameters are reviewed each refueling cycle to ensure that the individual parameter used in the analysis is bounding. If it is not bounded, an evaluation is performed to ensure the analysis would bound a cycle specific analysis or a new analysis is performed.

a. Isothermal Temperature Coefficient: A conservative hot full power value of 0.0 pcm/°F is assumed.

b. Doppler Temperature Coefficient: A conservative least negative value is assumed.

c. Scram Reactivity Curve: A conservative slow scram curve is assumed.

d. Nuclear Enthalpy Rise Hot Channel Factor: Is assumed to remain within the limits as defined in the COLR for allowable combinations of axial offset and power level.

Maximum reactivity feedback cases have been determined to be non-limiting with respect to both DNB and peak pressure concerns. These cases, which were included in the original FSAR analysis, are no longer analyzed.

14.4.9.3.3 Key System Parameter Assumptions

The following key system parameter assumptions are made to ensure the overall results of the analysis bound actual operation.

a. Initial Operating Concerns

For the DNB case, the initial core power, reactor coolant temperature, and reactor coolant pressure are assumed to be at their normal values. The DNBR calculations are performed using the Revised Thermal Design Procedure (RTDP), in which the uncertainties in the initial conditions are included in the DNBR limit value, as described in Reference 4. To address an increase in the analytical core power up to 1683 MWt and corresponding reduction in the power measurement uncertainty to 0.36%, the DNB case was explicitly reanalyzed. For the peak pressure calculations, uncertainties of 2%, -60/+40 psi and 4.0°F are applied in the most limiting direction to the initial core power, reactor coolant pressure and reactor coolant system average temperature, respectively. The peak pressure analyses using the 2% power measurement uncertainty remain bounding for the increase in analytical core power since the power increase is offset by a decrease in the power measurement uncertainty.

b. Reactor Control

From the standpoint of the minimum DNBR and maximum pressures attained, it is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

c. Steam Releases

No credit is taken for the operation of the steam dump system or steam generator power-operated relief valves. The steam generator pressure rises to the point where steam release through safety valves occurs thus limiting the secondary steam pressure increase.

d. Pressurizer spray and power-operated relief valves

Two cases are analyzed:

1. For the DNB case, 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.

2. For the overpressure case, no credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are operable. This case conservatively accounts for the effects of the pressurizer safety valve loop seals, as discussed in Reference 32.

e. Feedwater flow

Main feedwater flow to the steam generators is assumed to be 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 is normally assumed to occur; however, the auxiliary feedwater pumps would be expected to start on a trip of the main feedwater pumps. 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, with no credit taken for the direct reactor trip on the turbine trip. Trip signals are expected due to high pressurizer pressure, overtemperature ΔT, high pressurizer water level, or low-low steam generator water level.

Except as discussed, normal reactor control systems and engineered safety features are not required to function. Cases are presented in which pressurizer spray and power-operated relief valves are assumed, but the more limiting cases where those functions are not assumed are also presented.

The reactor protection system may be required to function following a turbine trip. Pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressures below allowable limits. No single active failure will prevent operation of any system required to function.

14.4.9.3.4 Single Active Failure Assumptions of a Safety Grade Component

Mitigation of the Loss of External Load transient is accomplished by a reactor trip along with relief of excess pressure through the safety valves. Not taking credit for the direct reactor trip caused by a turbine trip does not constitute the single failure because this trip relies on non-safety grade components.

The worst case single failure for the Loss of External Load transient is the failure of a reactor protection train. However, the reactor protection system is designed such that any single failure does not prevent proper operation of the protection system (see USAR Section 7.4). Therefore, the analysis assumes that the reactor protection system operates as designed.

14.4.9.4 Acceptance Criteria

1. The maximum reactor coolant and main steam system pressure must not exceed 110% of their design values.

2. The minimum departure from nucleate boiling ratio (DNBR) must be greater than the applicable limit for the DNBR correlation being used accounting for the penalties and factors described in Section 3.2.2, “Thermal Hydraulic Design Analysis”.

14.4.9.5 Results and Radiological Consequences

The transient responses for a turbine trip from full power operation are shown for two cases that assume minimum reactivity feedback with and without automatic pressure control (Figures 14.4-57 through 14.4-67). The calculated sequence of events for this accident is shown in Table 14.4-10. Figures 14.4-57 through 14.4-62 show the transient responses for the total loss of steam load with minimum reactivity feedback, assuming full credit for the pressurizer spray and pressurizer power-operated relief valves.
No credit is taken for the steam dump. The reactor is tripped by overtemperature ΔT trip channels. The minimum DNBR remains well above the safety analysis limit value. The steam generator safety valves open and limit the secondary steam pressure increase.

The turbine trip accident was also studied assuming the plant to be initially operating at full power with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump. The reactor is tripped on the high pressurizer pressure signal. Figures 14.4-63 through 14.4-67 show the transient responses for this case. In this case, the pressurizer safety valves are actuated, and maintain system pressure below 110 percent of the design value. The steam generator safety valves open and limit the secondary steam pressure increase.

The results of the analyses show that the plant design is such that a turbine trip without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure-relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits. The integrity of the core is maintained by operation of the reactor protection system, i.e., the DNBR will be maintained above the safety analysis limit values. Therefore, the applicable acceptance criteria are met and the analysis demonstrates the ability of the NSSS to safely withstand a full load rejection.

Radiological consequences are not evaluated for this transient because no fuel pins are expected to experience departure from nucleate boiling and thus experience cladding failure.

14.4.10 Loss of Normal Feedwater

14.4.10.1 Identification of Cause and Frequency Classification

A loss of normal feedwater (from a pipe break that can be isolated from the SGs, pump failures, valve malfunctions, or loss of outside AC power) results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core. For other evaluations of a Feedwater Line Break, refer to section 11.9 (inside containment) and Appendix I (outside containment).

This event is classified as a Condition II event (moderate frequency).

14.4.10.2 Expected Plant Response

This subsection describes the actual sequence of event and expected system response to a Loss of Normal Feedwater transient. It does not represent assumptions, requirements, or equipment used in the analysis.

A loss of normal feedwater transient is characterized by a rapid reduction in steam generator water level which results in a reactor trip, a turbine trip, and auxiliary feedwater actuation by the protection system logic. Following the reactor trip the power quickly falls to decay heat levels.

If the loss of normal feedwater is a result of a loss of offsite power, power will also be lost to the reactor coolant pumps. The coolant flow necessary for core cooling and removal of residual heat is then maintained by natural circulation in the reactor coolant loops. The decay heat is transferred to the steam generators, either through the steam dump valves to the condenser/atmosphere, through the power-operated relief valves, or through the steam generator safety valves.

Until the decay heat level decreases to the point where the feedwater makeup requirements are less than the auxiliary feedwater flow, the water level in the steam generators will continue to decrease. Without adequate heat removal, the reactor coolant system temperature will rise causing an increase in RCS pressure. Prior to completely depleting the SG inventory, the auxiliary feedwater flow will exceed the makeup requirements and steam generator water level will recover.

14.4.10.3 Analysis of Transient

14.4.10.3.1 Methodology

The computer code used to analyze this transient is described in section 14.3.

The Loss of Normal Feedwater transient is analyzed using the RETRAN computer code. The RETRAN model simulates the RCS, neutron kinetics, pressurizer, pressurizer relief and safety valves, pressurizer heaters, pressurizer spray, steam generators, feedwater system, and main steam safety valves (MSSVs). The code computes pertinent plant variables including steam generator mass, pressurizer water volume, and reactor coolant average temperature.

The Loss of Normal Feedwater analysis is performed to demonstrate the adequacy of the Reactor Protection System to trip the reactor and auxiliary feedwater (AFW) system to remove long-term decay heat, stored energy, and RCP heat. This prevents excessive heatup or overpressurization of the RCS. As such, the assumptions used in the analysis are designed to maximize the time to reactor trip and to minimize the energy removal capability of the AFW system.

14.4.10.3.2 Key Physics Parameter Assumptions

The following core physics parameter assumptions are made for the Loss of Normal Feedwater analysis:

a. Isothermal Temperature Coefficient: A conservative hot full power value of 0.0 pcm/°F is assumed.

b. Doppler-only Power Coefficient: A conservative most negative value is assumed.

c. Effective Delayed Neutron Coefficient: A conservative maximum value is assumed.

d. Scram Reactivity Curve: A conservatively slow rod insertion curve is assumed.

14.4.10.3.3 Key System Parameter Assumptions

The following key system parameter assumptions are made to ensure the overall results of the analysis bound actual operation.

These assumptions maximize the possibility of water relief from the RCS by maximizing the expansion of the RCS inventory.

a. The plant is initially operating at the nominal NSSS power plus uncertainty. The RCP heat is a maximum constant value. The RCPs run throughout the transient.

b. The initial reactor coolant vessel average temperature is assumed to be the nominal full-power value minus uncertainty.

c. The initial pressurizer pressure is assumed to be the nominal value plus uncertainty.

d. The initial pressurizer water level is assumed to be the programmed full-power value plus uncertainty.

e. The initial steam generator water level is assumed to be the programmed full-power value plus uncertainty.

f. Reactor trip occurs on steam generator low-low water level at 0% NRS. Turbine trip occurs as a result of reactor trip.

g. One minute after the low-low steam generator water level setpoint is reached, a minimum constant AFW flow of 190 gpm is initiated from one AFW pump, with flow split equally between the two steam generators (equal flow split is the limiting case).

h. Secondary system steam relief is achieved through the main steam safety valves (MSSVs). The MSSV opening pressures are the nominal settings plus a 3% tolerance.

i. Normal reactor control systems are not assumed to be operable if their operation leads to less limiting analysis results. However, the pressurizer power-operated relief valves (PORVs), pressurizer heaters, and pressurizer sprays are assumed to operate normally, since this results in a conservative transient with respect to the peak pressurizer water volume.

j. A conservative core residual heat generation is assumed based on the ANS 5.1-1979 decay heat model including uncertainty (Reference 36).

14.4.10.3.4 Single Active Failure Assumptions of a Safety Grade Component

Mitigation of the loss of normal feedwater transient is accomplished by a reactor trip along with makeup to the steam generators from the Auxiliary Feedwater System.

The worst case single failure for the loss of normal feedwater transient is the failure of one of the two Auxiliary Feedwater pumps to start.

14.4.10.4 Acceptance Criteria

1. The maximum reactor coolant and main steam system pressure must not exceed 110% of their design values.

2. The minimum departure from nucleate boiling ratio (DNBR) must be greater than the applicable limit for the DNBR correlation being used accounting for the penalties and factors described in Section 3.2.2, “Thermal Hydraulic Design Analysis”.

3. The event shall not progress to a more serious plant condition without other faults occurring independently. This criterion is met by demonstrating that the AFW system provides sufficient heat removal to preclude the pressurizer from becoming water-solid due to coolant expansion, such that there is no water relief from the pressurizer. This ensures that long-term core cooling is maintained.

14.4.10.5 Results and Radiological Consequences

Figures 14.4-68 through 14.4-73 show the significant plant responses following a loss of normal feedwater. The calculated sequence of events and results are listed in Table 14.4-11.

Following the reactor and turbine trip from full load, the water level in each steam generator falls due to the reduction of the steam generator void fraction, and because steam flow through the steam generator MSSVs continues to dissipate the stored and generated heat. One minute after the initiation of the low-low level trip, flow from the available motor-driven AFW pump begins, thus reducing the rate of water level decrease in the steam generators.

The capacity of one AFW pump is sufficient to dissipate core residual heat, stored energy, and RCP heat such that the pressurizer does not become water-solid, demonstrating the adequacy of the AFW system to provide long-term core cooling. Figure 14.4-71 shows the pressurizer water volume transient; the calculated peak pressurizer water volume is 934.1 ft^3 compared to the total pressurizer volume limit of 1000 ft^3. Plant procedures may be followed to further cool down the plant.

The maximum RCS and main steam system pressures for this event are bounded by the Loss of External Electrical Load analysis (Section 14.4.9), which demonstrates that the peak pressures remain below 110% of the respective design limit values. The DNBR is not calculated for this analysis since it is also bounded by the Loss of External Load analysis, for which the initial reactor coolant heatup is more severe.

Radiological consequences are not evaluated for this transient because no fuel pins are expected to experience departure from nucleate boiling and thus experience cladding failure.

14.4.11 Loss of All AC Power to the Station Auxiliaries (LOOP)

14.4.11.1 Identification of Cause and Frequency Classification

A loss of offsite power can result from a number of external or internal causes. The specific cause is not of concern as part of the analysis of this transient.

This event is classified as a Condition II event (moderate frequency).

14.4.11.2 Expected Plant Response

This subsection describes the actual sequence of events and expected system response to a LOOP. It does not represent assumptions, requirements, or equipment used in the analysis.

In the event of a complete loss of offsite power and a turbine trip, there will be a loss of power to the plant auxiliaries, e.g., the reactor coolant pumps, main feedwater pumps, etc. The following events would be expected to occur:

- The reactor is tripped and plant vital instruments are supplied by the emergency power sources.
- The emergency diesel generators will start on loss of voltage on the safeguard 4kV buses to supply plant vital loads.

- As the steam system pressure (and corresponding temperature) subsequently increases, the atmospheric steam dumps and/or the steam system power relief valves are automatically opened to the atmosphere. Steam bypass to the condenser is assumed not available because of loss of the circulating water pumps.

- If the steam flow rate through the atmospheric steam dumps and power relief valves is not sufficient (or if the valves are not available), the steam generator self-actuated safety valves may lift to dissipate the residual heat produced in the reactor.

- As the no-load temperature is approached, the steam power relief valves are used to dissipate the residual heat and to maintain the plant in Mode 3, Hot Standby. If the power relief valves are not available for any reason the safety valves are used to remove residual heat.

- The auxiliary feedwater system automatically starts.

- Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and removal of residual heat is maintained by natural circulation in the reactor coolant loops.

Until the decay heat level decreases to the point where the feedwater makeup requirements are less than the auxiliary feedwater flow, the water level in the steam generators will continue to decrease. Without adequate heat removal the reactor coolant system temperature will rise causing an increase in RCS pressure. Prior to completely depleting the SG inventory, the auxiliary feedwater flow will exceed the makeup requirements and steam generator water level will recover.

14.4.11.3 Analysis of Transient

14.4.11.3.1 Methodology

The Loss of All AC Power to the Station Auxiliaries transient is analyzed using the RETRAN computer code. The code simulates the neutron kinetics, RCS including natural circulation, pressurizer, pressurizer relief and safety valves, pressurizer heaters, pressurizer spray, steam generators, feedwater system, and MSSVs. The code computes pertinent plant variables including steam generator mass, pressurizer water volume, and reactor coolant average temperature.

The analysis does not assume that power is lost as the initiating event. Rather, the analysis conservatively models a loss of normal feedwater with a subsequent loss of offsite power following the reactor trip on low-low steam generator water level. This bounds the case of an immediate loss of all AC power as the initiating event, which would result in an immediate reactor trip.

14.4.11.3.2 Key Physics Parameter Assumptions

The core physics parameter assumptions are the same as those identified for the Loss of Normal Feedwater analysis (Section 14.4.10).

14.4.11.3.3 Key System Parameter Assumptions

The system parameter assumptions are the same as those identified for the Loss of Normal Feedwater analysis (Section 14.4.10), with the following exceptions.

a. The RCPs are assumed to lose power and begin coasting down 2 seconds following the reactor trip on low-low steam generator water level. Following the loss of power to the RCPs, coolant flow necessary for core cooling and removal of residual heat is maintained by natural circulation flow in the coolant loops. Heat addition from the RCPs to the primary coolant ceases.

b. Pressurizer sprays are lost when forced reactor coolant flow ceases as a result of RCP coastdown.

14.4.11.3.4 Single Active Failure Assumptions of a Safety Grade Component

The single failure assumption is the same as that identified for the Loss of Normal Feedwater analysis (Section 14.4.10).

14.4.11.4 Acceptance Criteria

1. The maximum reactor coolant and main steam system pressure must not exceed 110% of their design values.

2. The minimum departure from nucleate boiling ratio (DNBR) must be greater than the applicable limit for the DNBR correlation being used accounting for the penalties and factors described in Section 3.2.2, “Thermal Hydraulic Design Analysis”.

3. The event shall not progress to a more serious plant condition without other faults occurring independently. This criterion is met by demonstrating that the AFW system provides sufficient heat removal to preclude the pressurizer from becoming water-solid due to coolant expansion, such that there is no water relief from the pressurizer. This ensures that long-term core cooling is maintained.

14.4.11.5 Results and Radiological Consequences

Figures 14.4-74 through 14.4-79 show the significant plant responses following a Loss of All AC to the Station Auxiliaries. The calculated sequence of events and results are listed in Table 14.4-12.

Following the reactor and turbine trip from full load, the water level in each steam generator falls due to the reduction of the steam generator void fraction, and because steam flow through the steam generator MSSVs continues to dissipate the stored and generated heat. One minute after the initiation of the low-low level trip, flow from the available motor-driven AFW pump begins, thus reducing the rate of water level decrease in the steam generators.

The capacity of one AFW pump is sufficient to dissipate core residual heat, stored energy, and RCP heat such that the pressurizer does not become water-solid, demonstrating the adequacy of the AFW system and natural circulation flow conditions in the RCS to provide long-term core cooling. Figure 14.4-77 shows the pressurizer water volume transient; the calculated peak pressurizer water volume is 653.0 ft^3 compared to the total pressurizer volume limit of 1000 ft^3. The results are less limiting than those obtained for the loss of normal feedwater transient due to the loss of RCP heat addition. Plant procedures may be followed to further cool down the plant.

The maximum RCS and main steam pressures for this event are bounded by the Loss of External Electrical Load analysis (Section 14.4.9), which demonstrates that the peak pressures remain below 110% of the respective design limit values. In the case where a loss of all AC power is the initiating event, the first few seconds of the transient will closely resemble the simulation of the complete Loss of Reactor Coolant Flow event (Section 14.4.8), where DNB and core damage due to rapidly increasing core temperature is prevented by promptly tripping the reactor. For the specific scenario analyzed, the DNBR results would be less limiting since the reactor is already tripped when RCP coastdown begins. Thus, the DNBR is not calculated for this analysis since it is bounded by the Loss of Reactor Coolant Flow analysis.

Radiological consequences are not evaluated for this transient because no fuel pins are expected to experience departure from nucleate boiling and thus experience cladding failure.

14.5 STANDBY SAFETY FEATURES ANALYSIS

14.5.1 Fuel Handling

14.5.1.1 General

The following fuel handling accidents are evaluated to ensure that no hazards are created:

a. A fuel assembly becomes stuck inside the reactor vessel;

b. A fuel assembly or RCCA is dropped onto the floor of the refueling cavity or spent fuel pool;

c. A fuel assembly becomes stuck in the penetration valve;

d. A fuel assembly becomes stuck in the transfer carriage or the carriage becomes stuck.

The possibility of a fuel handling incident is very remote because of the many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor technically trained in nuclear safety. Also, before any refueling operations begin, verification of complete RCCA insertion is obtained by tripping all rods to obtain indication of rod drop. Boron concentration in the coolant is raised to the refueling concentration level and verified by sampling. Refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical by at least 5% with all RCCAs withdrawn. The refueling cavity is filled with water meeting the same boric acid specifications.

After the reactor vessel head is removed, the RCC drive shafts are disconnected from their respective assemblies using the manipulator crane and the shaft unlatching tool. A load cell is used to indicate that the drive shaft is free of the control cluster as the lifting force is applied.

The fuel handling manipulators and hoists are designed so that fuel cannot be raised above a position which provides adequate shield water depth for the safety of operating personnel. This safety feature applies to handling facilities in both the containment and in the spent fuel pool area. In the spent fuel pool, the design of storage racks and manipulation facilities is such that:

1. Fuel at rest is positioned by positive restraints in a safe, always subcritical, geometrical array.

2. Fuel is manipulated only one assembly at a time in either the refueling cavity or the spent fuel pool.

3. Violation of procedures by placing one fuel assembly in juxtaposition with any group of assemblies in racks will not result in criticality.

Adequate cooling of fuel during underwater handling is provided by convective heat transfer to the surrounding water. The fuel assembly is immersed continuously while in the refueling cavity or spent fuel pool.

Even if a spent fuel assembly becomes stuck in the transfer tube, the fuel assembly is completely immersed and natural convection will maintain adequate cooling to remove the decay heat. The fuel handling equipment is described in detail in Section 10.2.1.

Two Nuclear Instrumentation channels are continuously in operation during fuel handling and provide warning of any approach to critical during refueling operations. One channel is a source range channel, which provides both visual indication in the control room and audible indication in containment. The other channel is either the other source range channel, or one of the two Gamma-Metric neutron counters. This instrumentation provides a continuous audible signal in the containment, and would annunciate a local horn and an annunciator in the plant control room if the count rate increased above a preset low level.

Although, safety features make the probability of a fuel handling incident very low, it is possible that a fuel assembly could be dropped during the handling operations. Therefore, this incident is analyzed both from the standpoint of radiation exposure and accidental criticality.

Special precautions are taken in all fuel handling operations to minimize the possibility of damage to fuel assemblies during transport to and from the spent fuel pool and during installation in the reactor. All handling operations on irradiated fuel are conducted under water. The handling tools used in the fuel handling operations are conservatively designed and the associated devices are of a fail-safe design.

In the fuel storage area, the fuel assemblies are spaced in a pattern which prevents any possibility of a criticality accident.

The motions of the cranes which move the fuel assemblies are limited to a low maximum speed. Caution is exercised during fuel handling to prevent the fuel assembly from striking another fuel assembly or structures in the containment or spent fuel storage building.

The fuel handling equipment suspends the fuel assembly in the vertical position during fuel movements, except when the fuel is moved through the transfer tube.

The design of the fuel assembly is such that the fuel rods are restrained by grid clips which provide a total restraining force of approximately 40 pounds on each fuel rod at the end of life. The force transmitted to the fuel rods during normal handling is limited to the (grid frictional) restraining force and is not sufficient to breech the fuel rod cladding. If the fuel rods are not in contact with the fuel assembly bottom nozzle, the rods would have to slide against the 40 pound friction force. This would dissipate an appreciable amount of energy and thus limit the impact force on the individual fuel rods.

The evaluations of a fuel assembly drop event illustrate the defense-in-depth of the system design and the associated conservatism in the fuel handling accident analysis. In effect, mechanistic analyses show that actual damage to a fuel assembly in a drop event will be limited to a few rows of fuel rods or less whereas the radiological consequences conservatively assume that all the fuel rods of a single assembly will rupture. In this manner, the fuel handling accident will bound any conceivable drop accident. Mechanistic analysis supporting Technical Specifications (Reference NUREG-1431, Rev. 3) show that only the first few rows of a fuel assembly would fail from a hypothetical maximum drop. Furthermore, mechanistic drop analyses performed specifically for PINGP fuel designs have historically shown even less damage would actually occur (Reference 85). In any event, mechanistic drop analyses have only been performed historically to demonstrate margin to the TS Bases (i.e., less than a few failed rows) or to the accident analysis (all rods fail) when deemed appropriate.

If one assembly is lowered on top of another, no damage to the fuel rods would occur that would breech the cladding. Considerable deformation would have to occur before the fuel rods would contact the top nozzle adapter plate and apply any appreciable load to the rods. Based on the above, it is unlikely that any damage would occur to the individual fuel rods during handling.

If during handling and subsequent translatory motion the fuel assembly should strike against a flat surface, the fuel assembly lateral loads would be distributed axially along its length with reaction forces at the grid clips and essentially no damage would be expected in any fuel rod.

Analyses were performed that address the extremely remote situations where a fuel assembly is dropped vertically and strikes a solid unyielding surface, and where a single fuel assembly is dropped vertically onto another fuel assembly located in either the reactor vessel or the spent fuel racks (Ref. 85). The analyses demonstrate that the energy absorbed by the fuel rods in the dropped fuel assembly is less than the criterion for fuel rod fracture in compression. The analyses also show that buckling will not occur in an irradiated target fuel assembly when impacted by a dropped fuel assembly. The analyses also demonstrate that fuel rod buckling will occur in a fresh 422V+ fuel assembly when impacted by a vertical drop of another fuel assembly. However, even though mechanistic drop analyses were performed, a conservative upper limit of damage is assumed by considering the cladding rupture of all rods in one complete irradiated fuel assembly when evaluating the environmental consequences of a fuel handling accident.

For the assumed accident there would be a sudden release of the gaseous fission products held in the voids between the pellets and cladding of one fuel assembly. The low temperature of the fuel during handling operations precludes further significant release of gases from the pellets themselves after the cladding is breached. Molecular halogen release is also greatly minimized due to their low volatility at these temperatures. The strong tendency for iodine in vapor and particulate form to be scrubbed out of gas bubbles during their ascent to the water surface further reduces the quantity released from the water surface.

Under the accident methodology for the FHA, a fuel assembly is assumed to be dropped and damaged during fuel handling. The dose analysis is performed to determine the radiological consequences of the accident. For the FHA, Prairie Island has implemented the alternate source term (AST) in accordance with 10 CFR 50.67.

FHA Input Parameters and Assumptions

The major assumptions and parameters used in the analysis are itemized in Tables 14.5-1 and 14.5-2. The analysis involves dropping a recently discharged fuel assembly. It is assumed that all of the activity in the damaged assembly is released to the pool. Furthermore, it is assumed that the activity that escapes from the containment refueling cavity or the spent fuel pool is released to the environment over a two-hour time period per the guidance of RG 1.183. A constant release rate is assumed for the two-hour time period.

No credit is taken for ventilation filtration system operation in the spent fuel area (i.e., no credit is taken for spent fuel pool special ventilation). Similarly, no credit is taken for containment purge or in-service purge supply and exhaust system closure or filtration capability. In addition, no credit is taken for the containment equipment hatch placement or closure nor is credit taken for having the containment air lock doors closed. Since the assumptions and parameters used to model the release due to a FHA inside containment are identical to those for a FHA in the spent fuel pool, except for different control room intake atmospheric dispersion factor values (χ/Qs) for the different release paths, the activity released is the same regardless of the location of the accident. In order to bound the accident occurring either inside containment or in the present fuel pool, the location with the highest χ/Q value is assumed. More detailed discussion of the control room and offsite χ/Qs is included below.

Consistent with RG 1.183 (Position 1.2 of Appendix B), the radionuclides considered for release are xenons, kryptons, halogens, cesiums, and rubidiums. The list of xenons, kryptons, and halogens considered is given in Table D.3-2 in Appendix D. These values are based on 1683 MWt core power. The alkali metals, cesium and rubidium are not included in this analysis because they are not assumed to be released from the pool. Per RG 1.183, Appendix B, the cesium and rubidium (particulate radionuclides) released from the damaged fuel rods are assumed to be retained by the water in the refueling cavity and would not be available for release.

It is assumed that all of the fuel rods in the equivalent of one fuel assembly are damaged to the extent that all their gap activity is released. The inventory in the damaged assembly is based on the assumption that the subject fuel assembly has been operated at the maximum radial peaking factor of 1.90 times the average core power. It is assumed that the dropped assembly has been discharged from the core 50 hours after reactor shutdown; therefore, a decay time of 50 hours is applied to the activities in the analysis. The basis for the core activity is described in Appendix D, Section D.3. 01517422

The calculation of the radiological consequences following a FHA uses gap fraction of 8% for I-131, 10% for Kr-85 and 5% for all other noble gas and iodine nuclides. Footnote 11 to Table 3 in RG 1.183 indicates that these gap fractions are acceptable “with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU.” PINGP fuel design and fuel management practices provide for exceeding linear heat generation rate (LHGR) of 6.3 kw/ft with burnups exceeding 54 GWD/MTU. Site-specific analyses based on a Westinghouse OFA core were performed that show that the gap fractions in Table 3 of RG 1.183 are bounding. The site-specific analysis of the gap fractions is described in Appendix D, Section D.2. PINGP is licensed to utilize Westinghouse 422V+ fuel. (Ref. 95)

Due to the margin between the calculated gap fractions and those specified in Table 3 of RG 1.183, along with minimal differences between the fuel designs (i.e., pin diameter and mass of fuel), it is assumed that the RG 1.183 gap fractions would be bounding for the Westinghouse 422V+ fuel as well. There are no other significant differences in the PINGP fuel design and fuel management schemes that would cause the 422V+ gap fractions to be significantly larger than those for the Westinghouse OFA Fuel which are much less then the RG 1.183 values.

In accordance with RG 1.183, the iodine species released from the damaged fuel to the pool water are 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodine. It is assumed that all CsI instantaneously dissociates in the water and re-evolves as elemental iodine. In accordance with RG 1.183, the iodine species released from the pool water are 57% elemental iodine and 43% organic iodine.

An effective decontamination factor (DF) of 200 for iodine, as provided in RG 1.183 is used in the analysis to account for scrubbing of the iodine in the pool liquid. A DF of 200 is applicable to PINGP as the minimum water level requirement of RG 1.183, Appendix B, Section 2 is satisfied. Specifically, PINGP Technical Specification Surveillance Requirement (SR) 3.9.2.1, “Refueling Cavity Water Level,” requires that a minimum of 23 feet of water above the top of the reactor vessel flange be maintained during movement of irradiated assemblies within containment. Similarly, SR 3.7.15.1, “Fuel Storage Pool Water Level,” requires a minimum of 23 feet of water over the top of the assemblies be maintained during movement of irradiated fuel assemblies in the spent fuel storage pool. No DF is applied to the noble gas releases and an infinite DF is applied to the particulate radionuclides (i.e., the cesium and rubidium).

No credit is taken for removal of iodine by containment and spent fuel pool building ventilation systems' filters nor is credit taken for isolation of release paths. It is assumed that the activity is released from the pool to the outside atmosphere over a 2-hour period. Since no filters or containment isolation is modeled, this analysis supports refueling operation with the equipment hatch, air lock doors or containment penetrations remaining open at 50 hours after shutdown. Administrative controls are implemented during fuel handling operations to close any containment openings within one hour following a fuel handling accident.

The EAB dose is calculated for the worst 2-hour period, the LPZ dose is calculated for the release duration (i.e., two hours), and the control room doses are calculated for 30 days. As shown in the results, the EAB and LPZ dose are reported for the entire 30-day duration.

The RADTRAD software code was used to calculate the isotopic releases and resulting radiation doses offsite in the control room (Reference 107).

Control Room and Off-Site Atmospheric Dispersion

Control Room Atmospheric Dispersion

The control room intake χ/Q values for the potential FHA release points are determined using ARCON96, Atmospheric Relative Concentrations in Building Wakes methodology. (Reference 58). The determination of the χ/Q values is documented in Reference 59 and Reference 108.

Input data consists of hourly on-site meteorological data, release characteristic such as release height, the building area affecting the release, and various receptor parameters such as its distance and direction from the release to the control room air intake and intake height. A continuous temporally representative 5-year period of hourly average data from the PINGP meteorological tower (i.e., January 1, 1993 through December 31, 1997) is used in this calculation. The FHA could be postulated to occur in either the spent fuel pool or in containment.

Receptor

The CR ventilation intakes use bubble tight dampers to isolate the control room from the outside environment. However, the FHA analysis conservatively assumes that all of the inleakage to the Control Room Envelope (CRE) occurs through the ventilation intake. The CR ventilation intake is the closest point to the source of the leakage and provides the limiting χ/Q. Using a single point for the inleakage to the CRE also simplifies this determination, as a single receptor location can be used to bound other potential receptor locations for the CRE.

Source

The spent fuel pool (SFP) enclosure is inside of the Auxiliary Building, but outside of the Auxiliary Building Special Ventilation Zone (ABSVZ). This portion of the Auxiliary Building is a steel structure with metal siding that is not leak tight. This will be referred to as the "common area of the Aux Bldg" hence forth. If the Spent Fuel Pool special ventilation system were credited in the dose analyses the release is filtered before being exhausted through the Shield Building Stack. If normal ventilation is operating and credit is not taken for isolation by the high radiation signal, the release is out of the normal ventilation exhaust stack; which is farther away than other potential release locations. Without the ventilation systems operating, the radioactivity released from the damaged fuel assembly could exit the SFP enclosure and enter the common area of the Aux Bldg. Activity exiting the common area of the Aux Bldg at the closest point to the CR ventilation intake would provide a bounding atmospheric dispersion factor. Therefore, the analysis is performed assuming that the radioactivity is released through the common area of the Aux Bldg closest to the CR ventilation intake and no credit is taken for isolation of the spent fuel pool structure or operation of the spent fuel pool ventilation systems.

The analysis for the FHA inside of containment is performed assuming that there are no controls on containment boundary during fuel handling. Thus, the leakage could exit Containment and enter the ABSVZ through open containment penetrations, exit containment directly to the atmosphere through the open Equipment Hatch or enter this same common area of the Aux Bldg through an open Containment Maintenance Air Lock. If the leakage entered this common area of the Aux Bldg through an open Maintenance Air Lock it could have the same release path as that described above for the FHA in the SFP enclosure. Leakage into the ABSVZ would need to traverse a torturous path to exit the building and most likely would be filtered by the Auxiliary Building special ventilation system and released through the Shield Building Ventilation Stack. Leakage through the open Equipment Hatch would enter the Annulus and be released to the Shield Building stack or released directly to the outside environment. The distance from Shield Building Ventilation Stack and the Equipment Hatch to the CR Vent Intake is much further than the distance from the common area of the Aux Bldg to the CR Vent Intake. Thus, similar to the FHA in the spent fuel pool assuming all of the leakage escapes through the common area of the Aux Bldg in the area of the building closest to the CR ventilation intake provides a bounding result.

Thus, the following release locations were considered in the ARCON96 analyses:

- Common Area of Auxiliary Building to 121 Control Room Intake
- Common Area of Auxiliary Building to 122 Control Room Intake
- Spent Fuel Pool Normal Exhaust Stack to 121 Control Room Intake
- Spent Fuel Pool Normal Exhaust Stack to 122 Control Room Intake
- Unit 1 Equipment Hatch to 121 Control Room Intake
- Unit 1 Equipment Hatch to 122 Control Room Intake
- Unit 2 Equipment Hatch to 121 Control Room Intake
- Unit 2 Equipment Hatch to 122 Control Room Intake

The common area of the Auxiliary Building is modeled as a diffuse source. The spent fuel pool normal exhaust and equipment hatches are modeled as point sources. The χ/Qs calculated for the 0-2 hour time period for each of the above source/receptor pairs are summarized in Table 14.5-3.

As show in Table 14.5-3, the limiting χ/Q for the 0-2 hour time period is 6.17E-03 sec/m^3 for the common area of the Auxiliary Building to 121 Control Room intake.

Off-Site Atmospheric Dispersion

The χ/Q values for the PINGP EAB and the LPZ are those from Appendix H, Table XIV. The χ/Q value for the 0 - 8 hour time period is used for the duration of the analysis for the EAB and the LPZ.

Control Room Ventilation Operation

It is assumed that the control room (CR) HVAC system is initially operating in normal mode, whereby fresh air is being brought into the CR unfiltered at a rate of 2000 cfm. Post-accident, the activity level in the Control Room would cause a high radiation signal within the first few seconds. The high radiation signal causes dampers to close automatically isolating the control room envelope (CRE) from the outside air and directing a portion of the recirculated air through PAC filters. Actuation of the system in this manner due to the high radiation signal is conservatively delayed to 5 minutes after event initiation to increase the margin of safety. After isolation and initiation of filtered recirculation, 300 cfm of unfiltered air inleakage is assumed. The 300 cfm of unfiltered inleakage includes 290 cfm for boundary inleakage and 10 cfm for ingress and egress.

Acceptance Criteria

According to RG 1.183, the EAB and LPZ dose acceptance criteria for a fuel handling accident is 6.3 rem TEDE, which is approximately 25% of the 10 CFR 50.67 limit of 25 rem. The control room dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67.

Results and Conclusions

The FHA Dose analysis is documented in Reference 60.

Offsite

The offsite dose due to a design basis FHA are presented below. These doses are well within the dose limits 10 CFR 50.67 and less than the acceptance criteria of RG 1.1.83.

FHA Offsite Dose Results Assuming AST

Location	Acceptance Criteria (rem)	TEDE (rem)
Exclusion Area Boundary	6.3	2.28
Low Population Zone	6.3	0.62

Control Room

The Control Room dose due to a design basis FHA is presented below. The doses are less than the dose limit of 10 CFR 50.67 and acceptance criteria of RG 1.183.

FHA Control Room Dose Results Assuming AST

Unfiltered Inleakage (cfm)	Acceptance Criteria (rem)	TEDE (rem)
300	5	3.64

14.5.1.2 Deleted

14.5.1.3 Deleted

14.5.1.4 Deleted

14.5.1.5 Deleted

14.5.2 Accidental Release of Radioactive Liquids

Vessels in the waste disposal system which are used for waste storage are housed in a Class I portion of the Auxiliary Building or the Class I* portion of the Radwaste Building. All such vessels are located inside Class I structural enclosures such as sumps, dikes, or walls or specially constructed areas which will retain spilled liquids. This ensures that the structures are capable of containing the liquid wastes during seismic events.

Thus, there are no credible accidents which would result in the release of radioactive wastes to the river in excess of the limits given in the Offsite Dose Calculation Manual (ODCM). USAR, Section 9.2, contains a discussion of radioactive liquid waste storage and processing.

14.5.3 Accidental Release-Waste Gas

The waste gas accident is defined as an unexpected and uncontrolled release to the atmosphere of the radioactive xenon and krypton fission gases that are stored in the waste gas storage system. Failure of a gas decay tank or associated piping could result in a release of this gaseous activity. This analysis shows that even with the worst expected conditions, the offsite doses following release of this gaseous activity would be very low.

14.5.3.1 Gas Decay Tank Rupture

The gas decay tanks contain gases vented from the reactor coolant system, the volume control tank, and the liquid holdup tanks. Two independent process loops are provided to accumulate and store radioactive gases. The first system is designed to strip fission gases from the reactor coolant while the second accumulates all other potentially radioactive gases. The fission gases stripped from the reactor coolant will represent the more significant portion of the radioactive source and will be calculated as described below.

Nonvolatile fission product concentrations are greatly reduced as the cooled Reactor Coolant System liquid is passed through the purification demineralizers. (The removal factor for iodine, for example, is at least 10). The decontamination factor for iodine between the liquid and vapor phases, for example, is expected to be on the order of 10,000.

The components of the waste gas system are not subjected to high pressures or stresses, they are a Class I design, and are designed to the standards given in Table 9.1-2; thus, a rupture or failure is highly unlikely. However, a rupture of a gas decay tank is analyzed to define the limit of the hazard that could result from any malfunction in the radioactive waste disposal system.

The activity in a gas decay tank is taken to be the maximum amount that could accumulate over the plant lifetime from operation with one percent of the rated core thermal power being generated by rods with clad defects. For all isotopes except Kr 85, this postulated amount of activity is taken to be one Reactor Coolant System equilibrium cycle inventory as given in Appendix D, Table D.7-1. This value is particularly conservative because some of this activity would normally remain in the coolant, some would have been dispersed earlier through the stack via equipment leakage, and the shorter-lived isotopes will have decayed substantially. The Kr 85 inventory given in Appendix D, Table D.7-1, represents the activity at the end of the 60 year plant life. (Reference 112) 01448551

To define the maximum doses, the release is assumed to result from gross failure of any process system storage tank, here represented by a gas decay tank giving an instantaneous release of its volatile and gaseous contents to the atmosphere.

Gas decay tank rupture maximum doses are provided along those for volume control tank rupture, below. These gas decay tank maximum doses result from a postulated WGDT activity inventory of 140,000 Ci DEX. Note, however, that the maximum WGDT activity inventory of 78,800 Ci DEX allowed by TS 5.5.10b effectively imposes an upper limit of 0.5 rem whole body dose on the consequences of the hypothetical tank rupture described by the accident analysis. 01448551

The WGDT rupture analysis results have been approved by the NRC in License Amendment 215 and 203, for Units 1 and 2, respectively. (Reference 113)

14.5.3.2 Volume Control Tank Rupture

The volume control tank contains fission gases and low concentrations of halogens which are normally a source of waste gas activity vented to a gas decay tank. The iodine concentrations and volatility are quite low at the temperature, pH and pressure of the fluid in the volume control tank. The same assumptions detailed in the preceding subsection apply to this tank. As the volume control tank and associated piping are not subjected to any high pressures or stresses, failure is very unlikely. However, a rupture of the volume control tank is analyzed to define the maximum exposure that could result from such an occurrence.

Rupture of the volume control tank is assumed to release all the contained noble gases and 1% of the halogen inventory of the tank plus that amount contained in the 40 gpm flow from the demineralizers, which would continue for up to fifteen minutes before isolation would occur. The 1% halogen release is a very conservative estimate of the decontamination factor expected for these conditions.

Based on 1% fuel defects, the activities available for release are 7700 Ci of Xe^{133} dose equivalent noble gases and .022 Ci of I^{131} dose equivalent halogens.

Method of Analysis

In calculating off-site plume center-line exposure it is assumed that the activity is discharged to the atmosphere at ground level and is dispersed as a Gaussian plume downwind taking into account building wake dilution.

Dispersion coefficients based on the on-site meteorology program are used. A wind velocity of 0.89 meters per second is assumed to remain in one direction for the duration of the accident under Pasquill F conditions. The dispersion characteristics are discussed in Appendix H. Curves corrected for building wake effects by the volumetric source method, are presented on Figure 8 of Appendix H.

The following parameters have been used in the dose assessment:

a. A 0-8 hour EAB X/Q value of 6.49×10^{-4} sec/m^3

b. A 0-8 hour LPZ X/Q value of 1.77E-04 sec/m^3

c. Breathing rate equal to 3.47×10^{-4} m^3/sec

d. An I^{131} equivalent dose conversion factor equal to 1.48×10^6 rem/curie

e. A Kr^{85} dose equivalent conversion factor equal to 6.20×10^{-2} rem-m^3/curie-sec

f. A Xe^{133} dose equivalent conversion factor equal to 3.57×10^{-2} rem-m^3/curie-sec

The following tabulation summarizes the whole body and thyroid doses at the exclusion distance, consistent with a receptor on the plume centerline.

	Thyroid Dose		**Whole Body Dose**		
	EAB	**LPZ**	**EAB**	**LPZ**	
Gas Decay Tank Rupture (Ref. 112)	N/A	N/A	4.32 rem	1.18 rem	01448551
Volume Control Tank Rupture	7.3E-03 rem	1.7E-03	0.18 rem	0.05 rem	
10CFR100 Limits	300 rem	300 rem	25 rem	25 rem	

It is concluded that a rupture in the waste gas system or in the volume control tank would present no undue hazard to public health and safety.

14.5.4 Steam Generator Tube Rupture

14.5.4.1 General

The accident examined is the complete severance of a single steam generator tube with the reactor at power. This accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the Reactor Coolant System. In the event of a coincident loss of offsite power, or failure of the condenser steam dump system, discharge of activity to the atmosphere takes place via the steam generator safety and/or power operated relief valves.

The activity available for release from the system is limited by:

a. The activity concentration in the steam generator secondary that are a consequence of operational leakage prior to the complete tube rupture.

b. The activity concentration in the reactor coolant, which is conservatively assumed to arise from one percent defective fuel clad.

c. Operator actions to isolate the mixed primary and secondary leakage to atmosphere.

The steam generator tube material is Inconel 690 and as the material is highly ductile it is considered that the assumption of a complete severance is conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the Steam and Power Conversion System is subject to continuous surveillance and an accumulation of minor leaks which cause the activity to exceed the limits established in the technical specifications is not permitted during unit operation.

The main objective of the operator is to determine that a steam generator tube rupture has occurred, and to identify and isolate the ruptured steam generator on a restricted time scale in order to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the ruptured unit. The recovery procedure can be carried out in a time scale which ensures that break flow to the secondary system is terminated before water level in the affected steam generator rises into the main steam pipe. Sufficient indications and controls are provided to enable the Operator to carry out these functions satisfactorily. Consideration of the indications provided at the Control Board together with the magnitude of the break flow, leads to the conclusion that the isolation procedure can be completed within approximately 30 minutes of accident initiation.

The 30 minute time frame is not a strict acceptance criteria. The acceptance criteria are the basis for the 30 minute time frame. The acceptance criteria are two fold:

1. To prevent the water level in the Steam Generator from rising into the Main Steam pipe (i.e., ensure the ruptured Steam Generator does not become flooded).

2. To maintain offsite dose level acceptable values. The offsite dose is directly related to the activity transferred to the secondary side of the ruptured Steam Generator.

14.5.4.2 Description of Accident

This section describes the expected plant response to this accident. It does not represent the specific inputs and assumptions which are used in the analysis of the event. These are presented later.

Assuming normal operation of the various plant control systems, the following sequence of events is initiated by a tube rupture:

a. Pressurizer low pressure and low level alarms are actuated, and prior to plant trip, charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side there is a mismatch between steam flow/feedwater flow before trip as feedwater flow to the affected steam generator is reduced due to the additional break flow which is now being supplied to that unit.

b. Loss of reactor coolant inventory leads to falling pressure and level in the pressurizer until a reactor trip signal is generated by low pressurizer pressure. Resultant plant cooldown following reactor trip leads to a rapid change of pressurizer level, and the safety injection signal, initiated by low pressurizer pressure, follows soon after the reactor trip. The safety injection signal automatically terminates normal feedwater supply and initiates auxiliary feedwater addition. A single failure in the actuation circuitry will not prevent the actuation of the auxiliary feedwater system.

   The Engineered Safeguards Actuation System is designed in accordance with the criteria of IEEE-279-1968 and meets the single failure criteria in IEEE 279-1971 (See Section 7).

   The actuation circuitry, from the output of the Safety Injection logic down to the solenoid vent valves on the feedwater main and bypass valves, is redundant and meets the single failure criteria.

There are two auxiliary feedwater pumps, one motor driven and one turbine driven, either one of which will provide sufficient feedwater following a break. For Unit 1, the Safety Injection signal from train A starts the turbine driven pump, and the signal from train B starts the motor driven pump. For Unit 2, the Safety Injection signal from train A starts the motor driven pump, and the signal from train B starts the turbine driven pump. A single failure cannot disable both actuating circuits.

c. The steam generator blowdown liquid monitor and the air ejector radiation monitor will alarm, indicating an increase in radioactivity in the secondary system. These alarms generally provide the earliest diagnosis of a tube rupture.

d. The plant trip automatically shuts off steam supply to the turbine and if offsite power is available, the condenser steam dump valves open permitting steam dump to the condenser. In the event of a co-incident loss of offsite power (LOOP), the condenser steam dump valves would automatically close to protect the condenser. The steam generator pressure would rapidly increase resulting in steam discharge to the atmosphere through the steam generator safety valves, power operated relief valves or the atmospheric steam dump valves.

e. Following plant trip, the continued action of auxiliary feedwater supply and borated safety injection flow (supplied from the Refueling Water Storage Tank) provide a heat sink which absorbs some of the decay heat. Thus, steam bypass to the condenser, or in the case of loss of offsite power, steam relief to atmosphere, is attenuated during the time in which the recovery procedure leading to isolation is being carried out.

f. Safety injection flow results in increasing pressurizer water level. The time after trip at which the operator can clearly see returning level in the pressurizer is dependent upon the amount of operating auxiliary equipment and the size of the rupture.

14.5.4.3 Analysis

This section describes the determination of the mass transferred to the secondary side of the Steam Generator through the broken tube for determining the inputs to the radiological analysis. The subsequent section describes the radiological analysis based on this mass transfer.

In determining the mass transfer from the Reactor Coolant System through the broken tube several conservative assumptions were made:

a. Plant trip occurs automatically as a result of low pressurizer pressure.

b. Following the initiation of the Safety Injection Signal both Safety Injection Pumps are actuated and continue to deliver flow for 30 minutes.

c. After plant trip the break flow equilibrates at the point where incoming safety injection flow is balanced by outgoing break flow as shown in Figure 14.5-1 (for two SI Pumps). The resultant break flow persists from plant trip until 30 minutes after the accident.

d. The steam generators are controlled at the safety valve setting rather than the power operated relief valve setting.

e. The operator identifies the accident type and terminates break flow to the ruptured steam generator within 30 minutes of accident initiation.

The above assumptions lead to a conservative estimate of 140,000 lbs. for the total amount of reactor coolant transferred to the ruptured steam generator as a result of a tube rupture accident.

14.5.4.4 Environmental Consequences of a Tube Rupture

The key inputs and assumptions used in the Steam Generator Tube Rupture (SGTR) radiological consequence analysis analyzed using the Alternative Source Term (AST) are summarized below and provided in Table 14.5-12.

The SGTR accident is postulated as a complete severance of a single Steam Generator (SG) tube. The tube rupture results in the release of radioactive reactor coolant into the ruptured SG. For the intact SG, primary to secondary coolant leakage continues to transfer activity into the Secondary Coolant side. This makes it available for release into the environment via steaming through the Power Operated Relief Valves (PORV) and via the turbine driven auxiliary feedwater (TDAFW) pump steam exhaust. It has been shown that it is more conservative to assume the release is from the PORVs in lieu of the TDAFW pump steam exhaust. For the SG with the ruptured tube, coolant release will take two forms:

- Break Flow - un-flashed release of RCS coolant directly into the secondary loop, and made available for steaming release to the environment through the SG PORV and TDAFW pump steam exhaust.
- Flashed Break Flow - RCS coolant that flashes directly to steam when released from the ruptured tube, and is sent through the SG PORV and TDAFW pump steam exhaust to the environment.

The description of events subsequent to the tube failure is discussed in Section 14.5.4.2.

Consistent with Regulatory Guide 1.183, two reactor transients that maximize the radioactivity available for release were modeled. In addition to these two transients, the release of the maximum allowed operational concentration of iodine activity in the secondary coolant system, 0.1 μCi/gm is analyzed. This simulates the release of secondary coolant activity existing prior to the SGTR accident. The dose consequence of this simulation is added to each of the other modeled cases.

Case 1: Dose Due to Pre-accident Iodine Spike

The first case involves a 30 μCi/gm pre-accident Iodine spike. This 30 μCi/gm spike is consistent with the Technical Specification operational Reactor Coolant System (RCS) activity concentration limit for an assumed spike. In this scenario, it is assumed that all of the spike activity is homogeneously mixed in the primary coolant prior to accident initiation.

Case 2: Dose Due to Accident Initiated Concurrent Iodine Spike

The second case involves an accident initiated iodine spike that occurs concurrently with the release of fluid from the primary and secondary coolant systems. Regulatory Guide 1.183 specifies that this spike should result in a release rate from the operating limit defective fuel fraction (~1%) that is 335 times the normal rate, and lasts for an 8 hour duration.

Fuel Damage and Core Source Term

The design basis assumes no fuel damage for the postulated steam generator tube rupture event. For this SGTR accident, the source terms are defined by the Technical Specification activity release rates from a maximum failed fuel fraction assumed during operation, which are characterized by the equilibrium 0.5 μCi/gm Dose Equivalent (DE) I-131 iodine activity concentrations in the primary reactor coolant system. The noble gas inventory in the RCS is based on operation with a conservative worst-case 1% core fuel defects (i.e., 580 μCi/gm DE Xe-133). Because no fuel damage is assumed for this accident, only iodine and noble gas isotopes are modeled to contribute to dose. To identify the worst-case SGTR accident, however, the two different cases of iodine spiking described above are analyzed.

In addition, for both cases, a 0.1 μCi/gm DE I-131 equilibrium secondary coolant activity concentration limit from Technical Specifications is added.

Release Rates and Partitioning Factors

As previously discussed, a number of modes of release are indicative of this particular accident scenario. Therefore, the varying releases associated with the timing and sequence of events of this accident was derived.

Activity that originates in the RCS is released to the secondary coolant by means of the primaryto-secondary coolant leak rate. T his design basis leak rate value is 150 gpd into the intact SG. For input into RADTRAD this rate was converted to 0.0188 cubic feet per minute into the intact SG for 14 hours.

The methodology used to model steaming of activity through SG PORVs following the postulated SGTR event assumes an average cumulative release rate through the paths. The partitioning factors are applied to these release rates. Incremental steam mass releases are given in pounds per time interval. For the time intervals used in this accident scenario, release rates were derived by taking the averages of mass releases over each specified time interval. Then these mass flow rates were converted to volumetric flow rates using the assumption of cooled liquid conditions (i.e., 62.4 lbm/ft^3), as specified by the applicable guidance of Regulatory Guide 1.183.

The ruptured steam generator experiences two simultaneous release mechanisms. Primary to secondary coolant leakage through the ruptured tube of the ruptured SG that flashes conservatively goes directly to the environment, without mixing with any secondary coolant. Therefore, with thisrelease mechanism, no partitioning of iodine is expected to occur in this release. However, leakage that does mix with the volume of coolant in the ruptured SG is released by flashing to the environment, and the applicable partition factor is applied, as discussed in the following text.

For all post-accident releases through the SG PORVs, the mechanism for release to the environment is steaming of the coolant in the secondary system. Because of this release dynamic, Regulatory Guide 1.183 allows for a reduction in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water. For Iodine, the partitioning factor of 0.01 was taken directly from Regulatory Guide 1.183. Reviewing the specified AST release fractions, it is concluded that the only nuclides to be released from the core source term, other than iodines, are noble gas nuclides, and because of the volatility of noble gases, no partitioning is assumed for any such isotopes.

Control Room χ/Q Calculations (Meteorology)

The limiting control room atmospheric dispersion factors for SG PORVs releases are weighted by their portion of the total mass release to determine mass release weighted average atmospheric dispersion factors that are used to model releases the steam releases.

Acceptance Criteria

According to Regulatory Guide 1.183, the EAB and LPZ dose acceptance criteria for a steam generator tube rupture accident with a pre-accident iodine spike is the 10 CFR 50.67 limit of 25 rem TEDE. The EAB and LPZ dose acceptance criteria for a steam generator tube rupture accident with a concurrent iodine spike is 2.5 rem TEDE, which is 10% of the 10 CFR 50.67 limit of 25 rem TEDE.

The control room dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67

Dose Results

Radiological doses resulting from a design basis SGTR for a control room operator and a person located at EAB or LPZ are to be less than the regulatory dose limits as given below.

SGTR Dose Results with a Pre-Existing Iodine Spike		
Location	Acceptance Criteria (rem)	TEDE (rem)
Exclusion Area Boundary	25	1.09
Low Population Zone	25	0.30
Control Room	5.0	4.67

SGTR Dose Results with a Concurrent Iodine Spike		
Location	Acceptance Criteria (rem)	TEDE (rem)
Exclusion Area Boundary	2.5	0.96
Low Population Zone	2.5	0.27
Control Room	5.0	3.45

14.5.4.5 Recovery Procedure

In the event of an SGTR, the plant operators will diagnose the event and perform the required recovery actions to stabilize the plant and terminate the primary to secondary leakage. The operator actions for SGTR recovery are provided in the plant Emergency Operating Procedures.

Operator actions are described below.

1. Identify the ruptured steam generator

   High secondary side activity, as indicated by the condenser air ejector radiation alarm, steam generator blowdown liquid radiation alarm, and/or main steam line high radiation indication, typically will provide the first indication of an SGTR event. The ruptured steam generator can be identified by an unexpected increase in steam generator narrow range level, a radiation survey, or a chemistry laboratory sample. For an SGTR that results in a reactor trip at high power, the steam generator water level as indicated on the narrow range scale will decrease significantly for both steam generators. The auxiliary feedwater flow will begin to refill the steam generators, distributing flow to each steam generator. Since primary to secondary leakage adds additional liquid inventory to the ruptured steam generator, the water level will increase more rapidly than normally expected in that steam generator. This response provides confirmation of an SGTR event and also identifies the ruptured steam generator.

2. Isolate the ruptured steam generator from the intact steam generator and isolate feedwater to the ruptured steam generator

   After the steam generator with a tube rupture has been identified, recovery actions begin by isolating steam flow and feedwater flow for the ruptured steam generator. In addition to minimizing radiological releases, this also reduces the possibility of overfilling the ruptured steam generator with water by 1) minimizing the accumulation of feedwater flow and 2) enabling the operator to establish a pressure differential between the ruptured and intact steam generators as a necessary step toward terminating primary to secondary leakage.

3. Cool down the RCS using the intact steam generator

After isolating the ruptured steam generator, the RCS is cooled to less than the saturation temperature corresponding to the ruptured steam generator pressure by dumping steam from only the intact steam generator. This ensures adequate subcooling will exist in the RCS after depressurization of the RCS to the ruptured steam generator pressure in subsequent actions. If offsite power is available, the normal steam dump system to the condenser can be used to perform this cooldown. If offsite power is not available, the RCS is cooled using the intact steam generator power operated relief valve.

4. Depressurize the RCS to restore reactor coolant inventory

When the cooldown is completed, safety injection flow will increase RCS pressure until break flow matches safety injection flow. Consequently, safety injection flow must be terminated to stop primary to secondary leakage. Prior to terminating safety injection flow, adequate reactor coolant inventory must first be assured. The cooldown and depressurization provides both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after safety injection flow is stopped.

The RCS depressurization is performed using normal pressurizer spray, if it is available. However, if normal pressurizer spray is not available, RCS depressurization can be performed using the pressurizer power operated relief valve or auxiliary pressurizer spray.

5. Terminate safety injection to stop primary to secondary leakage

   The previous actions established adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that safety injection flow is no longer needed. When these actions are complete, safety injection flow is stopped to terminate primary to secondary leakage. Primary to secondary leakage will continue after safety injection flow is stopped until RCS and ruptured steam generator pressures equalize. Charging flow, letdown, and pressurizer heaters can then be used to control RCS pressure.

   Following safety injection termination, the plant conditions will be stabilized, the primary to secondary break flow will be terminated, and all immediate safety concerns will have been addressed. At this time a series of operator actions are performed to prepare the plant for cooldown to Mode 5, Cold Shutdown. Subsequently, actions are performed to cooldown and depressurize the RCS to Mode 5, Cold Shutdown and to depressurize the ruptured steam generator.

   There is ample time available to carry out the above recovery procedure such that isolation of the ruptured steam generator is established before water level rises into the main steam pipes. Normal operator vigilance assures that excessive water level will not be attained.

14.5.4.6 Steam Generator Tube Rupture Margin to Overfill

Analyses were performed of the limiting margin-to-overfill (MTO) scenarios to demonstrate that the rupture Steam Generator would not be overfilled. The analyses followed the methodology in WCAP-10698-P-A, with the exception of the assumption of a single failure.

The analyses were performed using the LOFTTR2 thermal hydraulic model consistent with the methodology in WCAP-10698-P-A.

The results indicate a margin-to-overfill of 186 ft^3 in the ruptured steam generator for the limiting scenario. The limiting scenario models 0% steam generator tube plugging (SGTP), low decay heat, maximum safety injection (SI) enthalpy and minimum auxiliary feedwater (AFW) enthalpy. No water is transferred into the steam lines.

The sequence of events for the limiting scenario analysis is presented in Table 1, below.

SGTR Sequence of Events	
Event	Time (sec)
Tube Rupture	0
Reactor Trip	49
AFW Initiaiton	50
SI Actuation	119
Ruptured SG AFW Isolation	251
Close MSIV	1130
Initiate Cooldown with Intact SGs	1190
Establish Charging Flow	1192
Terminate Cooldown	1626
Initiate Depressurization	1866
Terminate Depressuriztion	1962
Stop SI Flow	2082
Balance Charging and Letdown Flows	2982
Break Flow < 0	3212

Figure 14.5-12, pages 1 to 3, provides the time-dependant values of the following parameters for the limiting MTO scenario:

- Reactor Coolant System and Secondary Pressures (Intact and Ruptured Steam Generators)
- Primary-to-Secondary Break flow rate
- Steam Generator Water Volumes (Intact and Ruptured Steam Generators)
- Pressurizer Level
- Intact Steam Generator Inlet and Outlet Temperatures
- Ruptured Steam Generator Inlet and Outlet Temperatures
- Steam Generator Steam releases
- Steam Generator Narrow Range Level (Intact and Ruptured Steam Generators)

14.5.5 Rupture of a Steam Pipe

14.5.5.1 Identification of Cause and Frequency Classification

A Rupture of a Steam Pipe could be caused by a failure of the pipe itself, or the inadvertent opening and sticking of a valve, e.g., safety or PORV. The analyses in this section are used to evaluate: 1) the containment's response to a main steam line break (MSLB) inside of containment, 2) the core's response to a MSLB inside containment, 3) a small steam line break, and 4) the Dose analysis for a MSLB outside of containment. For an outside of containment MSLB, the effects to the core would be similar and are bounded by the inside containment MSLB analysis. The effects to structures and components for an outside of containment MSLB are evaluated in Appendix I.

This event is classified as a Condition IV event (limiting fault).

14.5.5.2 Expected Plant Response

This subsection describes the actual sequence of events and expected system response to a rupture of a steam pipe. It does not represent assumptions, requirements, or equipment used in the analysis.

The Steamline Rupture - Core Response transient is analyzed at both full-power and zero-power conditions. Increased steam flow from the steam generators causes an increase in the heat extraction rate from the reactor coolant system (RCS), resulting in a reduction of primary coolant temperature and pressure. Because negative moderator temperature and Doppler fuel temperature reactivity coefficients are a characteristic of the core design, the core power will inherently seek a level bounded by the steam load demand, assuming no intervention of control, protection, or engineered safeguards systems. The rate at which the plant approaches equilibrium power with the secondary load is greatest when the reactivity coefficients are the most negative, which corresponds to end-of-life in a fuel cycle. Thus, in the absence of any protective actions, a reactor power level dictated by steam flow rate could be established.

Each steam line has a fast-closing isolation valve with a downstream check valve. These four valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. For example, in the case of a break upstream of the isolation valve in one line, closure of either check valve in that line or the isolation valve in the other line will prevent blowdown of the other steam generator. In particular, the arrangement precludes blowdown of more than one steam generator inside the containment and thus prevents structural damage to the containment. In addition each main steam line incorporates a 16 inch diameter venturi type flow restriction which is located near the Steam Generator inside the containment. These components serve to limit the rate of release of steam for any break downstream of the venturi. The replacement steam generators have similar field restrictions that are integral to the steam exit nozzle.

Sustained high feedwater flow would cause additional cooldown. Therefore, a safety injection signal will rapidly close all feedwater control valves, close the main feedwater containment isolation valves and, trip the main feedwater pumps. Tripping the main feedwater pumps will cause the associated feedwater pump discharge valve to close (assuming off site power is available).

Depending on the size and location of the break, a safety injection signal will be generated by one of the actuation signals. For large breaks inside containment, the safety injection signal is generated from either a low steamline pressure or a high containment pressure. For smaller breaks inside containment, reactor trips on the Overpower ΔT function may be generated. For breaks initiated from zero-power conditions, the primary SI signal is generated from low steamline pressure or low pressurizer pressure. For large steam line breaks outside of containment, the primary SI signal is generated from low pressurizer pressure. For small steam line breaks outside containment, the primary SI signal is generated by low steam line pressure. The core is then shut down by the boric acid injection delivered by the Safety Injection System. Once the faulted SG has completely blown down, the cooldown will cease allowing the operators to stabilize the plant at a reduced temperature using the intact SG.

14.5.5.3 Analysis of Transient

The analyses described in this section model the NSSS response during the blow-down of the faulted SG. They do not include modeling of the NSSS response after the SG has completely blown down. It is assumed that following the blow-down that the operators will take the necessary actions to stabilize the plant at a reduced temperature.

14.5.5.3.1 Containment Response

14.5.5.3.1.1 Methodology

The steamline break mass and energy releases are generated using the NRC-approved LOFTRAN code (Reference 7). LOFTRAN is used for studies of the transient response of a PWR system to specified perturbations in process parameters. The code simulates a multi-loop system including the reactor vessel, hot and cold leg piping, steam generator (shell and tube sides), and the pressurizer. A neutron point kinetics model is used and the reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator is modeled as a homogeneous saturated mixture. Protection and control systems are simulated, as well as the Emergency Core Cooling System. The steamline break mass and energy release methodology was approved by the NRC (Reference 88) and is documented in WCAP-8822 (Reference 89).

The containment integrity analysis uses the GOTHIC code as documented in WCAP-16219-P (Reference 64).

14.5.5.3.1.2 Key Physics Parameter Assumptions

The physics modeling is reviewed each refueling cycle to ensure that the analysis is bounding. If it is not bounded a new analysis is performed. The core modeling includes a conservative combination of:

a. Moderator Density Coefficient

b. Doppler Temperature Coefficient

c. Boron Coefficient

d. Shutdown Margin

14.5.5.3.1.3 Key System Parameter Assumptions

The following key system parameter assumptions are made to ensure the overall results of the analysis bound actual operation.

a. The initial power level is assumed to be full power, 70% power, 30% power or Hot Zero Power.

b. There is no loss of offsite power (LOOP) during the event.

c. The most reactive control rod cluster is in the fully withdrawn position.

d. The automatic run out protection of the AFW pumps is conservatively modeled. Cases are also analyzed that model the failure of the AFW pump runout protection.

e. Liquid entrainment in the steam blowdown from the broken SG is modeled for large breaks.

f. The steamline non-return check valves are credited to prevent blowdown from the intact SG.

g. The reactor coolant pumps (RCPs) are not tripped during the MSLB event.

h. The deposition of RCP heat into the primary side coolant is included.

i. Reverse SG heat transfer from the intact loop is conservatively modeled.

j. The Pressurizer pressure and level are assumed to be at the nominal programmed values consistent with the assumed initial power level.

k. The initial SG secondary side liquid inventory is conservatively higher than the nominal programmed value.

l. All of the major primary and secondary side metal heat structures are modeled to maximize the available stored energy.

m. The assumed core residual heat generation is based on the 1979 American Nuclear Society (ANS) decay heat plus 2 sigma model (Reference 33). 01469415

n. The unisolated portion of the steam line volume blows down in the containment during the break.

o. The water in the unisolated portion of the Main FW line usually reaches saturated conditions as the faulted SG depressurizes. The decrease in density as flashing occurs causes most of the unisolable feedwater to enter the faulted SG.

p. The Main FW pumps are on at full capacity until the FRV closes. When the FRV is postulated to fail open, the trip and coastdown of the Main FW pumps and condensate pumps are credited after an SI signal.

q. When the AFW pumps get the signal to start, it is conservatively assumed that all the AFW flow goes to the broken SG.

r. The AFW water is assumed to be at a higher than nominal temperature.

s. AFW flow to the affected SG is assumed to be terminated at 10 minutes after the break occurs due to operator action. (Note: this is the event that the run-out protection does not trip the AFW pumps or if the operators restart the pumps.)

t. Deleted

u. The Main FW regulator valve on the broken loop is assumed to fully open at the beginning of the event due to the mismatch between steam and FW flow at the initiation of the MSLB. At the FW Isolation signal, the Main FW regulator valve on the intact loop is assumed to instantly close. The Main FW regulator valve on the broken loop is assumed to remain fully open until it is assumed to close at the end of the stroke time.

v. The boron concentration of the water being injected by the SI system is assumed to be the RWST minimum boron concentration.

w. The level of SG tube plugging is conservatively modeled to be less than the fraction of tubes actually plugged.

x. No credit is taken for charging flow.

y. The turbine is tripped at the time of reactor trip.

z. The non-return check valves (NRCVs) isolate the SG and closure of the MSIVs are not modeled.

aa. The NRCVs are assumed to close 0.5 second after forward flow ceases to exist in that steam line.

ab. Containment concrete and metal heat structures listed in Reference 64 are modeled.

ac. The initial containment pressure is assumed to be at the maximum allowed per Technical Specifications.

ad. The containment spray and FCU systems actuation times are modeled with conservative delay times.

ae. A conservatively long time is used for the time required to fill the piping of the Spray System.

af. A conservatively low value is used for the capacity of the Containment Spray Pumps.

ag. The temperature of the RWST water that is used by the containment spray is assumed to be conservatively high.

ah. Conservatively low heat removal capability is assumed for the FCUs.

14.5.5.3.1.4 Single Active Failure Assumptions of a Safety Grade Component

Mitigation of a Steam Line Break is accomplished by isolation of the faulted SG, and actuation of the Safety Injection System, Containment Fan Coil System and Containment Spray System. Sensitivity studies show that the peak containment pressure occurs with the failure of a safeguards train. 01469415

14.5.5.3.1.5 Deleted

14.5.5.3.2 Core Response

14.5.5.3.2.1 Methodology

The computer codes used to analyze this transient are described in Section 14.3.

The Steamline Rupture - Core Response transient is analyzed at both full-power and zero-power conditions using the RETRAN computer code. Using the RETRAN code, transient values of key plant parameters identified as statepoints (core average heat flux, core pressure, core inlet temperature, RCS flow rate, and core boron concentration) are calculated. Next, the Westinghouse advanced nodal code (ANC) core design code is used to evaluate the nuclear response to the RCS cooldown to confirm the RETRAN transient prediction of the average core power/reactivity, and to determine the peaking factors associated with the return to power in the region of the stuck RCCA. Finally, using the RETRAN-calculated statepoints and the ANC-calculated peaking factors, the VIPRE computer code is used to perform detailed thermal-hydraulic calculations and to determine the minimum departure from nucleate boiling ratio (DNBR), based on the W-3 or WRB-1 DNB correlations.

The purpose of the analysis from full-power conditions is to demonstrate that a reactor trip occurs in adequate time to ensure fuel and cladding damage is precluded. Breaks of various sizes are postulated to occur in the steamline upstream of the Main Steam Isolation Valve (MSIV). A range of break sizes up to 1.4 ft^2 is considered. The larger break sizes generate reactor trips on the low steamline pressure - safety injection - reactor trip function while smaller breaks trip on the Overpower ΔT (OPΔT) reactor trip function. The most limiting break size is typically the largest break case that results in a reactor trip on the OPΔT reactor trip function.

The purpose of the analysis from zero-power conditions is to demonstrate that with the high-power peaking factors that may exist when the most reactive rod cluster control assembly (RCCA) is stuck in its fully withdrawn position, the emergency core cooling system (safety injection) is actuated in adequate time to ensure fuel and cladding damage is precluded and the core is ultimately shut down by the boric acid injected into the RCS. The most limiting (largest) break size postulated to occur in the steamline upstream of the Main Steam Isolation Valve (MSIV) is analyzed, with and without offsite power available.

14.5.5.3.2.2 Key Physics Parameter Assumptions

The following key physics parameter assumptions are made in conservatively analyzing the Steamline Rupture - Core Response event from full-power conditions:

a. Moderator Density Coefficient: a most positive value is assumed.

b. Doppler Temperature Coefficient: a most negative value is assumed.

c. Doppler Power Defect: a least negative value is assumed.

d. Effective Delayed Neutron Fraction: a minimum value is assumed.

For the analysis of the Steamline Rupture - Core Response transient from zero-power conditions, the key physics parameter assumptions are consistent with an end-of-life shutdown margin of 1.7-percent $\Delta k/k$ corresponding to no-load, equilibrium xenon conditions, with the most reactive RCCA stuck in its fully withdrawn position. The stuck RCCA is assumed to be in the core location exposed to the greatest cooldown; that is, related to the faulted loop. The reactivity feedback model includes a positive moderator density coefficient (MDC) corresponding to an end-of-life rodded core with the most reactive RCCA in its fully withdrawn position. The variation of the MDC due to changes in temperature and pressure is accounted for in the model. The Doppler reactivity defect associated with power, assuming the stuck RCCA, is also accounted for in the model, as presented in Figure 14.5-9.

14.5.5.3.2.3 Key System Parameter Assumptions

Steamline Rupture - Full Power Core Response:

The following key system parameter assumptions are made to ensure the overall results of the analysis bound actual plant operation for the hot full power case up to 1677 MWt:

a. Initial conditions of core power, RCS coolant temperature and pressurizer pressure are assumed to be at their nominal values consistent with steady-state full power operation. Uncertainties in the initial conditions of these parameters are not considered, consistent with the application of the Revised Thermal design Procedure (RTDP) methodology. Steam generator water level is assumed to be at its nominal value.

b. Minimum measured flow is modeled consistent with the RTDP methodology.

c. 0% steam generator tube plugging (SGTP) level is assumed; this maximizes primary-to-secondary heat transfer and results in a more severe cooldown transient.

d. Pressurizer sprays and power-operated relief valves (PORVs) are assumed to be operational.

e. Manual rod control is assumed.

Reactor trip is initiated on a safety injection signal on low steam line pressure in any loop or directly on OPΔT in both loops, depending on the break size; turbine trip is initiated on the reactor trip signal. No decay heat is assumed.

Steamline Rupture - Zero Power Core Response:

The following key system parameter assumptions are made to ensure the overall results of the analysis bound the actual plant response to a steamline rupture at zero-power conditions:

a. Conditions corresponding to a subcritical reactor, an initial vessel average temperature at the no-load value of 547°F, and no core decay heat. These conditions are conservative for a steamline break transient because the resultant RCS cooldown does not have to remove any latent heat. Also, the steam generator water inventory is greatest at no-load conditions, which increases the capability for cooling the RCS.

b. The non-return check valves are neglected to conservatively allow blowdown from both steam generators up to the time of MSIV closure. This assumption is made, along with not crediting containment protection signals, to assure that any postulated break location or single failure assumption is bounded by a single analysis.

c. The closure of the MSIV in the faulted loop is conservatively modeled to be complete after receipt of a safety injection signal due to the coincidence of a hi-hi steam flow rate signal and a lo-lo steam line pressure signal from the same loop.

d. The safety injection pumps are assumed to provide flow to the RCS after receipt of the safety injection signal and delays that account for signal processing and pump startup delays, and, as applicable (for the case without offsite power available), diesel generator startup time.

e. The minimum capability for the injection of highly concentrated boric acid solution, corresponding to the most restrictive single active failure in the safety injection system (SIS), is assumed. The assumed safety injection flow (see Figure 14.5-10) corresponds to the operation of one high-head safety injection pump. Boric acid solution from the refueling water storage tank (RWST), with a minimum concentration and minimum temperature, is the assumed source of the safety injection flow. The safety injection lines downstream of the RWST are assumed to initially contain unborated water to conservatively maximize the time it takes to deliver the highly concentrated RWST boric acid solution to the reactor coolant loops.

f. The safety injection accumulator tanks (one per loop) provide a passive injection of borated water into the RCS. The accumulators are assumed to have a boron concentration of 2300 ppm and a minimum temperature. 01469415

g. Main feedwater flow equal to the nominal (100-percent power) value is assumed to initiate coincident with the postulated break, and is maintained until feedwater isolation occurs. The feedwater enthalpy is assumed to be 20.65 Btu/lbm, corresponding to 50°F.

h. A minimum SGTP level of 0 percent is assumed to maximize the cooldown of the RCS.

i. Maximum auxiliary feedwater at a minimum temperature is assumed to initiate coincident with the postulated break to maximize the cooldown of the RCS.

14.5.5.3.2.4 Single Active Failure Assumptions of a Safety Grade Component

For the Steamline Rupture - Core Response transient analyzed from full-power and zero-power conditions, the limiting single failure is assumed to be failure of one protection train. The loss of one protection train is the most limiting and does not affect the results of the transient since the protection function is carried out by the other train of the protection system.

14.5.5.3.3 Small Steam Line Break Response

14.5.5.3.3.1 Methodology

Small steam line break transients associated with the inadvertent opening of a steam dump or steam generator relief valve were not explicitly analyzed because the resultant reactor coolant system cooldown, and thus the minimum DNBR, would be less limiting compared to the double-ended rupture cases.

14.5.5.3.3.2 Deleted

14.5.5.3.3.3 Deleted

14.5.5.3.3.4 Deleted

14.5.5.3.4 Deleted

14.5.5.3.4.1 Deleted

14.5.5.3.4.2 Deleted

14.5.5.3.4.3 Deleted

14.5.5.3.4.4 Deleted

14.5.5.4 Acceptance Criteria

Several different break sizes are analyzed for the Steamline Rupture - Core Response transient. The applicable criteria for the steamline break event are discussed below. Note that depending upon the break size, the event is considered to be either a Condition III or IV event. However, some Condition II events are indistinguishable from a minor steamline break with respect to the primary system response and must satisfy Condition II criteria. Examples of such events include an excessive load increase and steam system valve malfunction events. Therefore, a subset of the Condition II criteria are applied for all break sizes analyzed for ease of interpretation.

The Main Steamline Rupture accident is classified as a Condition IV event. The design criteria for Condition IV events are as follows:

> Condition IV faults shall not cause a release of radioactive material that results in an undue risk to public health and safety exceeding the guidelines of 10CFR100, “Reactor Site Criteria.”
>
> A single Condition IV fault shall not cause a consequential loss of required functions of systems needed to cope with the fault including those of the reactor coolant system and the reactor containment system.

The applicable subset of design criteria for Condition II steamline breaks, per ANSI N18.2-1973, is as follows:

Any release of radioactive materials in effluents to unrestricted areas shall be in conformance with paragraph 20.1 of 10 CFR Part 20, “Standards for Protection Against Radiation.” 01517422

A single Condition II incident shall not cause consequential loss of function of any barrier to the escape of radioactive products.

The specific criteria applied in performing the analysis for this event are as follows:

1. The pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values. Since this event results in a decrease in both the primary and secondary side pressures, these criteria are not challenged by a steamline break event and are therefore not analyzed for this event.

2. The stringent criterion of satisfying the DNB design basis is applied for this Condition III/IV event, which requires that the minimum departure from nucleate boiling ratio (DNBR) must be greater than the applicable limit for the DNBR correlation being used accounting for the penalties and factors described in Section 3.2.2, “Thermal Hydraulic Design Analysis”. It has been traditional practice to assume that fuel failure will occur if fuel centerline melting takes place. Therefore, for the analysis from full-power conditions, the fuel damage criteria also include demonstrating that the peak linear heat generation rate (expressed in kW/ft) does not exceed a value which would cause fuel centerline melt.

3. The containment vessel internal pressure must not exceed the designed maximum listed in USAR Section 5.2.

The Steamline Rupture - Core Response transient is primarily analyzed for DNBR, and in the case of a steamline break from full-power conditions, over-power concerns.

14.5.5.5 Results and Radiological Consequences

The transient response for the Steamline Rupture - Full Power Core Response analysis is shown in Figures 14.5-2 through 14.5-8. Table 14.5-4 provides the time sequence of events for the limiting break size.

For the Steamline Rupture - Zero Power Core Response analysis, a hypothetical double-ended rupture (DER) of a main steamline was postulated and cases were considered both with and without offsite power available. The limiting (maximum) break size is effectively limited to the flow area of the steam generator outlet nozzle flow restrictors (1.4 ft^2 per steam generator). The transient response for the limiting Steamline Rupture - Zero Power Core Response analysis (with offsite power available) is presented in Figures 14.5-11 through 14.5-20. Table 14.5-5 provides the time sequence of events for the limiting analysis.

The results of the Unit 1 containment response following a MSLB provide a peak pressure of 44.2 psig at 211 seconds and a peak temperature of 310.8°F at 86 seconds, see Figures 14.5-23A and 14.5-24A.

The results of the Unit 2 containment response (modeling the RSGs) following a MSLB provide a peak pressure of 44.3 psig at 213 seconds and a peak temperature of 310.7°F at 86 seconds; see Figure 14.5-23B and Figure 14.5-24B, respectively. 01469415

Radiological consequences are not evaluated for MSLB inside of containment because no fuel pins are expected to experience departure from nucleate boiling and thus experience cladding failure.

14.5.5.6 Dose Analyses for MSLB Outside of Containment

RADTRAD is used to calculate the Control Room and Offsite dose due to airborne radioactivity releases following a MSLB. (See USAR Appendix D.)

The MSLB dose assessment supports the implementation of Alternate Repair Criteria (ARC) as defined in USNRC GL 95-05 (Reference 102) and previously approved in PINGP License Amendment Number 133 and 125. (Reference 101) In accordance with GL 95-05, the MSLB dose assessment utilizes the maximum allowable accident induced leakage that results in dose consequences that are just within the most limiting of the regulatory limits associated with the EAB, LPZ and the Control Room. (Reference 104)

ARC methodology is utilized herein for both Unit 1 and 2, which is conservative.

Table 14.5-8 lists some of the key assumptions / parameters utilized to develop the radiological consequences following the MSLB.

The radiological model used for the MSLB assessment conservatively assumes an almost immediate dry-out of the faulted SG following a MSLB resulting in a release of all of the contents of the steam generator. It is noted that for this release pathway, due to SG dryout, 107,100 lbs of secondary liquid is released to the environment in 10 minutes. The nitial concentration of iodine in the steam generator liquid is assumed to be at Tech Spec levels.

A simultaneous Loss of Offsite Power is assumed rendering the condenser unavailable, and environmental steam releases are postulated via the MSSVs / SG PORVs of the intact steam generator until shutdown cooling is initiated at T=45.4 hours. The elevated iodine activity in the primary coolant due to a postulated pre-accident or concurrent iodine spike, as well as the noble gas (at the Technical Specification concentrations for the primary coolant), leak into the faulted and intact steam generators, and are released to the environment from the common area of the Auxiliary Building, and from the MSSVs / SG PORVs, respectively.

The steam releases from the intact steam generator continue until shutdown cooling is initiated via operation of the Residual Heat Removal (RHR) System at T=45.5 hours, resulting in the termination of environmental releases via this pathway. Additionally, the releases from the faulted SG due to primary to secondary leakage is terminated at T=75 hours after the accident.

In accordance with the guidance provided in GL 95-05, increased primary-to-secondary leakage (i.e., in addition to that allowed by the Technical Specification) is postulated to occur via pre-existing tube defects as a result of the rapid depressurization of the secondary side due to the MSLB and the consequent high differential pressure across the faulted steam generator. In accordance with the referenced guidance, the MSLB dose analysis is performed to establish a maximum allowable accident-induced leakage, against which the cycle leakage projections can be compared.

14.5.5.6.1 Source Terms

Since there is no postulated fuel damage associated with this accident, the main radiation source is the activity in the primary and secondary coolant system. For the primary coolant, two spiking cases are addressed: a pre-incident iodine spike and a coincident iodine spike.

a) Pre-incident spike – the initial primary coolant iodine activity is assumed to be 60 times the Technical Specification Limit of 0.5 μCi/gm DE I-131 which is the transient Technical Specification limit for full power operation. The initial primary coolant noble gas activity is assumed to be at Technical Specification levels.

b) Coincident spike – Immediately following the accident the iodine appearance rate from the fuel to the primary coolant is assumed to increase to 500 times the equilibrium appearance rate corresponding to the Technical Specification coolant concentrations. The duration of the assumed spike is 8 hours. The initial primary coolant noble gas activity is assumed to be at Tech Spec levels.

The secondary coolant iodine activity, just prior to the accident is assumed to be at the Technical Specification limit of 0.1 μCi/gm DE I-131.

14.5.5.6.2 Coolant Activity

The design basis (1% fuel defects) primary coolant activity inventory used in the MSLB dose analysis reflects the changes in fuel design and fuel management schemes utilized by PINGP. The coolant inventory reflects the higher coolant isotopic inventories between the current OFA fuel and the Heavy Bundle Fuel (HBF). (See Appendix D for additional information.)

14.5.5.6.3 Release Path and Activity Transport

Following a MSLB, primary and secondary coolant activity is released to the environment via two pathways, i.e., via the break point of the faulted SG, and via the MSSVs / SG PORVs of the intact SG.

Faulted Steam Generator

The release from the faulted Steam Generator occurs via the postulated break point of the main steam line. The faulted steam generator is conservatively assumed to dry-out almost instantaneously (~10 minutes) following the MSLB, releasing all of the iodine in the secondary coolant that was initially contained in the steam generator. The secondary steam activity initially contained in the faulted steam generator is also released. The primary to secondary tube leakage in the faulted SG is assumed to increase from 150 gpd to 1 gpm (at 70°F) as a result of accident induced conditions. All iodine and noble gas activities in the referenced tube leakage are released directly to the environment without hold-up or decontamination. The primary to secondary leakage continues for 75 hours. Release of the SG inventory (including primary-to-secondary leakage) is via the common area of the Auxiliary Building.

Intact Steam Generator

The releases from the intact steam generator occur via the SG MSSVs / PORVs. The iodine activity in the intact SG liquid is released to the environment in proportion to the steaming rate and the partition coefficient. Steam releases from MSSVs / SG PORVs, terminate within 45.5 hours of the DBA due to initiation of RHR. Per the plant Technical Specifications, the primary to secondary leakage in the intact SG is 150 gpd. Steam releases to the environment occur via the MSSVs & SG PORV during the cool down phase. The analysis assumes that the release point is either the MSSVs or the SG PORV, which ever has the worse atmospheric dispersion factor. It is noted that the SG PORVs are located in the same area as the MSSVs. Steam can also be released from the Turbine Driven AFW Pump steam exhaust. Analyses showed that modeling the PORV was more conservative.

14.5.5.6.4 Accident Atmospheric Dispersion Factors (χ/Q)

Offsite Atmospheric Dispersion Factors

The Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors (χ/Q) are listed in Table 14.5-8. The EAB value is taken directly from Appendix H. The values for the LPZ are derived by linear interpolation from χ/Q values listed in Table XIV of Appendix H at a distance of 2,414 m.

Control Room Atmospheric Dispersion Factors

The Control Room air intake and center of Control Room ceiling (called center of control room hence forth) χ / Q values for the release points applicable to the MSLB are calculated using "**A**tmospheric **R**elative **CON**centrations in Building Wakes" (ARCON96) methodology (Ramsdell, 1997, Reference 93). Input data consist of hourly on-site meteorological data; release characteristics such as release height, stack radius, stack exit velocity, and stack flow rate; the building area affecting the release; and various receptor parameters such as its distance and direction from the release to the control room air intake and intake height.

On-site hourly met data (1993 through 1997) was utilized to develop the ARCON96 on-site atmospheric dispersion factors used in the MSLB dose consequence analyses.

All releases are conservatively treated as ground-level as there are no releases at the site that are high enough to escape the aerodynamic effects of the plant buildings (i.e., 2.5 times Shield Building height, per Reference 103). In addition, the stack/vent release flows are not necessarily maintained throughout the accident period.

The specific release-receptor combinations for which χ / Q values are calculated are as follows:

1. Unit 1 and Unit 2 Main Steam Safety Valves/Steam Generator Power Operated Relief Valve (MSSVs/SG PORVs) to the Unit 1 and Unit 2 Control Room Air Intakes and the Control Room Center (Diffuse Source)

2. Unit 1 and Unit 2 SG PORVs to the Unit 1 and Unit 2 Control Room Air Intakes and the Control Room Center (Point Source)

3. Common Area of Auxiliary Building to Unit 2 Control Room Air Intake (Diffuse Source). The determination of the values for this source-receptor pair are described in Section 14.5.1.1.

The following assumptions are made for these calculations:

- The MSSVs/SG PORV releases are from the centroid of a rectangle encompassing the valves and are treated as a diffuse area source only when releases occur simultaneously from both the MSSVs and SG PORVs

- The SG PORV releases are from the centroid of a rectangle encompassing the valves and are treated as a point source when releases occur only from the SG PORVs

- Initial diffusion coefficients for diffuse sources are based on the recommendations in Regulatory Guide 1.194 (Reference 103).

- The wind direction range (90 degrees), wind speed assigned to calm (0.5 m/sec), surface roughness length (0.2 meter), and sector averaging constant (4.3) are taken from the recommendations in Regulatory Guide 1.194. (Reference 103)

- All releases are conservatively treated as ground level releases as there are no release conditions that merit categorization as an elevated release (i.e., 2.5 times Shield Building height) at this site.

- The plume centerline from each release is conservatively transported directly over the receptor

- Control Room Unfiltered In-leakage: the χ/Q from the accident release point to the Control Room Vent Intake is conservatively utilized for Control Room in-leakage.

- Control Room Ingress/Egress: the χ/Q from the accident releases point to the Control Room Vent Intake is utilized for Control Room ingress/egress. The doors to the Control Room are located on the north side (i.e., Turbine Building side) of the Control Room as well as in the northeast and northwest corners. All release points for a MSLB are located south of the Control Room. Therefore, the distances from these release points to the Control Room Vent Intake are conservative (i.e., shorter) relative to the Control Room doors.

The bounding χ/Q values (taking into consideration an accident at either unit) for the release-receptor combinations applicable to the MSLB are selected and provided in Table 14.5-8.

14.5.5.6.5 Dose Model

Doses to the offsite and control room were calculated using RADTRAD. The code calculates an integrated release for defined time periods at a location of interest. The integrated activity is used to calculate a cumulative dose at the location of interest using the dose models and methodology described in USAR Appendix D.

14.5.5.6.6 Control Room Model

During normal plant operation, the maximum total supply of unfiltered outside air to the control room is 2000 cfm (1818 scfm + 10%).

The control room (with a calculated free volume of 61,315 ft3) is located in the Auxiliary Building at El. 735′ and is equidistant from both units. The control room envelope includes the chiller rooms located directly above the control room at elevation 755′ but does not include the cable spreading room located directly below the control room at Elevation 715′, or the Operations Lounge and Records Room located adjacent to the control room. Note that operator occupancy and habitability determination is limited to Elevation 735′ only.

Since the PINGP control rooms are contained in a single control room envelope, they are modeled as a single region. Isotopic concentrations in areas outside the control room envelope are assumed to be comparable to the isotopic concentrations at the center of the control room. To support development of bounding control room doses, the most limiting χ / Q associated with the release point/receptor for an accident at either unit is utilized.

Prior to isolation, the control room post-accident ventilation model utilized in the dose analysis corresponds to an assumed "single intake" which utilizes the worst case atmospheric dispersion factor (χ / Q) from release points associated with an accident occurring at either unit to the limiting control room intake.

The plant design will automatically isolate the control room and initiate control room filtered recirculation at 3600 cfm (4000 cfm - 10%) via the PINGP Control Room Emergency Ventilation System upon receipt of an SI signal from either unit, or a high radiation alarm from the control room in-duct radiation monitors. For specific operating configurations of the ventilation system, the high radiation signal is relied on as backup to the SI signal in the event of a single failure. The delay in crediting control room emergency isolation / filtered recirculation is assumed to be ~5 minutes based on receipt of a high radiation signal. This delay is sufficient to address a Loss of Offsite Power (LOOP) that takes into account the delay associated with the diesel generator becoming fully operational (including sequencing delays), damper closure / re-alignment, and the time it takes for the emergency recirculation filtration fans to come up to speed. The control room recirculation filters are 99% efficient for removing airborne particulates and 95% efficient for removing airborne organic and elemental iodine.

The dose model conservatively assumes that prior to achieving control room isolation the unfiltered intake flow into the control room is equivalent to the intake associated with normal operation, i.e. 2000 cfm. This is conservative since loss of normal ventilation would result in reducing the amount of contaminated air entering the control room via the intake prior to isolation.

A control room unfiltered inleakage of 300 cfm is assumed prior to and during the time it is isolated. The value for control room unfiltered inleakage is based on the results of tracer gas testing in the isolated mode of 290 cfm, and includes a 10 cfm unfiltered inleakage due to ingress / egress as recommended by NUREG-0800 SRP 6.4 (Reference 97).

Table 14.5-8 lists key assumptions / parameters associated with the control room design.

14.5.5.6.7 SUMMARY OF RESULTS AND CONCLUSIONS

The dose consequences at the exclusion area boundary, low population zone and Control Room due to a design basis main steamline break are as follows:

The dose at the EAB during the limiting 2 hours, and the dose at the LPZ "for the duration of the release" represent the post accident dose to the public due to inhalation and submersion for each of these events. These values are presented below. Due to distance/plant shielding, the dose contribution at the EAB/LPZ due to direct shine from contained sources is considered negligible for all accidents. The associated regulatory limits are also presented.

Per regulatory guidance, the dose at the Control Room is integrated over 30 days. The calculated doses address the fact that for events with durations less than 30 days, the CR dose needs to include the remnant radioactivity within the CR envelope after the event has terminated. The 30-day integrated dose to the control room operator, due to inhalation and submersion, is presented below.

The dose consequences at the EAB, LPZ, and Control Room following a MSLB remain below the regulatory limits in 10 CFR 50.67 and Regulatory Guide 1.183.

MSLB Dose Results with a Pre-Existing Iodine Spike		
Location	Acceptance Criteria (rem)	TEDE (rem)
Exclusion Area Boundary	25	0.11
Low Population Zone	25	0.05
Control Room	5.0	0.79

MSLB Dose Results with a Concurrent Iodine Spike		
Location	Acceptance Criteria (rem)	TEDE (rem)
Exclusion Area Boundary	2.5	0.55
Low Population Zone	2.5	0.26
Control Room	5.0	4.04

14.5.6 Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)

14.5.6.1 Description of Accident

This accident is a result of an extremely unlikely mechanical failure of a control rod mechanism pressure housing such that the Reactor Coolant System pressure would then eject the RCCA and drive shaft. The consequences of this mechanical failure, in addition to being a minor loss of coolant accident, may also be a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage for severe cases. The resultant core thermal power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by reactor trip actuated by high neutron flux signals.

14.5.6.2 Design Characteristics

Certain features in Westinghouse pressurized water reactors are intended to preclude the possibility of a rod ejection accident, and to limit the consequences if the accident were to occur. These include a sound, conservative mechanical design of the rod housings, together with a thorough quality control (testing) program during assembly, and a nuclear design which lessens the potential ejection worth of RCCAs and minimizes the number of assemblies inserted at high power levels.

The mechanical design is discussed in Section 3. An evaluation of the mechanical design and quality control procedures indicates that a failure of a control rod mechanism housing sufficient to allow a control rod to be rapidly ejected from the core should not be considered credible for the following reasons:

a. Each control rod drive mechanism housing is completely assembled and shop-tested.

b. The mechanism housings were hydrotested after assembly to the reactor vessel head during shop fabrication.

c. Stress levels in the mechanism are not affected by anticipated system transients at power, or by the thermal movement of the coolant loops. Moments induced by the design earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class 1 components.

d. The latch mechanism housing and rod travel housing are each a single length of forged Type-316 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered.

A significant margin of strength in the elastic range together with the large energy absorption capability in the plastic range gives additional assurance that gross failure of the housing will not occur. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are full penetration welds. Administrative requirements require periodic inspections of these types of welds.

Even if a rupture of the control rod mechanism housing is postulated, the operation of a chemical shim system is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with RCCAs inserted only far enough to permit load follow. Reactivity changes caused by core depletion and xenon transients are compensated by boron changes.

Further, the location and grouping of control rod banks are selected during nuclear design to lessen the severity of an ejected assembly. Therefore, should an RCCA be ejected from the reactor vessel during normal operation, there would probably be no reactivity excursion - since most of the RCCAs are fully withdrawn from the core - or a minor reactivity excursion if an inserted assembly is ejected from its normal position.

However, it may be occasionally desirable to operate with larger than normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the RCCAs above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all assemblies is continuously indicated in the control room. An alarm will occur if a bank of RCCAs approaches its insertion limit or if one assembly deviates from its bank. There are low and low-low level insertion monitors with visual and audio signals. The RCCA position monitoring and alarm systems are described in detail in Section 7.

The reactor protection system provides core protection in the event of a rod ejection accident. This system is described in more detail in Section 7.

Disregarding the remote possibility of the occurrence of a control rod mechanism housing failure, investigations have shown that failure of a control rod housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings that would increase the severity of the initial accident.

Due to the extremely low probability of a rod ejection accident, some fuel damage could be considered an acceptable consequence, provided there is no possibility of the offsite consequences exceeding the requirements of 10CFR50.67. Although severe fuel damage to a portion of the core may in fact be acceptable, it is difficult to treat this type of accident on a sound theoretical basis. For this reason, criteria for the threshold of fuel failure are established, and it is demonstrated that this limit will not be exceeded.

Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy, have been carried out as part of the SPERT project by the Idaho Nuclear Corporation (Reference 10). Extensive tests of UO_2 - Zirconium clad fuel rods representative of those in PWR type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design have exhibited failures as low as 225 cal/gm. These results differ significantly from the TREAT (Reference 11) results, which indicated a failure threshold of 280 cal/gm. Limited results have indicated that this threshold decreases by about 10% with fuel burnup. The clad failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure, (large fuel dispersal, large pressure rise) even for irradiated rods, did not occur below 300 cal/gm.

The ultimate acceptance criteria for this event is that any consequential damage to either the core or the RCS must not prevent long-term core cooling, and that any offsite dose consequences must be within the requirements of 10CFR50.67. To demonstrate compliance with these requirements, it is sufficient to show that the RCS pressure boundary remains intact, and that no fuel dispersal in the coolant, gross lattice distortions, or severe shock waves will occur in the core. Therefore, the following acceptance criteria are applied to the RCCA Ejection accident:

a. Maximum average fuel pellet enthalpy at the hot spot must remain below 200 cal/g (360 Btu/lbm).

b. Peak RCS pressure must remain below that which would cause the stresses in the RCS to exceed the Faulted Condition stress limits.

c. Maximum fuel melting must be limited to the innermost 10% of the fuel pellet at the hot spot, independent of the above pellet enthalpy limit.

Method of Analysis

The analysis of the control rod ejection accident requires modeling of the neutron kinetics coupled with the fuel and clad heat up condition and the thermal hydraulics of the coolant channel. The analysis is performed by first calculating the core average neutronic response and then using the resulting core average power response as a forcing function for the hot spot thermal evaluation.

The computer codes used to perform the analyses are described in Section 14.3. Additional details of the methodology are provided in WCAP-7588 (Reference 3).

A 1-D axial kinetics model is used in the TWINKLE code for the analysis of the core average response, since it allows for a more realistic representation of the spatial effects of axial moderator feedback, power distribution, and RCCA movement. The moderator reactivity effect is included by correlating reactivity with moderator density, thereby including the effects of coolant temperature, pressure, and voiding. The Doppler reactivity effect is correlated as a function of fuel temperature. The largest temperature rise during the transient, and hence the largest reactivity effects, occurs in channels where the power is higher than average. As a result, when a 3-D space time kinetics calculation is not performed, weighting factors are applied as multipliers to the average channel Doppler reactivity feedback to account for spatial reactivity feedback effects.

The average core energy addition, calculated as described above, is multiplied by the appropriate hot channel factors, and the hot spot analysis is performed using a detailed fuel and clad transient heat-transfer computer code, FACTRAN. This computer code calculates the transient temperature distribution in a cross-section of a metal-clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power vs time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A parabolic radial power distribution is used within the fuel rod.

The computer code uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandberg-Tong correlation to determine the film boiling coefficient after DNB (Reference 31). The DNB heat flux is not calculated, instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat-transfer coefficient may be calculated by the code; however, it is adjusted in order to force the full-power steady-state temperature distribution to agree with the fuel heat-transfer design codes.

The overpressurization of the RCS and number of rods in DNB, as a result of a postulated ejected rod, have both been analyzed on a generic basis for Westinghouse PWRs as detailed in Reference 3.

If the safety limits for fuel damage are not exceeded, there is little likelihood of fuel dispersal into the coolant or a sudden pressure increase from thermal-to-kinetic energy conversion. The pressure surge for this analysis can, therefore, be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

A detailed calculation of the pressure surge for an ejection worth of one dollar at BOL, hot full power, indicates that the peak pressure does not exceed that which would cause stresses in the RCS to exceed their Faulted Condition stress limits. Since the severity of the Prairie Island analysis does not exceed this worst case analysis, the RCCA Ejection accident will not result in an excessive pressure rise or further damage to the RCS.

Reference 3 also documents a detailed multi-channel thermal-hydraulics code calculation, which demonstrates an upper limit to the number of rods-in-DNB for the RCCA Ejection accident as 10%. Since the severity of the Prairie Island analysis does not exceed this worst case analysis, the maximum number of rods in DNB following a RCCA Ejection will be less than 10%, which is well within the value currently used in the radiological dose evaluation. The most limiting break size resulting from a RCCA Ejection will not be sufficient to uncover the core or cause DNB at any later time. Since the maximum number of fuel rods experiencing DNB is limited to 10%, the fission product release will not exceed that associated with the requirements of 10CFR50.67.

In calculating the nuclear power and hot spot fuel rod transients following RCCA Ejection, the following conservative assumptions are made:

a. Maximum uncertainties in initial conditions are employed. The analysis assumes uncertainties of +4.0°F in nominal vessel T_{avg}, and -60 psi in nominal system pressure. A reactor power level of 1683 MWt was modeled, consistent with the maximum reactor power including all applicable uncertainties.

b. A minimum value for the delayed neutron fraction for BOC and EOC conditions is assumed which increases the rate at which the nuclear power increases following RCCA Ejection.

c. A minimum value of the Doppler power defect is assumed which conservatively results in the maximum amount of energy deposited in the fuel following RCCA Ejection. A minimum value of the moderator feedback is also assumed. A positive moderator temperature coefficient is assumed for the beginning of cycle, zero power case.

d. Maximum values of ejected RCCA worth and post-ejection total hot channel factors are assumed for all cases considered. These parameters are calculated using standard nuclear design codes for the maximum allowed bank insertion at a given power level as determined by the rod insertion limits. No credit is taken for the flux flattening effects of reactivity feedback.

e. The start of rod motion occurs 0.45 seconds after the high neutron flux trip point is reached.

The analysis is performed to bound operation with Westinghouse fuel (UO_2 and up to 8 w/o gadolinia-doped UO_2) and a maximum loop-to-loop steam generator tube plugging imbalance of 10%. The analyses for both 400V+ and 422V+ fuel types are summarized in the results section.

Results

A summary of the cases is given in Table 14.5-6. The calculated sequence of events is presented in Table 14.5-7. The nuclear power and hot spot and cladding temperature transients are presented in Figures 14.5-25 through 14.5-32 for 400V+ fuel and Figures 14.5-33 through 14.5-44 for 422V+ fuel.

For the hot full power cases, the peak hot spot fuel centerline temperature reached the fuel melting temperature; however, melting was restricted to less than 10 percent of the pellet. For the hot zero power cases, no fuel melting was predicted. The UO_2 cases are bounding for all fuel types, including gadolinia-doped fuel.

For all cases, reactor trip occurs very early in the transient, after which the nuclear power excursion is terminated. The reactor will remain subcritical following reactor trip.

14.5.6.3 Radiological Consequences

The Rod Cluster Control Assembly (RCCA) Ejection Accident is also referred to as the Control Rod Ejection Accident (CREA). The key inputs and assumptions used in the CREA radiological consequence analysis analyzed using the Alternative Source Term (AST) are summarized below and provided in Table 14.5-13.

Consistent with Regulatory Guide 1.183, the CREA dose assessment is analyzed using two cases. The two release cases are combined to determine the dose consequences.

In the first case, the activity released from the failed fuel is assumed to be released instantaneously and homogeneously throughout the containment atmosphere and available for release to the environment via containment leakage.

In the second case, the activity released from the failed fuel is assumed to be completely dissolved in the reactor coolant system (RCS), which is also referred to as primary coolant. Primary to secondary coolant leakage transfers activity into the secondary side of the Steam Generators. This makes it available for release into the environment via steaming through the Power Operated Relief Valves (PORV) and via the turbine driven auxiliary feedwater (TDAFW) pump steam exhaust. It has been shown that it is more conservative to assume the release is from the PORVs in lieu of the TDAFW pump steam exhaust.

<u>Fuel Damage and Core Source Term</u>

For conservatism, the CREA core source term is that associated with a power level of 1,852 MWth.

The CREA results in clad damage to 10% of the fuel. The design basis of this accident assumes that 0.25% fuel melt is postulated to occur.

Core and RCS Release: The following activity is assumed to be instantaneously and homogeneously distributed in the containment and primary coolant following a CREA:

1. 10% of the core iodine and 10% of the core noble gases in the fuel gap of clad damaged fuel is released into containment and available for release from containment,

2. 10% of the core iodine and 10% of the core noble gases in the fuel gap of clad damaged fuel is released into the RCS and available for primary-to-secondary leakage,

3. 25% of the core iodine and 100% of the core noble gases in the melted fuel is released into containment and available for release from containment,

4. 50% of the core iodine and 100% of the core noble gases in the melted fuel is released into the RCS and available for primary-to-secondary leakage,

5. 100% of the iodine and noble gasses initially present (i.e., pre-rod ejection accident) in the primary coolant associated with 1% fuel defects. The modeling of the pre-accidentprimary coolant iodine and noble gas activity associated with 1% fuel defects represents discretionary conservatism relative to the 0.5 μCi/gm DE I-131 and 580 μCi/gm DE Xe-133 equilibrium primary coolant activity concentration Technical Specification limits.

Release Rates and Partitioning Factors

During the first 24 hours the containment is assumed to leak at its maximum Technical Specification leak rate of 0.15 volume percent per day and at 50% of this leak rate for the remaining duration of the accident. No credit is taken for a reduction in the amount of radioactive material available for leakage from the containment due to natural deposition and containment spray.

The containment leakage is initially released to the environment as unfiltered bypass leakage. Following a drawdown time, a portion of the leakage is released to the environment as filtered leakage via the Auxiliary Building Special Ventilation Zone (ABSVZ). Following a drawdown time, another portion of the leakage is released to the environment as unfiltered leakage via the ShieldBuilding (SB). After a period of time, Shield Building recirculation filters begin operation. Values for containment leakage, Shield Building Ventilation system and Auxiliary Building Special Ventilation system performance are provided in Section 14.9.

Activity that originates in the primary coolant is released to the secondary coolant by means of the primary-to-secondary coolant leak rate. This design basis leak rate value is 150 gpd into each of the two steam generators (SGs). For input into RADTRAD, this rate was converted to a total leak rate of 0.0376 cubic feet per minute into both SGs.

The methodology used to model steaming of activity through SG PORVs following the postulated CREA assumes an average cumulative release rate through these paths. The partitioning factors are applied to these release rates. Incremental steam mass releases are given in pounds per time interval. For the time intervals used in this accident scenario, release rates were derived by taking the averages of mass releases over each specified time interval. Then these mass flow rates were converted to volumetric flow rates using the assumption of cooled liquid conditions (i.e., 62.4 lbm/ft^3), as specified by the applicable guidance of Regulatory Guide 1.183.

For all post-accident releases through the SG PORVs, the mechanism for release to the environment is steaming of the coolant in the secondary system. Because of this release dynamic, Regulatory Guide 1.183 allows for a reduction in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water. For Iodine, the partitioning factor of 0.01 was used. Reviewing the specified AST release fractions, it is concluded that the only nuclides to be released from the core source term, other than iodines, are noble gas nuclides, and because of the volatility of noble gases, no partitioning is assumed for any such isotopes.

<u>Control Room χ/Q Calculations (Meteorology)</u>

The containment releases are via the Shield Building vent stack. The secondary releases are viathe SG PORVs. The limiting control room atmospheric dispersion factors for SG PORVs are weighted by their portion of the total mass release to determine mass release weighted average atmospheric dispersion factors that are used to model the steam releases.

<u>Acceptance Criteria</u>

According to Regulatory Guide 1.183, the EAB and LPZ dose acceptance criteria for a control rod ejection accident is 6.3 rem TEDE, which is 25% of the 10 CFR 50.67 limit of 25 rem TEDE. 01517422

The control room dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67.

<u>Dose Results</u>

Radiological doses resulting from a design basis CREA for a control room operator and a person located at EAB or LPZ are to be less than the regulatory dose limits as given below.

CREA Dose Results

Location	Acceptance Criteria (rem)	TEDE (rem)
Exclusion Area Boundary	6.3	0.67
Low Population Zone	6.3	0.39
Control Room	5.0	3.91

Conclusions

Even on the most pessimistic basis, the analyses indicate that the fuel and clad limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the primary coolant system. The amount of fission products released as a result of the assumed failure of fuel rods entering into DNB will not exceed the requirements of 10CFR50.67.

14.6 LARGE BREAK LOCA ANALYSIS

Note: The Large Break LOCA analyses described within this section bound cores containing either 422 Vantage + or 400 Vantage + fuel or a core containing a mixture of these two fuel types. The Large Break LOCA analyses described within this section is based on the combined Unit 1 and Unit 2 analysis with Replacement Steam Generators (Reference 83). 01454730

14.6.1 General

A loss-of-coolant accident (LOCA) may result from a rupture of the Reactor Coolant System (RCS) or of any line connected to that system up to the first closed valve. Ruptures of a very small cross section will cause expulsion of the coolant at a rate which would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. Breaks with a total cross-sectional area less than 1.0 ft^2, are discussed in Section 14.7. A small quantity of the coolant containing fission products normally present in the coolant would be released to the containment.

Should a major break occur, depressurization of the Reactor Coolant System results in a pressure decrease in the pressurizer. Reactor trip signal occurs and Safety injection signal occurs when the respective pressurizer low pressure trip setpoint is reached (including allowances for uncertainties, etc.). The large break LOCA analysis does not model control rod insertion and thus does not specifically model a reactor trip setpoint. The injection of the borated water limits the consequences of the accident in two ways:

1. Borated water injection complements void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat.
2. Injection of borated water provides heat transfer from the core and prevents excessive clad temperatures.

14.6.2 Acceptance Criteria

A major pipe break (large break), as considered in this section, is defined as a rupture of the reactor coolant pressure boundary with a total cross-sectional area greater than 1.0 ft^2. This is considered a Condition IV event, a limiting fault.

It must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met (Reference 77).

The Acceptance Criteria for the loss-of-coolant accident is described in 10 CFR 50.46 as follows:

a. The calculated peak fuel element cladding temperature is below the requirement of 2200°F.
b. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limits of 17% are not exceeded during or after quenching.

c. The amount of hydrogen generated by fuel element cladding that reacts chemically with water or steam does not exceed an amount corresponding to interaction of 1% of the total amount of Zircaloy in the reactor.

d. The core remains amenable to cooling during and after the break.

e. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

These criteria were established to provide significant margin in ECCS performance following a LOCA.

14.6.3 Method of Analysis

The analysis was performed using the Westinghouse Realistic Large Break LOCA Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM) (Reference 78). This methodology uses the Westinghouse WCOBRA/TRAC code as described in Reference 79 and as updated for Upper Plenum Injection (UPI) plants as described in Reference 80. The methodology using ASTRUM is nearly identical to the realistic methods described in References 79 and 80 except for a revised uncertainty treatment called ASTRUM.

The Prairie Island Units 1 and 2 PCT-limiting transients are cold leg split breaks which analyze conditions that fall within those listed in Table 14.6-1. Traditionally, cold leg breaks have been limiting for large break LOCA. This location is the one where flow stagnation in the core appears most likely to occur. Scoping studies with WCOBRA/TRAC have confirmed that the cold leg remains the limiting break location (Reference 79).

The realistic LOCA methodology described in Reference 78 uses the following computer codes:

- WCOBRA/TRAC for modeling the entire transient including system hydraulics and cladding temperature analysis.

- PAD (Reference 81) for generating the fuel parameters used in WCOBRA/TRAC.

- COCO (Reference 82) for confirming the containment backpressure used in the WCOBRA/TRAC model is conservatively low. The containment backpressure is conservatively minimized in LOCA Peak Clad Temperature Analysis, consistent with the methodology described in References 78, 79, and 80.

- HOTSPOT is used to apply localized uncertainties for the maximum local oxidation calculations.

The WCOBRA/TRAC analysis is performed using Prairie Island specific vessel and loop models. Prior to execution of the statistical uncertainty analysis, limiting conditions for offsite power availability, steam generator tube plugging, peripheral fuel assembly relative power are established. The identified limiting conditions are then used to perform a statistical based analysis using 124 WCOBRA/TRAC runs with random selection of 34 initial conditions, component configuration parameters and analytical parameters. In particular, break sizes from 1 ft^2 through a double ended guillotine are considered. Results from 124 calculations are tallied by ranking peak cladding temperature (PCT) from higher to lower. A similar procedure is used for local maximum oxidation (LMO) and core wide oxidation (CWO). The highest rank of PCT, LMO and CWO will bound 95-percent of their respective populations at a 95-percent confidence level. A summary of the analysis inputs is presented in Table 14.6.1. See Reference 83 for a more comprehensive list of input assumptions and a discussion of the detailed analysis techniques. 01454730

14.6.4 Description of a Nominal Large Break LOCA Transient

The large break LOCA transient can be divided into convenient time periods in which specific phenomena occur, such as various hot assembly heatup and cooldown transients. For a typical large break, the blowdown period can be divided into the Critical Heat Flux (CHF) phase, the upward core flow phase, and the downward core flow phase. These are followed by the refill, reflood, and long-term cooling periods. Specific important transient phenomena and heat transfer regimes are discussed below, with the transient results shown in Figures 14.6-1 to 14.6-12. (The PCT-limiting cases were chosen to show a conservative representation of the response to a large break LOCA.) The nuclear fuel rods were initialized with internal gas properties, radial power profiles, and fuel average temperatures from the Westinghouse Nuclear Fuel Core Technologies PAD code (Reference 81).

Critical Heat Flux Phase

Shortly after the break is assumed to open, the vessel depressurizes rapidly and the core flow decreases as subcooled liquid flows out of the vessel into the broken cold leg. The fuel rods go through departure from nucleate boiling (DNB) and the cladding rapidly 01454730 heats up (Figure 14.6-1), while the core power shuts down due to voiding in the core. Control rod insertion is not modeled. The hot water in the core and upper plenum flashes to steam. The water in the upper head flashes and is forced down through the guide tubes and the upper support plate holes. The break flow becomes saturated and is substantially reduced (Figure 14.6-2).

Upward core flow phase

As the reactor coolant system continues to depressurize, the colder water in the downcomer and lower plenum flashes and the mixture swells. This phase may be enhanced if voiding in the pumps is not significant or if the break discharge rate is low due to saturated fluid conditions at the break. If voiding in the pump is high or the break flow is large, the cooling effect due to upward flow may not be significant. However, there is sufficient upflow cooling to begin reducing the heat up in the fuel rods. As the lower plenum fluid depletes, upflow through the core ends (Figure 14.6-3).

Downward Core Flow Phase

The break flow begins to dominate and pulls flow down through the core. Figure 14.6-3, shows the total core flow at the bottom of the core. The blowdown peak clad temperature (PCT) occurs as the downflow increases in intensity and continues to decrease while downflow is sustained. At approximately 10-15 seconds, the pressure in the cold leg falls to the point where the accumulator begins injecting into the cold leg (Figure 14.6-4). Because the break flow is still high, much of the accumulator Emergency Core Cooling System (ECCS) water entering the downcomer is bypassed out the break. As the system pressure continues to decrease, the break flow, and consequently the core flow, is reduced. The break flow further reduces and the accumulator water begins to fill the downcomer and lower plenum. The core flow is nearly stagnant during this period and the hot assembly experiences a near adiabatic heat up.

Refill Phase

The accumulator blowdown fluid that is not bypassed fills the downcomer and lower plenum. The HHSI pump begins to inject (Figure 14.6-5) into the cold leg. Since the break flow has significantly reduced by this time, much of the ECCS entering the downcomer via the cold leg is retained in the downcomer and refills the lower plenum. The LHSI pump begins injecting (Figure 14.6-6) cold ECCS water into the upper plenum. The water enters the vessel at the hot-leg nozzle centerline elevation and falls down to the upper core plate through the outer global channels. The liquid drains down through the low-power region via the open hole channel of the counter-current flow limiting (CCFL) region. The hot assembly near-adiabatic heat-up is significantly reduced once the lower plenum fills with ECCS water (Figures 14.6-1 and 14.6-8).

Reflood

During the early reflood phase, the accumulators begin to empty and begin injecting nitrogen into the cold leg (Figure 14.6-4). The insurge in the downcomer forces the downcomer liquid into the lower plenum and core regions (Figures 14.6-7 through 14.6-9). During this time, core cooling is increased and the hot assembly clad temperature decreases slightly.

Steam generation from the water that enters the core forces some fluid back into the lower plenum and downcomer and ultimately out of the break. Meanwhile, the LHSI liquid flows down through the low power region and then across the core into the average assemblies near the bottom of the core. This water quenches the bottom of the core, which produces vapor that flows up through the average and hot assemblies, providing bottom-up cooling. The reflood PCT occurs at approximately 50 to 60 seconds.

By about 60 seconds, a quench front is established that progresses up the core moving the PCT elevation higher into the core until the rods quench at about 90 seconds (Figures 14.6-1 and 14.6-12). The system pressure is constant near atmospheric pressure by this time (Figure 14.6-11), and the vessel liquid mass shows a trend of increasing inventory with time by 300 seconds, which indicates that the increase in inventory due to the pumped safety injection is more than offsetting the loss of inventory through the break (Figure 14.6-10).

14.6.5 Deleted

14.6.6 Results

Confirmatory Sensitivity Studies

A number of sensitivity calculations were carried out to investigate the effect of the key LOCA parameters (Table 14.6-1), and to determine the limiting plant configuration for the uncertainty evaluation. In the sensitivity studies performed, LOCA parameters were varied one at a time. For each sensitivity study, a comparison between the base case and the sensitivity case transient results was made.

The results of these analyses lead to the following conclusions:

1. Modeling maximum steam generator tube plugging (10%) results in a higher PCT than minimum steam generator tube plugging (0%).

2. Modeling no loss-of offsite power (no-LOOP) results in a higher PCT than loss-of-offsite power (LOOP).

3. Modeling the maximum power fraction (P_{LOW}=0.7) in the low power/periphery channel of the core results in a higher PCT than minimum power fraction (P_{LOW}=0.2).

These assumptions were used in the uncertainty analysis.

Uncertainty Evaluation and Results

The ASTRUM methodology requires the execution of 124 WCOBRA/TRAC transients to determine a bounding estimate of the 95th percentile of the Peak Clad Temperature (PCT), Local Maximum Oxidation (LMO), and Core Wide Oxidation (CWO) with 95% confidence level. The results for Prairie Island Units 1 and 2 are given in Table 14.6-6, which shows the limiting peak clad temperatures (1,992°F), the limiting local maximum oxidation (0.62%), and the limiting core-wide oxidation (0.014%). Table 14.6-7 contains a sequence of events for the limiting PCT cases for Units 1 and 2. Figure 14.6-15 is a scatter plot for Units 1 and 2, which shows the effect of the effective break area on the analysis PCT.

14.6.7 Deleted

14.7 SMALL BREAK LOCA ANALYSIS

A small break LOCA (SBLOCA) analysis for Unit 1 has been completed at a core power of 1683 MWt using a full core of 422 Vantage+ fuel. It has been subsequently judged that the Unit 1 analysis can be applied to Unit 2 once the AREVA model 56/19 RSGs are installed; therefore, a single plant model can be applied to represent both units. The analysis for the 422 Vantage+ design bounds mixed cores utilizing the 422 Vantage+ and 400 Vantage+ fuel designs. The acceptance criteria for the SBLOCA analysis are Gpresented in Subsection 14.7.1, a general description of an SBLOCA transient is given in Subsection 14.7.2, the evaluation model used in the analysis is described in Subsection 14.7.3, the input parameters and initial conditions used in the analysis are discussed in Subsection 14.7.4, and the results of the analysis are summarized in Subsection 14.7.5. 01386642

14.7.1 Acceptance Criteria 01386642

A minor pipe break (small break), as considered in this section, is defined as a rupture of the reactor coolant system (RCS) pressure boundary with a total cross-sectional area less than 1.0 ft^2 in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered a Condition III event, an infrequent fault.

The Acceptance Criteria for the loss-of-coolant accident (LOCA) are described in 10 CFR 50.46 and are summarized as follows:

a. The calculated peak fuel element cladding temperature shall not exceed 2200°F.

b. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limits of 17% are not exceeded during or after quenching.

c. The amount of hydrogen generated by fuel element cladding that reacts chemically with water or steam does not exceed an amount corresponding to interaction of 1% of the total amount of Zircaloy in the reactor.

d. The core remains amenable to cooling during and after the break.

e. The core temperature is maintained at an acceptably low value and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

These criteria were established to provide significant margin in emergency core cooling system (ECCS) performance following a LOCA.

14.7.2 Description of Small Break LOCA Transient

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Ruptures of small cross-section will cause loss of coolant at a rate which can be accommodated by the charging pumps. These pumps would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. The coolant which would be released to the containment contains the fission products existing at equilibrium. In this instance, the break is considered as a leak.

The maximum size break for which the charging system can maintain the pressurizer level is obtained by comparing the calculated leak flow vs. the charging pump flow at normal Reactor Coolant System pressure. These calculations indicate that the charging system is more than capable of making up for a rupture of a 3/8″ tubing or failure of a 3/8″ compression fitting and maintaining pressurizer water level and system pressure.

Rupture of cross-sections up to about the equivalent of a 3/4″ connecting pipe will cause expulsion of coolant at a rate which can be accommodated by two of the three charging pumps well before the core is uncovered. However, in this instance the Reactor Coolant System would depressurize below the Reactor Trip and SI setpoints, and the SI Pumps would mitigate the event. Furthermore, for leaks smaller than the bore through 3/8″ tubing or compression fittings, if the charging pumps are not available, the plant response will be similar to the response for breaks larger than 3/8″; i.e., the SI System will mitigate the event.

Should a small break LOCA occur, the loss of RCS inventory causes fluid to flow into the loops from the pressurizer, resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the low pressurizer pressure trip setpoint is reached. During the early part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following reactor trip. Therefore, upward flow through the core is maintained. The Safety Injection System is actuated when the appropriate setpoint is reached. The consequences of the accident are limited in two ways:

1. Reactor trip and borated water injection complement void formation in the core and cause a rapid reduction of the nuclear power to a residual level corresponding to the delayed fission and fission product decay.

2. Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

Before the break occurs, the plant is in an equilibrium condition; i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from decay, hot internals, and the vessel continues to be transferred to the RCS. The heat transfer between the RCS and the secondary system may be in either direction, depending on the relative temperatures. In the case of continued heat addition to the secondary, system pressure increases and steam dump may occur. Makeup to the secondary side is automatically provided by the auxiliary feedwater pumps. The safety injection signal stops normal feedwater flow by closing the main feedwater line isolation valves and initiates auxiliary feedwater flow by starting auxiliary feedwater pumps. The secondary flow aids in the reduction of RCS pressures.

When the RCS depressurizes to the accumulator cover gas pressure, the cold leg accumulators begin to inject water into the reactor coolant loops. Due to the loss of offsite power assumption, the reactor coolant pumps are assumed to be tripped at the time of reactor trip during the accident and the effects of pump coastdown are included in the analyses.

14.7.3 Small Break LOCA Evaluation Model

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The NOTRUMP computer code is used in the analysis of loss-of-coolant accidents due to small breaks in the reactor coolant system. The NOTRUMP computer code is a one-dimensional general network code consisting of a number of advanced features. Among these features are the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. Also, safety injection into the broken loop is modeled using the COSI condensation model (Reference 73). The NOTRUMP small break LOCA emergency core cooling system (ECCS) evaluation model was developed to determine the RCS response to design basis small break LOCAs and to address the NRC concerns expressed in NUREG-0611, “Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants.”

In NOTRUMP, the RCS is nodalized into volumes interconnected by flowpaths. The intact loop and broken loop are modeled explicitly. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied throughout the system. A detailed description of NOTRUMP is given in WCAP-10054-P-A, WCAP-10054-P-A Addendum 2 Revision 1, and WCAP 10079-P-A. (References 72, 73 and 71 respectively)

The use of NOTRUMP in the analysis involves, among other things, the representation of the reactor core as heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multinode capability of the program enables an explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the loop seal during a loss-of-coolant transient.

Cladding thermal analyses are performed with the small break LOCA version of the LOCTA-IV code (SBLOCTA; Reference 90) which models annular pellets explicitly (Reference 74) and uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the NOTRUMP hydraulic calculations, as input. 01386642

A schematic representation of the computer code interfaces is given in Figure 14.7-1.

This model was developed to resolve TMI Action Item II.K.3.30. NRC acceptance of this model for Prairie Island was documented in an NRC staff letter dated June 6, 1985.

14.7.4 Small Break Input Parameters and Initial Conditions 01386642

Table 14.7-1 lists important input parameters and initial conditions used in the small break LOCA analyses. See Reference 91 for a more comprehensive discussion regarding the inputs used in the small break LOCA analysis. 01386642

The hot rod axial power shape utilized to perform the small break analysis is shown in Figure 14.7-2. This power shape represents a distribution of power versus core height and was chosen because it maximizes the integral power at the top half of the core where peak cladding temperature (PCT) occurs and minimizes power in the bottom half of the core, which reduces swell due to void formation thus providing a deeper core uncovery. Core decay heat is based on ANS1971 Standard as described in 10CFR Part 50, Appendix K. 01365252

Safety injection (SI) flow rates to the RCS as a function of the system pressure are used as part of the input. The SI Pump delivery is delayed 27 seconds to account for the time required for diesel startup and loading of the safety injection pumps onto the emergency buses following a loss of offsite power.

The ECCS system consists of gas pressurized accumulator tanks and pumped SI systems. The accumulators are modeled to inject into the cold legs when the RCS depressurizes to 699.7 psia. For this analysis, the SI delivery considers minimum ECCS availability from one high head SI (HHSI) and one residual heat removal (RHR; or low head SI (LHSI)) pump. Figure 14.7-3a represents injection flow from one degraded HHSI pump spilling to RCS pressure (for cold leg breaks smaller than HHSI injection line diameter), while Figure 14.7-3b represents injection flow from one degraded HHSI pump spilling to containment pressure (0 psig) for cold leg break sizes greater than the diameter of HHSI line.

The RCS depressurizes below the RHR cut-in pressure for the larger break sizes (6 inches and greater); therefore, the injection flow from a single RHR pump (Figure 14.7-3c) is modeled to inject directly into the upper plenum of the vessel.

For the accumulator line break, the HHSI flows are assumed to spill to RCS pressure (Figure 14.7-3a) and RHR flows inject to the upper plenum (Figure 14.7-3c).

The hydraulic analyses are performed with the NOTRUMP code at 1683 MWt core power. The core thermal transient analyses are performed with the small break LOCA version of the LOCTA-IV code at 1683 MWt core power.

14.7.5 Small Break Results 01386642

A range of break sizes (1.5-inch, 2-inch, 3-inch, 4-inch, 6-inch, and 8-inch cold leg breaks) including a 10.126-inch accumulator line break were analyzed to establish the limiting break size. The results of this analysis are summarized in Tables 14.7-2 and 14.7-3. 01386642

The 3-inch break is the limiting break size and the following transient parameters are presented for this case (Figures 14.7-4 through 14.7-13).

- Reactor Coolant System Pressure
- Core Mixture Level
- Total Reactor Coolant System Mass
- Top Core Exit Vapor Temperature
- Vapor Mass Flow Rate Out Top of Core
- Total Break Flow and Safety Injection Flow
- Cladding Surface Heat Transfer Coefficient at PCT Elevation
- Fluid Temperature at PCT Elevation
- Cladding Temperature at PCT Elevation 01386642
- Local ZrO_2 Thickness at Maximum Local ZrO_2 Elevation

Figures 14.7-14 through 14.7-35 present the non-limiting break results. Fuel rod heatup calculations were not performed for the 1.5-inch, 6-inch, 8-inch, and 10.126-inch breaks because core uncovery for these breaks was either minimal or did not occur. The non-limiting transient parameters are as follows: 01386642

- Reactor Coolant System Pressure
- Core Mixture Level
- Top Core Exit Vapor Temperature
- Cladding Temperature at PCT Elevation (2-inch and 4-inch only)
- Local ZrO_2 Thickness at Maximum Local ZrO_2 Elevation (2-inch and 4-inch) 01386642

The maximum calculated peak cladding temperature for the small breaks analyzed is 959°F (Reference 92 and 111). 01386642 These results are well below all Acceptance Criteria limits of 10 CFR 50.46. Small break LOCA PCTs could be affected by periodic error discovery and/or corrections. These are accounted for under 10 CFR part 50.46 Reportability requirements. Even accounting for additional penalties that might address errors in or corrections to the analysis, the peak clad temperature is well below the Acceptance Criteria limit of 10 CFR 50.46, and no case is limiting when compared to the results presented for the large break LOCA. 01386642

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14.8 ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)

As defined in 10CFR50.62, an ATWS is an expected operational transient (such as loss of feedwater, loss of load, or loss of offsite power) which is accompanied by a failure of the reactor protection system (RPS) to shutdown the reactor. Initial NRC Staff guidance on ATWS was provided in WASH-1270, “Technical Report on Anticipated Transients Without Scram for Water Cooled Power Reactors.” Westinghouse responded to WASH-1270 with a series of generic ATWS studies which were summarized in WCAP-8330 (1974). Additional studies, which conformed with guidance provided in NUREG-0460 (1978), were also performed by Westinghouse at the request of the NRC. The results of these studies showed that the analyzed ATWS events would have acceptable consequences, provided the turbine is tripped and auxiliary feedwater flow is initiated in a timely manner.

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The final ATWS rule, 10 CFR 50.62, “Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled-Nuclear Power Plants,” was published by the NRC on July 26, 1984. For Westinghouse plants, this rule required an ATWS Mitigating System Actuation Circuitry (AMSAC) system to initiate a turbine trip and actuate auxiliary feedwater flow independent of the RPS be installed. This requirement is based on the analyses performed by Westinghouse in response to WASH-1270 and NUREG-0460.

In December 1988, the Westinghouse Owners Group (WOG) produced WCAP-11993, “Assessment of Compliance with ATWS Rule Bases for Westinghouse PWRs.”

In early 1996 PINGP reviewed the design basis for the Auxiliary Feedwater system and determined that the existing pump low discharge pressure setpoints did not adequately protect the auxiliary feedwater pumps from runout conditions. New setpoints were calculated in an attempt to correct this situation. However, during a review of the design basis for the new setpoints, it was recognized that the AFWP runout protection impacts the operability of the auxiliary feedwater pumps during an ATWS event.

NSP determined that the preferred method for correction of this issue was the installation of a diverse scram system similar to that described in 10CFR50.62 for CE and B&W plants. The ATWS Mitigating System Actuation Circuitry/Diverse Scam System (AMSAC/DSS) is described in Section 7.11.

Plant parameters are shown in Figures 14.8-1 through 14.8-39.

14.8.1 Analytical Basis

To provide the actuation signal to this diverse scram system, modifications to the AMSAC logic were developed. To support these changes, new analytical analyses were performed by NSP Nuclear Analysis & Design (NAD) based on NSP's approved Reload Safety Evaluation (RSE) methods and more conservative acceptance criteria than those applied in the Westinghouse generic ATWS Analysis. The results of these analyses required a change to the process variable inputs used to generate an AMSAC/DSS actuation signal. It was determined that the new process variable inputs would be steam generator wide range levels and reactor coolant pump breaker position. The steam generator levels will provide a trip signal when 2 out of 2 wide range signals (per steam generator) indicate a level of less than 40%. In addition, a 52b contact located in the RCP switchgear, used to indicate breaker position, will be used to indicate loss of reactor coolant system flow.

PINGP submitted a license amendment to make the necessary changes to the AMSAC system along with the installation of a diverse scram system to resolve the issues discovered with the AFW Pumps during ATWS conditions (License Amendment 138/129). The AMSAC/DSS system is designed to meet the requirements of 10CFR50.62 including diversity and independence from the Reactor Trip System.

In support of the transition to Westinghouse safety analysis methods, separate AMSAC/DSS analyses were performed, which are the basis for the information presented in the subsections that follow.

14.8.2 Computer Codes Used for ATWS Analysis

AMSAC/DSS analyses were performed using the RETRAN and VIPRE computer codes, which are described in Section 14.3.

14.8.3 Transient Analyses Results

All USAR condition 2 transient events were evaluated with consideration towards explicitly analyzing each under ATWS conditions. For many of the condition 2 events, explicit analyses were performed. Events were not explicitly analyzed for ATWS conditions if the transient either (1) does not require a reactor trip to mitigate the consequences of the event in the analysis, or (2) results in consequences bounded by either another analyzed transient or an ATWS event transient selected for analysis.

The disposition of Condition 2 transients is provided in Table 14.8-1.

14.8.3.1 AMSAC/DSS Trips and Acceptance Criteria

The DSS installed in Prairie Island provides a reactor trip on low steam generator wide range level (WRL) signal and on a RCP breaker position signal. The DSS will provide a trip signal when two-out-of-two wide range signals (per steam generator) indicate a level of less than 40 percent in either steam generator. For the AMSAC/DSS analyses, a trip setpoint of 35 percent WRL was assumed. The DSS will also provide a trip signal after receiving an RCP breaker open position signal.

The following key assumptions regarding the DSS functionality were made for these analyses:

1. The AMSAC/DSS steam generator wide range level trip safety analysis setpoint is 35 percent WRL.

2. The AMSAC/DSS output signal is generated within 5 seconds.

3. The AMSAC/DSS AFW start delay is 65 seconds from the time that the AMSAC/DSS setpoint is reached, or 60 seconds from the time that the AMSAC/DSS output signal is generated.

4. The AMSAC/DSS turbine trip delay is 10 seconds from the time that the AMSAC/DSS setpoint is reached, or 5 seconds from the time that the AMSAC/DSS output signal is generated.

The AMSAC/DSS analyses were based on the following acceptance criteria, which are consistent with those originally used by NSP in the AMSAC/DSS analyses documented in Reference 48:

1. The minimum departure from nucleate boiling ratio (DNBR) must be greater than the DNBR correlation limit for the correlation being used.

2. The analytical limit for the RCS maximum pressure is 3200 psig throughout any AMSAC/DSS event.

14.8.3.2 AMSAC/DSS - Partial Loss of Reactor Coolant Flow, One RCP Trip

Accident Description

The partial loss-of-coolant-flow accident can result from a mechanical or electrical failure in an RCP, or from a fault in the power supply to the RCP. If the reactor is at power at the time of the accident, the immediate effect of the 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 is not tripped promptly.

The normal protection against a loss-of-coolant-flow accident is provided by the low primary coolant flow reactor trip signal, which is actuated in any reactor coolant loop by two-out-of-three low flow signals. For this event, the low coolant flow reactor trip is not modeled, and the AMSAC/DSS is assumed to actuate a diverse reactor trip after receiving a RCP breaker open position signal.

Method of Analysis

The loss of an RCP with both loops in operation event is analyzed to show that: (1) the integrity of the core is maintained as the DNBR remains above the correlation limit value of 1.17, and (2) the peak RCS pressure remains below 3200 psig.

The loss of an RCP event is analyzed with two computer codes. First, the RETRAN computer code is used to calculate the loop and core flow during the transient, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE computer code is then used to calculate the hot-channel heat flux transient and DNBR, based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN. The DNBR results presented represent the minimum of the typical or thimble cell.

The following key analysis assumptions are made:

1. The AMSAC/DSS system is functional and activated by an opening of the RCP breaker at 0 seconds. A 5-second delay is assumed to actuate the DSS signal, plus another 3 seconds for rod motion to begin. The total delay from transient initiation to the start of rod motion is therefore 8 seconds.

2. Initial reactor power, pressurizer pressure, and RCS temperature are assumed to be at their nominal values. Minimum measured flow is also assumed. Initial core power is assumed to be 1683 MWt.

3. A conservatively large absolute value of the Doppler-only power coefficient is used, along with the most-positive ITC limit for full-power operation (0 pcm/°F). The assumptions maximize the core power during the initial part of the transient when the minimum DNBR is reached.

4. A conservatively low trip reactivity value (4.0-percent/Δp) is used to minimize the effect of rod insertion following reactor trip and maximize the heat flux statepoint used in the DNBR evaluation for this event. This value is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position.

5. A conservative trip reactivity worth versus rod position was modeled in addition to a conservative rod drop time (2.4 seconds to dashpot).

6. The Westinghouse original steam generators were modeled. However, the analysis applies to the replacement steam generators since this event is not sensitive to the secondary-side modeling. 01386642

7. A maximum uniform steam generator tube plugging level of 25 percent was assumed in the RETRAN analysis. As noted above, although this tube plugging level was specific to the original Westinghouse steam generators, the analysis remains applicable to the replacement steam generators. 01386642

Results

Figures 14.8-1 through 14.8-4 illustrate the transient response for the loss of an RCP with both loops in operation. The minimum DNBR calculated is 1.39 and the maximum RCS pressure increased to 2391 psia. These results are still well within the acceptance criteria for this event.

Conclusions

The analysis performed has demonstrated that for the partial loss-of-coolant flow event, the DNBR does not decrease below the AMSAC/DSS limit value at any time during the transient. Additionally, the peak RCS pressure remains below the AMSAC/DSS limit of 3200 psig. Therefore, the AMSAC/DSS adequately protects the reactor.

14.8.3.3 AMSAC/DSS - Loss of Normal Feedwater Flow

Accident Description

A loss of normal feedwater (from a pipe break, pump failure, or valve malfunction) results in a reduction of the ability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, core damage could possibly occur from a sudden loss of heat sink. If an alternate supply of feedwater were not supplied to the steam generators, residual heat following reactor trip and reactor coolant pump (RCP) heat would cause the primary system water to expand to the point where water relief from the pressurizer would occur. A significant loss of water from the RCS could conceivably lead to core damage. 01365252

The following features provide normal protection against a loss of normal feedwater:

1. Reactor trip on low-low water level in either steam generator

2. Automatic start of one motor-driven auxiliary feedwater (AFW) pump and one turbine-driven AFW pump (via opening of the steam admission control valve) per unit on low-low water level in either steam generator

For the AMSAC/DSS LONF event, it is assumed that these protection functions are unavailable. The AMSAC/DSS is assumed to start the AFW pumps, actuate a turbine trip, and actuate a diverse reactor trip after receiving a steam generator wide range level ≤ 35 percent signal (safety analysis value). 01386642

Method of Analysis

The loss of normal feedwater transient is analyzed using the RETRAN computer code, which is described in Section 14.3. The major assumptions are summarized below: 01386642

1. The plant is initially operating at 100 percent of the nominal NSSS power of 1690 MWt. The RCP heat is a nominal constant value of 7 MWt. The RCPs run throughout the transient.

2. The initial reactor coolant vessel average temperature is assumed to be 560°F, the nominal full-power value.

3. The initial pressurizer pressure is assumed to be 2250 psia, the nominal value.

4. The initial pressurizer water level is assumed to be 33 percent level span, the programmed full-power value.

5. The initial steam generator water level is assumed to be 44 percent of narrow range span (NRS) (the programmed full-power value).

6. The transient is simulated by terminating main feedwater flow at 20 seconds.

7. A reactor trip signal is generated 5 seconds after the steam generator wide range level reaches 35 percent WRL. The rods begin to drop after an additional delay of 3 seconds. No credit is taken for any other reactor trip functions. Turbine trip occurs 5 seconds after the reactor trip signal (or 10 seconds after the steam generator wide range level reaches 35 percent WRL).

8. Both the turbine- and motor-driven AFW pumps are operable. Sixty-five seconds after the AMSAC/DSS wide range steam generator water level setpoint is reached, a constant AFW flow of 160 gpm is initiated from each AFW pump, with flow split equally between the two steam generators. The AFW enthalpy is assumed to be 44.7 Btu/lbm (73.6°F at 1100 psia).

9. Secondary system steam relief is achieved through the main steam safety valves (MSSVs). The MSSV opening pressures are the nominal settings plus 3 percent tolerance.

10. The pressurizer power-operated relief valves (PORVs), pressurizer heaters, and pressurizer sprays are assumed to operate normally.

11. A conservative core residual heat generation is assumed based on ANS 5.1 1979 decay heat model with no additional uncertainty.

Results

Figures 14.8-5 through 14.8-12 show the significant plant responses following a loss of normal feedwater. The capacity of the AFW pumps together with the AMSAC/DSS trip function are sufficient to dissipate core residual heat, stored energy, and RCP heat demonstrating the adequacy of the AFW system to provide long-term core cooling. The maximum RCS pressure for this event, 2380 psia, is well below the peak pressure limit of 3200 psig. The minimum calculated DNBR is 1.86, which is above the correlation limit of 1.17.

Conclusions

The results of the loss of normal feedwater analysis show that the AMSAC/DSS criteria are met. The AFW capacity and AMSAC/DSS reactor trip are sufficient to dissipate core residual heat, stored energy, and RCP heat such that reactor coolant water is not relieved through the pressurizer relief or safety valves, and the AMSAC/DSS maximum RCS pressure and minimum DNBR criteria are met. 01365252

14.8.3.4 AMSAC/DSS - Loss of AC Power

Accident Description

A complete loss of non-emergency AC power results in the loss of all power to the plant auxiliaries, such as the RCPs, main feedwater and condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of the onsite AC distribution system.

Upon the loss of power to the RCPs, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops. Following the RCP coastdown caused by the loss of AC power, the natural circulation capability of the RCS removes residual and decay heat from the core, aided by the AFW in the secondary system.

The AMSAC/DSS is assumed to start the AFW pumps, actuate a turbine trip, and actuate a diverse reactor trip after receiving a steam generator wide range level $\leq$ 35 percent signal (safety analysis value).

Method of Analysis

The loss of all AC power to the station auxiliaries transient is analyzed using the RETRAN computer code, which is described in Section 14.3.

The analysis does not assume that power is lost as the initiating event. Rather, the analysis conservatively models a loss of normal feedwater with a subsequent loss of offsite power following the reactor trip on the wide range steam generator water level AMSAC/DSS setpoint.

Major assumptions made in the loss of all auxiliary AC power analysis are the same as those made in the AMSAC/DSS loss of normal feedwater analysis (Section 14.8.3.3), with the following exceptions:

1. The RCPs are assumed to lose power and begin coasting down 2 seconds following the reactor trip on the wide range steam generator water level AMSAC/DSS trip function. Following the loss of power to the RCPs, coolant flow necessary for core cooling and removal of residual heat is maintained by natural circulation flow in the coolant loops. Heat addition from the RCPs to the primary coolant ceases.

2. Pressurizer sprays are lost when forced reactor coolant flow ceases as a result of RCP coastdown.

Results

Figures 14.8-13 through 14.8-20 show the significant plant responses following a loss of all AC to the station auxiliaries. The capacity of the AFW pumps together with the AMSAC/DSS trip function are sufficient to dissipate core residual heat, stored energy, and RCP heat (up to the point of RCP coastdown) demonstrating the adequacy of the AFW system to provide long-term core cooling. The maximum RCS pressure for this event, 2367 psia, is well below the peak pressure limit of 3200 psig. The minimum calculated DNBR is 1.86, which is above the correlation limit of 1.17.

Conclusions

The results of the loss of all AC power to the station auxiliaries AMSAC/DSS analysis show that the AMSAC/DSS acceptance criteria are satisfied. The AFW capacity and AMSAC/DSS reactor trip are sufficient to dissipate core residual heat and stored energy such that reactor coolant water is not relieved through the pressurizer relief or safety valves, and the AMSAC/DSS maximum RCS pressure and minimum DNBR criteria are met.

14.8.3.5 AMSAC/DSS - Loss of Load/Turbine Trip

Accident Description

The loss-of-external-electrical-load event is defined as a complete loss of steam load or a turbine trip from full power without a direct reactor trip. This anticipated transient is analyzed as a turbine trip from full power because it bounds both events - the loss of external electrical load and turbine trip. The turbine trip event is more severe than the total loss of external electrical load since it results in a more rapid reduction in steam flow.

If the reactor were not tripped during this event, the mismatch between heat production and heat removal would eventually boil the steam generators dry leading to consequences identical to those in the AMSAC loss of normal feedwater flow transient.

The AMSAC/DSS is assumed to start the AFW pumps and actuate a diverse reactor trip after receiving a steam generator wide range level $\leq$ 35 percent signal (safety analysis value).

Method of Analysis

This event is analyzed using the RETRAN computer code, which is described in Section 14.3.

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from full power with no credit taken for a direct reactor trip on turbine trip or for the normal reactor protection system functions. This assumption will delay reactor trip until conditions in the RCS cause a trip on the AMSAC/DSS steam generator wide range level setpoint. Therefore, the analysis assumes a worst case transient and demonstrates the adequacy of the pressure relieving devices and AMSAC/DSS setpoint assumed in the analysis for this event.

One case is performed to confirm that both the AMSAC/DSS DNBR and peak primary RCS pressure limits are met. The major assumptions for these cases are summarized as follows:

1. The AMSAC/DSS is assumed to initiate a reactor trip signal when the steam generator wide range level reaches 35 percent WRL. A reactor trip signal is generated 5 seconds after this AMSAC/DSS setpoint is reached. The rods begin to drop after an additional 3-second delay.

2. Initial core power, reactor coolant temperature, and reactor coolant pressure are assumed to be at the nominal values consistent with steady-state full-power operation.

3. The loss of load event results in a primary system heatup and, therefore, is conservatively analyzed assuming minimum reactivity feedback consistent with BOC conditions. An MTC of -4.1 pcm/°F was assumed. This is a bounding MTC for 95 percent of a representative cycle.

4. It is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

5. No credit is taken for the operation of the steam dump system or steam generator power-operated relief valves (PORVs). The steam generator pressure rises to the safety valve setpoints, where steam release through the MSSV limits the secondary side steam pressure to the setpoint values. The MSSV was explicitly modeled in the loss-of-load licensing basis analysis assuming a zero percent tolerance with a 5 psi pop to full open. The MSSV model also assumed a 5 psi pressure drop from the main steam line entrance to the MSSV header in determining the opening setpoints.

6. Automatic pressurizer pressure control is assumed. Therefore, full credit is taken for the effect of the pressurizer spray and PORVs in reducing or limiting the primary coolant pressure. Safety valves are also available and are modeled assuming a zero percent setpoint tolerance.

7. Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip.

Results

The transient responses for a total loss of load from full-power operation are shown in Figures 14.8-21 through 14.8-27. The reactor is tripped on the AMSAC/DSS steam generator wide range level trip signal. The PORVs and PSVs are actuated and maintain the primary RCS pressure below 3200 psig. The peak RCS pressure calculated in the transient is 2446 psia.

The minimum calculated DNBR for this case is 1.80, which is well above the AMSAC/DSS limit of 1.17.

Conclusions

The results of the analysis show that the plant design is such that a total loss of external electrical load without a direct or immediate reactor trip does not result in a violation of the AMSAC/DSS criteria. Pressure relieving devices that have been incorporated into the plant design are adequate to limit the maximum pressures.

The integrity of the core is maintained by operation of the AMSAC/DSS; that is, the minimum DNBR is maintained above the correlation limit value of 1.17.

14.8.3.6 AMSAC/DSS - RCCA Bank Withdrawal at Power

Accident Description

The uncontrolled RCCA bank withdrawal at power event is defined as the inadvertent addition of reactivity to the core caused by the withdrawal of RCCA banks when the core is above the power defined by the P-10 setpoint. The reactivity insertion resulting from the bank (or banks) withdrawal will cause an increase in the core nuclear power and subsequent increase in the core heat flux. An RCCA bank withdrawal can occur with the reactor subcritical, at HZP, or at power. The AMSAC/DSS uncontrolled RCCA bank at power event is analyzed for Mode 1 (power operation).

The event is simulated by modeling a constant reactivity insertion rate starting at time zero and continuing until the rods are fully withdrawn, based on a withdrawal rate of 72 steps/min. The D-bank of control rods is assumed to be at the nominal full-power position of 218 steps.

For this AMSAC/DSS event, a total reactivity of 100 pcm is assumed to be inserted by the rod withdrawal. This value is intended to bound typical cycle-specific values for the rod worth of D-bank at 218 steps.

No reactor trip is modeled in this analysis.

Method of Analysis

The AMSAC/DSS RCCA bank withdrawal at-power transient is analyzed to ensure that the AMSAC/DSS maximum RCS pressure and minimum DNBR criteria are met.

This event is analyzed with the RETRAN computer program, which is described in Section 14.3.

The following analysis assumptions are made:

1. Initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal value and the minimum measured RCS flow is assumed.

2. A -2 pcm/°F MTC is assumed at full power. This value is more conservative than a best-estimate MTC. A conservatively small (in absolute magnitude) Doppler power coefficient (DPC) is used in the analysis. 01386642

3. The transient is initiated with D-bank at 218 steps. A withdrawal rate of 72 steps/min is modeled from 218 steps to the maximum all-rods-out position (228 steps). The total reactivity inserted is 100 pcm.

Results

Figures 14.8-28 through 14.8-33 show the transient response of nuclear power, core reactivity, pressurizer pressure, RCS pressure, RCS loop temperatures, and DNBR to a RCCA withdrawal incident starting from full power, with no reactor trip.

The peak RCS pressure calculated, 2314 psia, remains below the ATWS limit of 3200 psig. In addition, the minimum calculated DNBR of 1.28 remains above the AMSAC/DSS limit of 1.17.

Conclusions

The AMSAC/DSS criteria are met for the RCCA bank withdrawal at-power event.

14.8.3.7 AMSAC/DSS - Uncontrolled Boron Dilution

Accident Description

The uncontrolled boron dilution event is defined as the inadvertent addition of reactivity to the core caused by the addition of unborated water into the RCS. The reactivity insertion resulting from this dilution will cause an increase in the core nuclear power and subsequent increase in the core heat flux. The AMSAC/DSS uncontrolled boron dilution event is analyzed for Mode 1 (power operation).

The event is simulated by modeling a constant reactivity insertion rate starting at time zero and continuing until operator action terminates the boron dilution at 10 minutes. The AMSAC/DSS does not actuate for this event.

For this event, a total reactivity of 219 pcm is assumed to be inserted by the boron dilution. This value is based on a maximum dilution flow of 60.5 gpm from one charging pump, and a boron worth of -7.4 pcm/ppm. 01365252

No reactor trip is modeled in this analysis. The event is assumed to be terminated by operator action after 10 minutes.

Method of Analysis

The AMSAC/DSS uncontrolled boron dilution event is analyzed to ensure that the AMSAC/DSS maximum RCS pressure and minimum DNBR criteria are met. Since this event is a slow gradual increase in power, RCS overpressurization is unlikely.

This event is analyzed with the RETRAN computer program, which is described in Section 14.3.

The following analysis assumptions are made:

1. Initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values and the minimum measured RCS flow is assumed.

2. A -5.3 pcm/°F is assumed at full power. This value represents a 95 percent MTC (bounding MTC for 95 percent of a representative cycle). Best-estimate values for the Doppler power and temperature coefficients are used in the analysis.

3. A maximum dilution rate of 60.5 gpm from one charging pump is modeled. The fluid conditions are calculated based on a temperature of 40°F and a pressure of 14.7 psia.

4. A boron worth of -7.4 pcm/ppm is assumed.

Results

Figures 14.8-34 through 14.8-39 show the transient response of nuclear power, core reactivity, pressurizer pressure, RCS pressure, RCS loop temperatures, and DNBR to a uncontrolled boron dilution incident starting from full power, with no reactor trip. The peak RCS pressure, 2306 psia, remains below the AMSAC/DSS limit of 3200 psig. In addition, the minimum DNBR of 1.31 remains above the AMSAC/DSS limit of 1.17,

Conclusions

The AMSAC/DSS criteria are met for the uncontrolled boron dilution event.

Calculation Results and Margins to Functional Goals

The results of the AMSAC/DSS analysis demonstrate that (1) the limiting reactor coolant system pressure occurs during the ATWS Loss of External Electrical Load/Turbine Trip transient, and that (2) the limiting minimum departure from nucleate boiling ratio (MDNBR) for all events dependent on AMSAC/DSS actuation occurs during the ATWS Partial Loss of Reactor Coolant Flow transient. For all events, the RCS pressure remains below 3200 psig thus not reducing the safety margin associated with the integrity of the Reactor Coolant Pressure Boundary, and the MDNBR remains higher than 1.17 thus not reducing the safety margin associated with the integrity of the fuel cladding and ensuring that a coolable geometry is maintained. Table 14.8-9 demonstrates that the analytical acceptance criteria for all functional goals were met in each ATWS transient that was explicitly analyzed.

14.8.4 Plant Mitigating Systems

Following the discovery that the AFW Pump runout protection impacted the operability of the auxiliary feedwater pumps during an ATWS event, NSP decided to install a diverse scram system similar to that described in 10CFR50.62 for CE and B&W plants and developed modifications to the AMSAC logic to provide this actuation signal. This modified logic uses steam generator wide range levels and RCP breaker position as inputs and provides outputs to start the AFW pumps, trip the turbine and actuate the diverse scram. Analysis performed demonstrates that this design adequately mitigates the consequences of an ATWS event.

License amendment 138/129 authorized the changes to the AMSAC system along with the installation of a diverse scram system (Refs 48, 49, & 50). The AMSAC/DSS system is designed to meet the requirements of 10 CFR 50.62 including diversity and independence from the Reactor Trip System. The revised AMSAC/DSS system is described in section 7.11. 01365252

Given an ATWS event, if the assumed postulated common mode failure (CMF) is mechanical in nature, i.e., in the rod drives or vessel internals, then the mitigating functions of auxiliary feedwater and turbine trip would be provided by the existing NSSS protection system. If the CMF is electrical in nature (i.e., protection logic or reactor trip breakers) then an AMSAC/DSS system as described above is provided.

A review of many Westinghouse PWR's has shown that many of the features of AMSAC are available in the balance of plant (BOP) design. These BOP features can be used to mitigate the consequences of ATWS.

The AMSAC/DSS system utilizes balance of plant features as diverse backup for the Reactor Protection System (RPS). The function of AMSAC/DSS is to provide an alternate means to initiate auxiliary feedwater, trip the main turbine and trip the reactor given a low probability ATWS event. Therefore, all of the Class 1E requirements imposed upon the RPS do not apply to AMSAC/DSS. Although all criteria for Class 1E systems do not apply to AMSAC/DSS, the Class 1E criteria is met for the interface relays which start the auxiliary feedwater pumps.

14.8.5 Continued Compliance With ATWS Rule

The transient analyses performed in support of the AMSAC/DSS modification will be evaluated during the reload transient analysis each fuel cycle to ensure that the analysis continues to bound plant operation.

14.9 ENVIRONMENTAL CONSEQUENCES OF LOSS-OF-COOLANT ACCIDENT

14.9.1 Introduction

The NRC has established guidelines in 10CFR50.67 for radiation doses resulting from accidental releases of radioactivity from a reactor plant. This section demonstrates the capability of the Prairie Island Nuclear Generating Plant to stay within the dose criteria set forth in 10CFR50.67 following the design basis accident and releases consistent with NRC Regulatory Guide 1.183 assumptions.

The Prairie Island Nuclear Generating Plant containment system is described in detail in Section 5. One feature of particular importance to the environmental consequences of a loss-of-coolant accident is the presence of two barriers in series to fission product leakage: the Reactor Containment Vessel and the Shield Building.

There are three activity release paths following a LOCA: (a) Containment Leakage, (b) ESF System Leakage, and (c) Refueling Water Storage Tank back leakage. During power operations, other pathways between containment atmosphere and the environment (i.e., in-service containment purge, main containment purge, containment vacuum relief, etc.) are isolated during Modes 1 through 4.

Containment vessel leakage is collected within an annular volume between these structures before release. Most of the potential leakage through containment penetrations will either be collected in the Annulus or the Auxiliary Building Special Ventilation Zone. A small amount of leakage could potentially bypass both leakage collection zones and leak directly to atmosphere. The annulus is therefore effective as a means of holding leakage for decay and providing additional dilution prior to release. Release to the environment is through absolute and charcoal filters provided in the Shield Building Ventilation System.

For reference in the evaluation of environmental consequences, a schematic diagram of the analysis model is shown in Figure 14.9-1. Long term uncontrolled leakage of radioactivity to the external atmosphere prior to filtration or decay is prevented by fans which establish a slight negative pressure with respect to the atmosphere in the annulus shortly after the accident. The amount of long term filtered exhaust released to the environment is sufficient to maintain the negative annulus pressure and compensate for inleakage. In general, all exhaust from the Shield Building will have experienced several passes of filtration as a result of the recirculation feature.

A measurable negative pressure with respect to atmosphere will be drawn in the Auxiliary Building Special Ventilation Zone within 20 minutes after initiation.

14.9.2 Assumed Accident and Activity Released

The postulated cause of radioactivity release to the environment analyzed in this section is an extremely improbable 3 ft^2 pump suction break which has been found to be the most conservative postulated accident (see analysis in Appendix K). Note that the size of the break is not significant to this analysis. Following the assumptions of NRC Regulatory Guide 1.183 the Design Basis accident will release to the Reactor Containment Vessel a spectrum of radionuclides present in the core fission product inventory. The release occurs in two phases. The gap release phase begins at 30 seconds and continues for 30 minutes. The early in-vessel release phase begins at 30 minutes and continues for 1.8 hours. Because of the multiple redundancies in engineered safety features, such a release is considered incredible.

During mitigation of a loss of coolant accident, the auxiliary feedwater pump(s) are used to maintain sufficient secondary side pressure and water volume to inhibit the leakage of radioactive material from the ruptured primary side to the secondary side.

14.9.3 Containment Vessel Inventory and Leak Rate

The noble gases and halogens released from the fuel are assumed to mix homogeneously throughout the free air volume of the primary containment. Fission product cleanup following a LOCA is accomplished by the Containment Spray System during the injection mode. Containment spray does not operate in the recirculation mode. Fission product clean-up by the containment spray system is not credited in the post-LOCA dose analysis.

The containment is surrounded by an annulus building (Shield Building). In the event of a postulated LOCA, the Shield Building Ventilation System (SBVS) is designed to collect and filter the containment leakage that has entered the Shield Building, and to exhaust the flow to the environment via the Shield Building vent stack.

It is expected that most of the containment leakage will be collected in the Shield Building; however, some of the containment leakage may by-pass the Shield Building and enter the Auxiliary Building where it is collected and filtered by the Auxiliary Building Special Ventilation System (ABSVS) and released to the environment via the Shield Building vent stack.

A small amount of containment leakage has the potential to bypass both leakage collection systems and leak directly to the atmosphere.

Timing of the release phases, release fractions, and the chemical form of the releases are consistent with Regulatory Guide 1.183 (Reference 109).

Isotopic decay, containment leakage and spray removal are credited to deplete the inventory of noble gases and iodines airborne in containment. As described in Section 14.9.6.3, since the long term sump water pH is controlled to greater than 7, iodine re-evolution is not considered.

Activity Leakage from the Containment Vessel is assumed to leak to the following locations at the given rates. During the initial 24 hours, 0.15%/day total leakage of which 0.084%/day leaks to the Annulus, 0.06% to the Auxiliary Building Special Ventilation Zone, and 0.006%/day directly to the outside environment. These leak rates 01466114 are consistent with the containment leak rate testing program requirements and RG 1.183 guidelines. From 24 hours to 30 days, 0.075%/day total leakage of which 0.042%/day to the Annulus, 0.03% to the Auxiliary Building Special Ventilation Zone, and 0.003%/day directly to the outside environment. These leak rates are consistent with containment leak testing program requirements and RG 1.183 guidelines. The portion released to the ABSVZ is filtered prior to release to the environment. Prior to establishing the negativepressure in the Annulus, the activity is postulated to leak directly to the environment. After the Shield Building Ventilation System is started with Recirculation filtration and has established a negative pressure in the Annulus, the activity is recirculated; which allows time for mixing, decay, and filtering. Following hold-up and decay in the Shield Building, activity leaves the Shield Building through filters. Of the activity passing through the filters, a portion is released and the remainder is recirculated back into the Shield Building.

Prior to establishing negative pressure in the Annulus and Auxiliary Building Special Ventilation Zone, the containment leakage is assumed to be released directly to the environment without filtering. After the Shield Building Ventilation System and the Auxiliary Building Special Ventilation System have established a negative pressure, activity is assumed to be filtered prior to release.

The initiating signal for the SBVS and the ABSVS is the Safety Injection (SI) signal. Startup of the ABSVS results in the automatic termination of the normal operation Auxiliary Building Ventilation System and closure of the associated exhaust dampers. It is determined by calculation that the normal operation Auxiliary Building Ventilation System exhaust dampers are closed prior to environmental release of any airborne activity in the Auxiliary Building due to containment leakage.

14.9.4 Sequence of Events Within the Shield Building and Auxiliary Building Special Vent Zone

As discussed previously, the Shield Building Ventilation System is designed to provide three functions during the course of the loss-of-coolant accident:

1. Provide a negative pressure region to control and limit environmental leakage.

2. Enhance mixing and dilution of any containment vessel leakage to the annulus.

3. Provide hold-up and long-term filtration of annulus air.

Immediately following the accident, the Shield Building pressure increases due to heat transferred from the containment shell. Operation of one of the two redundant sets of recirculation and exhaust fans establishes a negative pressure within about three minutes. The Shield Building Ventilation system continues to remove air from the annulus without recirculation for 12 minutes.

A careful examination of the sequence of events in the annulus is therefore required to assess the consequences of the postulated accident.

For convenience and conservatism in determining the effect of short term transient system behavior on the offsite dose, the following time periods have been defined for use in the dose analysis.

14.9.4.1 Containment Leakage that is collected in the annulus (Shield Building)

1. 0 to 36 seconds

Immediately following the accident (time t=0), the Shield Building annuluspressure increases due to containment shell expansion and heat transfer from the containment shell to the annulus air. The Shield Building Ventilation system's two redundant exhaust fans are not active during this time period. During this time period, no credit is taken for filtered exhaust from the Shield Building Ventilation system. The dose analysis assumes that all containment leakage entering the Shield Building during this time interval is released directly to the environment, without benefit of Shield Building dilution or recirculation filtration.

2. 36 seconds to 12 minutes

One Shield Building Ventilation system fan begins to draw air from the Shield Building annulus at 36 seconds. The fan is operated in the filtered exhaust mode with no credit being taken for filtered recirculation. The dose analysis assumes that all containment leakage entering the Shield Building during this time interval is released directly to the environment, without benefit of Shield Building dilution or recirculation filtration.

3. 12 minutes to 22 minutes

After 12 minutes, a negative pressure of approximately -2 inches of water with respect to the atmosphere is achieved in the shield building annulus. During this period the containment building leakage to the Shield Building is 0.084 weight percent per day. The dose analysis assumes that all containment leakage entering the Shield Building during this time interval is released directly to the environment, without benefit of Shield Building dilution or recirculation filtration. It is conservatively assumed that no filtered recirculation takes place during this time interval. During this time period, the shield building annulus is at a negative pressure with respect to the atmosphere.

4. <u>22 minutes to 1 day</u>

Equilibrium exhaust flow through the Shield Building Ventilation system is established at 22 minutes. Credit for filtered exhaust and recirculation to the shield building is taken during this interval. This time interval has been selectedon the basis of NRC Regulatory Guide 1.183 and is a conservative estimate of the calculated pressure-transient history for the Design Basis case.

Shield Building Ventilation system filter efficiencies of 0 percent for the removal of elemental iodine, 0 percent for the removal of organic iodine, and 99 percent for the removal of particulate isotopes have been conservatively assumed in the analysis.

5. <u>1 day to 30 days</u>

During this period the containment building leakage to the Shield Building is halved to 0.042 weight percent per day. This period is characterized by full equilibrium flow in the system. The filtered release to the atmosphere just balances the shield building inleakage. Filtered recirculation to the Shield Building annulus continues to reduce the annulus inventory of radioactive aerosol isotopes.

The amount of filtered venting following the termination of the initial pressuretransient (after 22 minutes) is dependent upon the negative pressure in the annulus as set by the exhaust fan.

14.9.4.2 Containment Leakage that is collected in the Auxiliary Building Special Ventilation Zone

1. <u>0 to 20 minutes</u>

A negative pressure with respect to atmosphere is achieved in the Auxiliary Building Special Ventilation Zone within 20 minutes after accident initiation. No credit is taken for filtration of releases via the ABSVS during this period. During this time period, the containment leakage is assumed to be released directly to the environment from the Shield Building wall.

2. <u>20 minutes to 30 days</u>

Negative pressure with respect to atmosphere is achieved in the Auxiliary Building Special Ventilation Zone. During this period, although no mixing is credited, releases from the containment into the Auxiliary Building Special Ventilation Zone are filtered by the ABSVS before release via the Shield Building Stack.

14.9.4.3 ESF Leakage

See Section 6.7.1.1

14.9.4.4 RWST Back-leakage

See Section 6.7.1.2

14.9.5 Method of Analysis

A radiological evaluation was performed to demonstrate that control room and off-site doses resulting from a loss-of-coolant accident are within the limits specified in 10 CFR 50 GDC 19 and 10 CFR Part 100 (Reference 41).

Doses due to Submersion and Inhalation

Control Room and Offsite dose due to airborne radioactivity releases following a LOCA are calculated using the RADTRAD computer code.

The inventory of fission products in the reactor core presented in Appendix D, Table D.1.-1 represent a conservative equilibrium reactor core inventory of dose significant isotopes, assuming maximum full power operation at 1.02 times the current licensed thermal power, and taking into consideration approved fuel enrichment and burnup. For the post-LOCA dose analysis, the fission product inventory in Table D.1-1 is extrapolated to 1852 MWt.

Control Room Dose due to Direct Shine from the External Cloud and Contained Sources:

Consistent with RG 1.183, that the LOCA dose analysis considers the following sources of radiation shine that will cause exposure to control room personnel:

1. Radiation shine from the external radioactive plume released from the facility (i.e., external airborne cloud shine dose),

2. Radiation shine from radioactive material in the reactor containment (i.e., containment shine dose), and

3. Radiation shine from radioactive material in systems and components inside or external to the control room envelope (e.g., radioactive material buildup in CR intake and recirculation filters [i.e., CR filter shine dose]).

RADTRAD was used to calculate the radiation source term in post-LOCA airborne source in the containment and in the Shield Building, in the external plume passing the Control Room due to containment leakage, ESF leakage and RWST back-leakage, and accumulated in the control room emergency ventilation filters. MicroShield is used to calculate the dose inside the Control Room by modeling the source-shield detector configuration.

14.9.5.1 Deleted

14.9.5.2 Deleted

14.9.5.3 Calculation of Offsite Dose

The dispersion coefficients (X/Q) have been derived from the onsite meteorology program and are discussed in Appendix H. The releases are assumed at ground level with a correction for building wake effects.

Table 14.9-5 lists the key assumptions / parameters utilized to develop radiological consequences.

The results of the integrated dose analyses are summarized in Table 14.9-2.

14.9.5.4 Dose to Control Room Personnel

Radiological doses resulting from a design basis LOCA for a control room operator are less than the regulatory dose limits as shown in Table 14.9-2.

14.9.6 Evaluation of Results

A comparison of offsite thyroid and whole body doses computed using releases consistent with the Regulatory Guide 1.183 model (see Table 14.9-2) demonstrates the conservatism of the Prairie Island Nuclear Generating Plant safeguards.

Shield Building leakage, mixing of leakage within the Annulus volume, and filter efficiency have been assigned values justified by the containment and safety features design. A discussion of the key analytical parameters is presented in the following sections.

14.9.6.1 Effect of Shield Building Mixing (Participation Fraction)

This section describes historical information that is retained but is not directly applicable to the Regulatory Guide 1.183 dose analysis.

Leakage from the Reactor Containment Vessel disperses and mixes with air in the Shield Building. Mixing is aided by several design features of the Shield Building.

1. During the period immediately following the loss-of-coolant accident, natural circulation flow upward along the Reactor Containment Vessel shell and downward near the Shield Building wall is promoted by the temperature differential across the annulus.

2. Except for the initial 4.5 minutes, all air leaving the annulus is collected by the Shield Building Ventilation System suction header located at the top of the annulus. This location maximizes the distance between the suction header and the most probable sources of leakage, the penetrations, which are all located in approximately the lower third of the Reactor Containment Vessel. Over the large transport distance from the penetrations to the suction header, mixing will be induced by diffusion and the flow of recirculated air.

3. The recirculated and filtered air which is returned to the Shield Building is directed by specially designed ducting to sweep past the Containment System penetrations and upward in the annulus with a turbulent motion.

The effect of non-uniform mixing in the Shield Building annulus volume is considered by defining a participation fraction which represents the fractional Shielding Building volume available for dilution, filtration and recirculation. The effect that variation of the Shield Building participation fraction has on the integrated 2 hour thyroid dose is shown in Figure 14.9-5.

In the range of .50 to 1.00 it can be seen that the marginal increase in dose is something less than proportional to the fractional decrease in participating volume. This is due primarily to the filtration capability of the recirculation system. For small participation fractions, the trend is to a direct proportionality with dose. Allowing for recirculation during the first 20 minute period would be expected to lower this part of the curve with consequent dose reduction.

14.9.6.2 Deleted

14.9.6.3 Effectiveness of Containment Spray System for Iodine Removal and Retention

No credit is taken in the dose analysis for iodine removal by operation of the Containment Spray system. Credit is taken for NaOH added to the containment sump water as a result of operation of the containment spray pump(s).

Regulatory Guide 1.183, Appendix A, Section 2, states:

> "If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event; radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form."

An evaluation of containment sump liquid pH was performed to ensure that the particulate iodine deposited into the containment water during the DBA LOCA does not re-evolve beyond that recognized in the DBA LOCA analysis. The objective of the analysis was to determine the transient containment sump pH so that the removal of elemental and particulate iodine (cesium iodide – CsI) from the containment atmosphere in the course of the DBA LOCA would not be overstated. The analysis credits the pH buffering effect of the addition of Sodium Hydroxide (NaOH) in the containment spray fluid during injection.

The pH of the containment sump water was determined using the MAAP-AST computer code (Reference 57).

In calculating the sump pH, the three major contributors to strong acid production are considered: boric acid from the reactor coolant system, the accumulators, and the refueling water storage tank (RWST); nitric acid from radiolysis of water; and, hydrochloric acid from radiolysis of chloride-bearing cable jacket/insulation.

The analysis results show the following. The sump water initially starts out acidic. After injection of NaOH begins the pH quickly increases to greater than 7.0 (well before significant radiolysis occurs) and remains above 7.0 for the remaining duration of the analysis period. Consistent with Regulatory Guide 1.183, Appendix A, re-evolution of iodine is not considered in the dose analysis.

14.9.6.4 Deleted

14.9.7 Charcoal Filter Ignition Hazard Due to Iodine Absorption

During a loss of coolant accident, the Shield Building ventilation system recirculates and filters the air volume in the Annulus. During this process, radioactive iodine would collect on the charcoal filters. This radioactive iodine which collects on the charcoal filters generates decay heat. During system operation, the ventilation flow through the filters provides forced cooling for the filter. Without cooling, if the elevated temperatures in the charcoal were reached, iodine desorption or spontaneous ignition may occur. The following describes an analysis which determines the effects on the charcoal of a loss of forced air flow.

The following analysis assessing the charcoal filter ignition hazard is based on Regulatory Guide 1.4 guidance which assumes that 95 percent of the iodines released from the damaged core are elemental and organic species that are subject to collection on a charcoal filter. Per Regulatory Guide 1.183 (Reference 109), the alternative source term (AST) methodology assumes that 5 percent of the iodines released from the damaged core are elemental and organic species that are subject to collection on a charcoal filter. Application of the AST methodology effectively reduces the quantity of iodine available for collection on a charcoal filter by a factor of nineteen, thereby greatly reducing the charcoal filter ignition hazard. It is also note that the LOCA AST dose calculation does not credit Shield Building Ventilation System charcoal filtration.

The maximum amount of heat that can be generated on the filters is limited to the iodine collected on the filters. This is a function of the rate at which the iodine leaks out of the Reactor Containment Vessel into the Shield Building Annulus, and by the decay of the same isotopes before they are collected on the filter.

The calculation (Reference 51) for this case determined the maximum temperature the charcoal would achieve after a loss of all air flow, with the Fire Protection system water spray to the filters unavailable. The analysis in Reference 51 is based on using OFA high burn-up fuel. There is very little difference between the curie content for the iodines in the OFA vs. Vantage fuels. This is to be expected since the nuclides with the short half lives will not increase with increased burn-up. Thus, the use of the Vantage fuel would not have a significant affect on the results of this analysis.

In the analysis, the following conservative assumptions are made:

1. Consistent with NRC Regulatory Guide 1.4, 25 percent of the halogens in the fuel are available for release from the containment environment immediately after the accident.

2. No credit is taken for plateout in the Shield Building, and no credit is taken for the iodine removal by the Containment Vessel Internal Spray System.

3. Containment leakage rate is assumed to 0.25 percent per day for the first 24 hours and 0.125 percent per day for the remainder of the accident. All containment leakage is assumed to collect in the Shield Building.

4. Consistent with R.G.1.4, credit is taken for the effects of radiological decay during holdup in the Shield Building.

5. Initially both trains of Shield Building Ventilation System are assumed to be operating. The iodine in the Shield Building is assumed to be evenly distributed between the two trains. If the assumption is made that only one train is operating at the initiation of the accident this would be the single active failure and would preclude a loss of air flow to a filter which has iodine loading.

6. To determine the iodine buildup on the filters, a filter efficiency of 100% is assumed.

7. When the maximum iodine loading is reached, one train of forced air flow is assumed to fail (i.e., the associated Motor Control Center fails). This is the single active failure assumed in this analysis. The iodine in the charcoal bed with the loss of air flow starts to generate heat due to the absence of forced cooling.

8. Credit is taken for heat transfer from the filter to the surroundings. This is based on the maximum expected temperatures in the surrounding area; i.e., the air temperature in the Shield Building is assumed to be 163°F (Appendix G), and the air temperature in the fourth floor of the Auxiliary Building is assumed to be 105°F.

Based on these assumptions, the maximum heat generation rate in the charcoal filters is determined to be approximately 207 watts per train. The temperature history of the filters is obtained by crediting the heat dissipation from the filter surfaces to the surroundings. Based on the calculated heat generation rate, the peak temperature of approximately 165°F is reached within one minute after the loss of forced air flow. This is well below the temperature at which iodine desorption begins (300°F), and is quite removed from the temperature of charcoal ignition.

Therefore, the Fire Protection water spray systems for the Shield Building Ventilation System (SBVS) filters are not required as essential support equipment for the filter units to be considered to be operable.

This analysis of the Shield Building Ventilation Filter envelopes the Auxiliary Building Special Ventilation System filters as well. The analysis above assumes the maximum allowable containment leakage is all loaded on the two SBVS filters. In an accident, some iodine may bypass the Shield Building and be adsorbed on the ABSVS filters. Since the maximum allowable leakage to the ABSVZ is 0.10%day at 46 psig per the Containment Leakage rate testing program, the loading on the two ABSVS filters would be bounded by the loading on the SBVS filters. Therefore, the ABSVS filter temperature could not exceed 165°F, and the Fire Protection water spray systems are not required for these filter units to be considered operable either.

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14.10 LONG TERM COOLING FOLLOWING A LOCA

14.10.1 General

Following a LOCA and prior to switchover from the RWST to the containment sump, the RWST can supply water to the RHR pumps (vessel injection), SI pumps (cold leg injection) and containment spray pumps (containment pressure control). After switchover to low-head recirculation, the RHR Pump(s) supply water directly to the reactor vessel. After switchover to high-head recirculation, the RHR Pump(s) supply water directly to the suction of the SI Pump(s) and the SI Pump(s) supply water to the RCS cold legs.

14.10.2 Minimum Flow Requirements

Following a LOCA event, the ECCS system must be able to maintain sufficient flow to the core to assure adequate Long Term Core Cooling (LTCC) per 10 CFR 50.46(b)(5). Per References 93 and 95 (Section 2.4.2.2.c), Long Term Core Cooling can be assured by confirming ECCS realignment for sump recirculation is complete, boiloff rates are less than injection flows, the core is quenched and the fuel cladding temperature is maintained below approximately 800 degF[1]. The water injected through either the RHR pumps or SI pumps provides the primary mechanism to meet these criteria. During transfer of the ECCS pumps to recirculation, injected flows can be significantly reduced or eliminated for short durations. For breaks sufficiently large to depressurize the RCS below the RHR pump shutoff head, continued single train ECCS injection through the recirculation realignment is maintained by the SI pump. For smaller breaks without RHR flow, there will be a flow interruption while the single operating SI pump is realigned for recirculation. Adequate Long Term Core Cooling has been evaluated for these two scenarios in References 93 and 94.

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The critical ECCS assumptions used to confirm the acceptance criteria are met are as follows:

- The minimum time before an RHR pump is shutdown for transfer to recirculation is 20 minutes.
- The maximum time before restarting an RHR pump following shutdown in preparation for transfer to recirculation is 14.4 minutes.
- The maximum time for full flow interruption prior to restarting a SI pump on recirculation is 8.4 minutes.
- For large breaks evaluated in Reference 93, the assumed minimum RHR and SI flow to the core during and after transfer to recirculation is 440 gpm and 290 gpm, respectively.
- For small breaks evaluated in Reference 94, the assumed minimum RHR and SI flow to the core before, during and after transfer to recirculation is shown in Table 14.10-2 through 14.10-4.

Using the above assumptions and success criteria, it has been shown that Long Term Core Cooling can be maintained.

(1) Per Reference 93, the 800 degree criteria can be sustained for up to 30 days. Shorter durations at higher temperatures may be justified to assure adequate LTCC.

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TABLE 14.1-1 ANS 51.1 CLASSIFICATION OF PLANT CONDITIONS

		Definition	Design Requirements
Condition I	Normal Operation	Operations that are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant.	Shall be accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action.
Condition II	Incidents of Moderate Frequency	Include Incidents, any one of which may occur during a calendar year for a particular plant.	Shall be accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action. Any release of radioactive materials in effluents to unrestricted areas shall be in conformance with paragraph 20.1 of 10CFR20, "Standards for Protection Against Radiation." By itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV type without other incidents occurring independently. A single Condition II Incident shall not cause consequential loss of function of any barrier to the escape of radioactive product.

TABLE 14.1-1 ANS 51.1 CLASSIFICATION OF PLANT CONDITIONS

		Definition	Design Requirements
Condition III	Infrequent Incidents	Include Incidents, any one of which may occur during the lifetime of a particular plant.	Incidents shall not cause more than a small fraction of the fuel elements in the reactor to be damaged, although sufficient fuel element damage might occur to preclude resumption of operation for a considerable outage time. The release of radioactive material due to Condition III Incidents may exceed guidelines of 10CFR20, "Standards for Protection Against Radiation," but shall not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III incident shall not, by itself, generate a Condition IV fault or result in a consequential loss of function of the reactor coolant system or reactor containment barrier.
Condition IV	Limiting Faults	Are faults that are not expected to occur, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. Condition IV faults are the most drastic that must be designed against, and thus represent the limiting design case.	Shall not cause a release of radioactive material that results in an undue risk to public health and safety exceeding the guidelines of 10CFR100, "Reactor Site Criteria." A single Condition IV fault shall not cause a consequential loss of required function of systems needed to cope with the fault including those of the reactor coolant system and the reactor containment system. 10CFR100 criteria were originally used for the radiological analysis and establishing the ANS 51.1 Classifications. Subsequently, the radiological analyses have been performed to meet the criteria in 10CFR50.67. This does not affect the classifications.

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TABLE 14.3-1
SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED FOR NON-LOCA ACCIDENT ANALYSES

Transient/Event	Computer Codes Used	DNB Correlation	Revised Thermal Design Procedure	Initial Core Power MWt	Vessel Coolant Flow (gpm)	Vessel Avg. Coolant Temp. (°F)	RCS Pressure (psia)
Uncontrolled RCCA Withdrawal from a Subcritical Condition (Section 14.4.1)	TWINKLE FACTRAN VIPRE	W-3[1] WRB-1[2]	No	0	79,922[4]	547.0	2190
Uncontrolled RCCA Withdrawal at Power (Section 14.4.2)	RETRAN	WRB-1	Yes (DNB) N/A (Pressure)	100, 60, 10% (DNB/MSS Pressure) 60% (RCS Pressure) 100% = 1683	183,400 (DNB) 178,000 (Pressure)	560.0 (100%-DNB) 556.0 (100%-Press.) 554.8 (60%-DNB) 550.8 (60%-Press.) 548.3 (10%-DNB) 544.3 (10%-Press.) 558.8 (60%-RCS Press)	2250 (DNB) 2190 (RCS Pressure) 2290 (MSS Pressure)
RCCA Misalignment (Dropped Rod) (Section 14.4.3)	LOFTRAN[3] ANC VIPRE	WRB-1	Yes	1677	183,400	560.0	2250
Chemical and Volume Control System Malfunction (Section 14.4.4)	N/A	N/A	N/A	(Mode 1) (Mode 2)	N/A	564.0 (Mode 1) 551.65 (Mode 2)	2250 (modes 1 and 2)
Startup of an Inactive Reactor Coolant Loop (Section 14.4.5)	Event precluded by the Technical Specifications						
Feedwater Temperature Reduction Incident (Section 14.4.6)	Event bounded by the Excessive Load Increase Incident						
Feedwater Flow Increase (Section 14.4.6)	RETRAN VIPRE	WRB-1 (HFP) W-3 (HZP)	Yes (HFP) No (HZP)	1683 (HFP) 0 (HZP)	183,400 (HFP) 178,000 (HZP)	560.0 (HFP) 547.0 (HZP)	2250
Excessive Load Increase (Section 14.4.7)	RETRAN	WRB-1	Yes	1683	183,400	560.0	2250

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TABLE 14.3-1
SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED FOR NON-LOCA ACCIDENT ANALYSES

Page 2 of 2

Transient/Event	Computer Codes Used	DNB Correlation	Revised Thermal Design Procedure	Initial Core Power MWt	Vessel Coolant Flow (gpm)	Vessel Avg. Coolant Temp. (°F)	RCS Pressure (psia)
Loss of Reactor Coolant Flow - Flow Coast Down (Section 14.4.8)	RETRAN VIPRE	WRB-1	Yes	1677	183,400	560.0	2250
Loss of Reactor Coolant Flow - Locked Rotor (Section 14.4.8)	RETRAN VIPRE	WRB-1	Yes (DNB) N/A (Hot Spot, Pressure)	1677 (DNB) 1683 (Hot Spot, Pressure)	183,400 (DNB) 178,000 (Hot Spot Pressure)	560.0 (DNB) 564.0 (Hot Spot, Pressure)	2250 (DNB) 2310 (Hot Spot, Pressure)
Loss of External Electrical Load (Section 14.4.9)	RETRAN	WRB-1	Yes (DNB) N/A (Pressure)	1683 (DNB) 1683 (Pressure)	183,400 (DNB) 178,000 (Pressure)	560.0 (DNB) 564.0 (Pressure)	2250 (DNB) 2190 (Pressure)
Loss of Normal Feedwater (Section 14.4.10)	RETRAN	N/A	N/A	1683	178,000	556.0	2290
Loss of All AC Power to the Station Auxiliaries (Section 14.4.11)	RETRAN	N/A	N/A	1683	178,000	556.0	2290
Rupture of a Steam Pipe - Full Power core Response (Section 14.5.6)	RETRAN VIPRE	W-3[1] WRB-1[2]	No[1] Yes[2] No-(kw/ft)	1683 [1] 1683[2]	178,000[1] 183,400[2]	564.0[1] 560.0[2]	2190[1] 2250[2]
Rupture of a Steam Pipe - Zero Power Core Response (Section 14.5.5)	RETRAN VIPRE	W-3	No	0	178,000	547.0	2250
RCCA Ejection (Section 14.5.6)	TWINKLE FACTRAN	N/A	N/A	1683 0 (HZP)	178,000 (HFP) 79,922[4] (HZP)	564.0 (HFP) 547.0 (HZP)	2190

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(1) Below the first mixing vane grid. (2) Above the first mixing vane grid. (3) The LOFTRAN portion of the analysis is generic; the DNB evaluation performed with VIPRE utilizes the plant-specific values presented. (4) Single-loop flow = 0.449*TDF

TABLE 14.3-2 NOMINAL VALUES OF PERTINENT PARAMETERS FOR NON-LOCA ACCIDENT ANALYSES

Parameter	RTDP	Non-RTDP
Thermal Output of NSSS (MWt)	1684	1690
Vessel Average Coolant Temperature (°F)	560.0	560.0 $\pm$ 4.0
Pressurizer Pressure (psia)	2250.0	2250 -60/+40
Reactor Coolant Loop Flow (GPM)	91,700	89,000
Maximum Steam Generator Tube Plugging	10%	10%
Steam Generator Pressure (psia)	765 (0% SGTP) 751 (10% SGTP)	778 (0% SGTP) 763 (10% SGTP)
Assumed Feedwater Temperature at Steam Generator Inlet (°F)	437.5	437.5

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TABLE 14.3-3 DESIGN BASIS LIMITS FOR FISSION PRODUCT BARRIERS

Barrier	Design Bases Parameter	Design Basis Limit
Fuel Cladding	DNBR (RTDP, WRB-1) DNBR (non-RTDP, W-3)	1.17 (correlation limit) 1.45 (correlation limit, 500-1000 psia) 1.30 (correlation limit, > 1000 psia
	Fuel Temperature	4700°F (except Rod Ejection)
	Fuel enthalpy Failed fuel pins (Locked Rotor) Fuel Melt	200 cal/g < 20% < 10% of fuel pellet at Hot Spot7 (Rod Ejection only)
	Clad temperature	2700°F (Locked Rotor, Standard ZIRLO clad) 2375°F (Locked Rotor, Optimized ZIRLO clad) 2200°F (LOCA)
	Clad Oxidation	17% local and 1% overall
RCS Boundary	Pressure	2750 psia (except Rod Ejection) 3200 psia (Rod Ejection only)
Main Steam	Pressure	1210 psia
Containment	Pressure	46 psig

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TABLE 14.4-1 SEQUENCE OF EVENTS
UNCONTROLLED RCCA WITHDRAWAL FROM A SUBCRITICAL CONDITION

Event	Time (seconds)	
	400V+	422V+
Initiation of Uncontrolled RCCA Bank Withdrawal	0	0
Power Range High Neutron Flux Low Setpoint Reached	10.0	10.0
Peak Nuclear Power Occurs	10.1	10.1
Rod Motion Begins	10.45	10.45
Peak Heat Flux Occurs	12.3	12.3
Minimum DNBR Occurs	12.3	12.3
Peak Cladding Temperature Occurs	12.6	12.65
Peak Fuel Average Temperature Occurs	13.0	12.9
Peak Fuel Centerline Temperature Occurs	14.5	14.1

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TABLE 14.4-2, DELETED

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TABLE 14.4-3 UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER TIME SEQUENCE OF EVENTS REPLACEMENT STEAM GENERATORS

Accident	Event	Time (seconds)
Uncontrollable RCCA bank withdrawal at power		
1. Case A	Initiation of uncontrollable RCCA withdrawal at a high reactivity insertion rate (100 pcm/sec)	0
	High Positive Flux rate trip setpoint reached	0.75
	Rods begin to fall into core	2.25
	Minimum DNBR occurs	2.00
2. Case B	Initiation of uncontrollable RCCA withdrawal at a small reactivity insertion rate (1 pcm/sec)	0
	Overtemperature ΔT reactor trip setpoint reached	65.41
	Rods begin to fall into core	67.91
	Minimum DNBR occurs	68.00

TABLE 14.4-4 CVCS MALFUNCTION TYPICAL SHUTDOWN MARGIN REQUIREMENTS

Plant Conditions[1]	Number of Charging Pumps Running		
	0-1 Pump	2 Pumps	3 Pumps
Mode 1	1.7	1.7	1.7
Mode 2	1.7	1.7	1.7
Mode 3, $T_{avg} \geq 520°F$	2.0	2.0	2.0
Mode 4	2.0	4.5	7.0
Mode 5	2.5	5.0	7.5
Mode 6, ARI[2]	5.129	5.129	7.0
Mode 6, ARO[3]	5.129	6.0	9.0

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(1) The Operational Mode Definitions are located in the Prairie Island Technical Specifications.

(2) ARI - All Rods In

(3) ARO - All Rods Out

TABLE 14.4-5, DELETED

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TABLE 14.4-6, DELETED

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TABLE 14.4-7 FRAMATOME ANP RSGs - SEQUENCE OF EVENTS FOR FEEDWATER SYSTEM MALFUNCTION EVENT AT FULL POWER (AUTOMATIC ROD CONTROL)

Event	**Time (Seconds)**	
	Single Loop Failure	**Multi Loop Failure**
Main Feedwater Control Valve(s) Fail Full Open	0.0	0.0
Minimum DNBR Occurs	83.3	32.0
Hi-Hi Steam Generator Water Level Trip Setpoint is Reached	103.3	98.8
Turbine Trip Occurs Due to Hi-Hi Steam Generator Level	104.2	99.7
Reactor Trip Occurs Due to Turbine Trip	106.2	101.8
Feedwater Isolation Valves Fully Closed	128.7	124.2

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TABLE 14.4-8 FRAMATOME ANP RSGs - SEQUENCE OF EVENTS FOR FEEDWATER SYSTEM MALFUNCTION EVENT AT FULL POWER (MANUAL ROD CONTROL)

Event	Time (Seconds)	
	Manual Rod Control Single Loop Failure	Manual Rod Control Multi Loop Failure
Main Feedwater Control Valve(s) Fail Full Open	0.0	0.0
Hi-Hi Steam Generator Water Level Trip Setpoint is Reached	102.9	97.9
Turbine Trip Occurs Due to Hi-Hi Steam Generator Level	103.8	98.8
Minimum DNBR Occurs	96.0	100.5
Reactor Trip Occurs Due to Turbine Trip	105.9	100.8
Feedwater Isolation Valves Fully Closed	128.3	123.3

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TABLE 14.4-9 TIME SEQUENCE OF EVENTS FOR EXCESSIVE LOAD INCREASE INCIDENT

	Case	Event	Time(s)	
1.	Minimum Reactivity Feedback (automatic rod control)	20-percent step load increase	10.0	
		Equilibrium conditions reached (approximate time)	70	01183316
2.	Minimum Reactivity Feedback (manual rod control)	20-percent step load increase	10.0	
		Equilibrium conditions reached (approximate time)	150	01183316
3.	Maximum Reactivity Feedback (automatic rod control)	20-percent step load increase	10.0	
		Equilibrium conditions reached (approximate time)	60	01183316
4.	Maximum Reactivity Feedback (manual rod control)	20-percent step load increase	10.0	
		Equilibrium conditions reached (approximate time)	60	01183316

TABLE 14.4-10 TIME SEQUENCE OF EVENTS FOR LOSS OF ELECTRICAL LOAD

Case	Event	RSGs Time (sec)
1. With Pressurizer Control		
	Turbine trip, loss of main feedwater flow	20.0
	Overtemperature ΔT reactor trip setpoint reached	30.1
	Rods begin to drop	32.6
	Minimum DNBR occurs	33.2
	Initiation of steam release from steam generator safety valves	36.5
2. Without Pressurizer Control		
	Turbine trip, loss of main feedwater flow	20.0
	High pressurizer pressure trip setpoint reached	28.0
	Rods begin to drop	29.0
	Pressurizer safety valves lift	30.8
	Peak RCS Pressure occurs	32.0
	Initiation of steam release from steam generator safety valves	35.3

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TABLE 14.4-11 SEQUENCE OF EVENTS - LOSS OF NORMAL FEEDWATER

Event	**Time (seconds)** **RSGs**
Main Feedwater Flow Stops	20.0
Low-Low Steam Generator Water Level Trip Setpoint Reached	59.1
Rods Begin to Drop	60.6
Both Steam Generators Begin to Receive AFW Flow from One Pump	119.1
Peak Water Volume in the Pressurizer Occurs*, Core Decay Heat (plus RCP Heat) Decreases to AFW Heat Removal Capacity	~7800
*Peak Pressurizer Water Volume, ft^3	934.1
Pressurizer Water Volume Limit, ft^3	1000

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TABLE 14.4-12 SEQUENCE OF EVENTS - LOSS OF ALL AC POWER TO THE STATION AUXILIARIES

Event	**Time (seconds)**
	RSGs
Main Feedwater Flow Stops	20.0
Low-Low Steam Generator Water Level Setpoint Reached	59.1
Rods Begin to Drop	60.6
RCPs Begin to Coast Down	62.6
Both Steam Generators Begin to Receive AFW Flow from One Pump	119.1
Peak Water Volume in the Pressurizer Occurs*, Core Decay Heat Decreases to AFW Heat Removal Capacity	~1560
* Peak Pressurizer Water Volume, ft^3	653.0
Pressurizer Water Volume Limit, ft^3	1000

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TABLE 14.4-13 LOCKED ROTOR ACCIDENT DOSE CONSEQUENCE PARAMETERS AND ASSUMPTIONS

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Parameter	Value	
Rated Core Termal Power Assumed (licensed value)	1,852 MWt	
Plant Status Assumed:		
Offswite Power	Not Available	
Main Condensers	Not Available	
Nominal Reactor Coolant System (RCS) Volume	5,290 ft^3	
Unit 1 Steam Generator Liquid Mass (Framatome 56/19)	107,100 lbm	
Unit 2 Steam Generator Liquid Mass (Westinghouse 51) (subsequently replaced with SG similar to Unit 1)	107,420 lbm	
Primary & Secondary Coolant Parameter		
Primary Coolant iodine specific activity	0.5 µCi/gm DE I-131	
Primary Coolant non-iodine specific activity	580 µCi/gm DE Xe-133	
Secondary Coolant iodine specific activity	0.1 µCi/gm DE I-131	
Fuel Damage as a Result of the Accident	20% clad damage	
Primary-to-Secondary (P-T-S) Leakage	150 gpd per SG	
Partition Coefficients	Iodine	Noble Gas
Steam Generators (P-T-S)	100	1.0
Steam Generators (Secondary Liquid)	100	1.0
Activity Release Duration for the Accident	45.5 hours	
Steam Releases from the Intact SG via PORVs to Environment		
0 – 2 hr	226,414 lbm	
2 – 8 hr	406,952 lbm	
8 – 24 hr	796,899 lbm	
24 – 45.5 hr	863,053 lbm	

TABLE 14.4-13 LOCKED ROTOR ACCIDENT DOSE CONSEQUENCE PARAMETERS AND ASSUMPTIONS

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Parameter	Value
Control Room Atmospheric Dispersion Factors (X/Q) for Intact SG Releases (Unit 2 Group 1 PORV to Unit 2 CR Vent Intake)	
0 – 2 hr	3.07E-02 sec/m^3
2 – 8 hr	2.49E-02 sec/m^3
8 – 24 hr	1.12E-02 sec/m^3
24 – 96 hr	7.78E-03 sec/m^3
Control Room Atmospheric Dispersion Factors (X/Q) for Intact SG Releases (Unit 2 Group 2 PORV to Unit 2 CR Vent Intake)	
0 – 2 hr	2.20E-03 sec/m^3
2 – 8 hr	1.81E-03 sec/m^3
8 – 24 hr	7.97E-04 sec/m^3
24 – 96 hr	5.16E-04 sec/m^3
Control Room (CR) Parameters	
CR Volume	61,315 ft^3
CR HVAC Emergency Mode Actuation Delay	5 minutes
Unfiltered In-leakage	300 cfm
Unfiltered Normal Mode Make-up Flow (< 5 minutes)	2,000 cfm
Filtered Recirculation Mode Flow (> 5 minutes)	3,600 cfm
Filter Efficiencies	
Elemental	95%
Organic	95%
Particulate	99%
CR Breathing Rate	3.5E-04 m^3/sec
CR Occupancy Factors	
0 – 24 hr	1.0
1 – 4 days	0.6
4 – 30 days	0.4
EAB Atmospheric Dispersion Factor (X/Q)	6.49E-04 sec/m^3
EAB Parameters	
EAB Breathing Rate	3.5E-04 m^3/sec
EAB Occupancy Factor	1.0 (any 2-hour period)

TABLE 14.4-13 LOCKED ROTOR ACCIDENT DOSE CONSEQUENCE PARAMETERS AND ASSUMPTIONS

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Parameter	Value
LPZ Atmospheric Dispersion Factors (X/Q)	
0-8 hr	1.77E-04 sec/m^3
8-24 hr	3.99E-05 sec/m^3
24-96 hr	7.12E-06 sec/m^3
96-720 hr	1.04E-06 sec/m^3
LPZ Parameters	
LPZ Breathing Rate	
0-8 hr	3.5E-04 m^3/sec
8-24 hr	1.8E-04 m^3/sec
24-720 hr	2.3E-04 m^3/sec
LPZ Occupancy Factor	1.0

TABLE 14.5-1 ASSUMPTIONS USED FOR FHA DOSE ANALYSIS (AST)

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Parameter	**Input**
Core Power Level	1852 MWt
Radial Peaking Factor	1.90
Fuel Damaged	All Rods in 1 assembly
Time from Shutdown before Fuel Movement	50 hrs
Activity in the Damaged Fuel Assembly	Appendix D, Table D.3-2
Gap Fractions	Appendix D, Table D.2-1
Chemical Form of Iodine in Pool	
Cesium iodide (CsI)	95%
Elemental	4.85%
Organic	0.15%
Water Depth (minimum)	23 feet
Overall Pool Iodine Scrubbing Factor (DF)	200
Noble Gas Scrubbing Factor (DF)	1.0
Particulate Scrubbing Factor (DF)	Infinite
Filter Efficiency - (SFP Special Vent)	No filtration assumed
Isolation of Release	No isolation assumed
Time to Release All Activity	2 hrs
Atmospheric Dispersion Factors (χ/Q)	
Control Room	4.19E-03 sec/m^3
Exclusion Boundary Area	6.49E-04 sec/m^3
Low Population Zone	1.77E-04 sec/m^3
Breathing Rate	
0-8 hours*	3.5E-04 m^3/sec
8-24 hours	1.8E-04 m^3/sec
7-24 hours	2.3E-04 m^3/sec

*Breathing rate used for control room analysis for the duration of the event.

TABLE 14.5-2 CONTROL ROOM PARAMETERS FOR FHA DOSE ANALYSES

Parameter	**Input**
Control Room Volume	61,315 ft^3
Control Room Unfiltered In-Leakage***	300 cfm
Normal Mode Ventilation Flow Rates	
Filtered Makeup Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Makeup Flow Rate	2000 cfm
Emergency Mode Ventilation Flow Rates	
Filtered Makeup Flow Rate	0 cfm
Filtered Recirculation Flow Rate *	3600 cfm
Unfiltered Makeup Flow Rate	0 cfm
Filter Efficiencies	
Elemental	95%
Organic	95%
Particulate	99%
CR Radiation Monitor Setpoint	< 1.0E-04 μCi/cc for Xe-133
CR Radiation Monitor Location (R-23 & R-24)	HVAC Duct downstream of filter
CR HVAC Emergency Mode Actuation Delay **	5 minutes
Occupancy Factors	
0 - 24 hours	1.0
1 - 4 days	0.6
4 - 30 days	0.4

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* Minimum flow rate provides limiting case.

** This is the amount of time needed to align the CR HVAC from Normal Mode of operation to Emergency Mode and includes delay time to reach the high radiation setpoint.

*** Includes 290 CFM for boundary in leakage and 10 CFM for ingress and egress. 01406407

TABLE 14.5-3 SUMMARY OF 0-2 HOURS χ/Q RESULTS FOR CONTROL ROOM INTAKE FUEL HANDLING ACCIDENT

Source	**Receptor**	**0-2 Hour χ/Q (Sec/m^3)**
Common Area of Aux Bldg	121 Control Room Intake	6.71E-03
Common Area of Aux Bldg	122 Control Room Intake	4.79E-03
Spent Fuel Pool Vent Normal Exhaust Stack	121 Control Room Intake	1.09E-03
Spent Fuel Pool Vent Normal Exhaust Stack	122 Control Room Intake	2.82E-03
Unit 1 Equipment Hatch	121 Control Room Intake	1.73E-03
Unit 1 Equipment Hatch	122 Control Room Intake	4.79E-04
Unit 2 Equipment Hatch	121 Control Room Intake	6.04E-04
Unit 2 Equipment Hatch	122 Control Room Intake	3.11E-03

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TABLE 14.5-4 STEAMLINE RUPTURE - FULL POWER CORE RESPONSE TIME SEQUENCE OF EVENTS - LIMITING BREAK SIZE

Event	RSGs ($1.01\ ft^2$) Time (seconds)
Break Initiated with Reactor at Full Power	20.01
OPΔT Condition Reached in Loop 1	31.84
OPΔT Condition Reached in Loop 2	33.47
Rod Motion on OPΔT Reactor Trip	35.97
Minimum DNBR Reached	36.35
Maximum Core Heat Flux Reached	36.35
Turbine Trip following Reactor Trip	36.97

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TABLE 14.5-5 STEAMLINE RUPTURE - ZERO POWER CORE RESPONSE TIME SEQUENCE OF EVENTS - LIMITING (WITH OFFSITE POWER) ANALYSIS

Event	RSGs Time (seconds)
Reactor Trip Initiated	0.01
Break Initiated with Reactor at Zero Power	10.01
Maximum AFW Initiated	10.01
Faulted Loop Steam Flow Reaches Hi-Hi Setpoint	10.01
Faulted Loop Steam Pressure Reaches Lo-Lo Setpoint	10.64
SI Signal Actuation Due to Coincidence of Hi-Hi Steam Flow and Lo-Lo Steam Pressure	11.66
Steam Line Isolation (MSIV Closure) Due to SI Signal Actuation	16.66
SI Pump Reaches Full Speed	21.66
Feedwater Isolation (Main Feedwater Isolation Valve Closure) Due to SI Signal Actuation	62.16
Minimum DNBR Reached	~101.00
Maximum Core Heat Flux Reached	101.00

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TABLE 14.5-6 PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

Parameters	BOL-HZP	BOL-HFP	EOL-HZP	EOL-HFP	
Initial core power level, percent	0	100[(1)]	0	100[(1)]	01193868
Ejected rod worth, % Δk	0.77	0.38	0.954	0.30	
Delayed neutron fraction, %	0.49	0.49	0.47	0.47	
Doppler reactivity defect (absolute value), pcm	1000	1000	980	980	
Doppler feedback reactivity weighting	2.008	1.139	2.755	1.316	
Trip reactivity, percent Δk	1.0	4.0	1.0	4.0	
F_Q before rod ejection	N/A	2.5	N/A	2.5	
F_Q after rod ejection	11.0	4.2	18.42	5.69	
Number of operational pumps	1	2	1	2	

[1.] The full power cases considered a reactor power of 1683 MWt, which includes all applicable uncertainties. 01193868

TABLE 14.5-7 SEQUENCE OF EVENTS RCCA EJECTION

Beginning of Cycle - Hot Zero Power	Time (seconds)	
	400V+	422V+
RCCA Ejection Occurs	0.000	0.000
High Neutron Flux Setpoint (Low Setting) is Reached	0.211	0.211
Peak Nuclear Power Occurs	0.252	0.252
Rods Begin to Fall Into the Core	0.661	0.661
Peak Cladding Average Temperature Occurs	2.297	2.191
Peak Fuel Average Temperature Occurs	2.463	2.432

Beginning of Cycle - Hot Full Power	Time (seconds)	
	400V+	422V+
RCCA Ejection Occurs	0.000	0.000
High Neutron Flux Setpoint (High Setting) is Reached	0.030	0.030
Peak Nuclear Power Occurs	0.135	0.135
Rods Begin to Fall Into the Core	0.480	0.480
Peak Fuel Average Temperature Occurs	1.962	1.974
Peak Cladding Average Temperature Occurs	2.106	2.085

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TABLE 14.5-7 SEQUENCE OF EVENTS RCCA EJECTION

End of Cycle - Hot Zero Power	Time (seconds)	
	400V+	**422V+**
RCCA Ejection Occurs	0.000	0.000
High Neutron Flux Setpoint (Low Setting) is Reached	0.156	0.156
Peak Nuclear Power Occurs	0.183	0.183
Rods Begin to Fall Into the Core	0.606	0.606
Peak Cladding Average Temperature Occurs	1.723	1.574
Peak Fuel Average Temperature Occurs	1.956	1.865

End of Cycle - Hot Full Power	Time (seconds)	
	400V+	**422V+**
RCCA Ejection Occurs	0.000	0.000
High Neutron Flux Setpoint (High Setting) is Reached	0.035	0.035
Peak Nuclear Power Occurs	0.130	0.130
Rods Begin to Fall Into the Core	0.485	0.485
Peak Fuel Average Temperature Occurs	2.002	2.025
Peak Cladding Average Temperature Occurs	2.165	2.154

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TABLE 14.5-8 MSLB DOSE CONSEQUENCE ANALYSIS INPUT PARAMETERS

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Parameter	Value	
Rated Core Termal Power Assumed (licensed value)	1,852 MWt	
Plant Status Assumed:		
Offswite Power	Not Available	
Main Condensers	Not Available	
Nominal Reactor Coolant System (RCS) Volume	5,290 ft^3	
Unit 1 Steam Generator Liquid Mass (Framatome 56/19)	107,100 lbm	
Unit 2 Steam Generator Liquid Mass (Westinghouse 51) (subsequently replaced with SG similar to Unit 1)	107,420 lbm	
Primary & Secondary Coolant Parameter		
Primary Coolant iodine specific activity	0.5 µCi/gm DE I-131	
Primary Coolant non-iodine specific activity	580 µCi/gm DE Xe-133	
Maximum Primary Coolant iodine specific activity	30 µCi/gm DE I-131	
Concurrent iodine spiking factor:	500	
Duration of concurrent iodine spike	8 hrs	
Secondary Coolant iodine specific activity	0.1 µCi/gm DE I-131	
Fuel Damage as a Result of the Accident	No failed fuel	
Primary-to-Secondary (P-T-S) Leakage	150 gpd per SG	
Faulted Steam Generator Dryout Time	10 minutes	
Steam Generator Tube Uncovery and Flashing Duration		
Faulted Steam Generator	Event Duration	
Intact Steam Generator	0 to 2 hours	
Steam Generator Partition Coefficients	Iodine	Noble Gas
Faulted SG (P-T-S and Secondary Liquid)	1.0	1.0
Intact SG (0 – 2 hours, P-T-S and Secondary Liquid)	1.0	1.0
Intact SG (> 2 hours, P-T-S and Secondary Liquid)	10	1.0
Activity Release Duration for the Accident		
Termination of Release from Faulted Steam Generator	75 hours	
Termination of Release from Intact Steam Generator	45.5 hours	

TABLE 14.5-8 MSLB DOSE CONSEQUENCE ANALYSIS INPUT PARAMETERS

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Parameter	Value
Steam Releases from the Intact SG to Environment	
0 – 2 hours	226,414 lbm
2 – 8 hours	406,952 lbm
8 – 24 hours	796,899 lbm
24 – 45.5 hours	863,053 lbm
Control Room Atmospheric Dispersion Factors (X/Q) for Faulted SG Releases	
0 – 2 hours	4.79E-03 sec/m^3
2 – 8 hours	3.60E-03 sec/m^3
8 – 24 hours	1.60E-03 sec/m^3
24 – 96 hours	1.21E-03 sec/m^3
96 – 720 hours	9.55E-04 sec/m^3
Control Room Atmospheric Dispersion Factors (X/Q) for Intact SG Releases	
0 – 2 hours	3.07E-02 sec/m^3
2 – 8 hours	2.49E-02 sec/m^3
8 – 24 hours	1.12E-02 sec/m^3
24 – 96 hours	7.78E-03 sec/m^3
96 – 720 hours	6.17E-03 sec/m^3
Control Room (CR) Parameters	
CR Volume	61,315 ft^3
CR HVAC Emergency Mode Actuation Delay	5 minutes
Unfiltered In-leakage	300 cfm
Unfiltered Normal Mode Make-up Flow (< 5 minutes)	2,000 cfm
Filtered Recirculation Mode Flow (> 5 minutes)	3,600 cfm
Filter Efficiencies	
Elemental	95%
Organic	95%
Particulate	99%
CR Breathing Rate	3.5E-04 / m^3sec
CR Occupancy Factors	
0 – 24 hours	1.0
1 – 4 days	0.6
4 – 30 days	0.4

TABLE 14.5-8 MSLB DOSE CONSEQUENCE ANALYSIS INPUT PARAMETERS

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Parameter	Value
EAB Atmospheric Dispersion Factor (X/Q)	6.49E-04 sec/m^3
EAB Parameters	
EAB Breathing Rate	3.5E-04 m^3/sec
EAB Occupancy Factor	1.0 (any 2-hour period)
LPZ Atmospheric Dispersion Factors (X/Q)	
0-8 hours	1.77E-04 sec/m^3
8-24 hours	3.99E-05 sec/m^3
24-96 hours	7.12E-06 sec/m^3
96-720 hours	1.04E-06 sec/m^3
LPZ Parameters	
LPZ Breathing Rate	
0-8 hours	3.5E-04 m^3/sec
8-24 hours	1.8E-04 m^3/sec
24-720 hours	2.3E-04 m^3/sec
LPZ Occupancy Factor	1.0

TABLE 14.5-9, DELETED

TABLE 14.5-10, DELETED

TABLE 14.5-11, DELETED

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TABLE 14.5-12 STEAM GENERATOR TUBE RUPTURE ACCIDENT DOSE CONSEQUENCE PARAMETERS AND ASSUMPTIONS

Parameter	Value	
Rated Core Termal Power Assumed (licensed value)	1,852 MWt	
Plant Status Assumed:		
Offswite Power	Not Available	
Main Condensers	Not Available	
Nominal Reactor Coolant System (RCS) Volume	5,290 ft^3	
Unit 1 Steam Generator Liquid Mass (Framatome 56/19)	107,100 lbm	
Unit 2 Steam Generator Liquid Mass (Westinghouse 51) (subsequently replaced with SG similar to Unit 1)	107,420 lbm	
Primary & Secondary Coolant Parameter		
Primary Coolant iodine specific activity	0.5 μCi/gm DE I-131	
Primary Coolant non-iodine specific activity	580 μCi/gm DE Xe-133	
Maximum Primary Coolant iodine specific activity	30 μCi/gm DE I-131	
Concurrent iodine spiking factor:	335	
Duration of concurrent iodine spike	8 hrs	
Secondary Coolant iodine specific activity	0.1 μCi/gm DE I-131	
Fuel Damage as a Result of the Accident	No failed fuel	
Primary-to-Secondary (P-T-S) Leakage	150 gpd per SG	
Partition Coefficients	Iodine	Noble Gas
Ruptured Steam Generator (flashed P-T-S)	1.0	1.0
Ruptured Steam Generator (unflashed P-T-S)	100	1.0
Intact Steam Generator (P-T-S)	100	1.0
Steam Generators (Secondary Liquid)	100	1.0
Reactor Trip Time	172 seconds	
Activity Release Duration for the Accident	14 hours	
Primary Coolant Released From Ruptured Tube		
Total Ruptured Steam Generator Break Flow (0 – 0.5 hr)	140,000 lbm	
Pre-trip Break Flow with Flashing	14,600 lbm	
Post-trip Break Flow with Flashing	125,400 lbm	
Pre-trip Flashed Break Flow	2,630 lbm	
Post-trip Flashed Break Flow	15,050 lbm	

TABLE 14.5-12 STEAM GENERATOR TUBE RUPTURE ACCIDENT DOSE CONSEQUENCE PARAMETERS AND ASSUMPTIONS

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Parameter	Value
Steam Releases from the Ruptured SG via PORV to Environment	
0 – 0.5 hr	80,500 lbm
Steam Releases from the Intact SG via PORVs to Environment	
0 – 2 hr	237,100 lbm
2 – 8 hr	569,000 lbm
8 – 14 hr	416,000 lbm
Control Room Atmospheric Dispersion Factors (X/Q) for Intact SG Releases (Unit 2 Group 1 PORV to Unit 2 CR Vent Intake)	
0 – 2 hr	3.07E-02 sec/m^3
2 – 8 hr	2.49E-02 sec/m^3
8 – 24 hr	1.12E-02 sec/m^3
24 – 96 hr	7.78E-03 sec/m^3
Control Room Atmospheric Dispersion Factors (X/Q) for Intact SG Releases (Unit 2 Group 2 PORV to Unit 2 CR Vent Intake)	
0 – 2 hr	2.20E-03 sec/m^3
2 – 8 hr	1.81E-03 sec/m^3
8 – 24 hr	7.97E-04 sec/m^3
24 – 96 hr	5.16E-04 sec/m^3
Control Room (CR) Parameters	
CR Volume	61,315 ft^3
CR HVAC Emergency Mode Actuation Delay	5 minutes
Unfiltered In-leakage	300 cfm
Unfiltered Normal Mode Make-up Flow (< 5 minutes)	2,000 cfm
Filtered Recirculation Mode Flow (> 5 minutes)	3,600 cfm
Filter Efficiencies	
Elemental	95%
Organic	95%
Particulate	99%
CR Breathing Rate	3.5E-04 m^3/sec
CR Occupancy Factors	
0 – 24 hr	1.0
1 – 4 days	0.6
4 – 30 days	0.4

TABLE 14.5-12 STEAM GENERATOR TUBE RUPTURE ACCIDENT DOSE CONSEQUENCE PARAMETERS AND ASSUMPTIONS

Parameter	Value
EAB Atmospheric Dispersion Factor (X/Q)	6.49E-04 sec/m^3
EAB Parameters	
EAB Breathing Rate	3.5E-04 m^3/sec
EAB Occupancy Factor	1.0 (any 2-hour period)
LPZ Atmospheric Dispersion Factors (X/Q)	
0-8 hr	1.77E-04 sec/m^3
8-24 hr	3.99E-05 sec/m^3
24-96 hr	7.12E-06 sec/m^3
96-720 hr	1.04E-06 sec/m^3
LPZ Parameters	
LPZ Breathing Rate	
0-8 hr	3.5E-04 m^3/sec
8-24 hr	1.8E-04 m^3/sec
24-720 hr	2.3E-04 m^3/sec
LPZ Occupancy Factor	1.0

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TABLE 14.5-13 CONTROL ROD EJECTION ACCIDENT DOSE CONSEQUENCE PARAMETERS AND ASSUMPTIONS

Parameter	Value
Rated Core Thermal Power Assumed (licensed value)	1,852 MWt
Plant Status Assumed:	
Offsite Power	Not Available
Main Condensers	Not Available
Nominal Reactor Coolant System (RCS) Volume	5,290 ft^3
Unit 1 Steam Generator Liquid Mass (Framatome 56/19)	107,100 lbm
Unit 2 Steam Generator Liquid Mass (Westinghouse 51) (subsequently replaced with SG similar to Unit 1)	107,420 lbm
Containment Volume	1,320,000 ft^3
Shield Building Free Air Volume	374,000 ft^3
Primary & Secondary Coolant Parameter	
Primary Coolant iodine specific activity	0.5 μCi/gm DE I-131
Primary Coolant non-iodine specific activity	580 μCi/gm DE Xe-133
Secondary Coolant iodine specific activity	0.1 μCi/gm DE I-131
Fuel Damage as a Result of the Accident	
Clad Damage (percent fuel rods in DNB)	10%
Fuel Melt (percent core)	0.25%
Clad Damaged Fuel Gap Activity Released into Containment and Available for Release from Containment	
Iodine	10%
Noble Gases	10%
Clad Damaged Fuel Gap Activity Released into RCS and Available for Primary-to-Secondary Leakage	
Iodine	10%
Noble Gases	10%
Melted Fuel Activity Released into Containment and Available for Release from Containment	
Iodine	25%
Noble Gases	100%

TABLE 14.5-13 CONTROL ROD EJECTION ACCIDENT DOSE CONSEQUENCE PARAMETERS AND ASSUMPTIONS

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Parameter	Value	
Melted Fuel Activity Released into RCS and Available for Primary-to-Secondary Leakage		
Iodine	50%	
Noble Gases	100%	
Primary-to-Secondary (P-T-S) Leakage	150 gpd per SG	
Partition Coefficients	Iodine	Noble Gas
Steam Generators (P-T-S)	100	1.0
Steam Generators (Secondary Liquid)	100	1.0
Activity Release Duration for the Accident	45.5 hours	
Steam Releases from the Intact SG to Environment		
0 – 2 hours	226,414 lbm	
2 – 8 hours	406,952 lbm	
8 – 24 hours	796,899 lbm	
24 – 45.5 hours	863,053 lbm	
Control Room Atmospheric Dispersion Factors (X/Q) for Containment Releases		
0 – 2 hours	4.53E-03 sec/m^3	
2 – 8 hours	3.93E-03 sec/m^3	
8 – 24 hours	1.73E-03 sec/m^3	
24 – 96 hours	1.22E-03 sec/m^3	
96 – 720 hours	9.16E-04 sec/m^3	
Control Room Atmospheric Dispersion Factors (X/Q) for Intact SG Releases (Unit 2 Group 1 PORV to Unit 2 CR Vent Intake)		
0 – 2 hours	3.07E-02 sec/m^3	
2 – 8 hours	2.49E-02 sec/m^3	
8 – 24 hours	1.12E-02 sec/m^3	
24 – 96 hours	7.78E-03 sec/m^3	
Control Room Atmospheric Dispersion Factors (X/Q) for Intact SG Releases (Unit 2 Group 2 PORV to Unit 2 CR Vent Intake)		
0 – 2 hours	2.20E-02 sec/m^3	
2 – 8 hours	1.81E-02 sec/m^3	
8 – 24 hours	7.97E-02 sec/m^3	
24 – 96 hours	5.16E-02 sec/m^3	

TABLE 14.5-13 CONTROL ROD EJECTION ACCIDENT DOSE CONSEQUENCE PARAMETERS AND ASSUMPTIONS

01406407

Parameter	Value
Control Room (CR) Parameters	
CR Volume	61,315 ft^3
CR HVAC Emergency Mode Actuation Delay	5 minutes
Unfiltered In-leakage	250 cfm
Unfiltered Normal Mode Make-up Flow (< 5 minutes)	2,000 cfm
Filtered Recirculation Mode Flow (> 5 minutes)	3,600 cfm
Filter Efficiencies	
Elemental	95%
Organic	95%
Particulate	99%
CR Breathing Rate	3.5E-04 m^3/sec
CR Occupancy Factors	
0 – 24 hours	1.0
1 – 4 days	0.6
4 – 30 days	0.4
EAB Atmospheric Dispersion Factor (X/Q)	6.49E-04 sec/m^3
EAB Parameters	
EAB Breathing Rate (0 – 720 hours)	3.5E-04 m^3/sec
EAB Breathing Rate (0 – 720 hours)	1.0 (any 2-hour period)
LPZ Atmospheric Dispersion Factors (X/Q)	
0 – 8 hours	1.77E-04 sec/m^3
8 – 24 hours	3.99E-05 sec/m^3
24 – 96 hours	7.12E-06 sec/m^3
96 – 720 hours	1.04E-06 sec/m^3
LPZ Parameters	
LPZ Breathing Rate	
0 – 8 hours	3.5E-04 m^3/sec
8 – 24 hours	1.8E-04 m^3/sec
24 – 720 hours	2.3E-04 m^3/sec
LPZ Occupancy Factor	1.0

TABLE 14.6-1 MAJOR PLANT PARAMETER ASSUMPTIONS USED IN THE BELOCA ANALYSIS FOR PRAIRIE ISLAND UNITS 1 AND 2

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PARAMETER	VALUE
Plant Physical Description	
• SG Tube Plugging	≤ 10%
Plant Initial Operating Conditions	
• Reactor Power	≤ 100% of 1683 MW_t
• Peaking Factors	$F_Q \leq 2.5$ $F_{\Delta H} \leq 1.77$
• Axial Power Distribution	See Figure 14.6-13
Fluid Conditions	
• T_{AVG}	T_{AVG} = 560.0 +/- 4°F
• Pressurizer Pressure	2190 psia ≤ P_{RCS} ≤ 2310 psia
• Reactor Coolant Flow	≥ 178,000 gpm
• Accumulator Temperature	70°F ≤ T_{ACC} ≤ 120°F
• Accumulator Pressure	699.7 psia ≤ P_{ACC} ≤ 809.7 psia
• Accumulator Water Volume	1245 ft^3 ≤ V_{ACC} ≤ 1295 ft^3
• Accumulator Boron Concentration	≥ 1900 ppm
Accident Boundary Conditions	
• Single Failure Assumptions	Loss of one ECCS train
• Safety Injection Flow	Minimum
• Safety Injection Temperature	60°F ≤ T_{SI} ≤ 120°F
• Low Head Safety Injection Initiation Delay Time	≤ 15 sec (with offsite power) ≤ 28 sec (without offsite power)
• High Head Safety Injection Initiation Delay Time	≤ 10 sec (with offsite power) ≤ 28 sec (without offsite power)
• Containment Pressure	Bounded (minimum)

TABLE 14.6-2 PRAIRIE ISLAND UNITS 1 AND 2 LARGE-BREAK LOCA HIGH-HEAD SAFETY INJECTION (HHSI) DELIVERED FLOW VERSUS PRESSURE

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Prairie Island Units 1 and 2 Large-Break LOCA High-Head Safety Injection (HHSI)	
Pressure (psia)	**Flow (gpm)**
14.7	276.5
114.7	259.6
214.7	213.7
314.7	165.0
414.7	113.1
514.7	0

TABLE 14.6-3 PRAIRIE ISLAND UNITS 1 AND 2 LARGE-BREAK-LOCA LOW-HEAD SAFETY INJECTION (LHSI) FLOW VERSUS PRESSURE

01129648

Prairie Island Units 1 and 2 Large-Break LOCA Low-Head Safety Injection (LHSI)	
Pressure (psia)	**Flow (gpm)**
14.7	1605.4
34.7	1473.1
54.7	1330.0
74.7	1165.1
94.7	972.7
114.7	741.2
134.7	404.9
145.6*	0*

Note:

*Actual shutoff head point for UPI LHSI.

TABLE 14.6-4 LARGE-BREAK LOCA CONTAINMENT DATA USED FOR CALCULATION OF CONTAINMENT PRESSURE

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Maximum Net Free Volume	1,370,000 ft^3
Initial Conditions	
Minimum pressure	14.2 psia
Minimum temperature	70.0°F
Minimum refueling water storage tank (RWST) temperature	60.0°F
Minimum service water temperature	32.0°F
Minimum temperature outside containment	-35.0°F
Minimum initial spray temperature	60.0°F
Spray System	
Maximum number of spray pumps operating	2
Maximum post-accident spray system initiation delay	11.8 sec*
Maximum spray system flow	3200 gal/min
Containment Fan Coolers	
Minimum post-accident initiation fan coolers	0 sec*
Maximum number of fan coolers operating	4
Maximum number of containment purge lines open at onset of transient	2
Maximum containment purge valve closure time	10 sec
Maximum containment purge valve inside diameter	17.625 in.
Structural Heat Sinks	See Table 14.6-5
Note: *Bounds both LOOP and OPA (Offsite Power Available)	

TABLE 14.6-5 PRAIRIE ISLAND UNITS 1 AND 2 LARGE-BREAK LOCA STRUCTURAL HEAT SINK TABLE

Prairie Island Units 1 and 2 Large-Break LOCA

Wall Description	Material of Each Layer	Thickness of Layer (ft)	Surface Area (ft^2)	Symmetric or Insulated
Containment Cylinder	Carbon Steel	0.125*	41,300	(1)
Containment Dome	Carbon Steel	0.063*	17,340	(1)
Reactor Vessel Cavity Liner	Carbon Steel	0.016*	1,300	Insulated (2)
	Concrete	1.0		
Refueling Canal	Stainless Steel	0.016	6,600	Insulated (2)
	Concrete	1.0		
Exposed Piping	Carbon Steel	0.031*	8,095	Insulated (3)
	Stainless Steel	0.018	130	Insulated (3)
	Copper	0.005	50	Insulated (3)
Steel Structures	Carbon Steel	0.021*	9,405	Symmetric
		0.042*	31,820	Symmetric
		0.063*	53,375	Symmetric
		0.125*	3,690	Symmetric
		0.167*	925	Symmetric
Handrails and Ladders	Carbon Steel	0.012*	2,315	Symmetric
Grating	Carbon Steel	0.008	17,075	Symmetric
Cable Trays and Conduit	Carbon Steel	0.008	23,765	Insulated (2)
Ductwork	Carbon Steel	0.009	31,495	Symmetric
Accumulators (Two)	Carbon Steel	0.120*	3,600	Insulated (4)
Ventilation Equipment	Carbon Steel	0.016*	14,565	Symmetric
	Copper	0.004	4,780	Symmetric
Heavy Walls	Concrete	1.0	40,800	25,800-Symmetric 15,000-Insulated (2)
Heavy Floors	Concrete	0.5	25,070	19,240-Symmetric 5,830-Insulated (2)
Light Floors	Concrete	0.25	7,570	Symmetric

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Notes:

* These items assume a minimum paint thickness of 0.007 inches in the analysis (paint thickness is minimized since paint is considered an insulator).

"Symmetric" structures are exposed to containment atmosphere on both sides, while structures indicated as "Insulated" are exposed to containment atmosphere on one side only. The surface area of symmetric structures includes the surface area for both sides. Insulated structures may be exposed to other materials such as water or air on the side not exposed to containment atmosphere, as discussed in the following notes: 01129648

(1) Side not exposed to containment atmosphere is assumed in contact with outside air at a minimum temperature (see Table 14.6-4).

(2) Side not exposed to containment atmosphere is assumed in contact with air 70°F.

(3) Side not exposed to containment atmosphere is assumed in contact with water 60°F and a flow of 10 ft/s.

(4) Side not exposed to containment atmosphere is assumed in contact with water 70°F and stagnant conditions.

TABLE 14.6-6 PRAIRIE ISLAND UNIT 1 AND 2 BEST ESTIMATE LARGE-BREAK LOCA RESULTS 01386642

10 CFR 50.46 Requirement	Value	Criteria
95/95 PCT (Peak Cladding Temperature, °F)	1,992	<2200
95/95 LMO (Local Maximum Oxidation, %)	0.62	<17
95/95 CWO (Core Wide Oxidation, %)	0.014	<1

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TABLE 14.6-6a, DELETED
TABLE 14.6-6b, DELETED

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TABLE 14.6-7 PRAIRIE ISLAND UNIT 1 AND 2 BEST ESTIMATE LARGE-BREAK SEQUENCE OF EVENTS FOR THE LIMITING PCT CASE

Event	Time (sec)
Start of Transient	0.0
Safety Injection Signal	3.8
PCT Occurs	7.1
Accumulator Injection Begins	9.0
High Head Safety Injection Begins	13.8
Low Head Safety Injection Begins	18.8
End of Blowdown	24.0
Bottom of Core Recovery	34.0
Accumulator Empty	~35.0
End of Transient	450.0

TABLE 14.6-7a, DELETED
TABLE 14.6-7b, DELETED

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TABLE 14.7-1
PARAMETERS USED IN THE SMALL BREAK LOCA ANALYSES

Parameter	**Unit 1 & 2**	
Analyzed Core Power(1)	1683 MWt	01386642
Total Core Peaking Factor, F_Q	2.5	
Channel Enthalpy Rise Factor, F_{DH}	1.77	
Hot Assembly Power Factor, P_{HA}	1.71	
Axial Power Shape	Figure 14.7-2	
Fuel Assembly Array	14 x 14 422Vantage+ with ZIRLO®(2)	01386642
Minimum Accumulator Cover Gas Pressure (including uncertainties)	699.7 psia	
Water/Gas Temperature	120°F	
Nominal Accumulator Water Volume	1270 ft^3	01365252
Thermal Design Flow	89,000 gpm	
Pumped Safety Injection Flow	Figure 14.7-3a Figure 14.7-3b Figure 14.7-3c	
Steam Generator Tube Plugging	10%	01386642
Reactor Trip Signal	1700 psia	
Safety Injection Signal	1700 psia	01365252
Rod Drop Time	2.4 seconds	
Reactor Trip Signal Delay Time	2.0 seconds	
Auxiliary Feedwater Flow Rate	90 gpm/SG	01365252 01386642

Notes:

1. This value includes power measurement uncertainty. Reactor coolant pump heat is not modeled in LOCA analysis.
2. The analysis bounds 14 x 14 422V+ fuel and a mixed core of the two fuel types.

TABLE 14.7-1a, DELETED

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TABLE 14.7-2
SMALL BREAK LOCA TIME SEQUENCE OF EVENTS

EVENT (sec)	1.5-inch	2-inch	3-inch	4-inch	6-inch	8-inch	10.126-inch
Transient Initiated	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Reactor Trip Signal	57.6	29.8	13.2	8.1	5.7	5.2	5
Safety Injection Signal	57.6	29.8	13.2	8.1	5.7	5.2	5
Safety Injection Begins[(1)]	84.6	56.8	40.2	35.1	32.7	32.2	32
Loop Seal Clearing Occurs[(2)]	1191	653	265	158	27	15	10
Top of Core Uncovered	N/A[(3)]	1401	574	393	N/A[(3)]	N/A[(3)]	N/A[(3)]
Accumulator Injection Begins	6704	2301	705	362	164	94	53
Top of Core Recovered	N/A[(3)]	2548	889	429	N/A[(3)]	N/A[(3)]	N/A[(3)]
RWST Low Level	N/A[(4)]	5288.0	2304.3	2283.8	1668.7	1523.4	1448.9

Notes:

1. Safety Injection (SI) begins 27.0 seconds (SI delay time) after the SI signal is generated.
2. Loop seal clearing is considered to occur when the broken loop (BL) loop seal vapor flow rate is sustained above 1 lbm/s.
3. There is no core uncovery for the 1.5-inch break case and only brief core uncovery for the 6-inch, 8-inch and 10.126-inch break cases.
4. The RWST low level is not reached for this break size.

TABLE 14.7-2a, DELETED

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TABLE 14.7-3
SMALL BREAK LOCA FUEL ROD HEATUP RESULTS

<table>
<tr><th>RESULTS</th><th>1.5-inch</th><th>2-inch</th><th>3-inch</th><th>4-inch</th><th>6-inch</th><th>8-inch</th><th>10.126-inch</th></tr>
<tr><td>PCT, °F</td><td rowspan="8">N/A[2]</td><td>954.2</td><td>958.9</td><td>603.7</td><td rowspan="8">N/A[2]</td><td rowspan="8">N/A[2]</td><td rowspan="8">N/A[2]</td></tr>
<tr><td>PCT Time, sec</td><td>1922.9</td><td>787.1</td><td>414.4</td></tr>
<tr><td>PCT Elevation, ft</td><td>11.25</td><td>10.75</td><td>11.00</td></tr>
<tr><td>Burst Time[1], sec</td><td>N/A</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Burst Elevation[1], ft</td><td>N/A</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Maximum ZrO_2, %</td><td>0.01</td><td>0.01</td><td>0.0</td></tr>
<tr><td>Maximum ZrO_2 Elevation, ft</td><td>11.25</td><td>11.0</td><td>12.0</td></tr>
<tr><td>Average ZrO_2, %</td><td>0.0</td><td>0.0</td><td>0.0</td></tr>
</table>

Notes:

1. Neither the hot rod nor the hot assembly average rod burst during the SBLOCTA calculations.
2. The core either does not uncover or only uncovers for a very short time; therefore, SBLOCTA calculations are not warranted for these break sizes.

TABLE 14.7-3a, DELETED

TABLE 14.7-4, DELETED

TABLE 14.7-5, DELETED

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TABLE 14.8-1 AMSAC/DSS EVENT APPROACH

Condition II Event	Approach
Uncontrolled RCCA Withdrawal from a Subcritical Condition	The only operation mode considered for the AMSAC/DSS events is full-power operation. Therefore, this event is not analyzed.
Uncontrolled RCCA Bank Withdrawal at Power	Analyzed. No reactor trip is credited from either the normal RPS or the AMSAC/DSS.
Dropped Rod/Control Rod Misalignment	Bounded by RWAP and uncontrolled boron dilution since the reactivity insertion from those events is larger.
Uncontrolled Boron Dilution	Analyzed. No reactor trip is credited from either the normal RPS or the AMSAC/DSS.
Startup of an Inactive Loop	Reactor trip not required in Westinghouse methodology.
Feedwater System Malfunction	Reactor trip not required in Westinghouse analysis methodology.
Excessive Load Increase	Reactor trip not required in Westinghouse analysis methodology.
Loss of External Load/Turbine Trip (LOL/TT)	Analyzed. Credit SG wide range level DSS reactor trip.(1)
Loss of Normal Feedwater Flow	Analyzed. DNBR results bounded by LOL/TT event. Credit SG wide range DSS reactor trip.
Loss of AC Power to the Station Auxiliaries	Analyzed. DNBR results bounded by LOL/TT event. Credit SG wide range DSS reactor trip.
Loss of Reactor Coolant Flow - 1/2 Pump Trip	Analyzed. Credit RCP breaker DSS reactor trip.
Isolation of Main Condenser	Bounded by LOL/TT event.(1)

(1) All normal feedwater is assumed to be lost coincident with the Loss of Load/Turbine Trip, and the steam dump system is assumed to be unavailable. Therefore, the LOL/TT analysis bounds the Isolation of Main Condenser event.

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TABLE 14.8-9 AMSAC/DSS ANALYSIS RESULTS

	Maximum RCS pressure (psia)	MDNBR
Loss of Normal Feedwater	2380	1.86
Loss of External Load/Turbine Trip	2446	1.80
1 of 2 Reactor Coolant Pump Trip	2391	1.39
Loss of AC	2367	1.86
RCCA Bank Withdrawal at Power	2314	1.28
Uncontrolled Boron Dilution	2306	1.31

TABLE 14.9-1, DELETED

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TABLE 14.9-2 OFFSITE AND CONTROL ROOM DOSE FOR DESIGN BASIS LOSS-OF-COOLANT ACCIDENT

Location	Acceptance Criteria (rem)	TEDE (rem)
Exclusion Area Boundary	25	2.58
Low Population Zone	25	2.42
Control Room	5.0	4.52

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TABLE 14.9-3 DELETED

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TABLE 14.9-4 DELETED

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TABLE 14.9-5 ANALYSIS ASSUMPTIONS AND KEY PARAMETER VALUES FOR DESIGN BASIS LOSS-OF-COOLANT ACCIDENT

Parameter	**Value**
Rated Core Termal Power Assumed (licensed value)	1,852 MWt
Containment Free Air Volume	1,320,000 ft^3
Shield Building Free Air Volume	374,000 ft^3
<u>Total Primary Containment Leak Rate</u>	
0 – 24 hours	0.15 w%/day
After 24 hours	0.075 w%/day
<u>Auxiliary Building Special Ventilation Zone (ABSVZ) Leak Rate</u>	
0 – 24 hours	0.06 w%/day
After 24 hours	0.03 w%/day
Drawdown Time	20 minutes
<u>Shield Building Leak Rate</u>	
0 – 24 hours	0.084 w%/day
After 24 hours	0.042 w%/day
Drawdown Time	12 minutes
<u>Bypass Leak Rate</u>	
0 – 24 hours	0.006 w%/day
After 24 hours	0.003 w%/day
<u>Shield Building (SB) Ventilation Parameters</u>	
SB Filtered Recirculation Mode Flow Start Time	22 minutes
SB Filtered Recirculation Mode Flow Rate	3,600 cfm
SB Filter Efficiencies	
Elemental	0%
Organic	0%
Particulate	99%
SB Exhaust Rate to Environment	2,000 cfm
<u>ABVS Ventilation Parameters</u>	
ABSVZ Filter Efficiencies	
Elemental	80%
Organic	80%
Particulate	99%

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TABLE 14.9-5 ANALYSIS ASSUMPTIONS AND KEY PARAMETER VALUES FOR DESIGN BASIS LOSS-OF-COOLANT ACCIDENT

Parameter	Value
Minimum Containment Sump Water Volume	230,000 gallons
Engineered Safety Features (ESF) Leakage Rate	
Allowable	2 gallons/hour
As modeled in dose analysis	4 gallons/hour
ESF Leakage Initiation Time	0 minutes
ESF Leakage Iodine Flashing Factors	
0 – 5.56 hours	4.27 percent
5.56 – 8.33 hours	1.87 percent
after 8.33 hours	3 percent
Refueling Water Storage Tank (RWST) Capacity	275,000 gallons
RWST Backleakage Leak Rate	
Allowable	5 gallons/hour
As modeled in dose analysis	10 gallons/hour
Minimum RWST Leakage Transit Time	35.0 hours
RWST Release Iodine Flashing Factors	
0 – 5.56 hours	4.27 percent
5.56 – 8.33 hours	1.87 percent
after 8.33 hours	3 percent

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TABLE 14.9-5 ANALYSIS ASSUMPTIONS AND KEY PARAMETER VALUES FOR DESIGN BASIS LOSS-OF-COOLANT ACCIDENT

Parameter	Value
Control Room Atmospheric Dispersion Factors (X/Q) for Containment and ESF Leakage Releases	
0 – 2 hours	4.53E-03 sec/m^3
2 – 8 hours	3.93E-03 sec/m^3
8 – 24 hours	1.73E-03 sec/m^3
24 – 96 hours	1.22E-03 sec/m^3
96 – 720 hours	9.16E-04 sec/m^3
Control Room Atmospheric Dispersion Factors (X/Q) for RWST Releases	
0 – 2 hours	2.53E-02 sec/m^3
2 – 8 hours	2.13E-02 sec/m^3
8 – 24 hours	9.65E-03 sec/m^3
24 – 96 hours	7.14E-03 sec/m^3
96 – 720 hours	6.15E-03 sec/m^3
Control Room (CR) Parameters	
CR Volume	61,315 ft^3
CR HVAC Emergency Mode Actuation Delay	5 minutes
Unfiltered In-leakage	250 cfm
Unfiltered Normal Mode Make-up Flow (< 5 minutes)	2,000 cfm
Filtered Recirculation Mode Flow (> 5 minutes)	3,600 cfm
Filter Efficiencies	
Elemental	95%
Organic	95%
Particulate	99%
CR Breathing Rate	3.5E-04 m^3/sec
CR Occupancy Factors	
0 – 24 hours	1.0
1 – 4 days	0.6
4 – 30 days	0.4
EAB Atmospheric Dispersion Factor (X/Q)	6.49E-04 sec/m^3
EAB Parameters	
EAB Breathing Rate	3.5E-04 m^3/sec
EAB Occupancy Factor	1.0 (any 2-hour period)

TABLE 14.9-5 ANALYSIS ASSUMPTIONS AND KEY PARAMETER VALUES FOR DESIGN BASIS LOSS-OF-COOLANT ACCIDENT

Parameter	Value
LPZ Atmospheric Dispersion Factors (X/Q)	
0 – 8 hours	1.77E-04 sec/m^3
8 – 24 hours	3.99E-05 sec/m^3
24 – 96 hours	7.12E-06 sec/m^3
96 – 720 hours	1.04E-06 sec/m^3
LPZ Parameters	
LPZ Breathing Rate	
0 – 8 hours	3.5E-04 m^3/sec
8 – 24 hours	1.8E-04 m^3/sec
24 – 720 hours	2.3E-04 m^3/sec
LPZ Occupancy Factor	1.0

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TABLE 14.9-6 DELETED,
SUPERSEDED BY NEW TABLE 14.9-5

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TABLE 14.10-1

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TABLE 14.10-2 HHSI Flow for 1 HHSI Pump with the Faulted Loop Spilling to RCS Pressure (Breaks less than 5.187 inches)

RCS Pressure psia	Spilled Flow gpm	Injected Flow gpm
14.7	292.3	276.4
114.7	285.0	269.5
214.7	277.5	262.5
314.7	270.0	255.4
414.7	262.4	248.2
514.7	254.6	240.9
614.7	246.7	233.4
714.7	238.2	225.3
814.7	229.5	217.1
914.7	220.6	208.7
1,014.7	211.6	200.2
1,114.7	202.5	191.5
1,214.7	192.1	181.6
1,314.7	181.1	171.2
1,414.7	169.7	160.6
1,514.7	158.1	149.5
1,614.7	144.0	136.2
1,714.7	127.6	120.7
1,814.7	110.0	104.0
1,914.7	86.2	81.6
2,014.7	55.9	52.9
2,114.7	0.0	0.0

Note: These SI Flows have been reduced an additional 11% below the calculated degraded SI flows and do not necessarily represent the SI flows assumed in other analyses.

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TABLE 14.10-3 HHSI Flow for 1 HHSI Pump with the Faulted Loop Spilling to Containment Pressure (0 psig) (Breaks greater than 5.187 inches)

RCS Pressure psia	Spilled Flow gpm	Injected Flow gpm
14.7	292.3	276.5
114.7	380.1	259.6
214.7	418.0	213.7
314.7	457.5	165.1
414.7	498.9	113.1
514.7	542.9	56.8
614.7	586.5	0.0
2,314.7	586.5	0.0

Note: These SI Flows have been reduced an additional 11% below the calculated degraded SI flows and do not necessarily represent the SI flows assumed in other analyses.

TABLE 14.10-4 RHR Flows for 1 RHR Pump Injecting from RWST (No Spilling Flows)

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RCS Pressure psia	Injected Flow gpm
14.7	1,605.4
34.7	1,473.1
54.7	1,330.0
74.7	1,165.1
94.7	972.7
114.7	741.2
134.7	404.9
145.6	0.0

Note: These RHR flow rates are just for the injection phase of the SBLOCA Long Term Core Cooling analysis. No RHR flow was credited following transfer to recirculation for the SBLOCA scenarios.

Illustration of OTΔT and OPΔT Protection

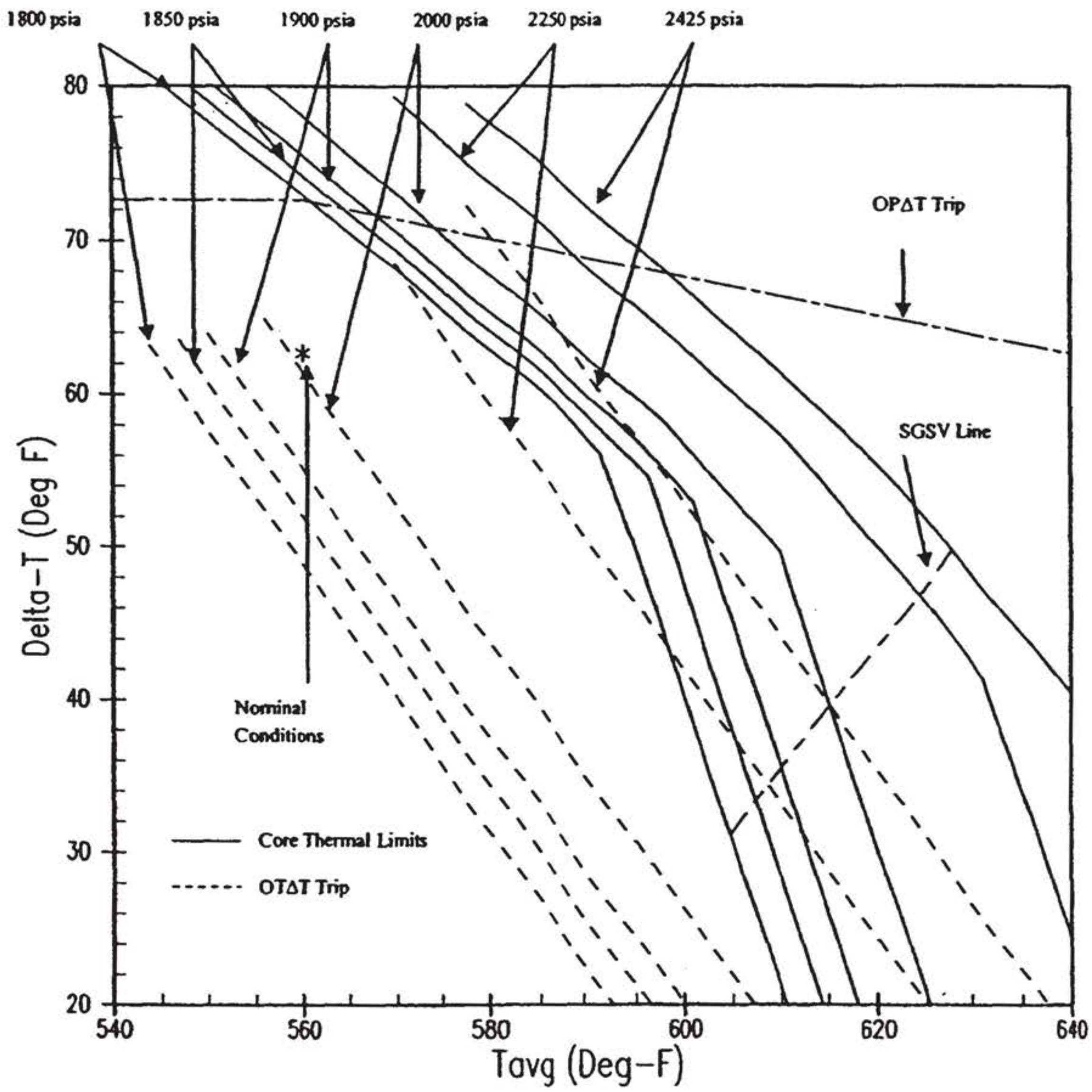


Figure 14.3-1

Rev. 27

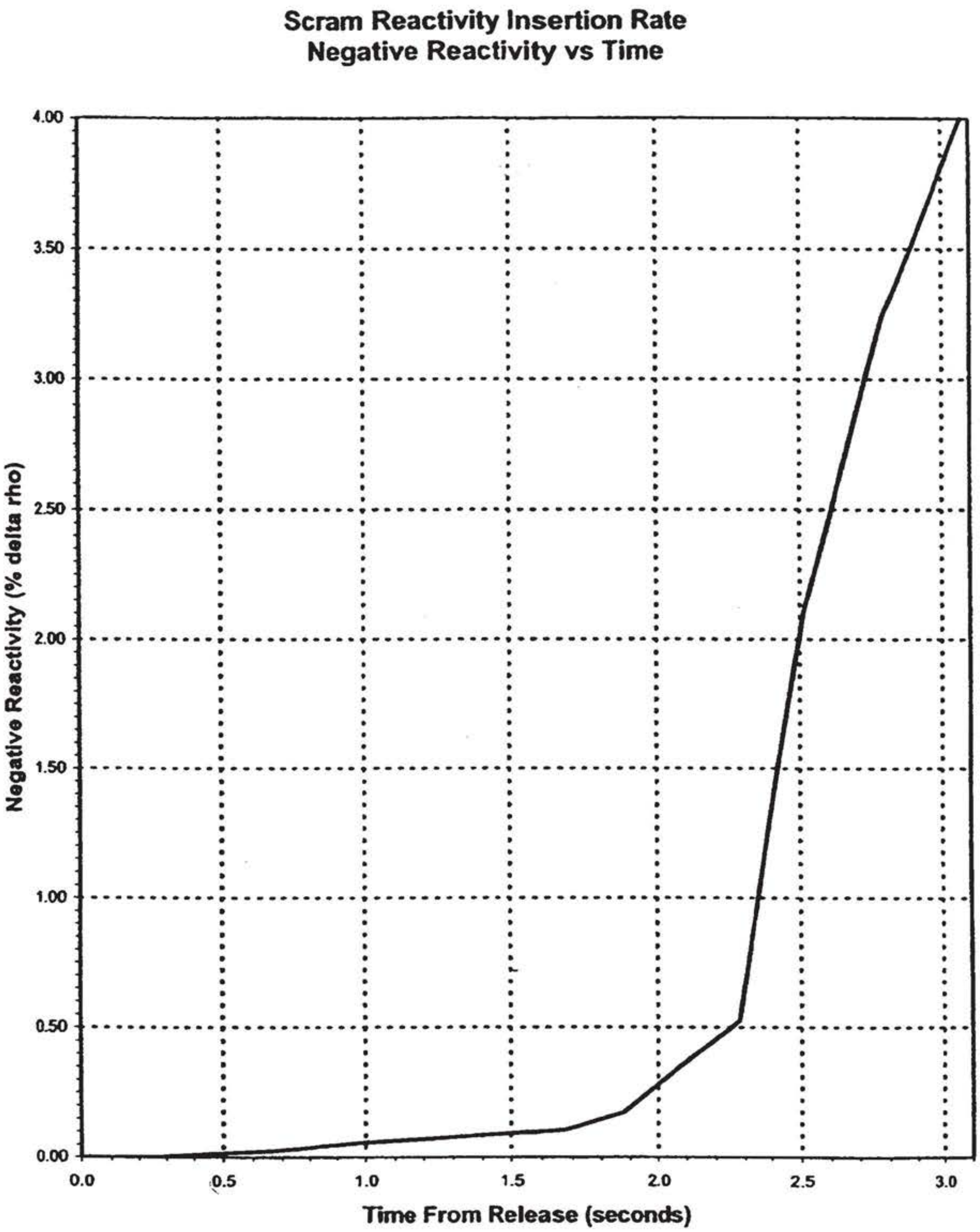


Figure 14.3-2

Rev. 27

Core Thermal Limits
Vessel Average Temperature vs. Fraction of Rated Thermal Power

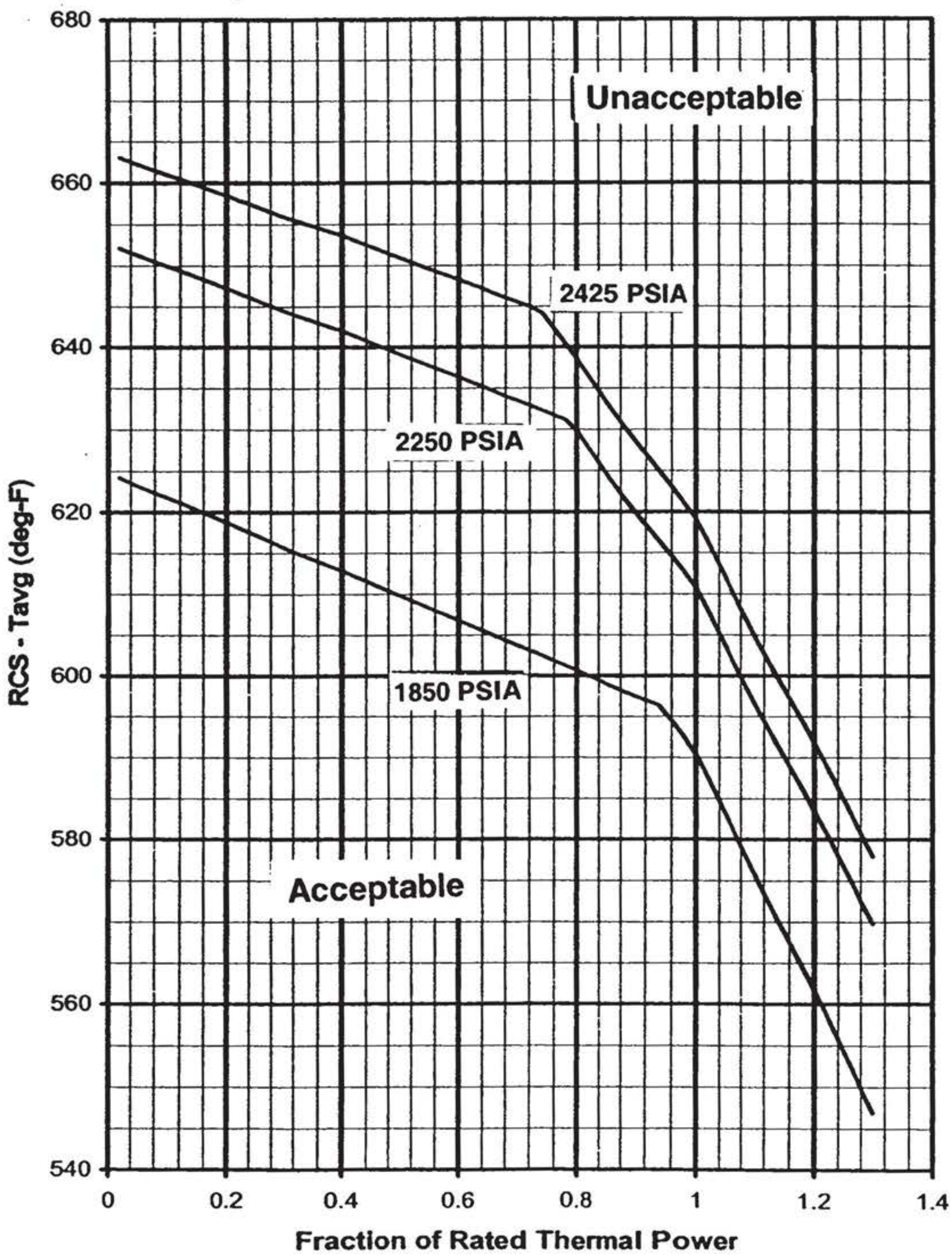
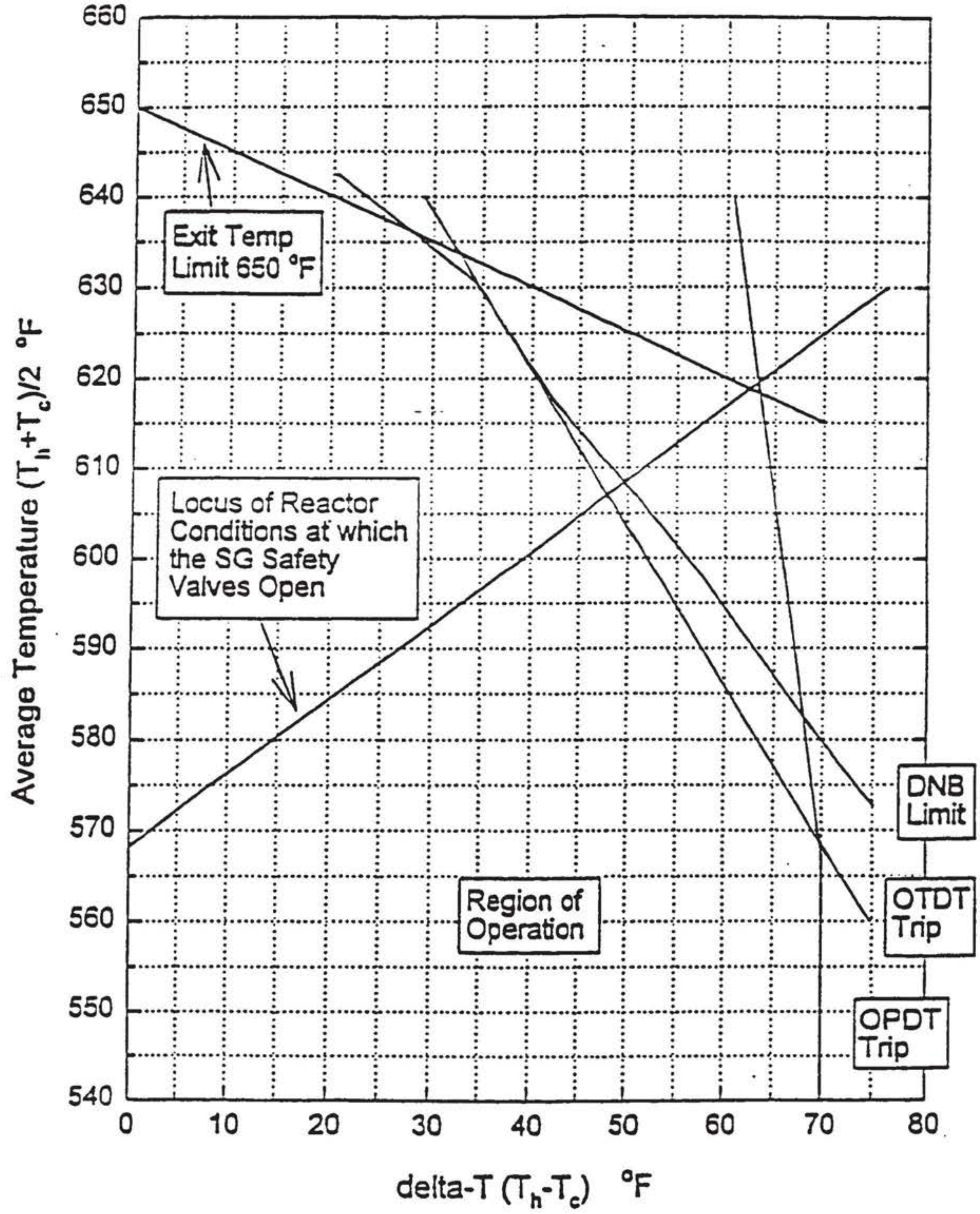


Figure 14.3-3

Rev. 27



REACTOR CORE LIMIT CURVES AT 2235 PSIG WITH DELTA - T TRIPS

DWN T. MILLER	DATE 6-23-99	NORTHERN STATES POWER COMPANY	SCALE: NONE	
CHECKED	CAD FILE U14304.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	FIGURE 14.3-4	REV. 18

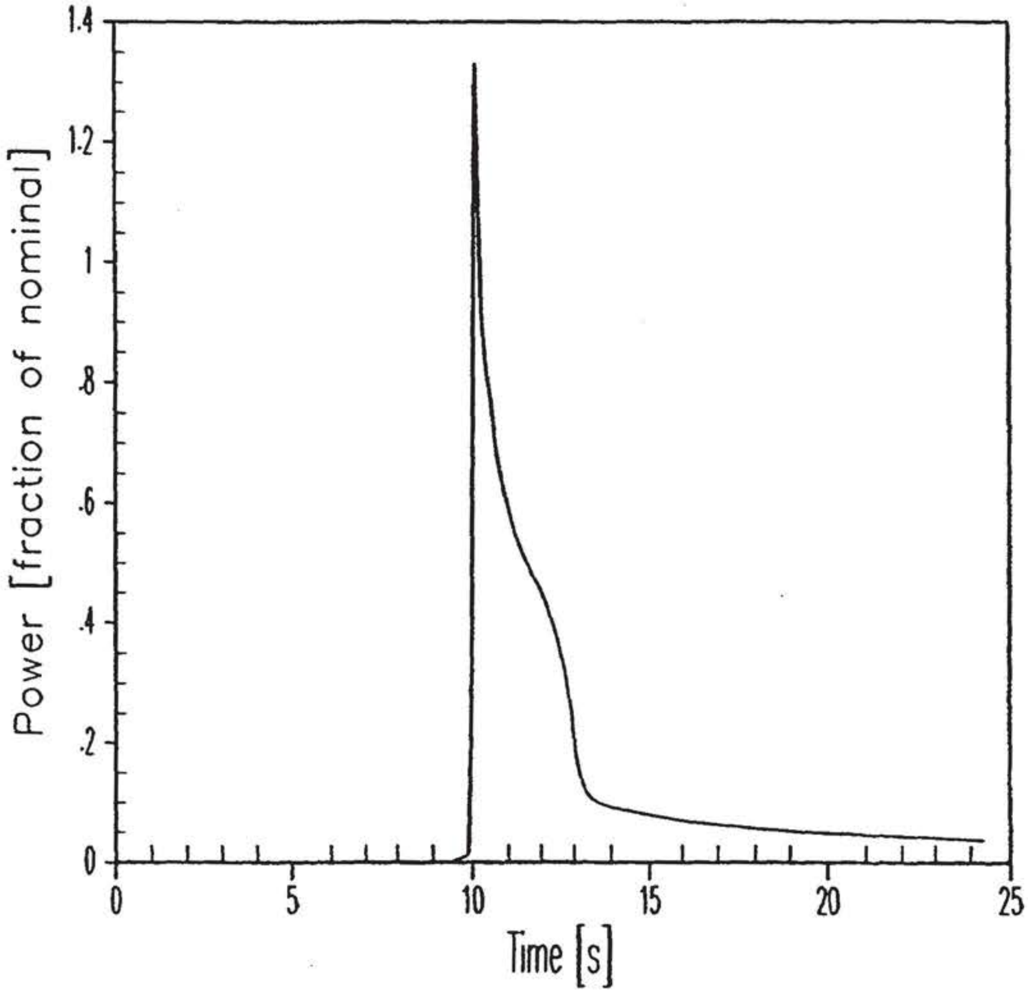
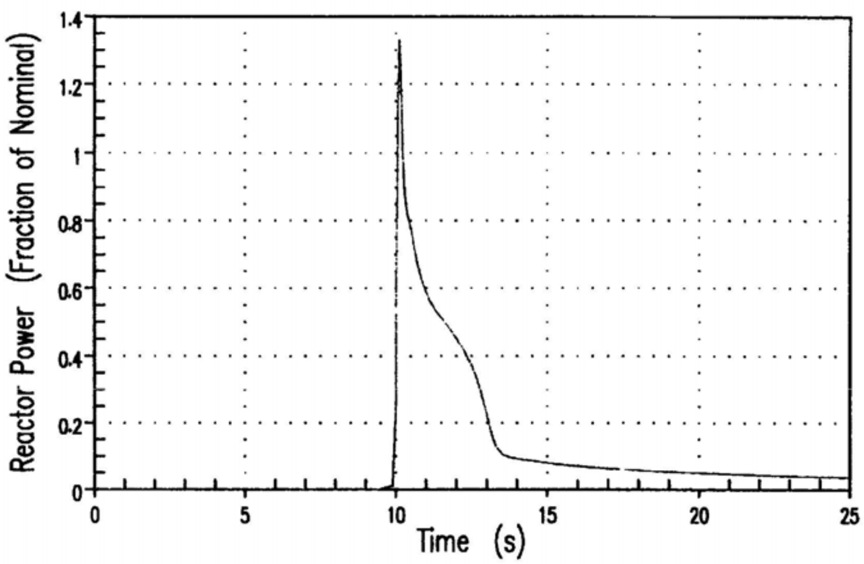


Figure 14.4-1 **Uncontrolled RCCA Withdrawal from a Subcritical Condition – Reactor Power Versus Time** **Rev. 27**



UNCONTROLLED RCCA WITHDRAWAL FROM A SUBCRITICAL CONDITION – REACTOR POWER VERSUS TIME

DWN KJF	DATE 1-28-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14401A.DGN		*FIGURE 14.4-1A REV. 31*

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FIGURE 14.4-1A

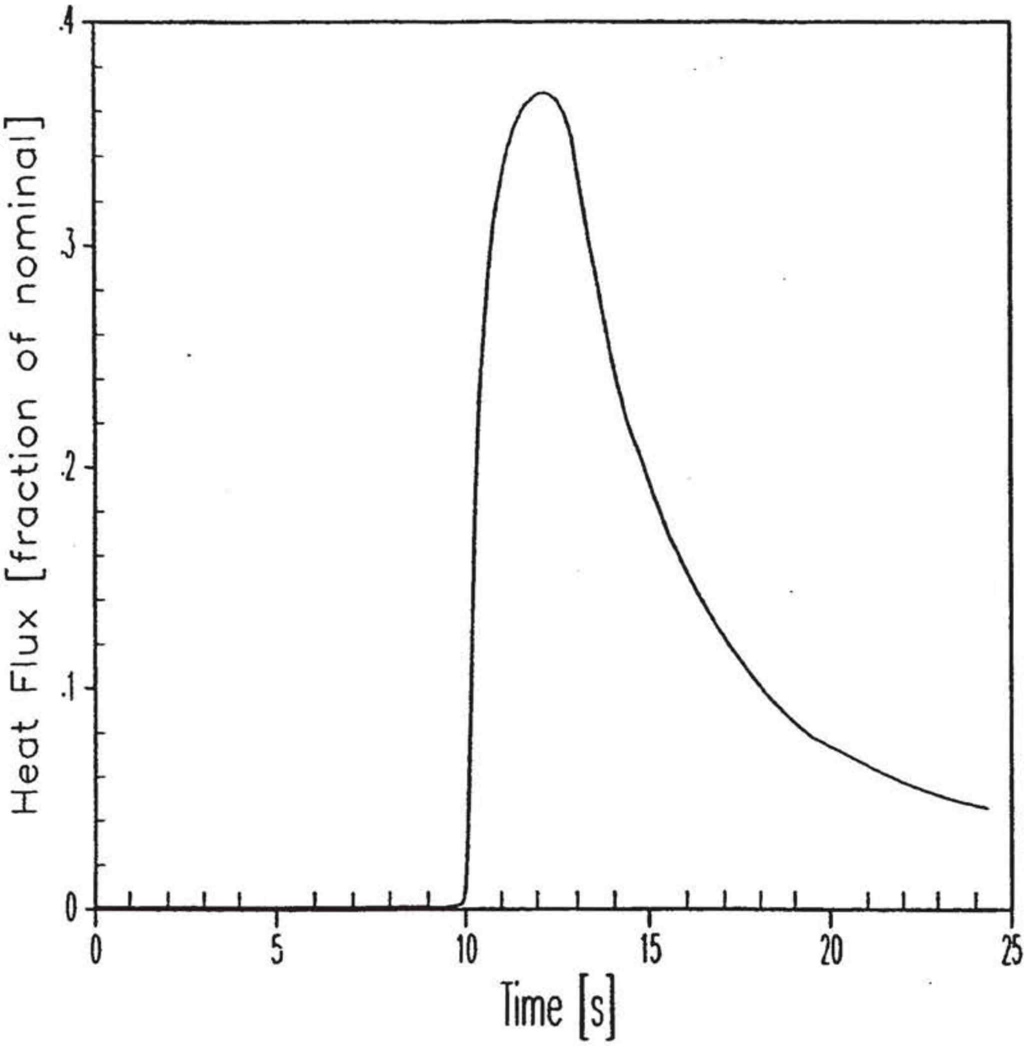
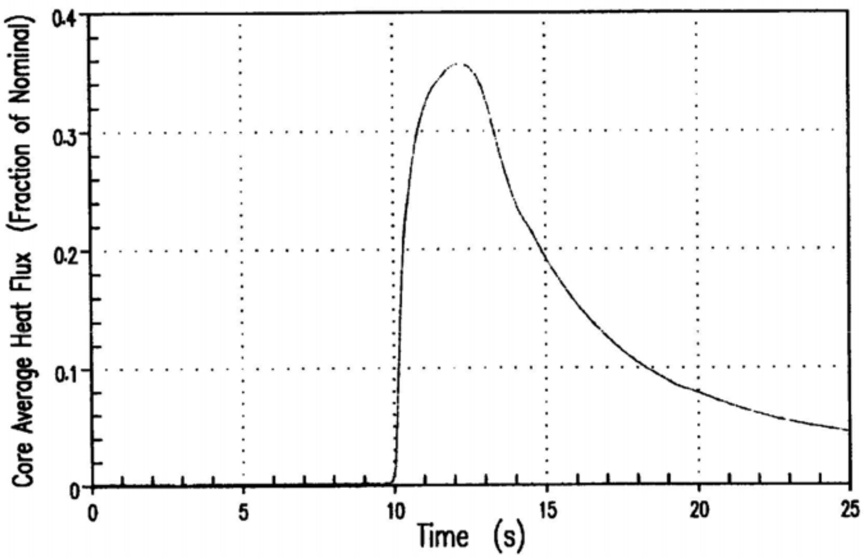


Figure 14.4-2 **Uncontrolled RCCA Withdrawal from a Subcritical Condition – Heat Flux Versus Time** **Rev. 27**



UNCONTROLLED RCCA WITHDRAWAL FROM A SUBCRITICAL CONDITION – HEAT FLUX VERSUS TIME

DWN KJF	DATE 1-28-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14402A.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-2A REV. 31*

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FIGURE 14.4-2A

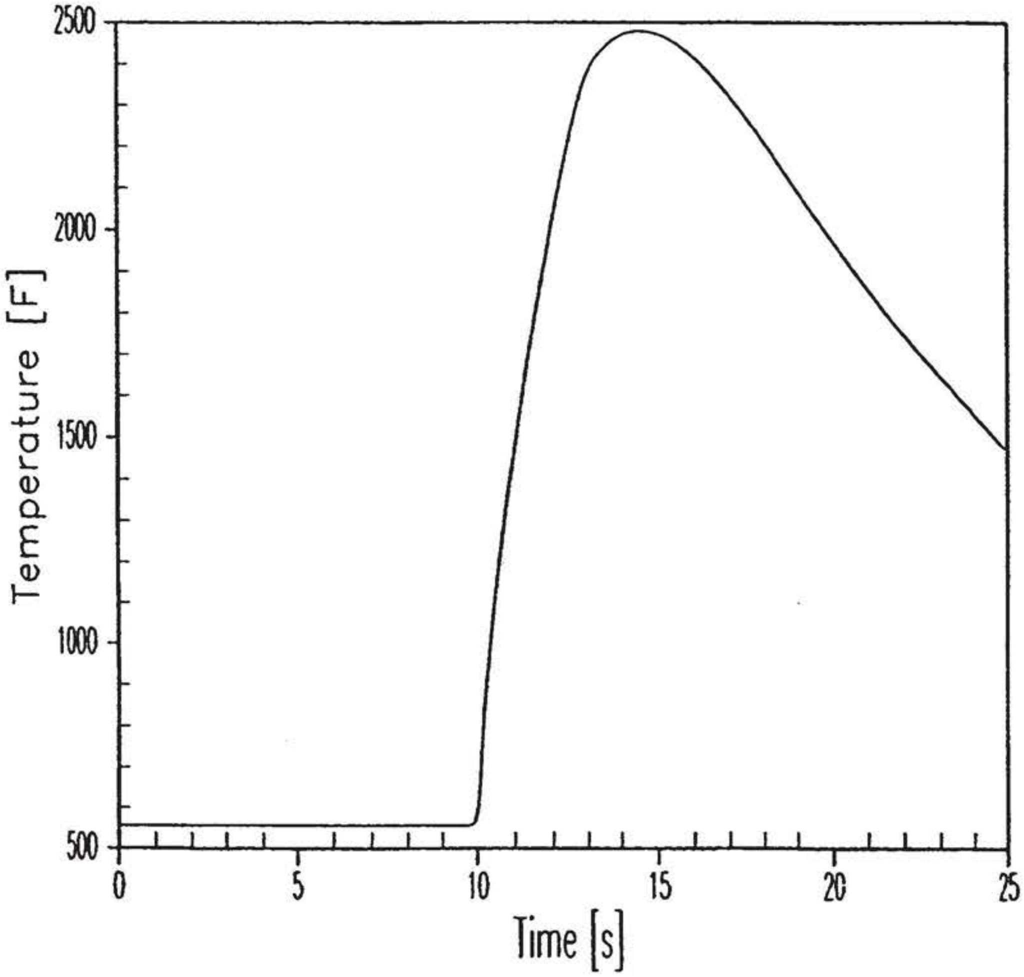
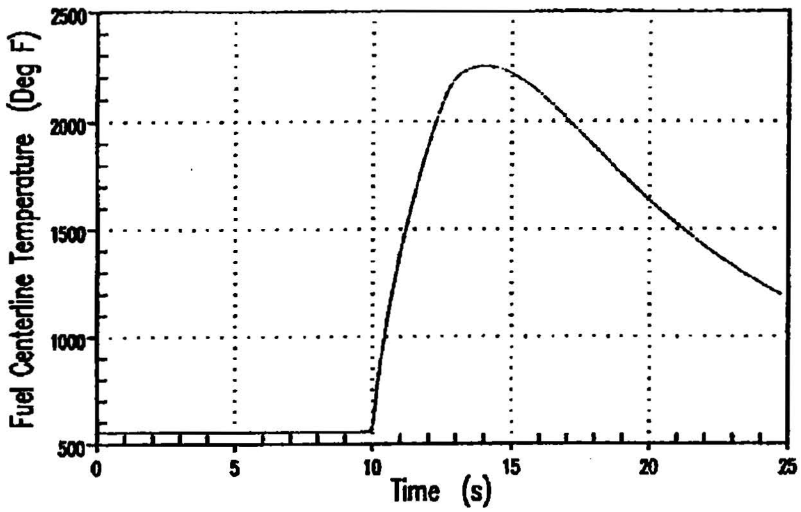


Figure 14.4-3 **Uncontrolled RCCA Withdrawal from a Subcritical Condition – Hot-Spot Fuel Centerline Temperature Versus Time** **Rev. 27**



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UNCONTROLLED RCCA WITHDRAWAL FROM A SUBCRITICAL CONDITION – HOT-SPOT FUEL CENTERLINE TEMPERATURE VERSUS TIME

DWN KJF	DATE 1-28-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14403A.DGN		*FIGURE 14.4-3A REV. 31*

FIGURE 14.4-3A

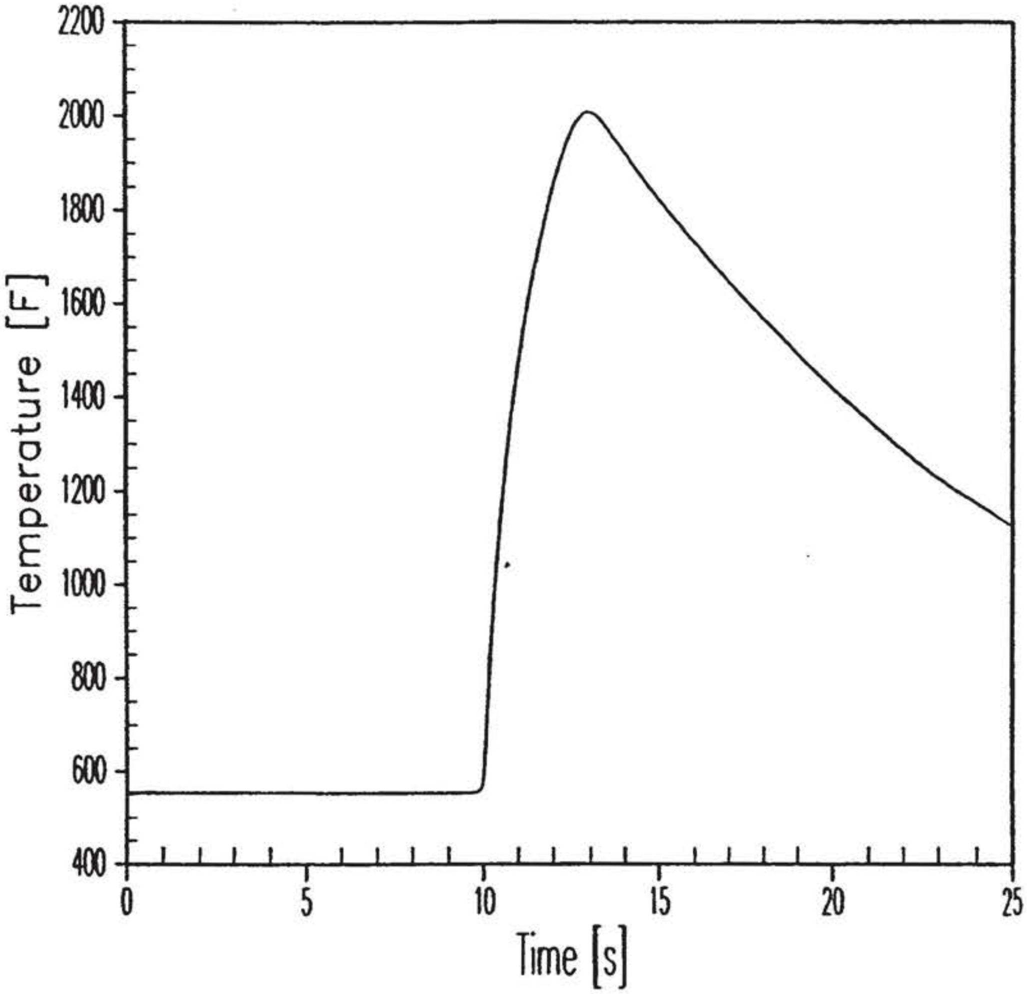
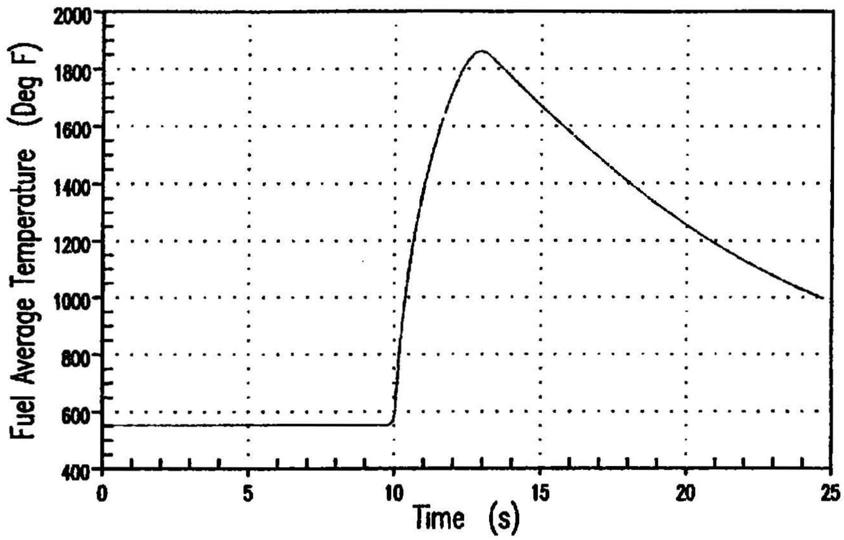


Figure 14.4-4 **Uncontrolled RCCA Withdrawal from a Subcritical Condition – Hot-Spot Fuel Average Temperature Versus Time** **Rev. 27**



UNCONTROLLED RCCA WITHDRAWAL FROM A SUBCRITICAL CONDITION – HOT-SPOT FUEL AVERAGE TEMPERATURE VERSUS TIME

DWN KJF	DATE 1-28-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14404A.DGN		*FIGURE 14.4-4A REV. 31*

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FIGURE 14.4-4A

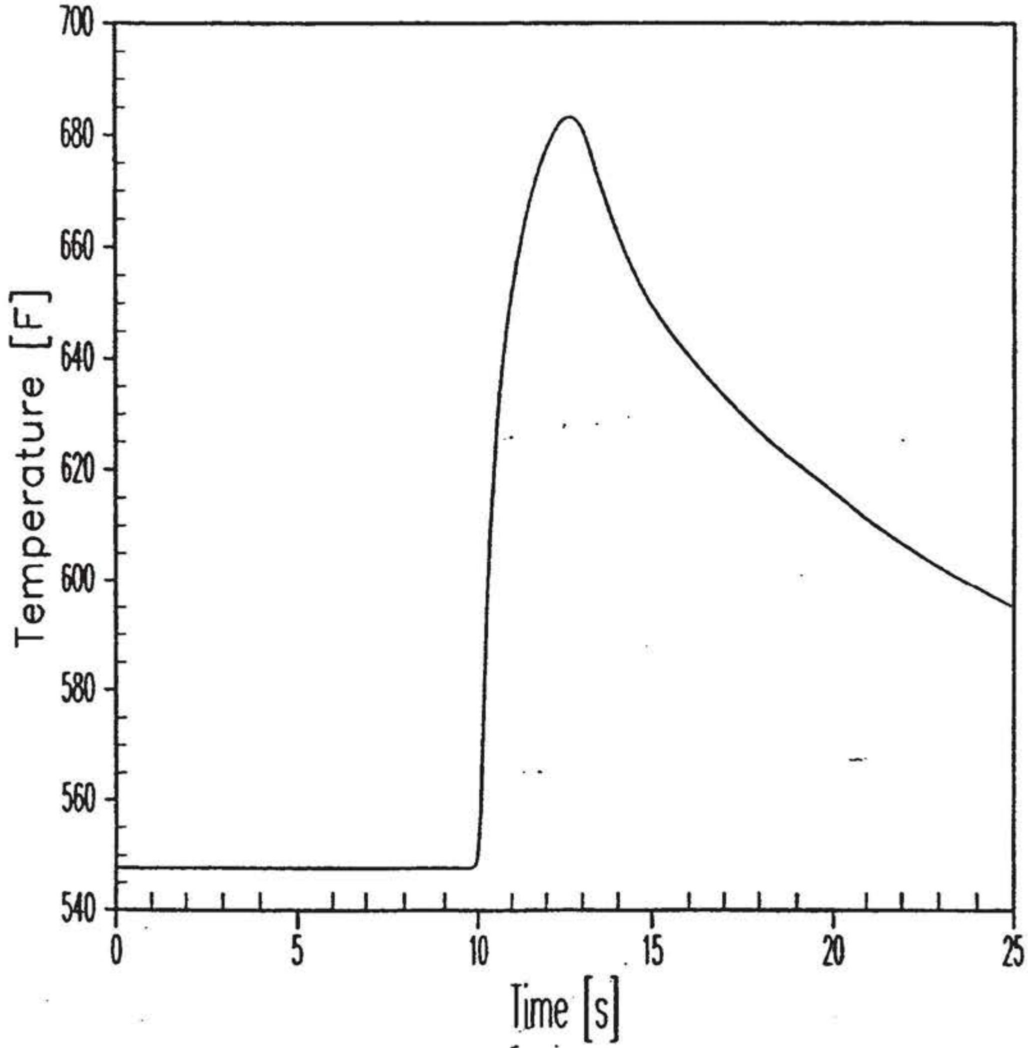
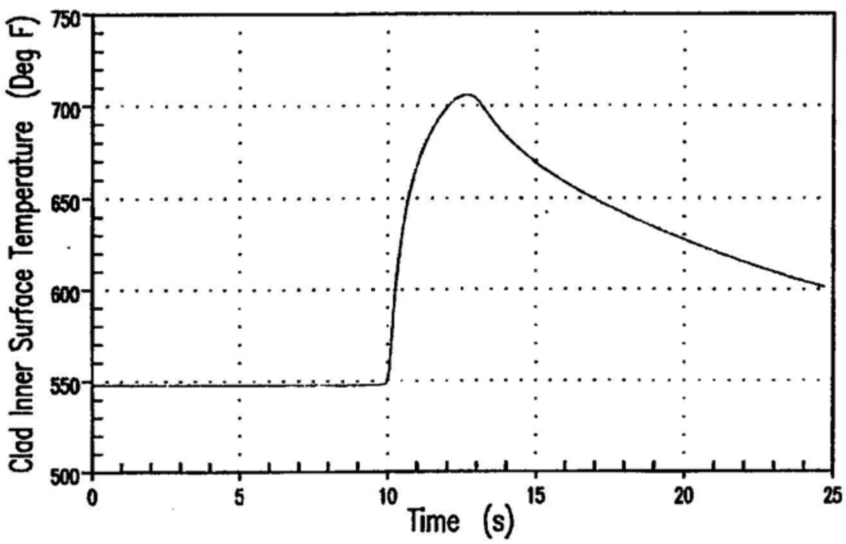


Figure 14.4-5	**Uncontrolled RCCA Withdrawal from a Subcritical Condition – Hot-Spot Cladding Temperature Versus Time**	**Rev. 27**



UNCONTROLLED RCCA WITHDRAWAL FROM A SUBCRITICAL CONDITION – HOT-SPOT CLADDING TEMPERATURE VERSUS TIME

DWN KJF	DATE 1-28-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14405A.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-5A*	*REV. 31*

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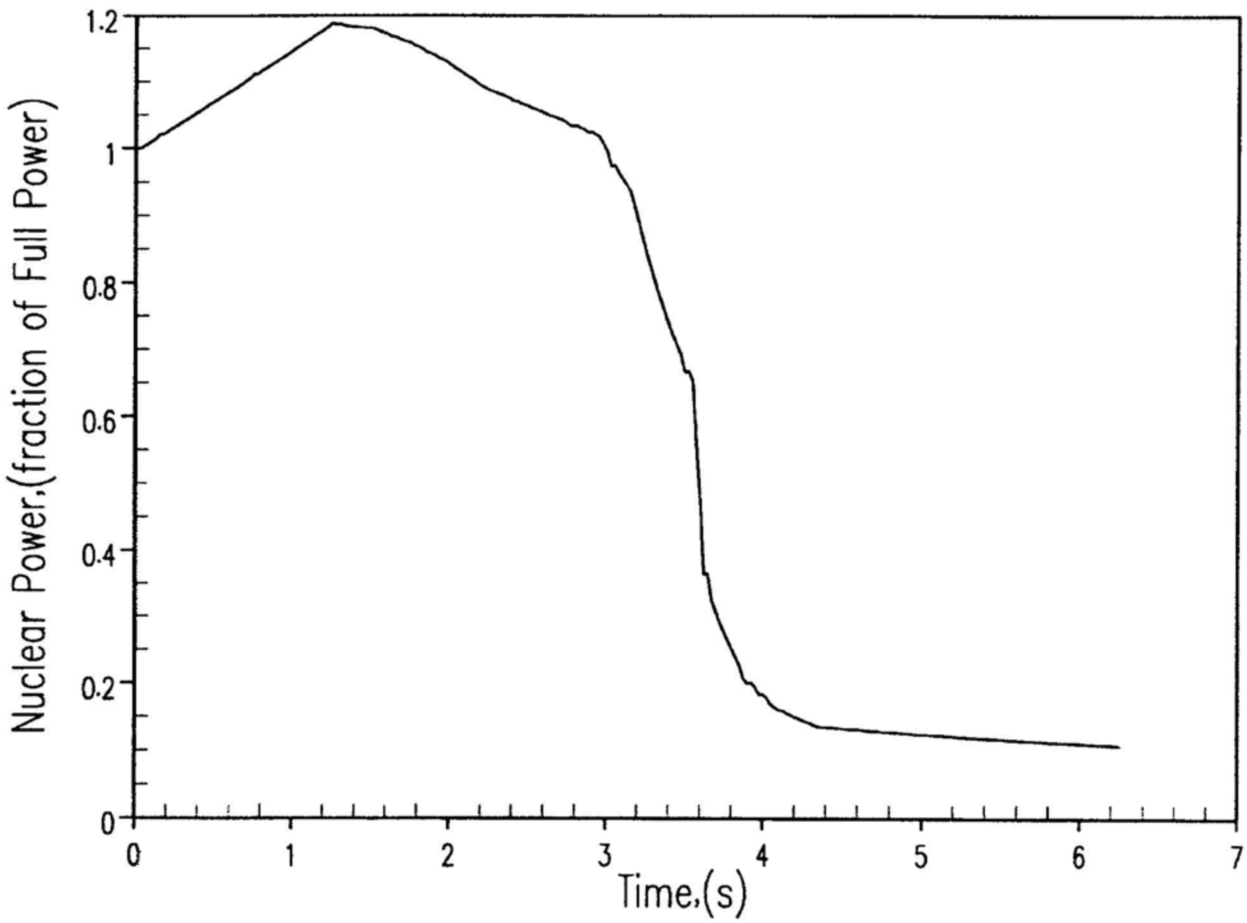
FIGURE 14.4-5A



UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER (110 pcm/sec – FULL POWER)
MINIMUM REACTIVITY FEEDBACK – NUCLEAR POWER vs. TIME

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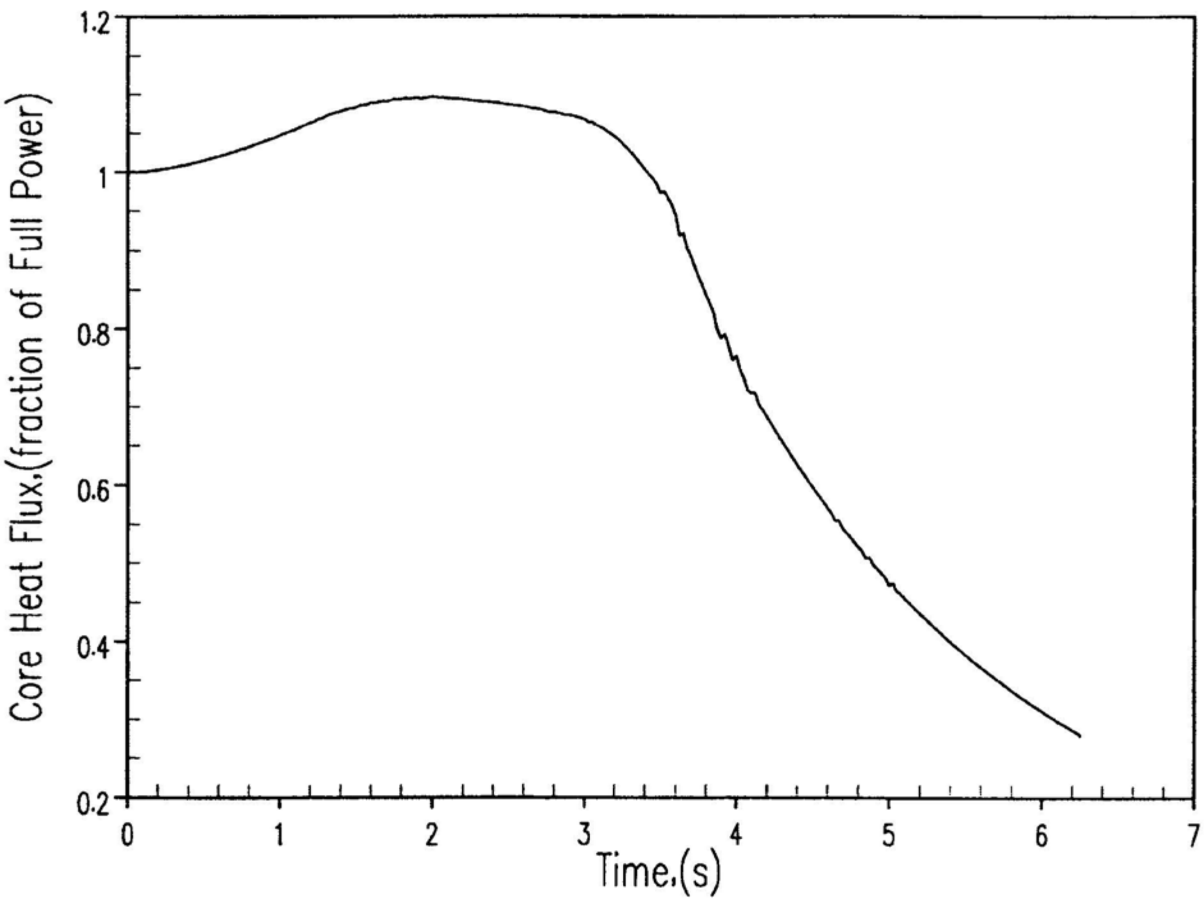
CHECKED:	CAD FILE: U14406.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-6*	*REV. 33*

01386642



UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER (110 pcm/sec – FULL POWER)
MINIMUM REACTIVITY FEEDBACK – CORE HEAT FLUX vs. TIME

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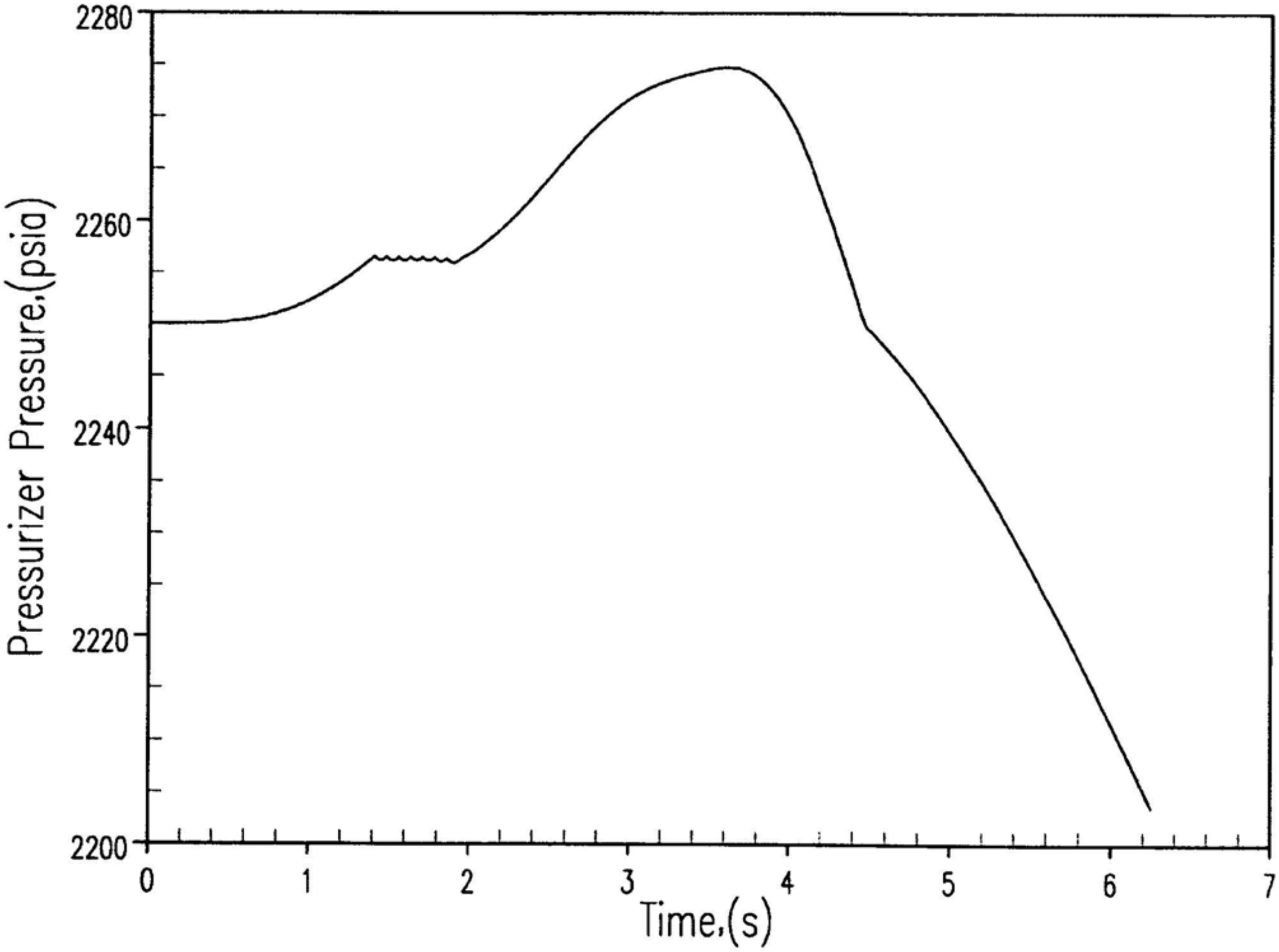
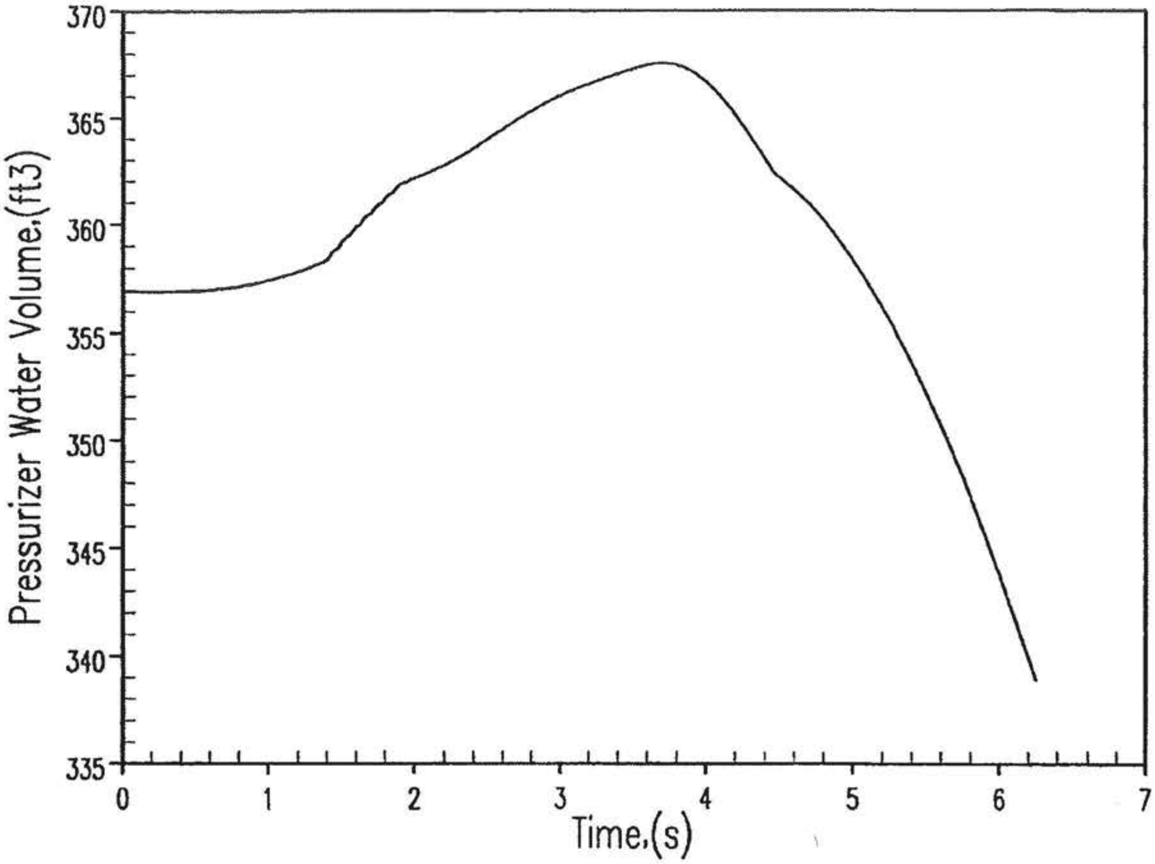


UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER (110 pcm/sec – FULL POWER)
MINIMUM REACTIVITY FEEDBACK – PRESSURIZER PRESSURE vs. TIME

DWN: KJF	DATE: 2–19–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U14408.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-8*	*REV. 33*

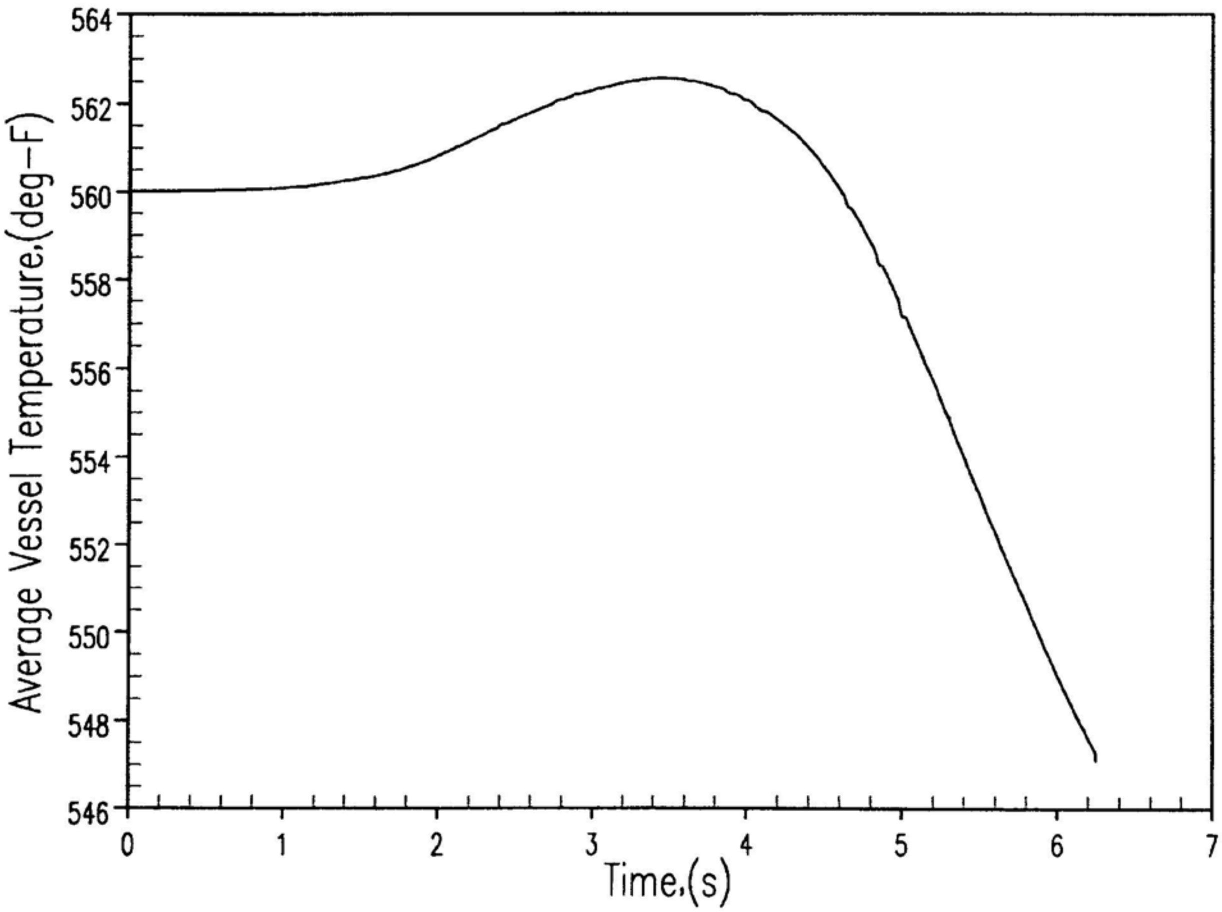
01386642



UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER (110 pcm/sec – FULL POWER)
MINIMUM REACTIVITY FEEDBACK – PRESSURIZER WATER VOLUME vs. TIME

DWN: KJF	DATE: 2-19-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U14409.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	FIGURE 14.4-9	REV. 33

01386642



UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER (110 pcm/sec – FULL POWER)
MINIMUM REACTIVITY FEEDBACK – AVERAGE VESSEL TEMPERATURE vs. TIME

DWN: KJF	DATE: 2-19-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED:	CAD FILE: U14410.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-10* *REV. 33*

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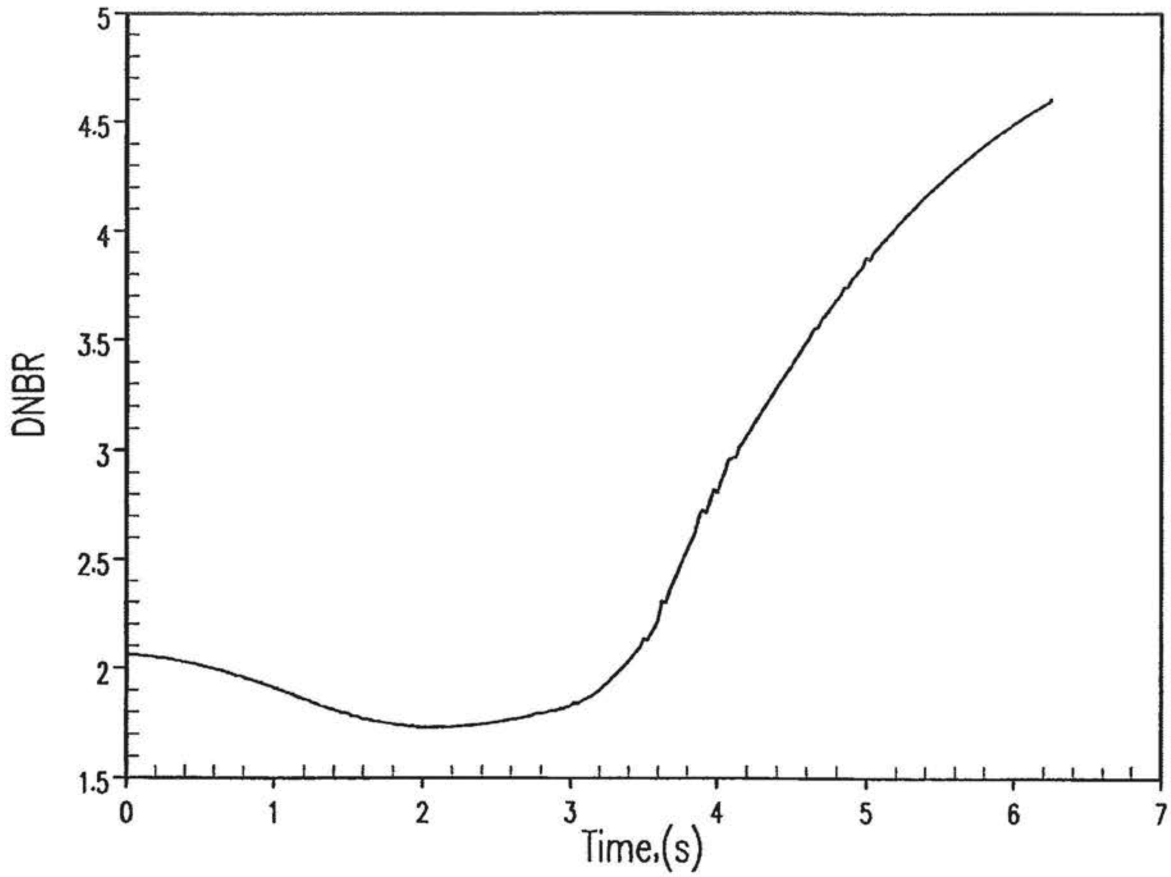

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UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER (110 pcm/sec – FULL POWER)
MINIMUM REACTIVITY FEEDBACK – DNBR vs. TIME

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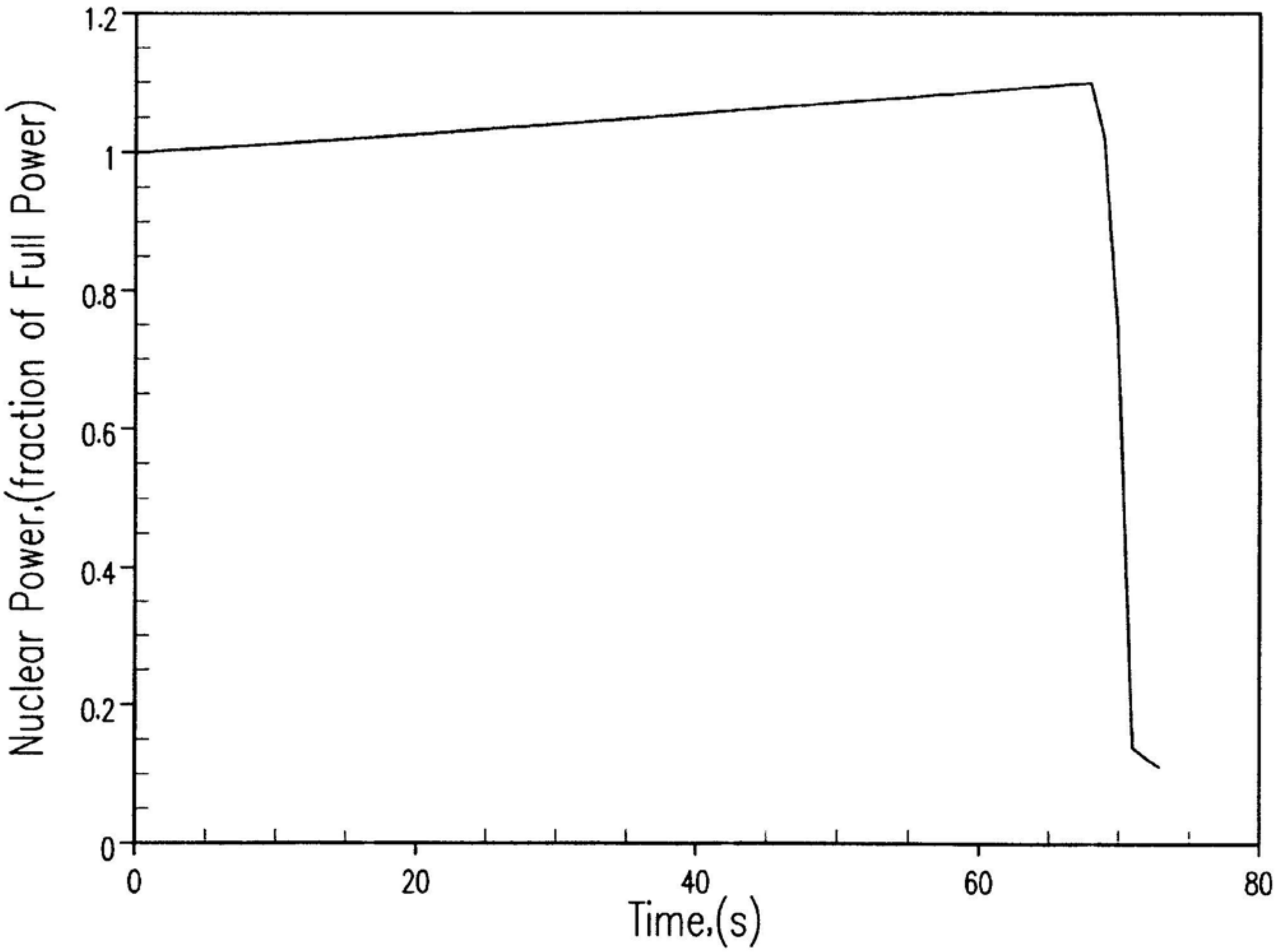

UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER (1 pcm/sec – FULL POWER)
MINIMUM REACTIVITY FEEDBACK – NUCLEAR POWER vs. TIME

DWN: KJF	DATE: 2–19–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
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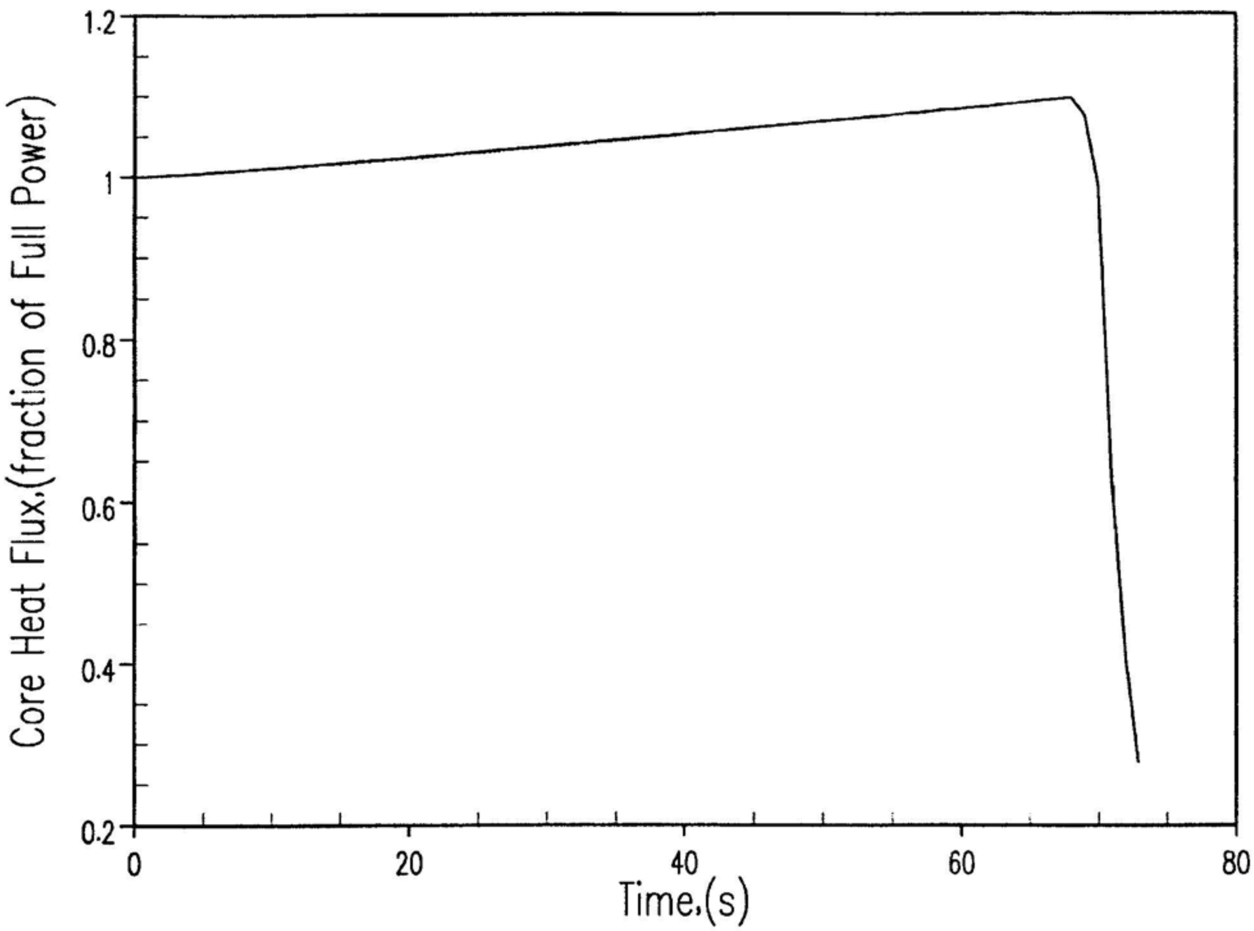

UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER (1 pcm/sec – FULL POWER)
MINIMUM REACTIVITY FEEDBACK – CORE HEAT FLUX vs. TIME

DWN: KJF	DATE: 2–19–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U14413.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-13*	*REV. 33*

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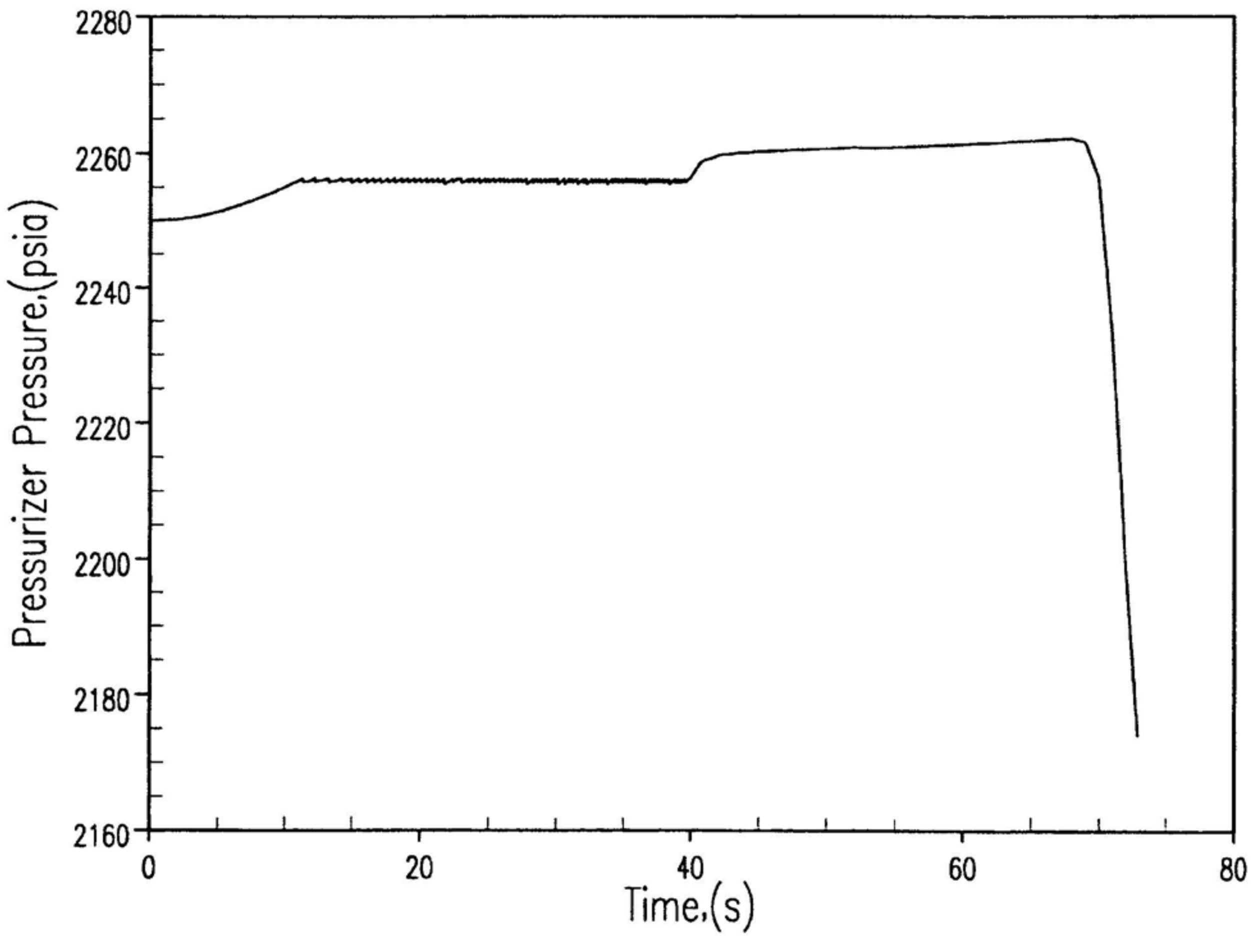

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UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER (1 pcm/sec – FULL POWER)
MINIMUM REACTIVITY FEEDBACK – PRESSURIZER PRESSURE vs. TIME

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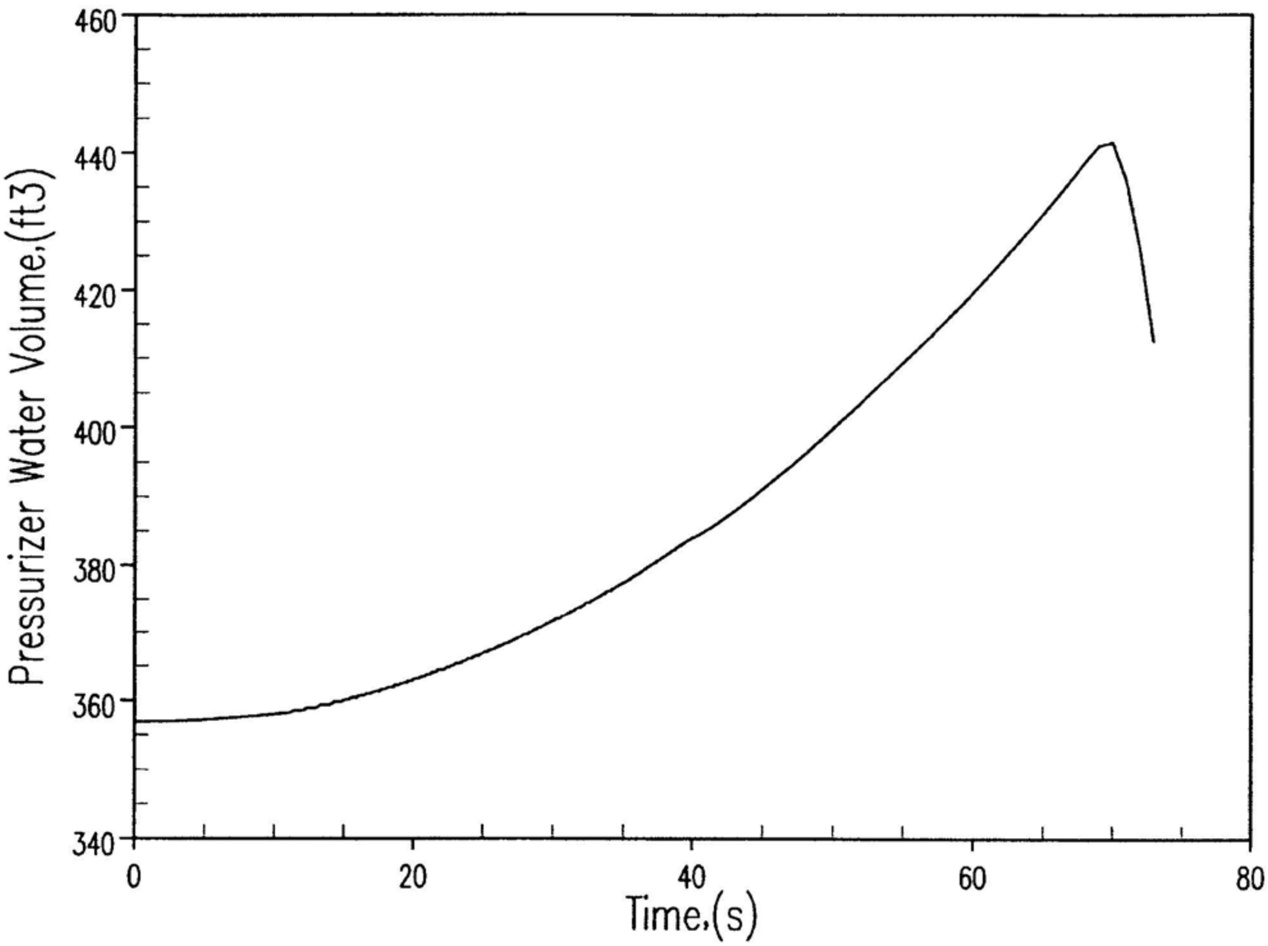

UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER (1 pcm/sec – FULL POWER)
MINIMUM REACTIVITY FEEDBACK – PRESSURIZER WATER VOLUME vs. TIME

DWN: KJF	DATE: 2–19–14	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE	
CHECKED:	CAD FILE: U14415.DGN		*FIGURE 14.4-15*	*REV. 33*

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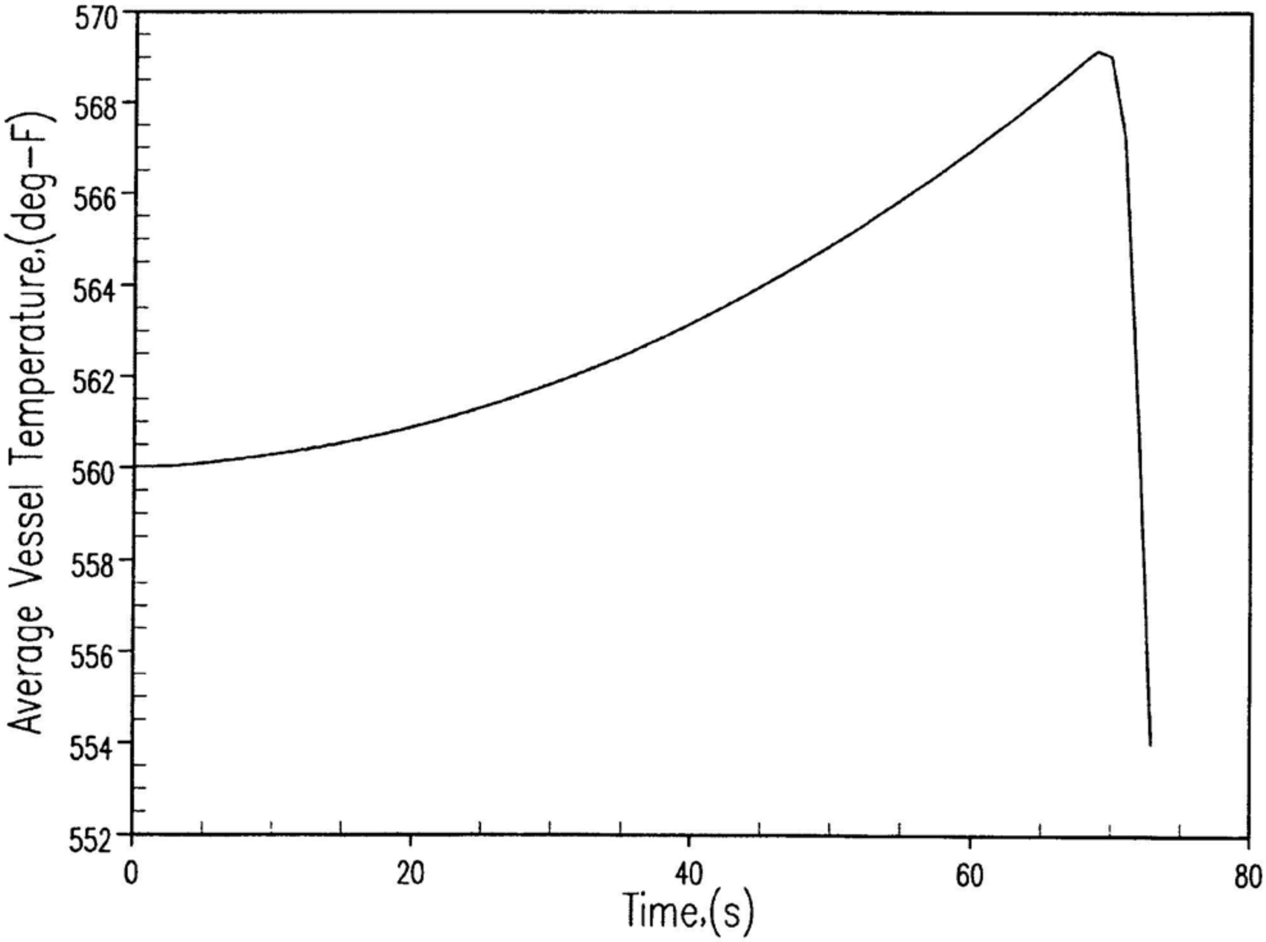

UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER (1 pcm/sec – FULL POWER)
MINIMUM REACTIVITY FEEDBACK – AVERAGE VESSEL TEMPERATURE vs. TIME

DWN: KJF	DATE: 2–19–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED:	CAD FILE: U14416.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-16* *REV. 33*

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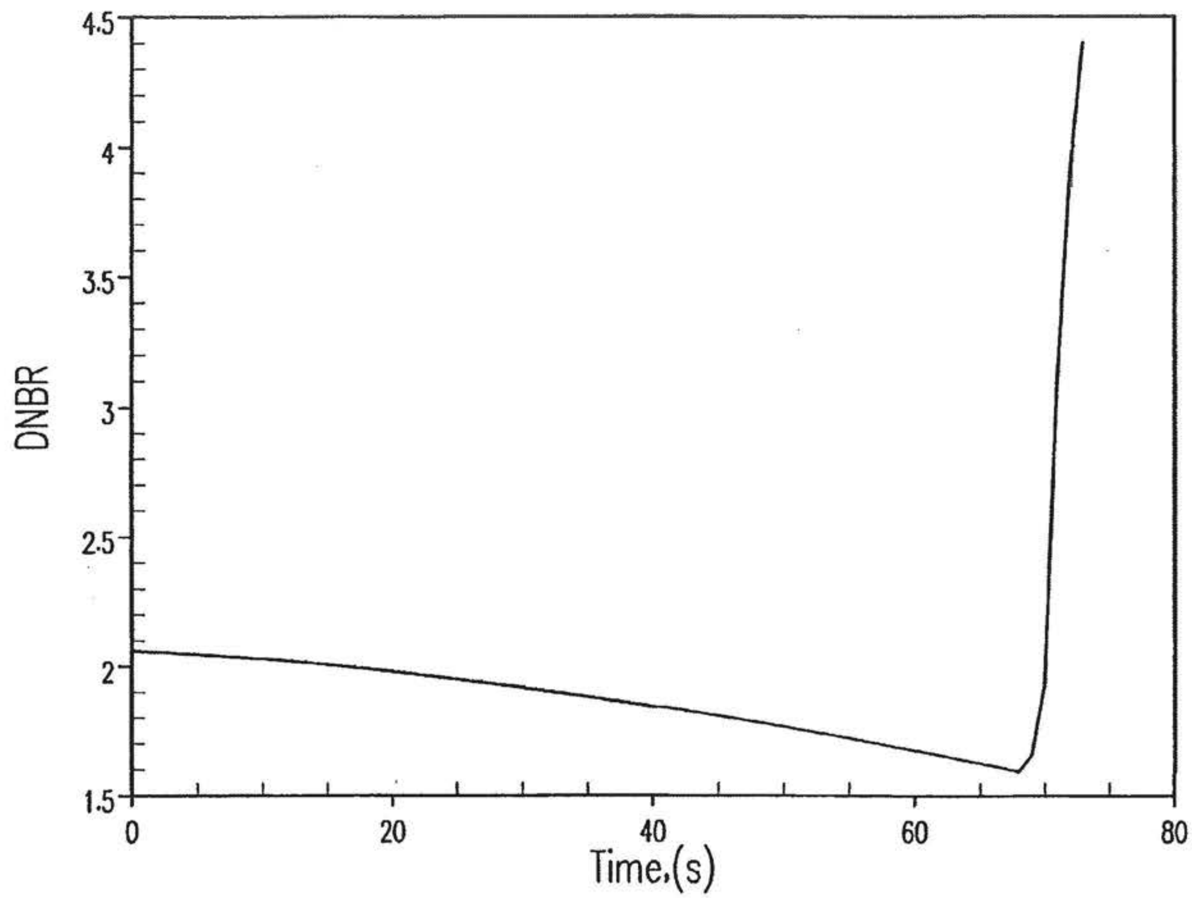

UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER (1 pcm/sec – FULL POWER)
MINIMUM REACTIVITY FEEDBACK – DNBR vs. TIME

DWN: KJF	DATE: 2-19-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
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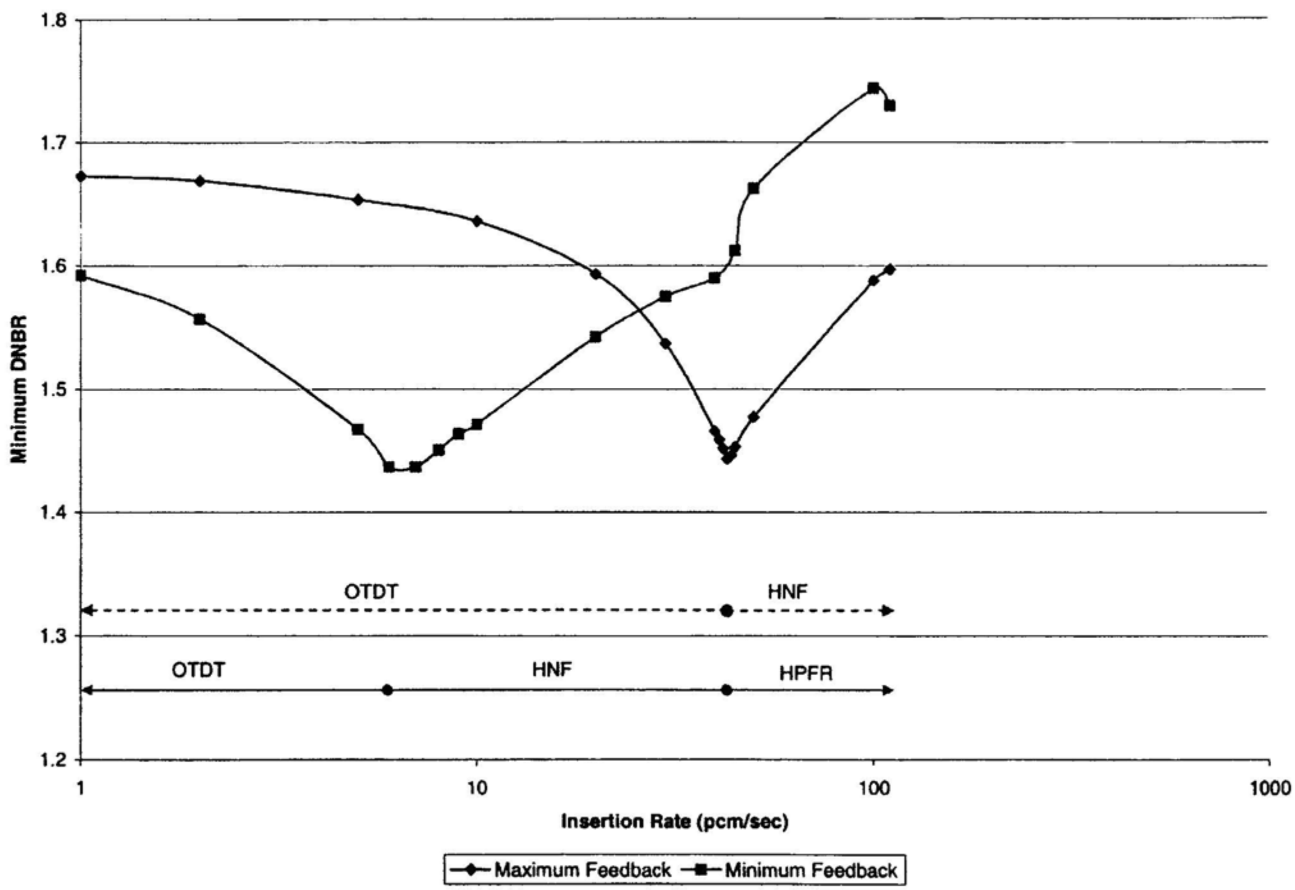

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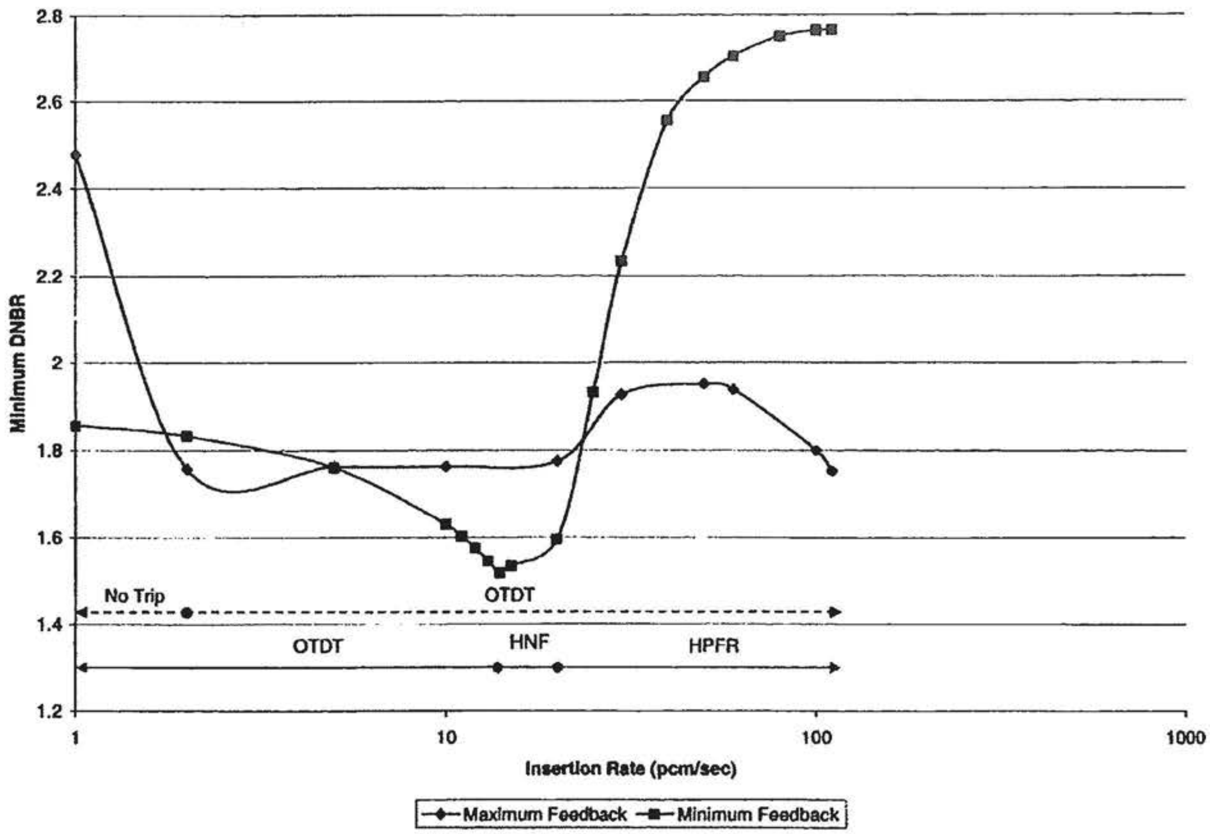
UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER, 100% POWER

DWN: KJF	DATE: 2-19-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U14418.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-18*	*REV. 33*



UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER, 60% POWER

DWN: KJF	DATE: 2-19-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
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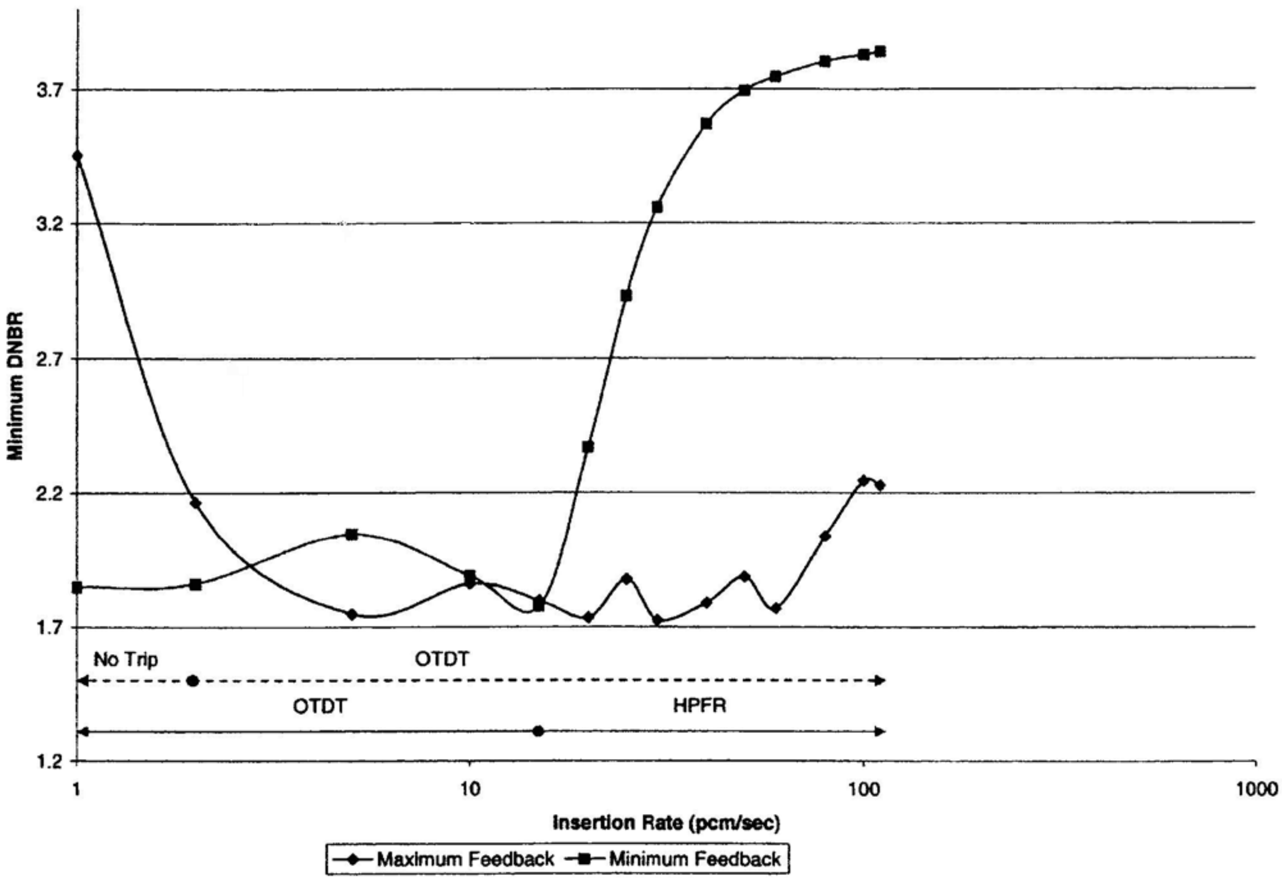
01386642



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UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER, 10% POWER

DWN: KJF	DATE: 2–19–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U14420.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-20*	*REV. 33*

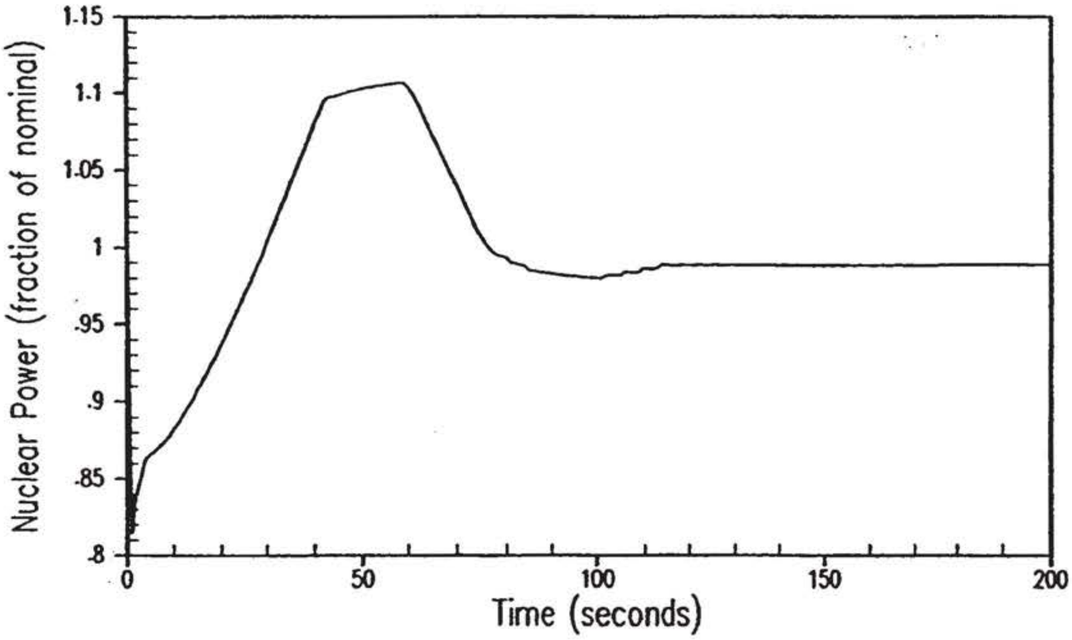

Figure 14.4-21
Dropped RCCA – Representative Transient Response -
Nuclear Power vs. Time

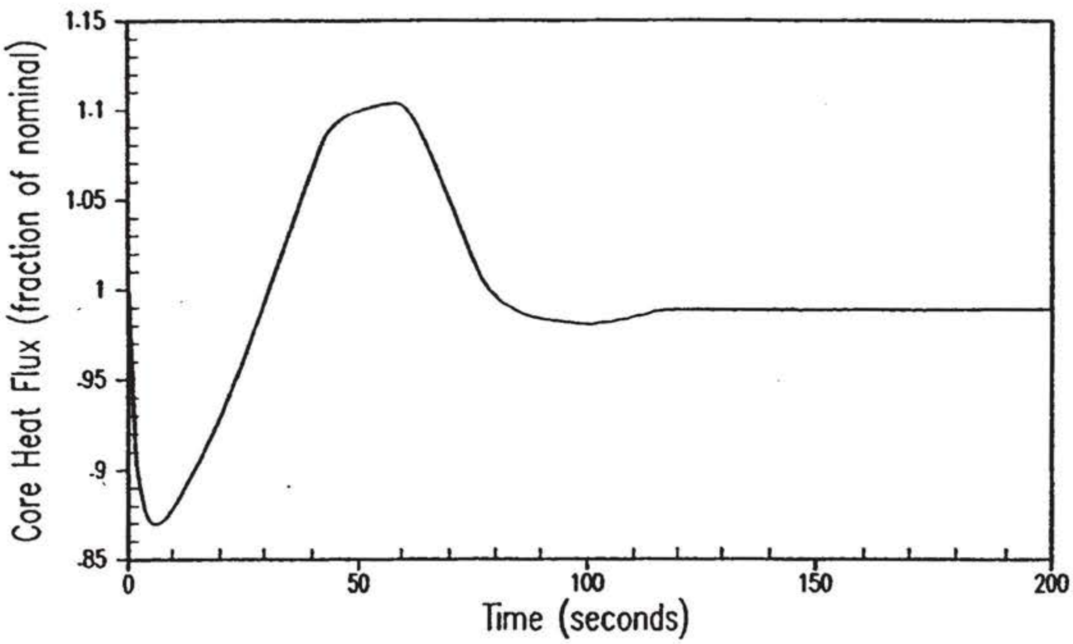

Figure 14.4-22
Dropped RCCA – Representative Transient Response -
Core Heat Flux vs. Time

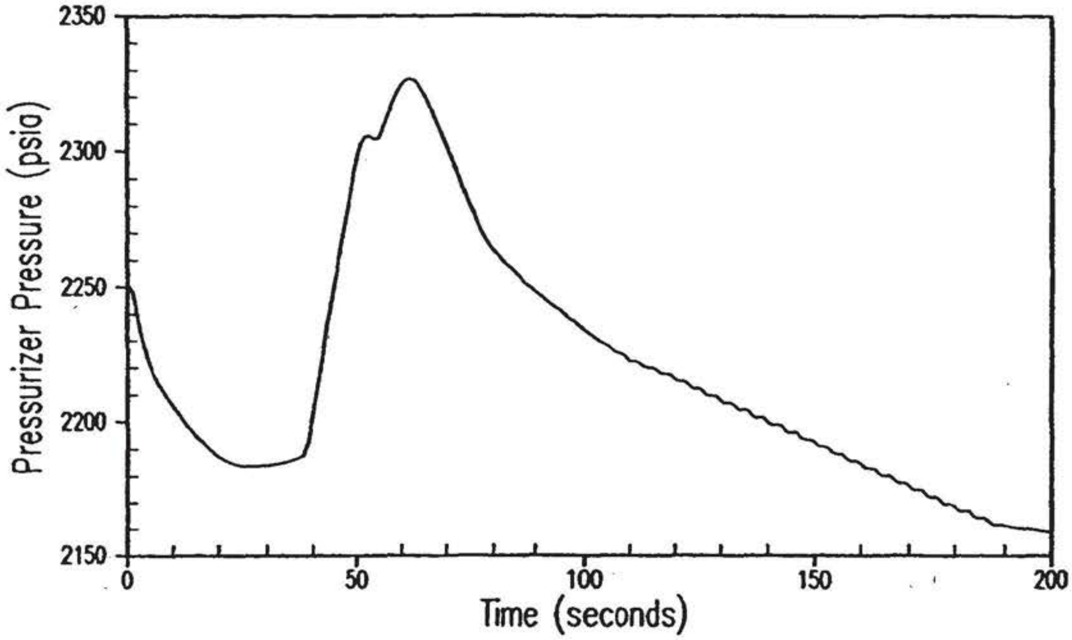

Figure 14.4-23
Dropped RCCA – Representative Transient Response -
Pressurizer Pressure vs. Time

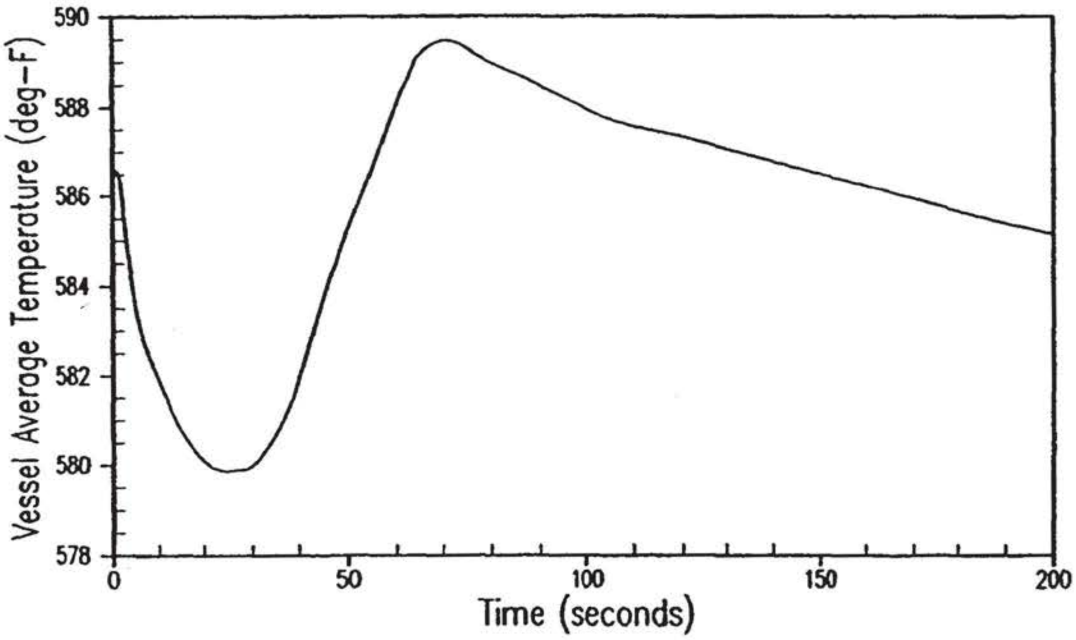

Figure 14.4-24
Dropped RCCA – Representative Transient Response -
Vessel Average Temperature vs. Time

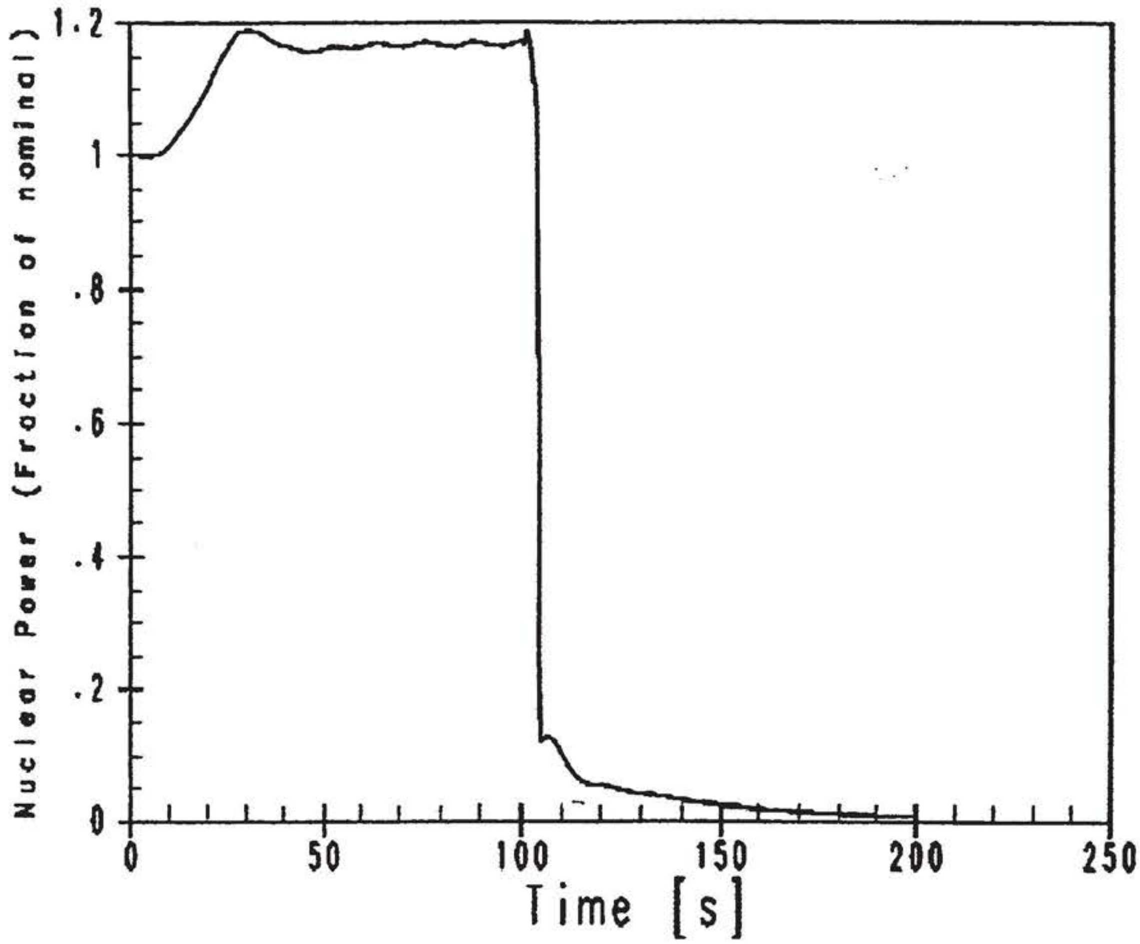

Figure 14.4-30
RSG – Feedwater Flow Increase to Both Loops – Hot Full Power - Automatic Rod Control Case – Reactor Power Versus Time

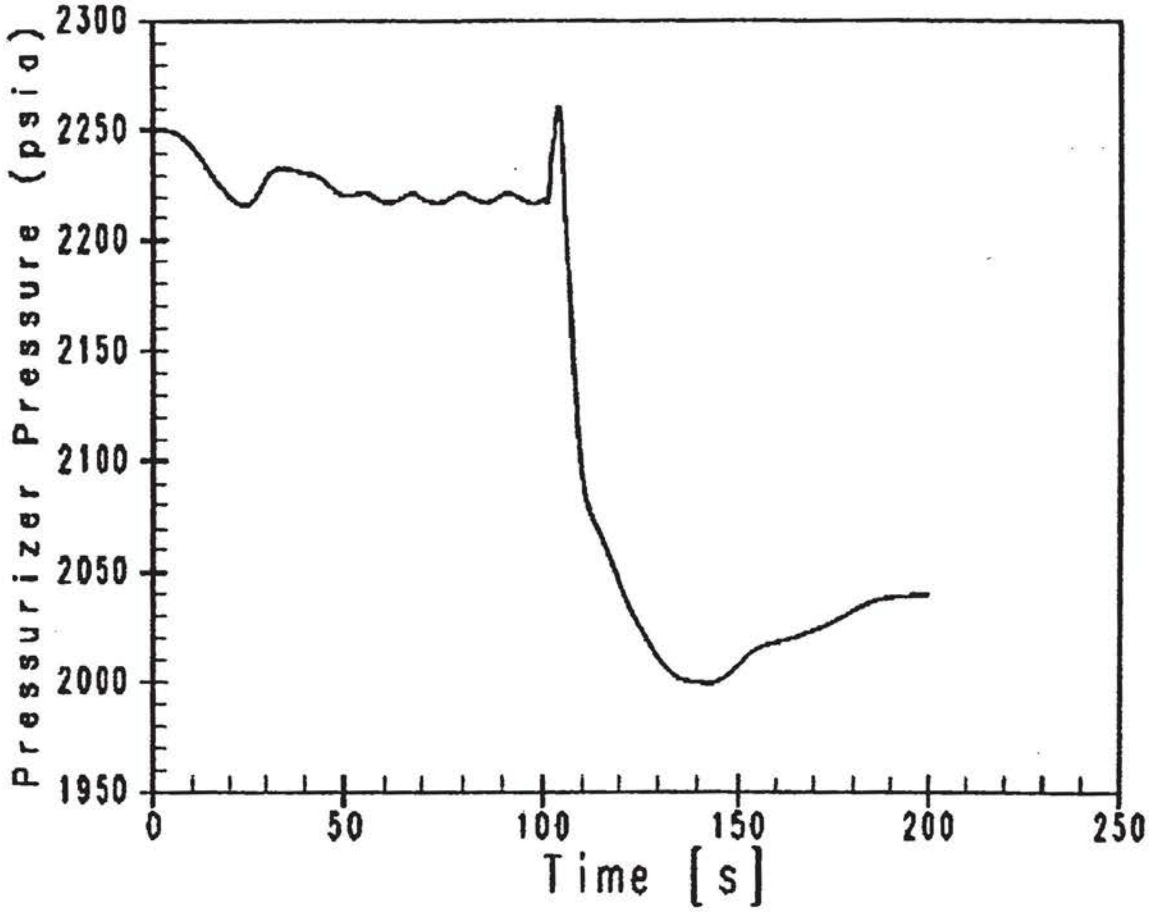

Figure 14.4-31
RSG – Feedwater Flow Increase to Both Loops – Hot Full Power - Automatic Rod Control Case – Pressurizer Pressure Versus Time

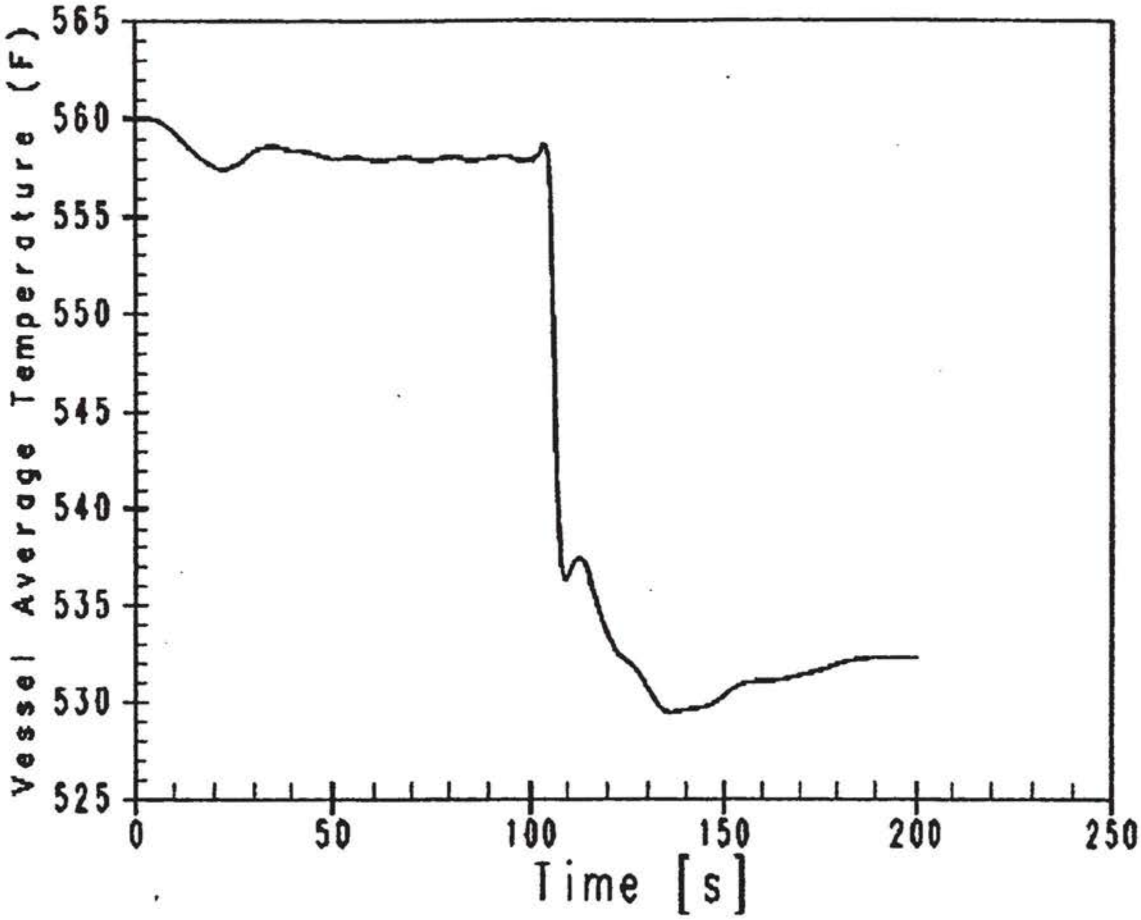

Figure 14.4-32
RSG – Feedwater Flow Increase to Both Loops – Hot Full Power -
Automatic Rod Control Case – Core Average Temperature Versus Time

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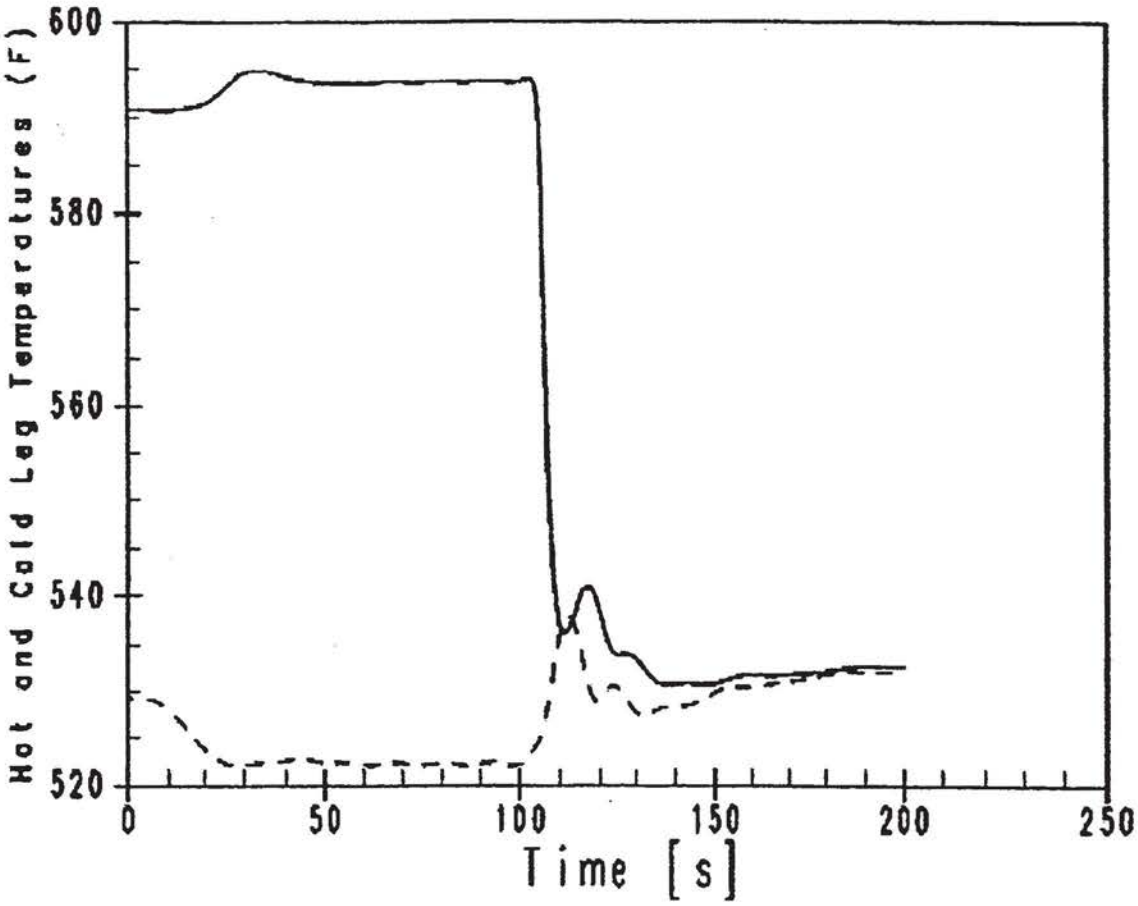


Figure 14.4-33
RSG – Feedwater Flow Increase to Both Loops – Hot Full Power - Automatic Rod Control Case – Vessel Outlet and Inlet Temperatures Versus Time

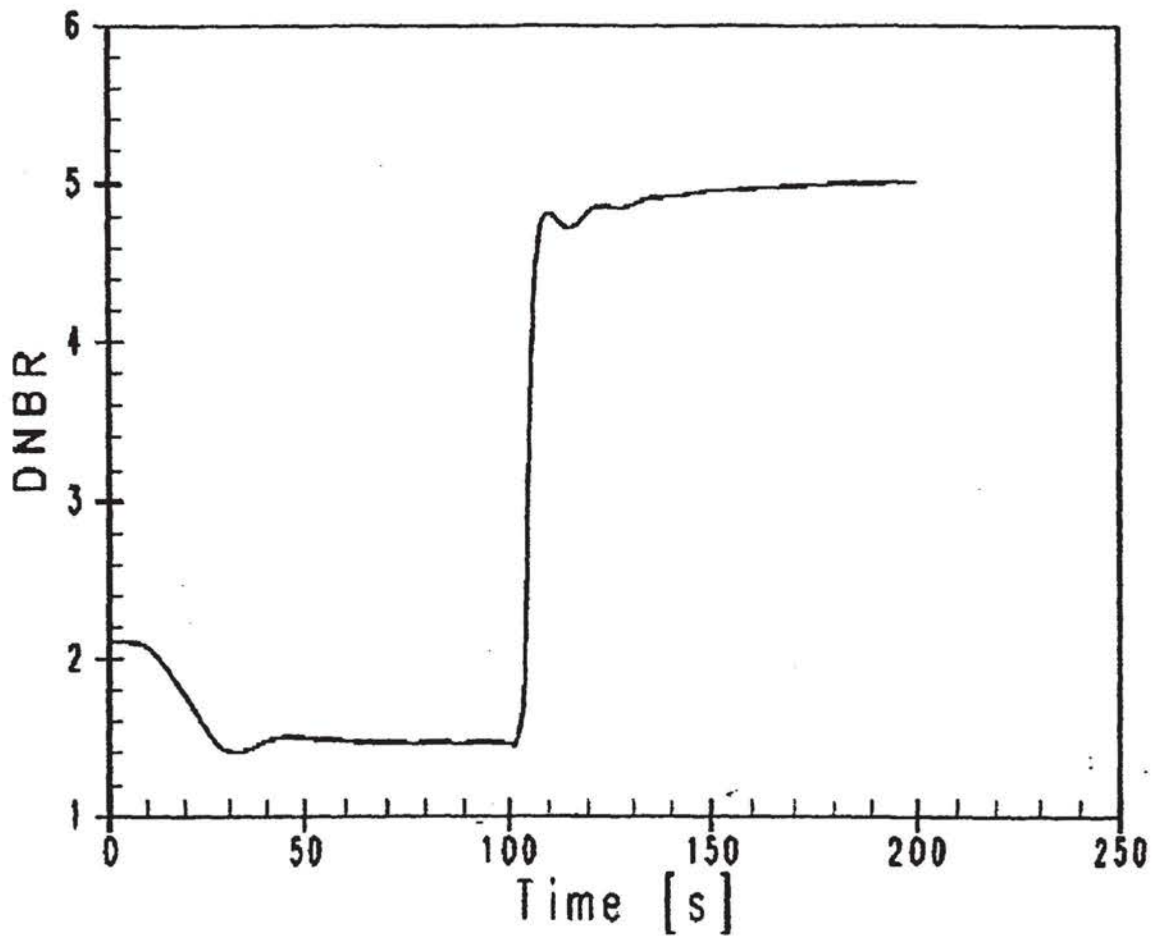
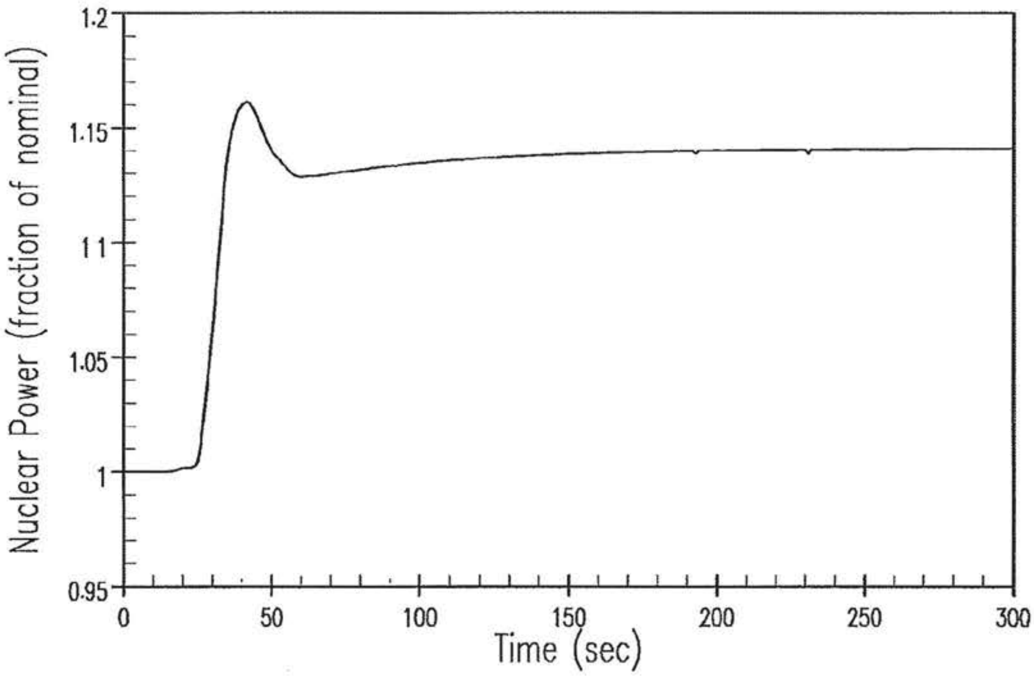


Figure 14.4-34
RSG – Feedwater Flow Increase to Both Loops – Hot Full Power - Automatic Rod Control Case – DNBR Versus Time



TWENTY PERCENT STEP LOAD INCREASE –
MINIMUM REACTIVITY FEEDBACK /AUTOMATIC ROD CONTROL
NUCLEAR POWER VERSUS TIME

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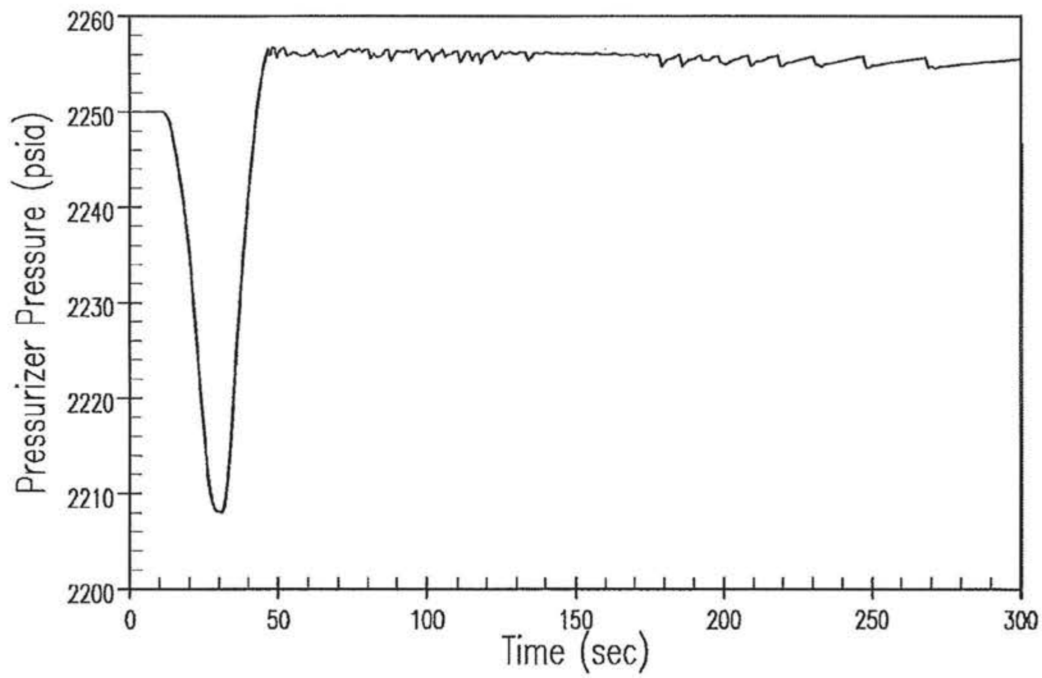

TWENTY PERCENT STEP LOAD INCREASE –
MINIMUM REACTIVITY FEEDBACK /AUTOMATIC ROD CONTROL
PRESSURIZER PRESSURE VERSUS TIME

DWN: KJF	DATE: 2-19-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
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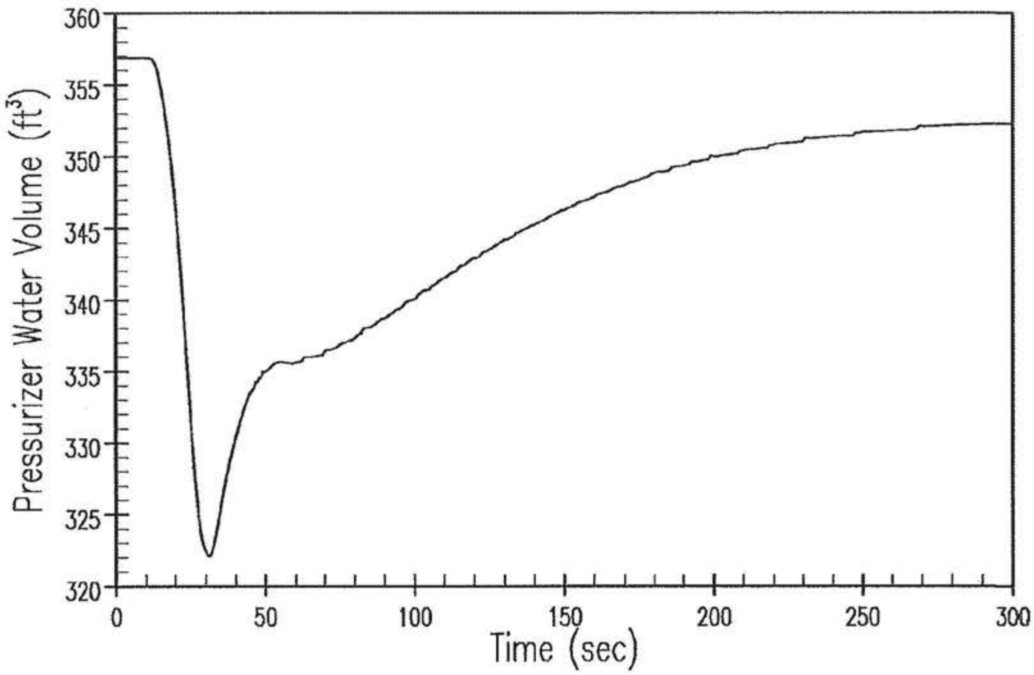

TWENTY PERCENT STEP LOAD INCREASE –
MINIMUM REACTIVITY FEEDBACK /AUTOMATIC ROD CONTROL
PRESSURIZER WATER VOLUME VERSUS TIME

DWN: KJF	DATE: 2-19-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
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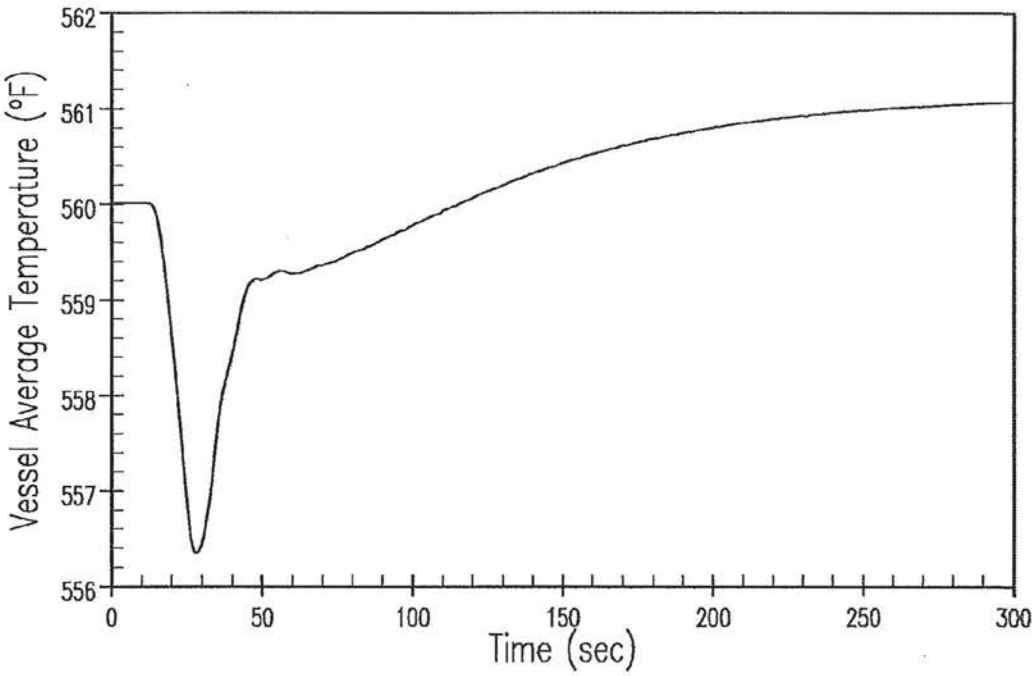

TWENTY PERCENT STEP LOAD INCREASE –
MINIMUM REACTIVITY FEEDBACK /AUTOMATIC ROD CONTROL
VESSEL AVERAGE TEMPERATURE VERSUS TIME

DWN: KJF	DATE: 2-19-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
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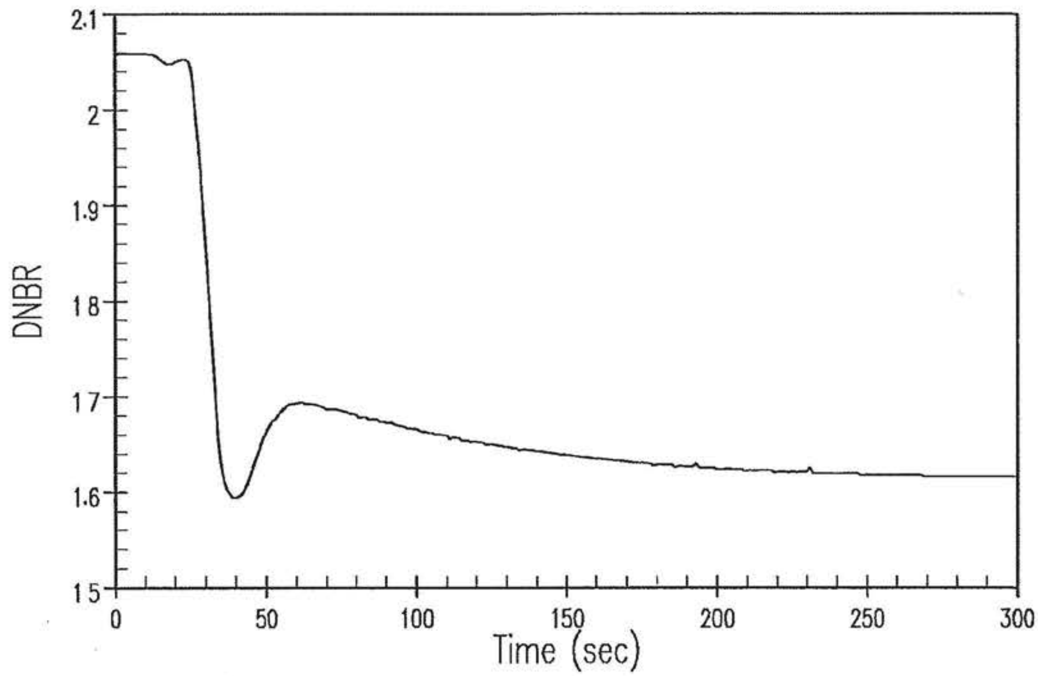
01386642



TWENTY PERCENT STEP LOAD INCREASE –
MINIMUM REACTIVITY FEEDBACK /AUTOMATIC ROD CONTROL
DNBR VERSUS TIME

DWN: KJF	DATE: 2-19-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
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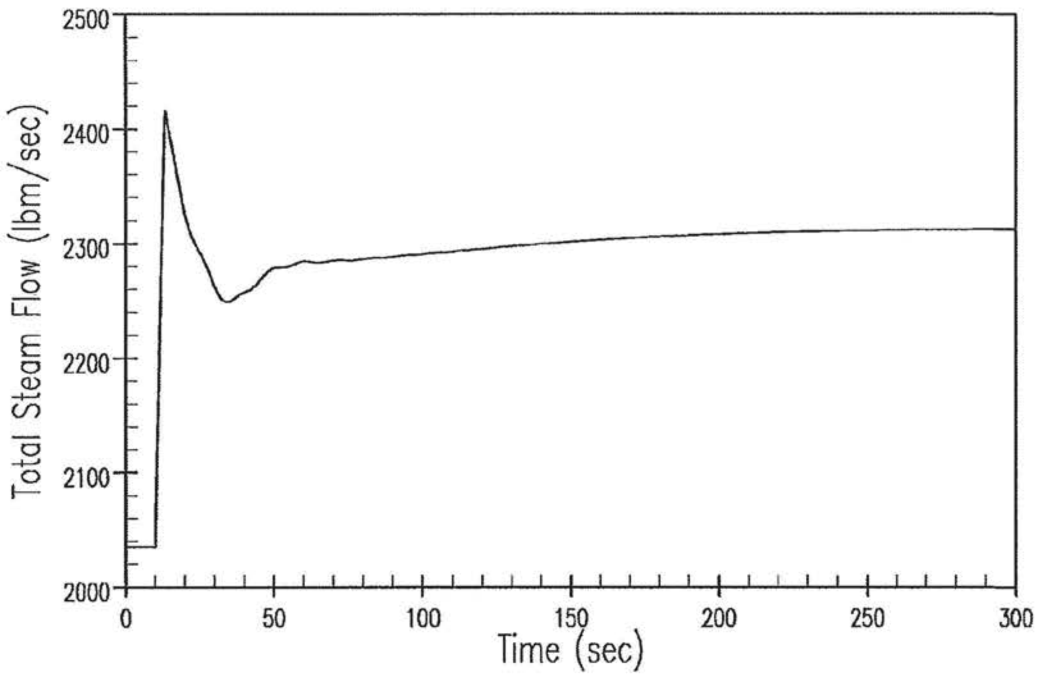

TWENTY PERCENT STEP LOAD INCREASE –
MINIMUM REACTIVITY FEEDBACK /AUTOMATIC ROD CONTROL
TOTAL STEAM FLOW VERSUS TIME

DWN: KJF	DATE: 2-19-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
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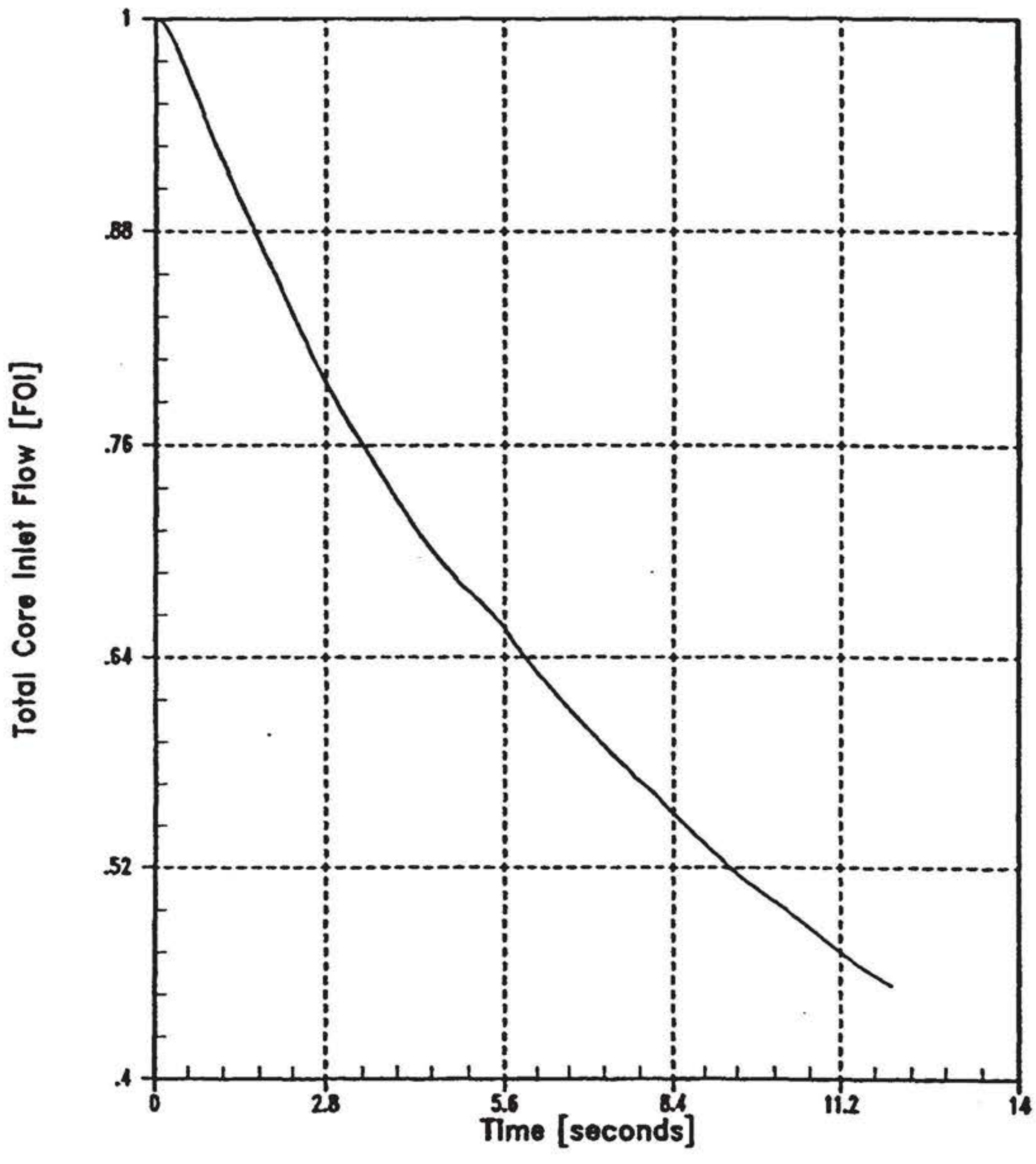
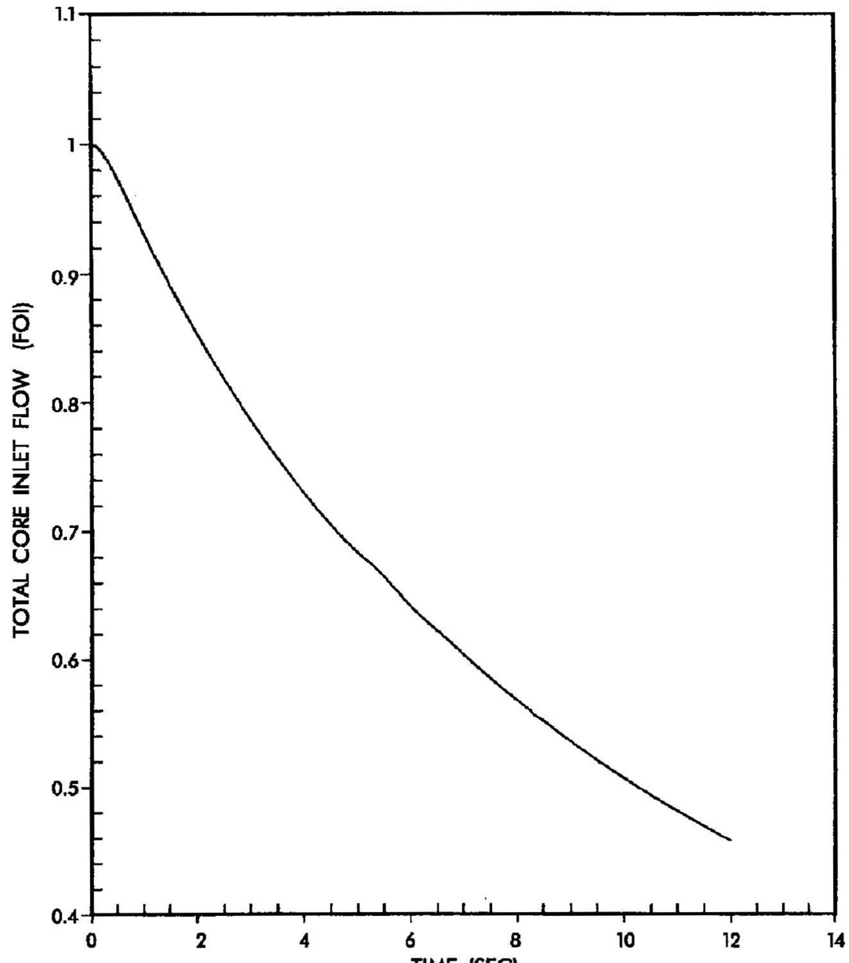


Figure 14.4-41
Total Core Inlet Flow vs. Time
Complete Loss of Flow, Two Pumps Coasting Down (CLOF)



TOTAL CORE INLET FLOW vs. TIME
COMPLETE LOSS OF FLOW, TWO PUMPS COASTING DOWN (CLOF)

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14441A.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-41A REV. 31*

01193868

FIGURE 14.4-41A

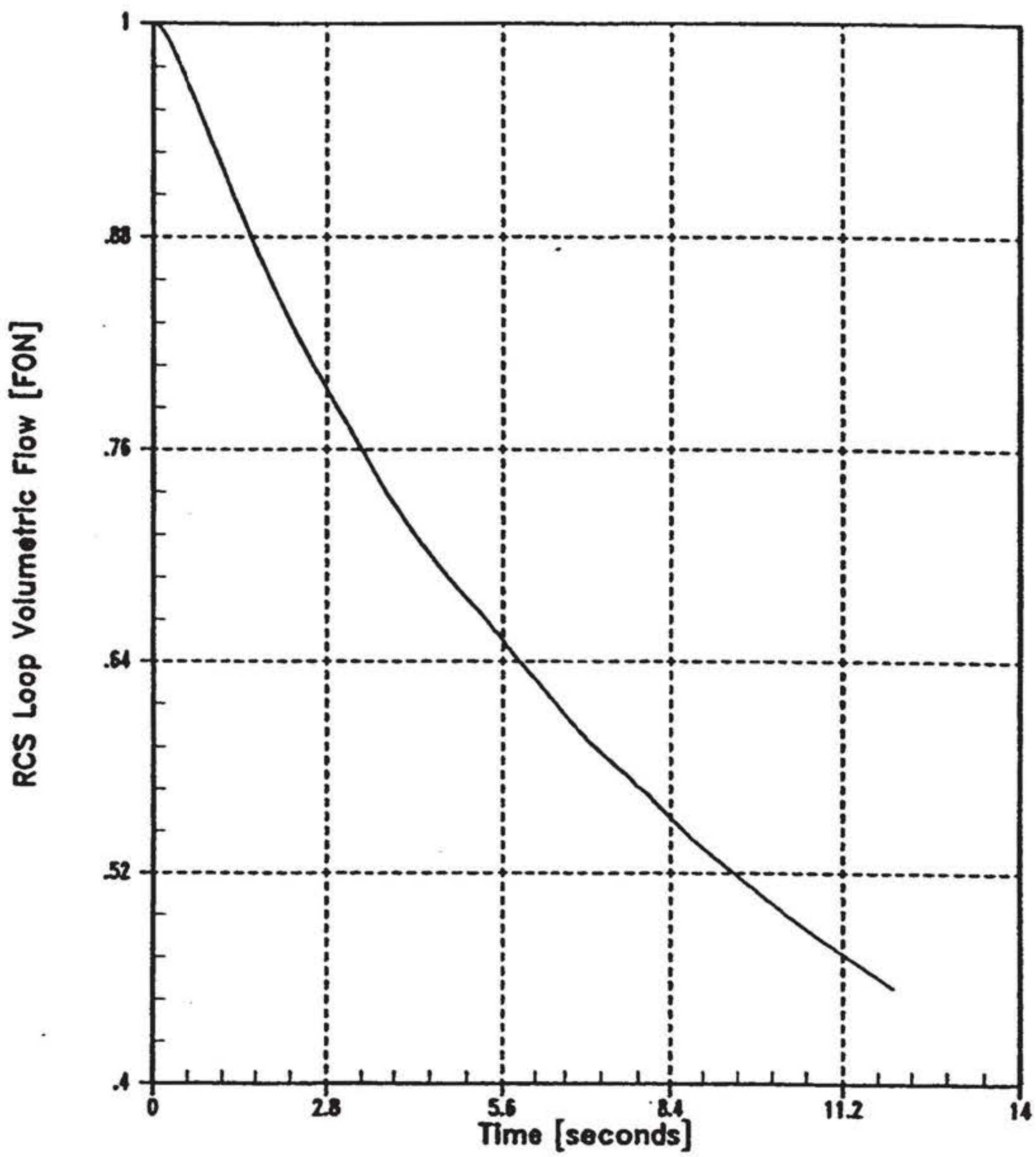
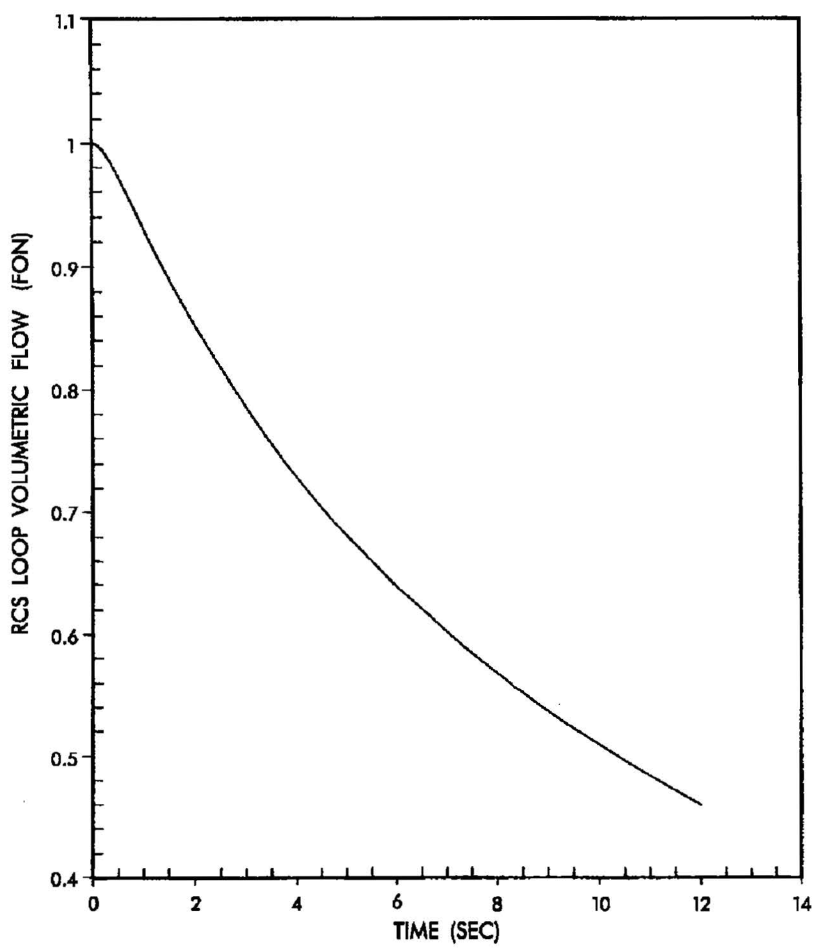


Figure 14.4-42
RCS Loop Flow vs. Time
Complete Loss of Flow, Two Pumps Coasting Down (CLOF)



RCS LOOP FLOW vs. TIME
COMPLETE LOSS OF FLOW, TWO PUMPS COASTING DOWN (CLOF)

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE NONE
CHECKED	CAD FILE U14442A.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-42A REV. 31*

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FIGURE 14.4-42A

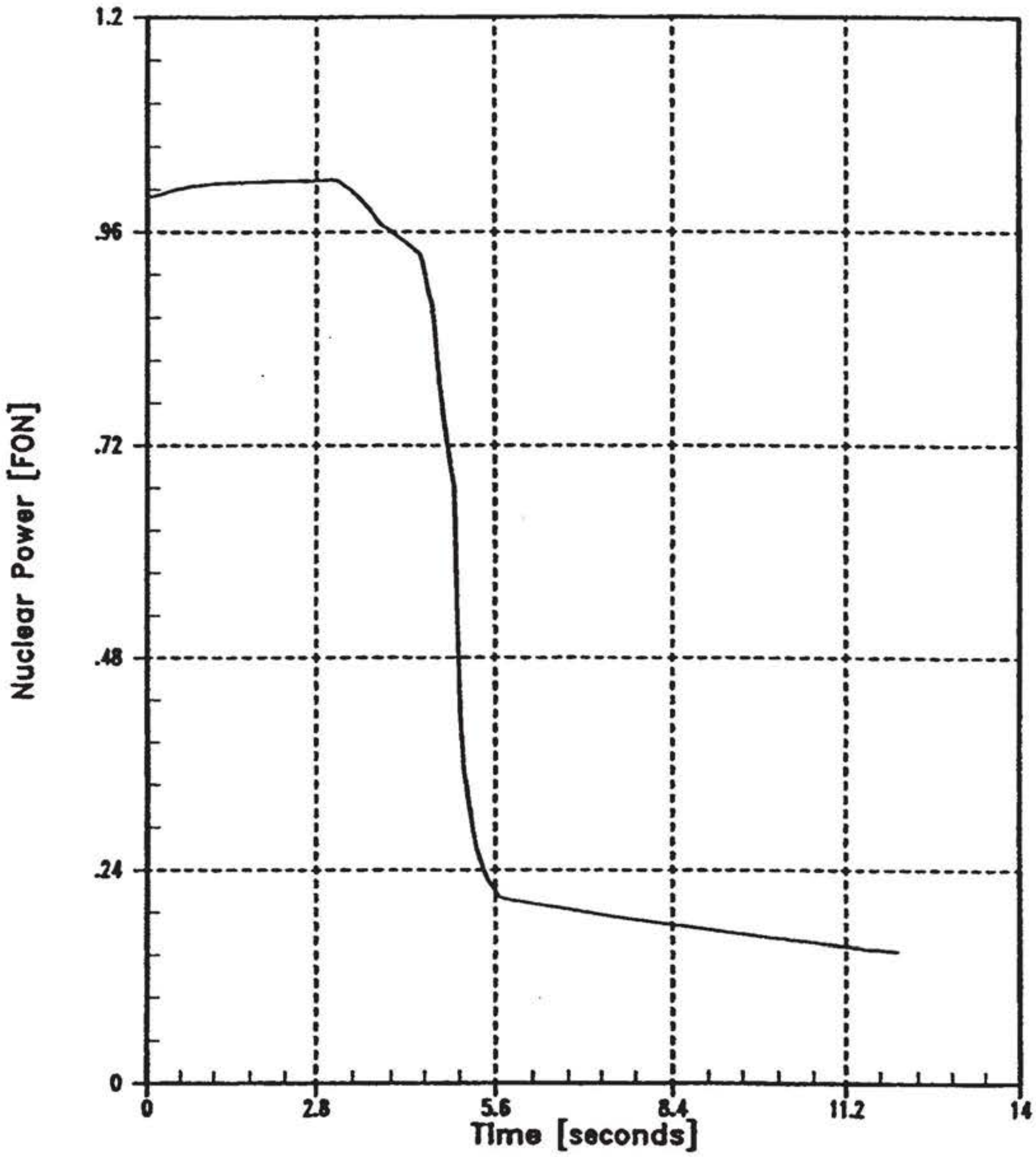
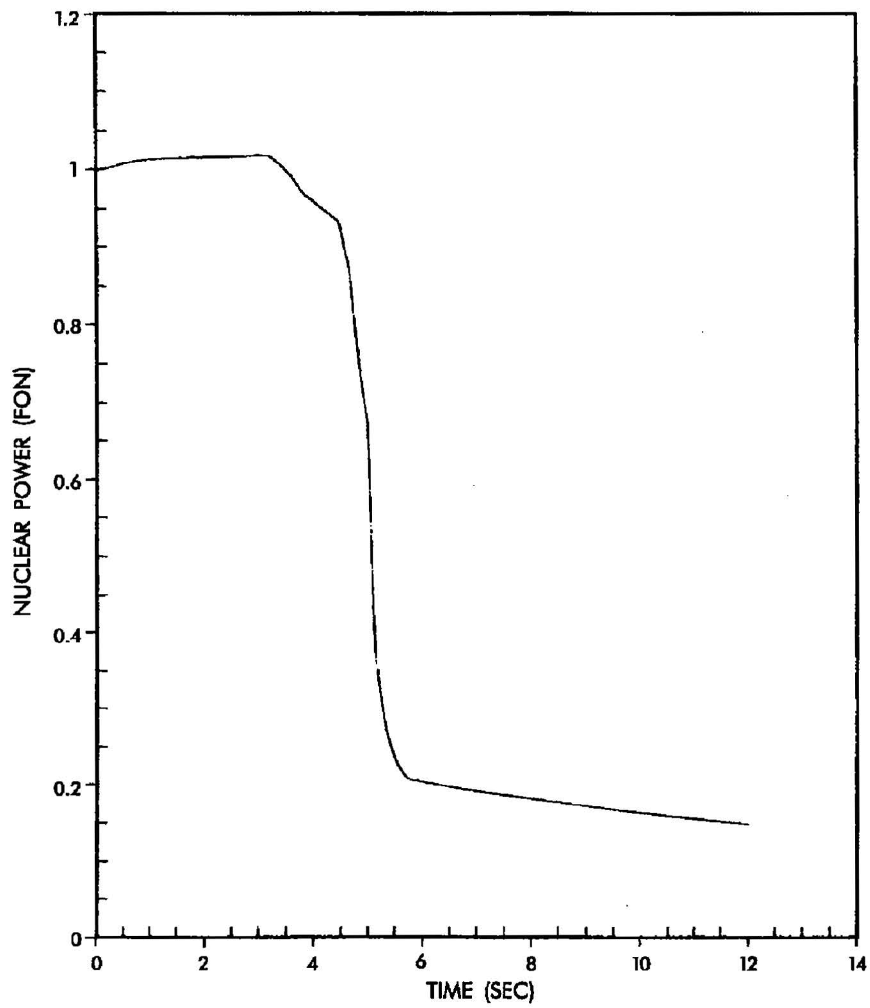


Figure 14.4-43
Nuclear Power vs. Time
Complete Loss of Flow, Two Pumps Coasting Down (CLOF)



NUCLEAR POWER vs. TIME
COMPLETE LOSS OF FLOW, TWO PUMPS COASTING DOWN (CLOF)

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE UI4443A.DGN		*FIGURE 14.4-43A REV. 31*

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FIGURE 14.4-43A

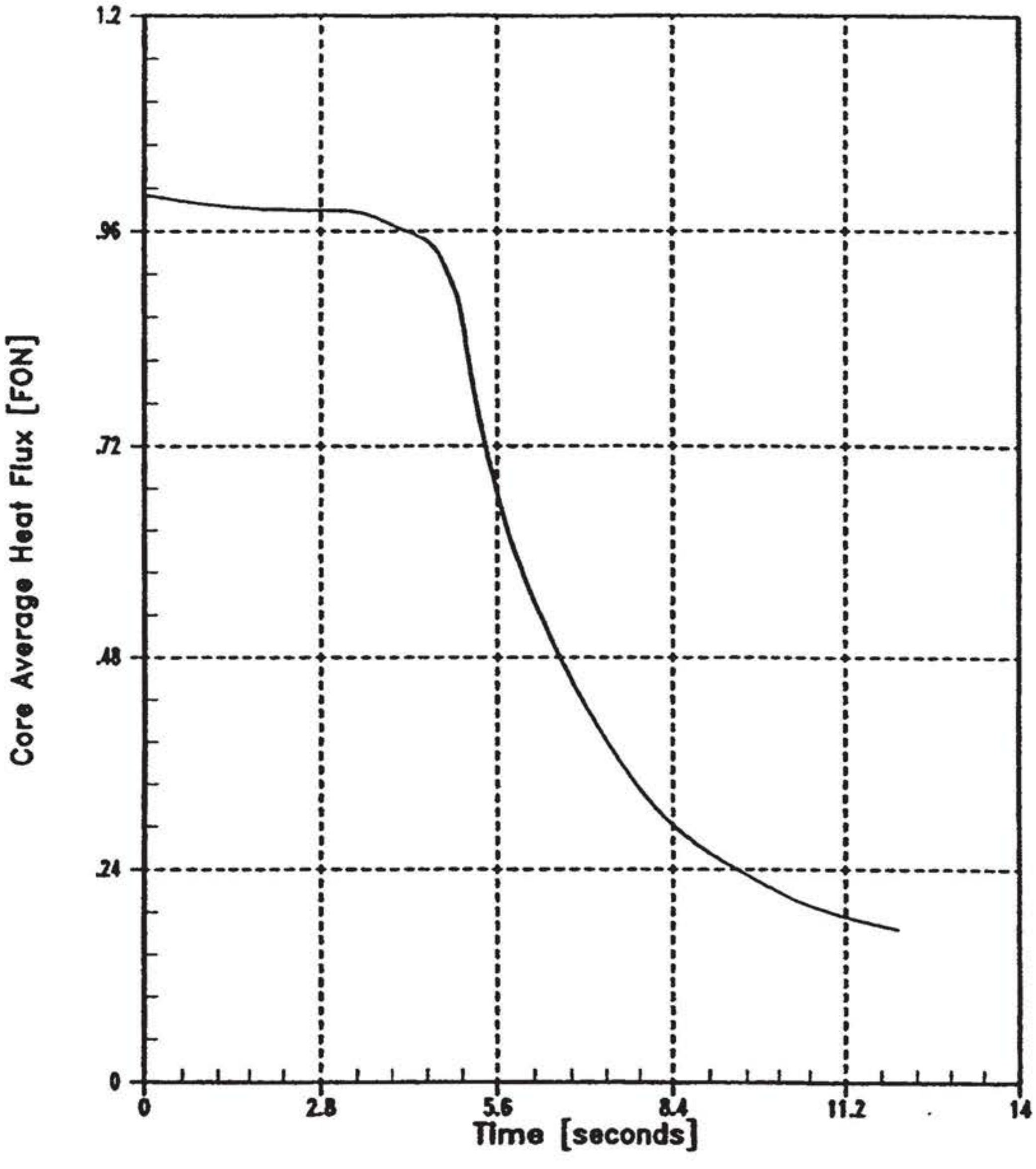
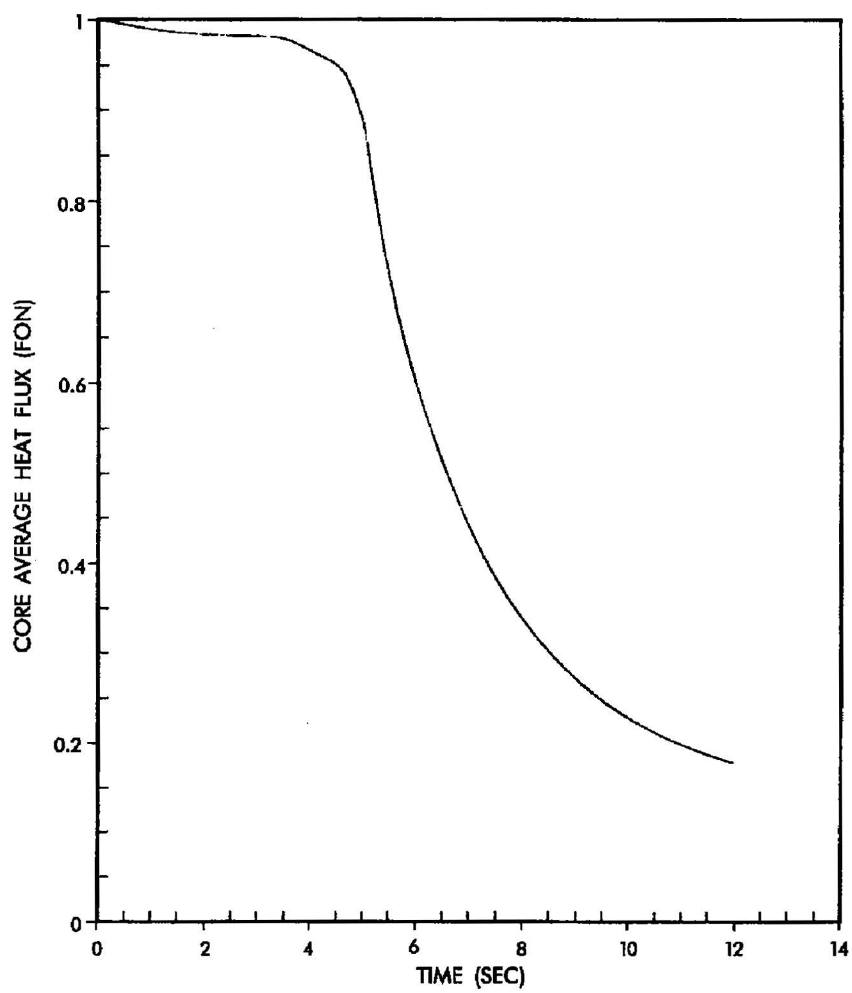


Figure 14.4-44
Core Average Heat Flux vs. Time
Complete Loss of Flow, Two Pumps Coasting Down (CLOF)



CORE AVERAGE HEAT FLUX vs. TIME
COMPLETE LOSS OF FLOW, TWO PUMPS COASTING DOWN (CLOF)

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14444A.DGN		*FIGURE 14.4-44A REV. 31*

01193868

FIGURE 14.4-44A

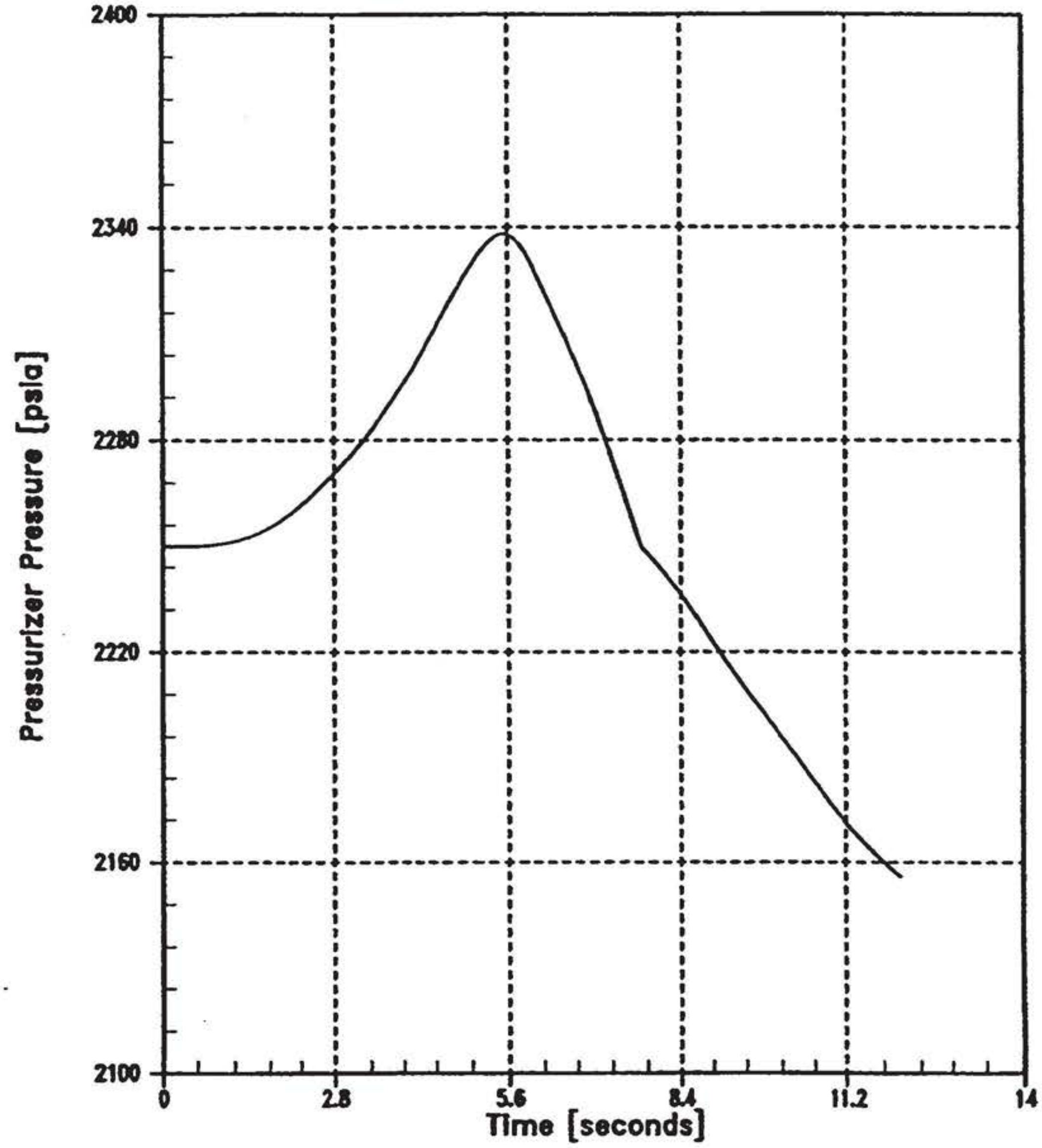


Figure 14.4-45
Pressurizer Pressure vs. Time
Complete Loss of Flow, Two Pumps Coasting Down (CLOF)

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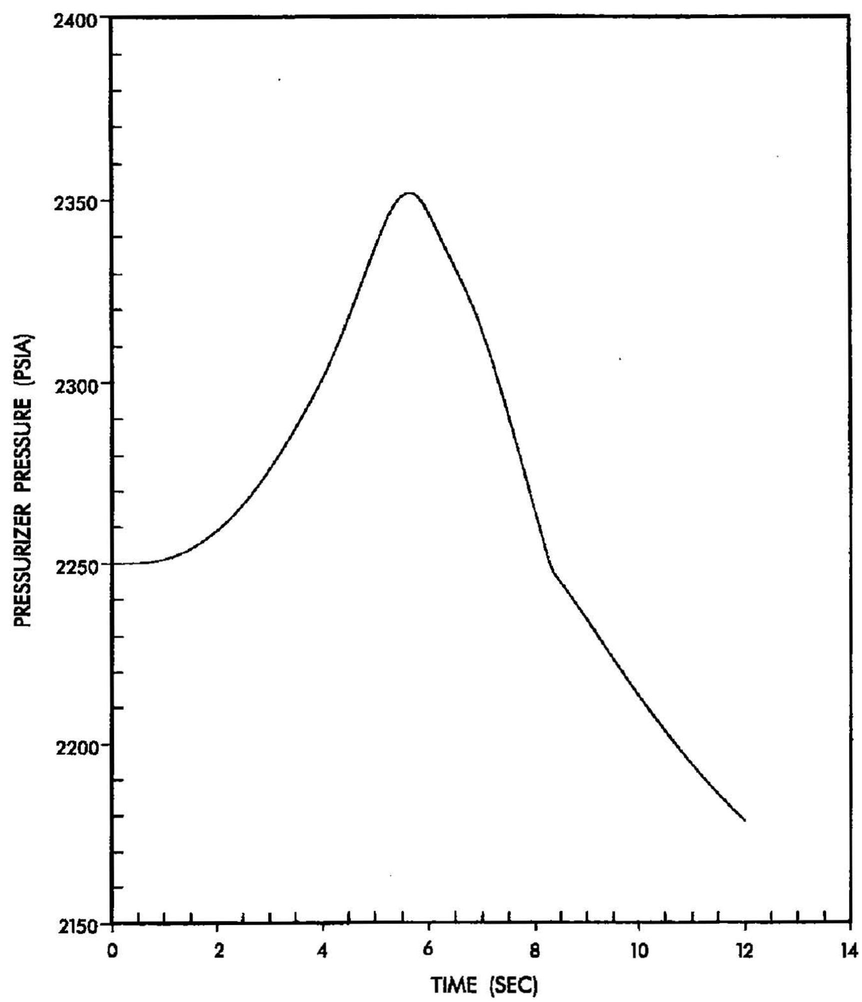


PRESSURIZER PRESSURE vs. TIME
COMPLETE LOSS OF FLOW, TWO PUMPS COASTING DOWN (CLOF)

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE NONE
CHECKED	CAD FILE U14445A.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-45A REV. 31*

01193868

FIGURE 14.4-45A

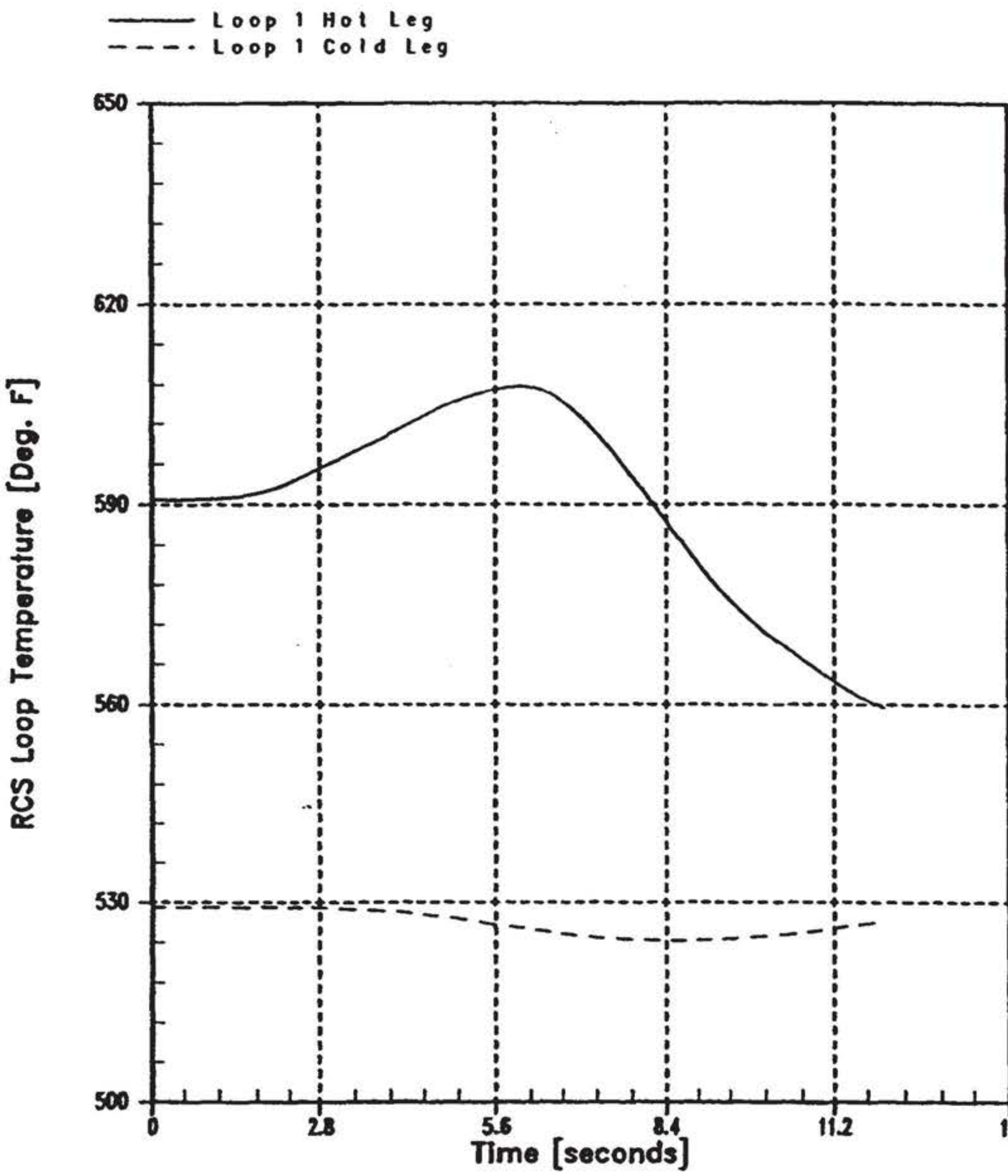
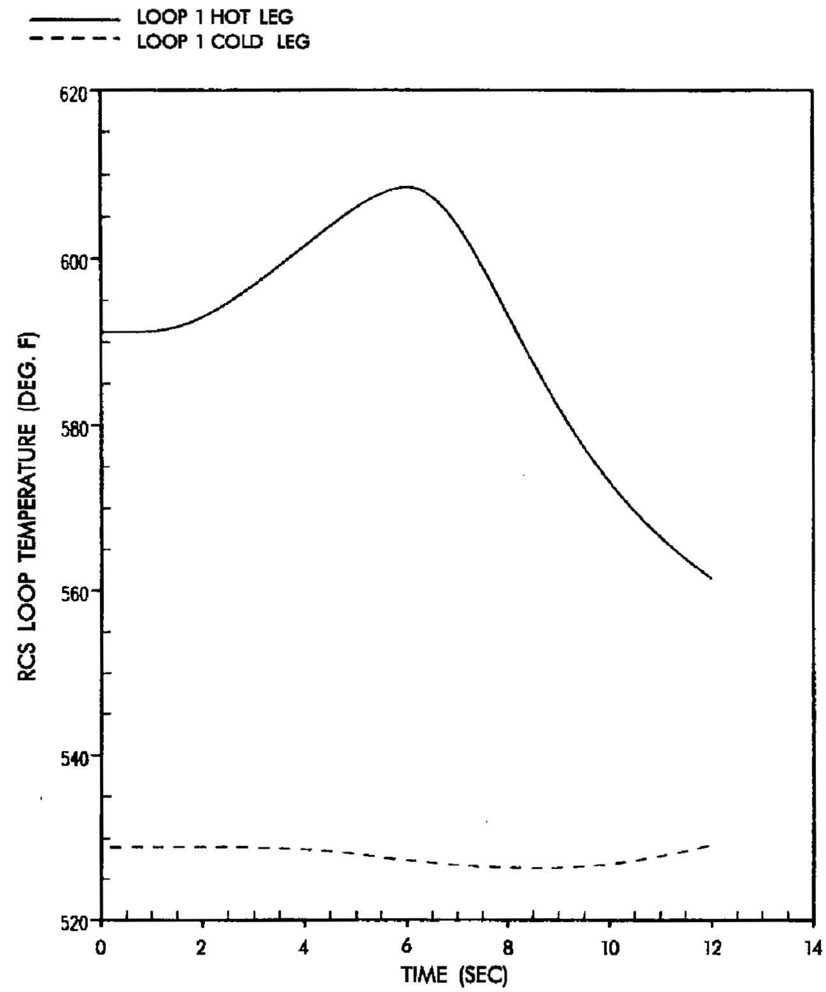


Figure 14.4-46
RCS Faulted Loop Temperature vs. Time
Complete Loss of Flow, Two Pumps Coasting Down (CLOF)



RCS FAULTED LOOP TEMPERATURE vs. TIME
COMPLETE LOSS OF FLOW, TWO PUMPS COASTING DOWN (CLOF)

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14446A.DGN		*FIGURE 14.4-46A REV. 31*

01193868

FIGURE 14.4-46A

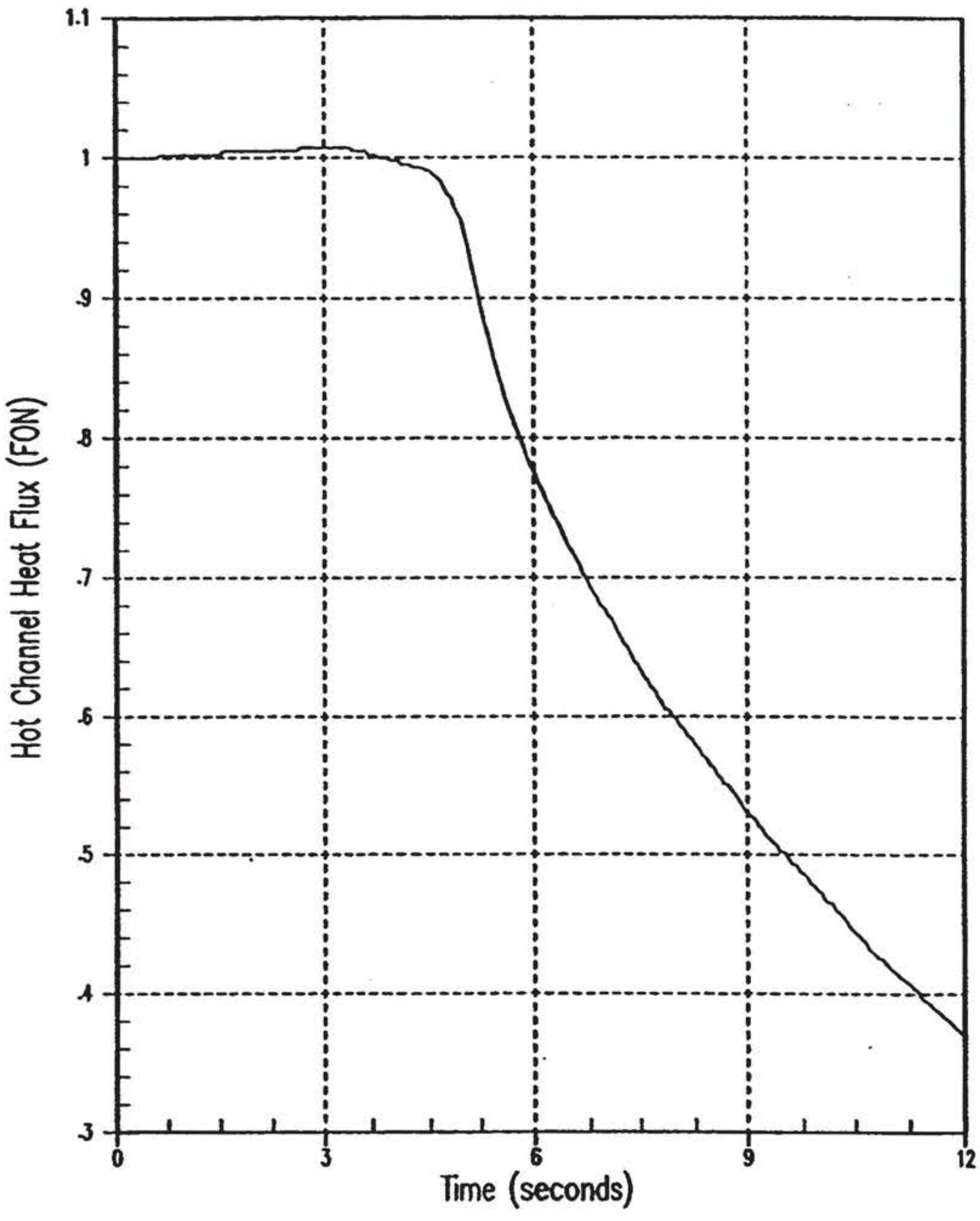
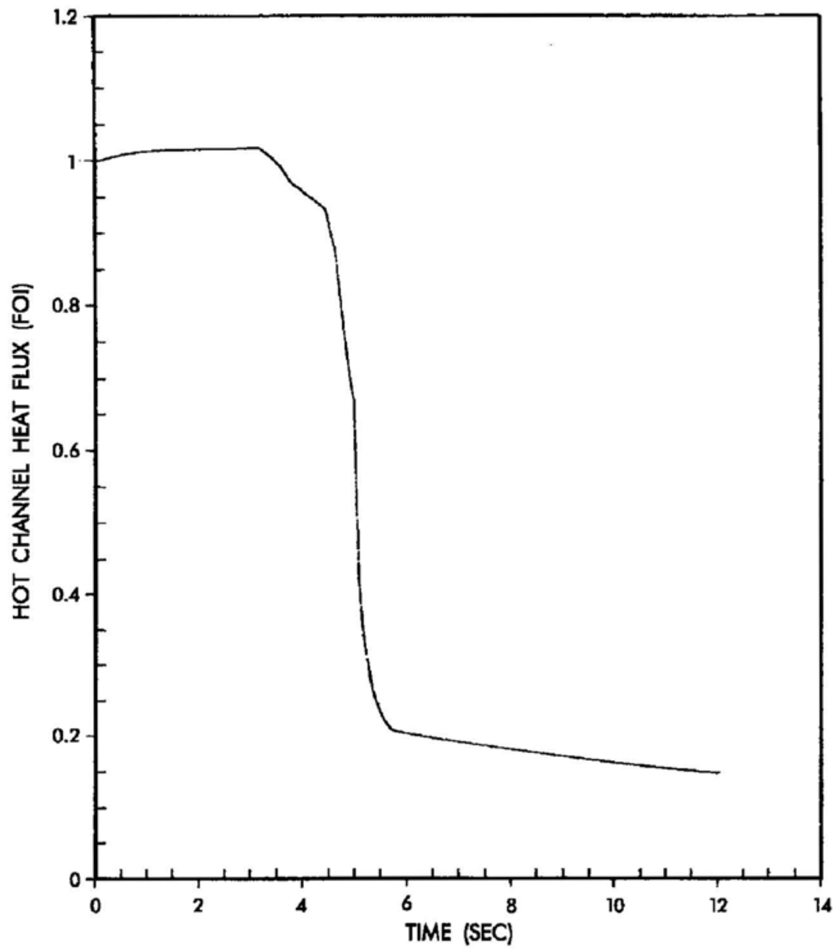


Figure 14.4-47
Hot Channel Heat Flux vs. Time
Complete Loss of Flow, Two Pumps Coasting Down (CLOF)



HOT CHANEL HEAT FLUX vs. TIME
COMPLETE LOSS OF FLOW, TWO PUMPS COASTING DOWN (CLOF)

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14447A.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-47A REV. 31*	

01193868

FIGURE 14.47A

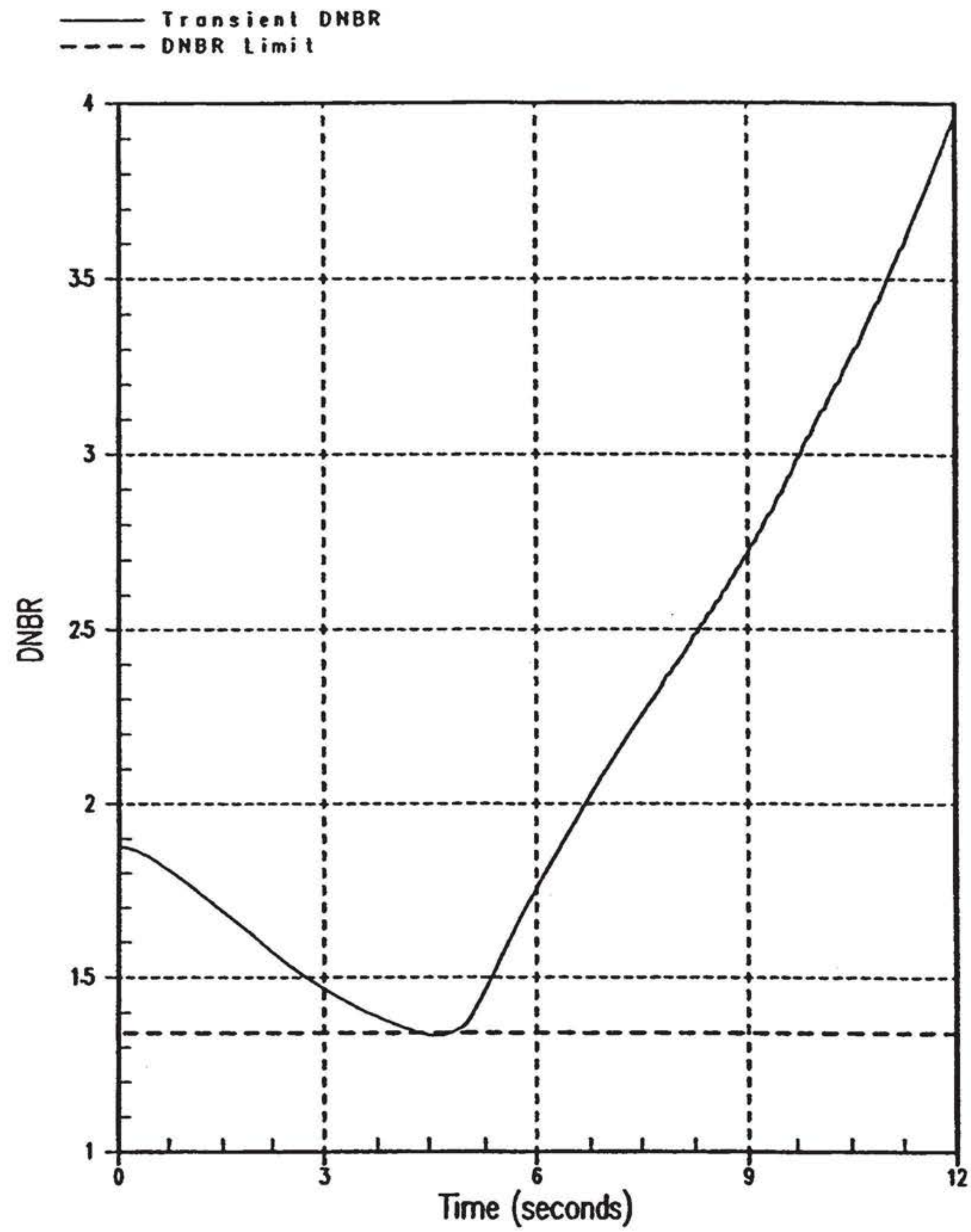
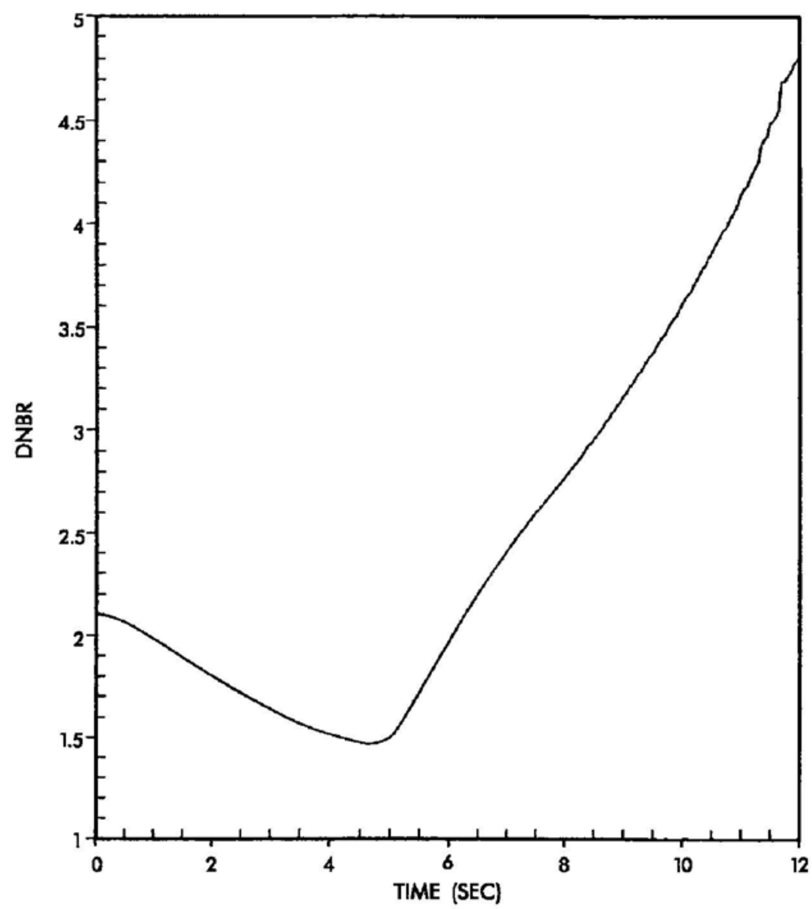


Figure 14.4-48
DNBR vs. Time
Complete Loss of Flow, Two Pumps Coasting Down (CLOF)



DNBR vs. TIME
COMPLETE LOSS OF FLOW, TWO PUMPS COASTING DOWN (CLOF)

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE	
CHECKED	CAD FILE U14448A.DGN		*FIGURE 14.4-48A REV. 31*	

01193868

FIGURE 14.4-48A

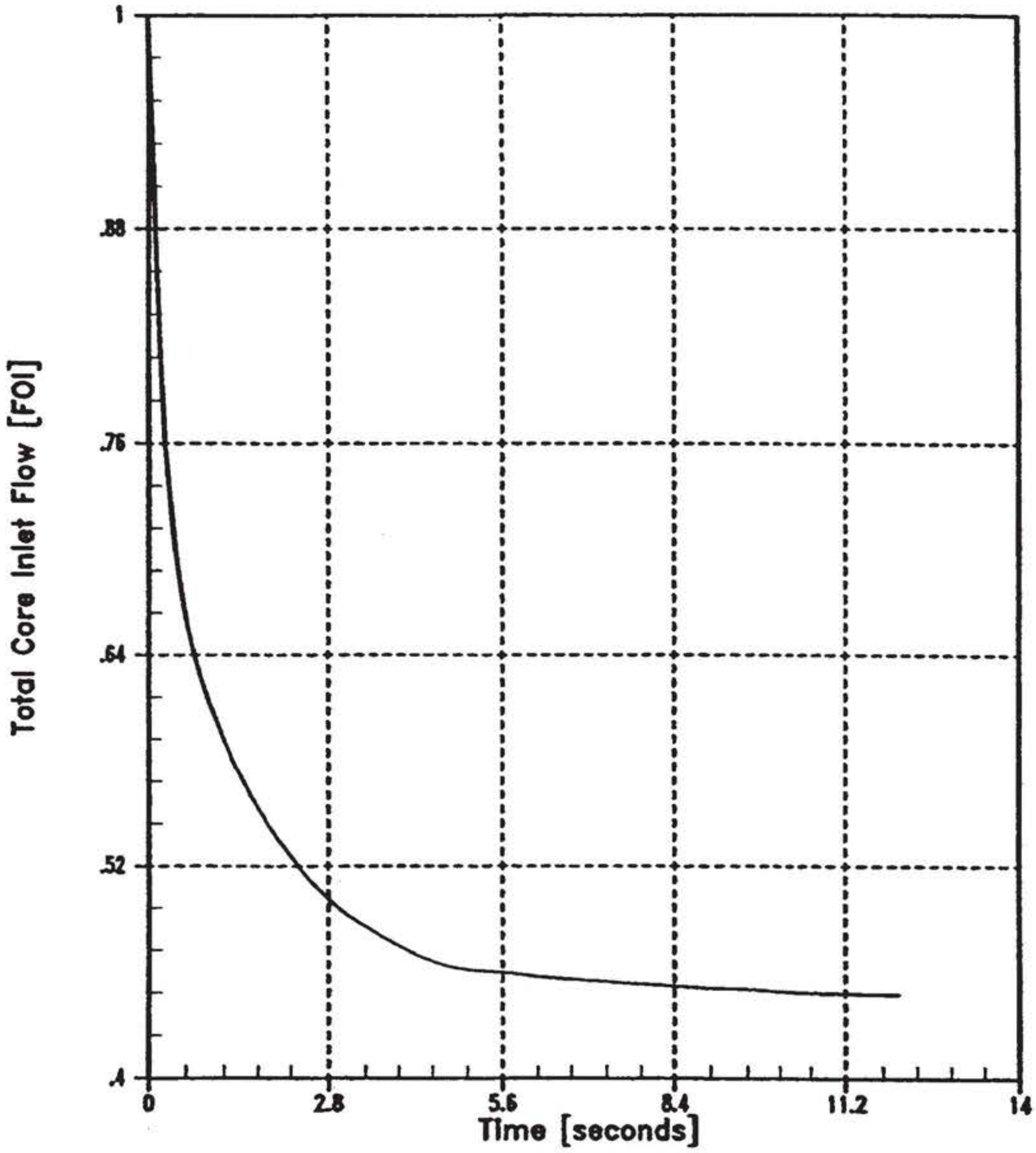
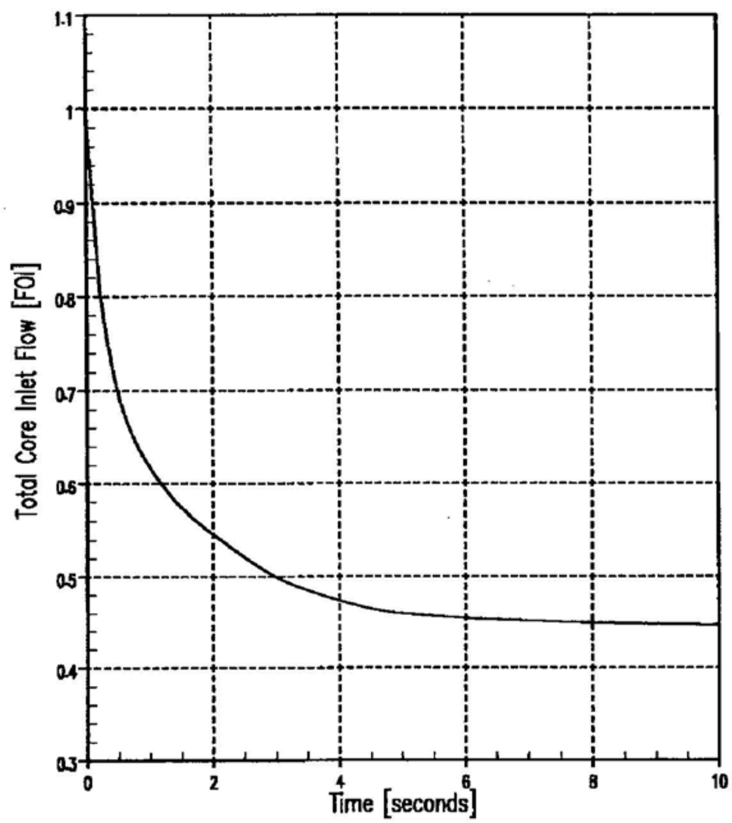


Figure 14.4-49
Total Core Inlet Flow vs. Time
Locked Rotor / Shaft Break – RCS Pressure / PCT Case



TOTAL CORE INLET FLOW vs. TIME
LOCKED ROTOR /SHAFT BREAK – RCS PRESSURE /PCT CASE

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE UI4449A.DGN		*FIGURE 14.4-49A REV. 31*

01193868

FIGURE 14.4-49A

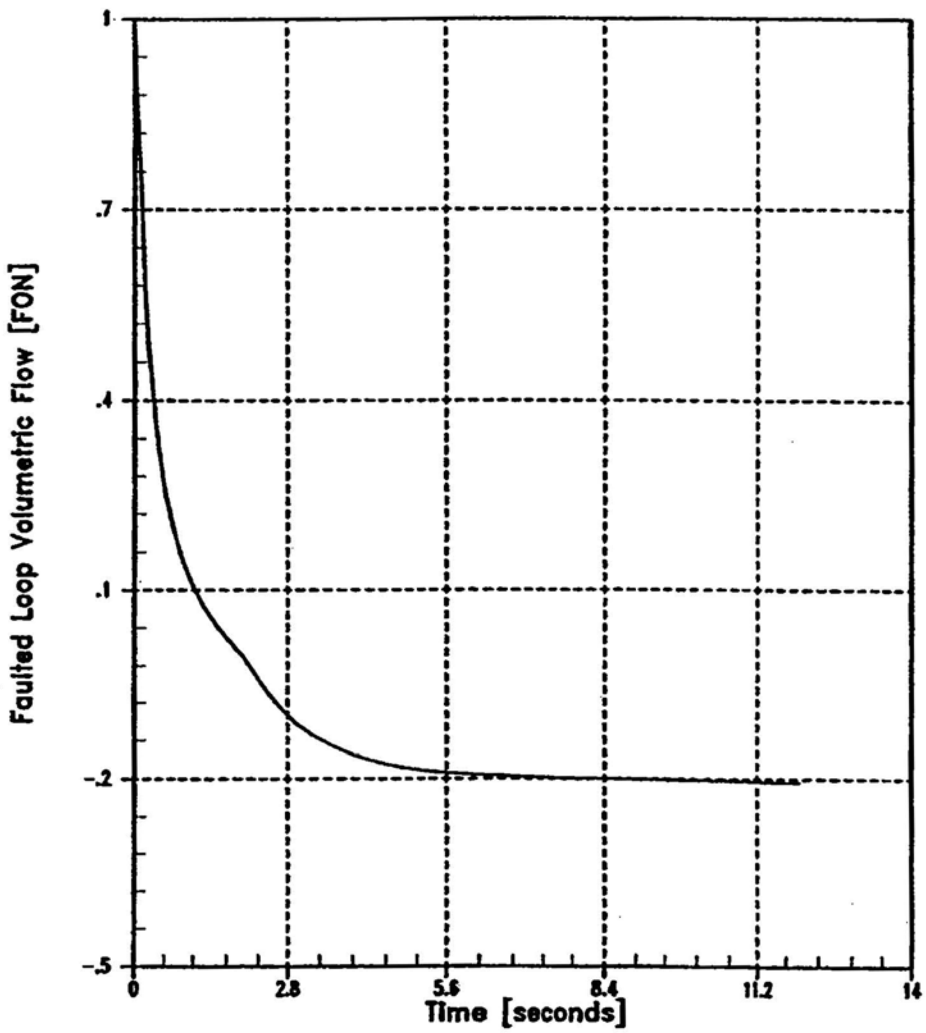
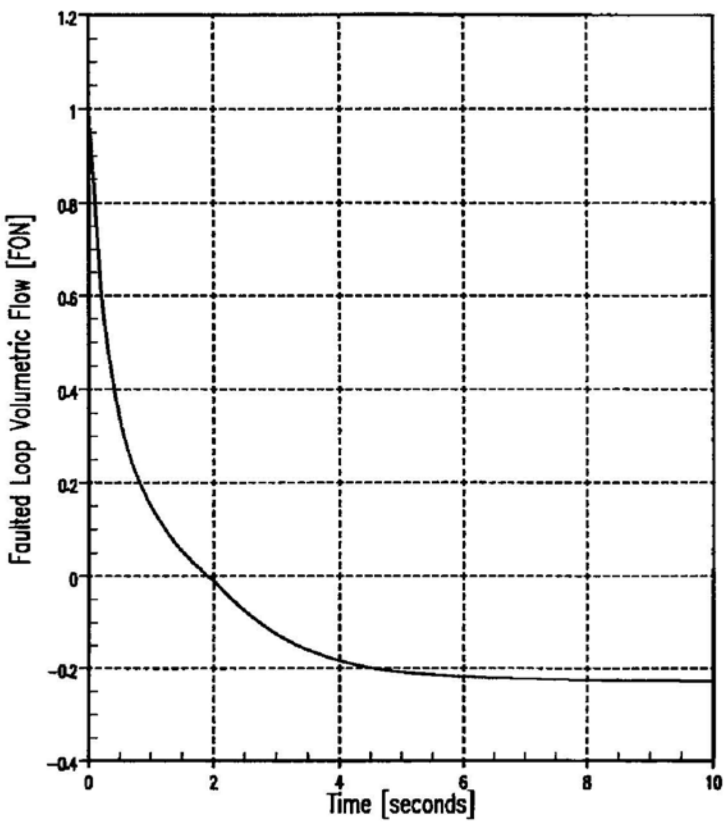


Figure 14.4-50
RCS Loop Flow vs. Time
Locked Rotor / Shaft Break – RCS Pressure / PCT Case



RCS LOOP FLOW vs. TIME
LOCKED ROTOR /SHAFT BREAK – RCS PRESSURE /PCT CASE

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14450A.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-50A REV. 31*

01193868

FIGURE 14.4-50A

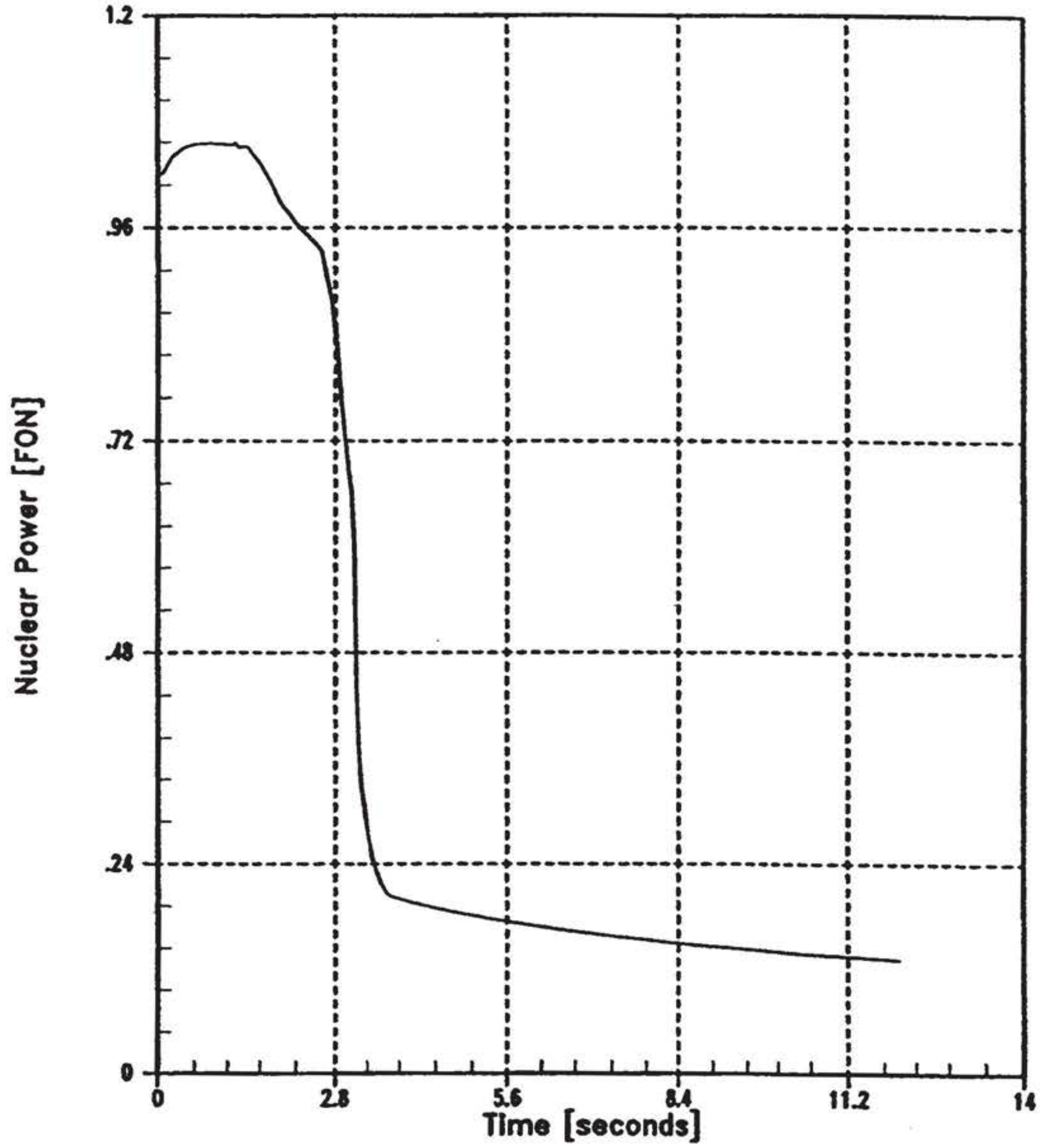
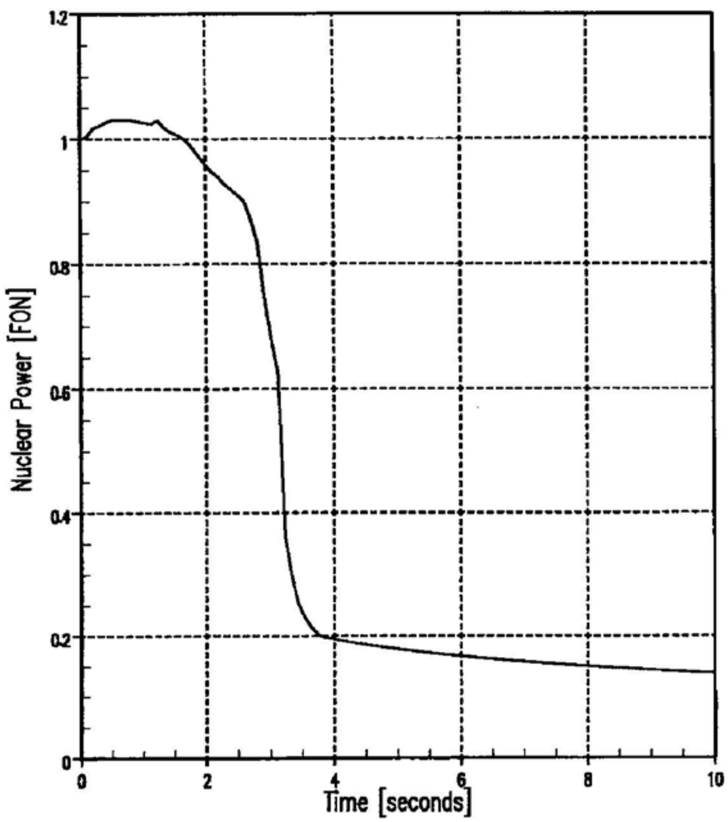


Figure 14.4-51
Nuclear Power vs. Time
Locked Rotor / Shaft Break – RCS Pressure / PCT Case



NUCLEAR POWER vs. TIME
LOCKED ROTOR /SHAFT BREAK – RCS PRESSURE /PCT CASE

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14451A.DGN		*FIGURE 14.4-51A REV. 31*

01193868

FIGURE 14.4-51A

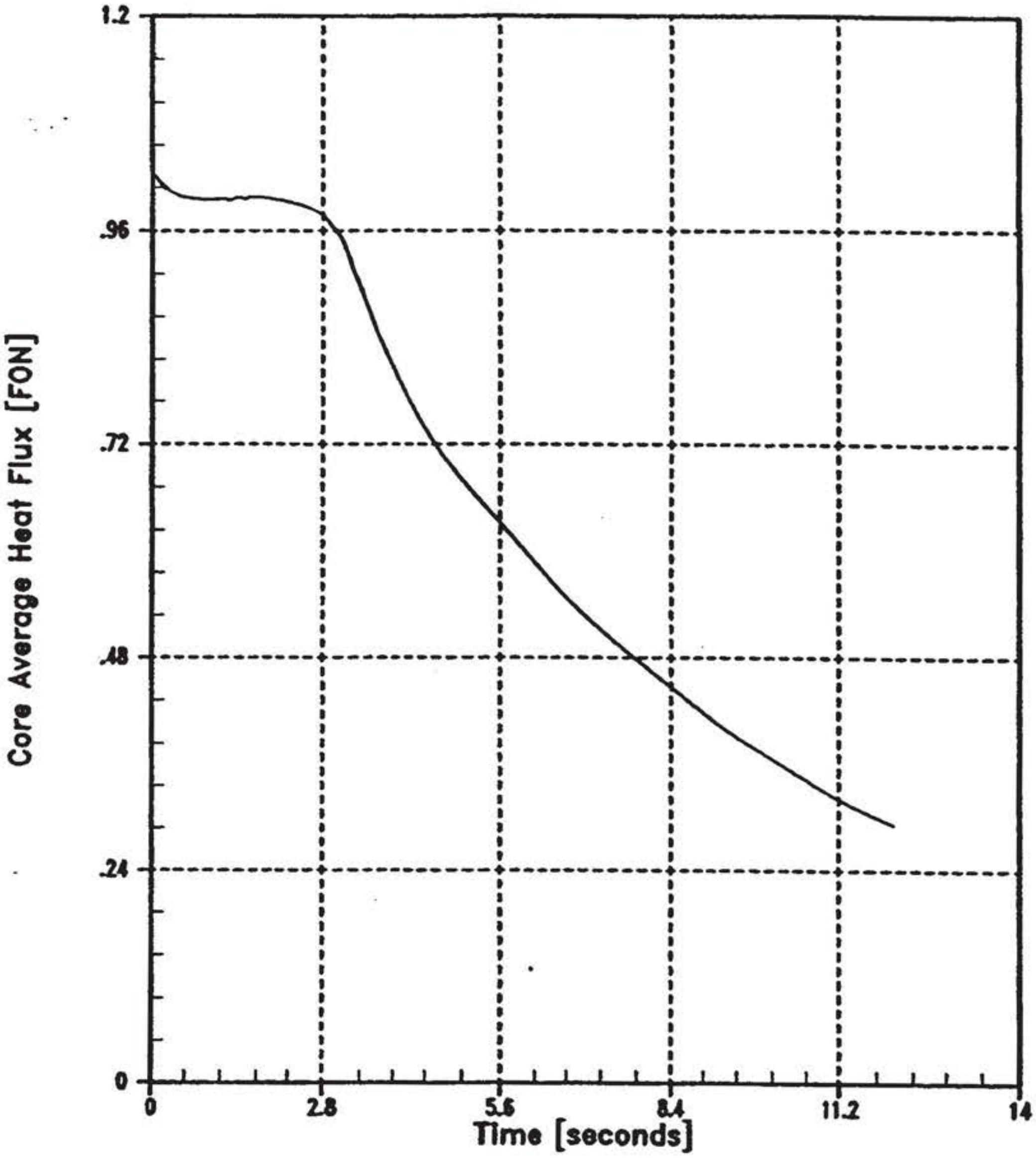
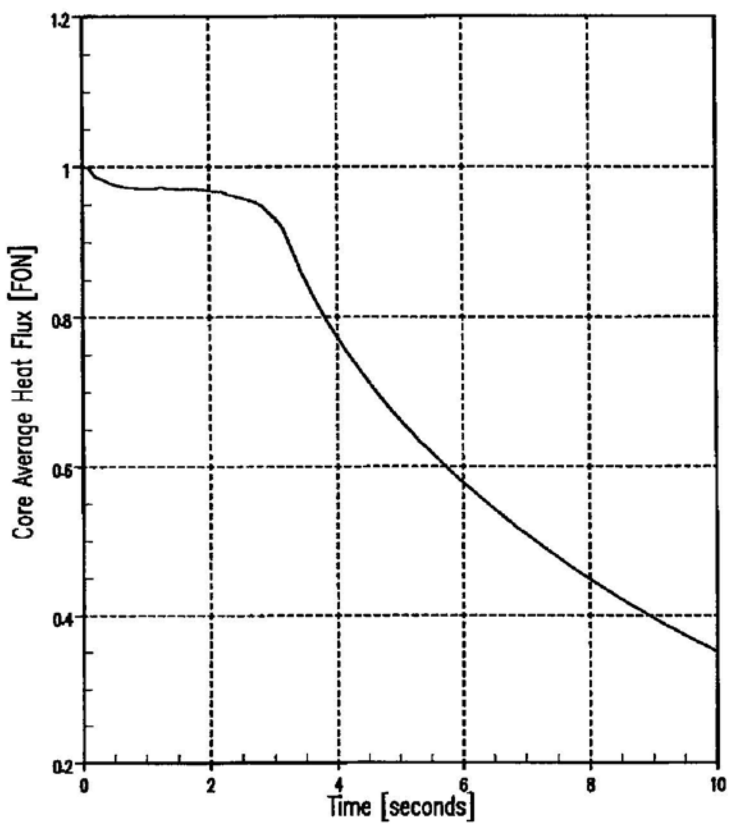


Figure 14.4-52
Core Average Heat Flux vs. Time
Locked Rotor / Shaft Break – RCS Pressure / PCT Case



CORE AVERAGE HEAT FLUX vs. TIME
LOCKED ROTOR /SHAFT BREAK – RCS PRESSURE /PCT CASE

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14452A.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-52A REV. 31*	

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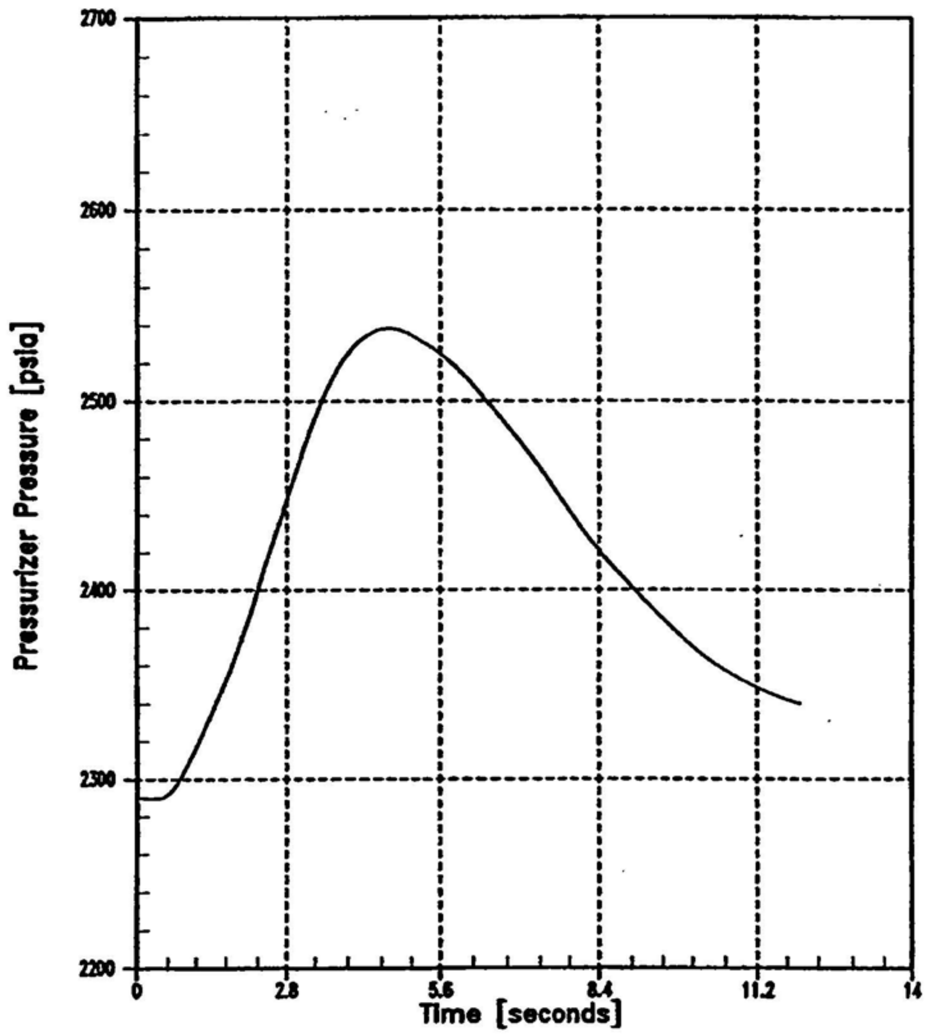
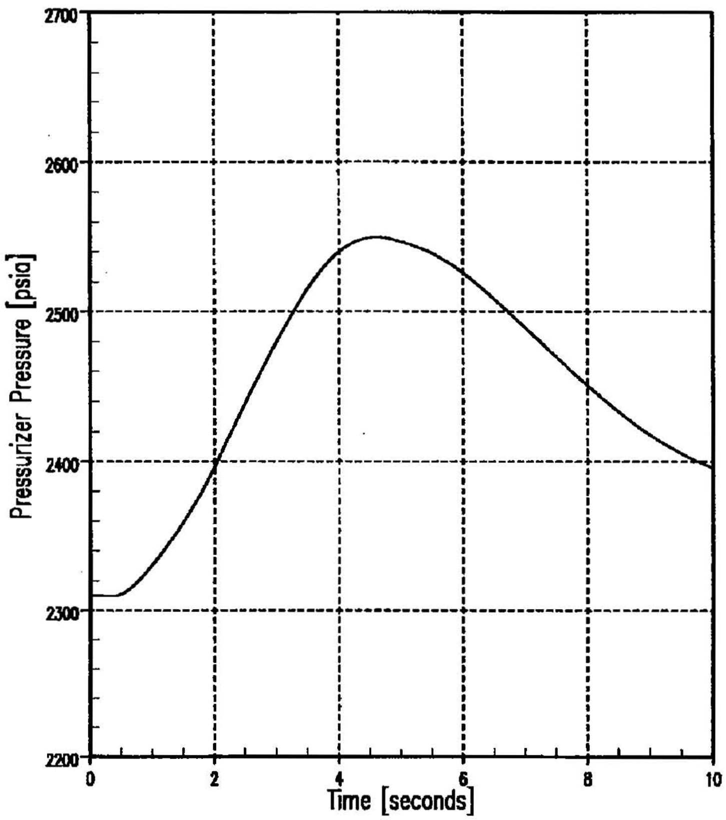


Figure 14.4-53
Pressurizer Pressure vs. Time
Locked Rotor / Shaft Break – RCS Pressure / PCT Case



PRESSURIZER PRESSURE vs. TIME
LOCKED ROTOR /SHAFT BREAK – RCS PRESSURE /PCT CASE

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14453A.DGN		*FIGURE 14.4-53A REV. 31*

FIGURE 14.4-53A

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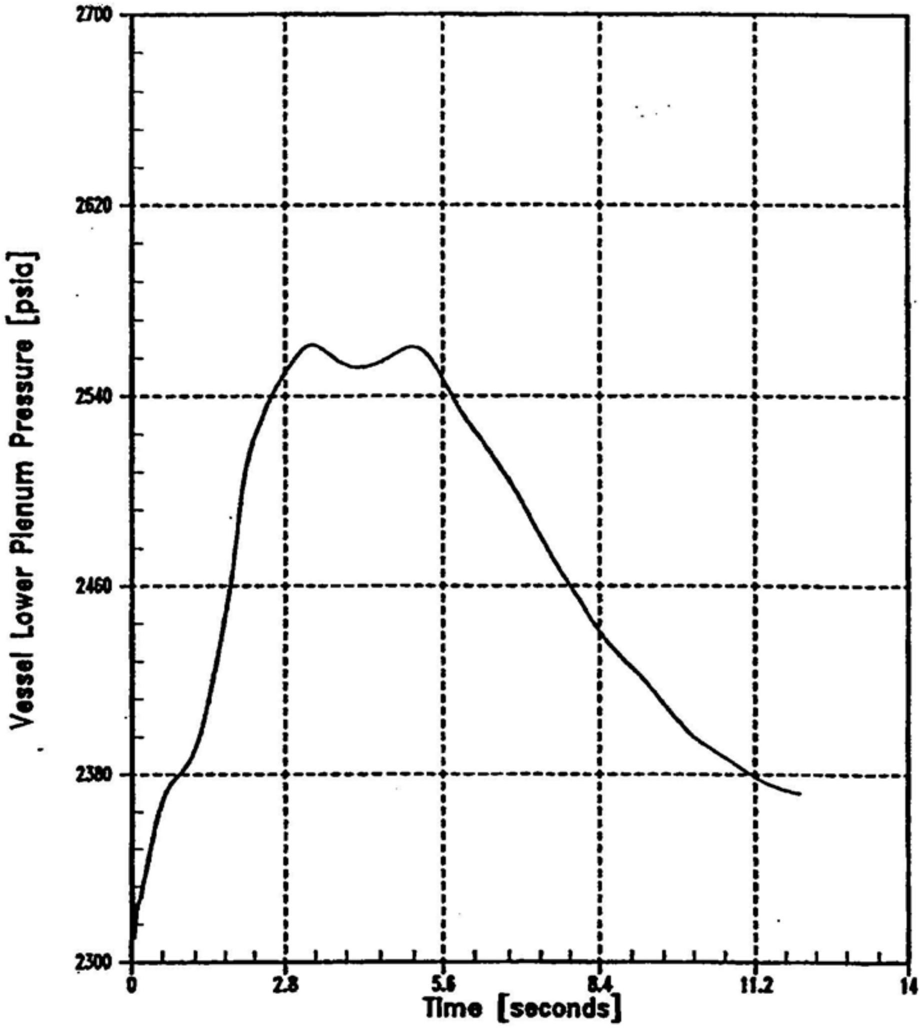
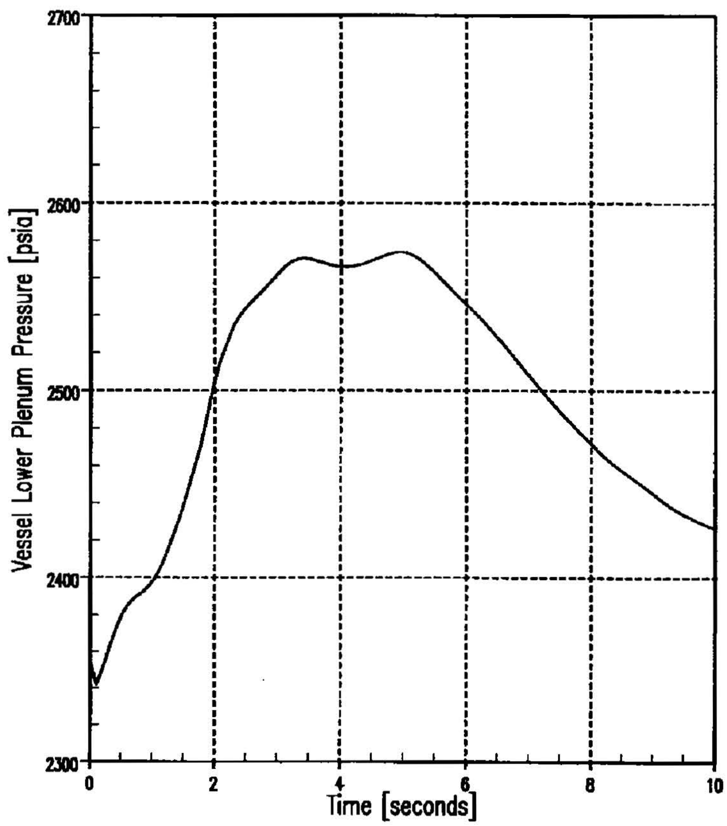


Figure 14.4-54
Vessel Lower Plenum Pressure vs. Time
Locked Rotor / Shaft Break – RCS Pressure / PCT Case

Rev. 27



VESSEL LOWER PLENUM PRESSURE vs. TIME
LOCKED ROTOR /SHAFT BREAK – RCS PRESSURE /PCT CASE

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14454A.DGN		*FIGURE 14.4-54A REV. 31*

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FIGURE 14.4-54A

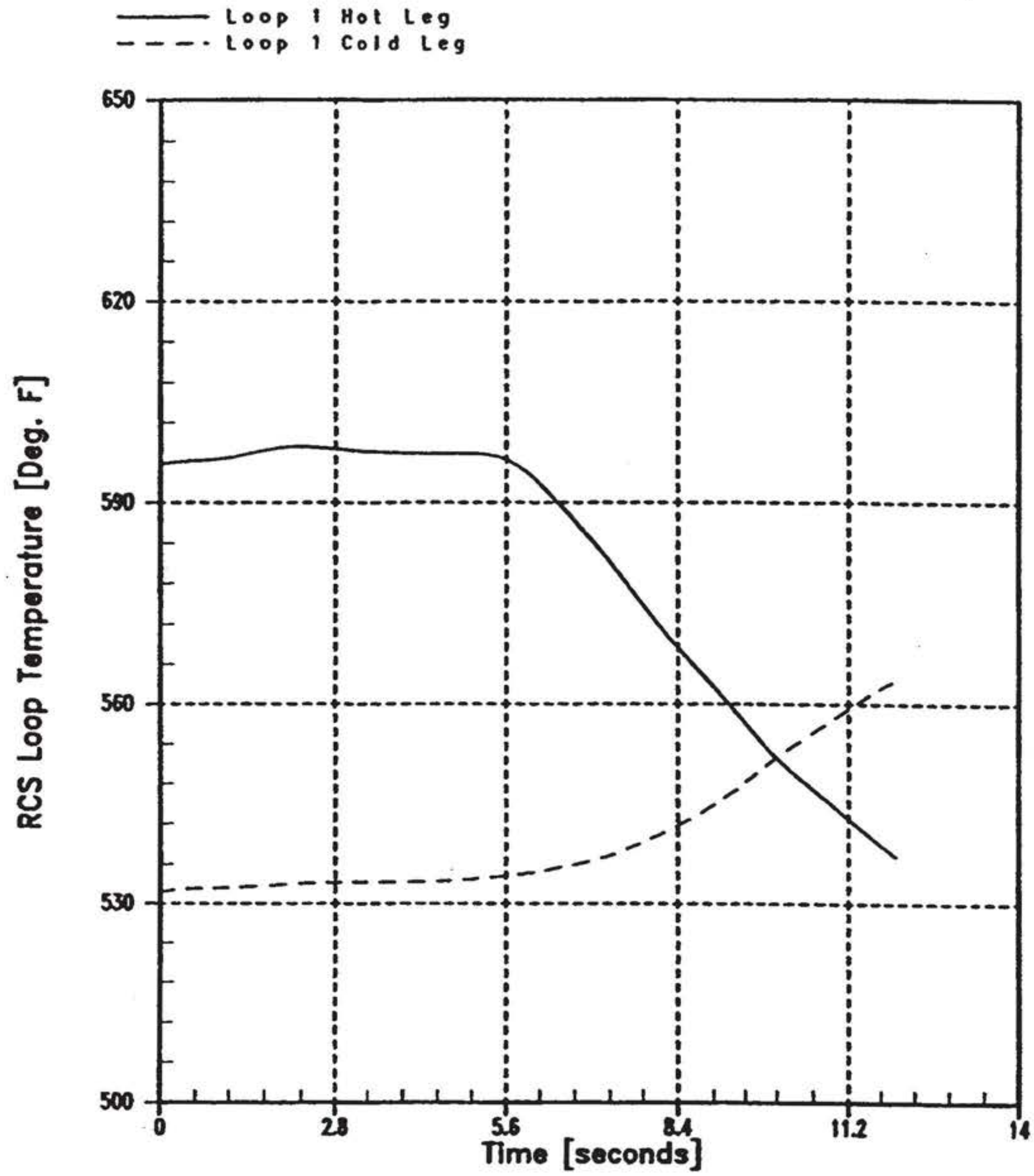
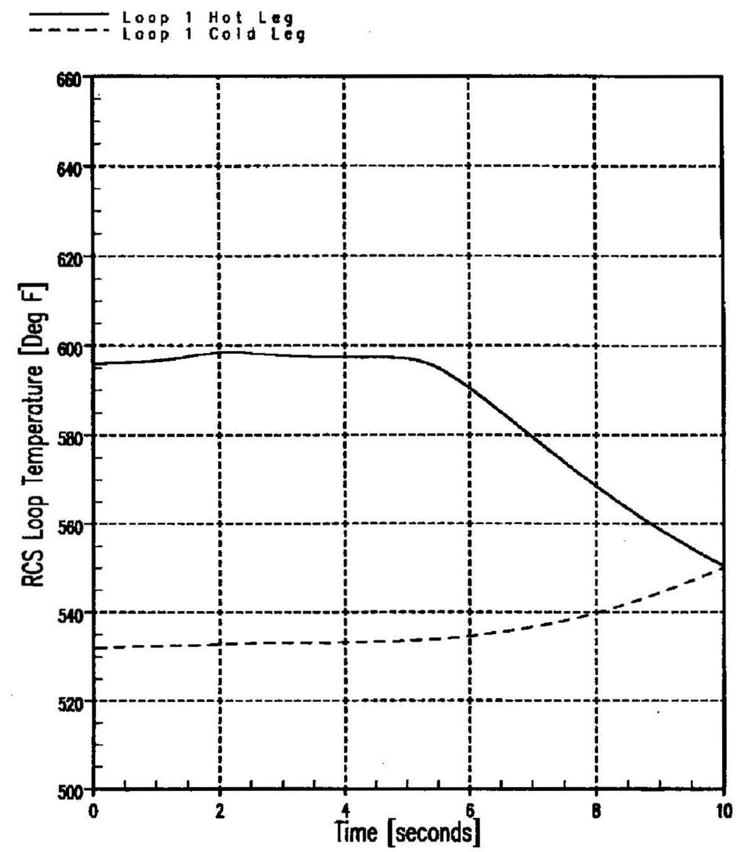


Figure 14.4-55
RCS Loop Temperature vs. Time
Locked Rotor / Shaft Break – RCS Pressure / PCT Case



RCS LOOP TEMPERATURE vs. TIME
LOCKED ROTOR /SHAFT BREAK – RCS PRESSURE /PCT CASE

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14455A.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-55A REV. 31*	

01193868

FIGURE 14.4-55A

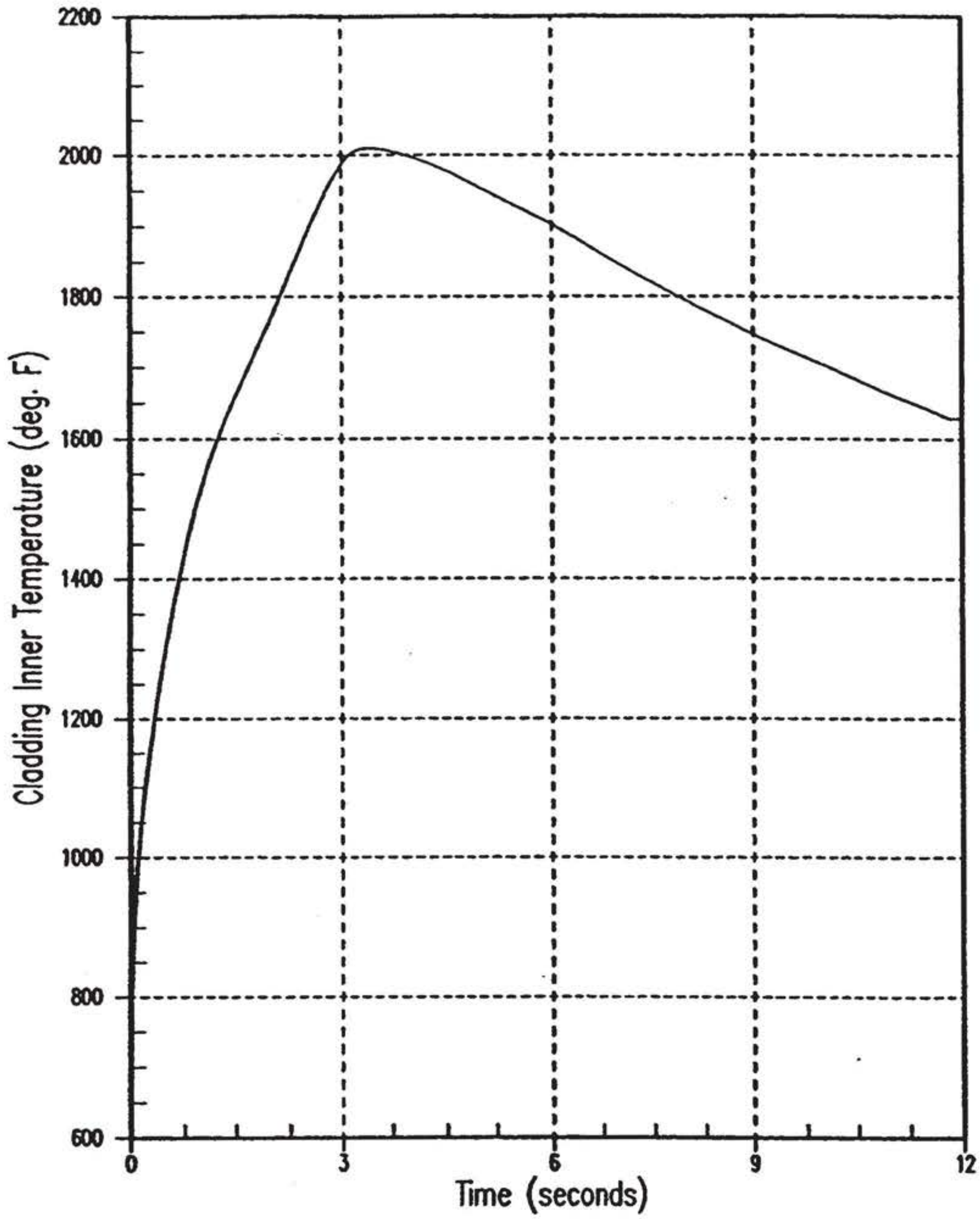
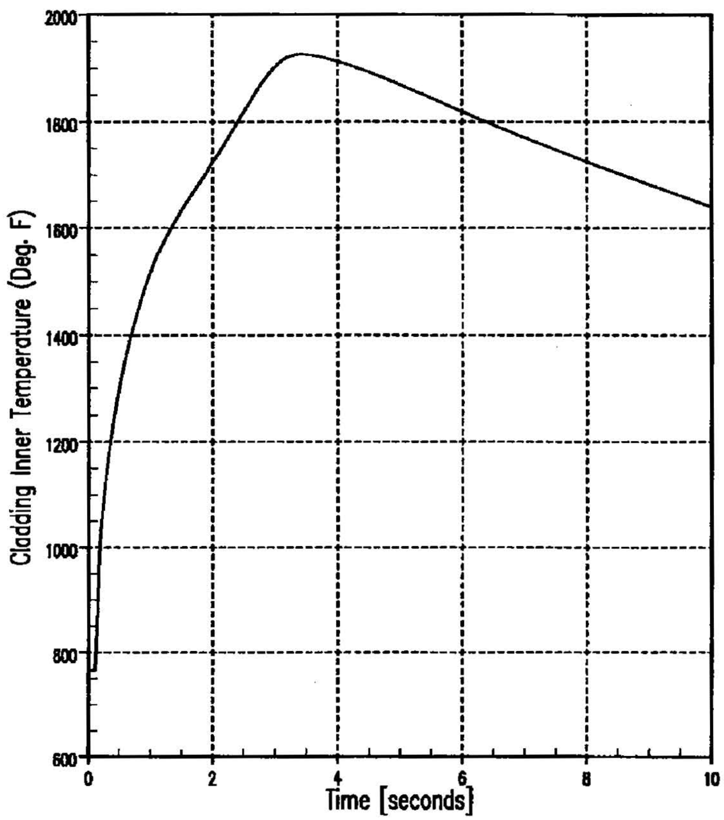


Figure 14.4-56
Hot Spot Cladding Inner Temperature vs. Time
Locked Rotor / Shaft Break – RCS Pressure / PCT Case

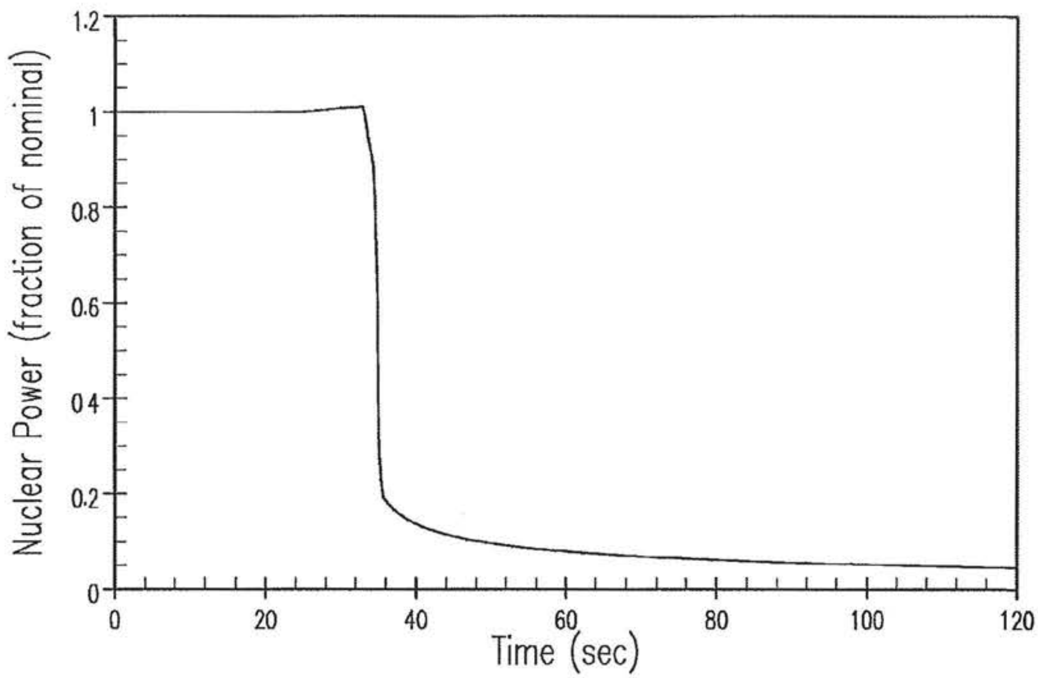


HOT SPOT CLADDING INNER TEMPERATURE vs. TIME
LOCKED ROTOR /SHAFT BREAK – RCS PRESSURE /PCT CASE

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14456A.DGN		*FIGURE 14.4-56A REV. 31*

01193868

FIGURE 14.4-56A



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LOSS OF EXTERNAL ELECTRICAL LOAD WITH AUTOMATIC PRESSURE CONTROL – NUCLEAR POWER VERSUS TIME

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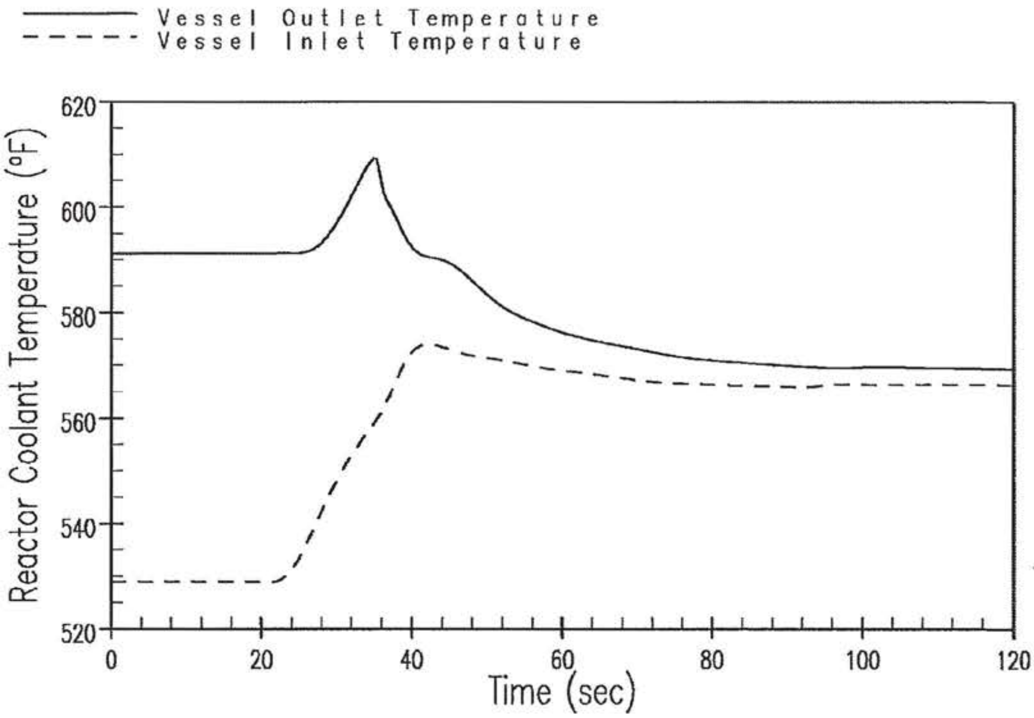
CHECKED:	CAD FILE: UI4457.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-57*	*REV. 33*



LOSS OF EXTERNAL ELECTRICAL LOAD WITH AUTOMATIC PRESSURE CONTROL – VESSEL INLET AND OUTLET TEMPERATURES VERSUS TIME

DWN: KJF	DATE: 2-19-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED:	CAD FILE: U14458.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-58* *REV. 33*

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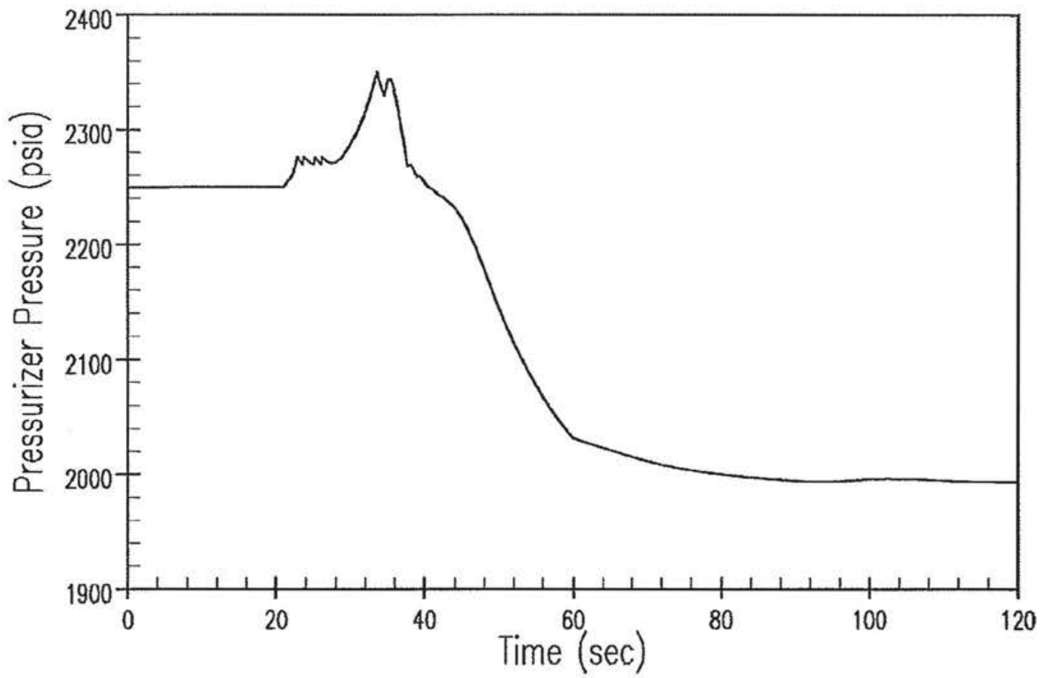

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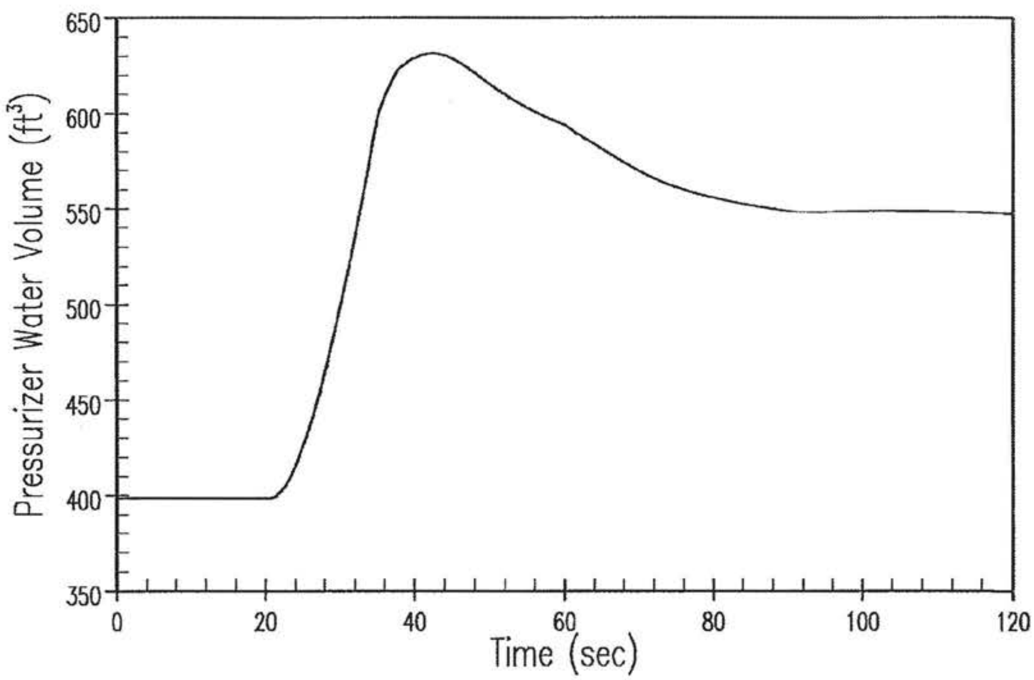
LOSS OF EXTERNAL ELECTRICAL LOAD WITH AUTOMATIC PRESSURE CONTROL – PRESSURIZER PRESSURE VERSUS TIME

DWN: KJF	DATE: 2-19-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U14459.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-59*	*REV. 33*



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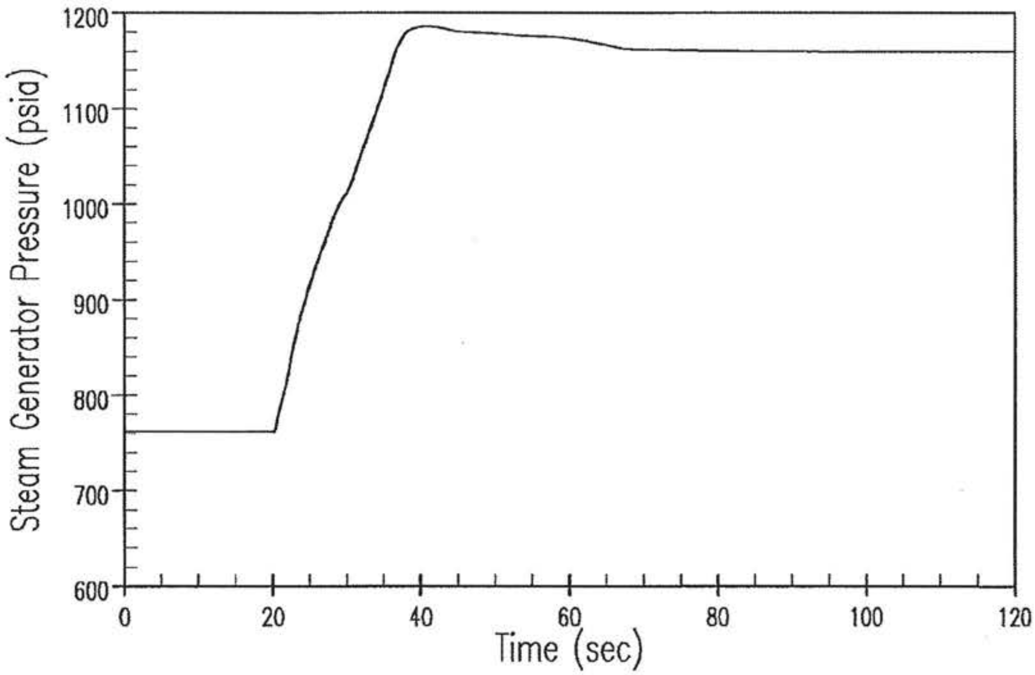
LOSS OF EXTERNAL ELECTRICAL LOAD WITH AUTOMATIC PRESSURE CONTROL – PRESSURIZER WATER VOLUME VERSUS TIME

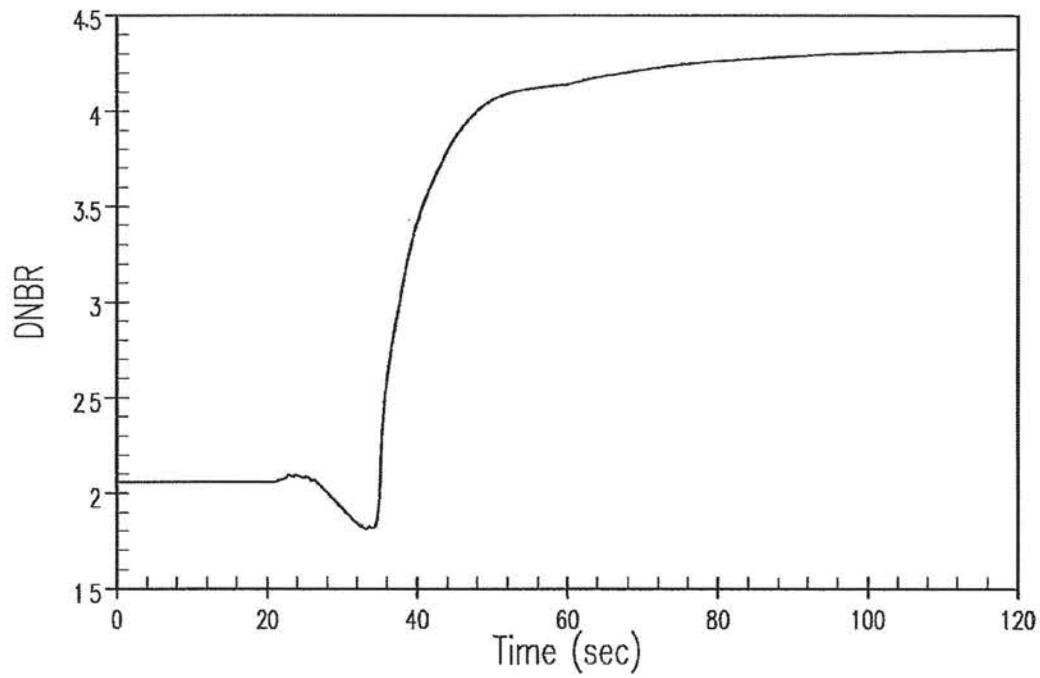
DWN: KJF	DATE: 2-19-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U14460.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	FIGURE 14.4-60	REV. 33



LOSS OF EXTERNAL ELECTRICAL LOAD WITH AUTOMATIC PRESSURE CONTROL – STEAM GENERATOR PRESSURE VERSUS TIME

DWN: KJF	DATE: 2-19-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U14461.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-61*	*REV. 33*

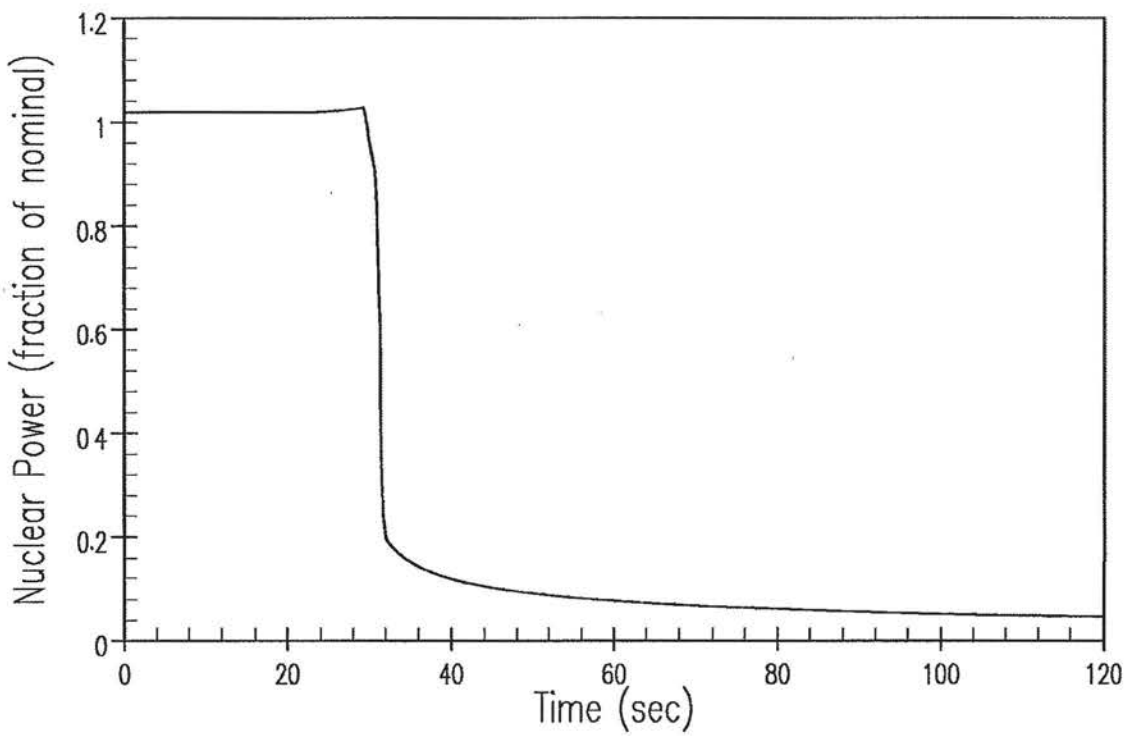
01386642



LOSS OF EXTERNAL ELECTRICAL LOAD WITH AUTOMATIC PRESSURE CONTROL – DNBR VERSUS TIME

DWN: KJF	DATE: 2-19-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U14462.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-62*	*REV. 33*

01386642



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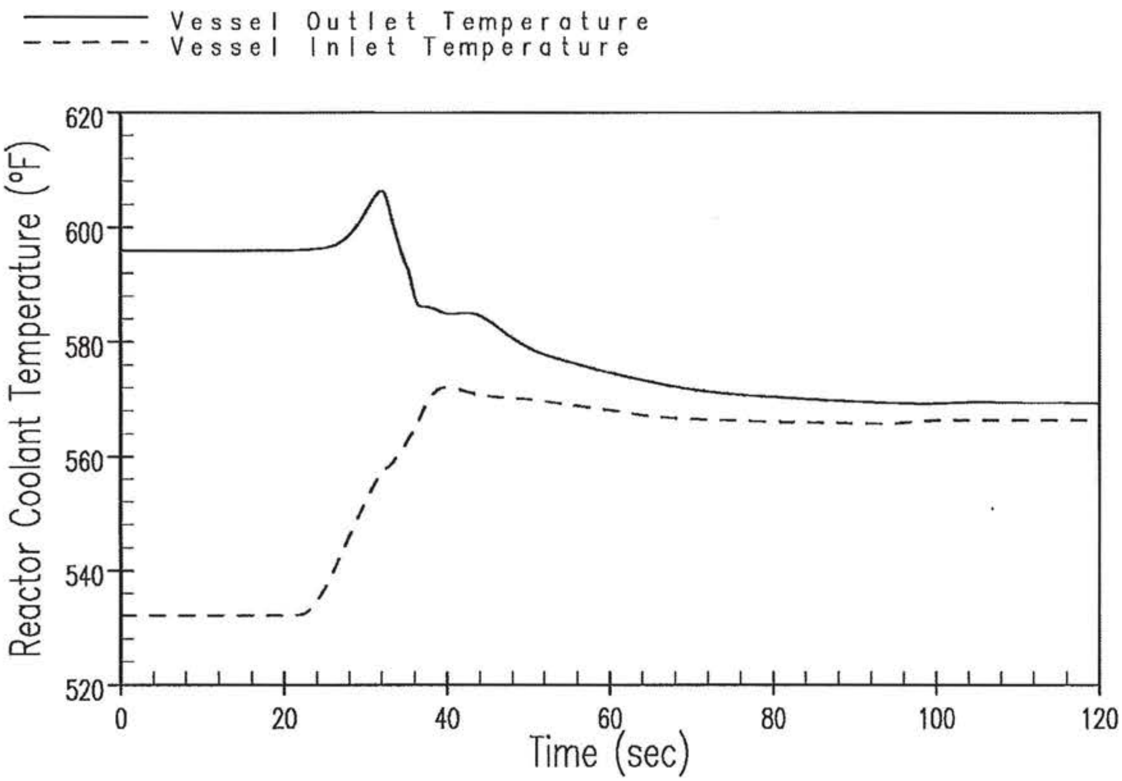
LOSS OF EXTERNAL ELECTRICAL LOAD WITHOUT AUTOMATIC PRESSURE CONTROL – NUCLEAR POWER VERSUS TIME

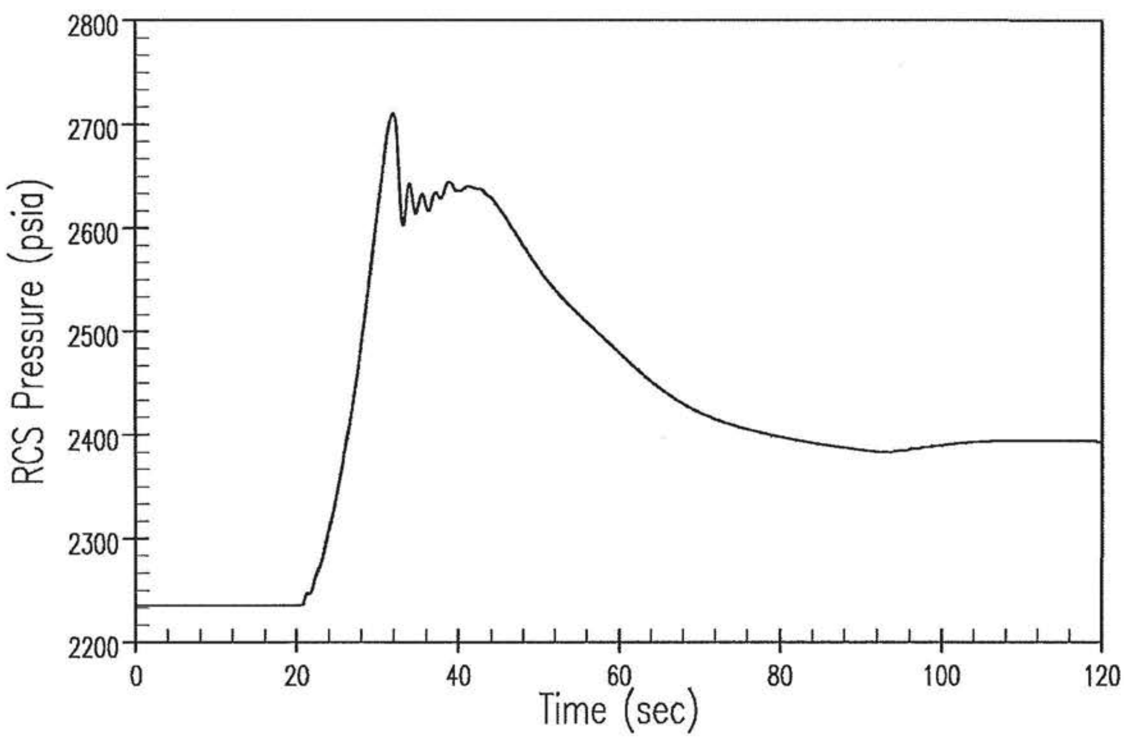
DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14463.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-63 REV. 33*



LOSS OF EXTERNAL ELECTRICAL LOAD WITHOUT AUTOMATIC PRESSURE CONTROL – VESSEL INLET AND OUTLET TEMPERATURES VERSUS TIME

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14464.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-64 REV. 33*

01386642



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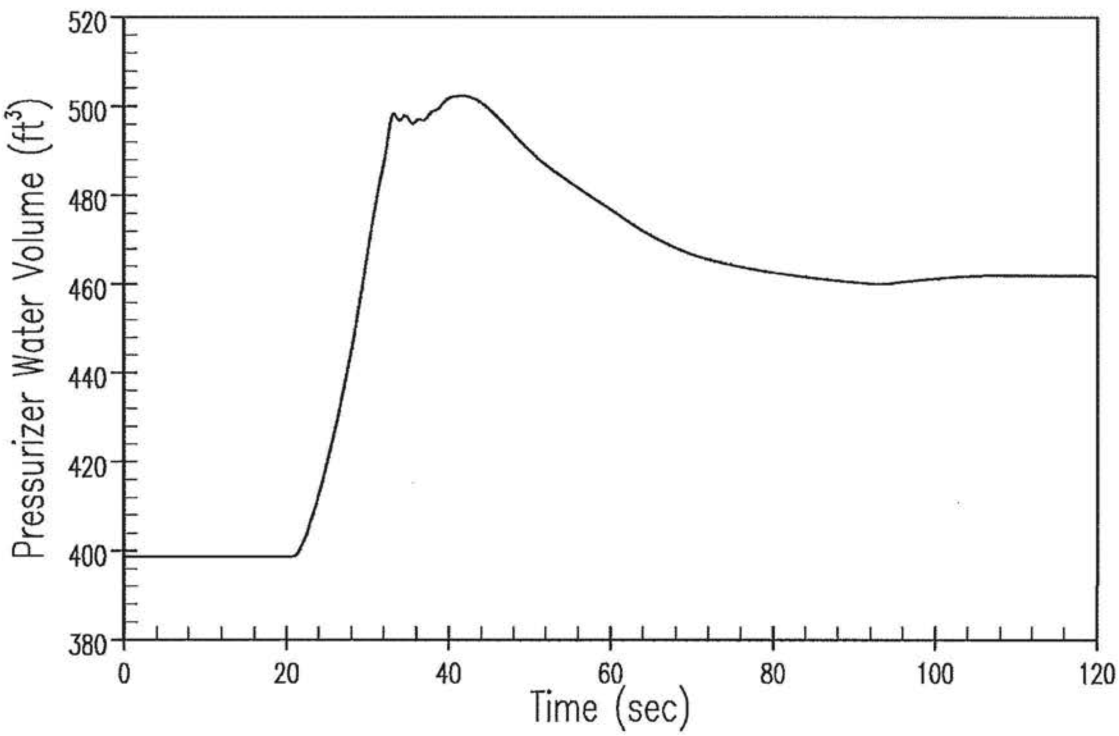
LOSS OF EXTERNAL ELECTRICAL LOAD WITHOUT AUTOMATIC PRESSURE CONTROL – RCS PRESSURE VERSUS TIME

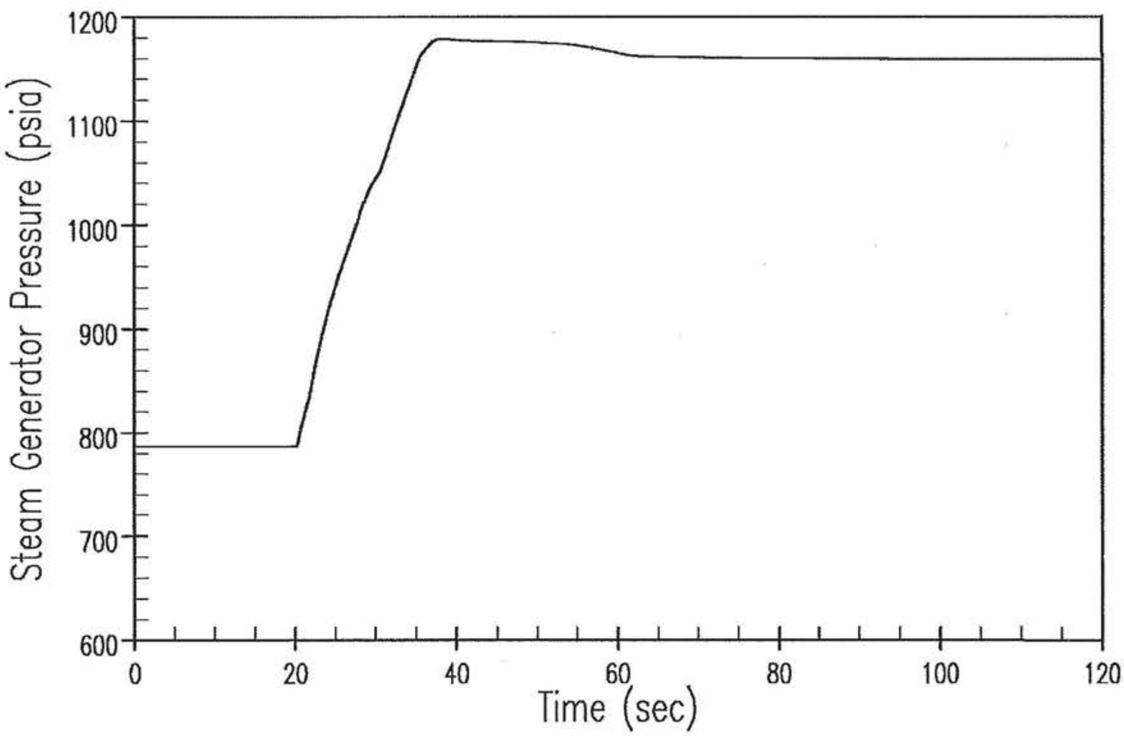
DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14465.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-65 REV. 33*



LOSS OF EXTERNAL ELECTRICAL LOAD WITHOUT AUTOMATIC PRESSURE CONTROL – PRESSURIZER WATER VOLUME VERSUS TIME

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14466.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-66 REV. 33*

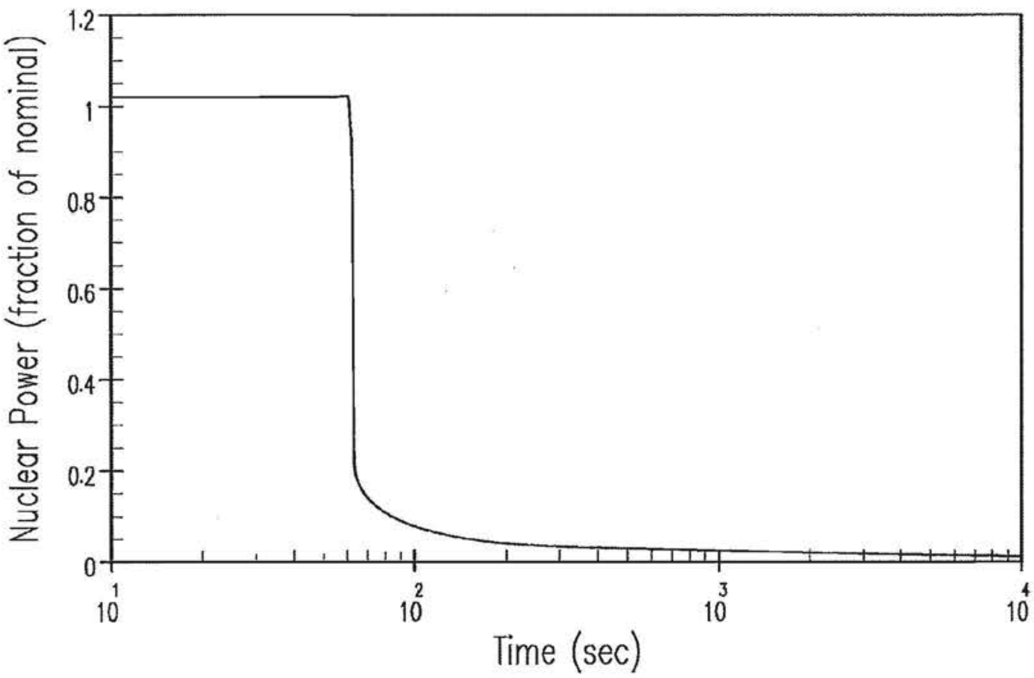
01386642



LOSS OF EXTERNAL ELECTRICAL LOAD WITHOUT AUTOMATIC PRESSURE CONTROL –
STEAM GENERATOR PRESSURE VERSUS TIME

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14467.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-67 REV. 33*	

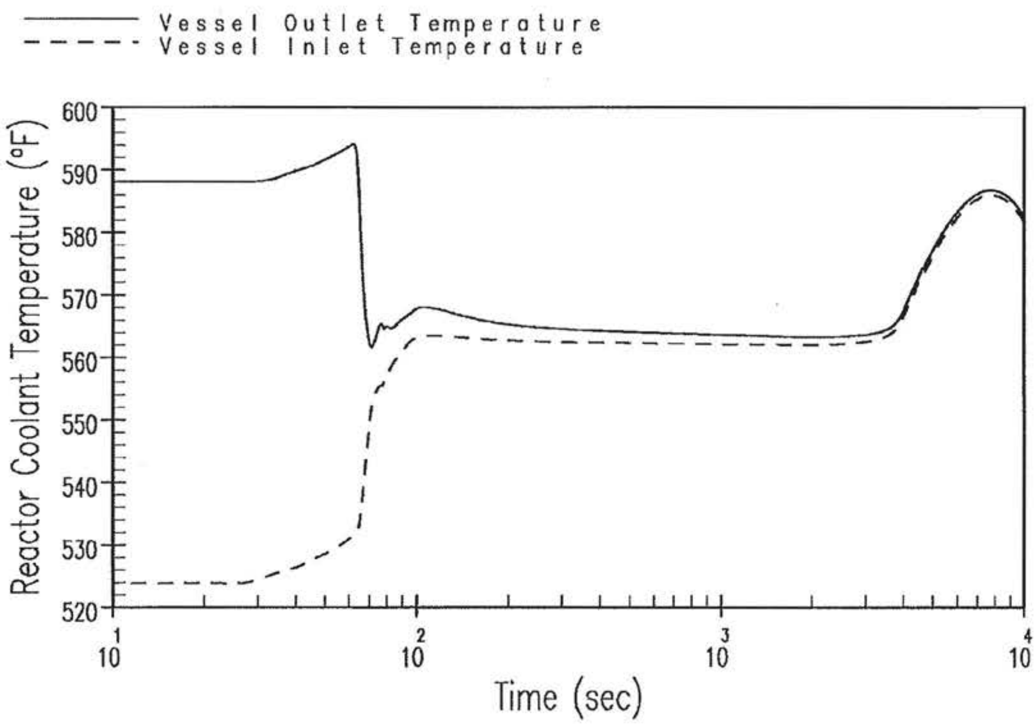
01386642



LOSS OF NORMAL FEEDWATER – NUCLEAR POWER

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14468.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-68 REV. 33*

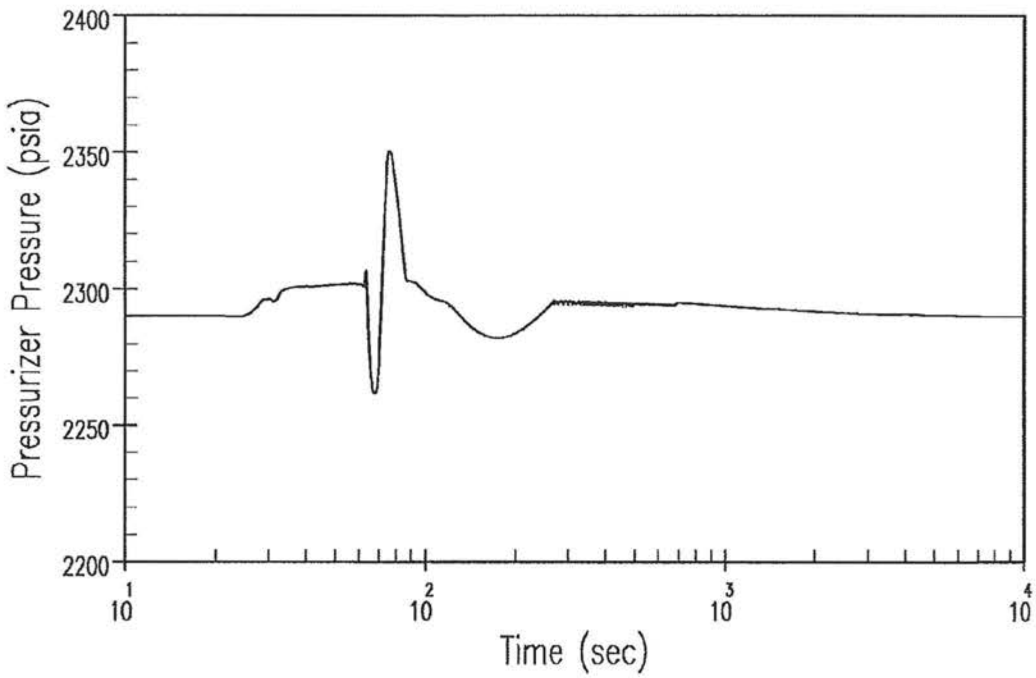
01386642



LOSS OF NORMAL FEEDWATER – REACTOR COOLANT TEMPERATURES

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14469.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-69 REV. 33*	

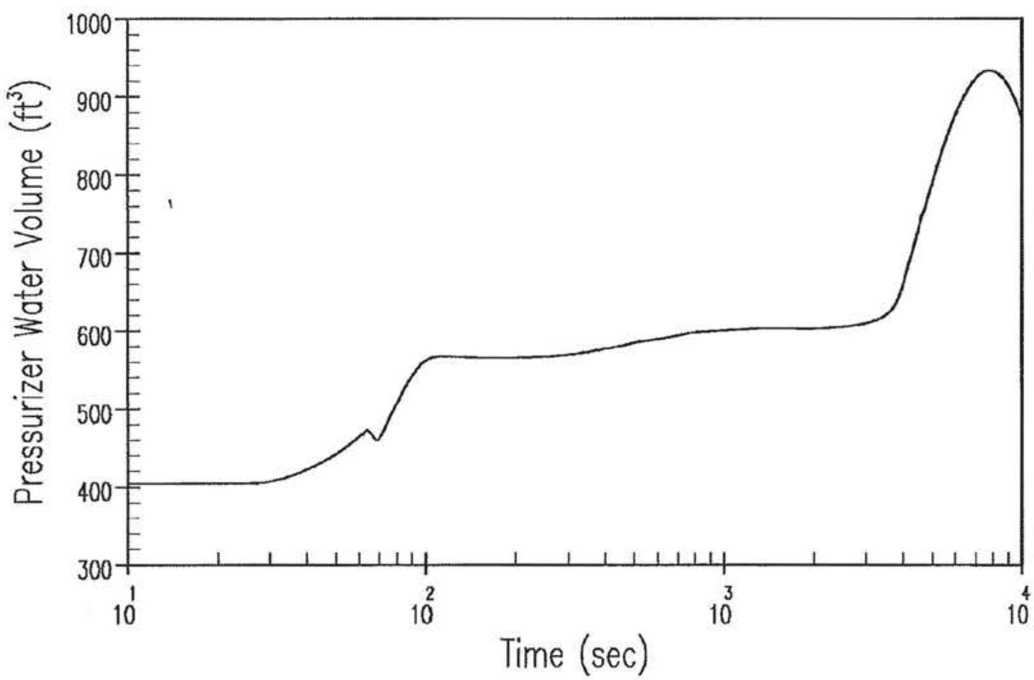
01386642



LOSS OF NORMAL FEEDWATER – PRESSURIZER PRESSURE

DWN	KJF	DATE	2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED		CAD FILE	U14470.DGN		*FIGURE 14.4-70 REV. 33*

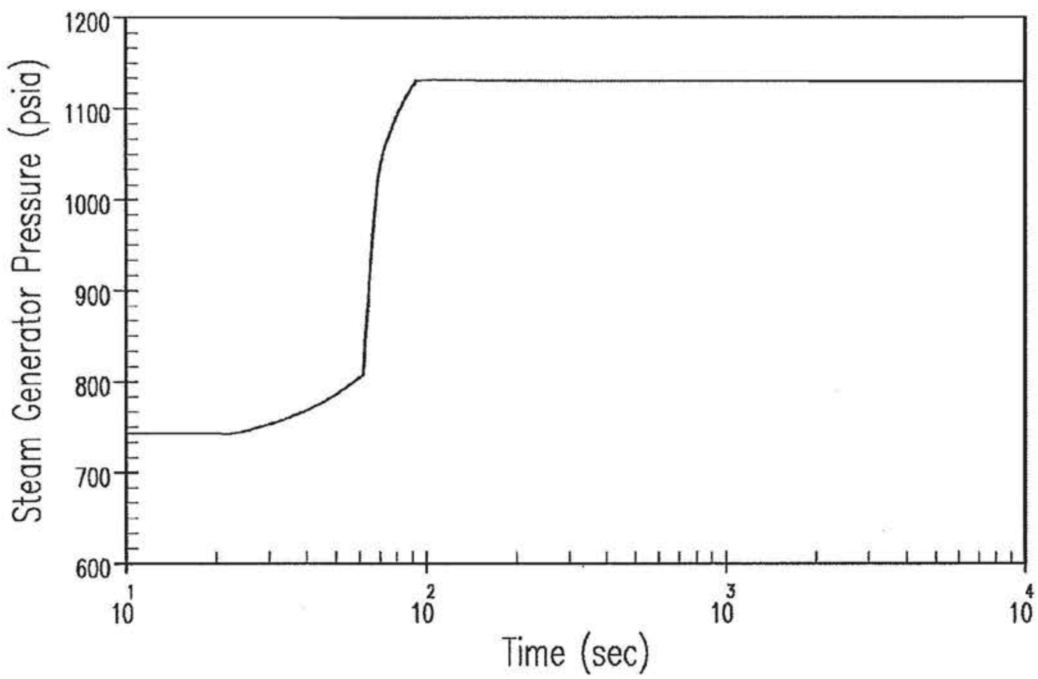
01386642



LOSS OF NORMAL FEEDWATER – PRESSURIZER WATER VOLUME

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14471.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-71 REV. 33*

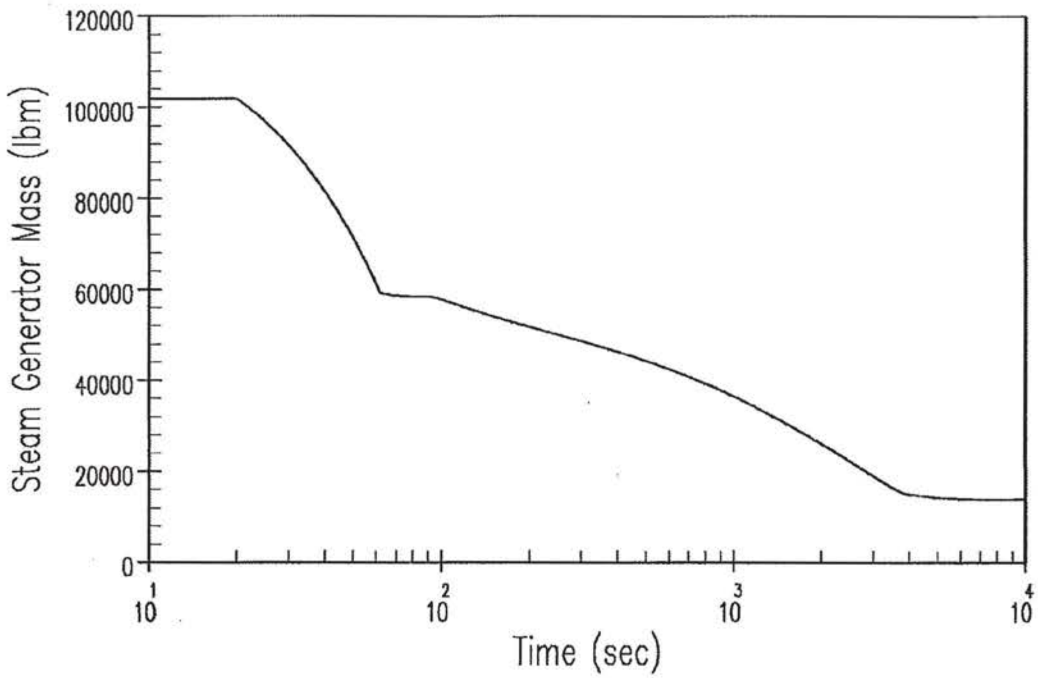
01386642



LOSS OF NORMAL FEEDWATER – STEAM GENERATOR PRESSURE

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14472.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-72 REV. 33*

01386642



LOSS OF NORMAL FEEDWATER – STEAM GENERATOR MASS

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14473.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-73 REV. 33*	

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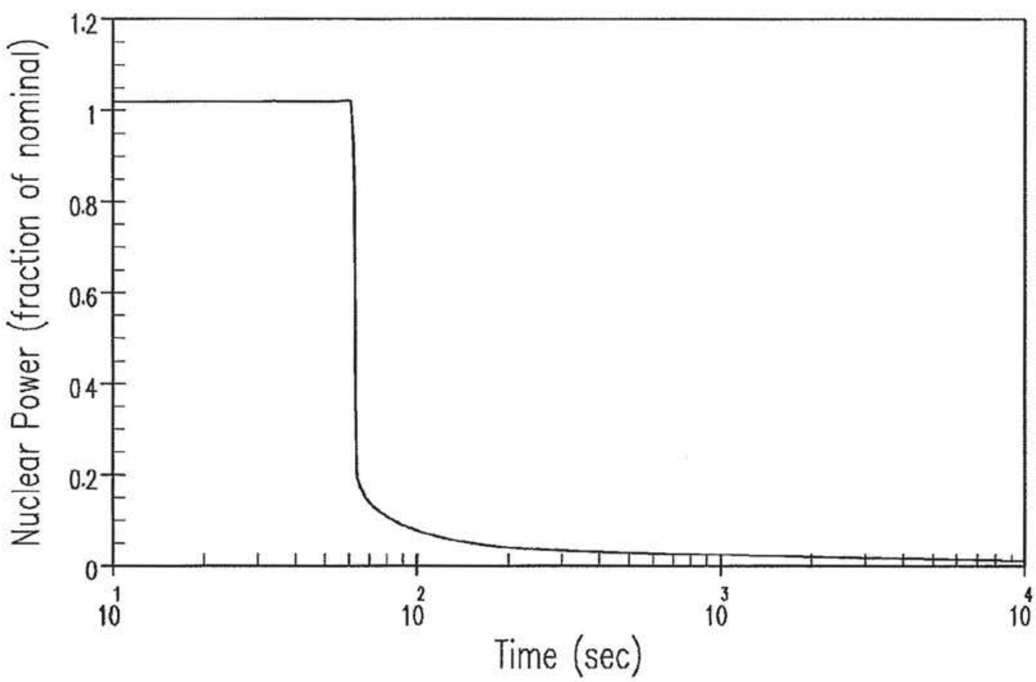
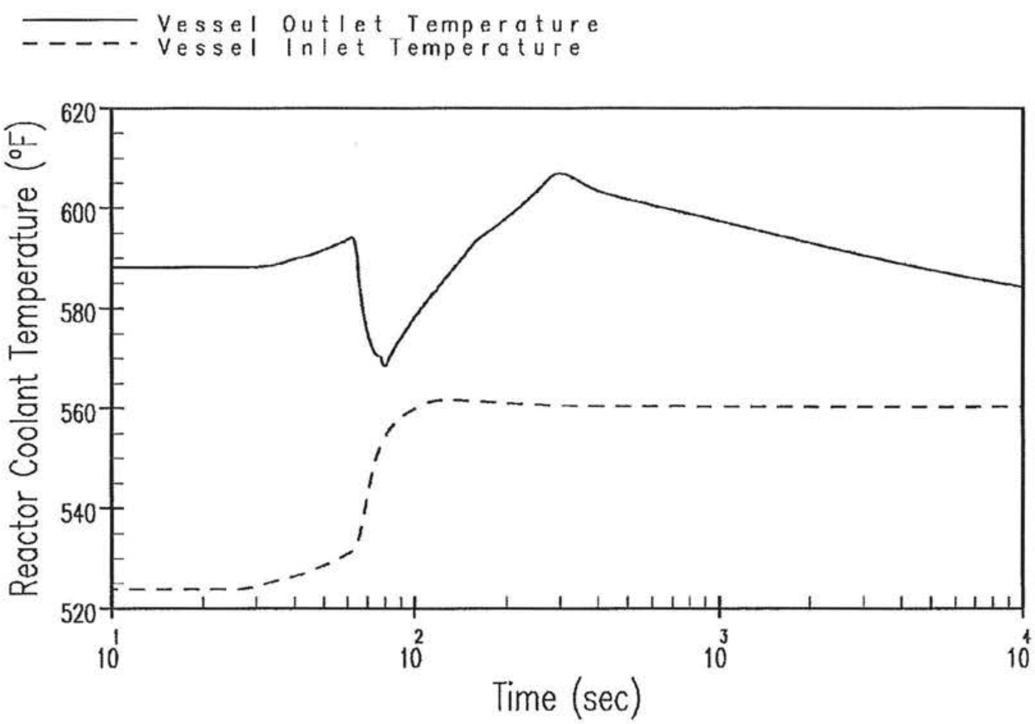


LOSS OF ALL AC POWER TO THE STATION AUXILIARIES – NUCLEAR POWER

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14474.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-74 REV. 33*

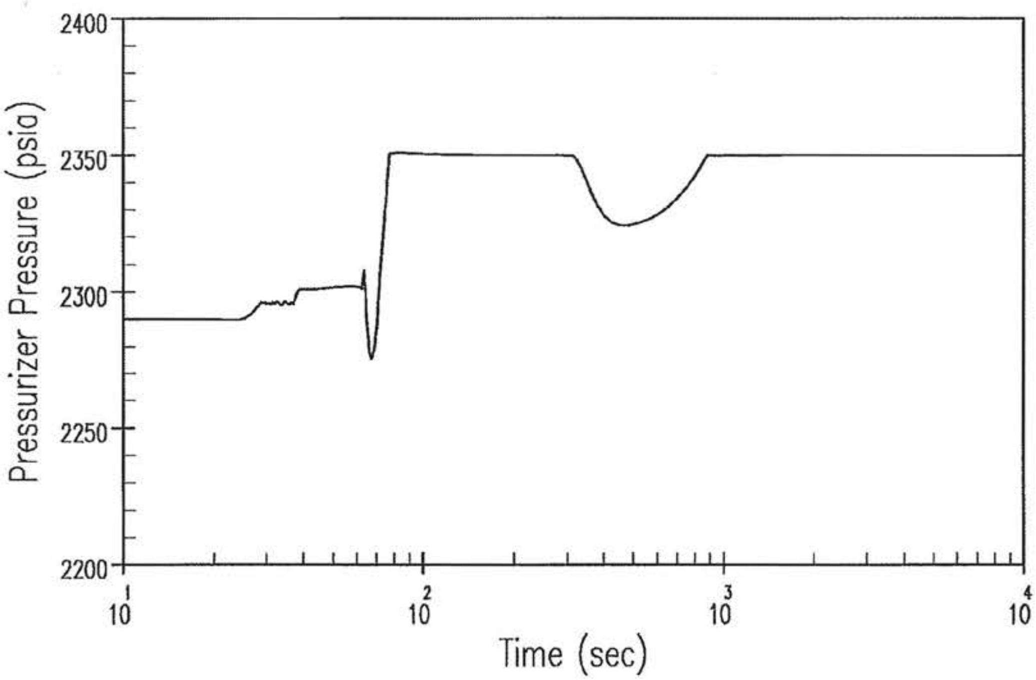
01386642



LOSS OF ALL AC POWER TO THE STATION AUXILIARIES –
REACTOR COOLANT TEMPERATURES

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14475.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-75 REV. 33*

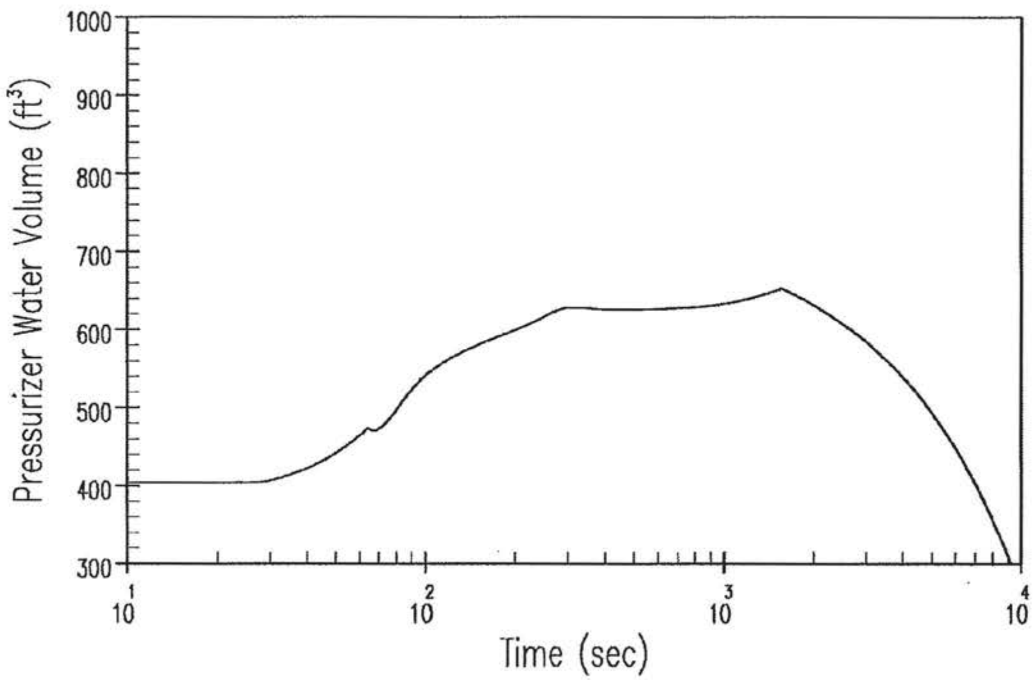
01386642



LOSS OF ALL AC POWER TO THE STATION AUXILIARIES –
PRESSURIZER PRESSURE

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14476.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-76 REV. 33*	

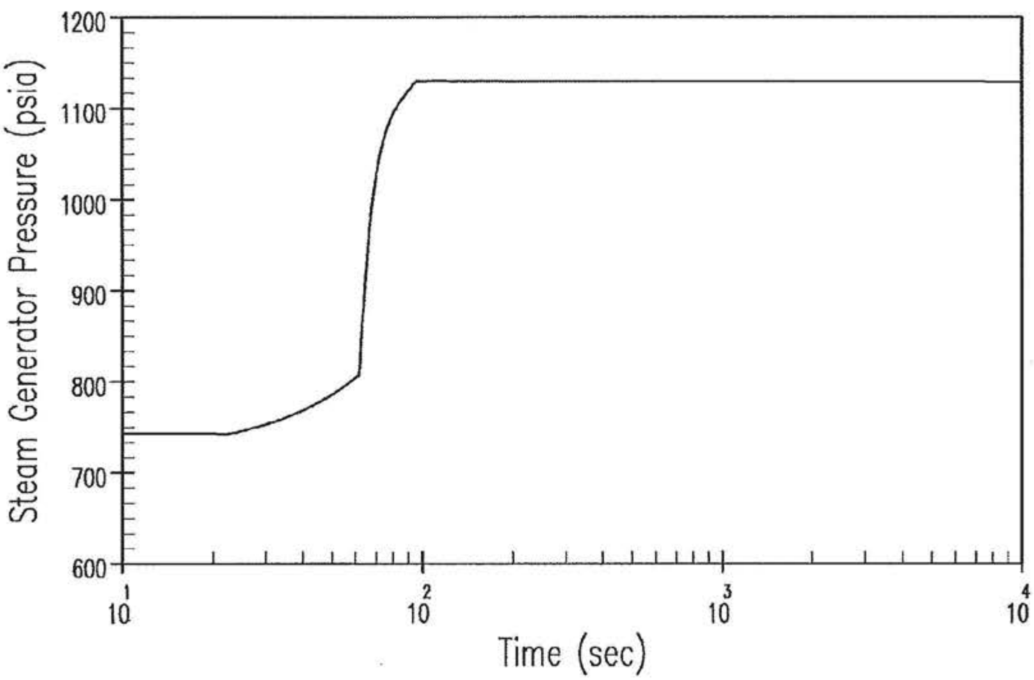
01386642



LOSS OF ALL AC POWER TO THE STATION AUXILIARIES – PRESSURIZER WATER VOLUME

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14477.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-77 REV. 33*	

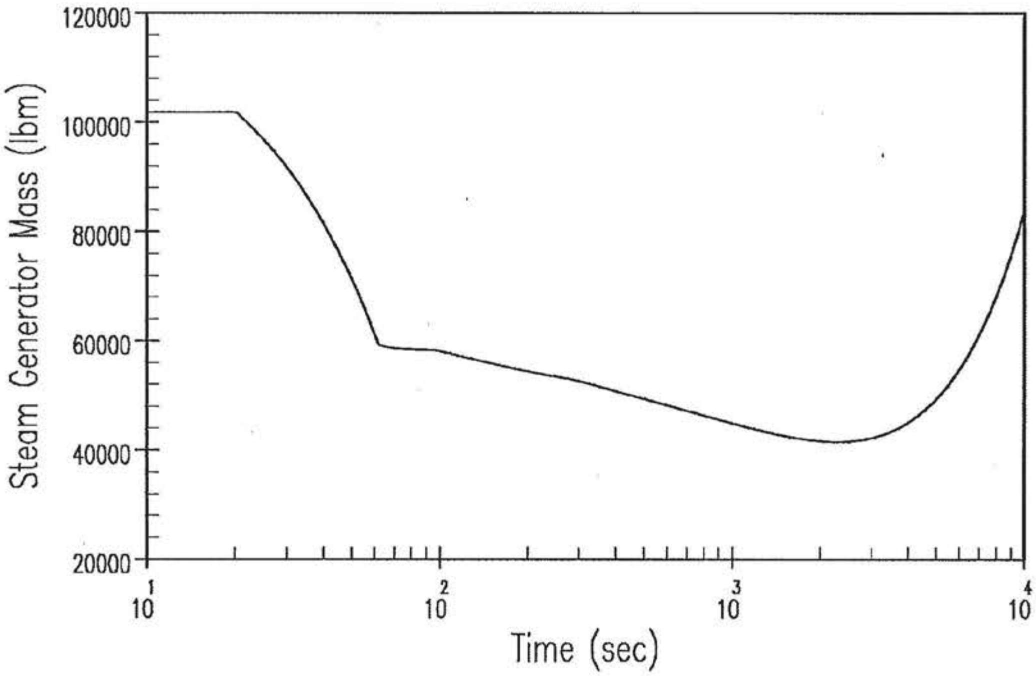
01386642



LOSS OF ALL AC POWER TO THE STATION AUXILIARIES –
STEAM GENERATOR PRESSURE

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14478.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.4-78 REV. 33*

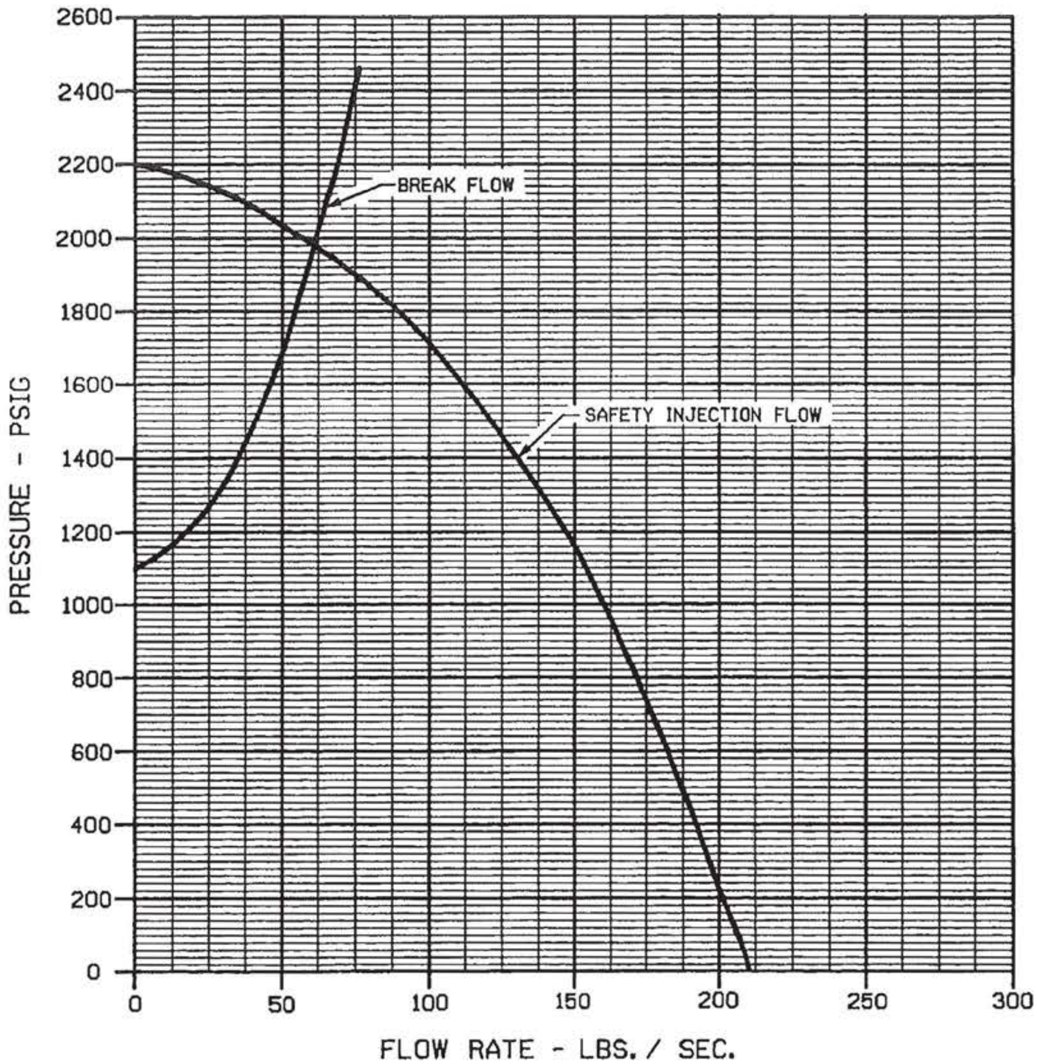
01386642



LOSS OF ALL AC POWER TO THE STATION AUXILIARIES – STEAM GENERATOR MASS

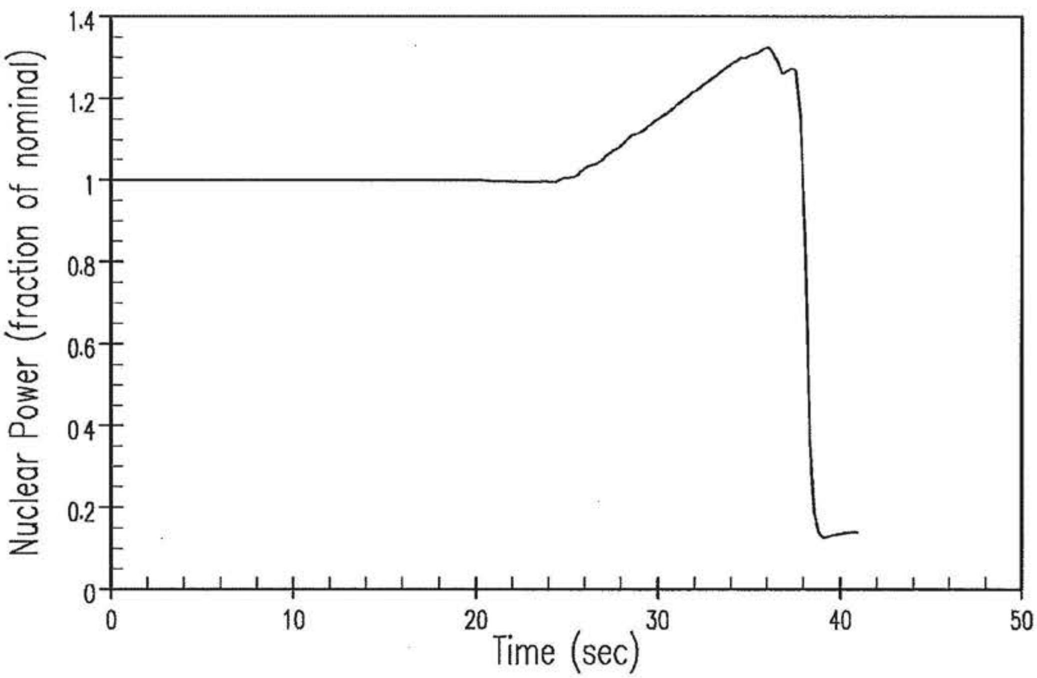
DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14479.DGN		*FIGURE 14.4-79 REV. 33*

01386642



BREAK FLOW AFTER TRIP VS. 2 S. I. PUMP
INJECTION FLOW

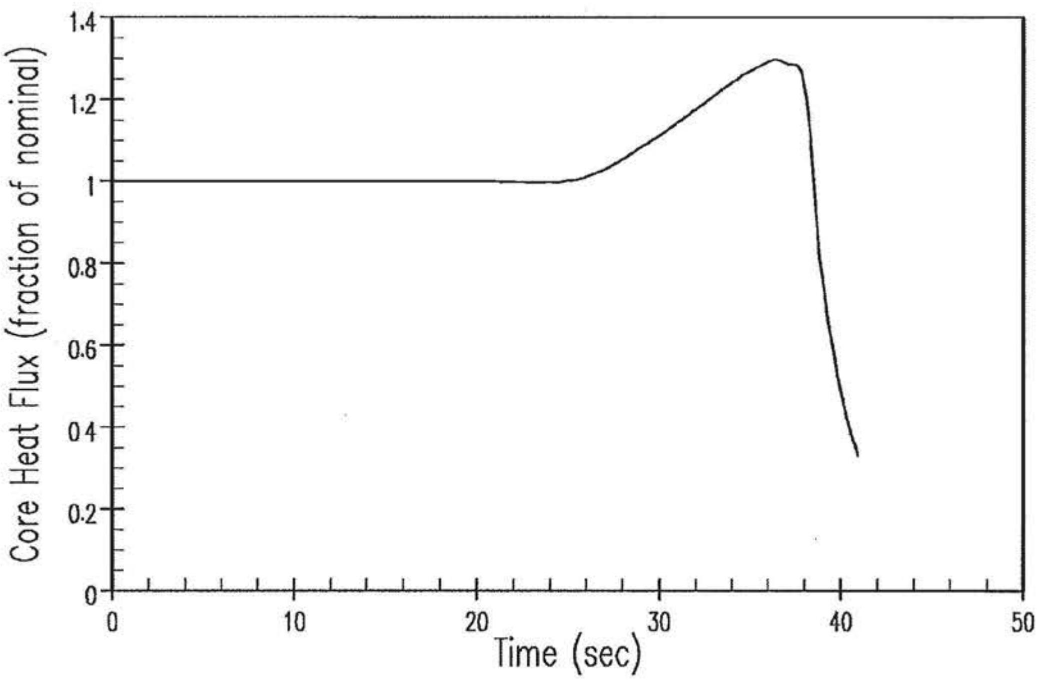
DWN T. MILLER	DATE 6-23-99	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE	
CHECKED	CAD FILE U14501.DGN		FIGURE 14.5-1	REV. 18



01386642

STEAMLINE RUPTURE – FULL POWER CORE RESPONSE –
NUCLEAR POWER VERSUS TIME

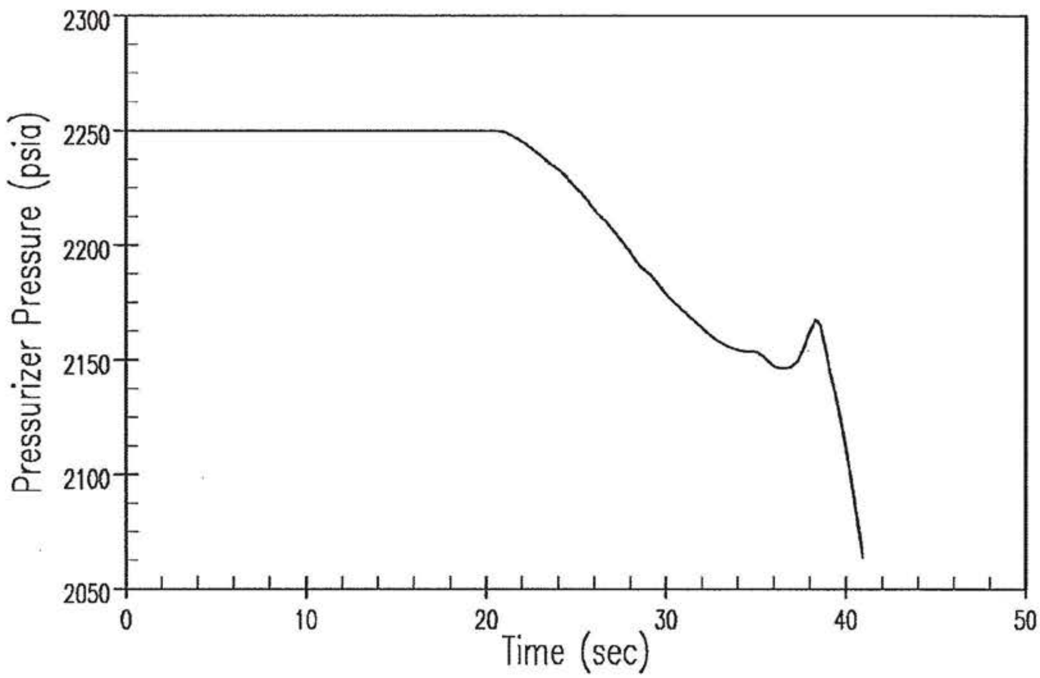
DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14502.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-2 REV. 33*



STEAMLINE RUPTURE – FULL POWER CORE RESPONSE –
CORE HEAT FLUX VERSUS TIME

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14503.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-3 REV. 33*

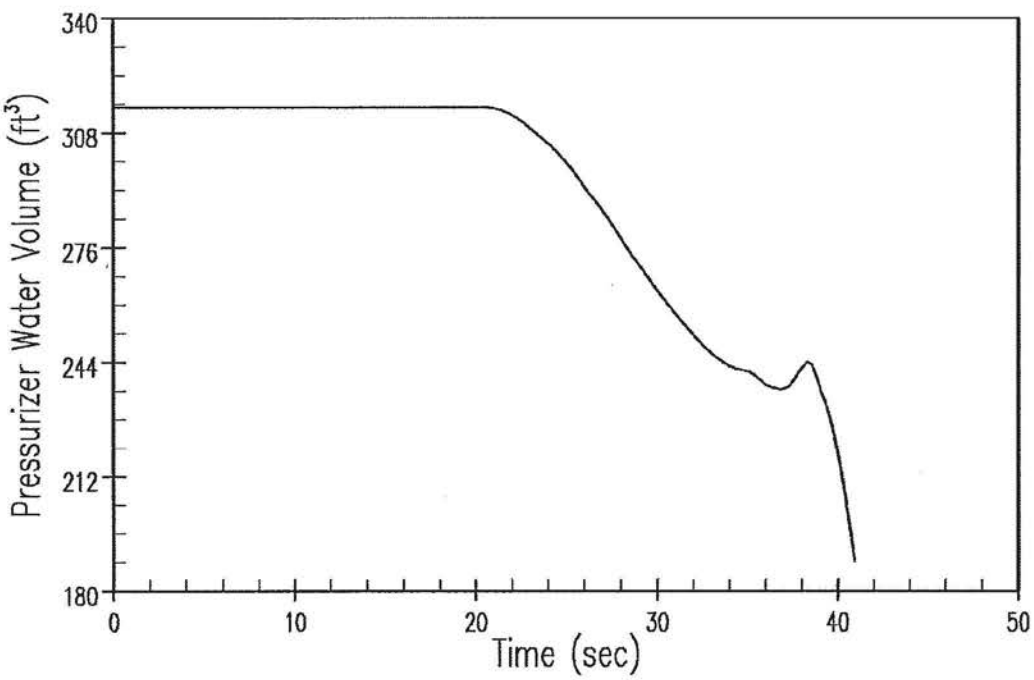
01386642



STEAMLINE RUPTURE – FULL POWER CORE RESPONSE –
PRESSURIZER PRESSURE VERSUS TIME

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14504.DGN		*FIGURE 14.5-4 REV. 33*

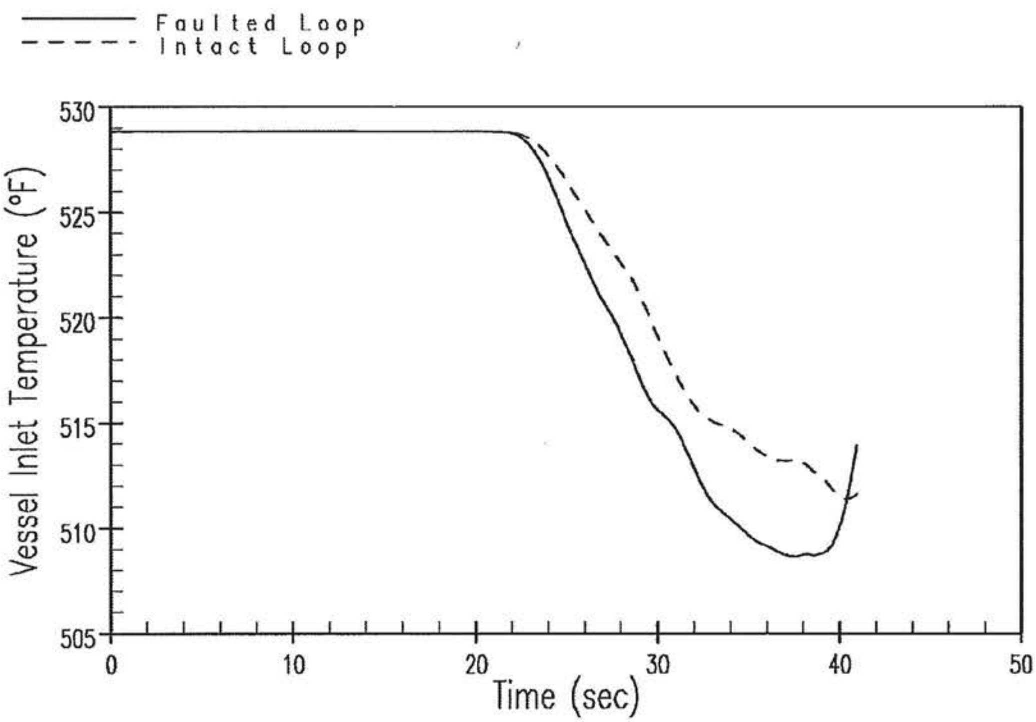
01386642



STEAMLINE RUPTURE – FULL POWER CORE RESPONSE –
PRESSURIZER WATER VOLUME VERSUS TIME

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14505.DGN		*FIGURE 14.5-5 REV. 33*

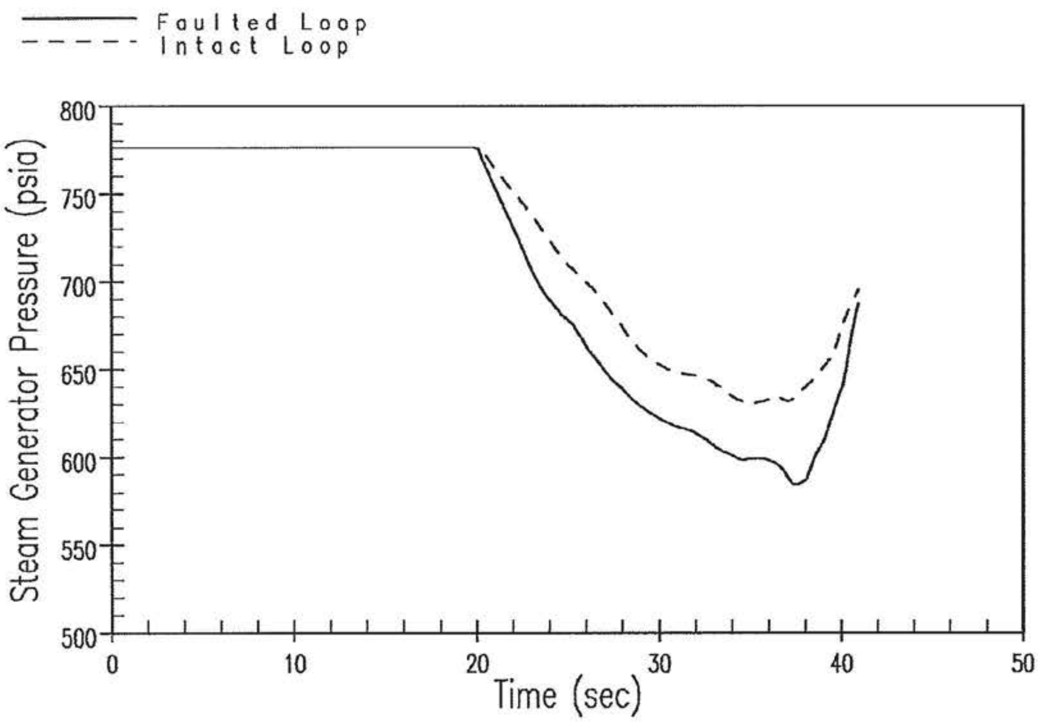
01386642



STEAMLINE RUPTURE – FULL POWER CORE RESPONSE –
VESSEL INLET TEMPERATURE VERSUS TIME

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14506.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-6 REV. 33*

01386642



STEAMLINE RUPTURE – FULL POWER CORE RESPONSE –
STEAM GENERATOR PRESSURE VERSUS TIME

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14507.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-7 REV. 33*	

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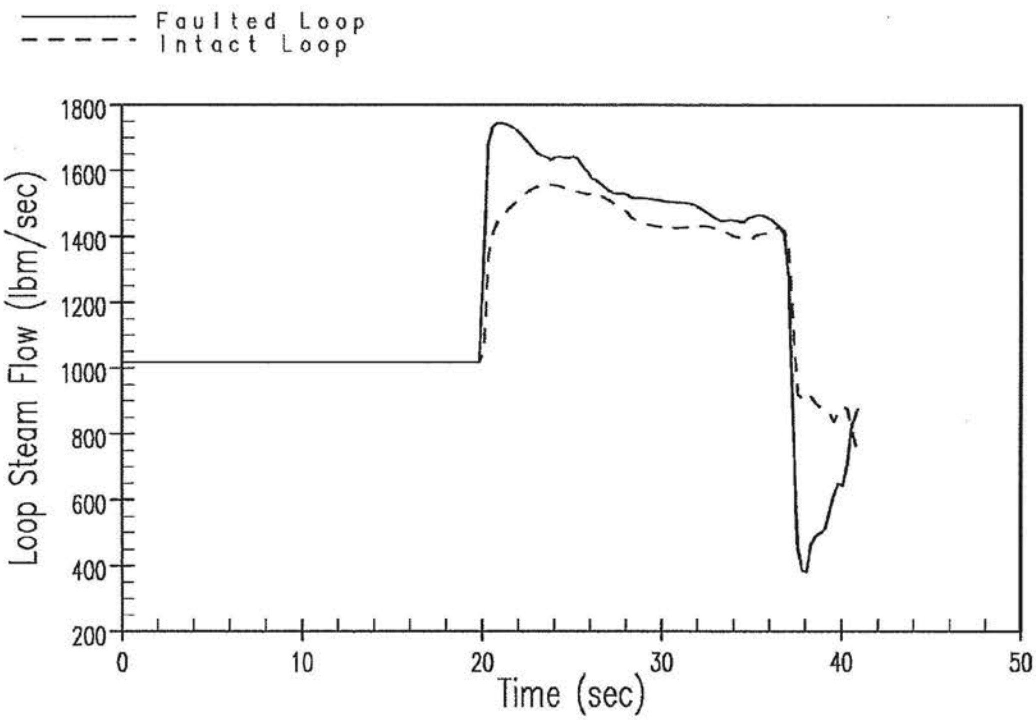
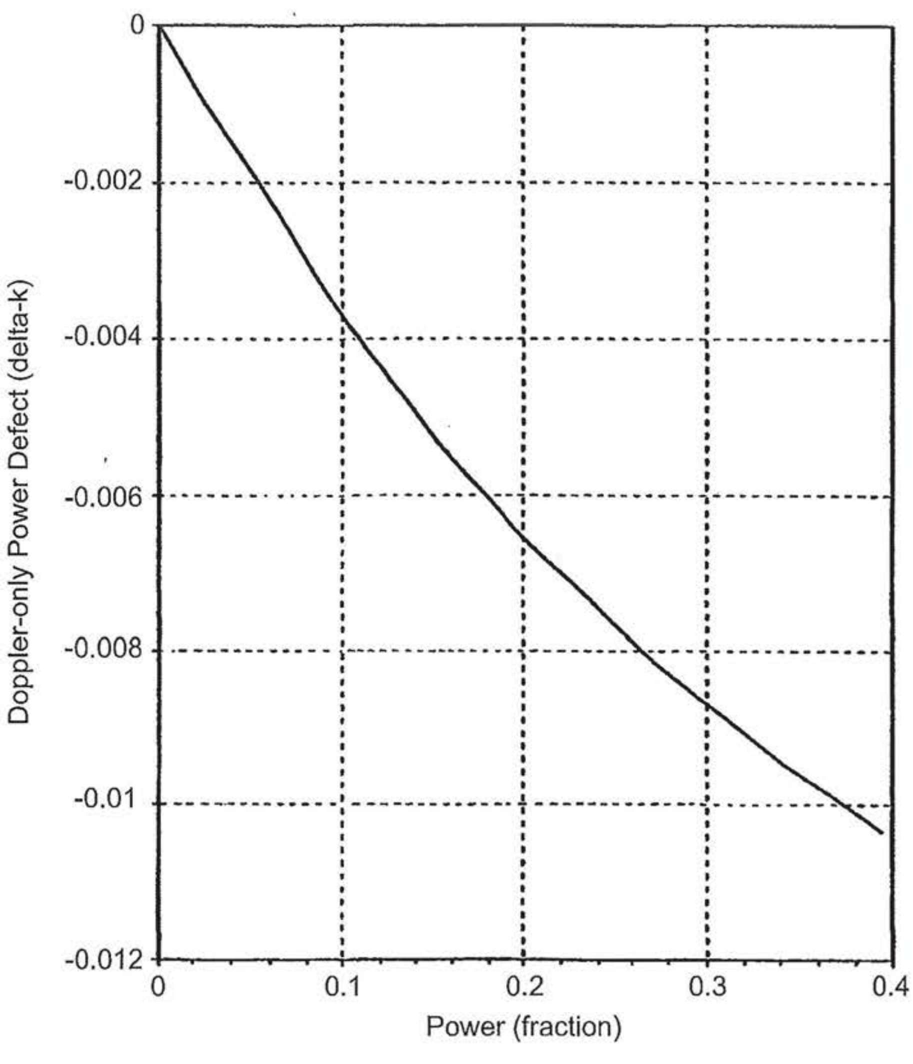


01386642

STEAMLINE RUPTURE – FULL POWER CORE RESPONSE –
LOOP STEAM FLOW VERSUS TIME

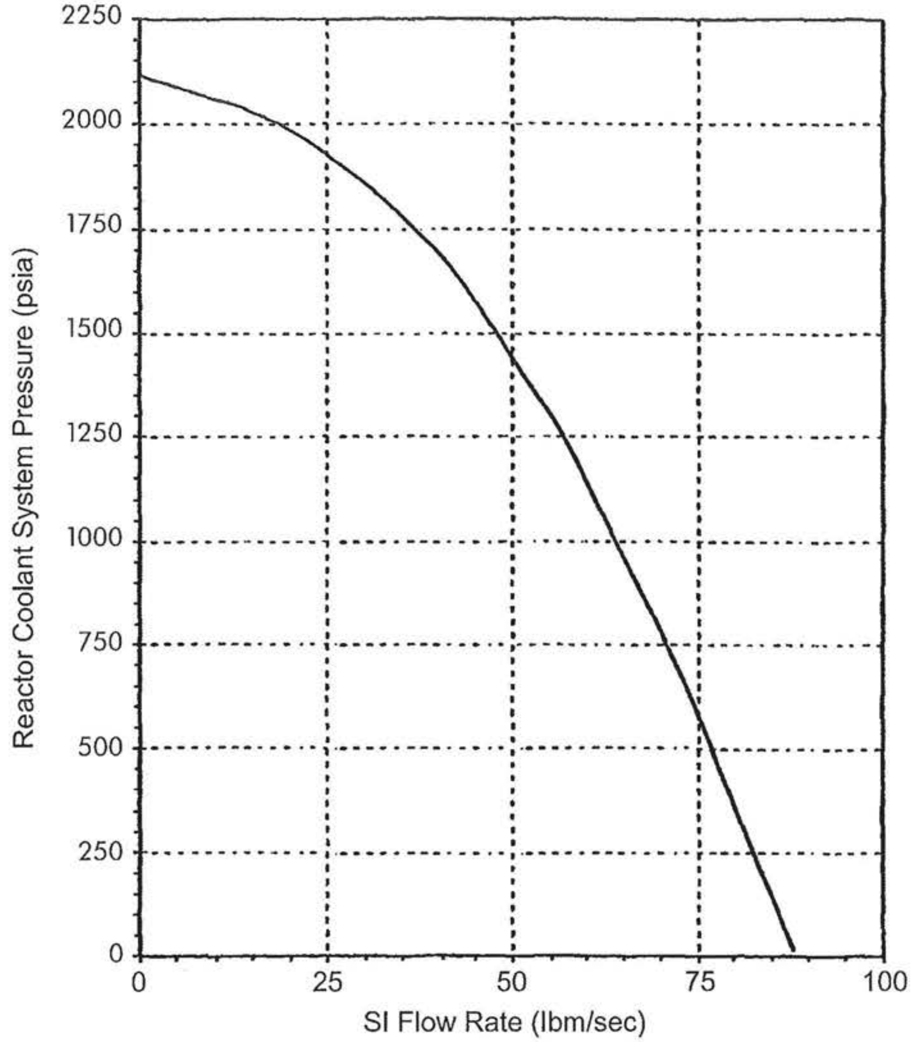
DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14508.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-8 REV. 33*	



STEAMLINE RUPTURE – ZERO POWER CORE RESPONSE –
DOPPLER–ONLY POWER DEFECT WITH STUCK RCCA

DWN KJF	DATE 2–20–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14509.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-9 REV. 33*

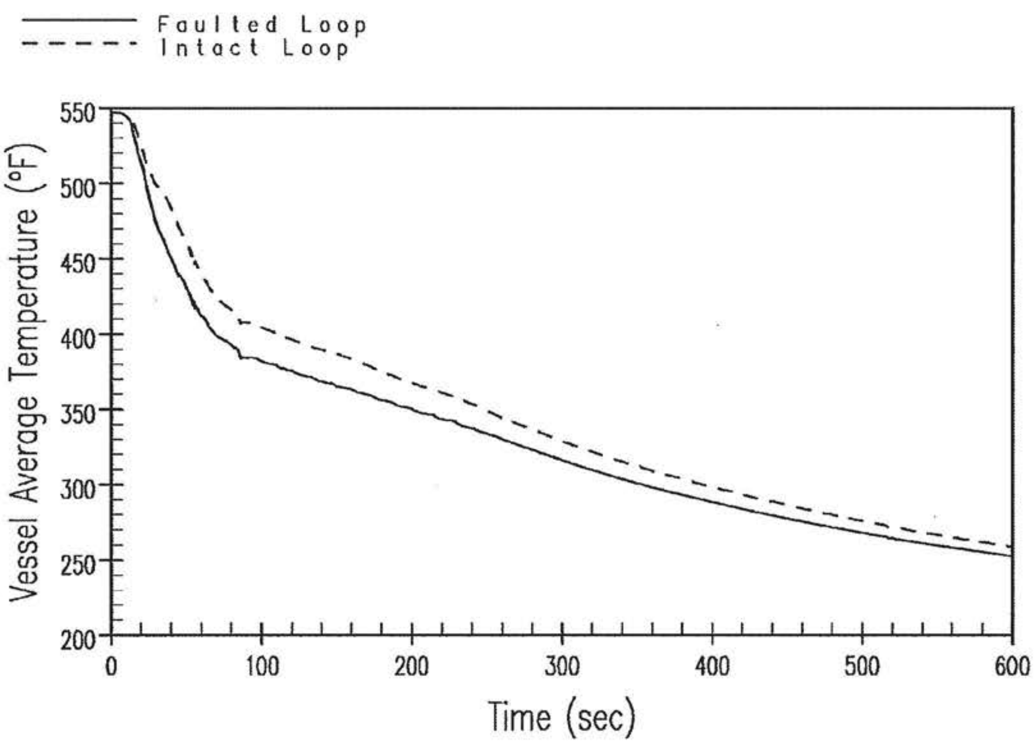
01386642



01386642

STEAMLINE RUPTURE – ZERO POWER CORE RESPONSE –
SAFETY INJECTION FLOW VERSUS RCS PRESSURE

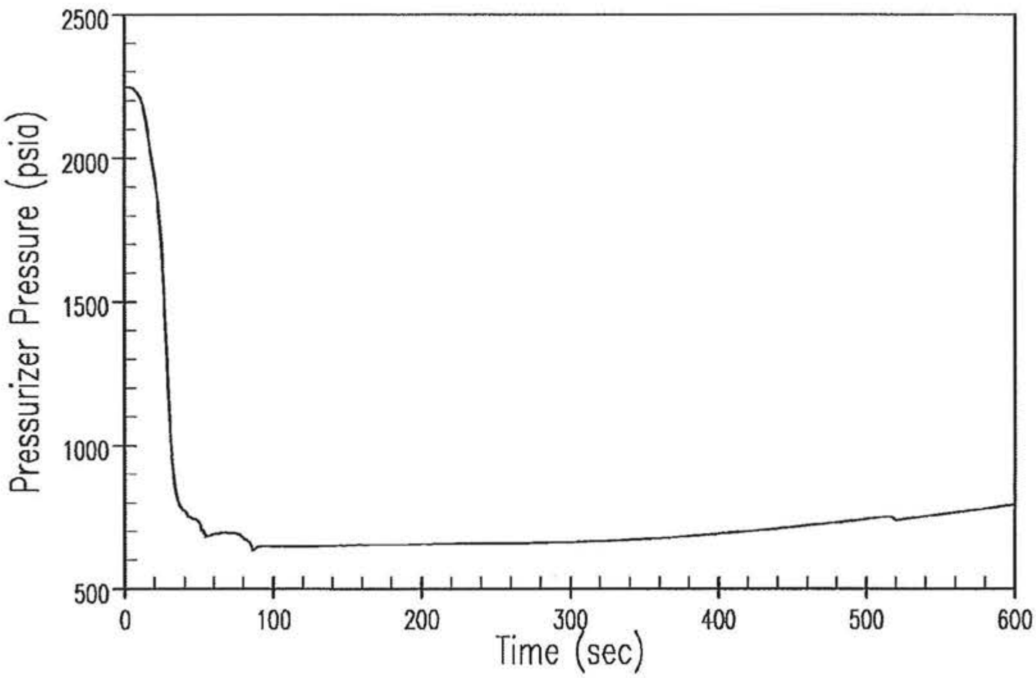
DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14510.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-10 REV. 33*



STEAMLINE RUPTURE – ZERO POWER CORE RESPONSE –
VESSEL AVERAGE TEMPERATURE VERSUS TIME

DWN KJF	DATE 2–20–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14511.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-11 REV. 33*	

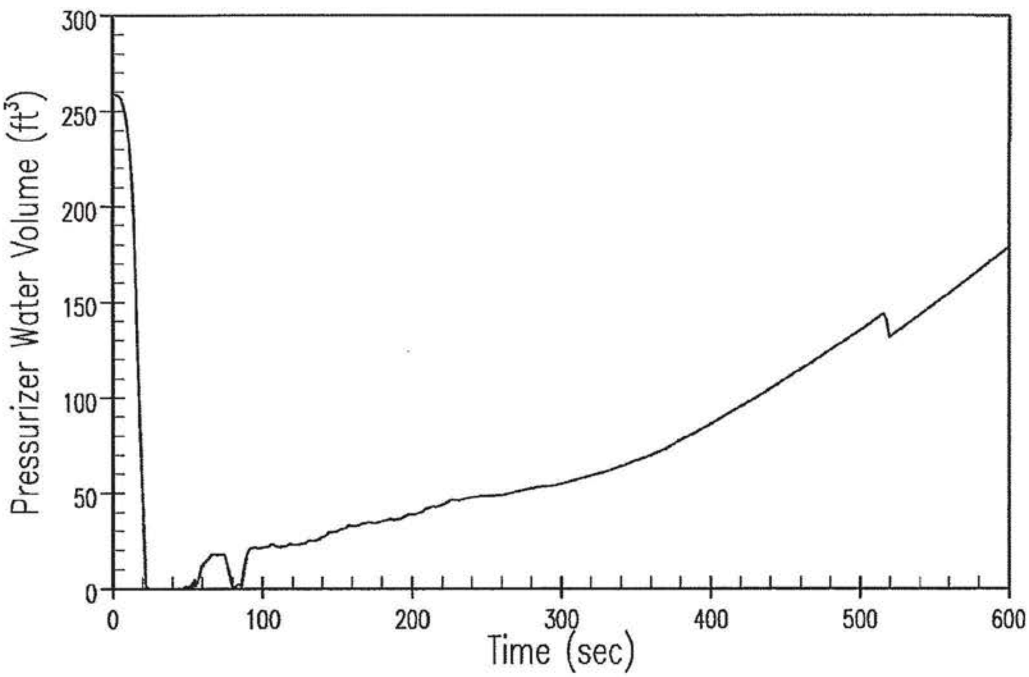
01386642



STEAMLINE RUPTURE – ZERO POWER CORE RESPONSE –
PRESSURIZER PRESSURE VERSUS TIME

DWN KJF	DATE 2–20–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14512.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-12 REV. 33*

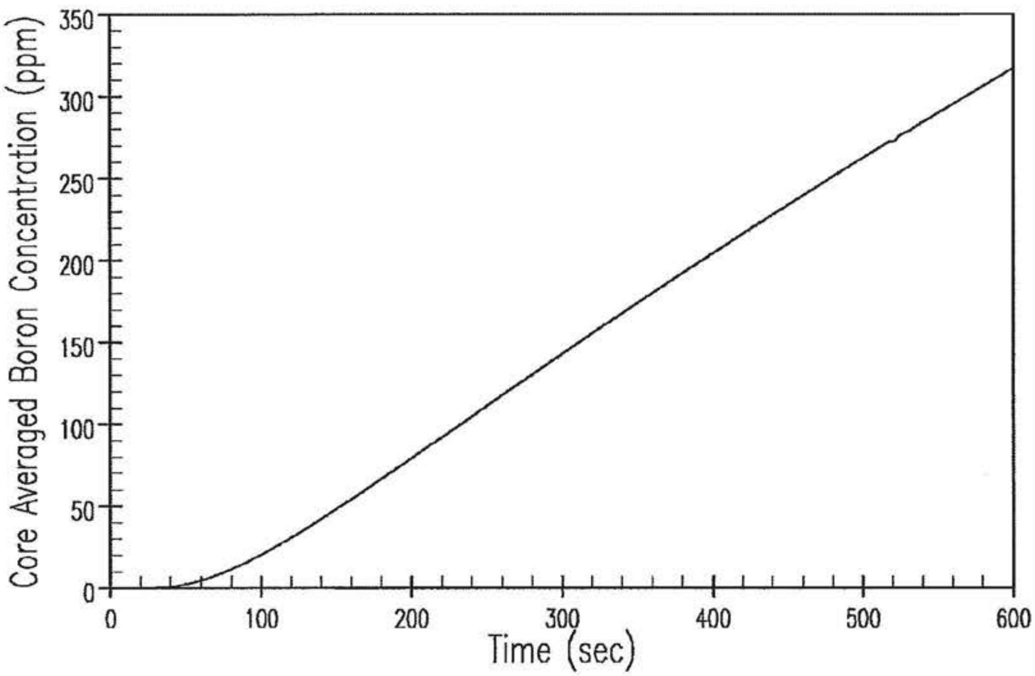
01386642



STEAMLINE RUPTURE – ZERO POWER CORE RESPONSE –
PRESSURIZER WATER VOLUME VERSUS TIME

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14513.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-13 REV. 33*

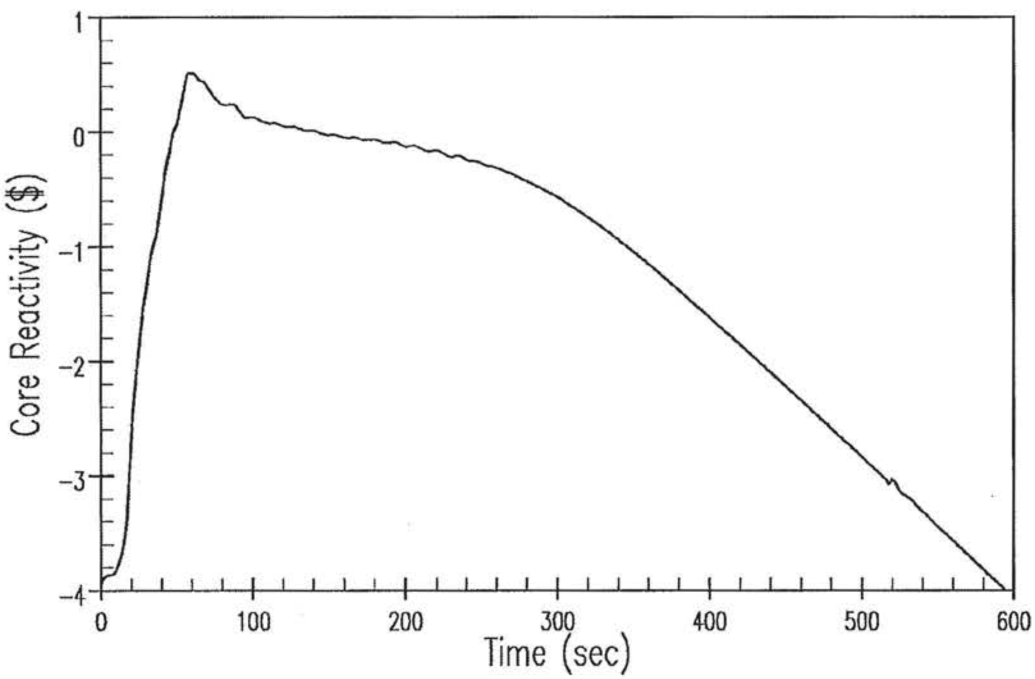
01386642



STEAMLINE RUPTURE – ZERO POWER CORE RESPONSE –
CORE AVERAGED BORON CONCENTRATION VERSUS TIME

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14514.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-14 REV. 33*

01386642



STEAMLINE RUPTURE – ZERO POWER CORE RESPONSE –
CORE REACTIVITY VERSUS TIME

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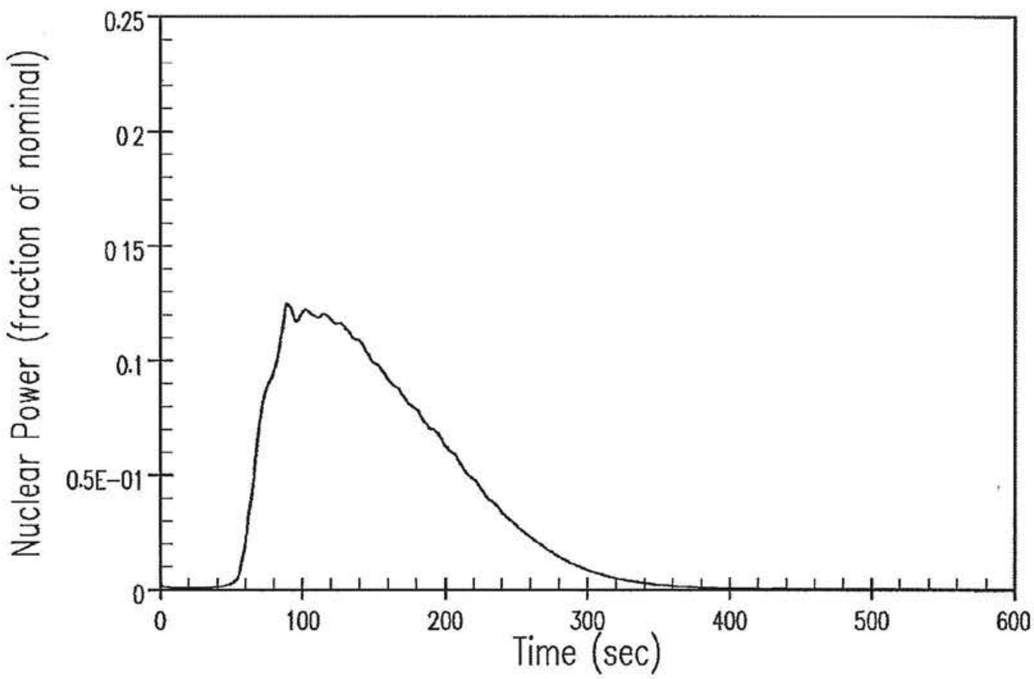
CHECKED	CAD FILE U14515.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-15 REV. 33*

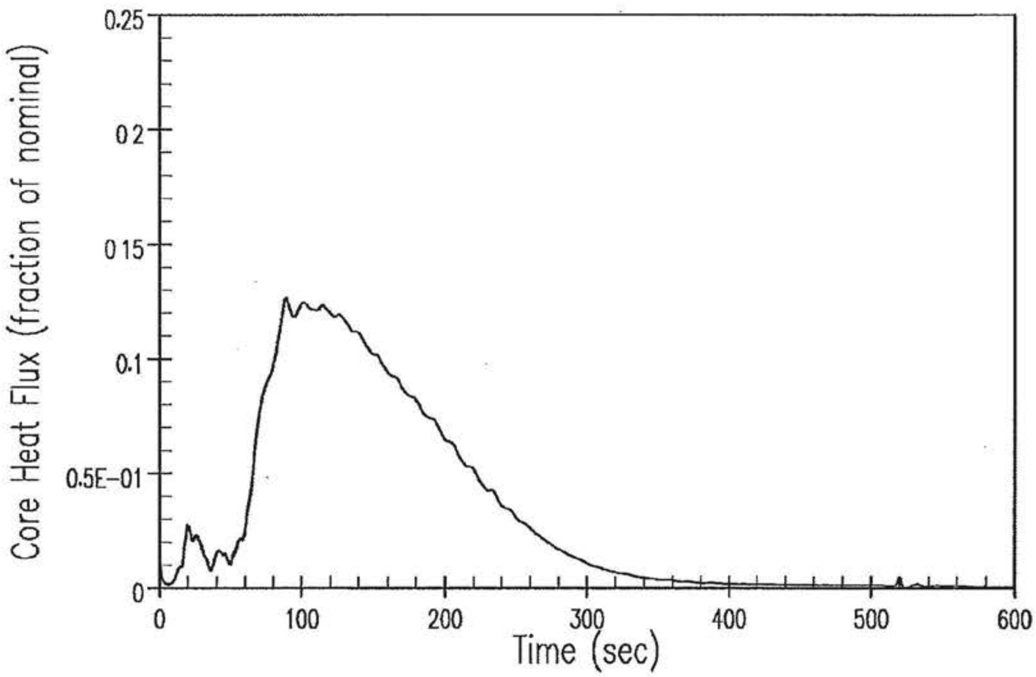
01386642



STEAMLINE RUPTURE – ZERO POWER CORE RESPONSE – NUCLEAR POWER VERSUS TIME

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14516.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-16 REV. 33*

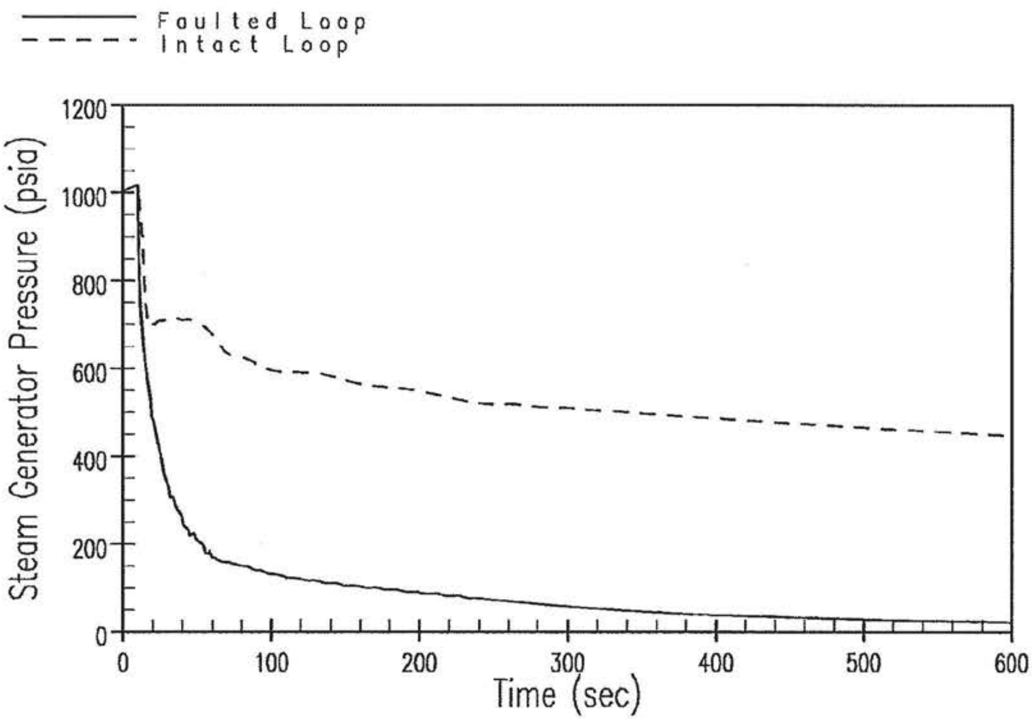
01386642



STEAMLINE RUPTURE – ZERO POWER CORE RESPONSE –
CORE HEAT FLUX VERSUS TIME

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14517.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-17 REV. 33*

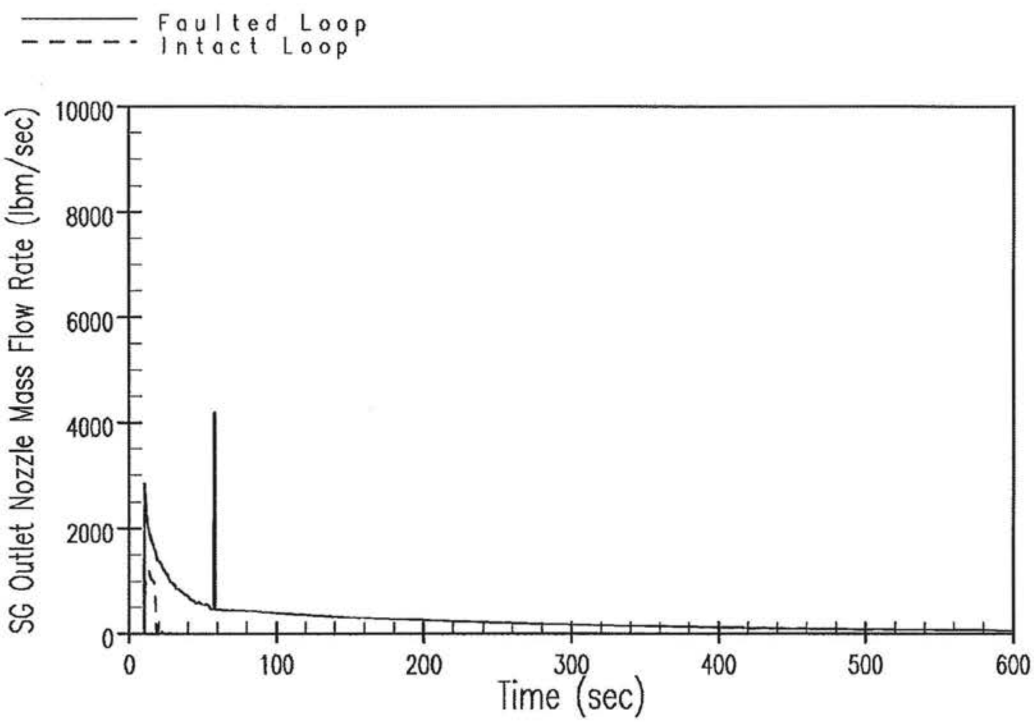
01386642



STEAMLINE RUPTURE – ZERO POWER CORE RESPONSE –
STEAM GENERATOR PRESSURE VERSUS TIME

DWN KJF	DATE 2–20–14	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA
CHECKED	CAD FILE U14518.DGN	SCALE: NONE
		FIGURE 14.5-18 REV. 33

01386642



STEAMLINE RUPTURE – ZERO POWER CORE RESPONSE –
STEAM GENERATOR OUTLET NOZZLE MASS FLOW RATE VERSUS TIME

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
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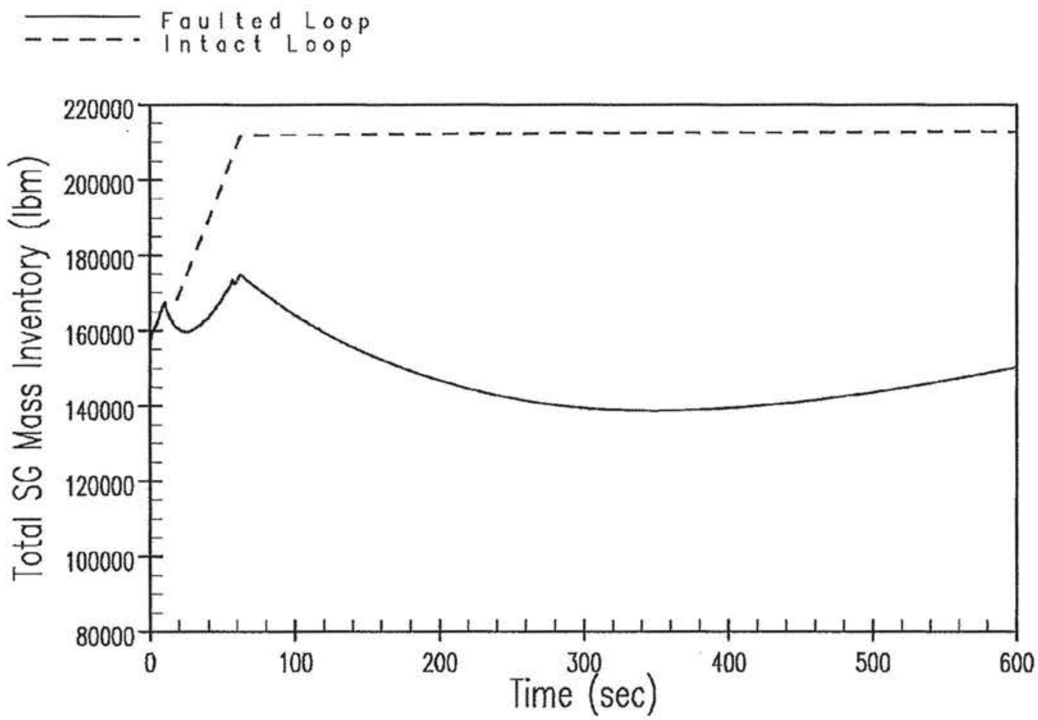
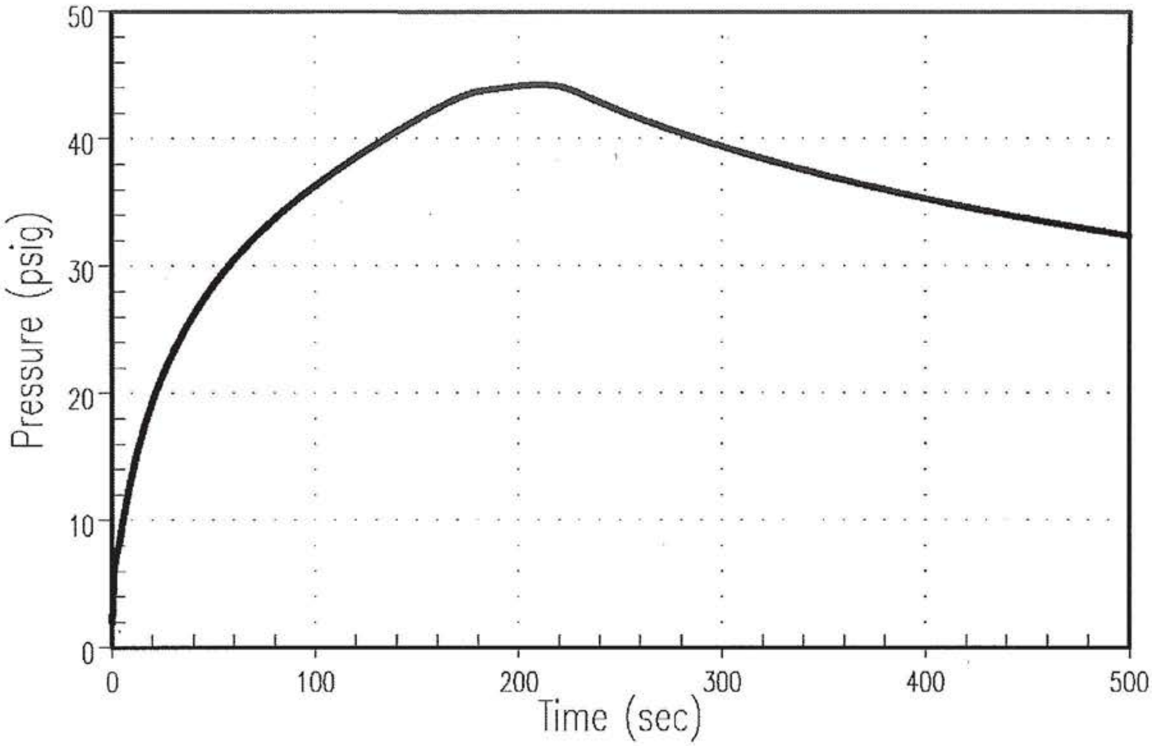
01386642



01386642

STEAMLINE RUPTURE – ZERO POWER CORE RESPONSE –
TOTAL STEAM GENERATOR MASS INVENTORY VERSUS TIME

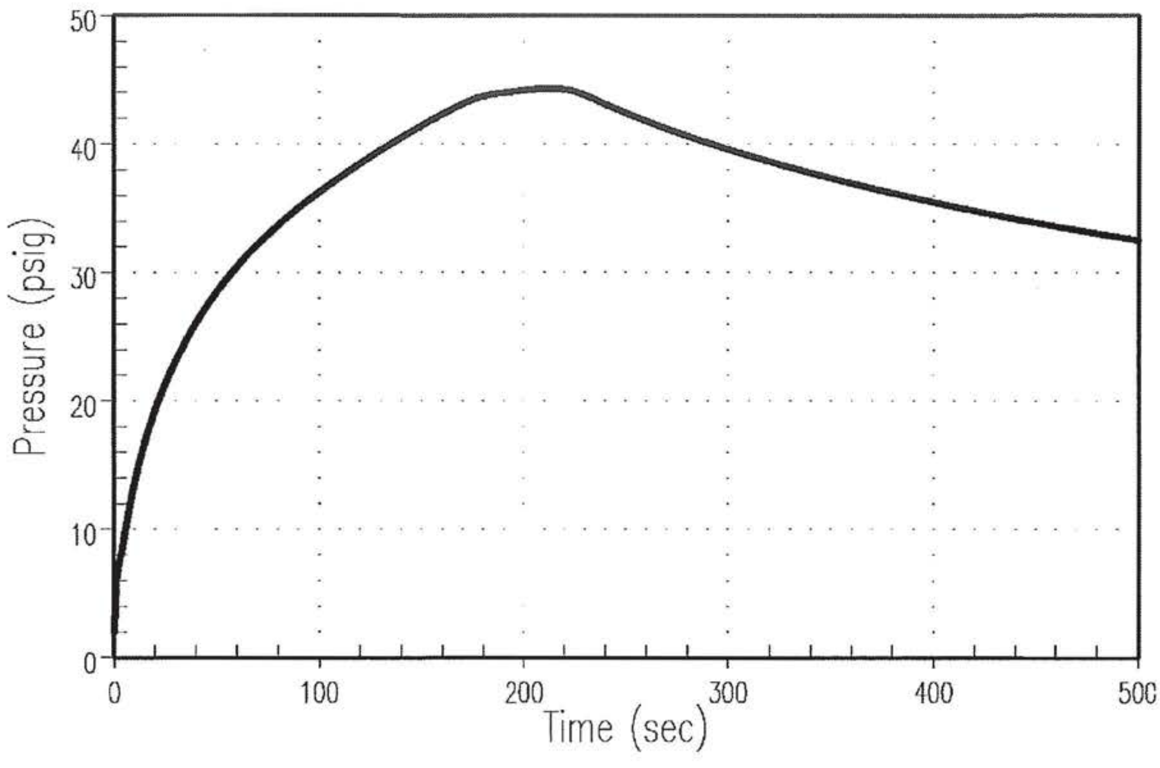
DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14520.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-20 REV. 33*



UNIT 1 PEAK CONTAINMENT PRESSURE FROM A STEAMLINE BREAK

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14523A.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-23A REV. 33*

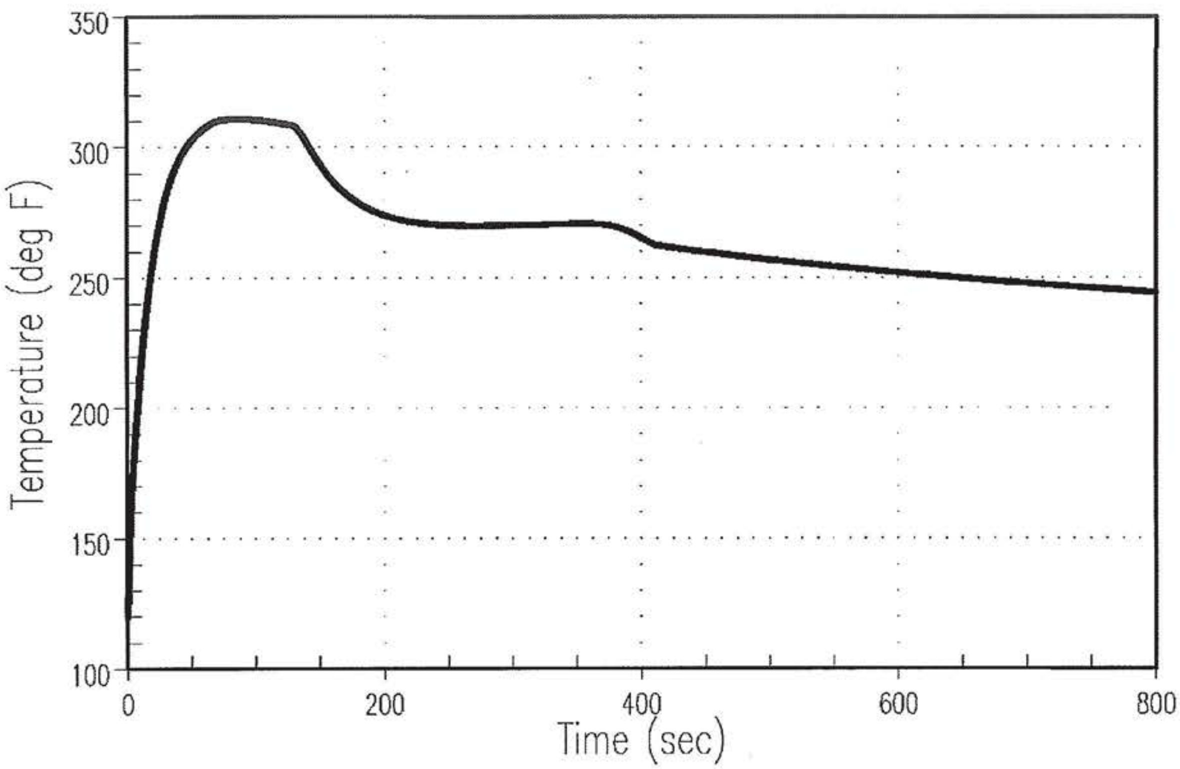
01386642



UNIT 2 PEAK CONTAINMENT PRESSURE FROM A STEAMLINE BREAK

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14523B.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-23B REV. 33*

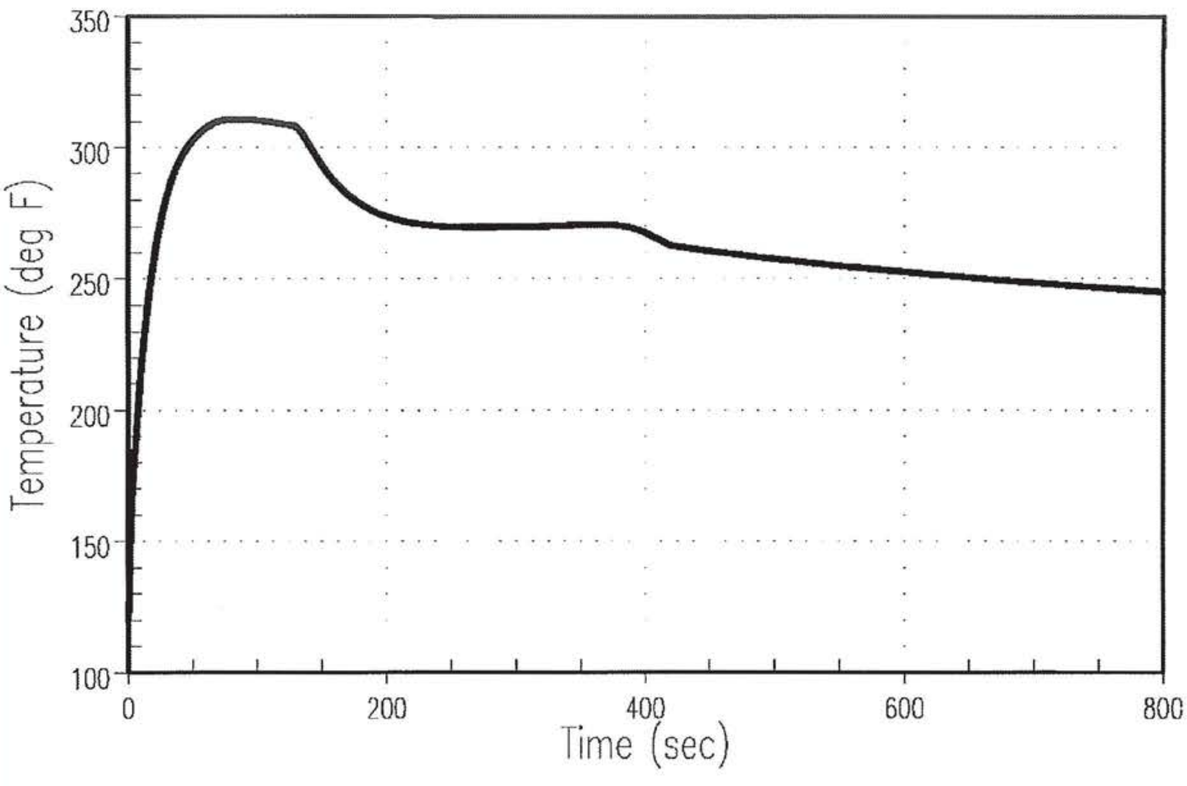
01386642



UNIT 1 PEAK CONTAINMENT TEMPERATURE FROM A STEAMLINE BREAK

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14524A.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-24A REV. 33*

01386642



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UNIT 2 PEAK CONTAINMENT TEMPERATURE FROM A STEAMLINE BREAK

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
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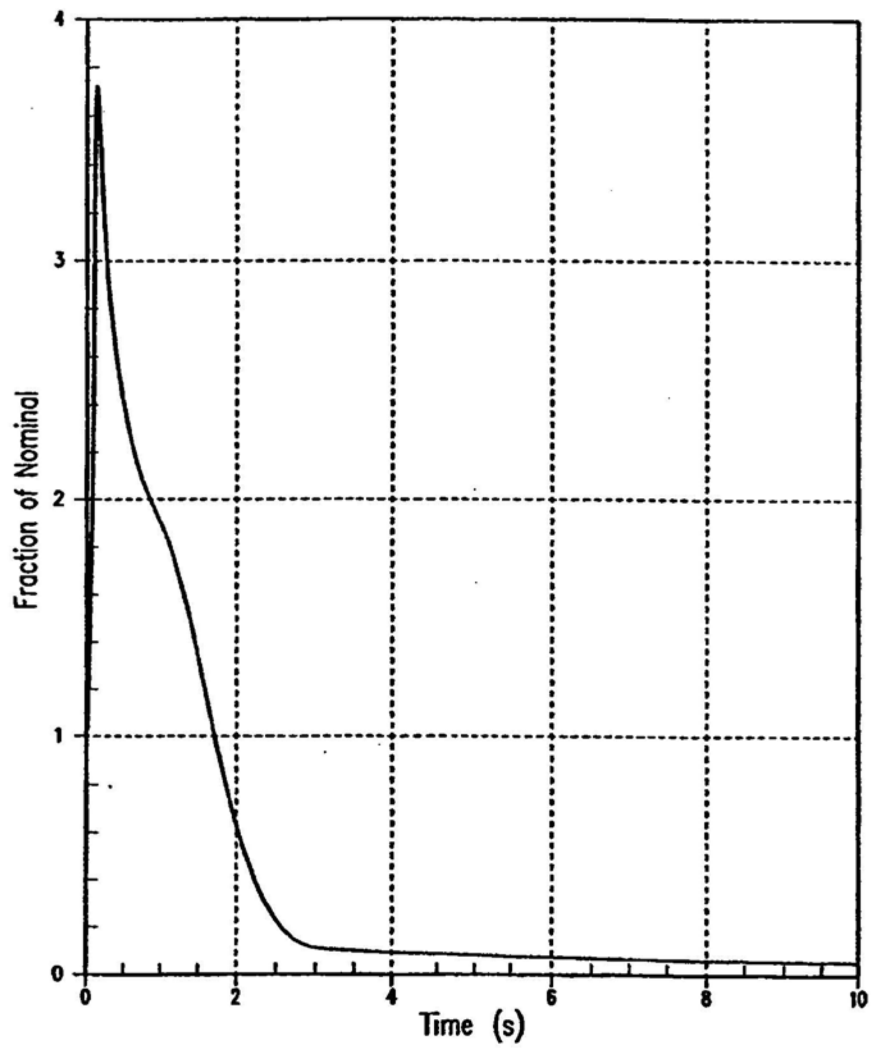


Figure 14.5-25
RCCA Ejection – BOC Full Power
Reactor Power vs. Time

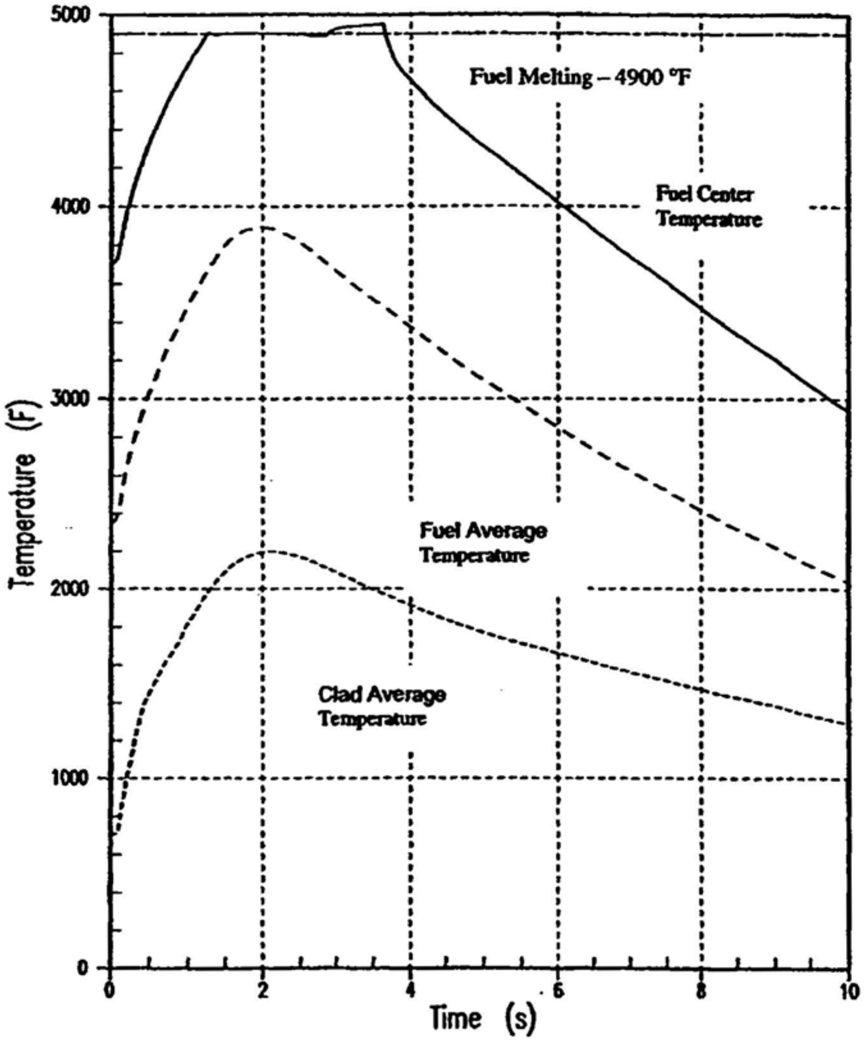


Figure 14.5-26
RCCA Ejection – BOC Full Power
Fuel and Clad Temperatures vs. Time

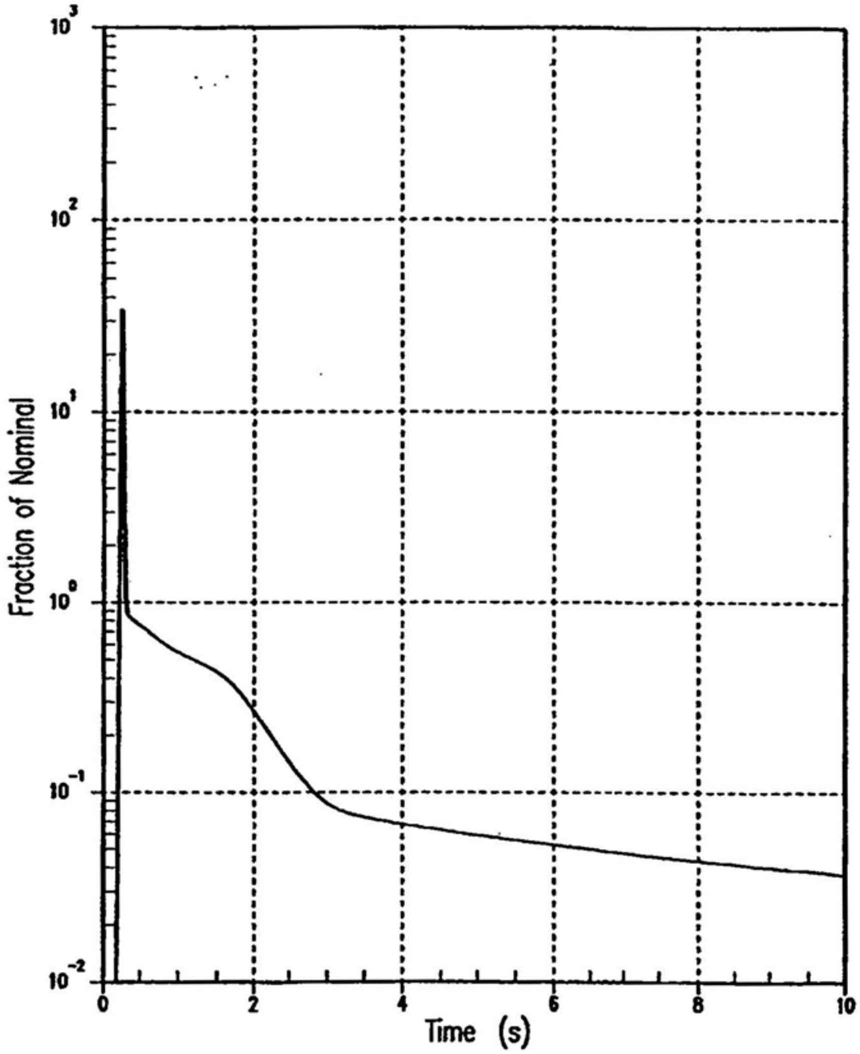


Figure 14.5-27
RCCA Ejection – BOC Zero Power
Reactor Power vs. Time

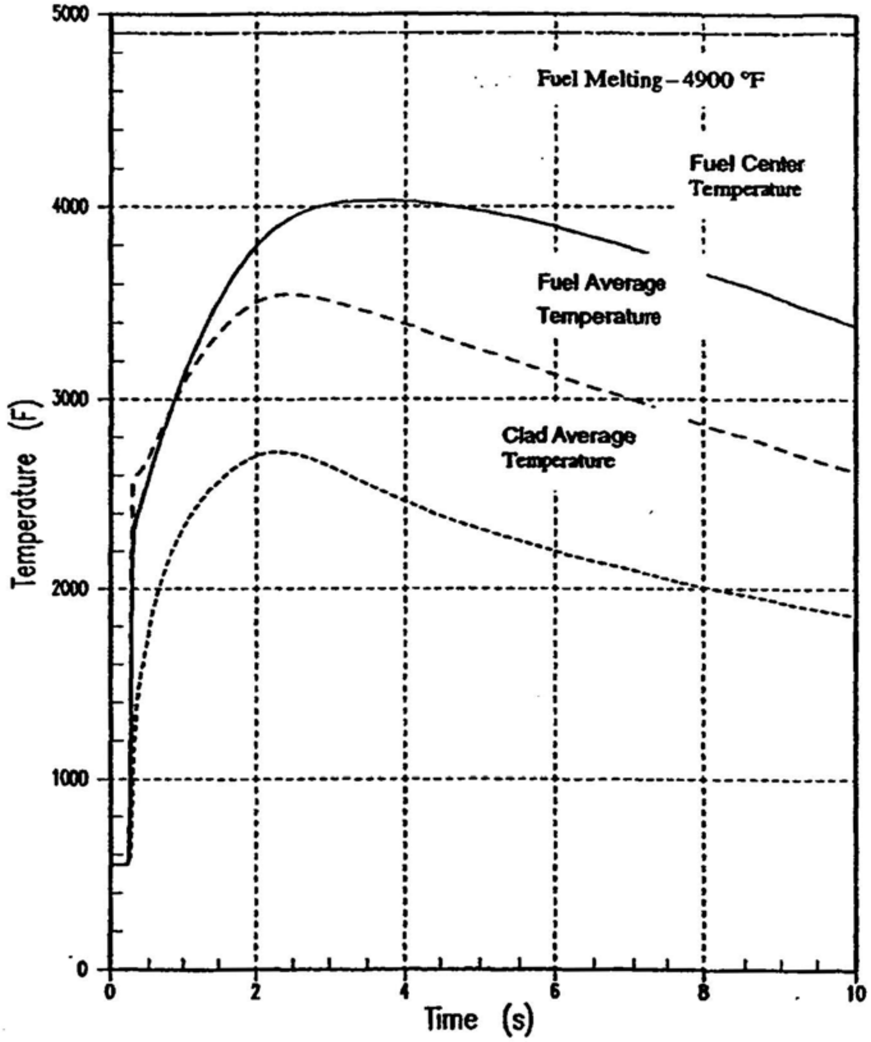


Figure 14.5-28
RCCA Ejection – BOC Zero Power
Fuel and Clad Temperatures vs. Time

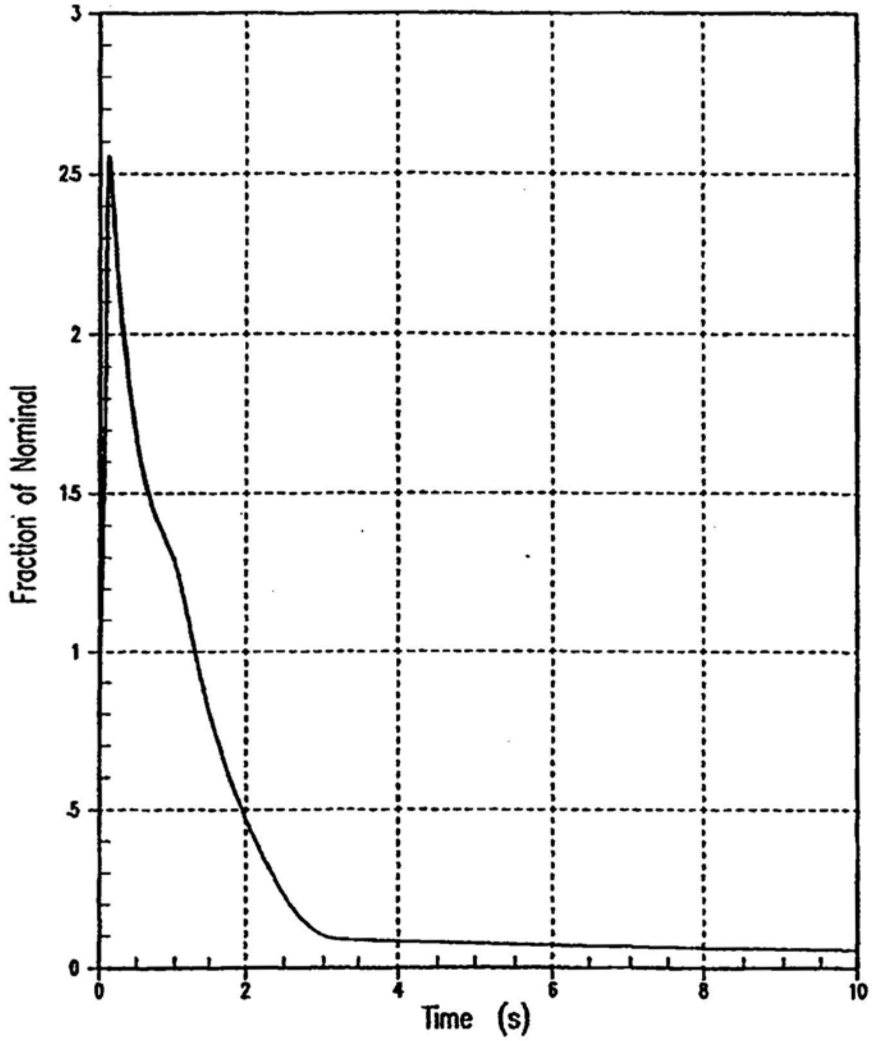


Figure 14.5-29
RCCA Ejection – EOC Full Power
Reactor Power vs. Time

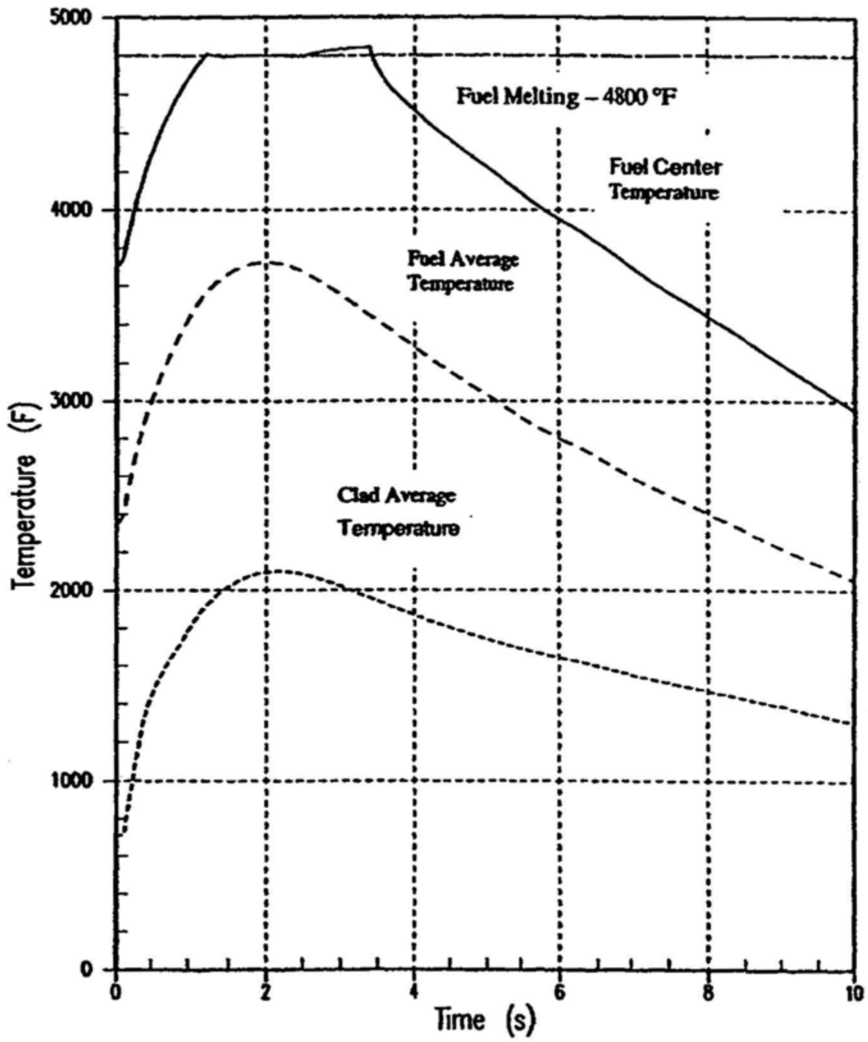


Figure 14.5-30
RCCA Ejection – EOC Full Power
Fuel and Clad Temperatures vs. Time

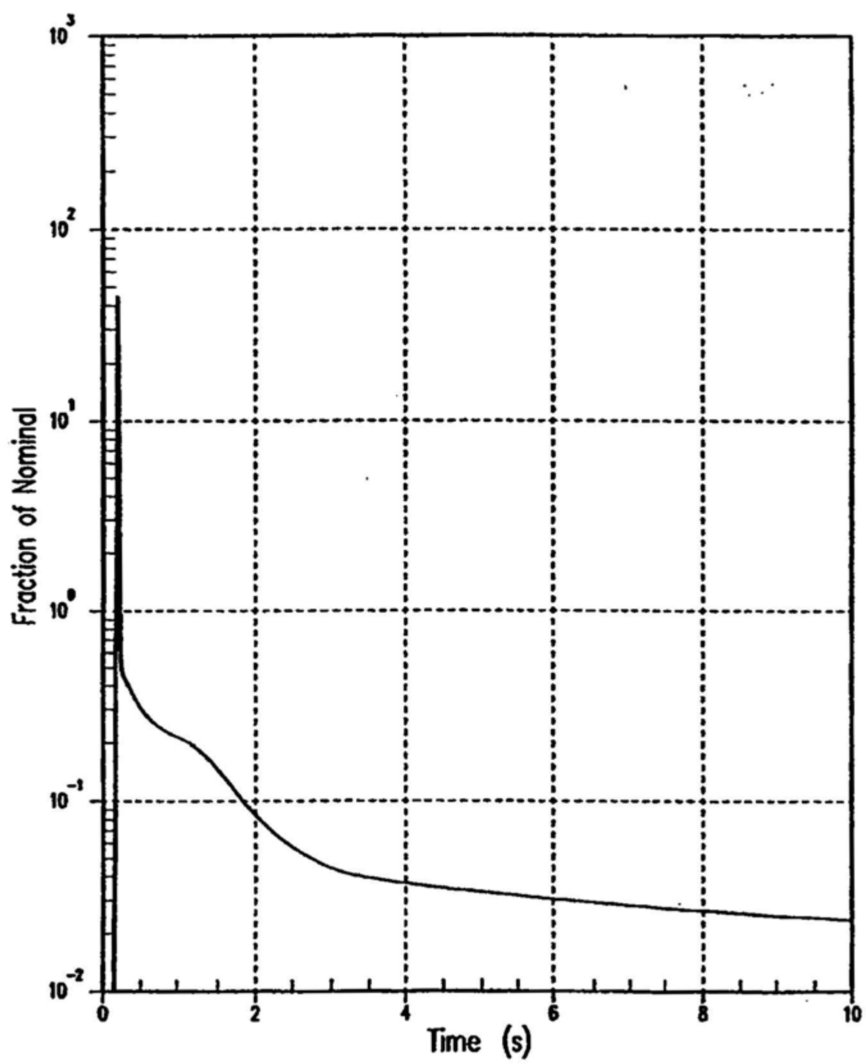


Figure 14.5-31
RCCA Ejection – EOC Zero Power
Reactor Power vs. Time

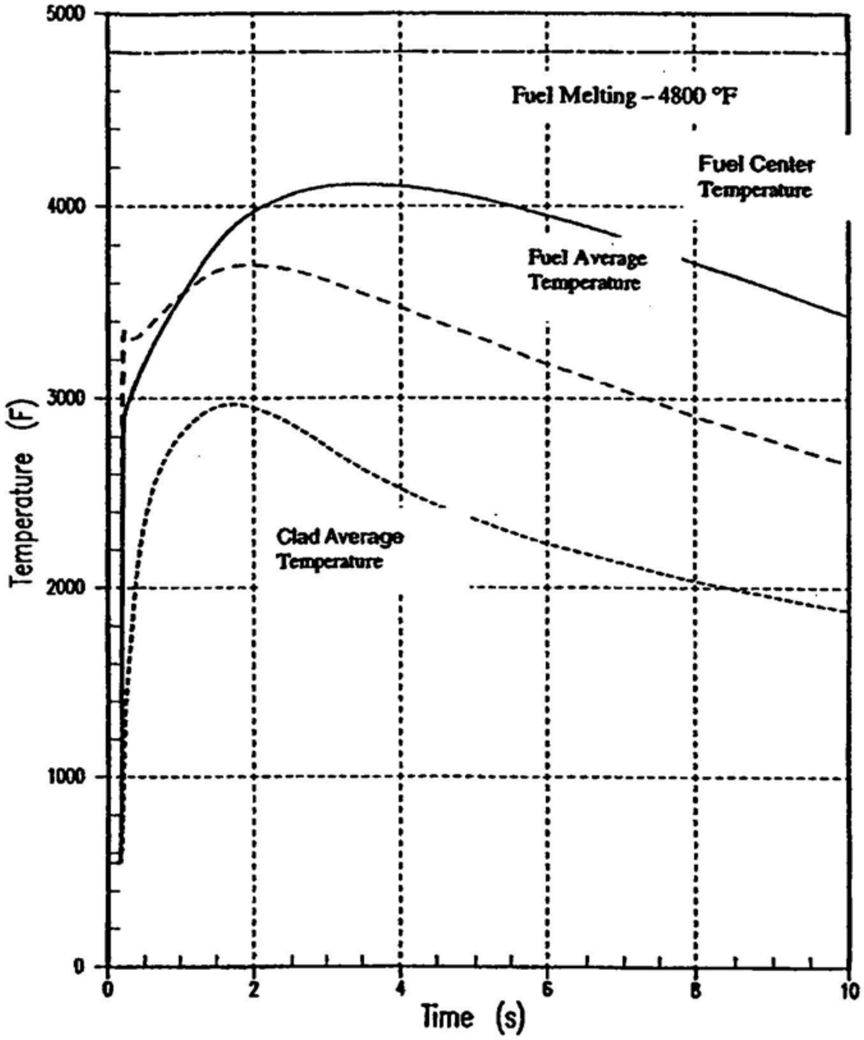
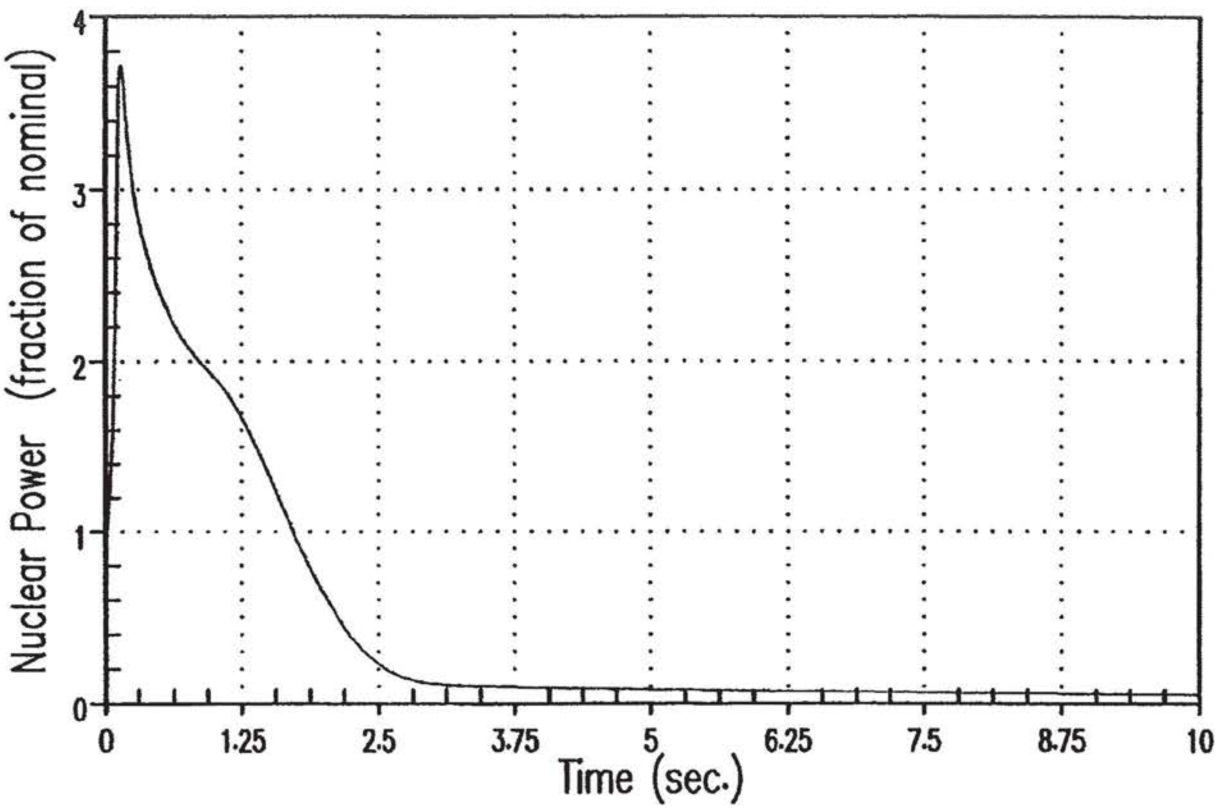


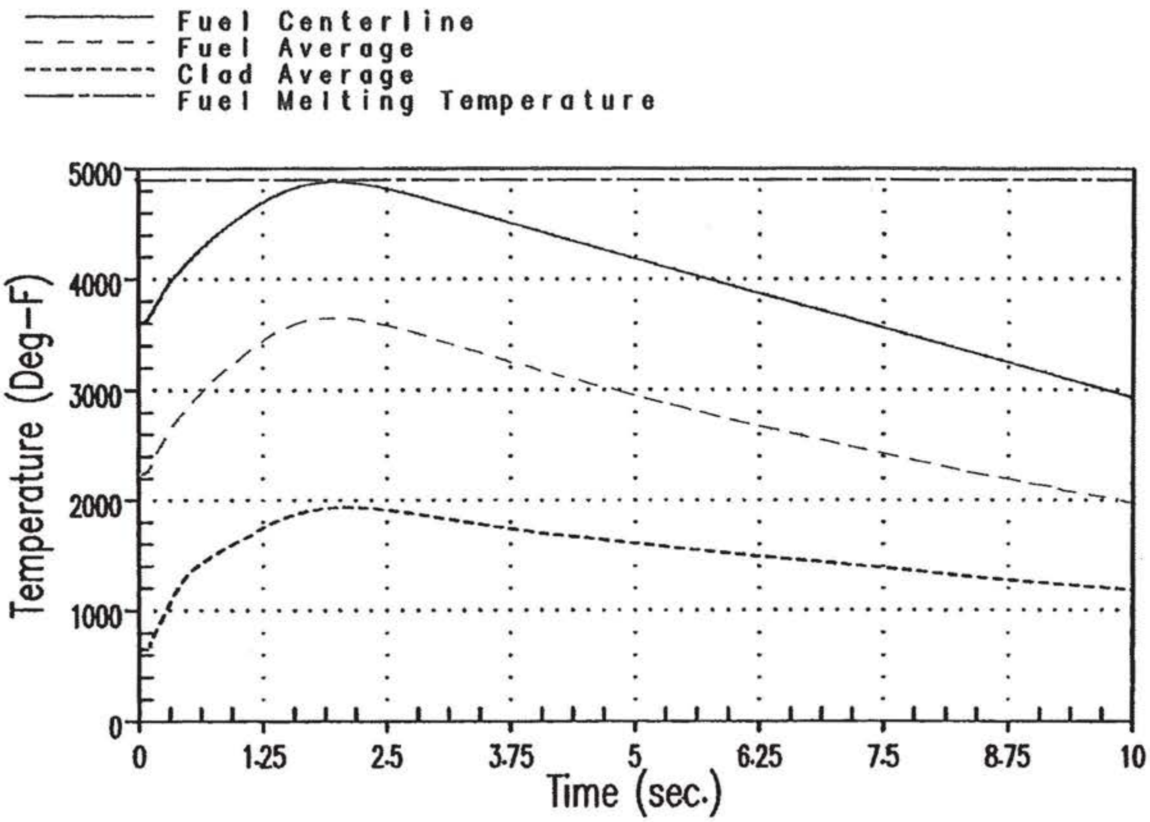
Figure 14.5-32
RCCA Ejection – EOC Zero Power
Fuel and Clad Temperatures vs. Time



RCCA EJECTION – BOC FULL POWER
REACTOR POWER vs. TIME

DWN: KJF	DATE: 10–25–12	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED:	CAD FILE: U14533.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-33 REV. 31*

01193868



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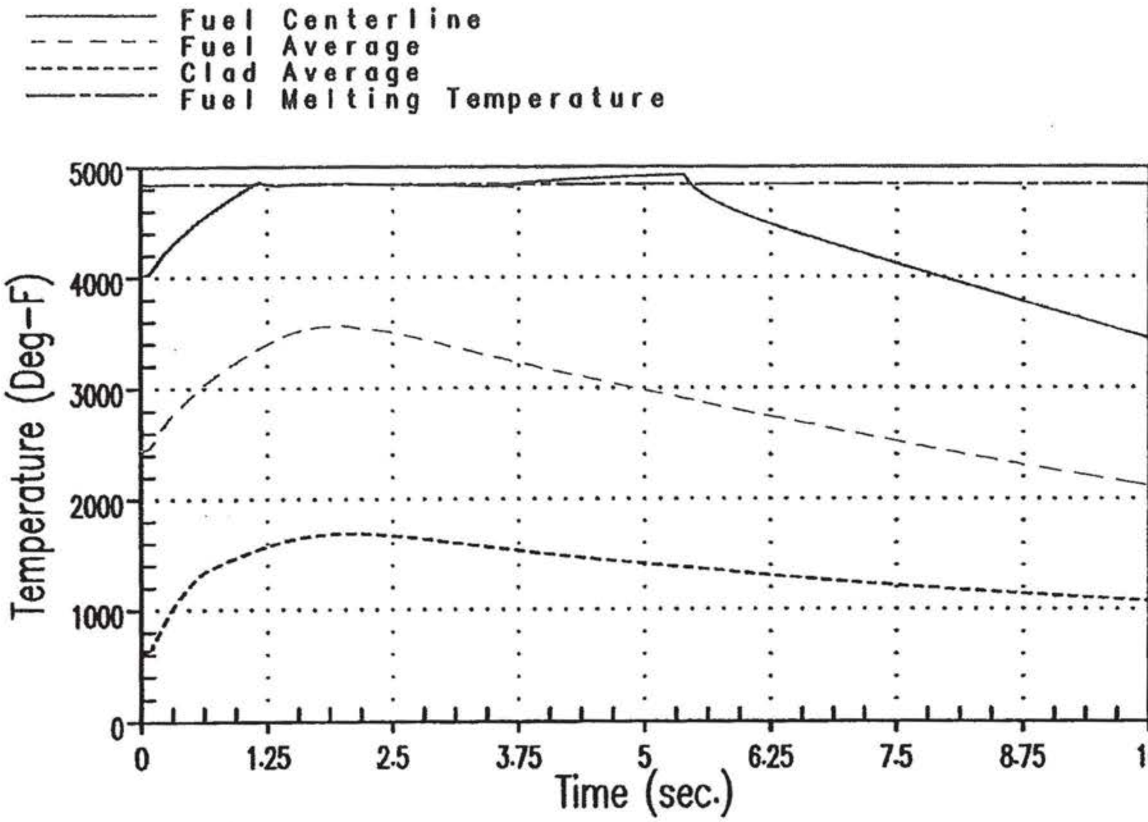
RCCA EJECTION – BOC FULL POWER
FUEL AND CLAD TEMPERATURES vs. TIME FOR UO_2 FUEL CASE

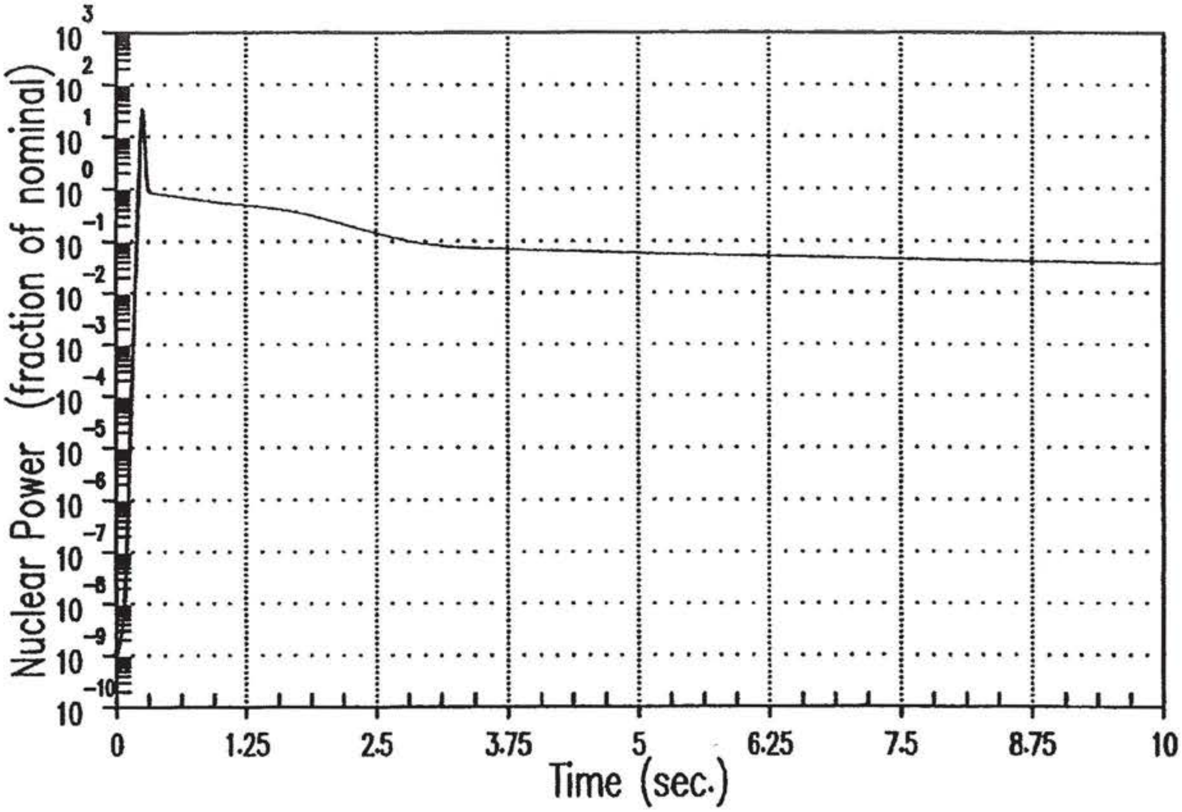
DWN: KJF	DATE: 10-25-12	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U14534.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-34 REV. 31*	



01193868

RCCA EJECTION – BOC ,FULL POWER
FUEL AND CLAD TEMPERATURES vs. TIME FOR GADOLINIA FUEL CASE

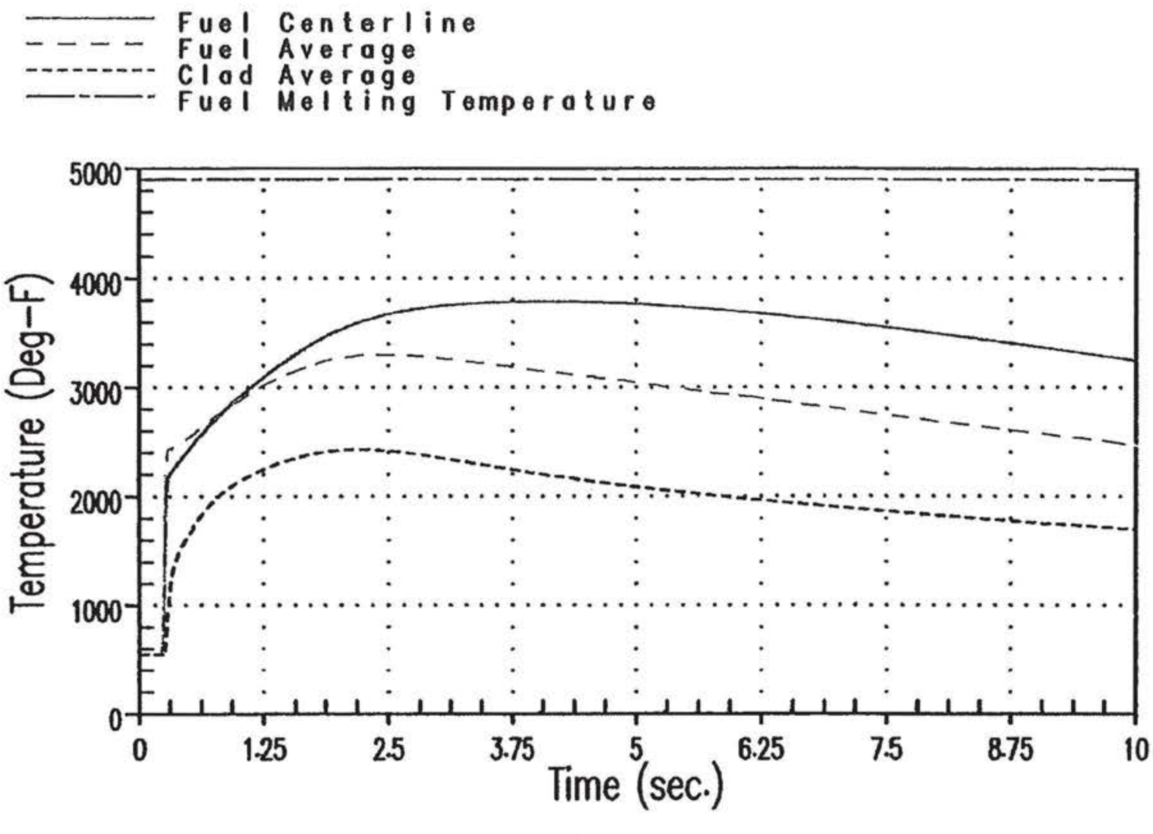
DWN: KJF	DATE: 10–25–12	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U14535.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-35 REV. 31*	



RCCA EJECTION – BOC ZERO POWER
REACTOR POWER vs. TIME

DWN: KJF	DATE: 10-25-12	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED:	CAD FILE: U14536.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	FIGURE 14.5-36 REV. 31

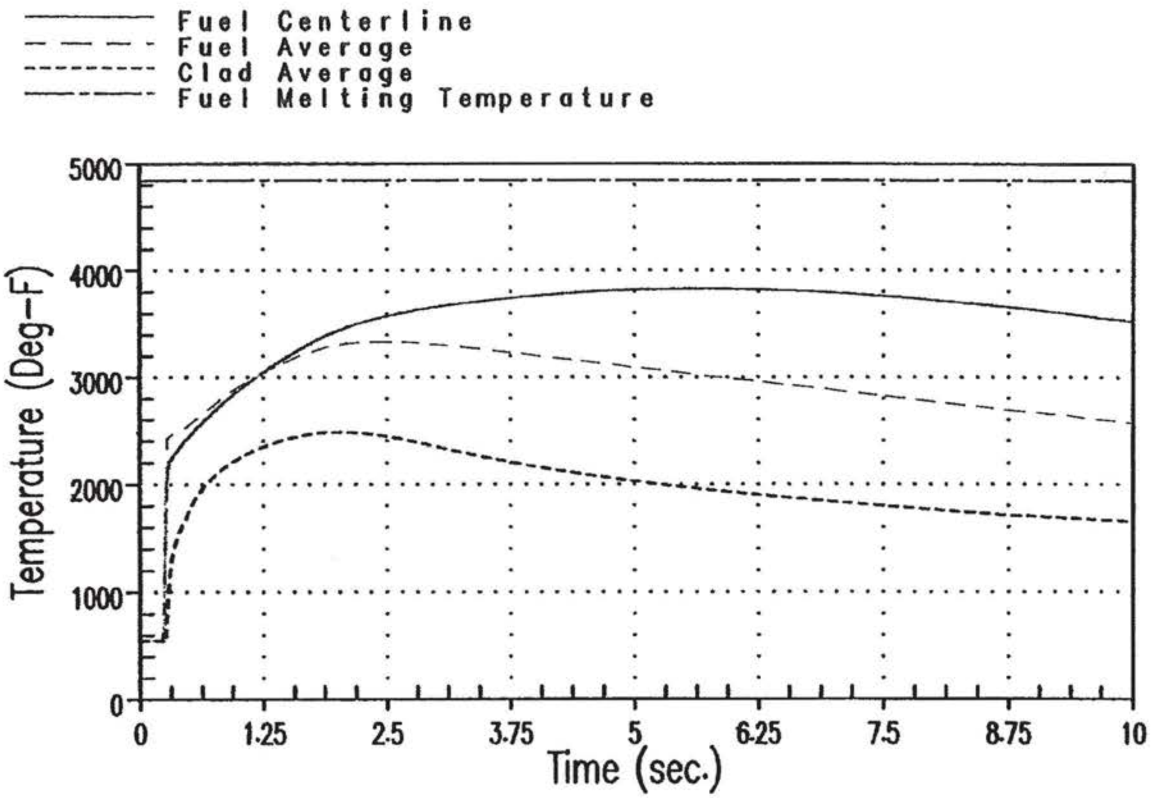
01193868



RCCA EJECTION – BOC ZERO POWER
FUEL AND CLAD TEMPERATURES vs. TIME FOR UO_2 FUEL CASE

DWN: KJF	DATE: 10-25-12	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED:	CAD FILE: U14537.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-37 REV. 31*

01193868



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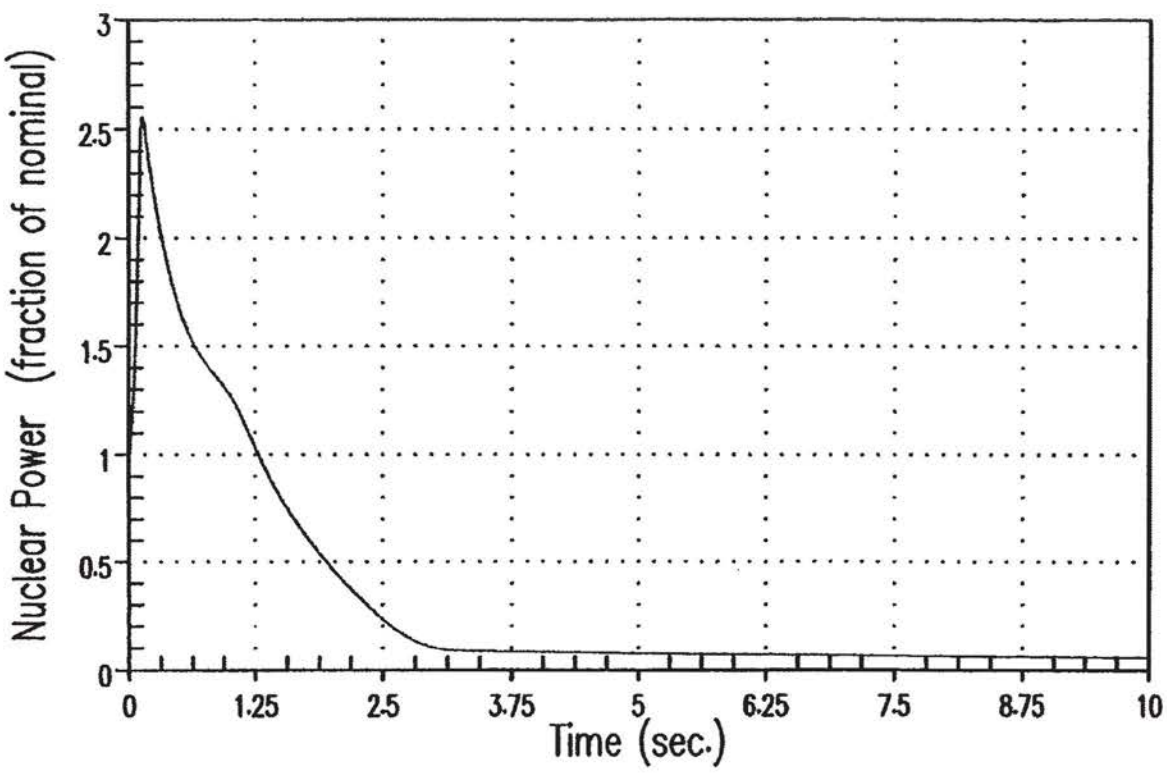
RCCA EJECTION – BOC ZERO POWER
FUEL AND CLAD TEMPERATURES vs. TIME FOR GADOLINIA FUEL CASE

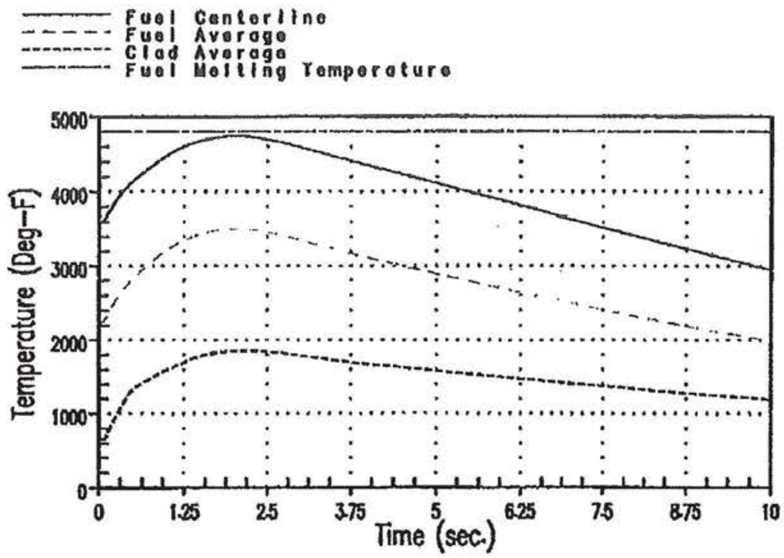
DWN: KJF	DATE: 10–25–12	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U14538.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	FIGURE 14.5-38 REV. 31	



RCCA EJECTION – EOC FULL POWER
REACTOR POWER vs. TIME

DWN: KJF	DATE: 10–25–12	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED:	CAD FILE: U14539.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-39 REV. 31*

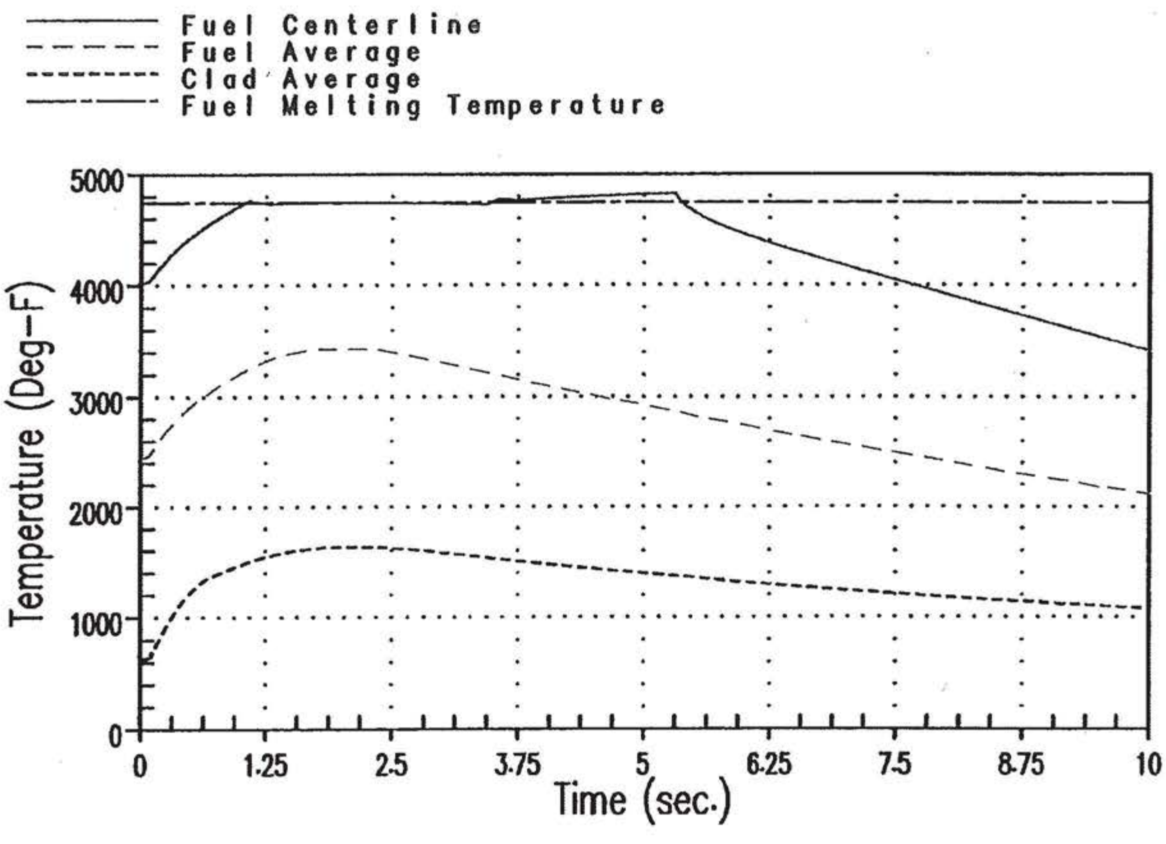
01193868



RCCA EJECTION - EOC FULL POWER
FUEL AND CLAD TEMPERATURES vs. TIME FOR UO_2 FUEL CASE

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE NONE	
CHECKED	CAD FILE U14540.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	FIGURE 14.5-40 REV. 31	

01193868



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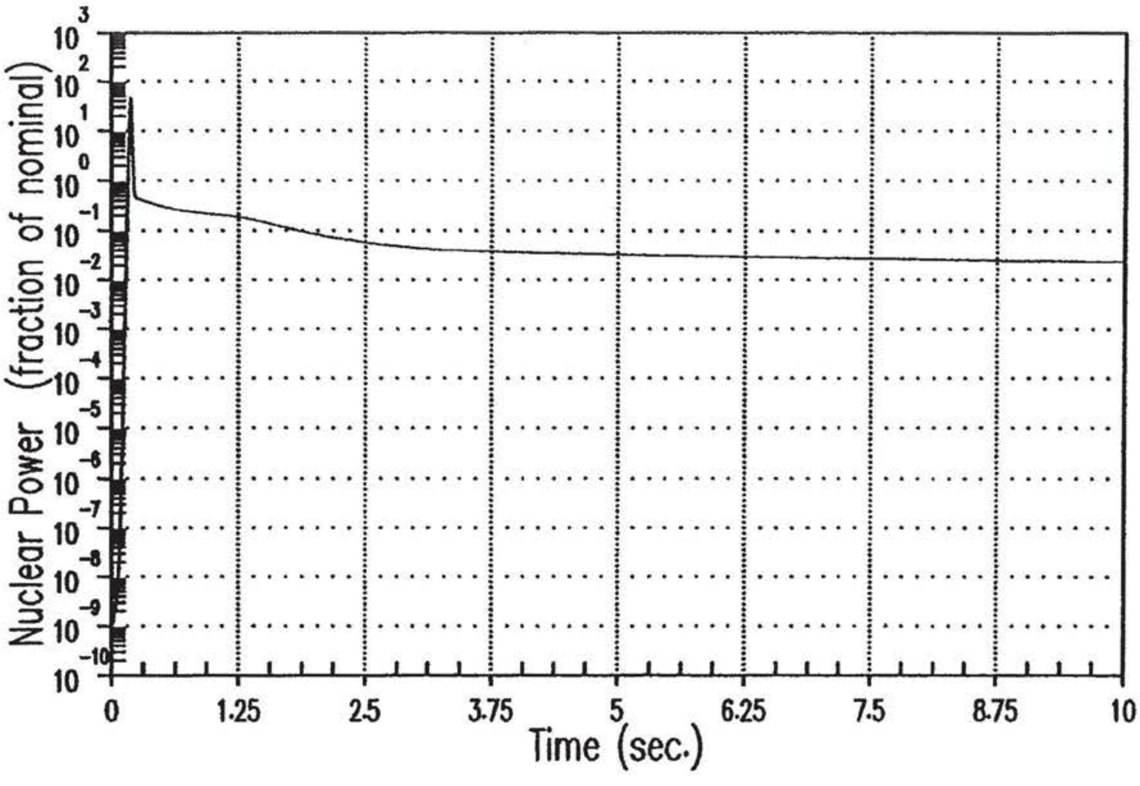
RCCA EJECTION – EOC FULL POWER
FUEL AND CLAD TEMPERATURES vs. TIME FOR GADOLINIA FUEL CASE

DWN: KJF	DATE: 10-25-12	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U14541.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-41 REV. 31*	



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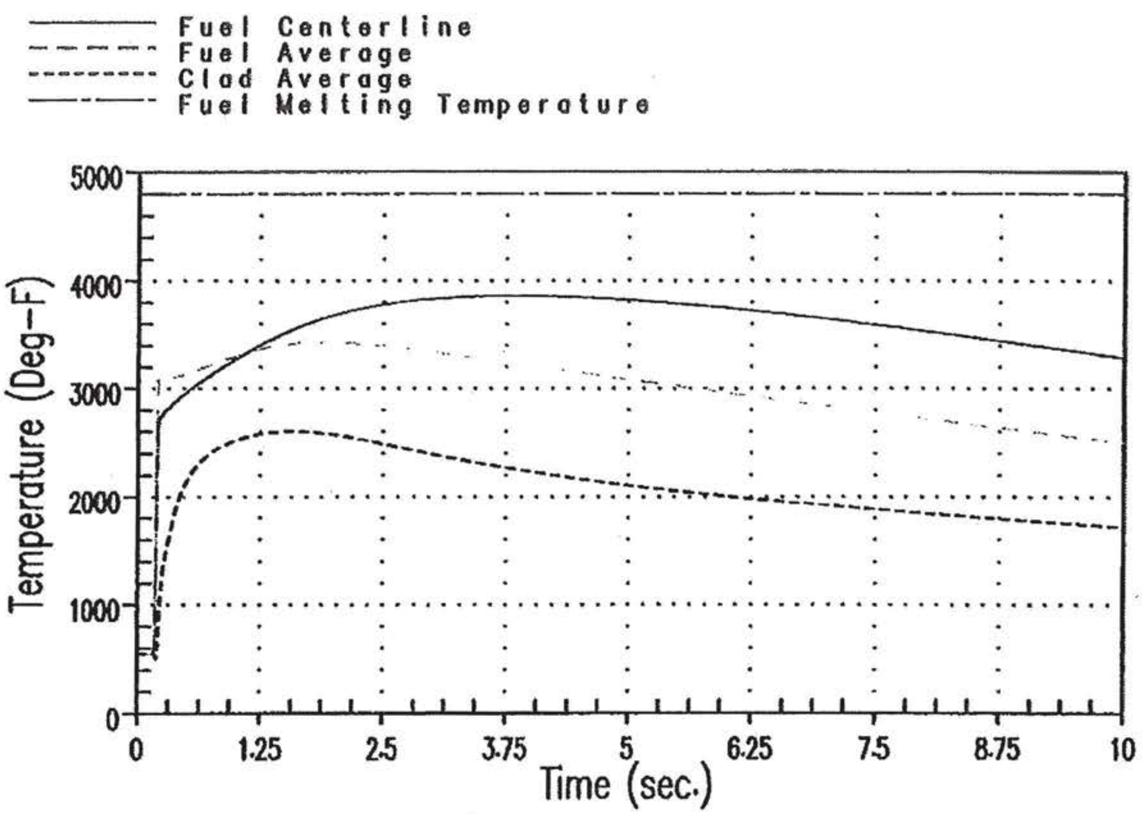
RCCA EJECTION – EOC ZERO POWER
REACTOR POWER vs. TIME

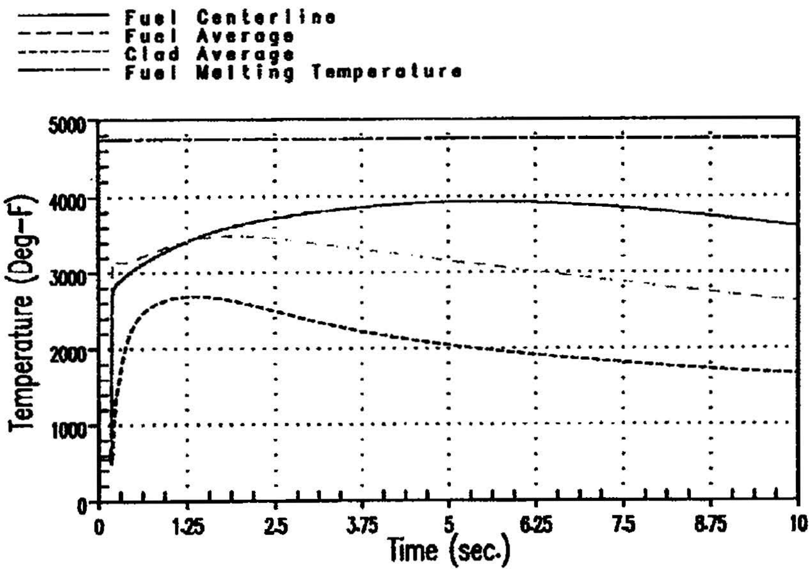
DWN: KJF	DATE: 10–25–12	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U14542.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-42 REV. 31*	



RCCA EJECTION – EOC ZERO POWER
FUEL AND CLAD TEMPERATURES vs. TIME

DWN: KJF	DATE: 10-25-12	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED:	CAD FILE: U14543.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	FIGURE 14.5-43 REV. 31

01193868

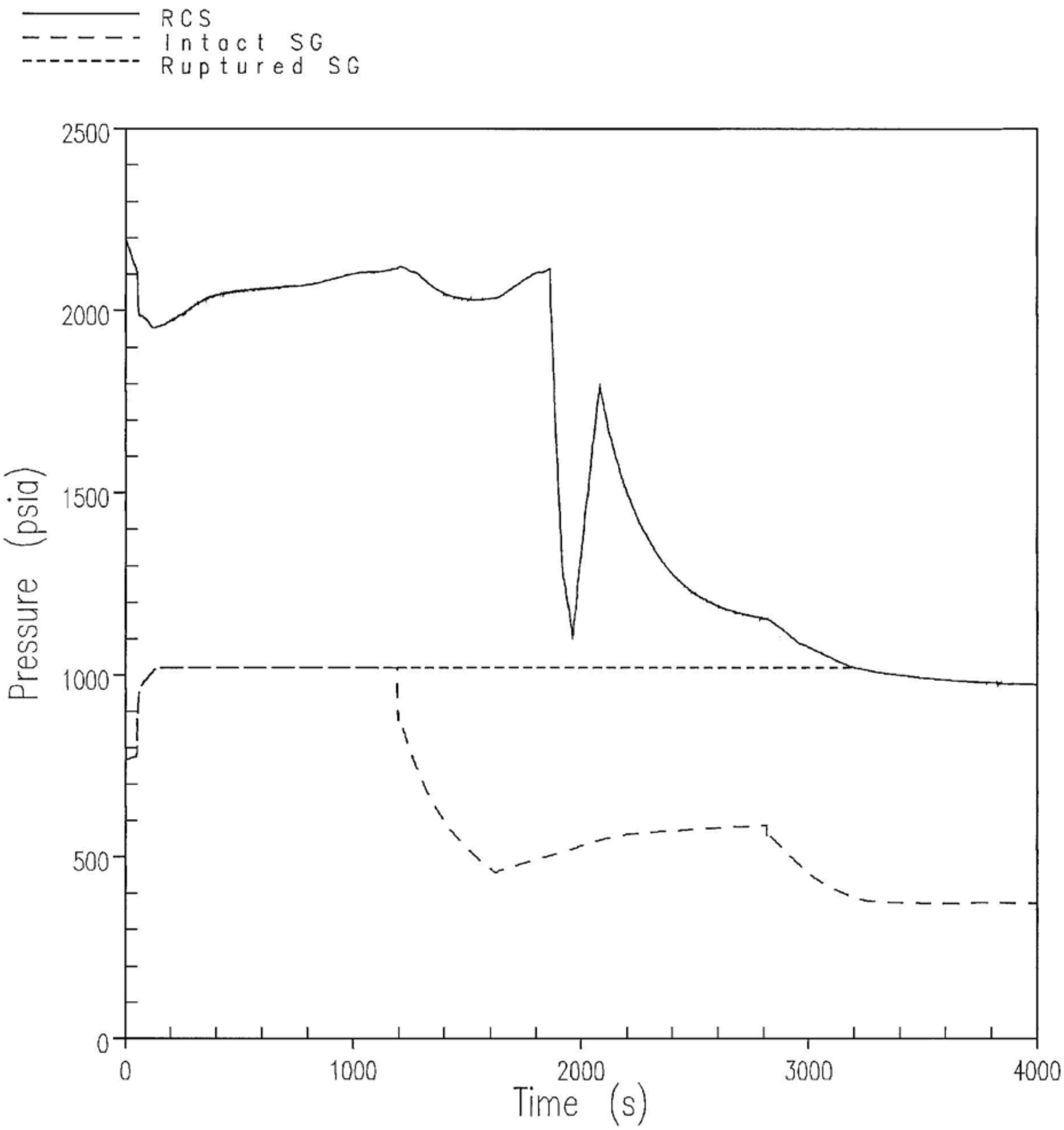


RCCA EJECTION – EOC ZERO POWER
FUEL AND CLAD TEMPERATURES vs. TIME FOR GADOLINIA FUEL CASE

DWN KJF	DATE 2-4-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE UI4544.DGN		*FIGURE 14.5-44 REV. 31*

01193868

FIGURE 14.5-44

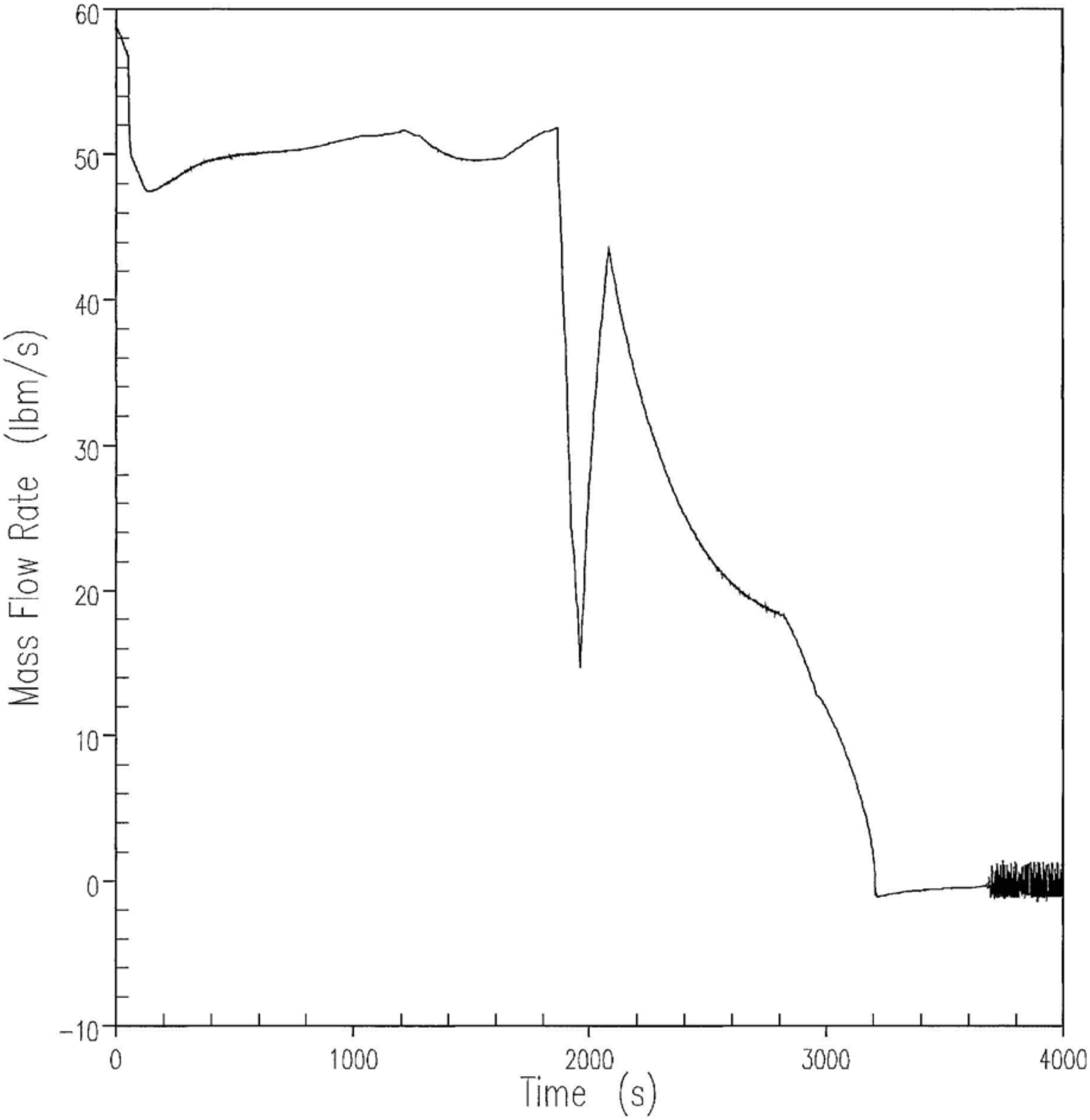


Steam Generator Margin to Overfill Analysis

RCS and Secondary Pressures

DWN: KJF	DATE: 5–14–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE		
CHECKED:	CAD FILE: U14545.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-45*	*REV. 33*	

01406407

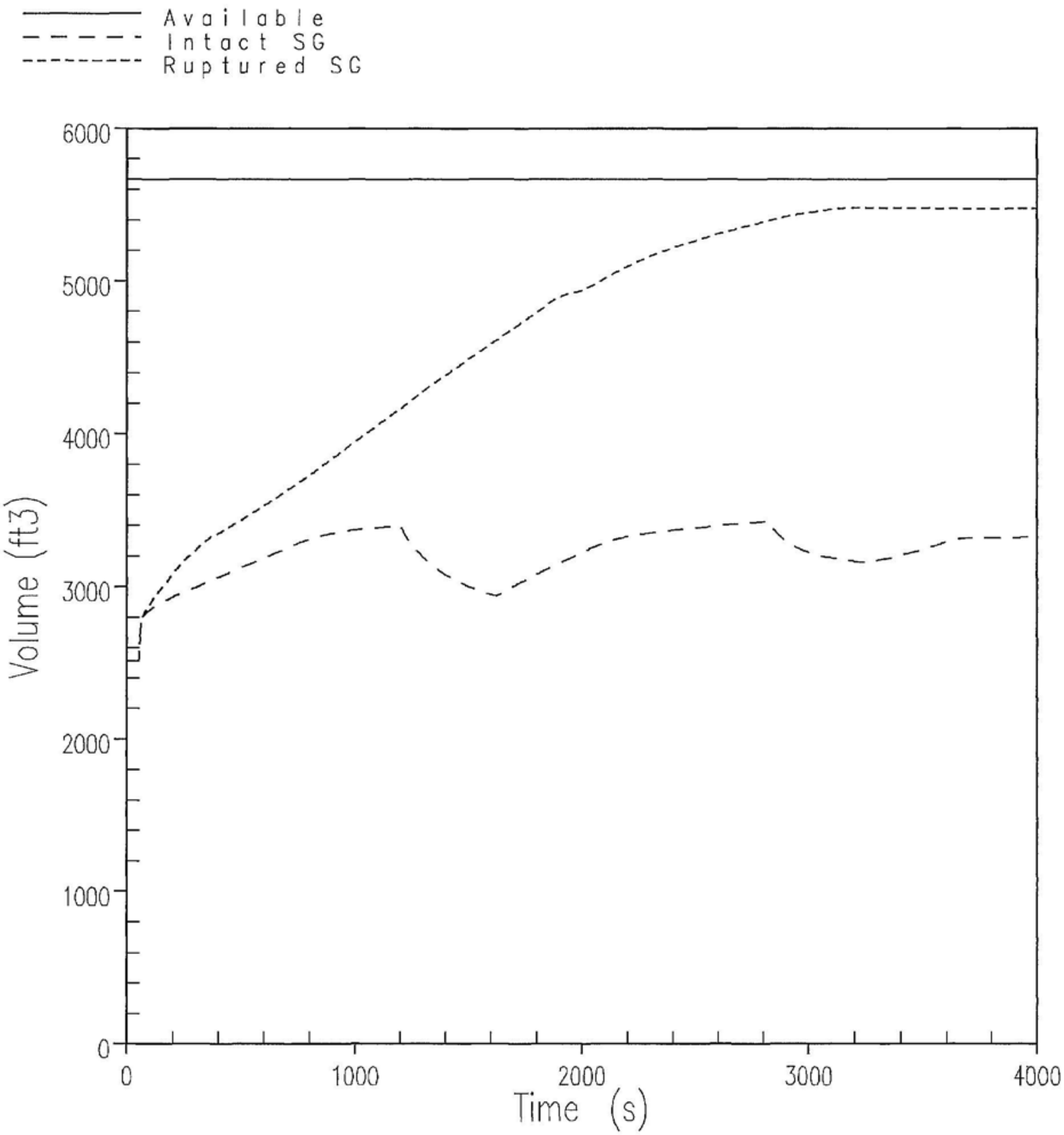


Steam Generator Margin to Overfill Analysis

Primary-to-Secondary Break Flow Rate

DWN: KJF	DATE: 5–14–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED:	CAD FILE: U14546.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-46* *REV. 33*

01406407



Steam Generator Margin to Overfill Analysis

Steam Generator Water Volumes

DWN: KJF	DATE: 5-14-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
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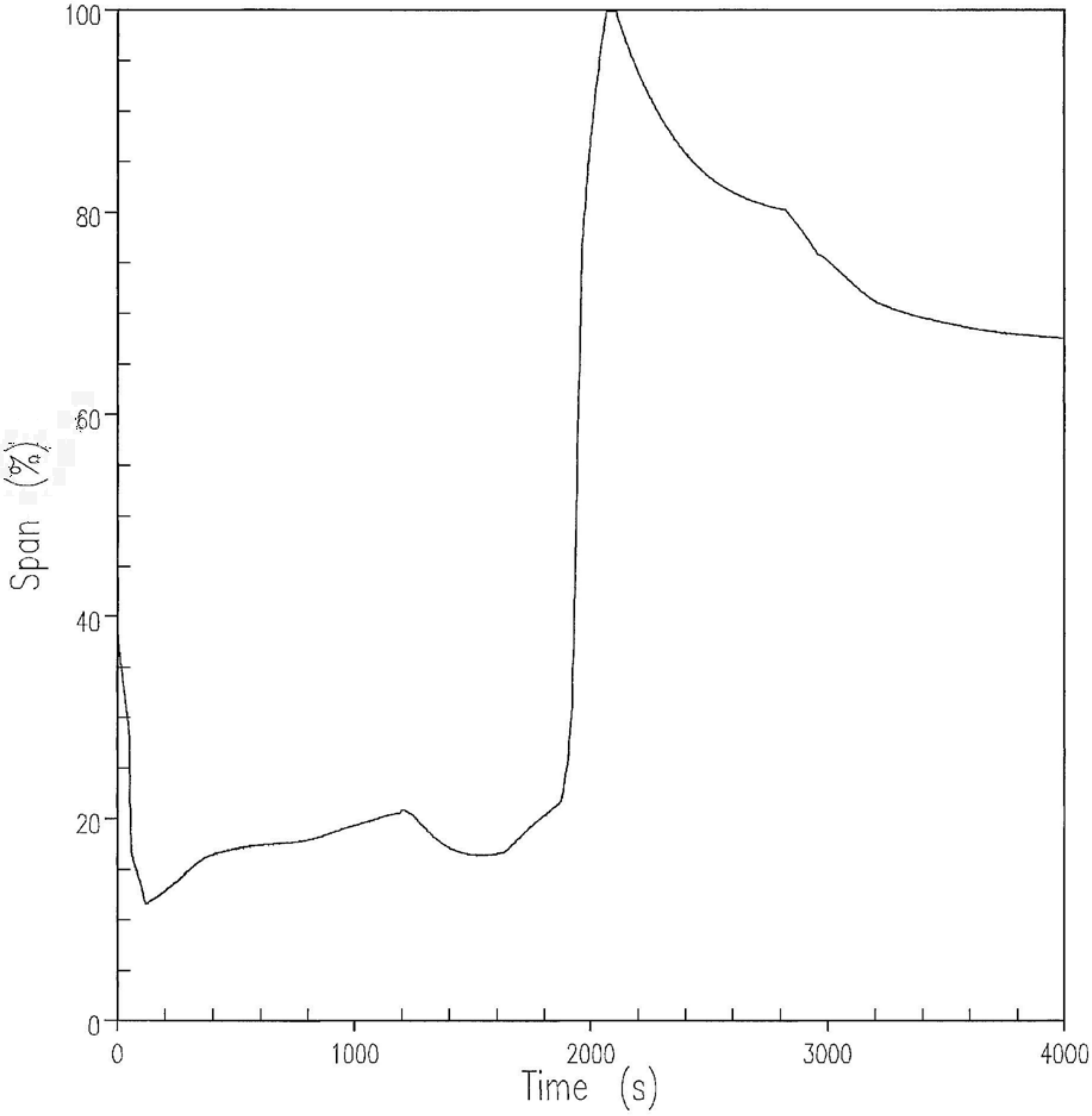
01406407



Steam Generator Margin to Overfill Analysis

Pressurizer Level

DWN: KJF	DATE: 5–14–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
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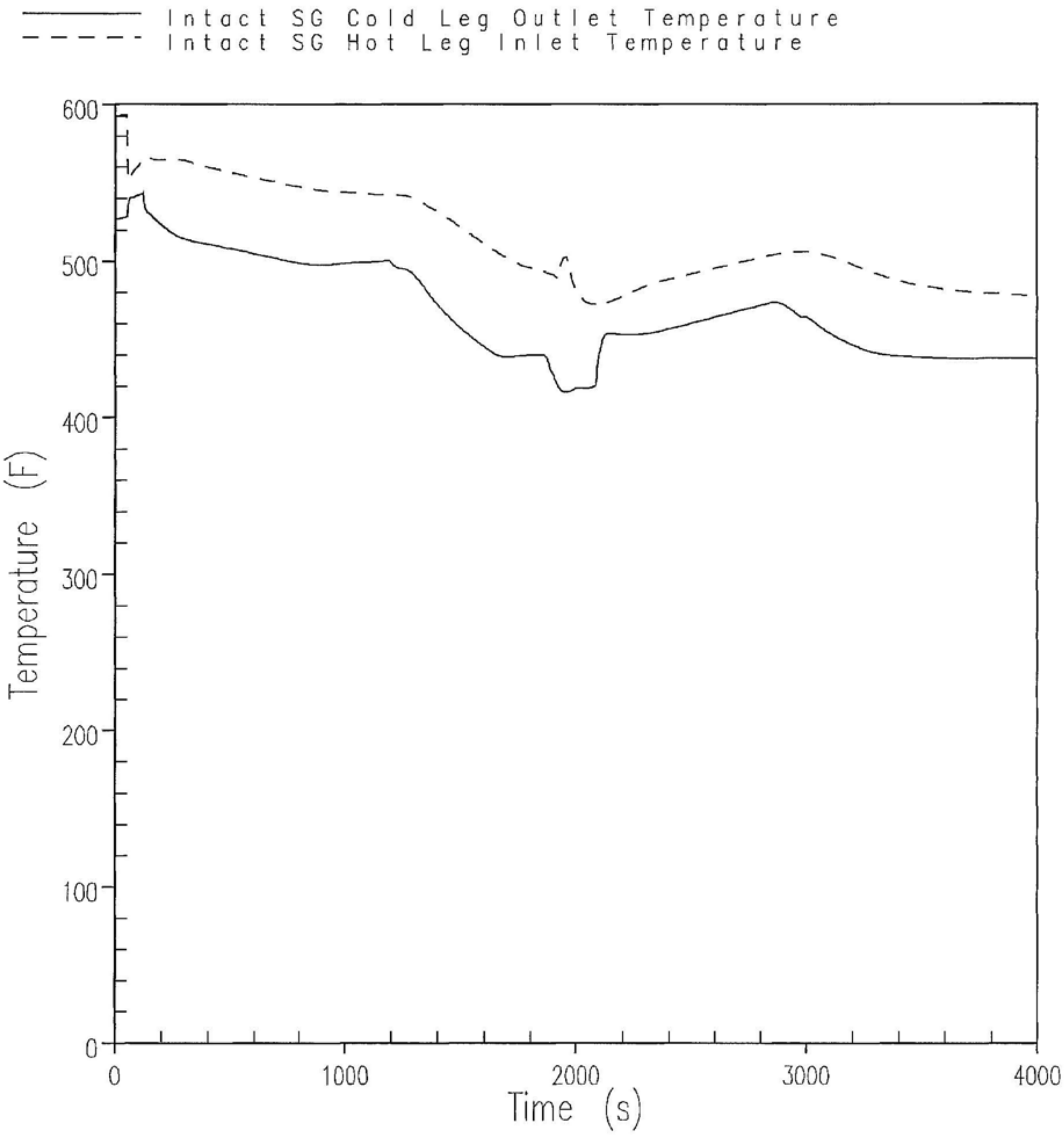
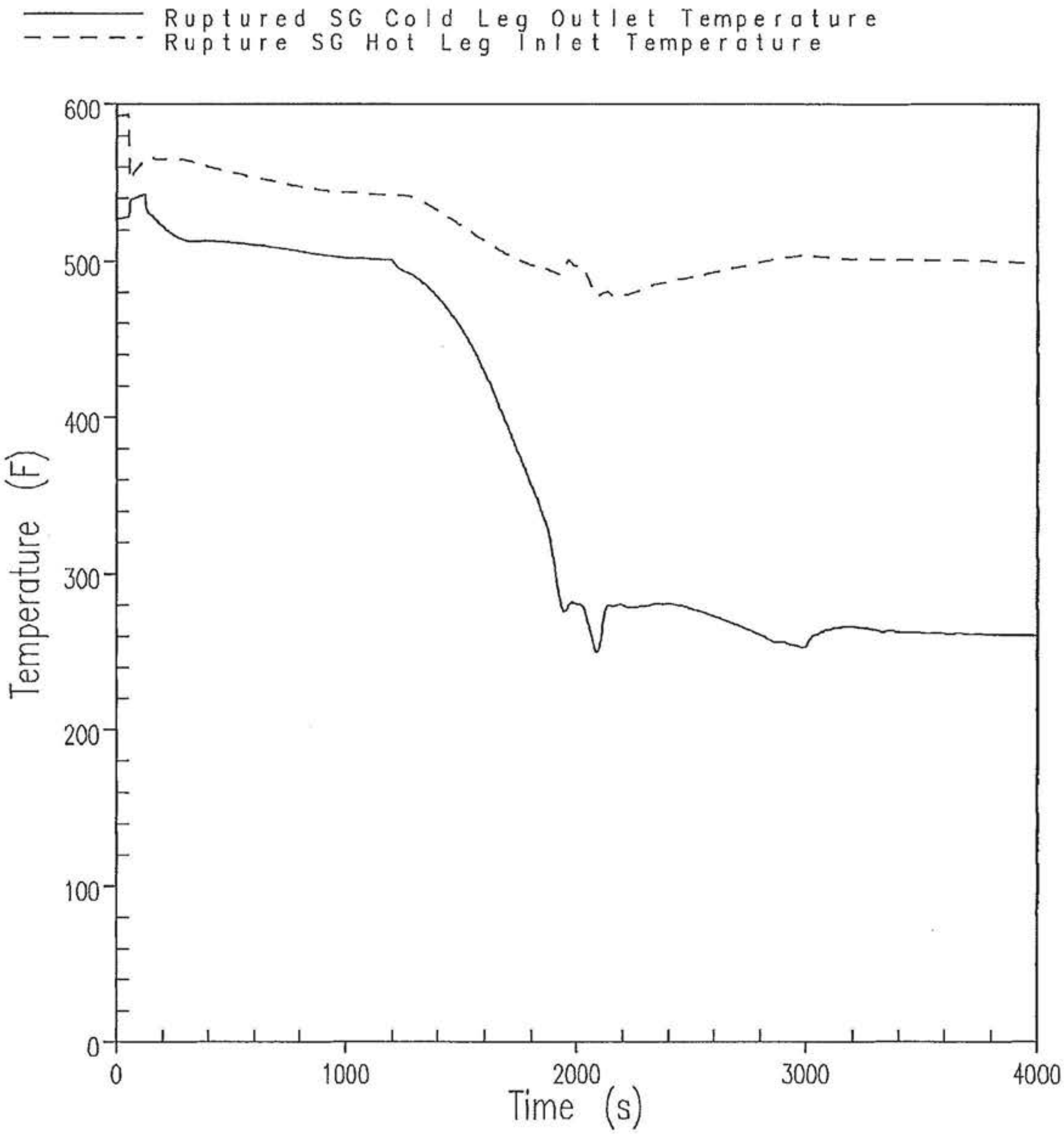
01406407



01406407

Steam Generator Margin to Overfill Analysis
Intact SG Inlet and Outlet Temperatures

DWN: KJF	DATE: 5-14-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U14549.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-49*	*REV. 33*

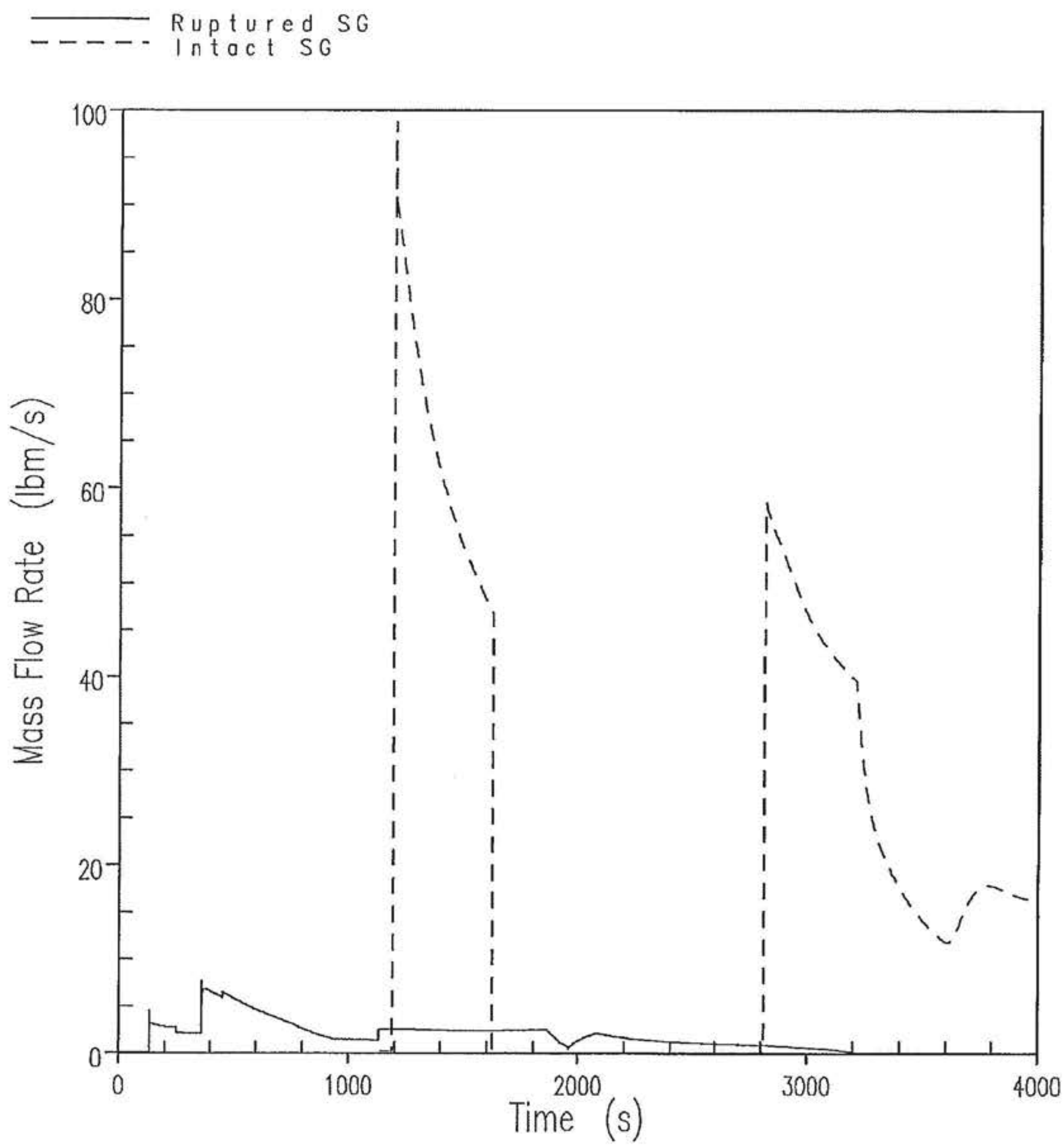


01406407

Steam Generator Margin to Overfill Analysis

Ruptured SG Inlet and Outlet Temperatures

DWN: KJF	DATE: 5-14-14	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE	
CHECKED:	CAD FILE: U14550.DGN		*FIGURE 14.5-50*	*REV. 33*

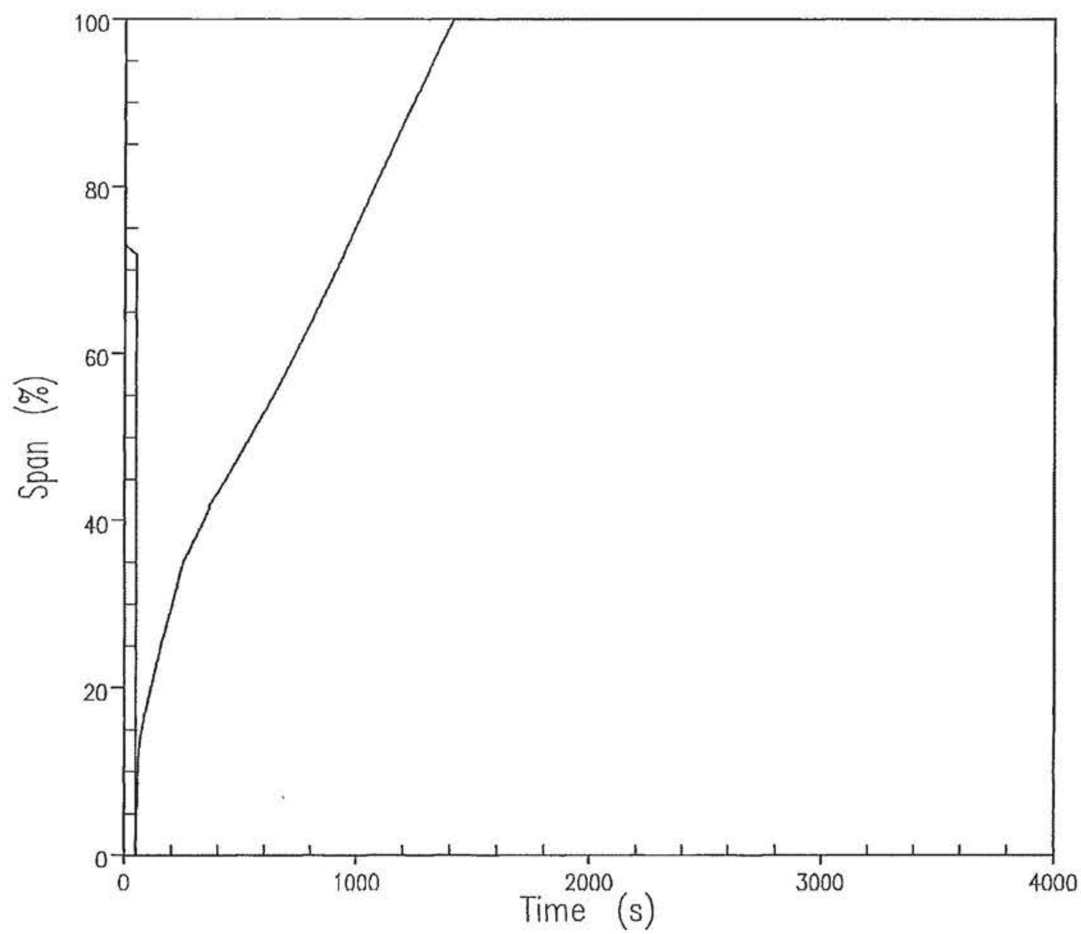


01406407

Steam Generator Margin to Overfill Analysis

SG Steam Releases

DWN: KJF	DATE: 5-14-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U14551.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-51*	*REV. 33*

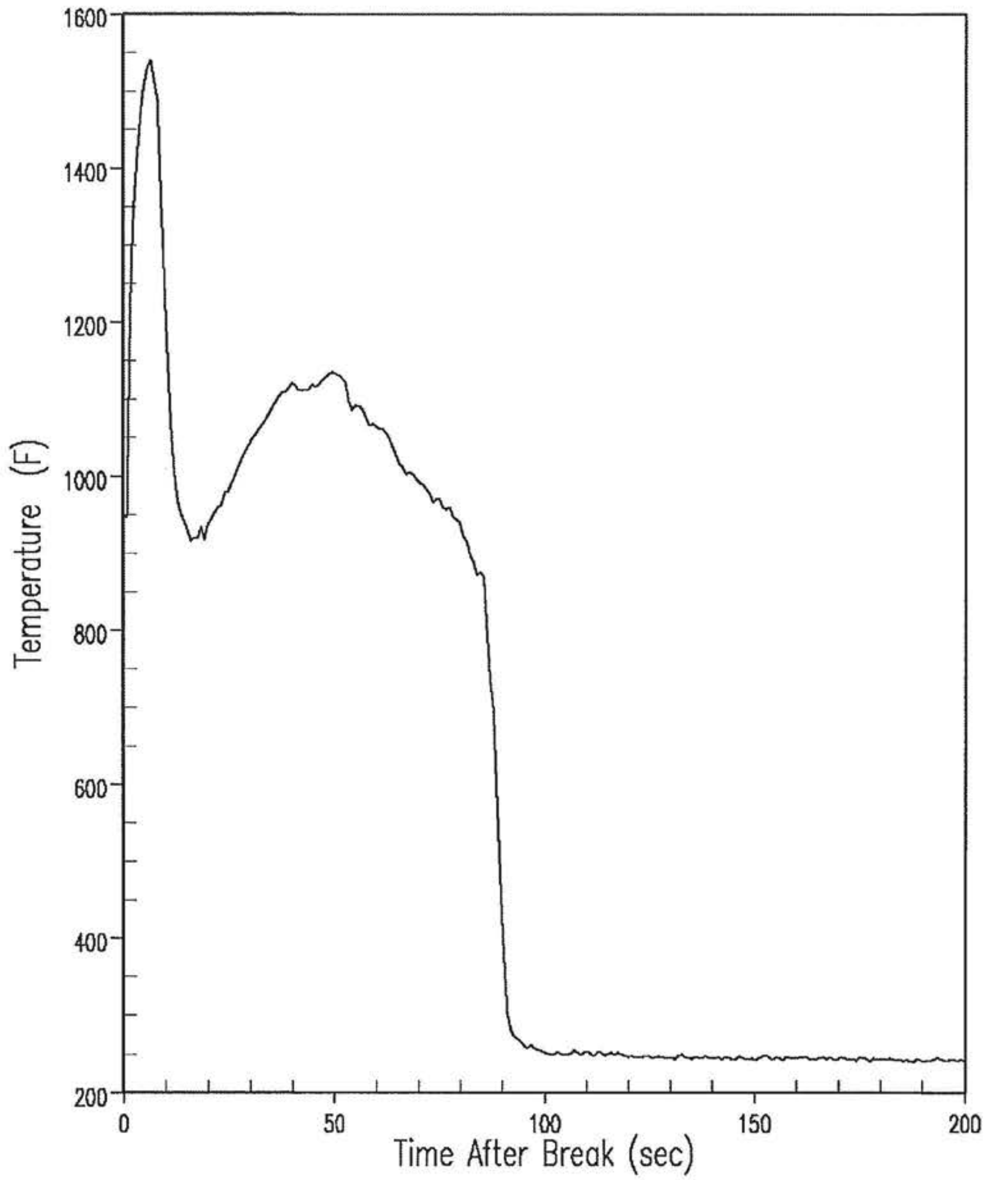


Steam Generator Margin to Overfill Analysis

Ruptured SG Narrow Range Level

DWN: KJF	DATE: 5-14-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U14552.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.5-52*	*REV. 33*

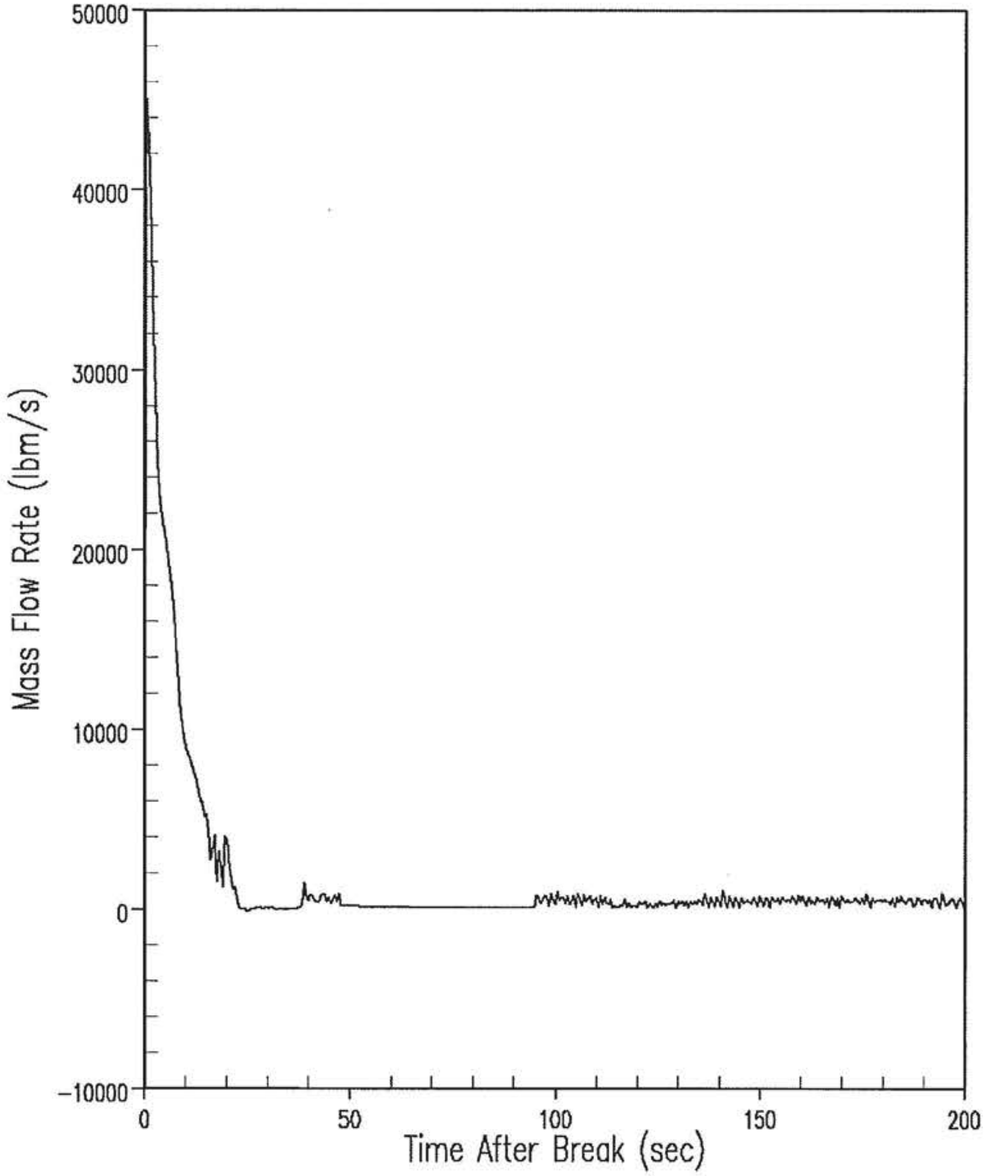
01406407



PRAIRIE ISLAND UNIT 1 AND 2 CLAD TEMPERATURE TRANSIENT AT THE LIMITING ELEVATION FOR THE LIMITING PCT CASE

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14601.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.6-1*	*REV. 33*

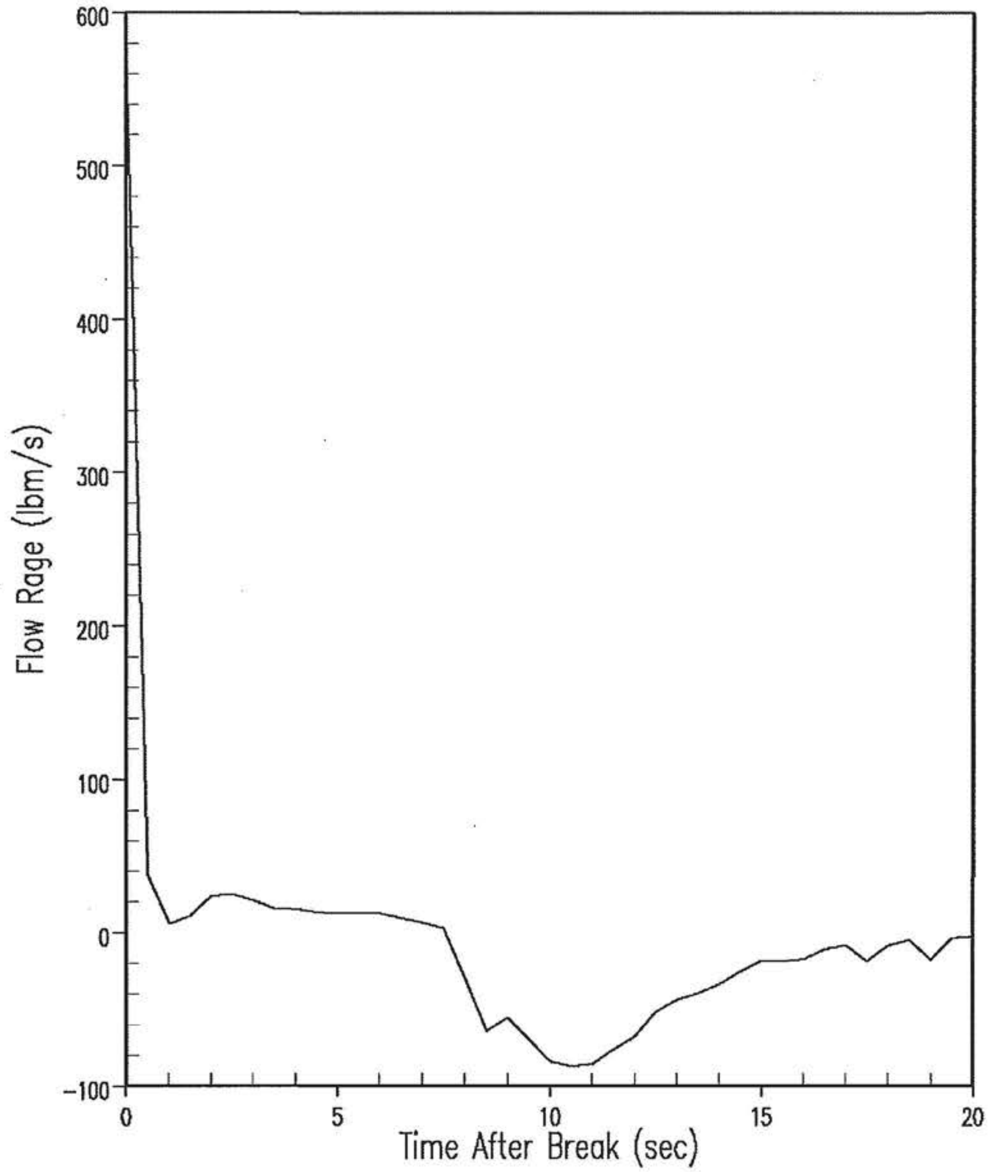
01386642



PRAIRIE ISLAND UNIT 1 & 2 VESSEL SIDE BREAK FLOW FOR THE LIMITING PCT TRANSIENT

DWN KJF	DATE 2-20-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14602.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.6-2 REV. 33*

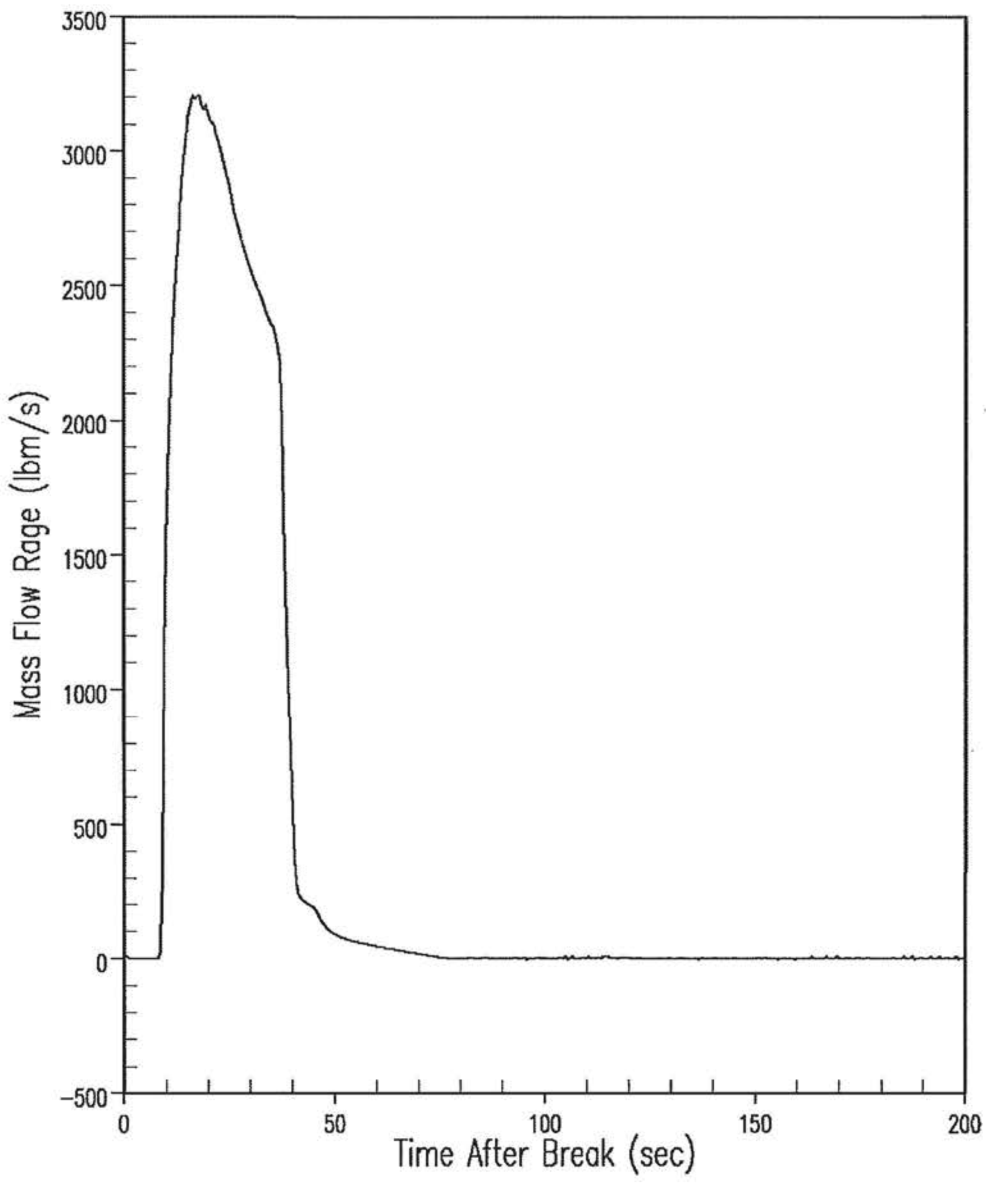
01386642



PRAIRIE ISLAND UNIT 1 AND 2
TOTAL FLOW AT THE BOTTOM OF THE CORE FOR THE LIMITING PCT TRANSIENT

DWN KJF	DATE 2–24–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14603.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.6-3 REV. 33*

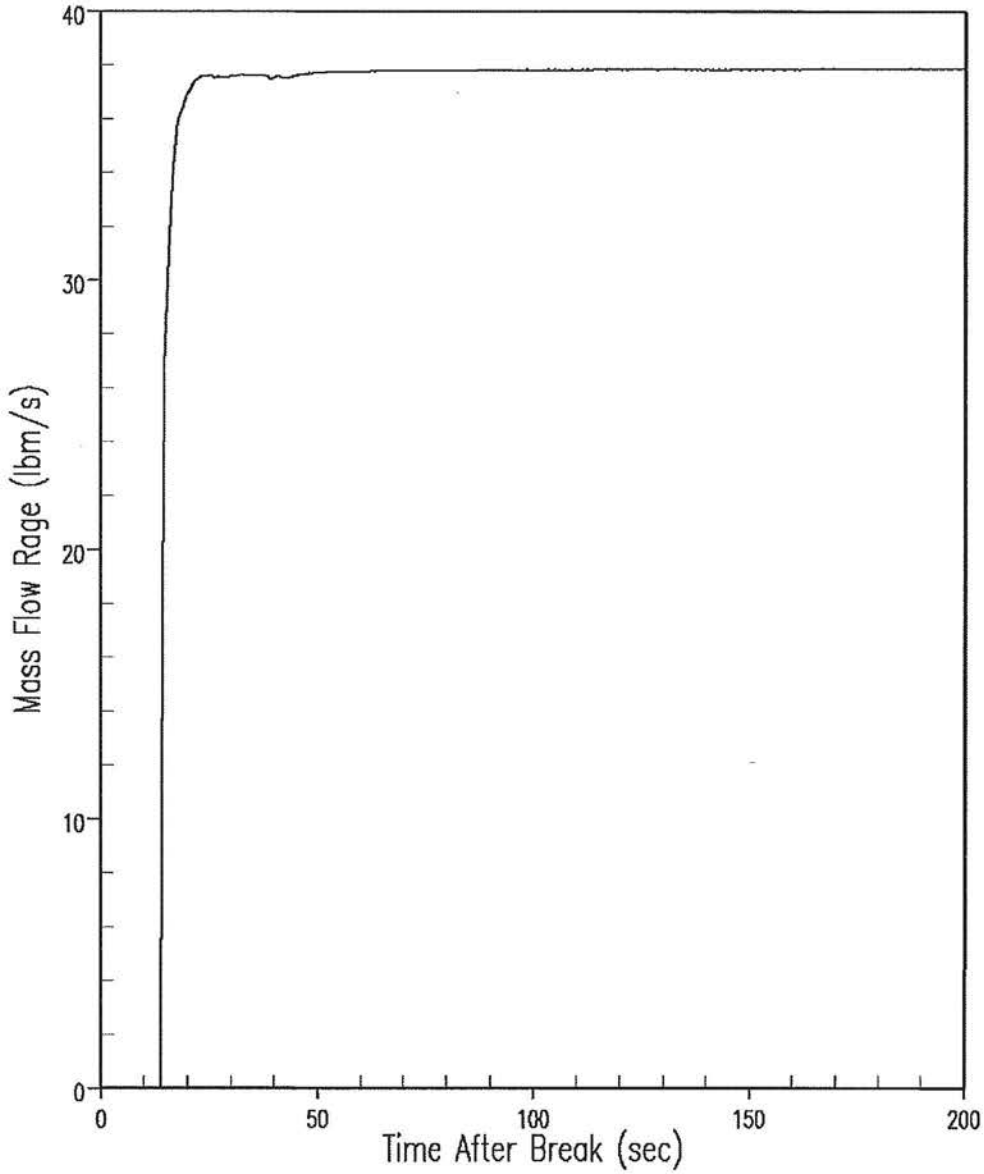
01386642



PRAIRIE ISLAND UNIT 1 AND 2
ACCUMULATOR INJECTION FLOW FOR THE LIMITING PCT TRANSIENT

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14604.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.6-4* *REV. 33*

01386642



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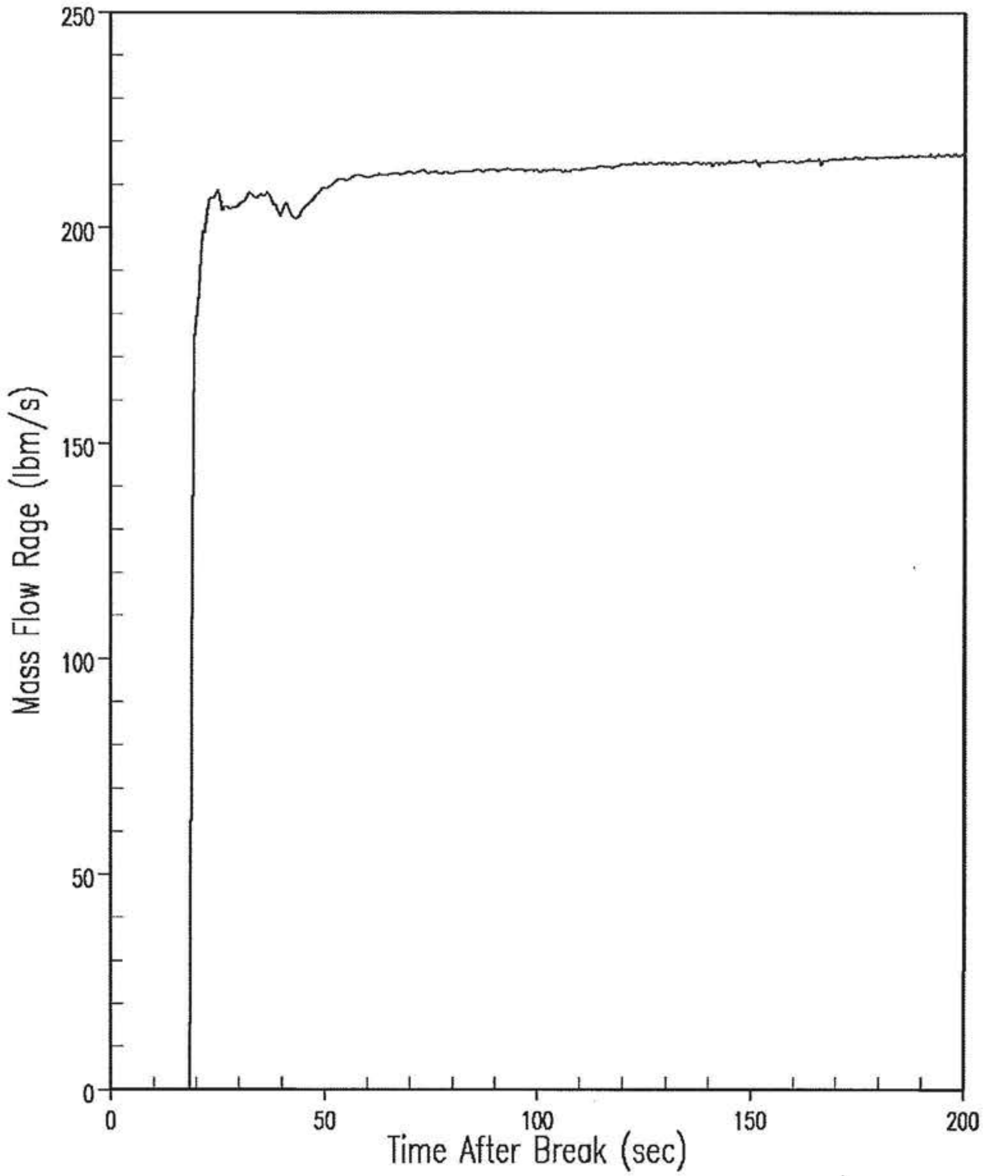
PRAIRIE ISLAND UNIT 1 AND 2
HIGH–HEAD SAFETY INJECTION FLOW FOR THE LIMITING PCT TRANSIENT (NO LOOP)

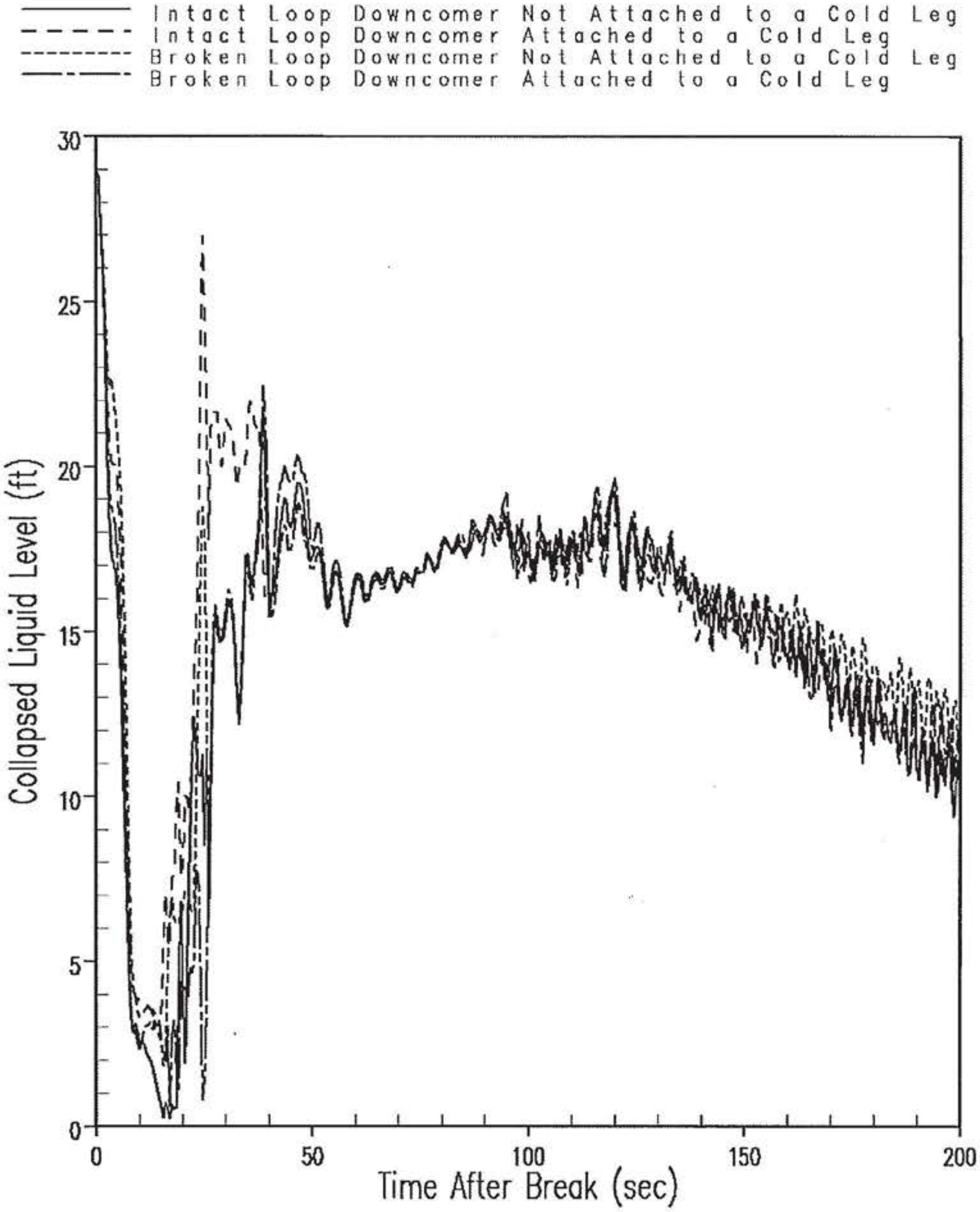
DWN	KJF	DATE	2–24–14	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED		CAD FILE	U14605.DGN		*FIGURE 14.6-5* *REV. 33*



PRAIRIE ISLAND UNIT 1 AND 2
LOW–HEAD SAFETY INJECTION FLOW FOR THE LIMITING PCT TRANSIENT (NO LOOP)

DWN KJF	DATE 2–24–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14606.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.6-6* *REV. 33*

01386642



PRAIRIE ISLAND UNIT 1 AND 2
DOWNCOMER COLLAPSED LIQUID LEVELS FOR THE LIMITING PCT TRANSIENT
(THE BOTTOM OF ACTIVE FUEL IS AT 5.8 FEET)

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14607.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.6-7*	*REV. 33*

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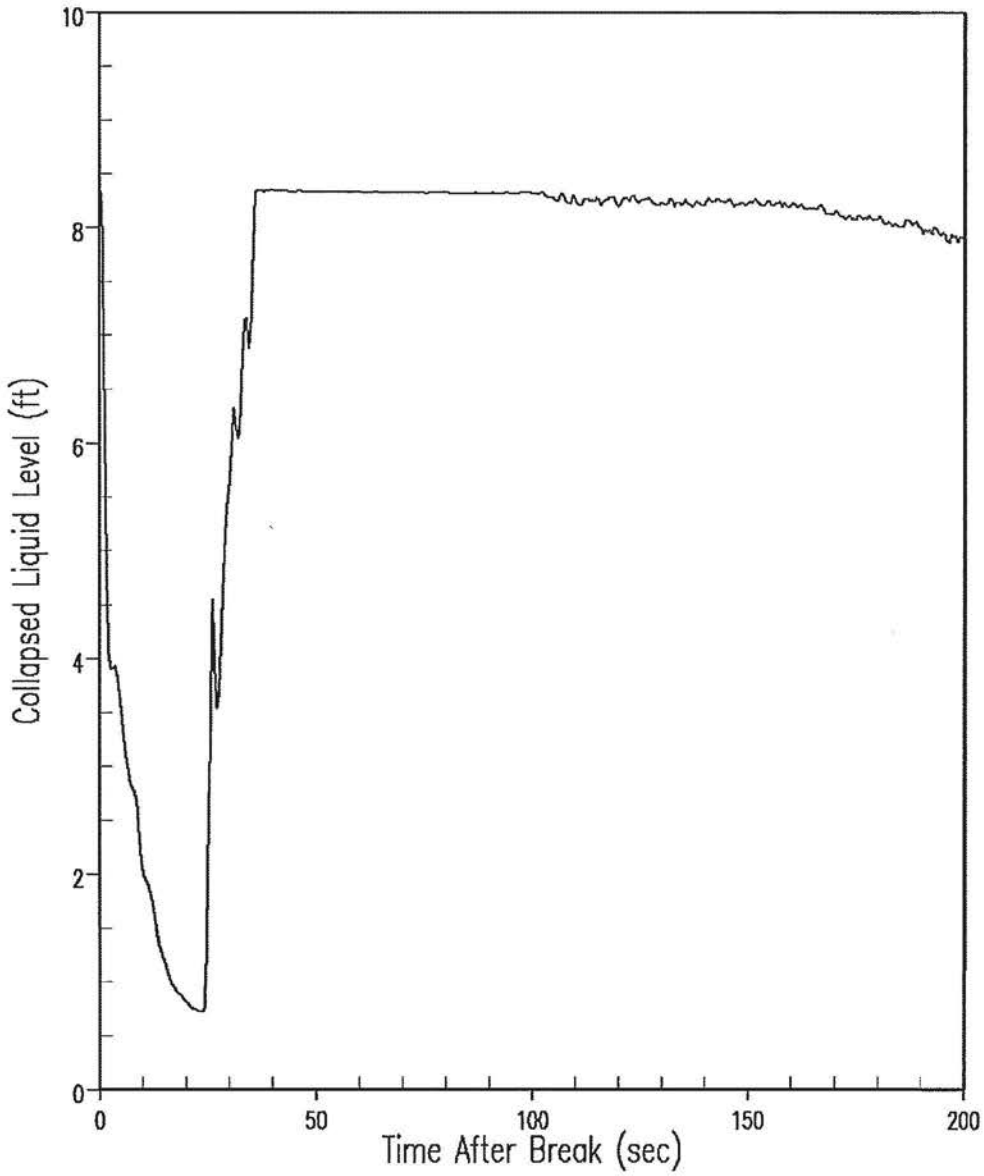
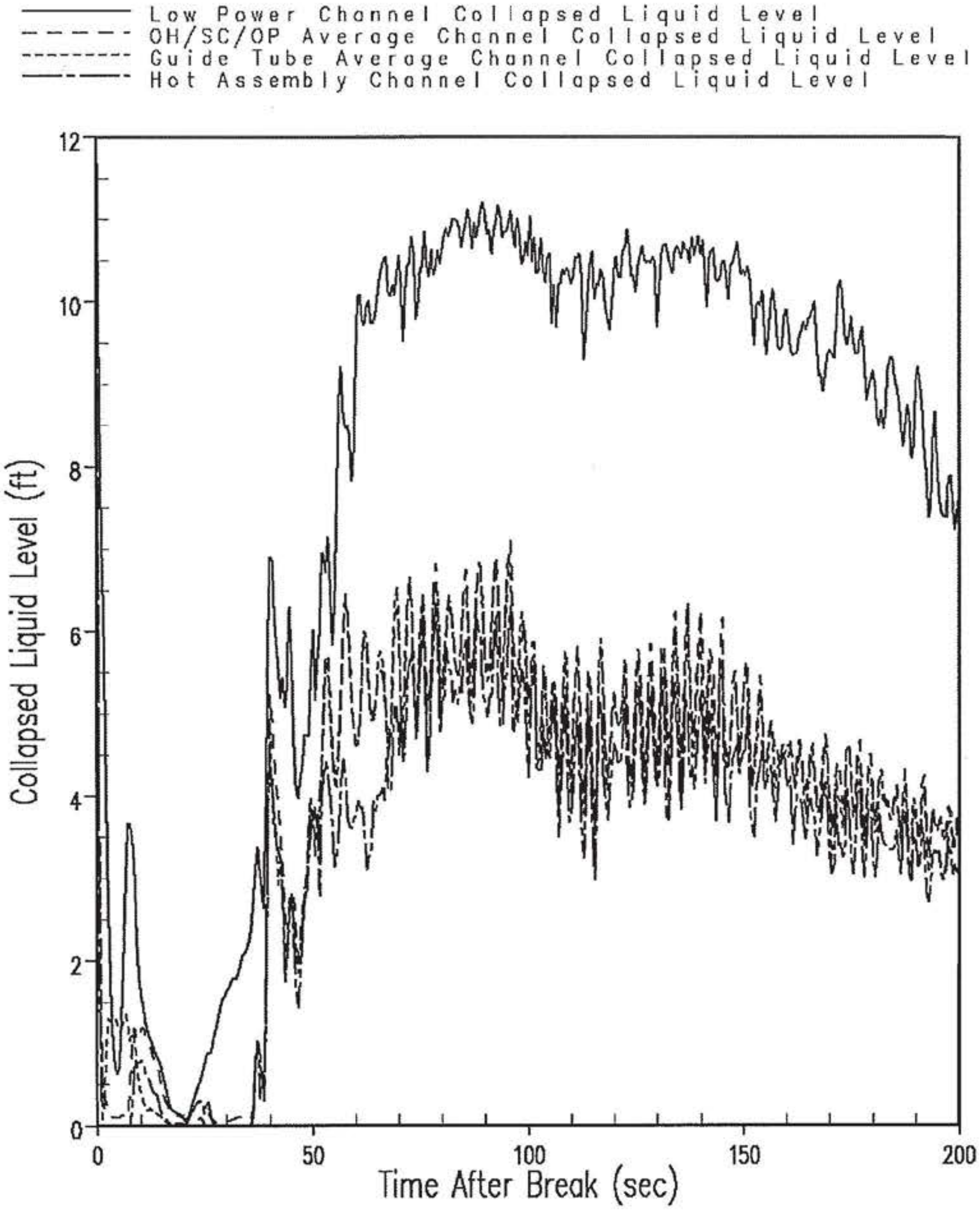


PRAIRIE ISLAND UNIT 1 & 2
LOWER PLENUM COLLAPSED LIQUID LEVEL FOR THE LIMITING PCT TRANSIENT
(THE BOTTOM OF ACTIVE FUEL IS AT 8.4 FEET)

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14608.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.6-8* *REV. 33*

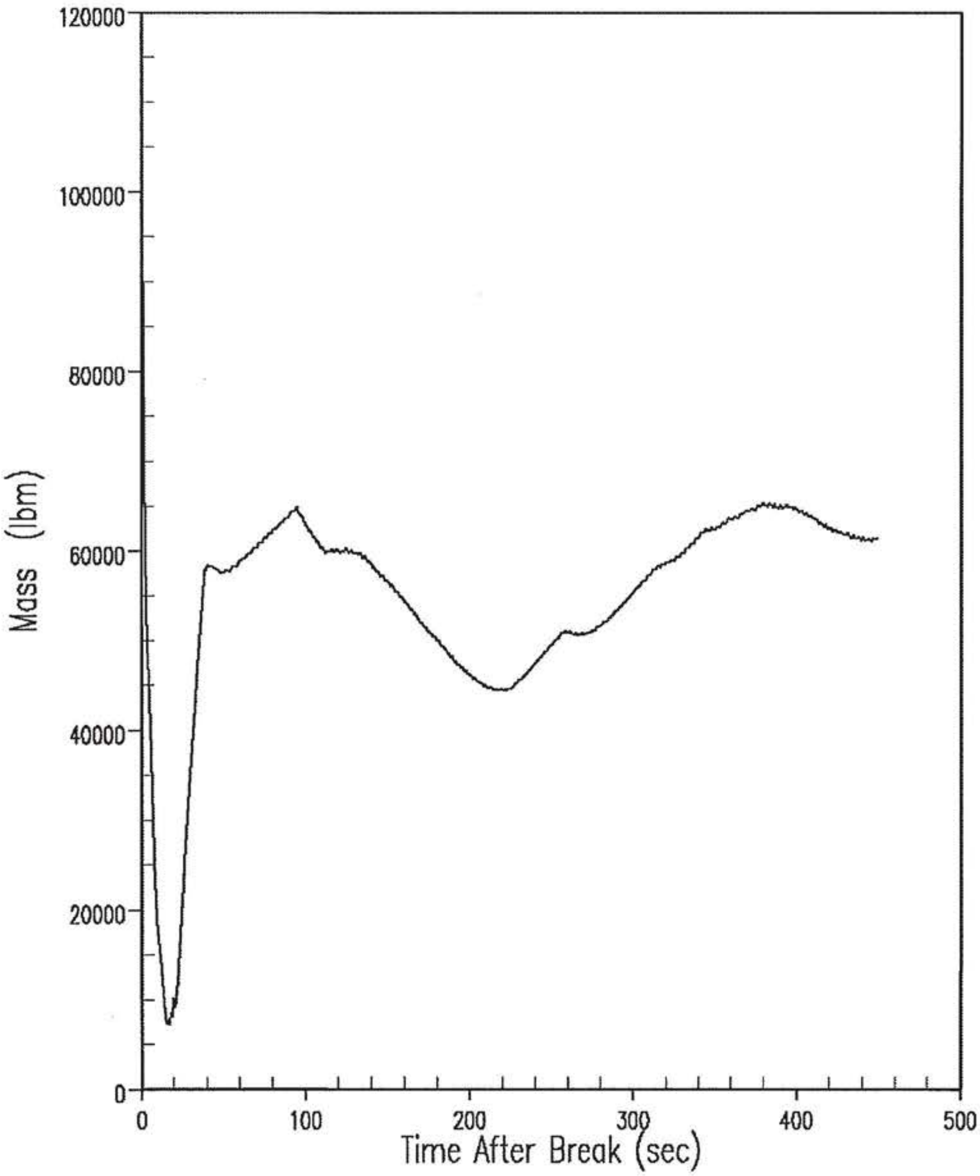
01386642



PRAIRIE ISLAND UNIT 1 & 2
CORE COLLAPSED LIQUID LEVELS FOR THE LIMITING PCT TRANSIENT
(THE BOTTOM OF ACTIVE FUEL IS AT 0.0 FEET)
(OH=OPEN HOLES, SC=SUPPORT COLUMN, OP=ORIFICE PLATE)

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14609.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.6-9*	*REV. 33*

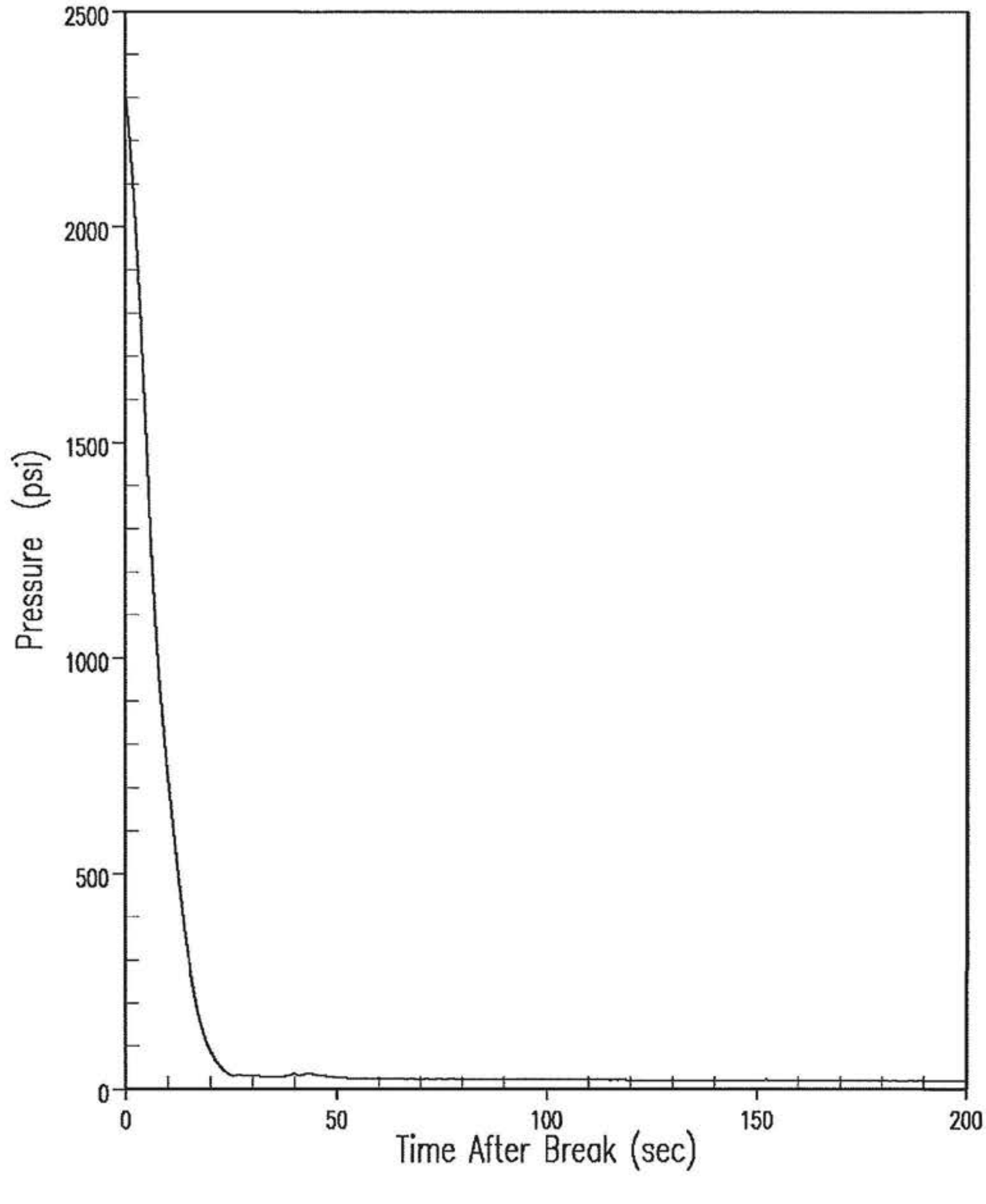
01386642



PRAIRIE ISLAND UNIT 1 AND 2 VESSEL LIQUID MASS FOR THE LIMITING PCT TRANSIENT

DWN KJF	DATE 2–24–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14610.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.6-10 REV. 33*

01386642



PRAIRIE ISLAND UNIT 1 AND 2 PRESSURIZER PRESSURE FOR THE LIMITING PCT TRANSIENT

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14611.DGN		*FIGURE 14.6-11 REV. 33*

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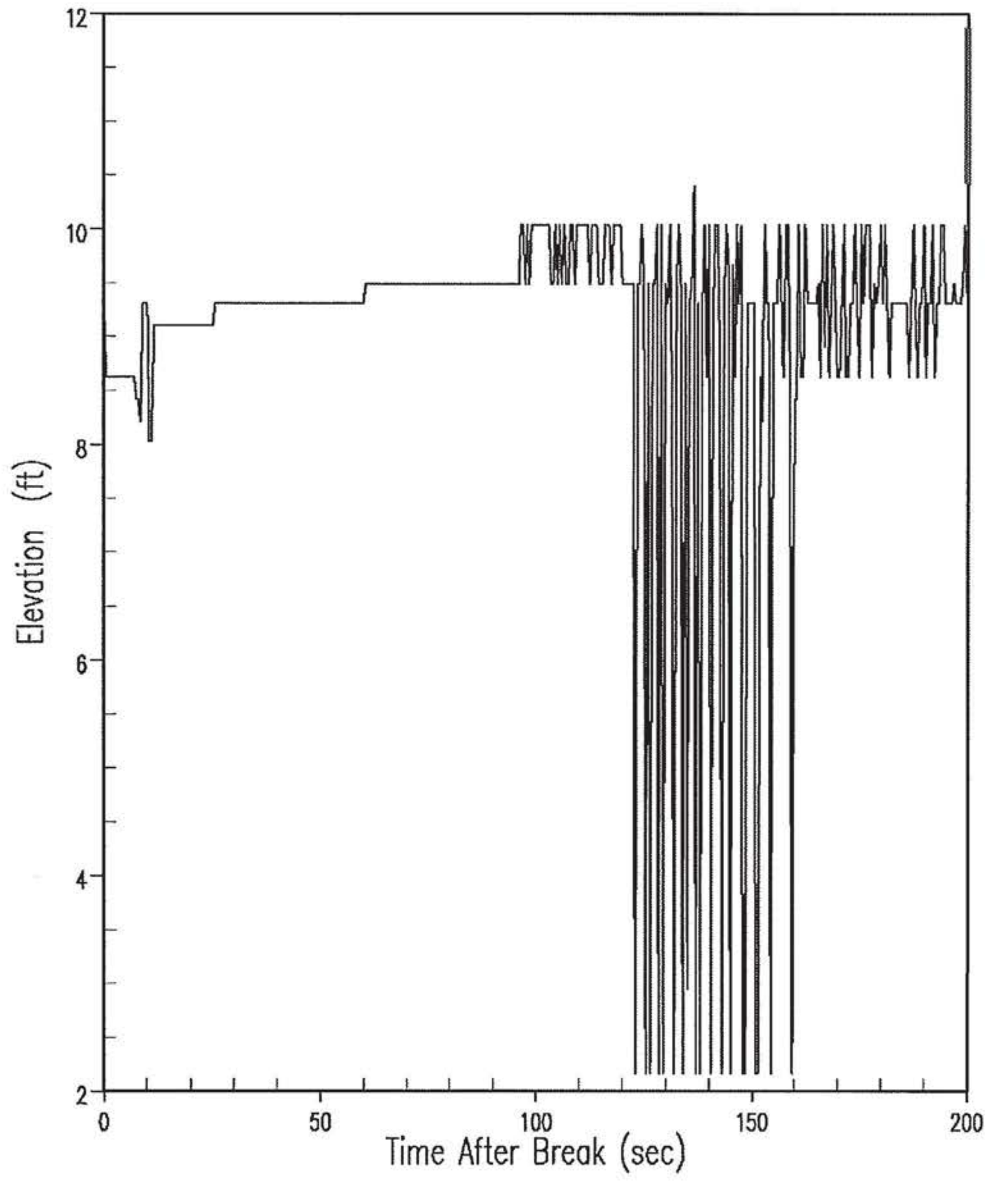
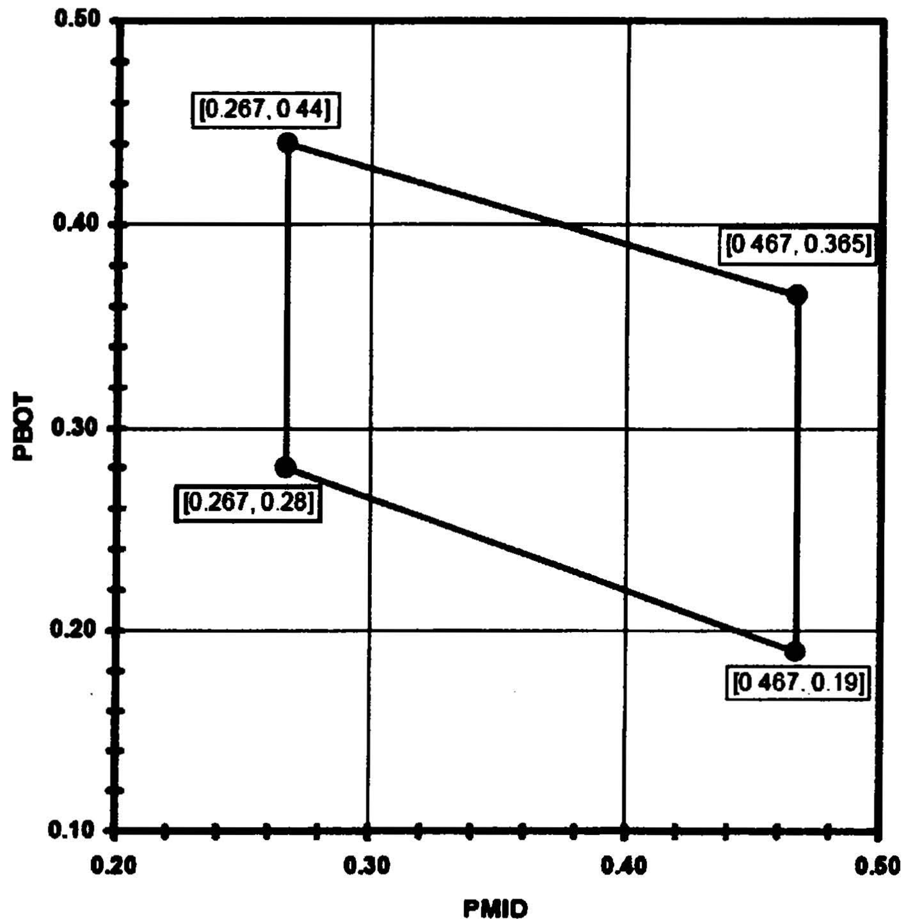


PRAIRIE ISLAND UNIT 1 AND 2
PEAK CLAD TEMPERATURE ELEVATION FOR THE LIMITING PCT TRANSIENT

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14612.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	FIGURE 14.6-12	REV. 33

01386642

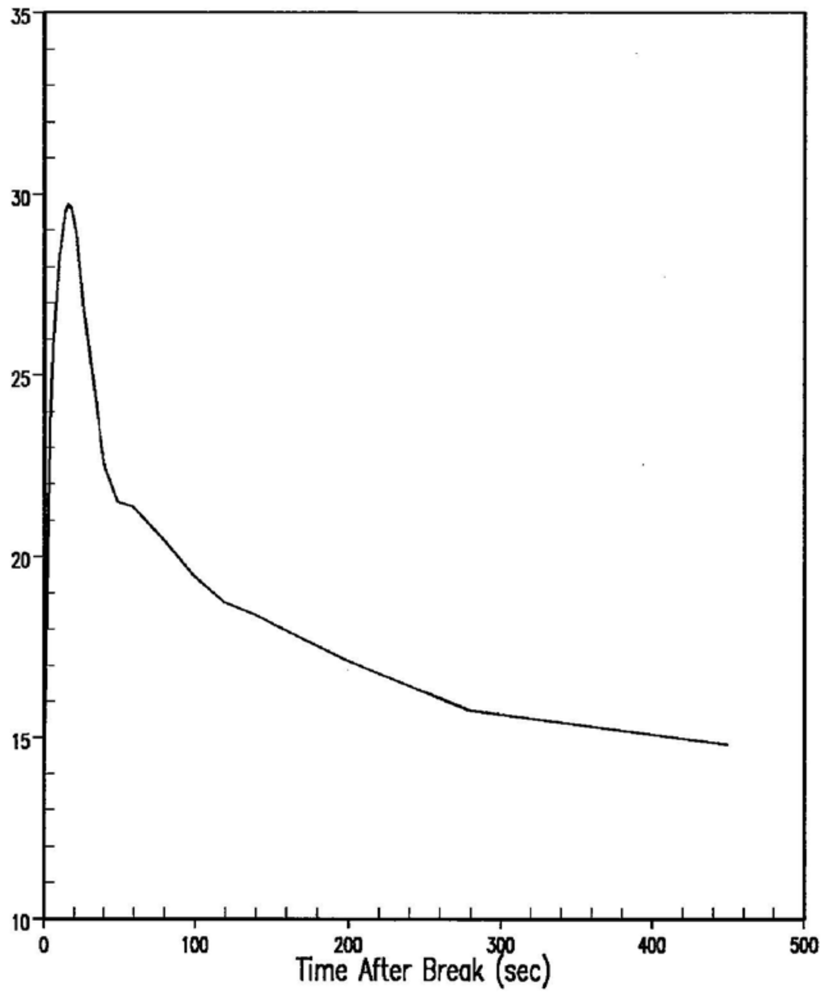


PRAIRIE ISLAND UNITS 1 AND 2 P_{BOT}/P_{MID} ANALYSIS RANGE
PBOT=INTEGRATED POWER FRACTION IN THE BOTTOM THIRD OF THE CORE
PMID=INTEGRATED POWER FRACTION IN THE MIDDLE THIRD OF THE CORE

DWN KJF	DATE 1-29-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14613.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.6-13 REV. 31*

01193868

FIGURE 14.6-13

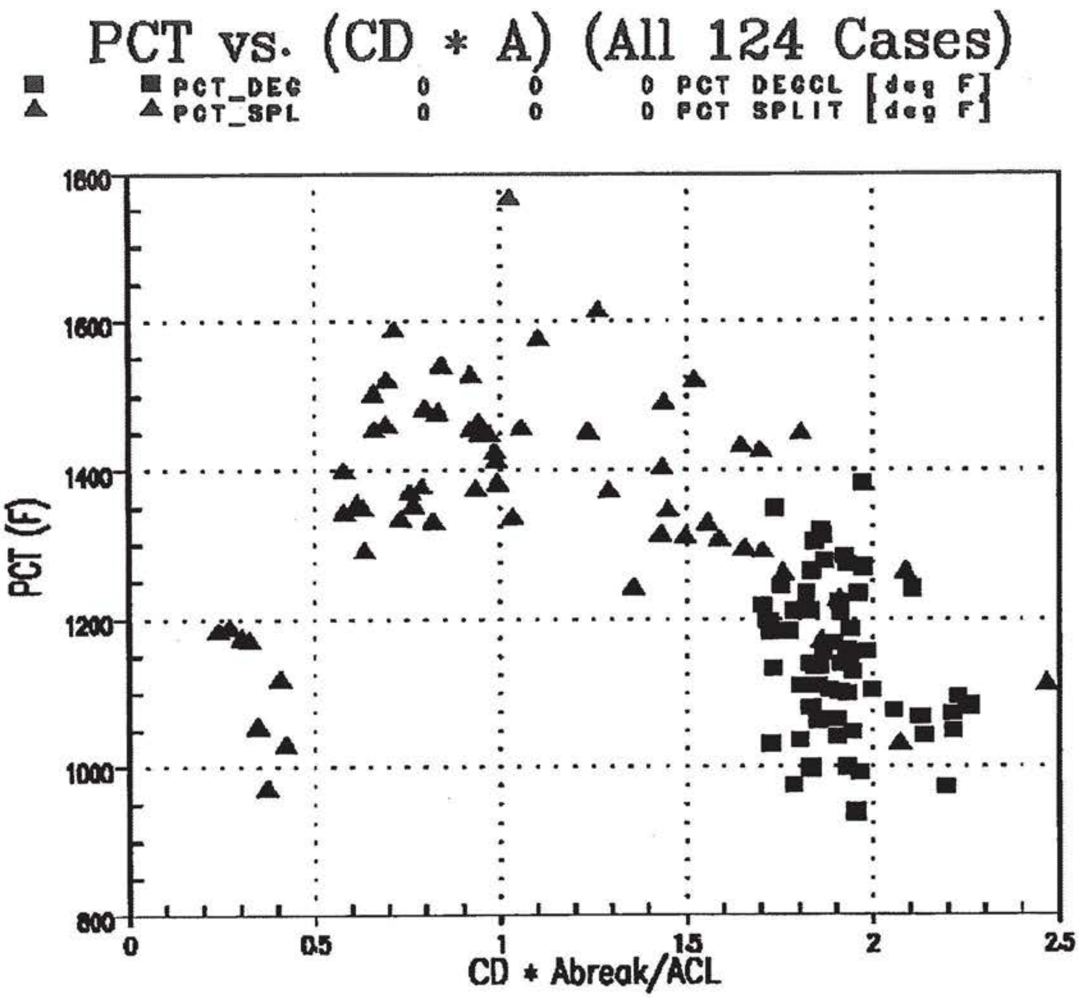


PRAIRIE ISLAND UNITS 1 AND 2 LOWER BOUND CONTAINMENT PRESSURE

DWN KJF	DATE 1-29-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14614.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.6-14 REV. 31*

01193868

FIGURE 14.6-14



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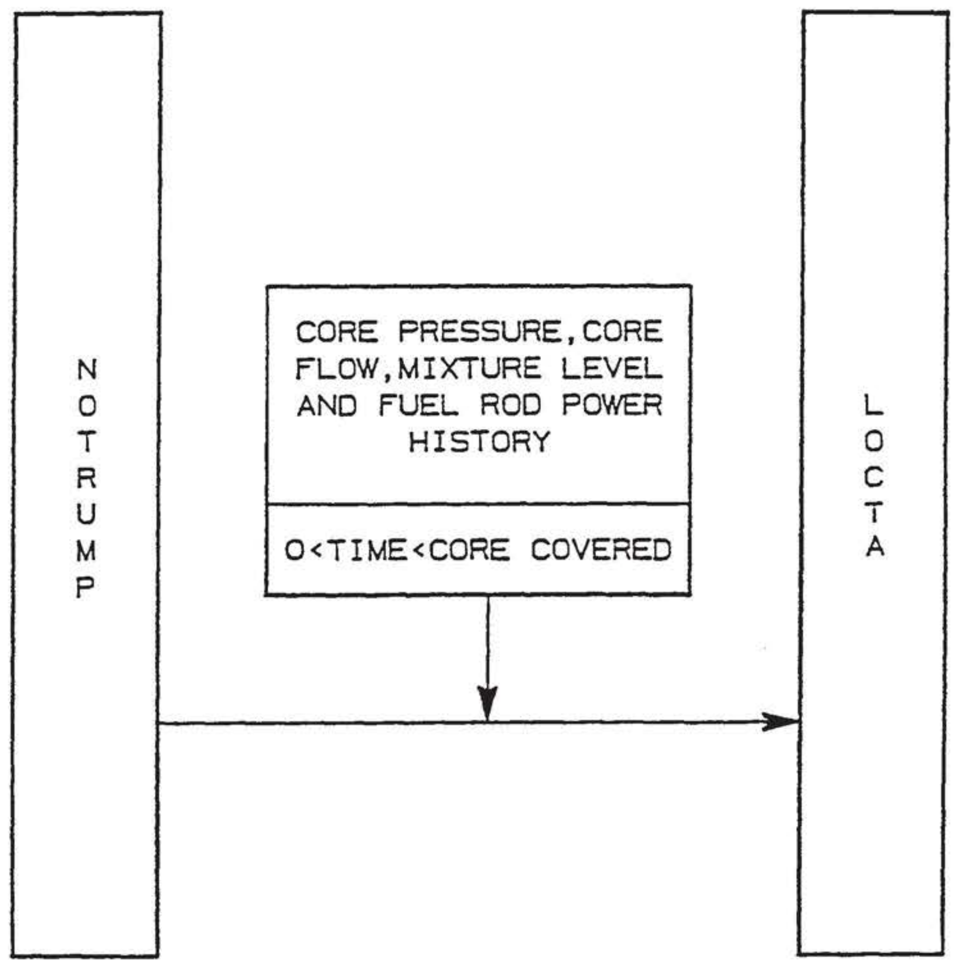
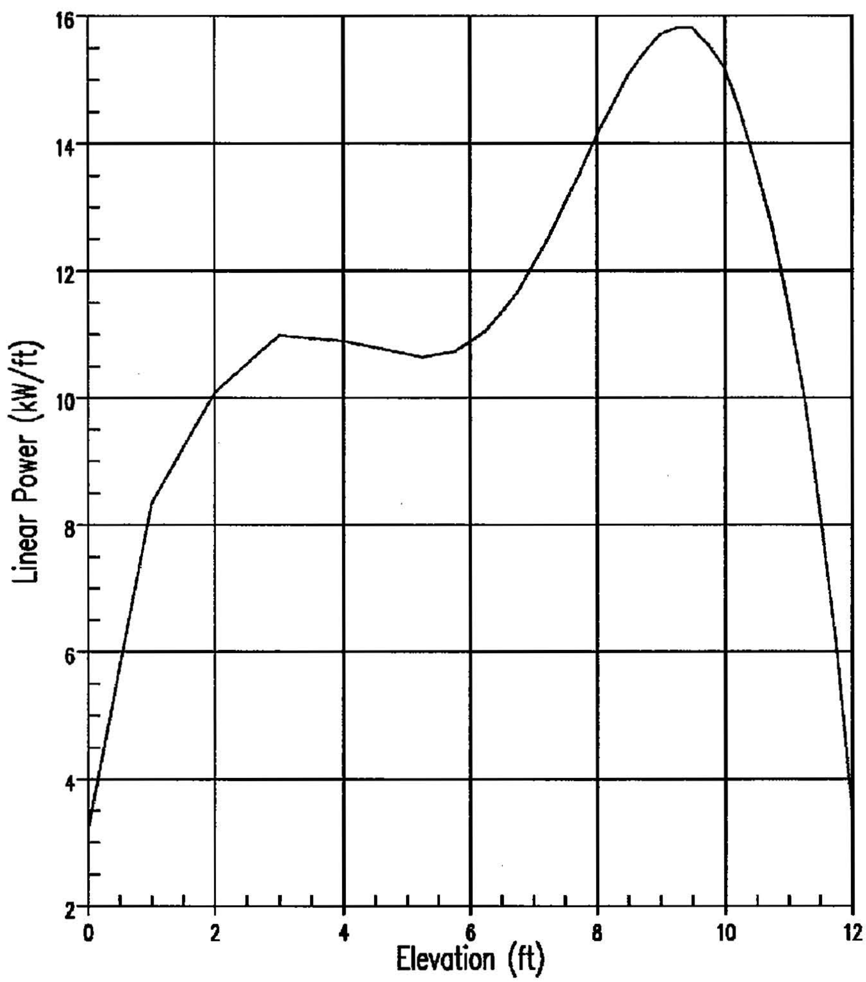
PRAIRIE ISLAND UNIT 1 AND 2 PCT VERSUS EFFECTIVE BREAK AREA SCATTER PLOT
(CD = DISCHARGE COEFFICIENT, ABREAK = BREAK AREA, ACL = COLD LEG AREA)

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14615.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.6-15*	*REV. 33*



CODE INTERFACE DESCRIPTION FOR SMALL BREAK MODEL

DWN G.L.D	DATE 6-23-99	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14701.DGN		**FIGURE 14.7-1 REV. 18**



HOT ROD AXIAL POWER SHAPE

DWN KJF	DATE 1-29-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14702.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-2 REV. 31*

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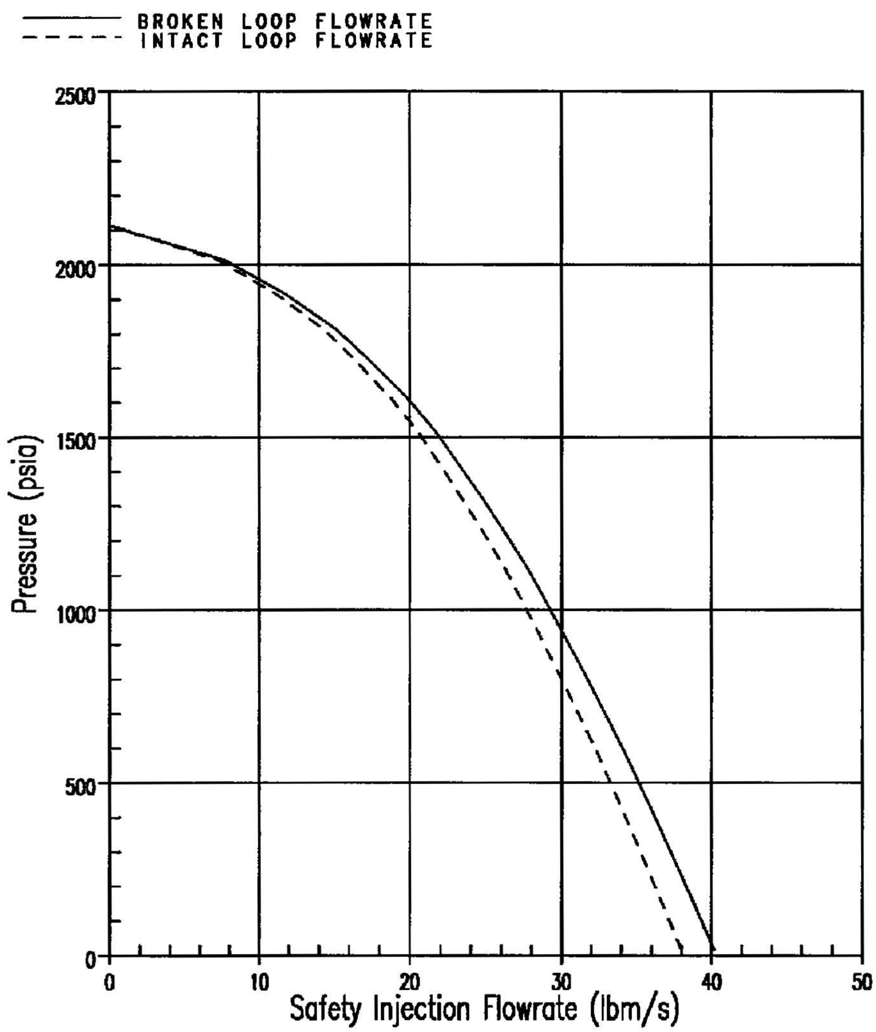
FIGURE 14.7-2



PUMPED HIGH HEAD SAFETY INJECTION FLOW RATE
FAULTED LOOP SPILLING TO RCS PRESSURE

DWN KJF	DATE 1-29-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14703A.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-3A REV. 31*

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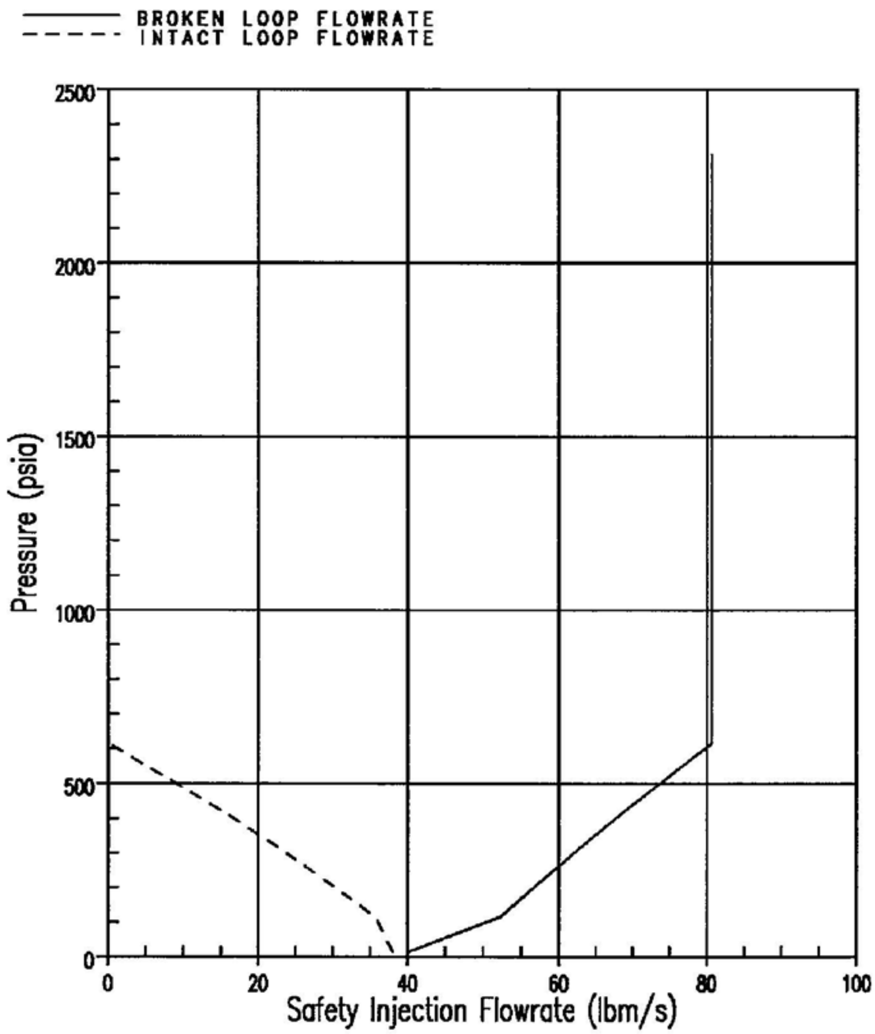
FIGURE 14.7-3A



PUMPED HIGH HEAD SAFETY INJECTION FLOW RATE
FAULTED LOOP SPILLING TO CONTAINMENT PRESSURE (0 psig)

DWN KJF	DATE 1-29-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE NONE
CHECKED	CAD FILE U14703B.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-3B REV. 31*

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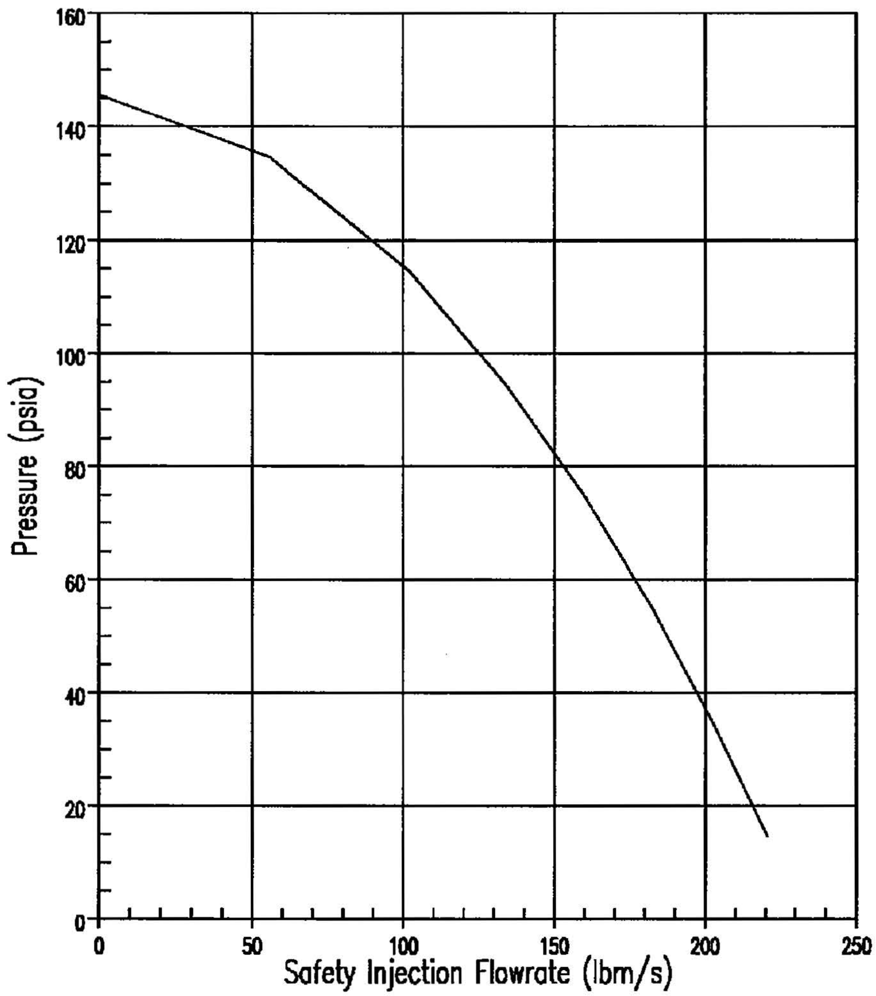
FIGURE 14.7-3B



PUMPED RESIDUAL HEAT REMOVAL INJECTION FLOW RATE
ONE PUMP, NO SPILLING FLOWS

DWN KJF	DATE 1-29-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14703C.DGN		FIGURE 14.7-3C REV. 31

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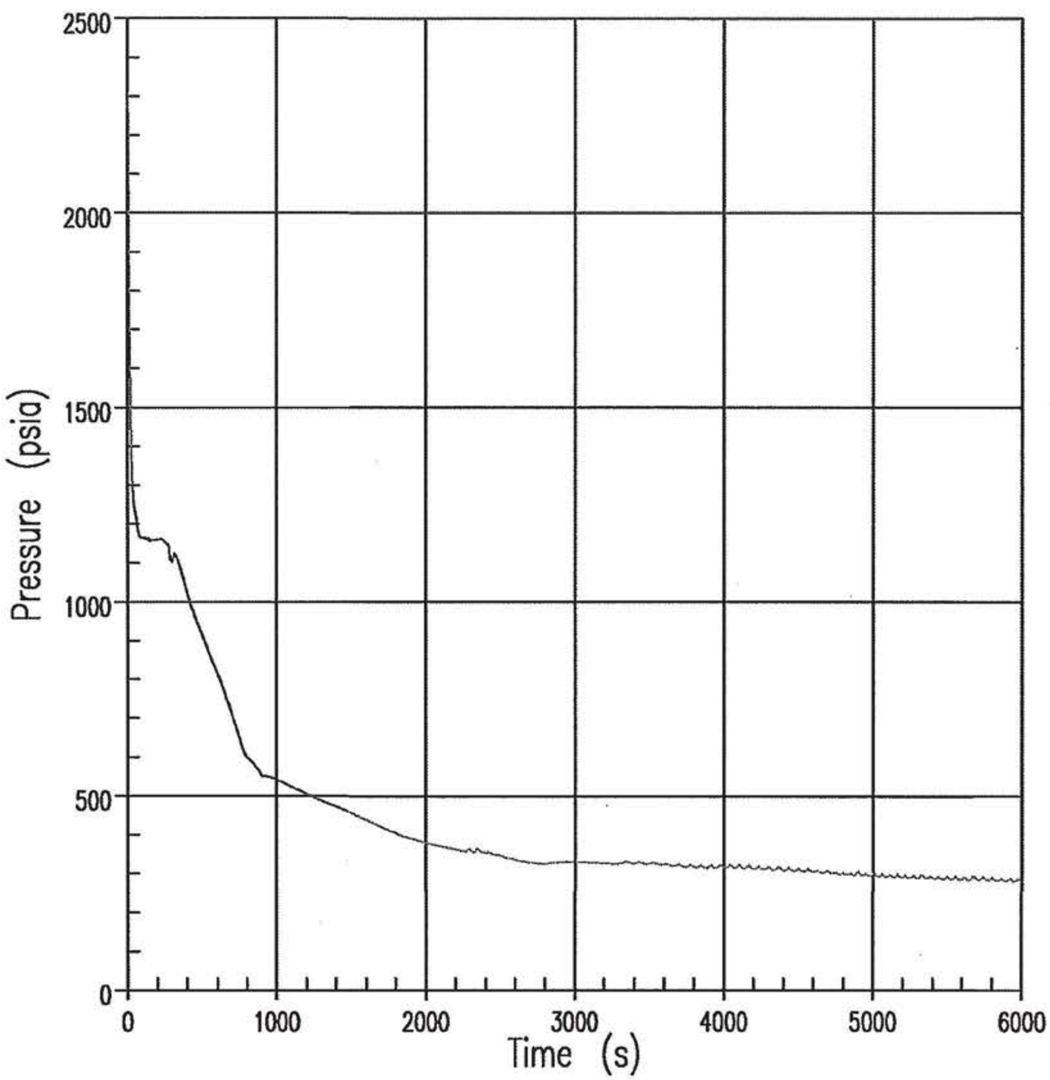
FIGURE 14.7-3C



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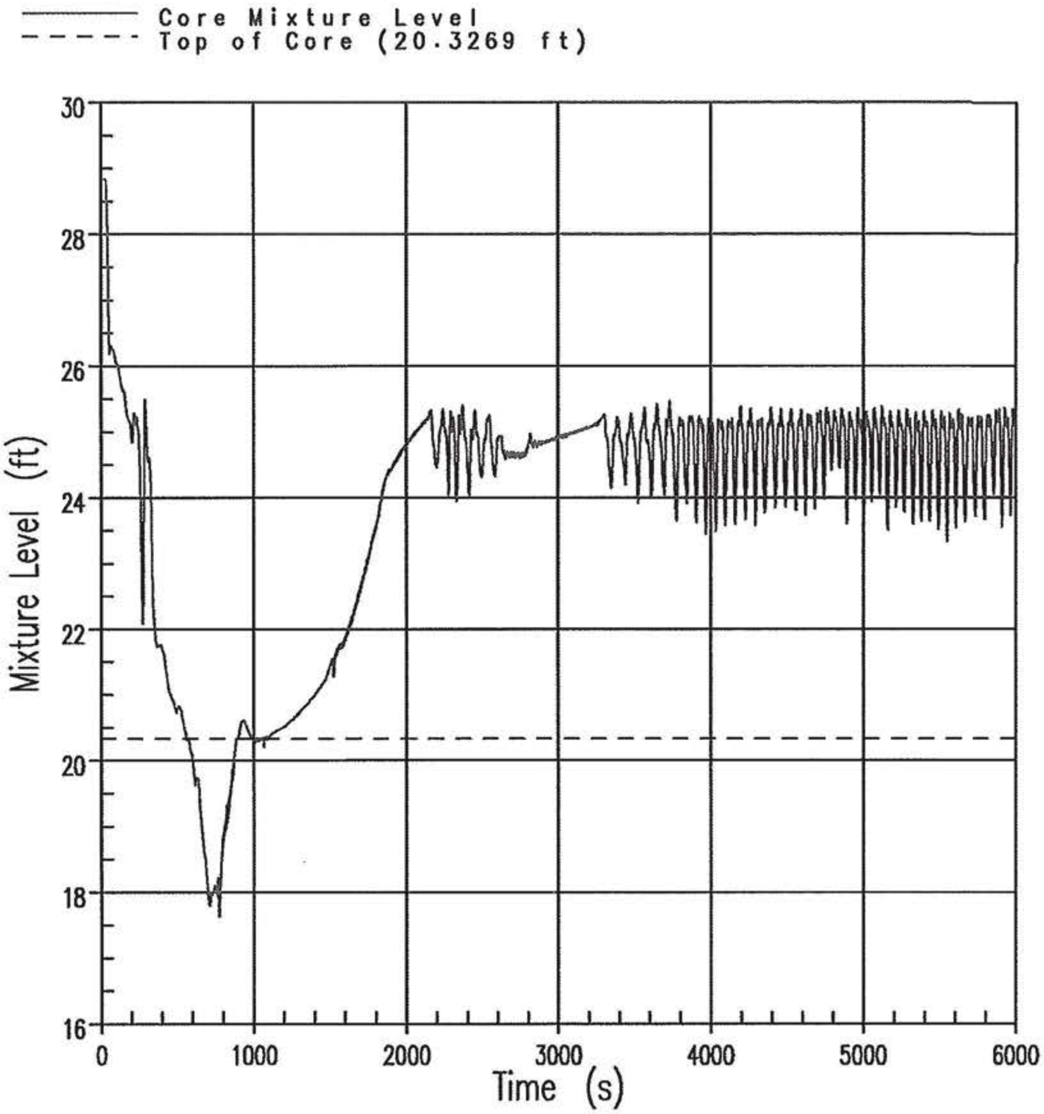
REACTOR COOLANT SYSTEM PRESSURE
3-INCH BREAK

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14704.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-4 REV. 33*



CORE MIXTURE LEVEL
3-INCH BREAK

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14705.DGN		*FIGURE 14.7-5 REV. 33*

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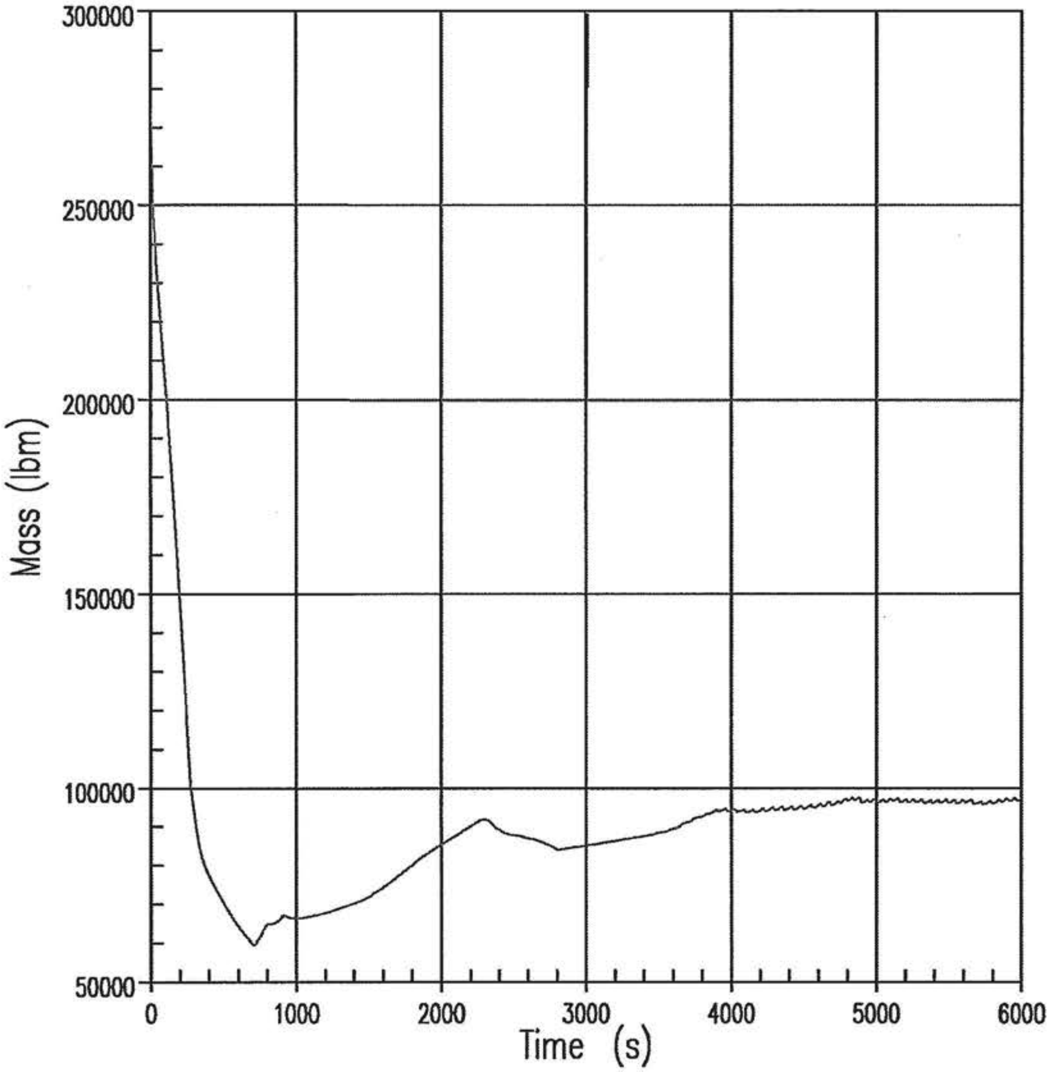

TOTAL REACTOR COOLANT SYSTEM MASS
3-INCH BREAK

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14706.DGN		*FIGURE 14.7-6 REV. 33*

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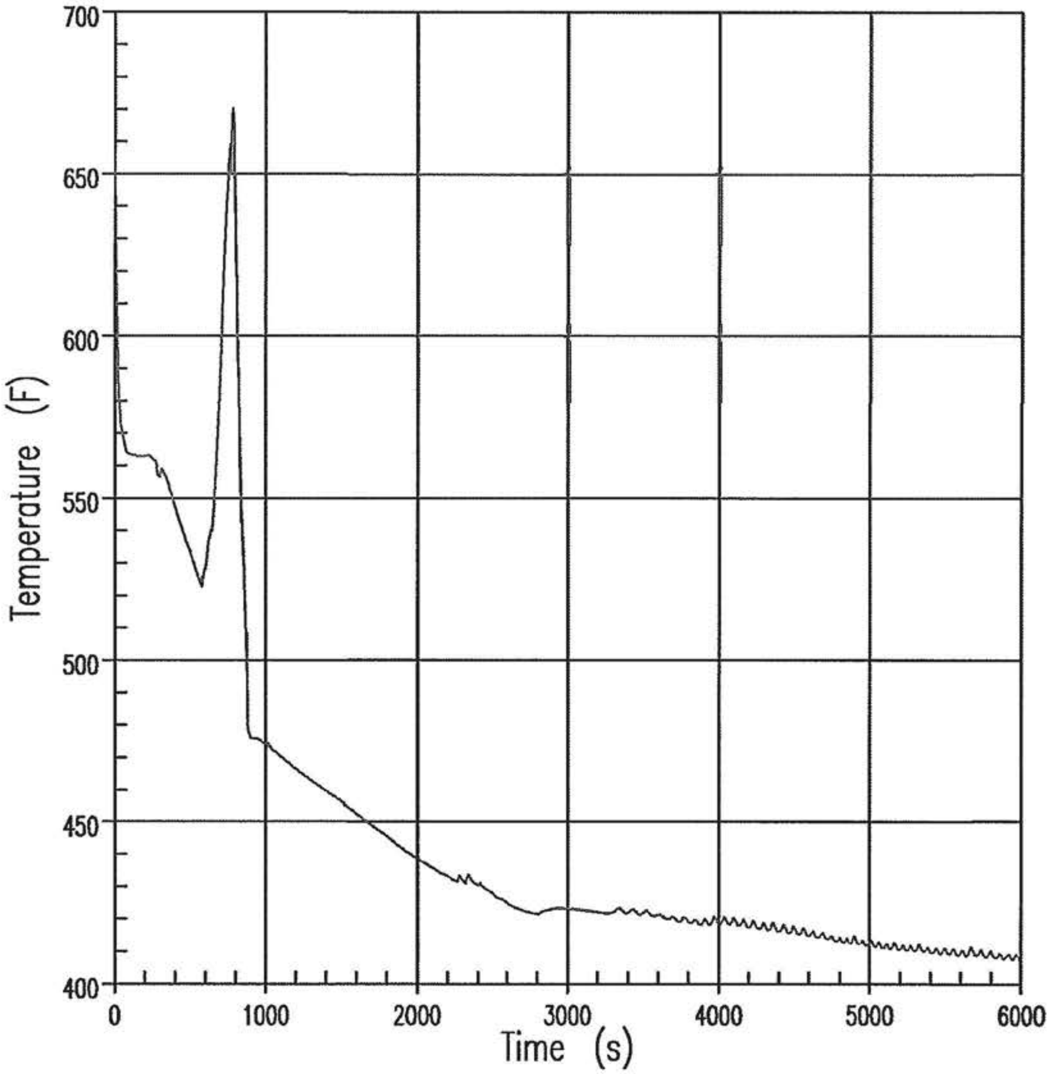

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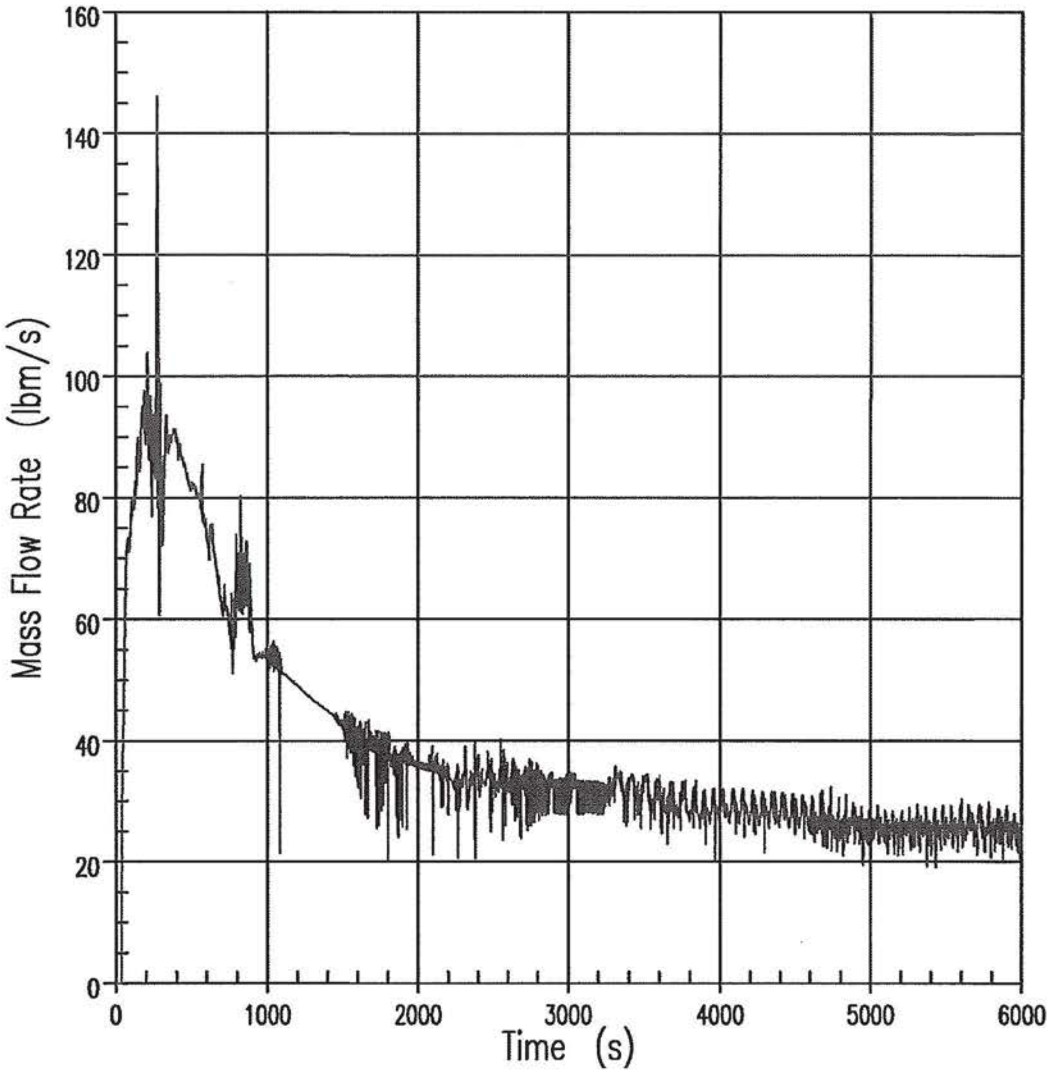
TOP CORE EXIT VAPOR TEMPERATURE
3–INCH BREAK

DWN KJF	DATE 2–24–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14707.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-7*	*REV. 33*



VAPOR MASS FLOW RATE OUT TOP OF CORE
3-INCH BREAK

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14708.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-8 REV. 33*

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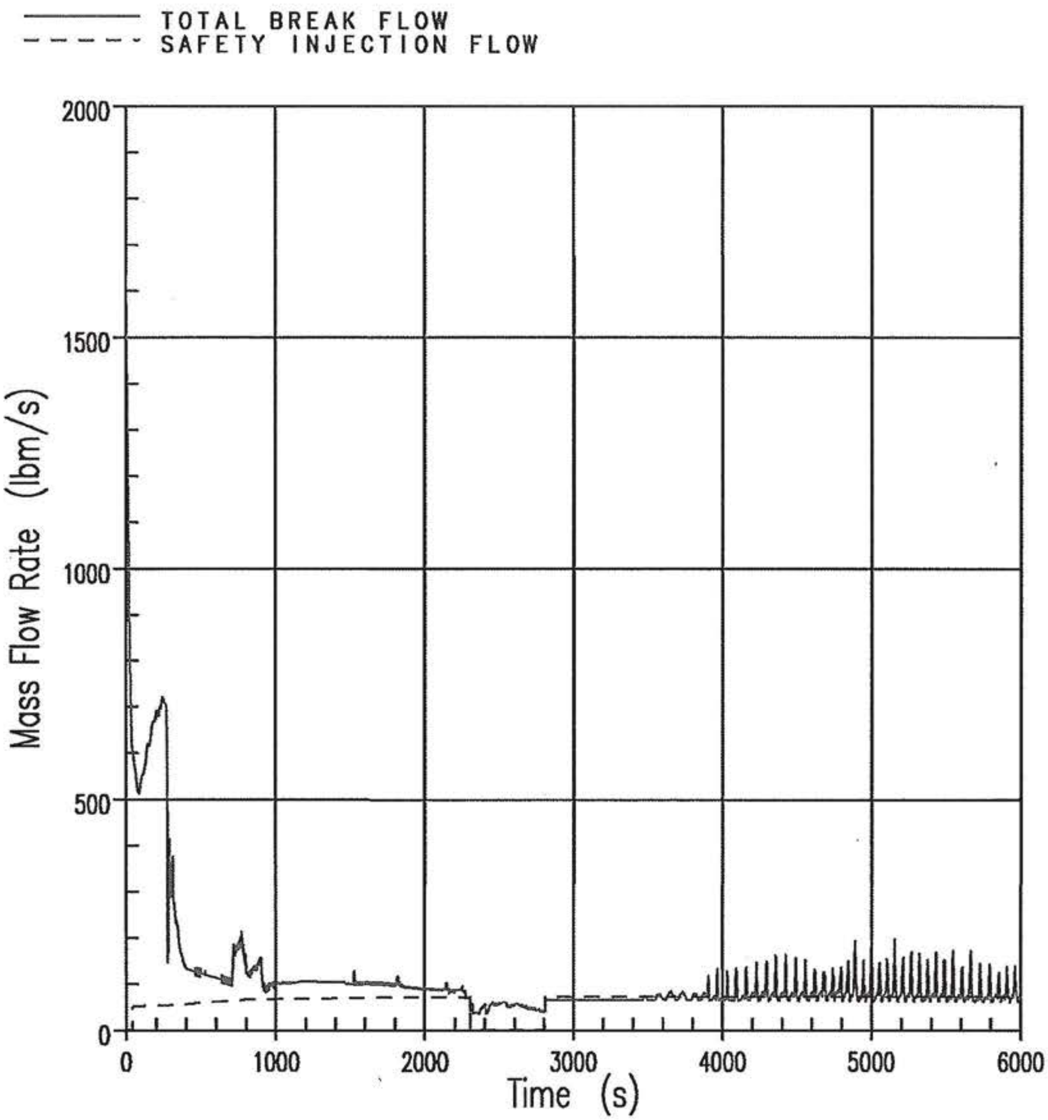

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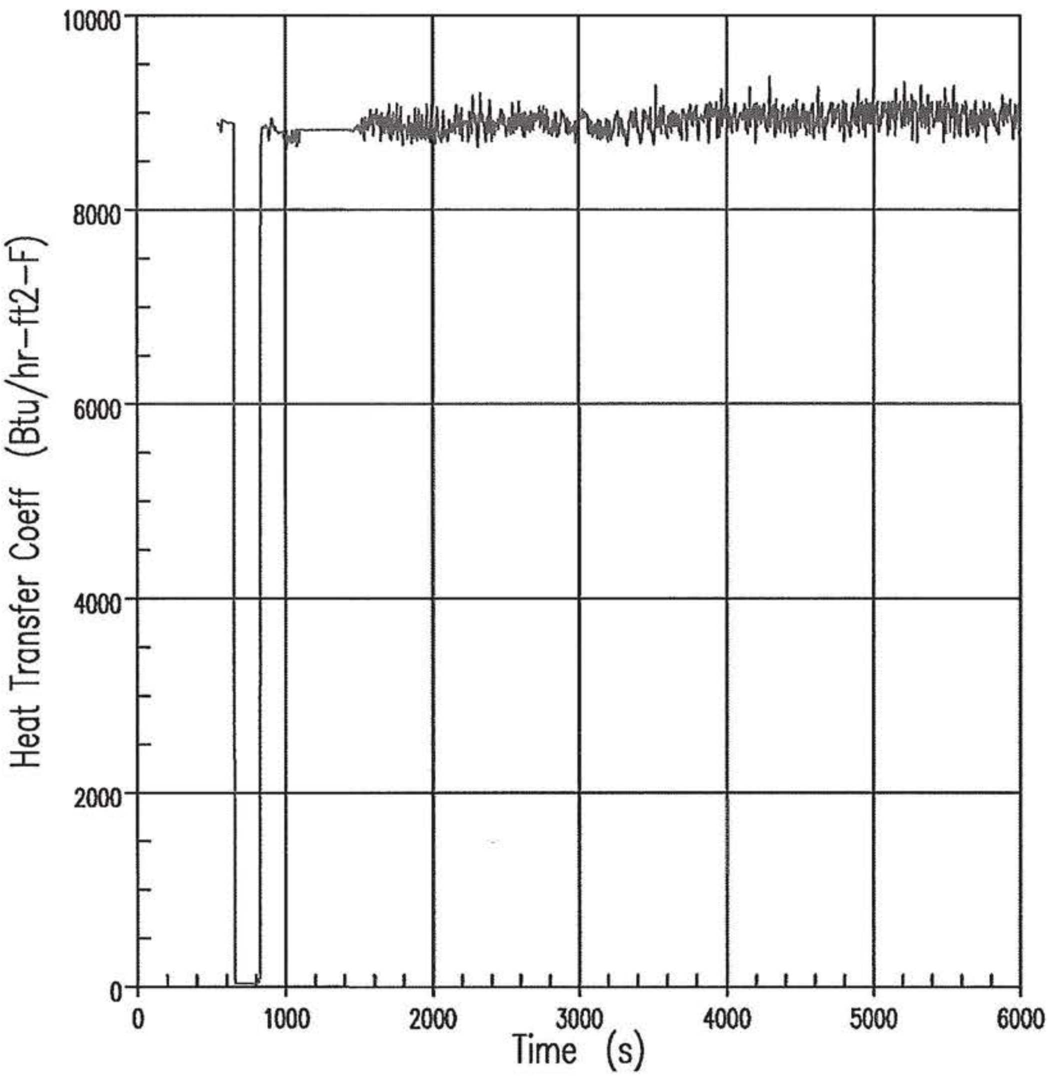
TOTAL BREAK FLOW AND SAFETY INJECTION FLOW
3-INCH BREAK

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14709.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	FIGURE 14.7-9 REV. 33



CLADDING SURFACE HEAT TRANSFER COEFFICIENT AT PCT ELEVATION
3-INCH BREAK

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14710.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-10 REV. 33*

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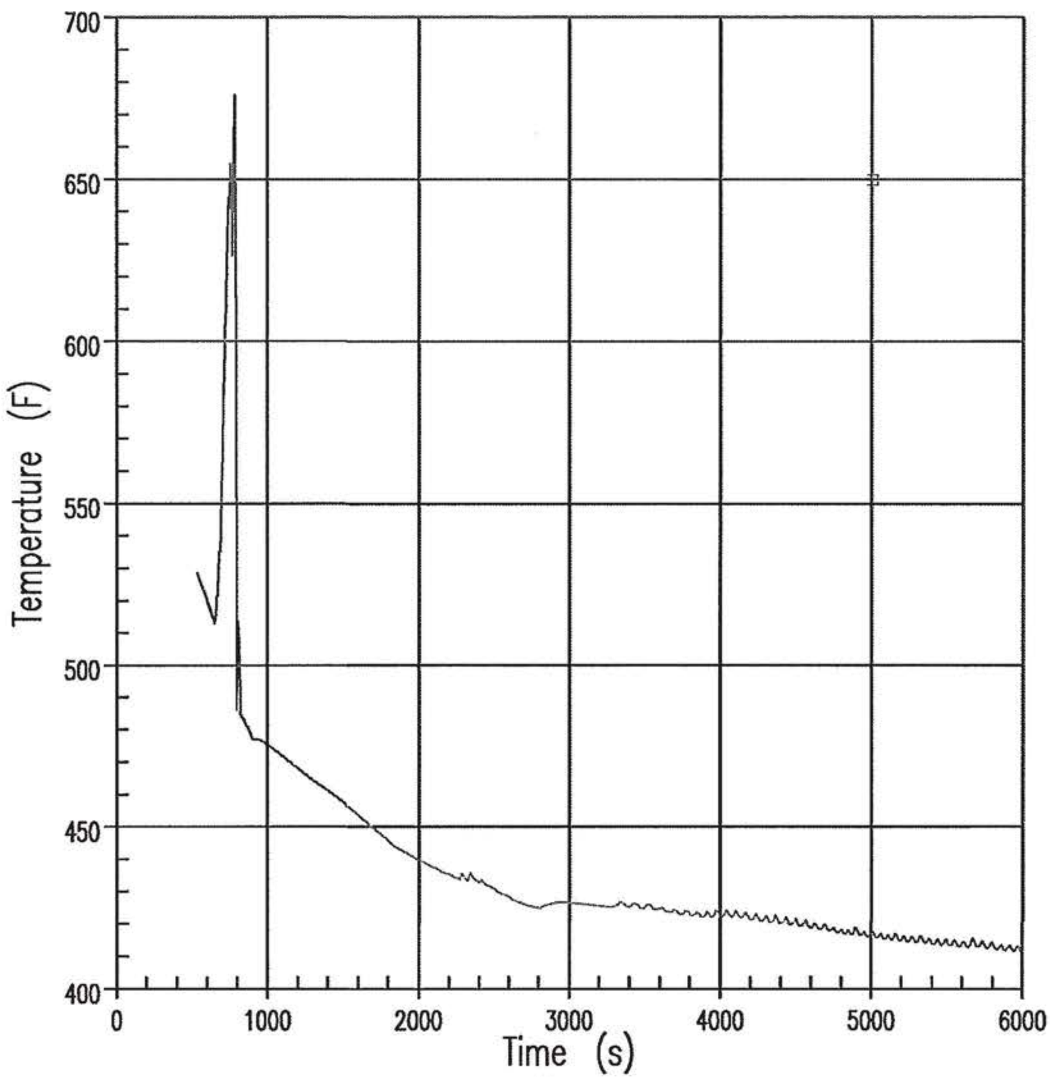

FLUID TEMPERATURE AT PCT ELEVATION
3-INCH BREAK

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14711.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-11 REV. 33*

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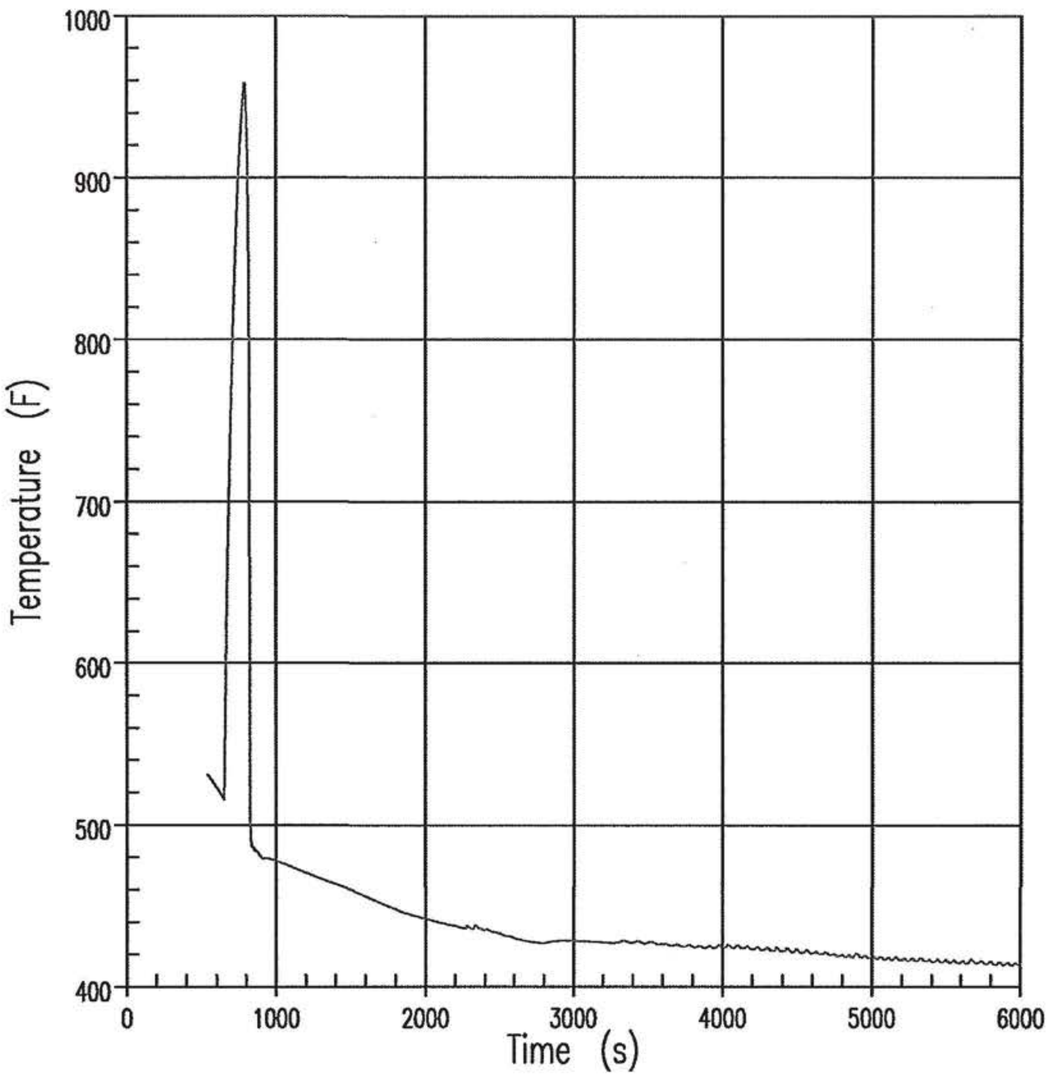

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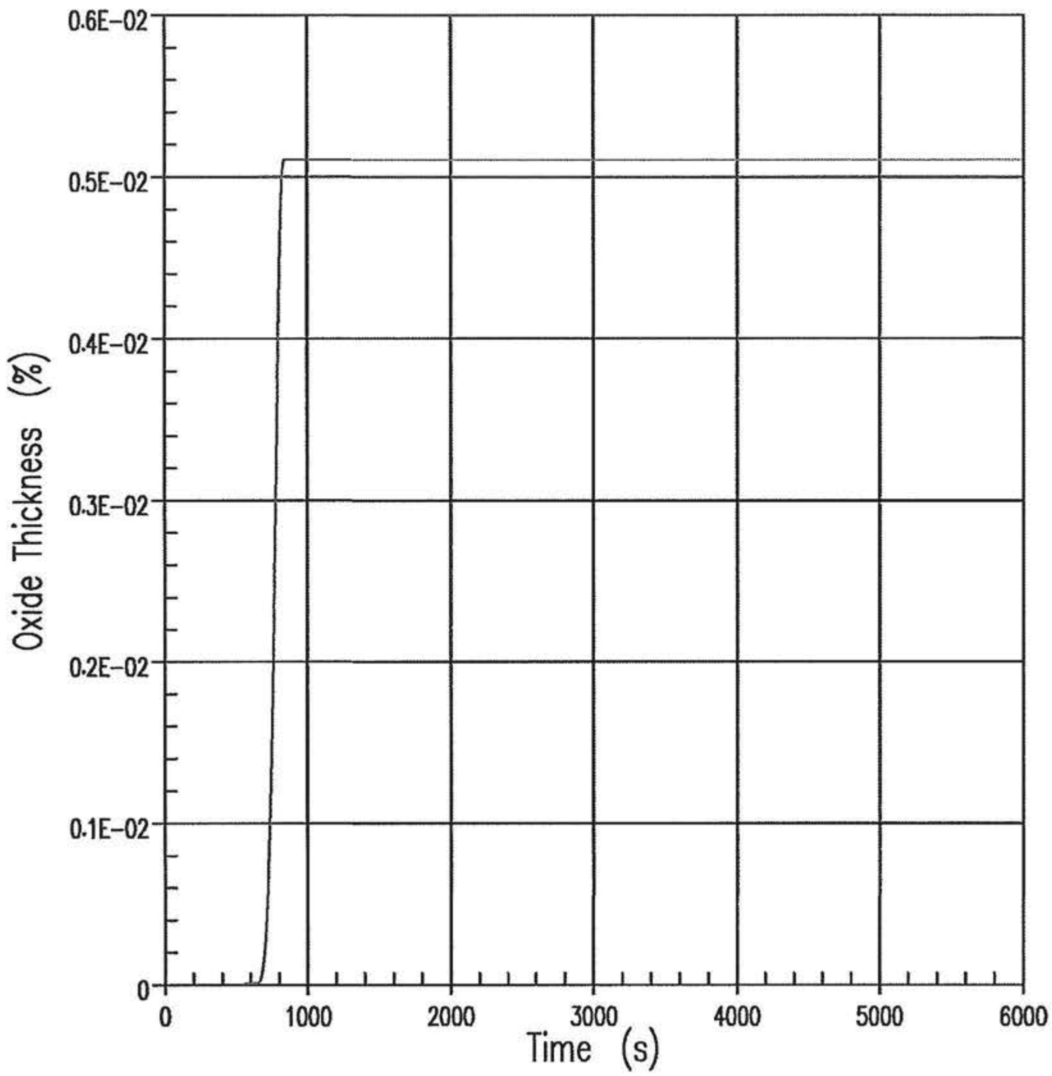
CLADDING TEMPERATURE AT PCT ELEVATION
3–INCH BREAK

DWN KJF	DATE 2–24–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14712.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-12 REV. 33*



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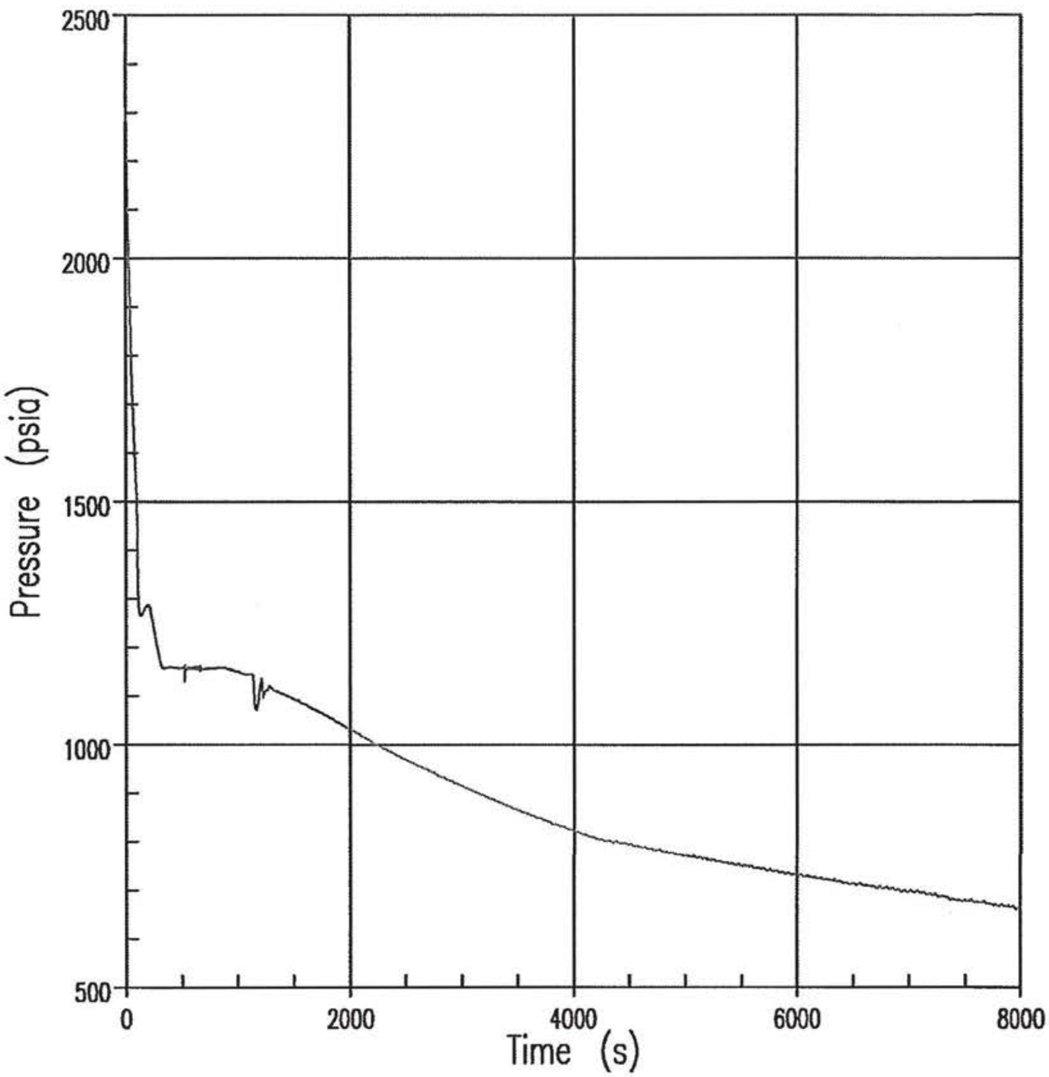
LOCAL ZrO_2 THICKNESS AT MAXIMUM LOCAL ZrO_2 ELEVATION
3–INCH BREAK

DWN KJF	DATE 2–24–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14713.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-13 REV. 33*



REACTOR COOLANT SYSTEM PRESSURE
1.5–INCH BREAK

DWN KJF	DATE 2–24–14	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14714.DGN		*FIGURE 14.7-14 REV. 33*

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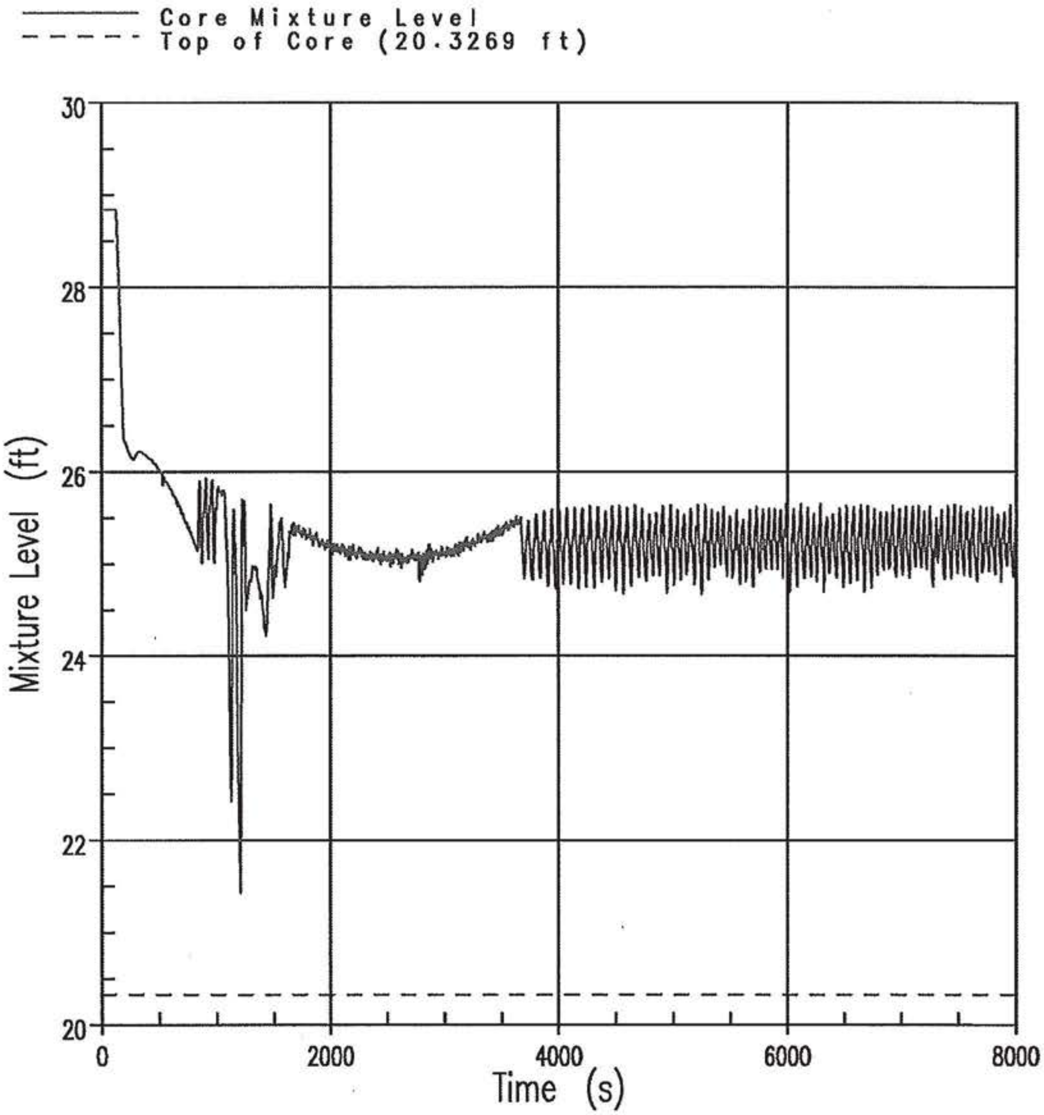

CORE MIXTURE LEVEL
1.5–INCH BREAK

DWN KJF	DATE 2–24–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14715.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-15 REV. 33*

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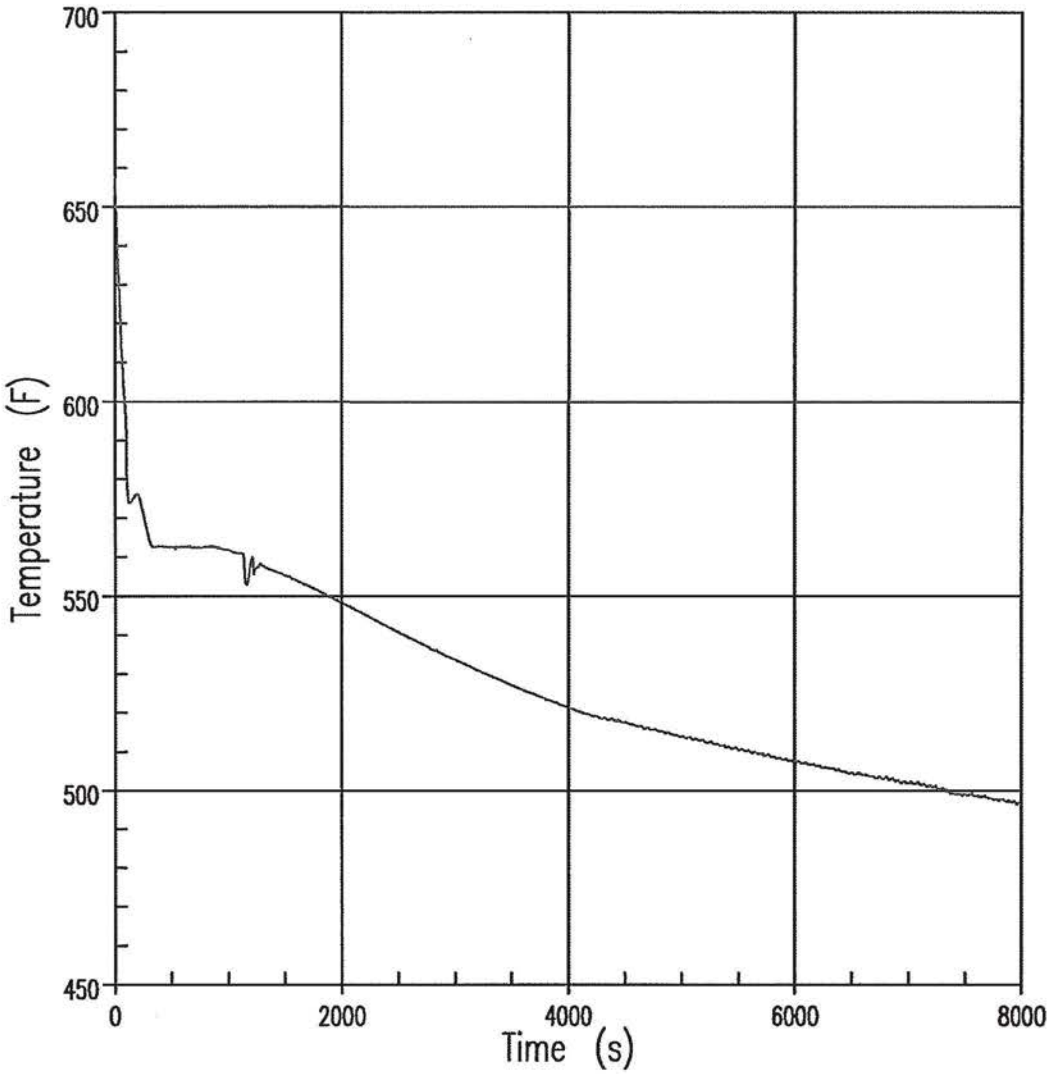

TOP CORE EXIT VAPOR TEMPERATURE
1.5–INCH BREAK

DWN KJF	DATE 2–24–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14716.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-16*	*REV. 33*

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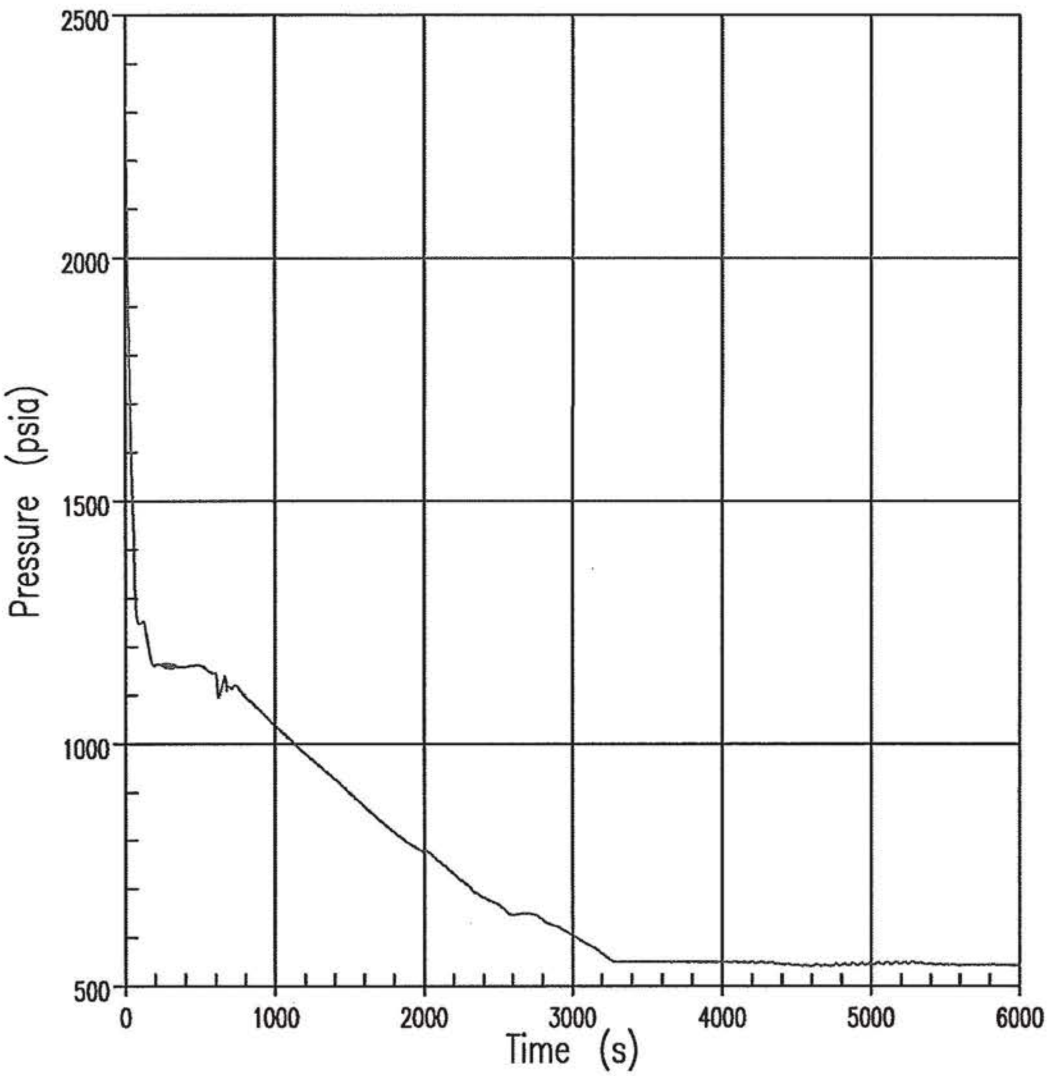

REACTOR COOLANT SYSTEM PRESSURE
2-INCH BREAK

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14717.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-17*	*REV. 33*

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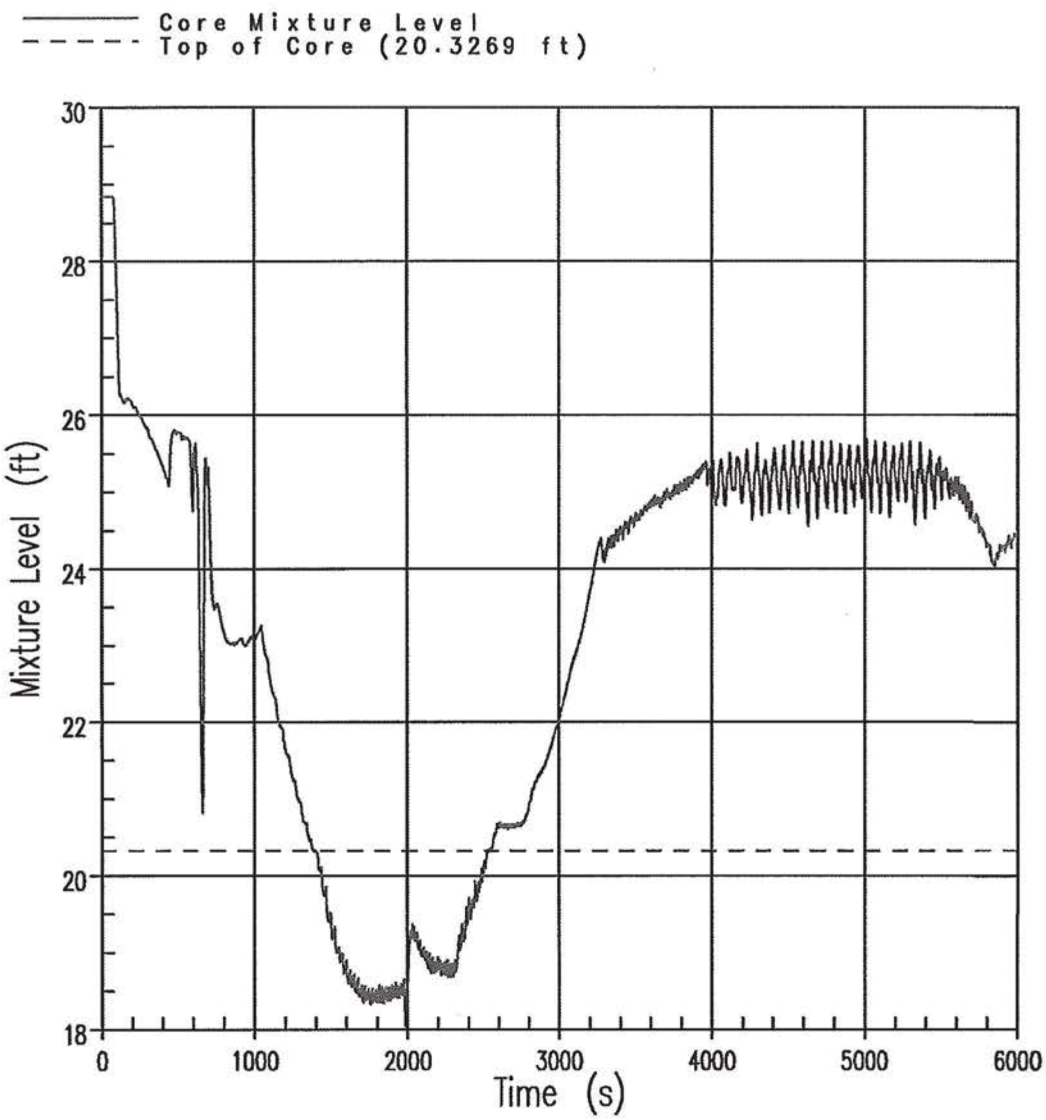

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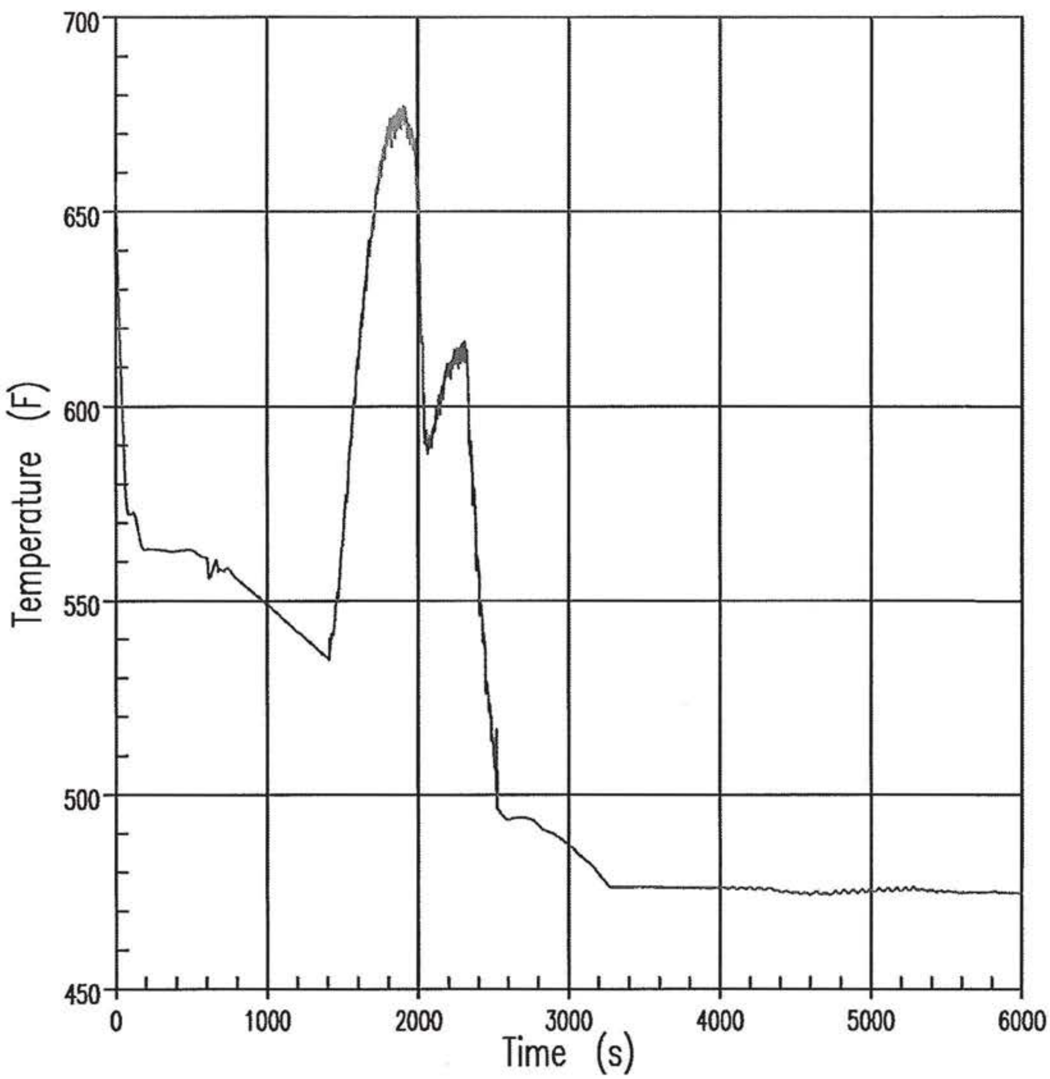
CORE MIXTURE LEVEL
2-INCH BREAK

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14718.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-18*	*REV. 33*



TOP CORE EXIT VAPOR TEMPERATURE
2-INCH BREAK

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14719.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-19*	*REV. 33*

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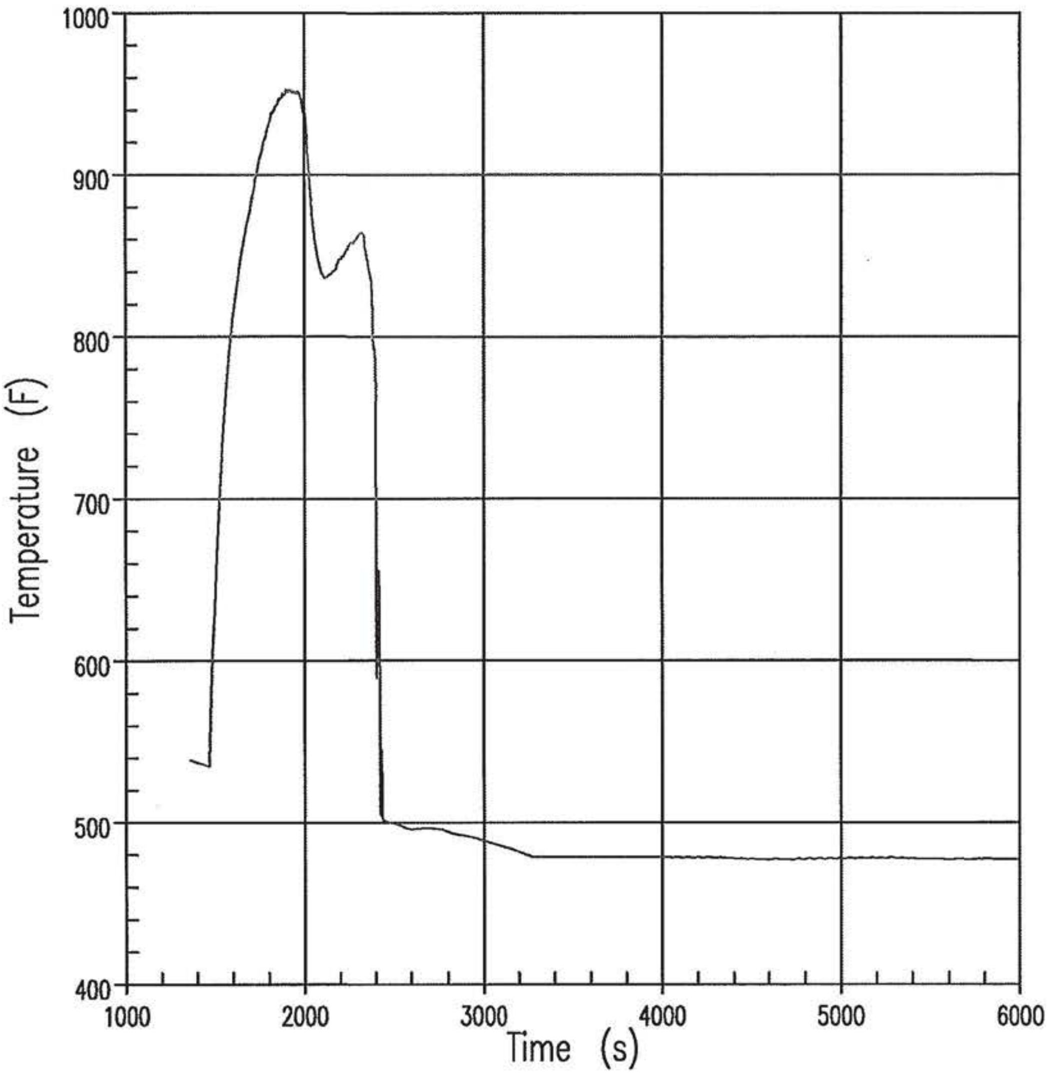

CLADDING TEMPERATURE AT PCT ELEVATION
2-INCH BREAK

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14720.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-20 REV. 33*

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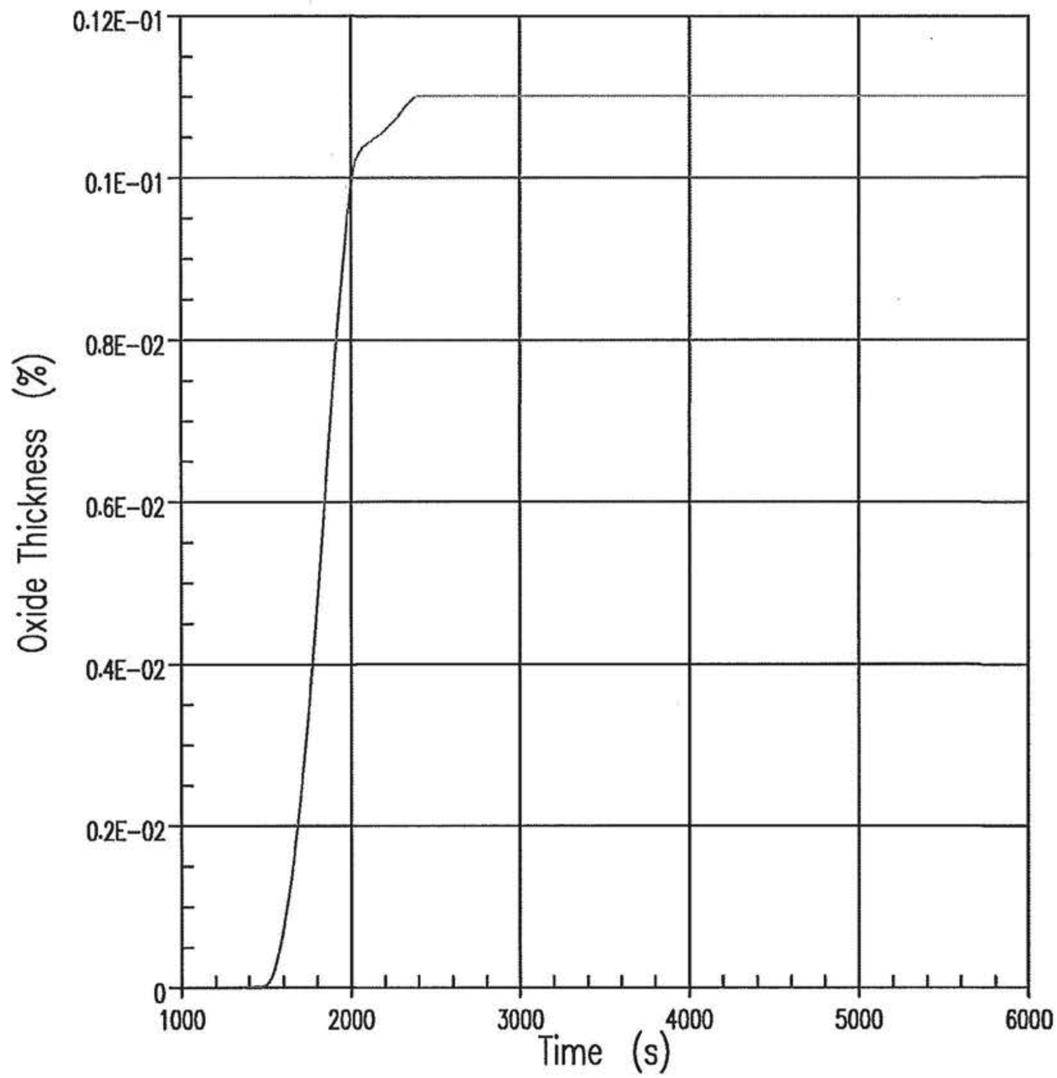

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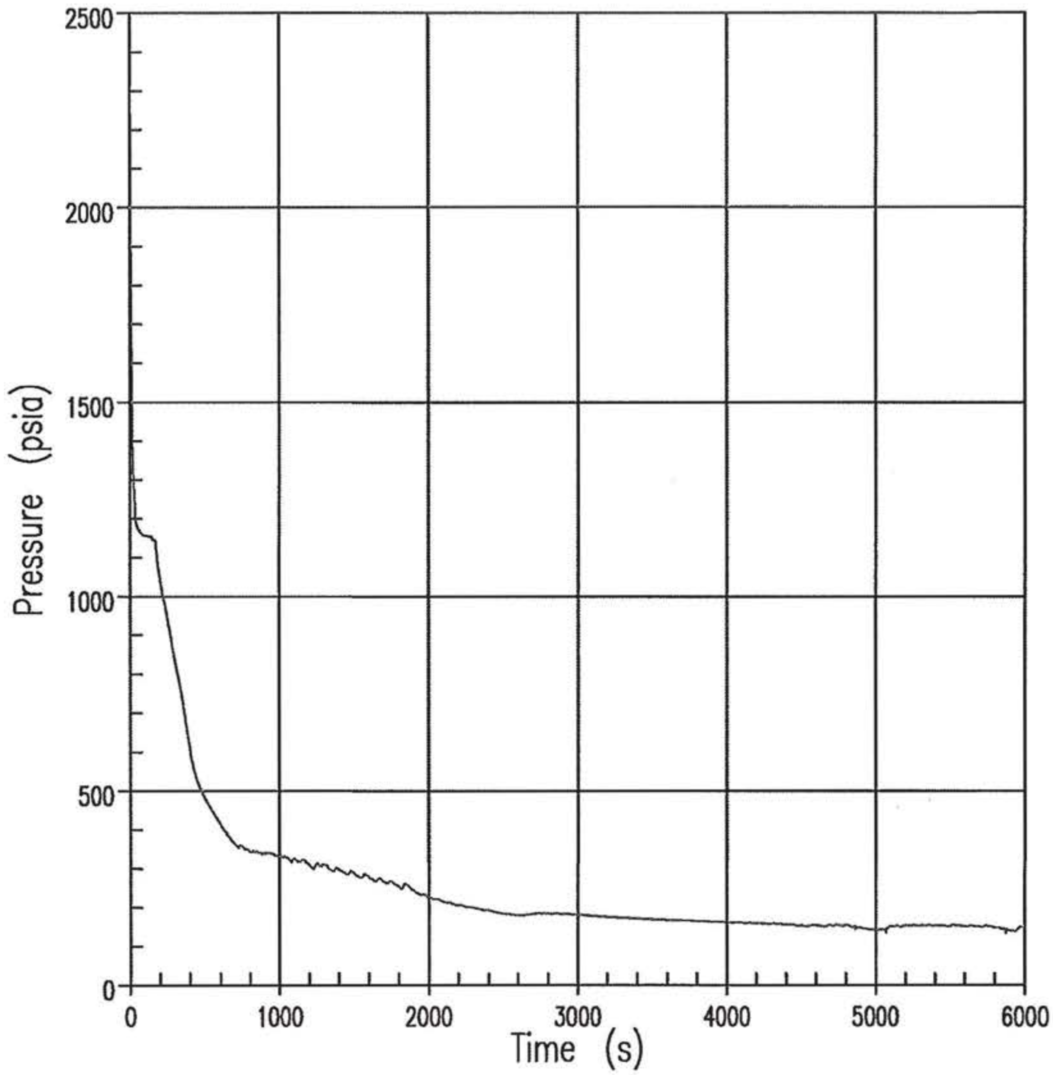
LOCAL ZrO_2 THICKNESS AT MAXIMUM LOCAL ZrO_2 ELEVATION
2-INCH BREAK

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14721.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-21*	*REV. 33*



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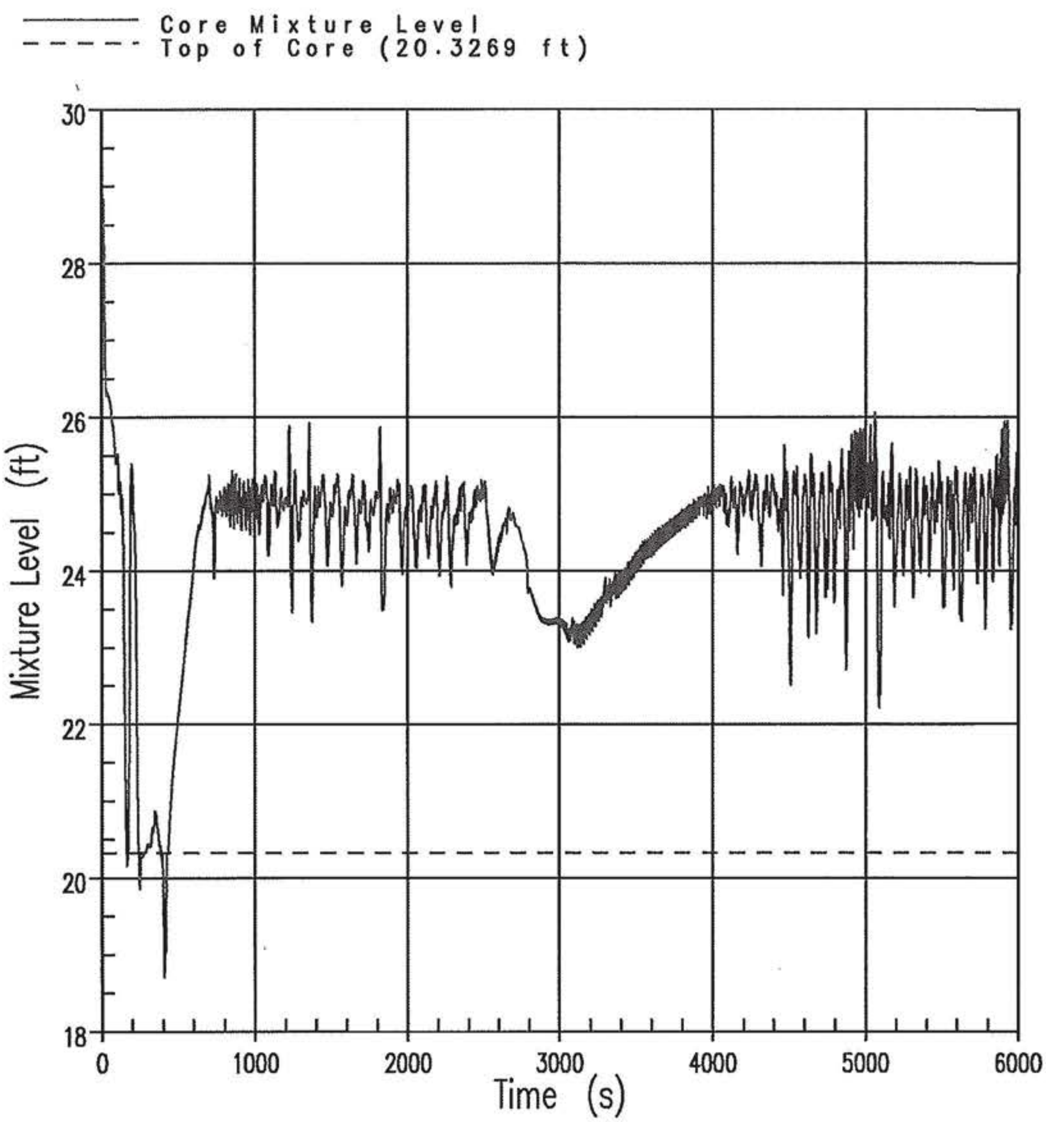
REACTOR COOLANT SYSTEM PRESSURE
4-INCH BREAK

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14722.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-22*	*REV. 33*



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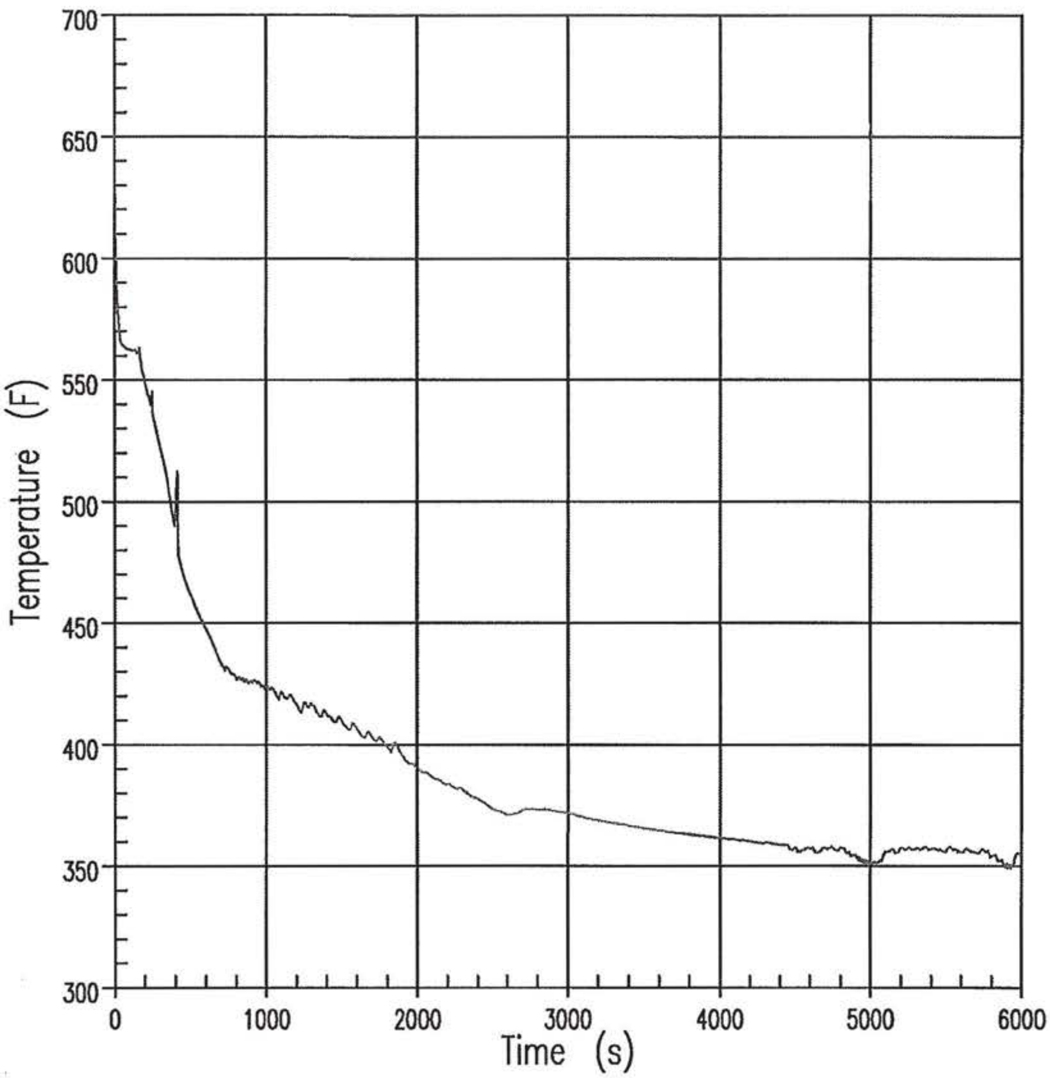
CORE MIXTURE LEVEL
4–INCH BREAK

DWN KJF	DATE 2–24–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14723.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-23 REV. 33*



TOP CORE EXIT VAPOR TEMPERATURE
4-INCH BREAK

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14724.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-24 REV. 33*

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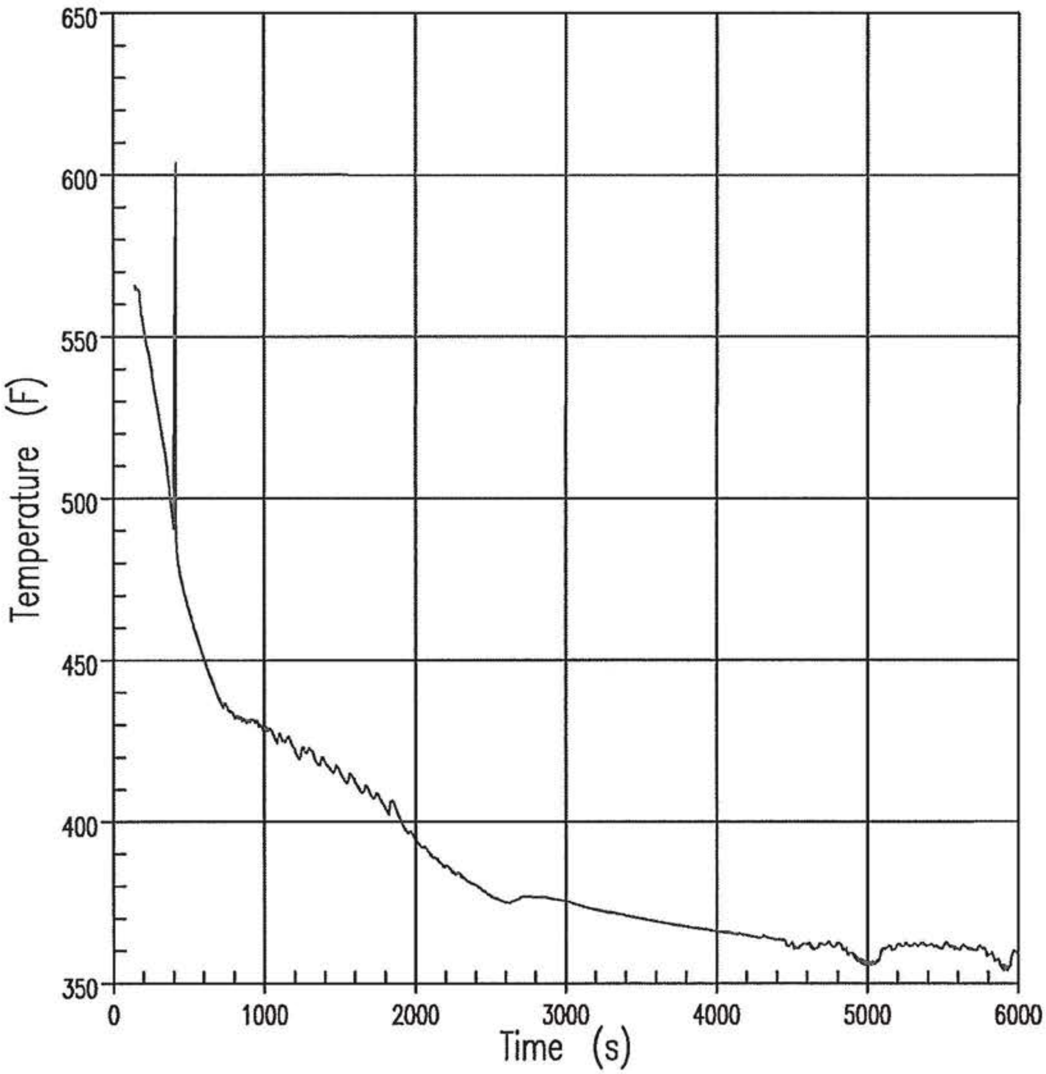

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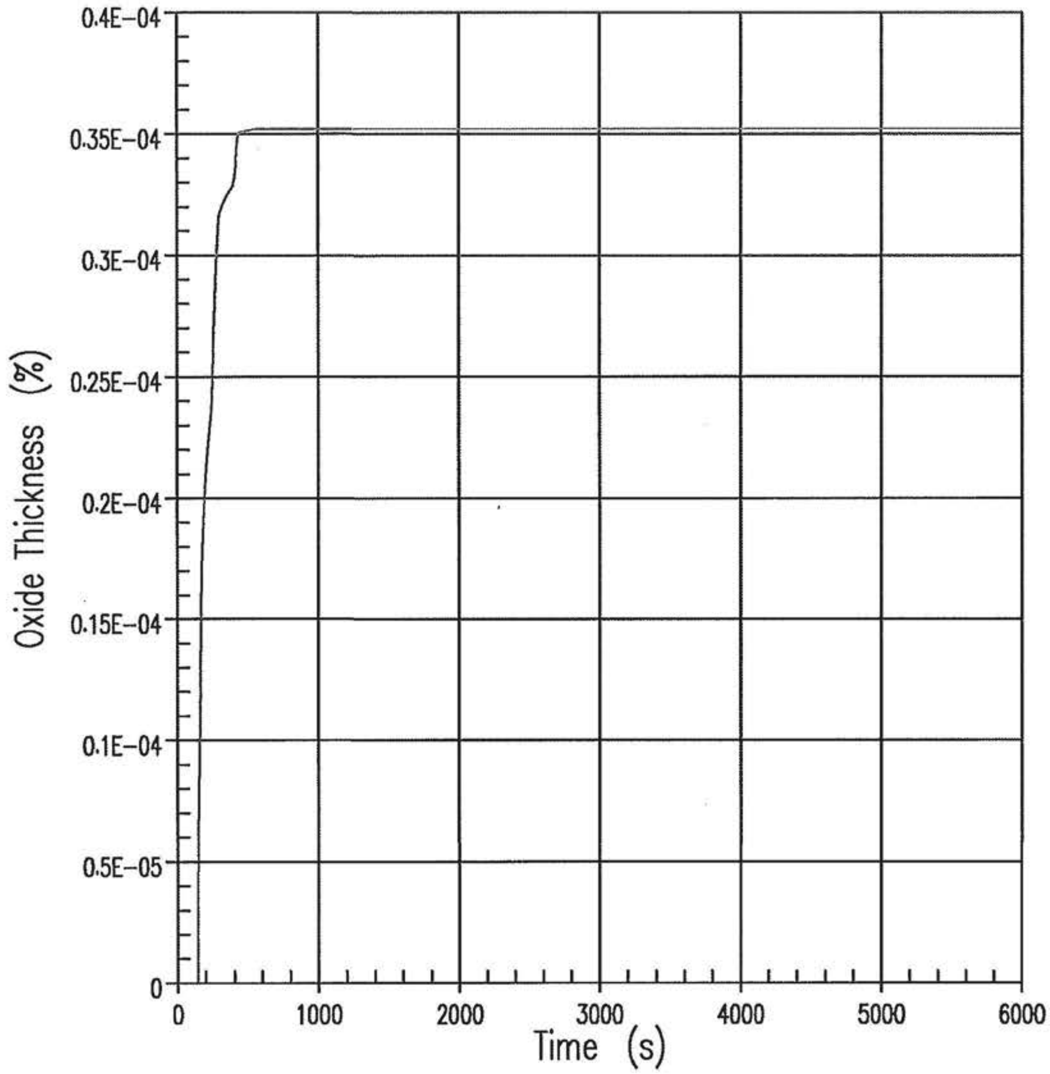
CLADDING TEMPERATURE AT PCT ELEVATION
4-INCH BREAK

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14725.DGN		*FIGURE 14.7-25 REV. 33*



LOCAL ZrO_2 THICKNESS AT MAXIMUM LOCAL ZrO_2 ELEVATION
4–INCH BREAK (UNIT 1)

DWN KJF	DATE 2–24–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14726.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-26* *REV. 33*

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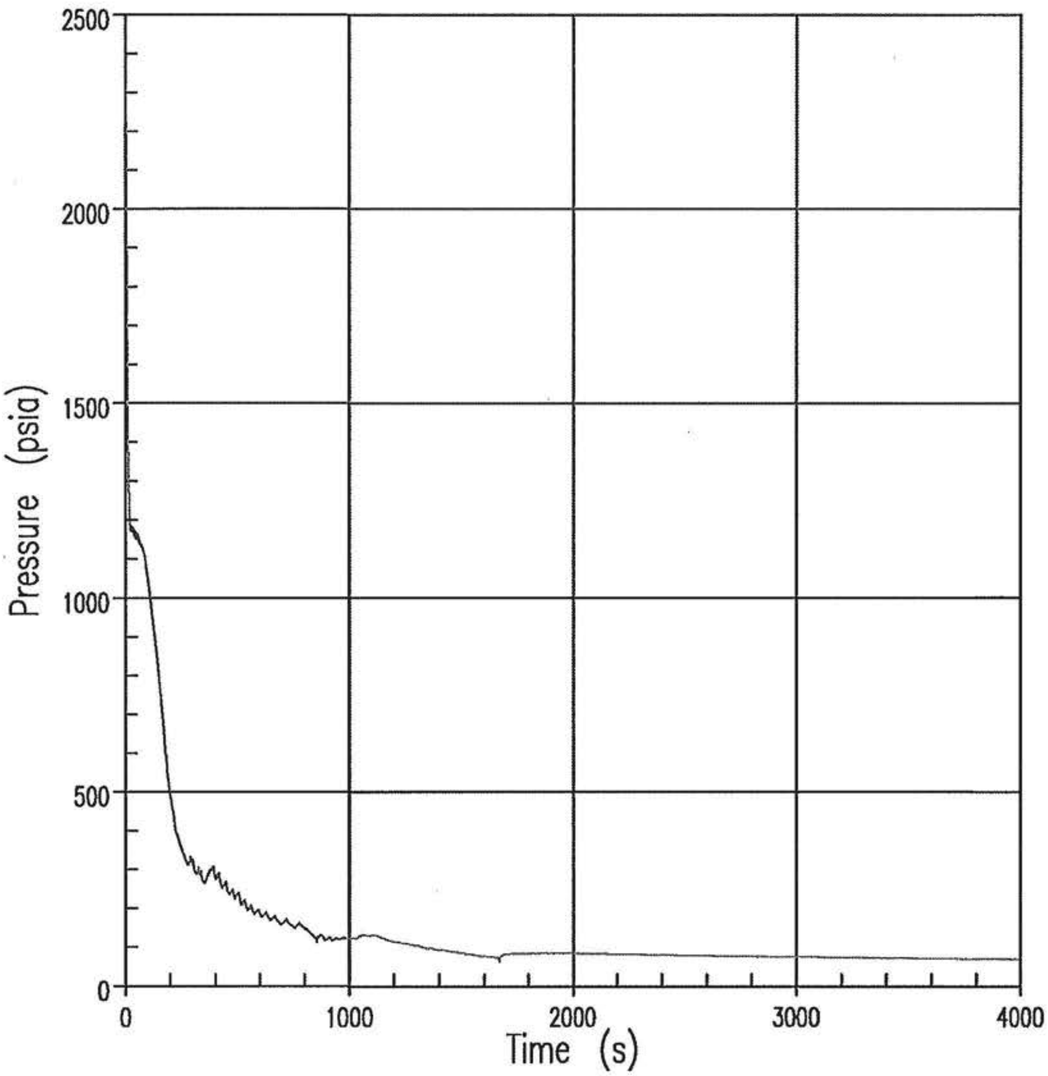

REACTOR COOLANT SYSTEM PRESSURE
6-INCH BREAK (UNIT 1)

DWN KJF	DATE 2-24-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14727.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-27 REV. 33*

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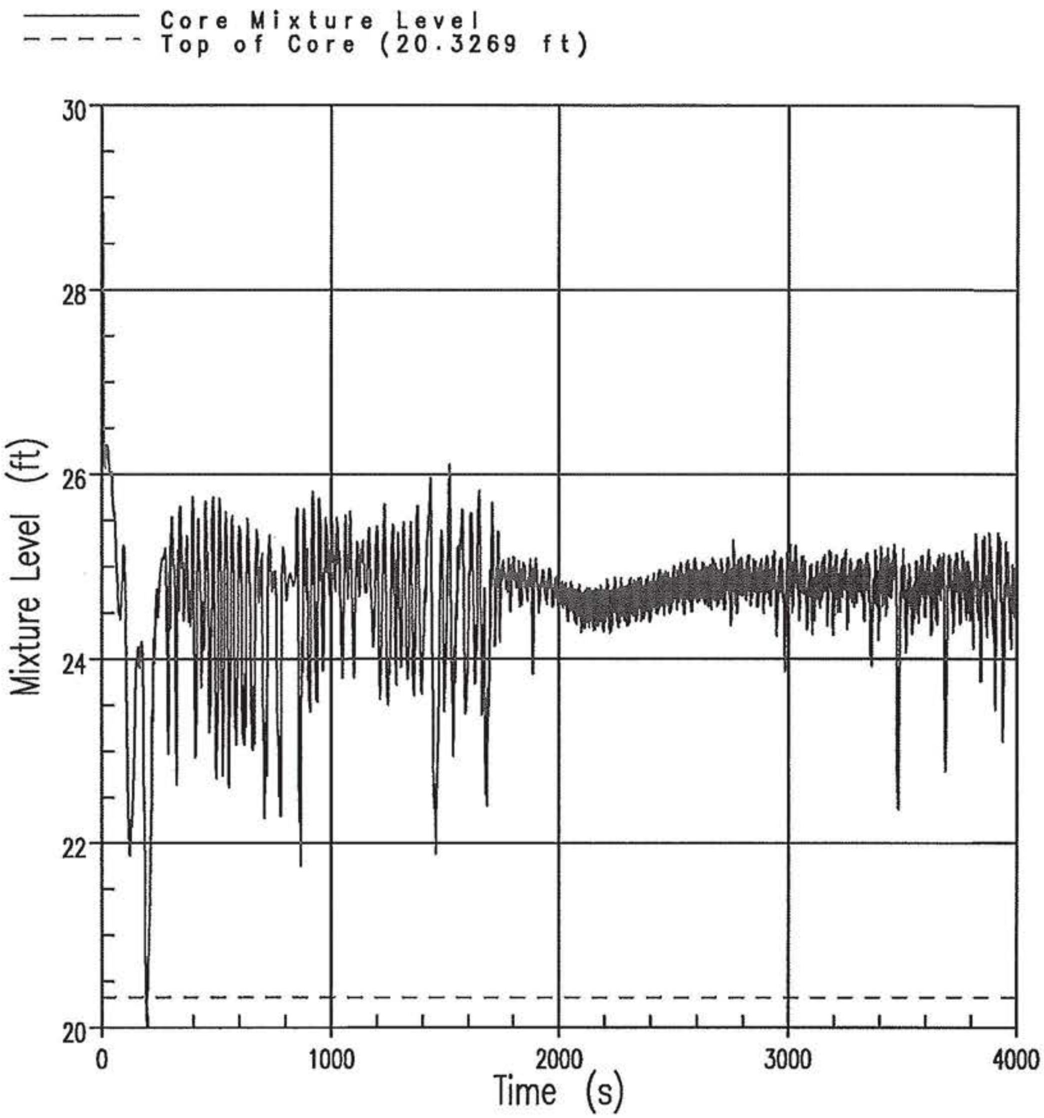

CORE MIXTURE LEVEL
6-INCH BREAK

DWN KJF	DATE 2-26-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14728.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-28 REV. 33*	

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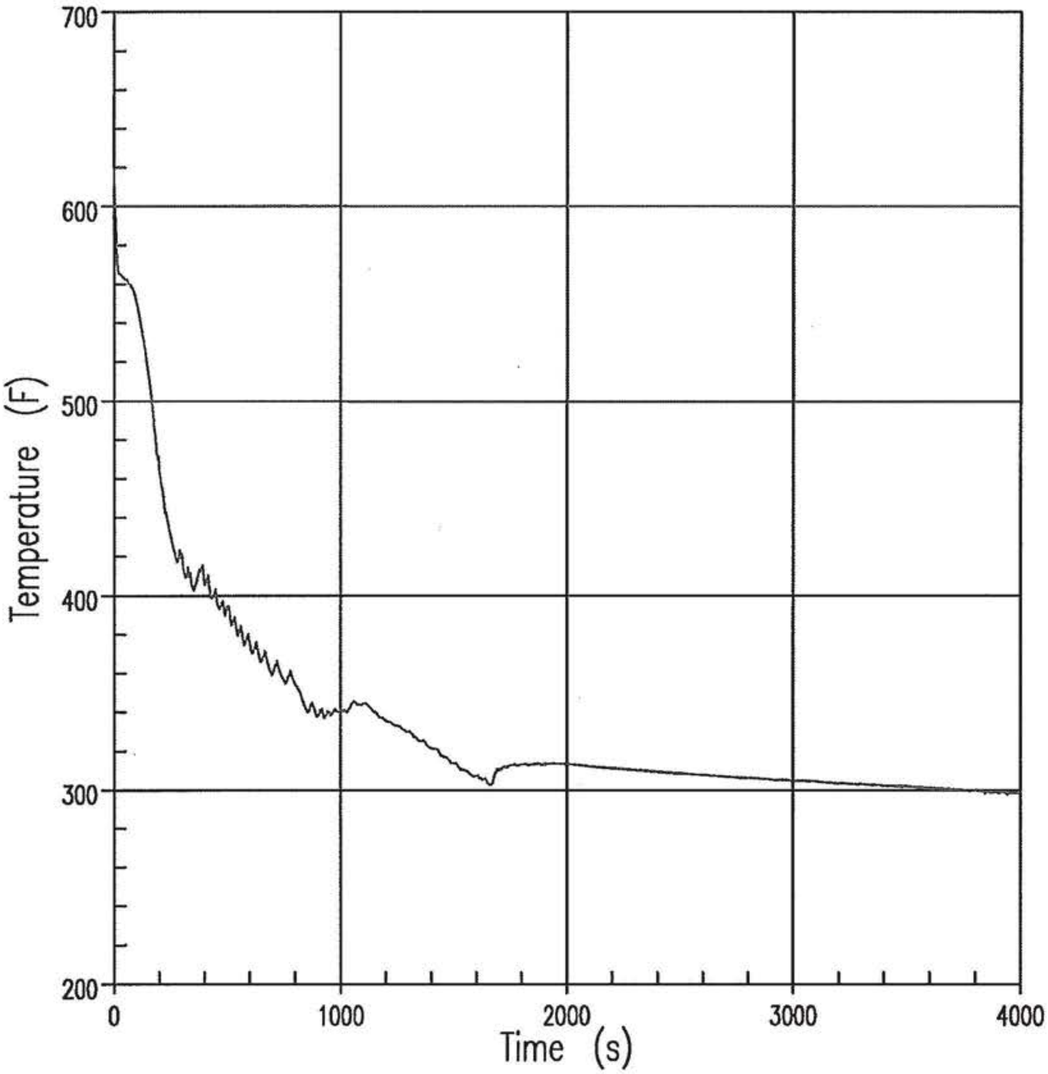

TOP CORE EXIT VAPOR TEMPERATURE 6–INCH BREAK

DWN KJF	DATE 2–26–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14729.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-29*	*REV. 33*

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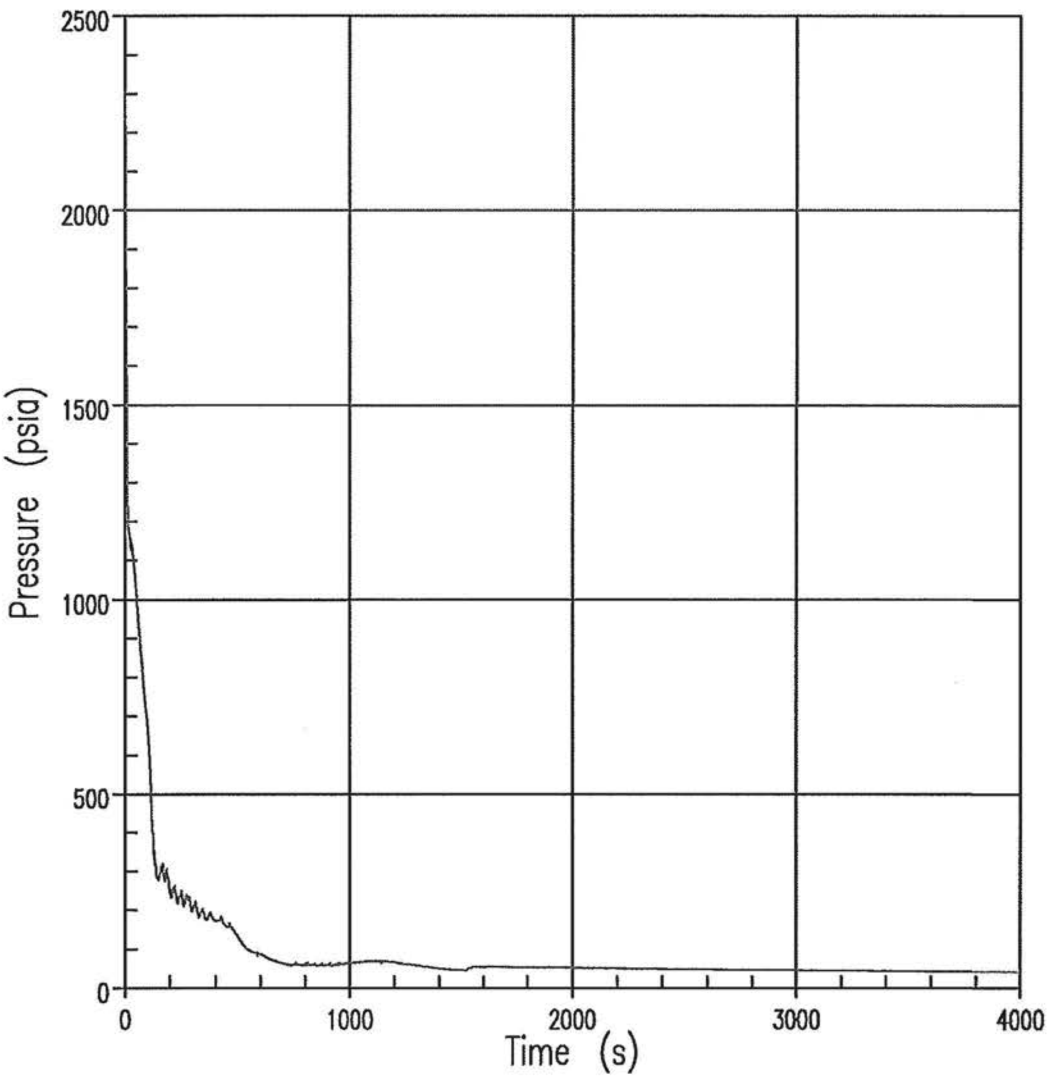

REACTOR COOLANT SYSTEM PRESSURE 8–INCH BREAK

DWN KJF	DATE 2–26–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14730.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-30 REV. 33*

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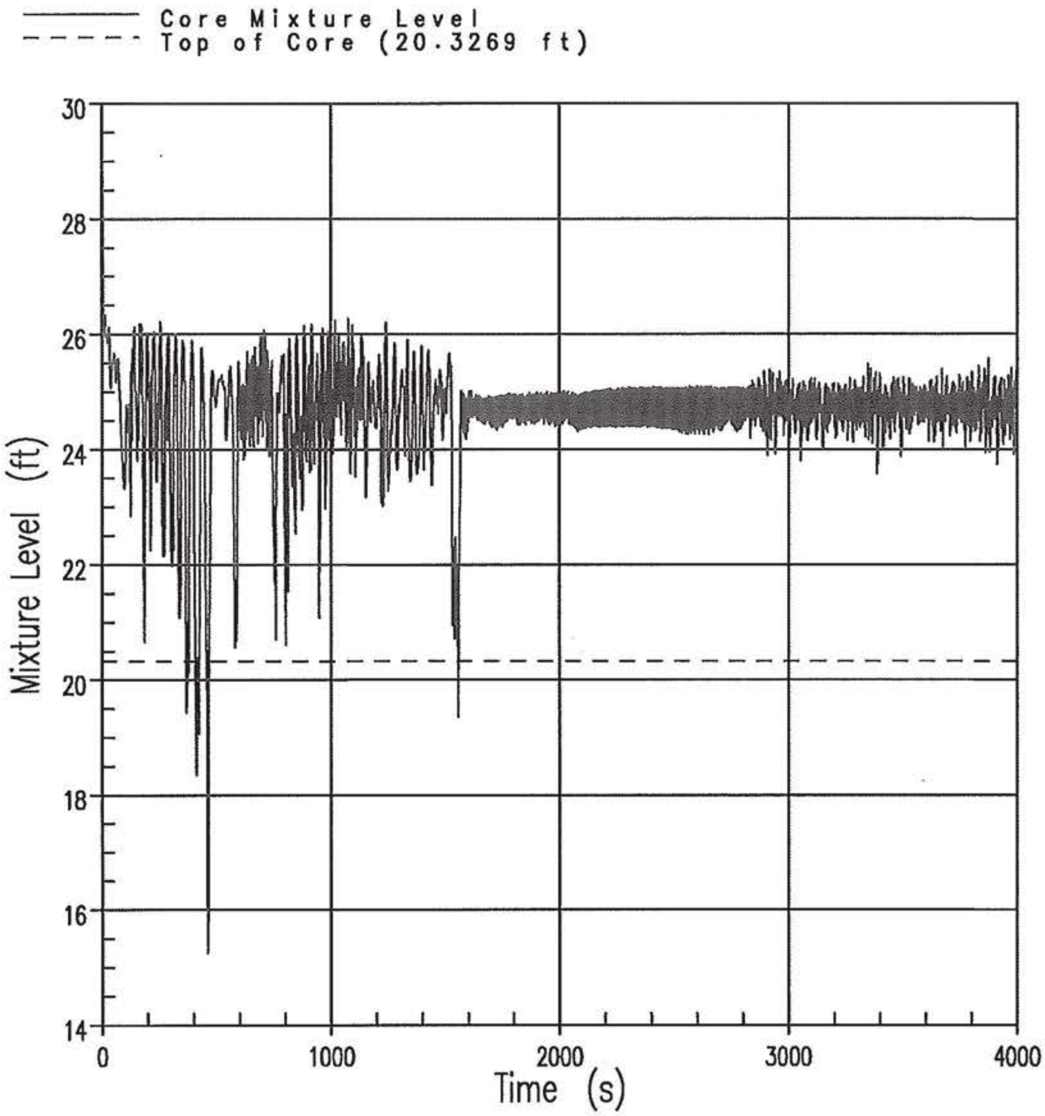

CORE MIXTURE LEVEL 8–INCH BREAK

DWN KJF	DATE 2–26–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14731.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-31*	*REV. 33*

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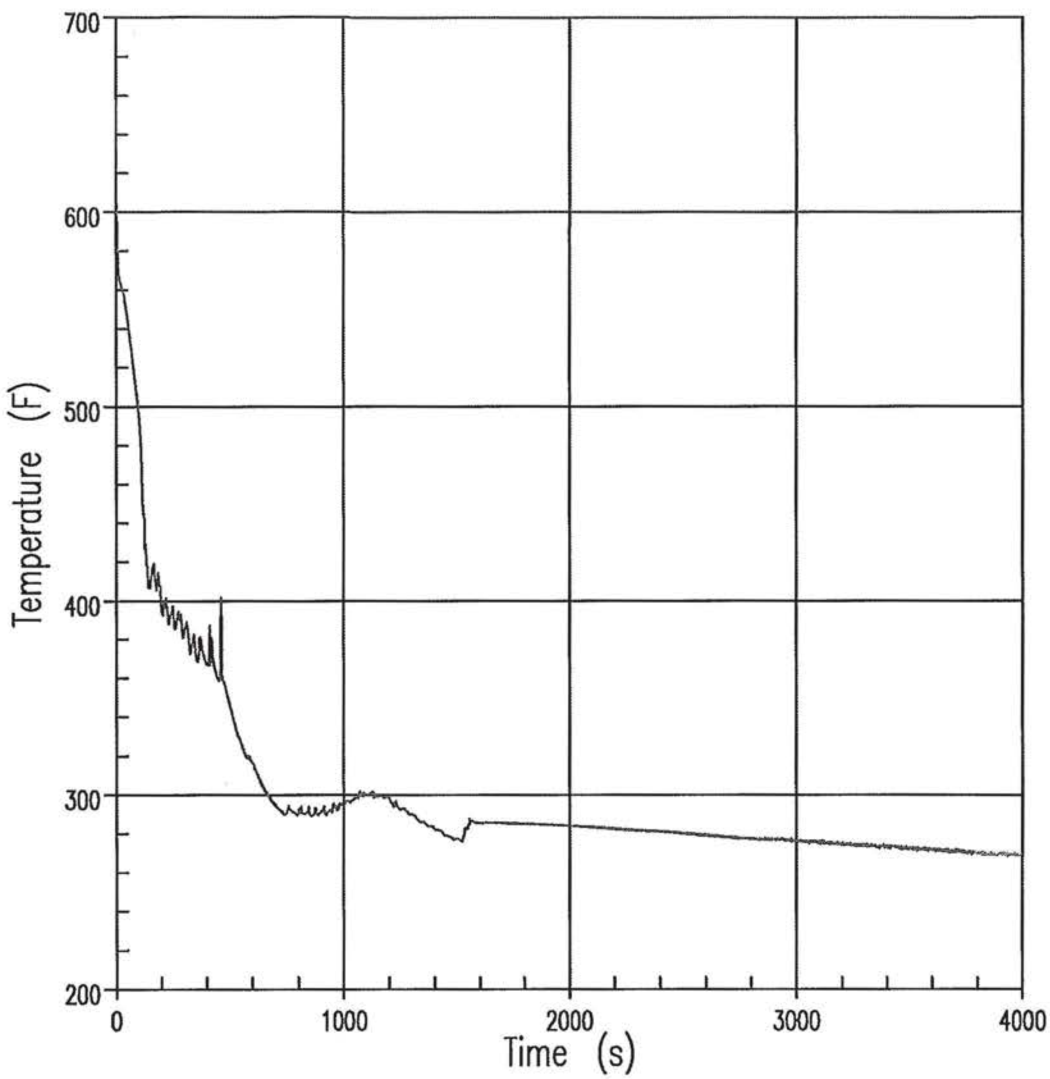

TOP CORE EXIT VAPOR TEMPERATURE 8–INCH BREAK

DWN KJF	DATE 2–26–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14732.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-32 REV. 33*

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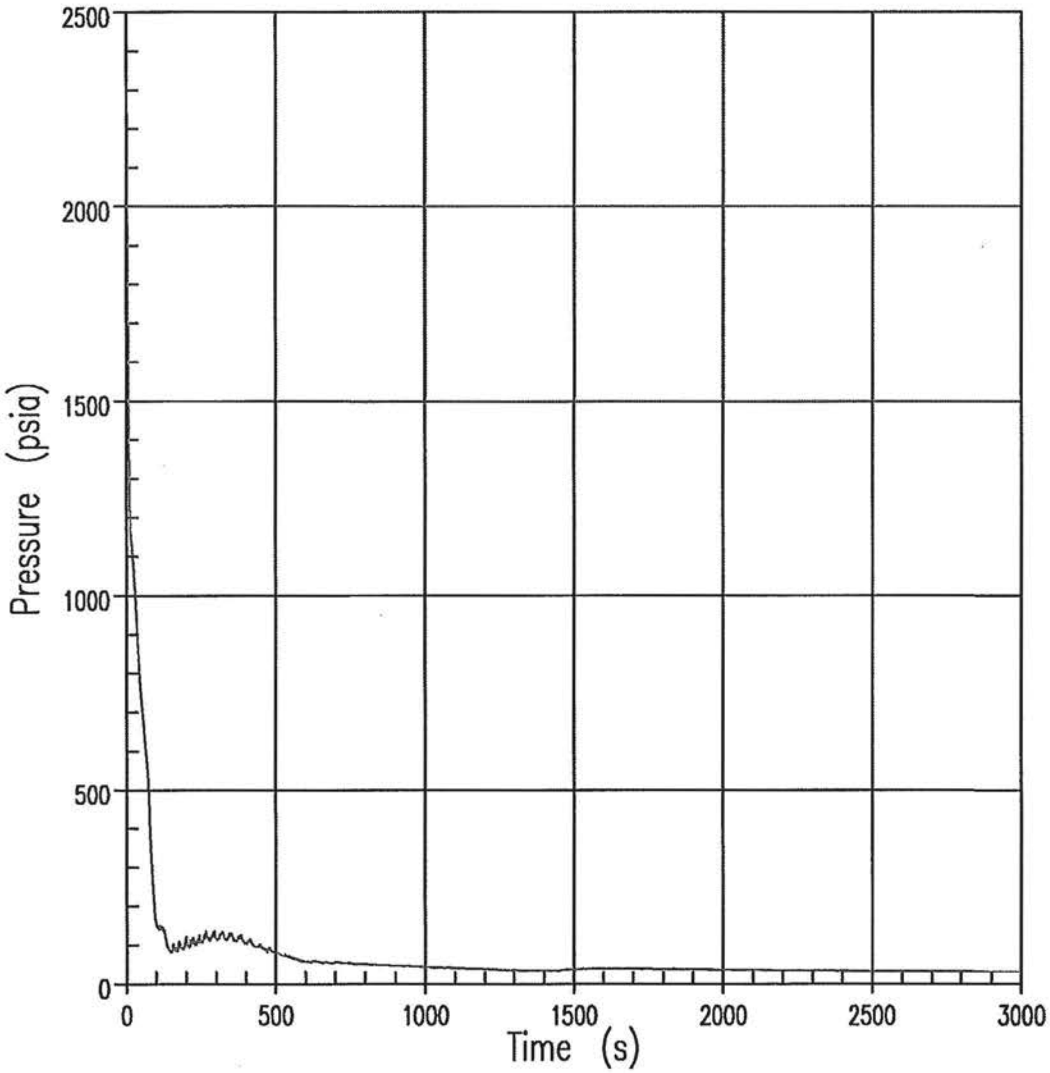

REACTOR COOLANT SYSTEM PRESSURE 10.126–INCH BREAK

DWN KJF	DATE 2–26–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14733.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-33* *REV. 33*

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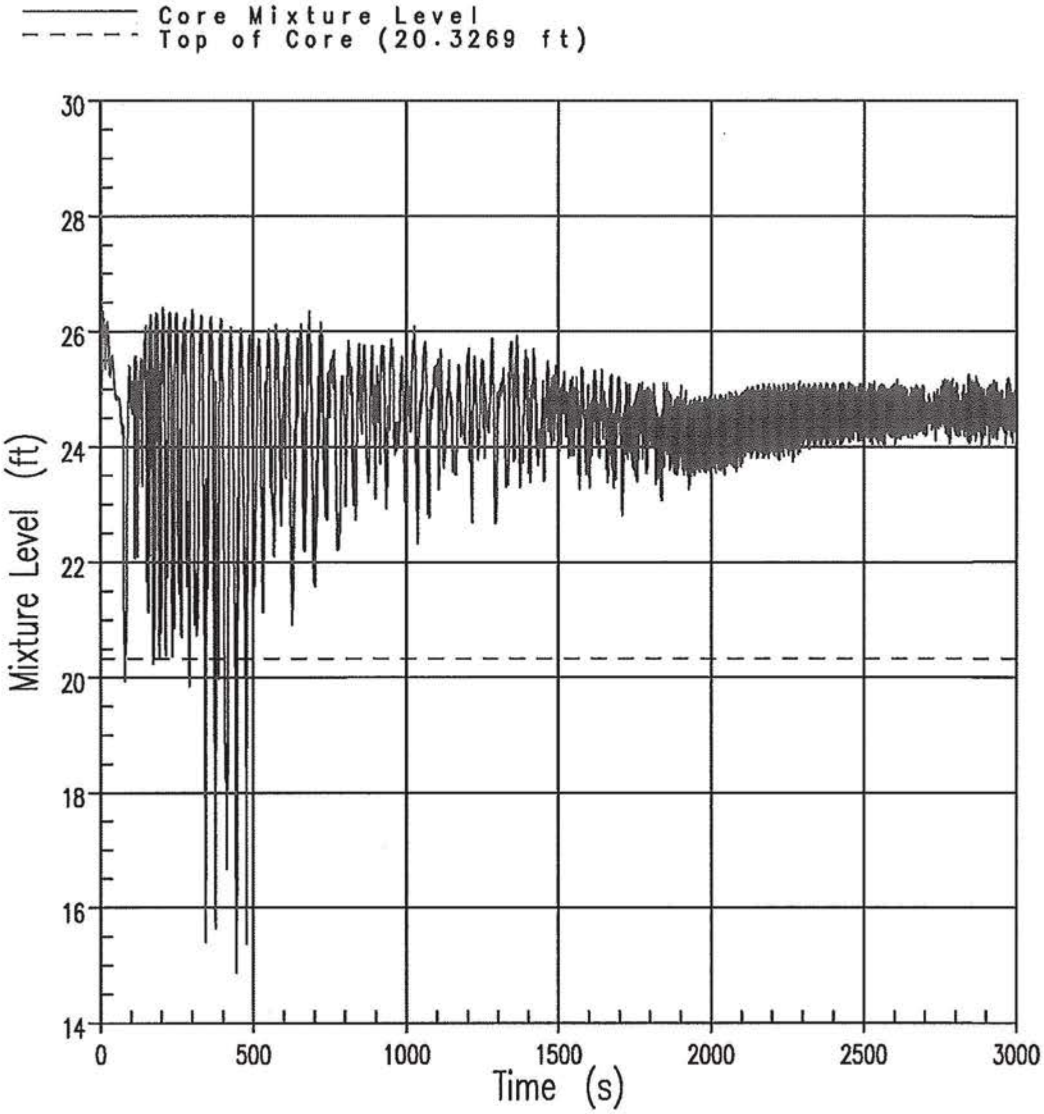

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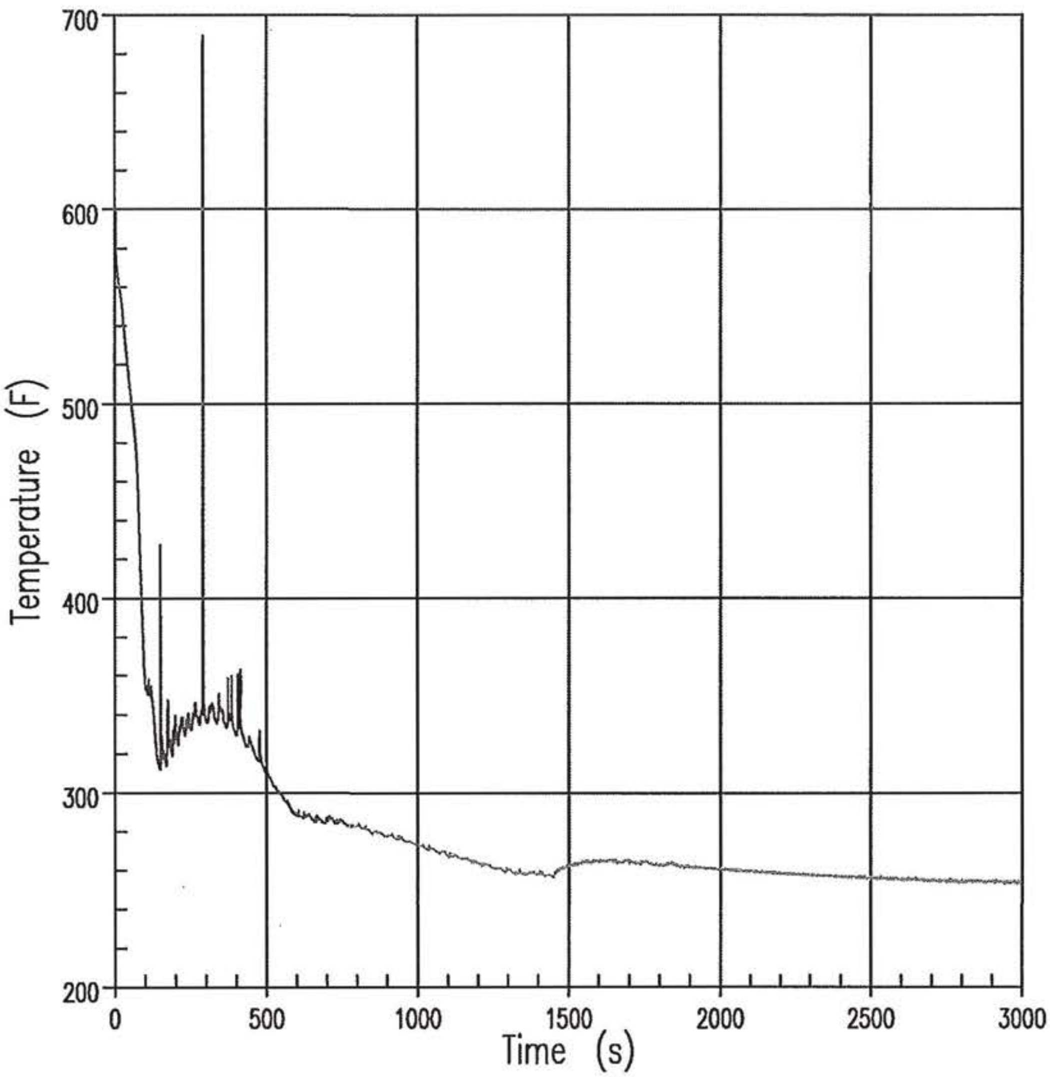
CORE MIXTURE LEVEL 10.126–INCH BREAK

DWN KJF	DATE 2–26–14	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14734.DGN		*FIGURE 14.7-34 REV. 33*



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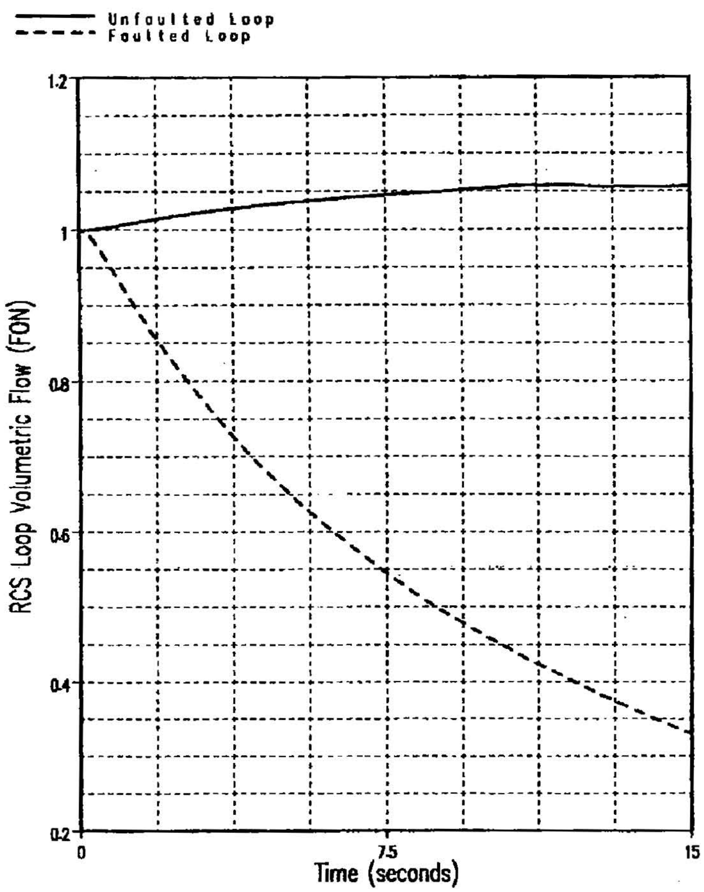
TOP CORE EXIT VAPOR TEMPERATURE 10.126–INCH BREAK

DWN KJF	DATE 2–26–14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14735.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.7-35*	*REV. 33*

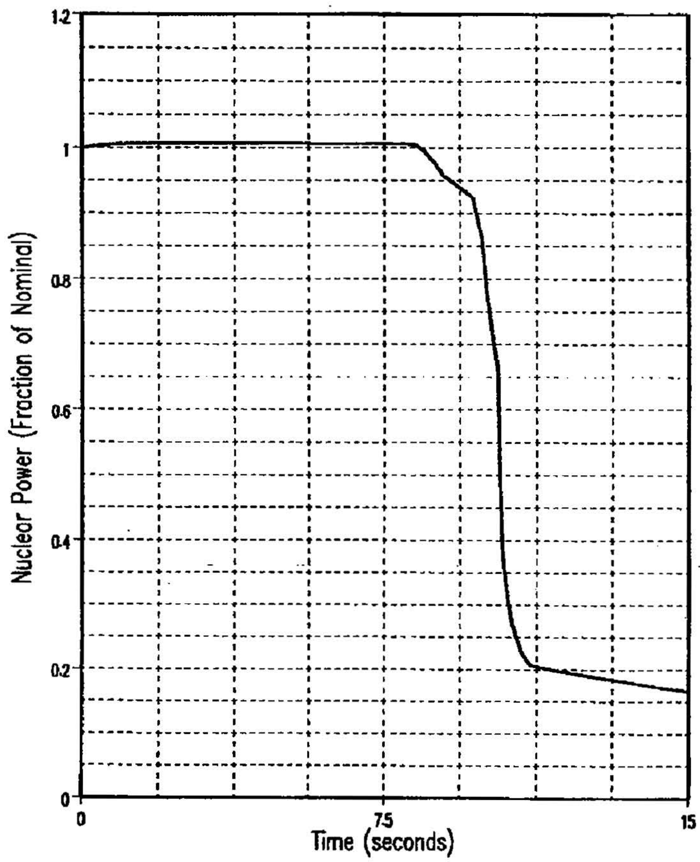


AMSAC/DSS:
PARTIAL LOSS OF FLOW,
ONE PUMP COASTING DOWN – RCS LOOP FLOW VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14801.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-01 REV. 31*	

01183316

FIGURE 14.8-01



AMSAC/DSS:
PARTIAL LOSS OF FLOW,
ONE PUMP COASTING DOWN – NUCLEAR POWER VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE UI4802.DGN		*FIGURE 14.8-02 REV. 31*

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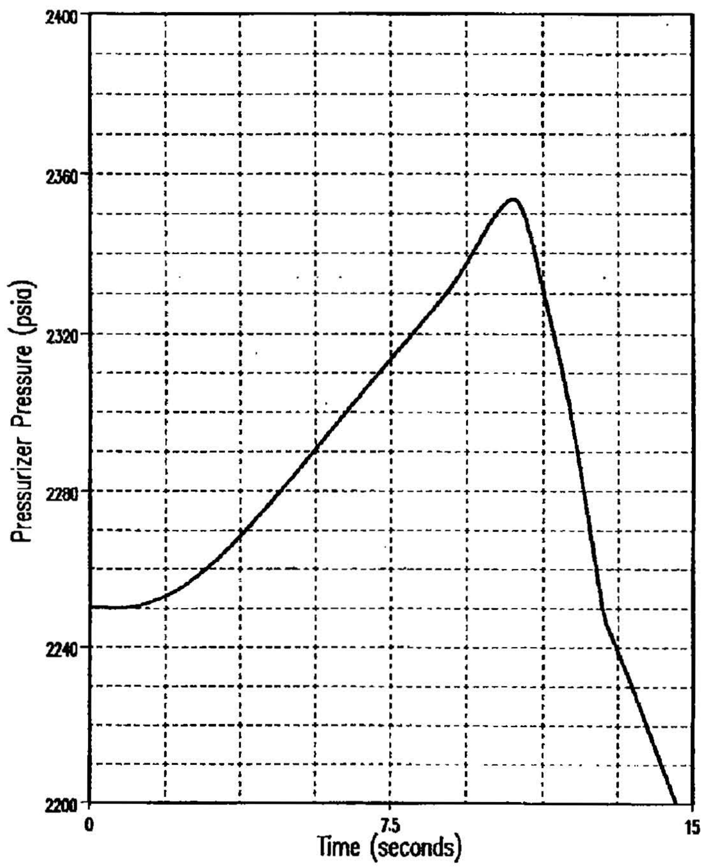
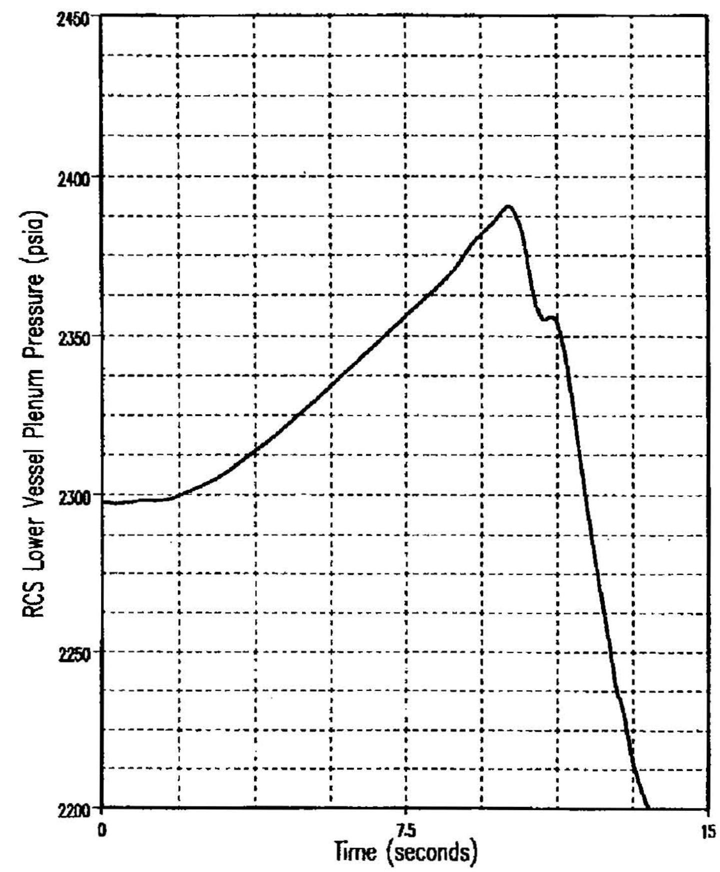
FIGURE 14.8-02



AMSAC/DSS:
PARTIAL LOSS OF FLOW,
ONE PUMP COASTING DOWN – PRESSURIZER PRESSURE VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14803.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-03 REV. 31*

01183316

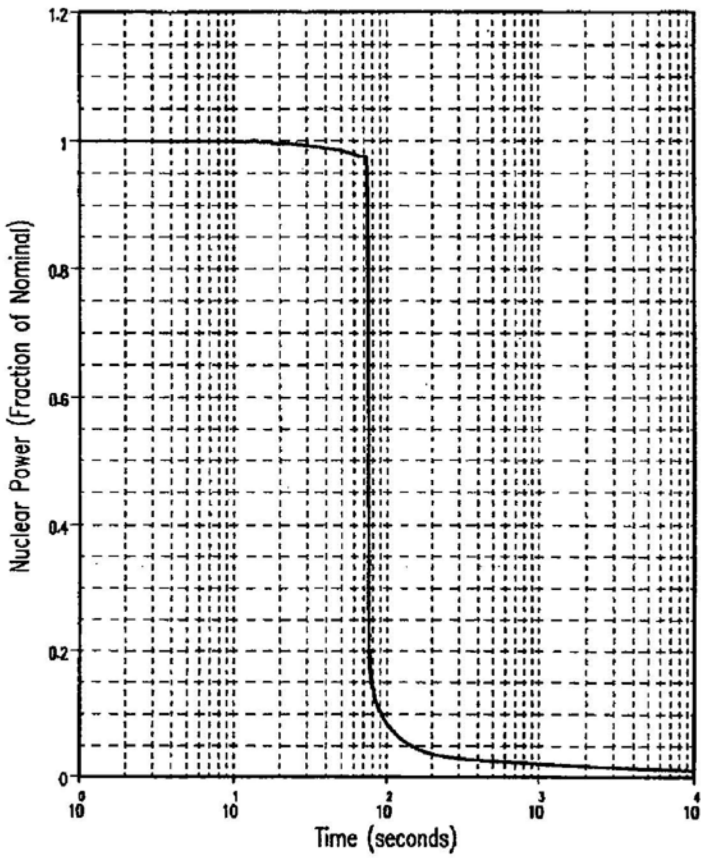


AMSAC/DSS:
PARTIAL LOSS OF FLOW,
ONE PUMP COASTING DOWN – RCS PRESSURE VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE NONE
CHECKED	CAD FILE U14804.DGN		*FIGURE 14.8-04 REV. 31*

01183316

FIGURE 14.8-04

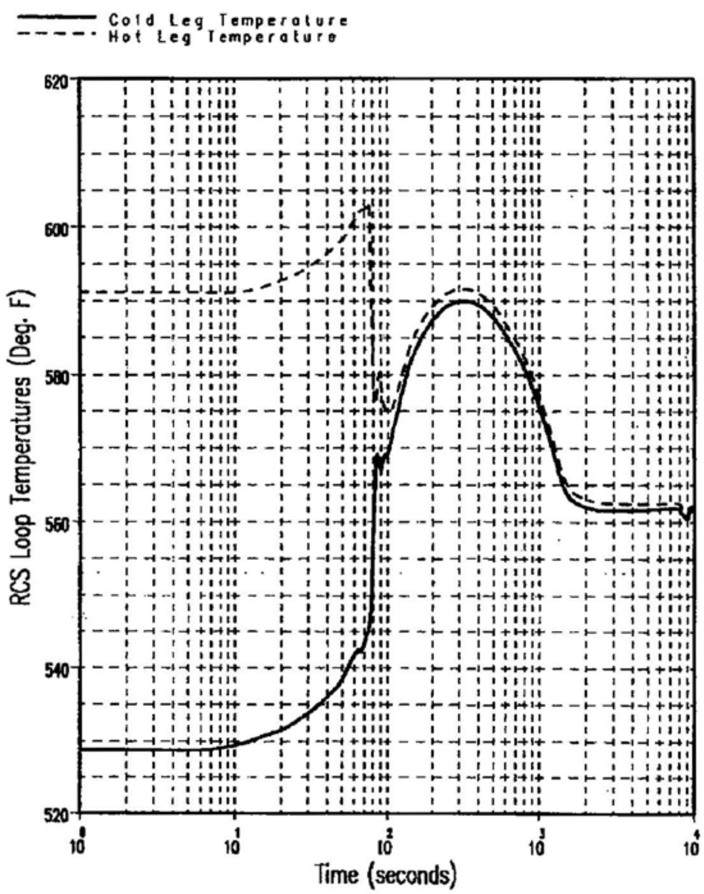


AMSAC/DSS:
LOSS OF NORMAL FEEDWATER – NUCLEAR POWER VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14805.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-05 REV. 31*

01183316

FIGURE 14.8-05

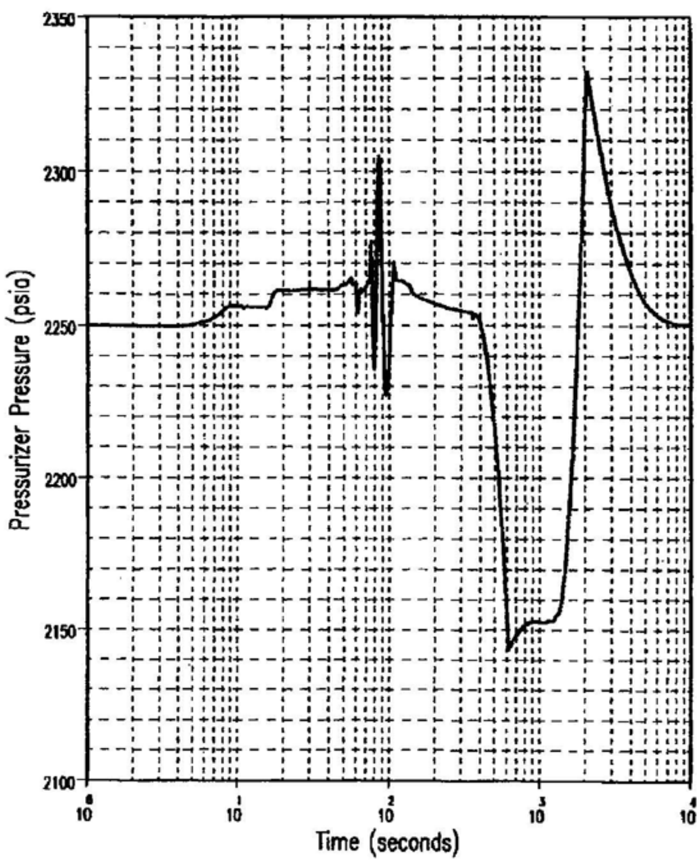


AMSAC/DSS:
LOSS OF NORMAL FEEDWATER – REACTOR COOLANT LOOP TEMPERATURES VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14806.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-06 REV. 31*

FIGURE 14.8-6

01183316

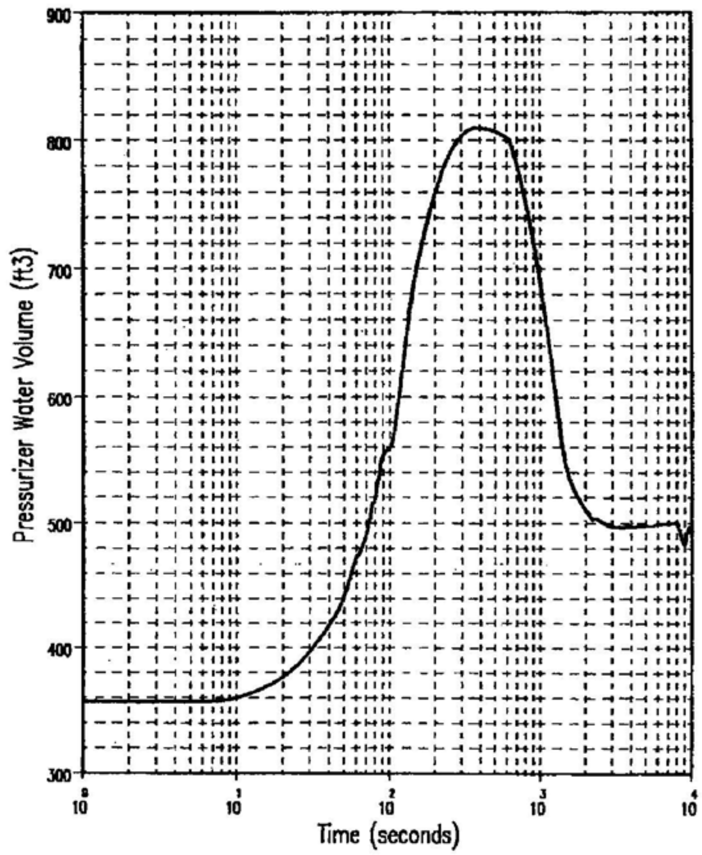


AMSAC/DSS:
LOSS OF NORMAL FEEDWATER – PRESSURIZER PRESSURE VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14807.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-07 REV. 31*

01183316

FIGURE 14.8-07

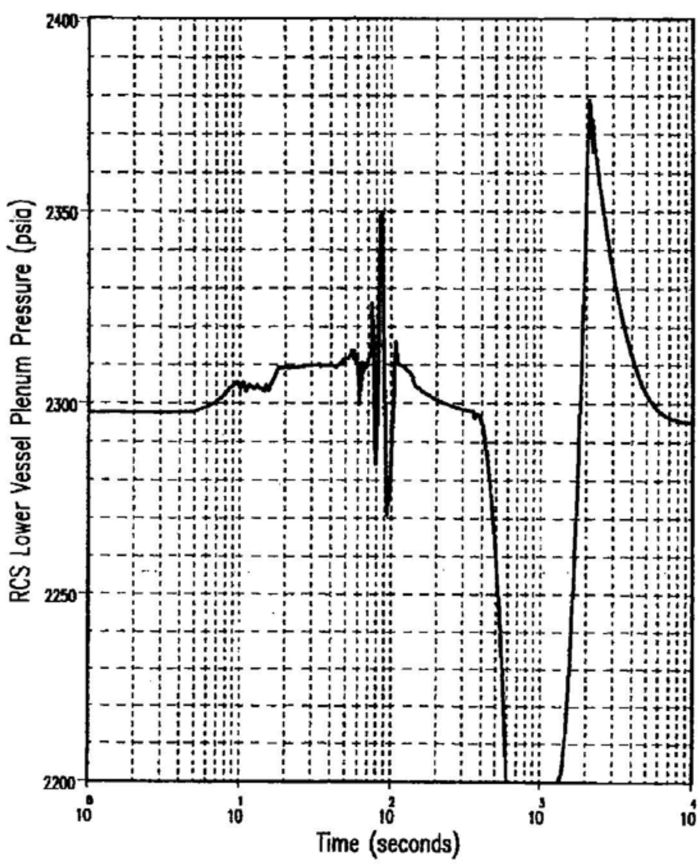


AMSAC/DSS:
LOSS OF NORMAL FEEDWATER – PRESSURIZER WATER VOLUME VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14808.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-08 REV. 31*	

01183316

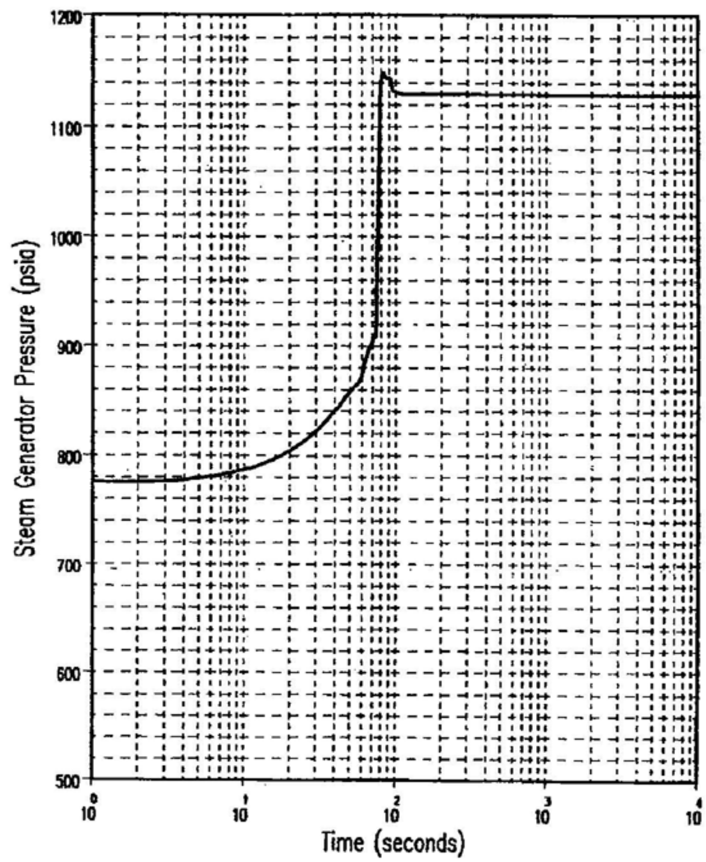
FIGURE 14.8-08



AMSAC/DSS:
LOSS OF NORMAL FEEDWATER – RCS PRESSURE VERSUS TIME

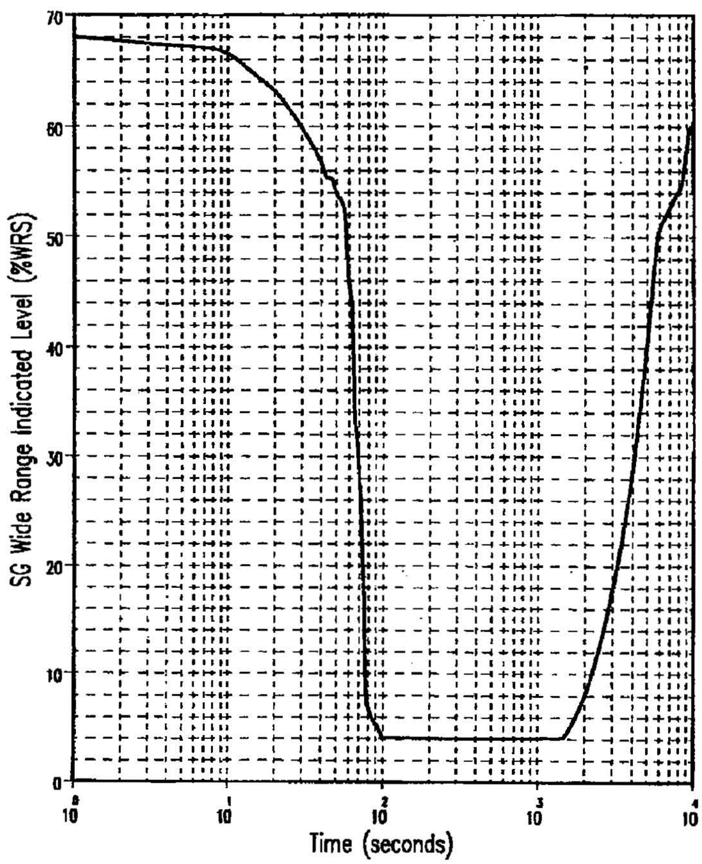
DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14809.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-09 REV. 31*	

01183316



AMSAC/DSS:
LOSS OF NORMAL FEEDWATER – STEAM GENERATOR PRESSURE VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14810.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-10 REV. 31*	

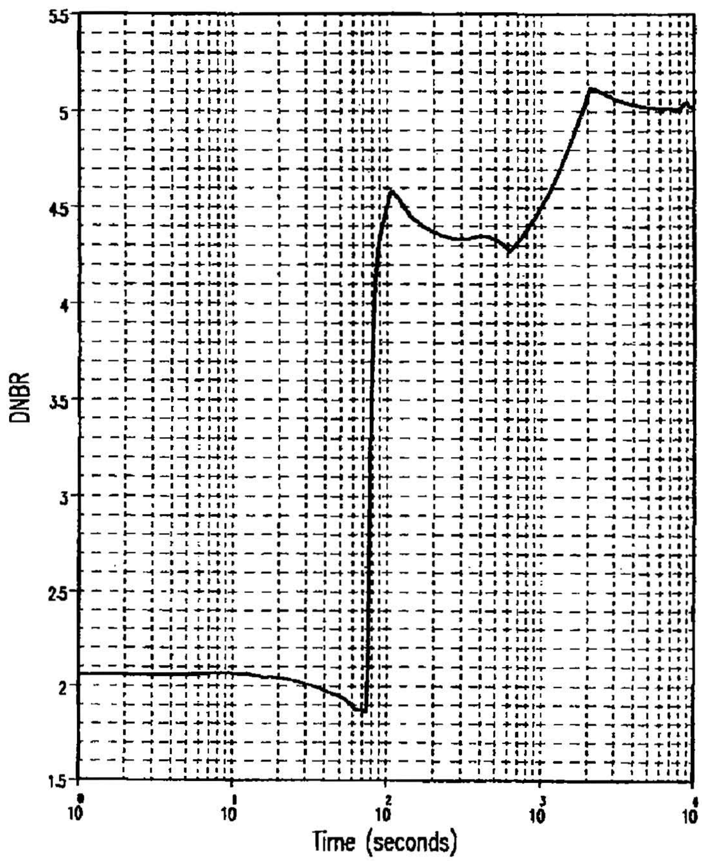


AMSAC/DSS:
LOSS OF NORMAL FEEDWATER – SG WIDE RANGE INDICATED LEVEL VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14811.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-11 REV. 31*

01183316

FIGURE 14.8-11

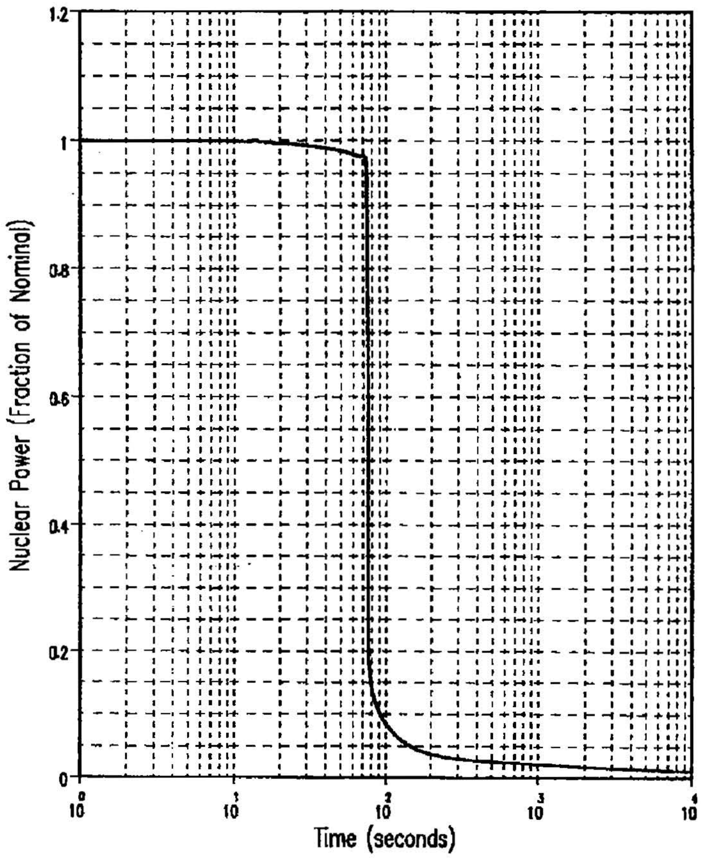


AMSAC/DSS:
LOSS OF NORMAL FEEDWATER – DNBR VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14812.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-12 REV. 31*

01183316

FIGURE 14.8-12

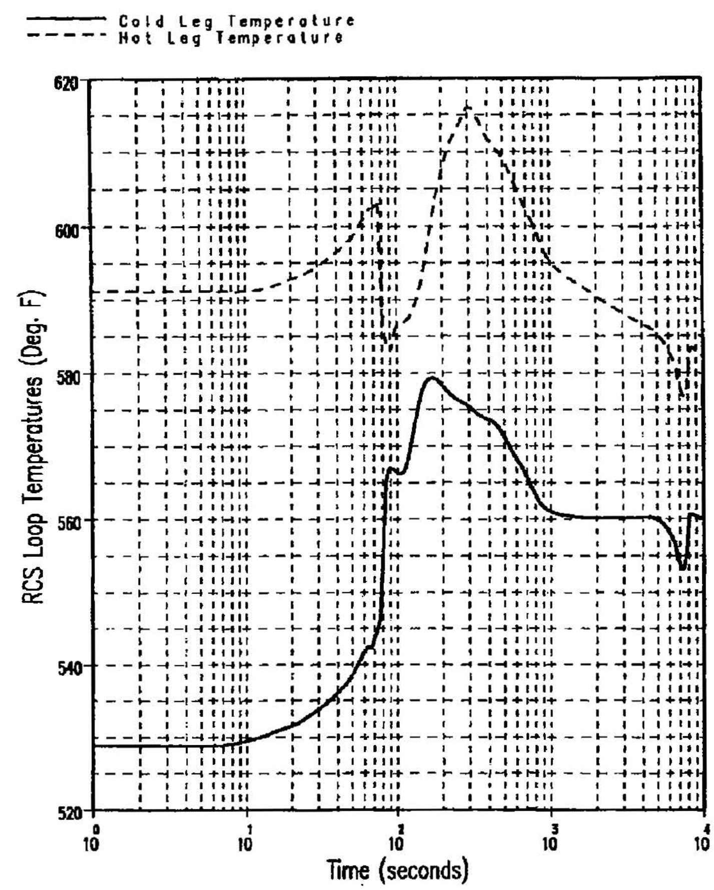


AMSAC/DSS:
LOSS OF ALL AC POWER TO THE STATION AUXILIARIES – NUCLEAR POWER VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14813.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-13 REV. 31*

01183316

FIGURE 14.8-13

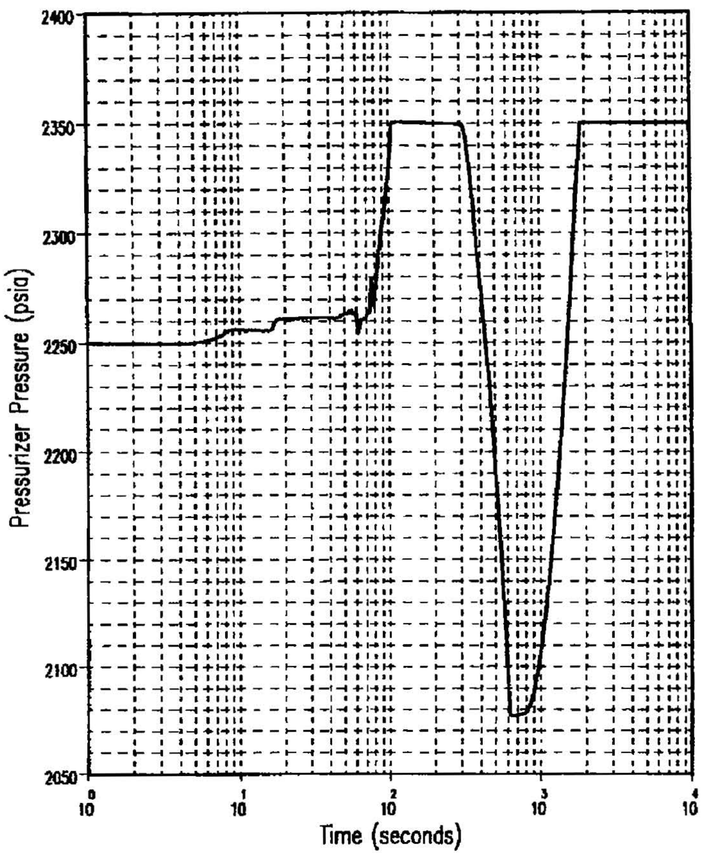


AMSAC/DSS:
LOSS OF ALL AC POWER TO THE STATION AUXILIARIES
REACTOR COOLANT LOOP TEMPERATURES VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14814.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-14 REV. 31*

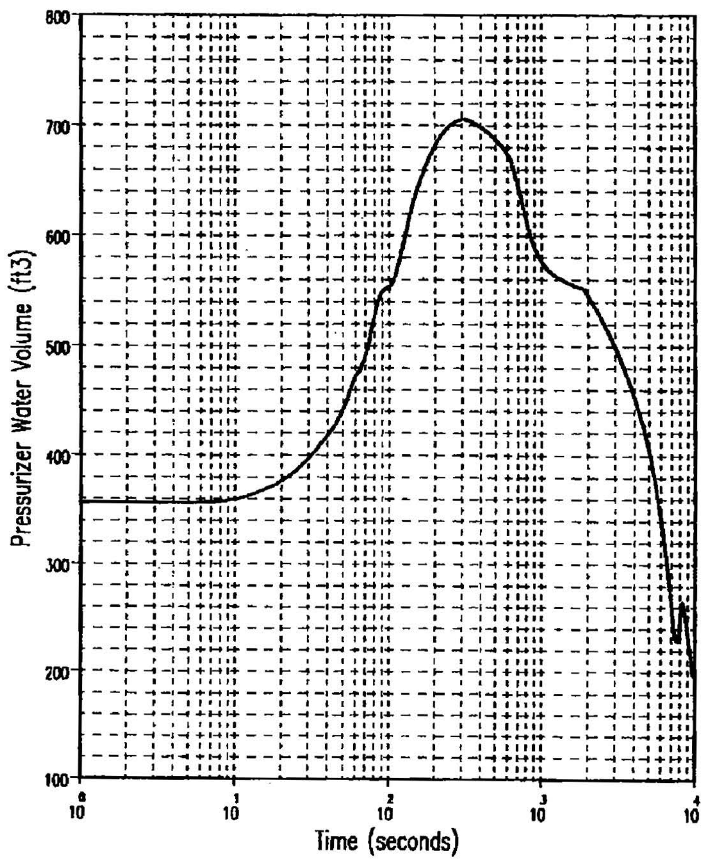
01183316

FIGURE 14.8-14



AMSAC/DSS:
LOSS OF ALL AC POWER TO THE STATION AUXILIARIES
PRESSURIZER PRESSURE VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14815.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-15 REV. 31*

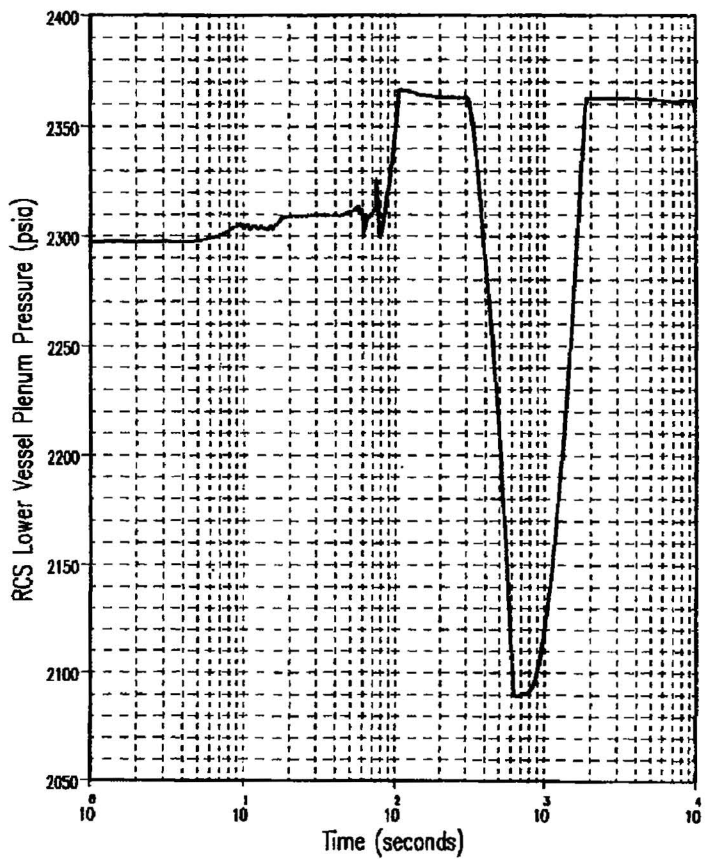


AMSAC/DSS:
LOSS OF ALL AC POWER TO THE STATION AUXILIARIES
PRESSURIZER WATER VOLUME VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE NONE
CHECKED	CAD FILE U14816.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-16 REV. 31*

01183316

FIGURE 14.8-16

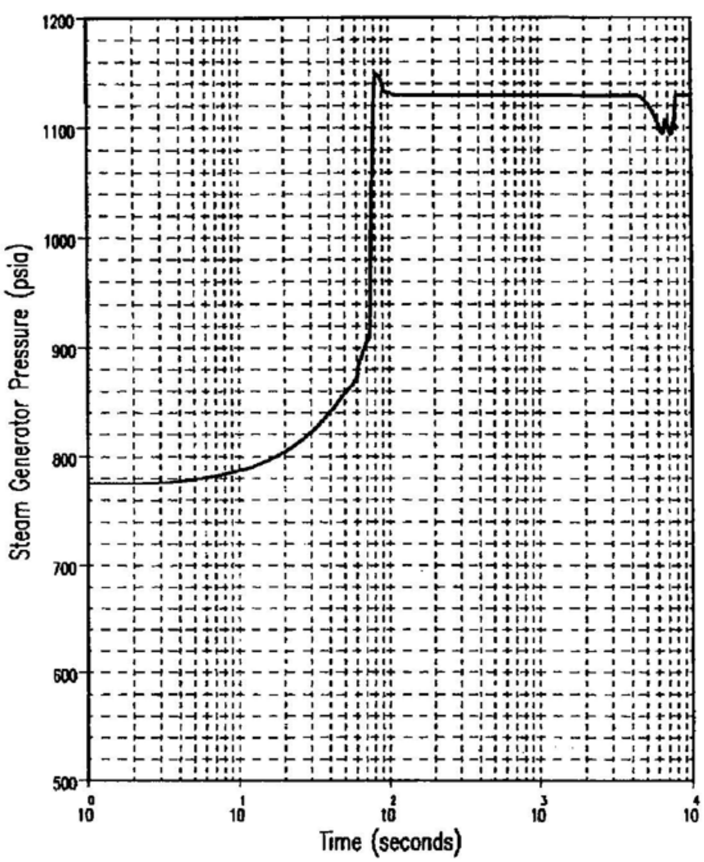


AMSAC/DSS:
LOSS OF ALL AC POWER TO THE STATION AUXILIARIES –
RCS PRESSURE VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14817.DGN		*FIGURE 14.8-17 REV. 31*

01183316

FIGURE 14.8-17

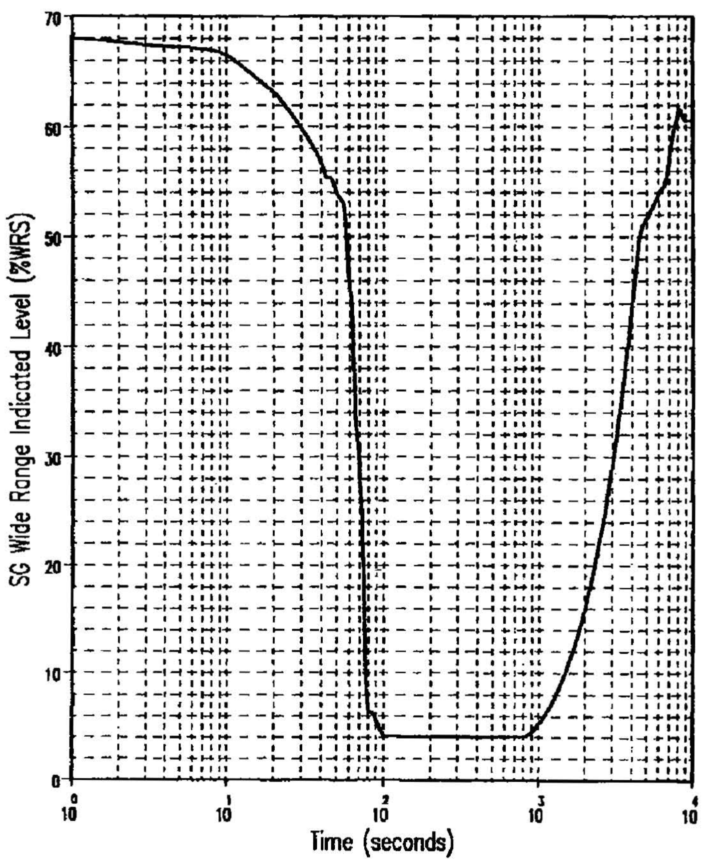


AMSAC/DSS:
LOSS OF ALL AC POWER TO THE STATION AUXILIARIES –
STEAM GENERATOR PRESSURE VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14818.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-18 REV. 31*

01183316

FIGURE 14.8-18

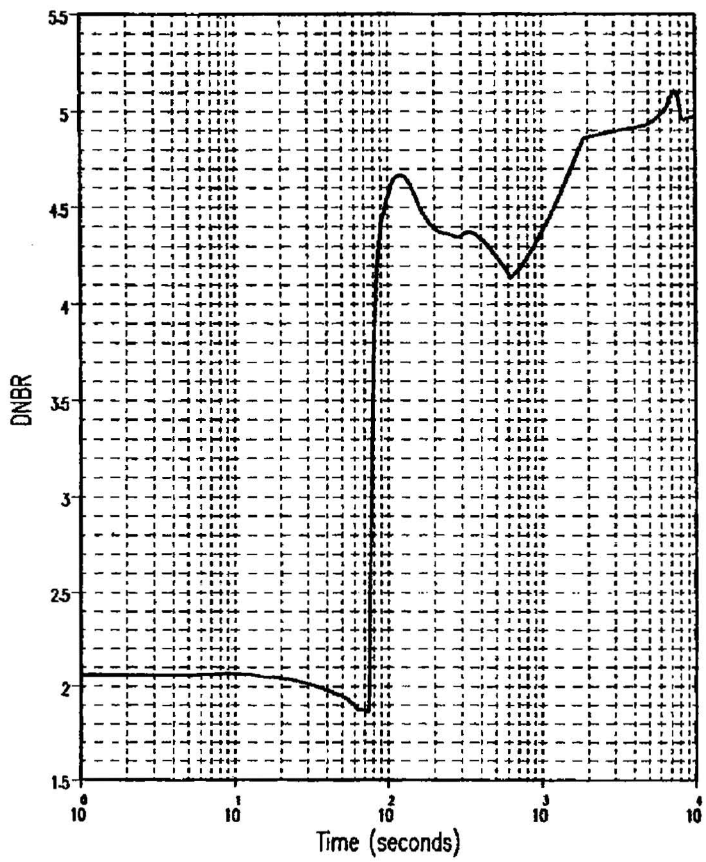


AMSAC/DSS:
LOSS OF ALL AC POWER TO THE STATION AUXILIARIES -
SG WIDE RANGE INDICATED LEVEL VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14819.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	FIGURE 14.8-19 REV. 31

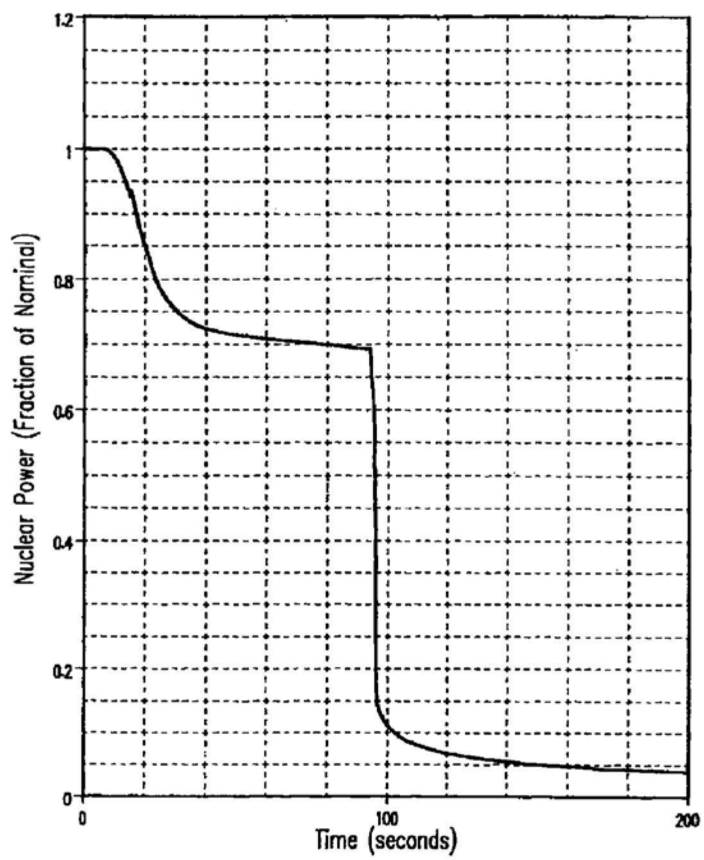
01183316

FIGURE 14.8-19



AMSAC/DSS:
LOSS OF ALL AC POWER TO THE STATION AUXILIARIES –
DNBR VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE	
CHECKED	CAD FILE U14820.DGN		*FIGURE 14.8-20 REV. 31*	

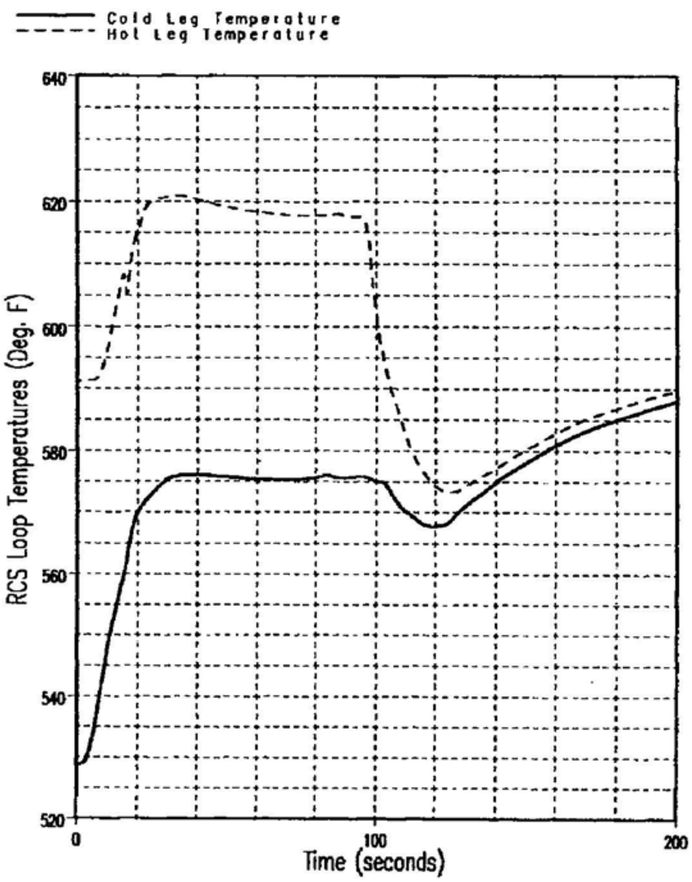


AMSAC/DSS:
LOSS OF EXTERNAL ELECTRICAL LOAD – NUCLEAR POWER VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14821.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-21 REV. 31*

01183316

FIGURE 14.8-21

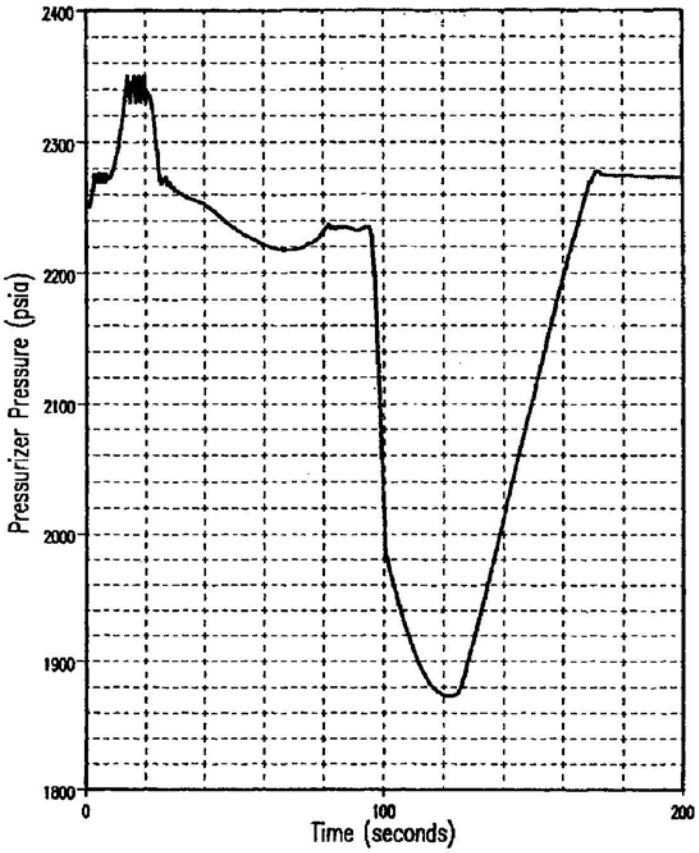


AMSAC/DSS:
LOSS OF EXTERNAL ELECTRICAL LOAD – REACTOR COOLANT LOOP TEMPERATURES VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14822.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-22 REV. 31*

0118316

FIGURE 14.8-22

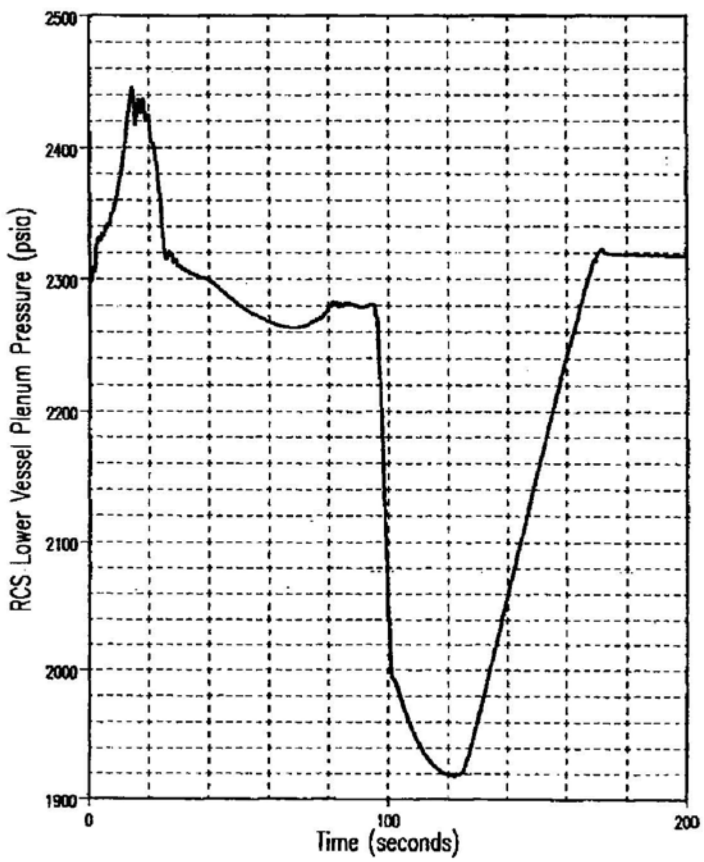


AMSAC/DSS:
LOSS OF EXTERNAL ELECTRICAL LOAD – PRESSURIZER PRESSURE VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE NONE
CHECKED	CAD FILE U14823.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	FIGURE 14.8-23 REV. 31

01183316

FIGURE 14.8-23

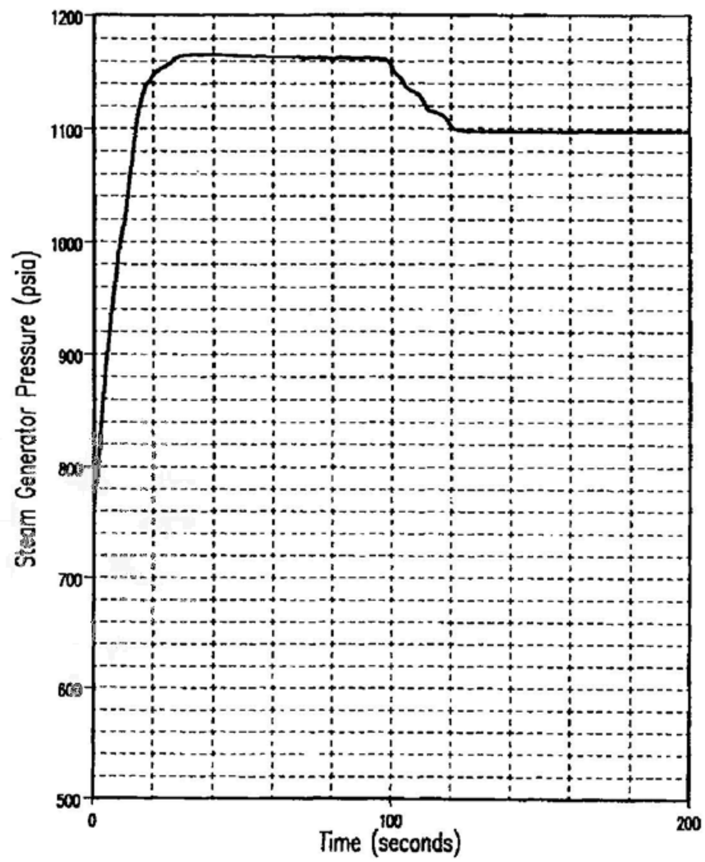


AMSAC/DSS:
LOSS OF EXTERNAL ELECTRICAL LOAD – RCS PRESSURE VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE UI4824.DGN		*FIGURE 14.8-24 REV. 31*

01183316

FIGURE 14.8-24

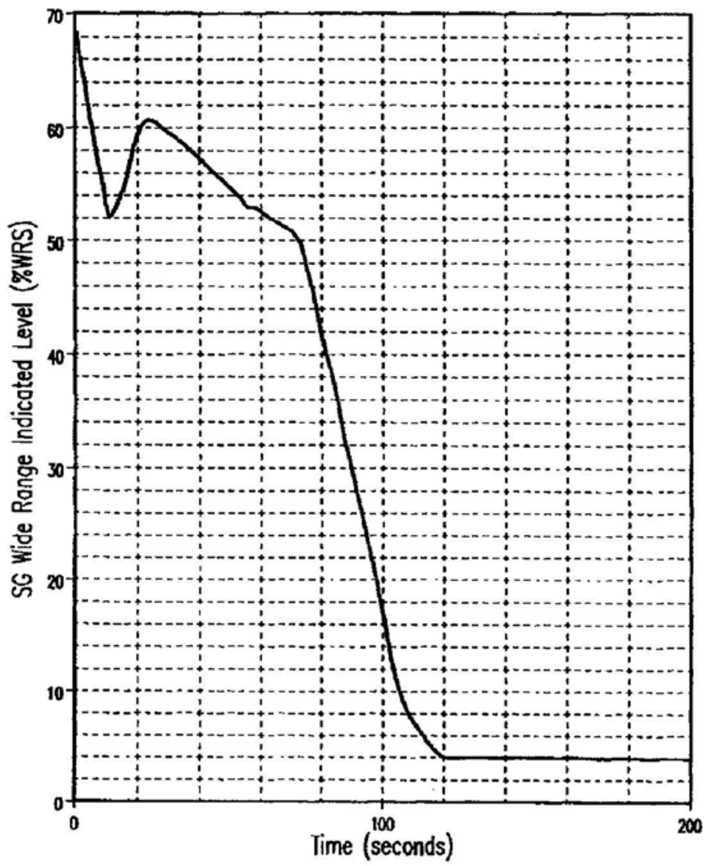


AMSAC/DSS:
LOSS OF EXTERNAL ELECTRICAL LOAD – STEAM GENERATOR PRESSURE VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14825.DGN		*FIGURE 14.8-25 REV. 31*

01183316

FIGURE 14.8-25

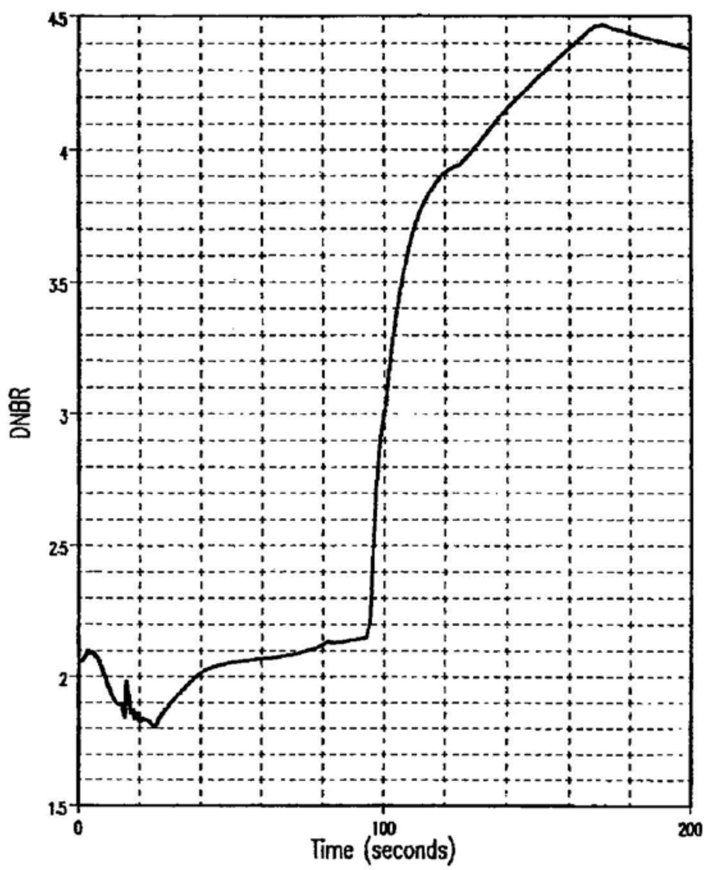


AMSAC/DSS:
LOSS OF EXTERNAL ELECTRICAL LOAD – SG WIDE RANGE INDICATED LEVEL VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14826.DGN		*FIGURE 14.8-26 REV. 31*

01183316

FIGURE 14.8-26

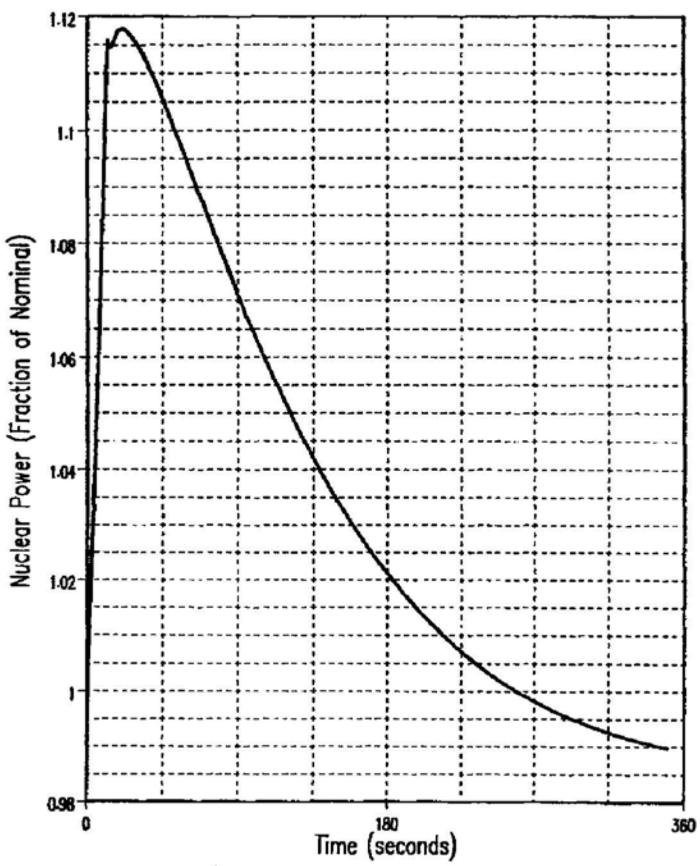


AMSAC/DSS:
LOSS OF EXTERNAL ELECTRICAL LOAD – DNBR VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE	
CHECKED	CAD FILE U14827.DGN		FIGURE 14.8-27 REV. 31	

FIGURE 14.8-27

01183316

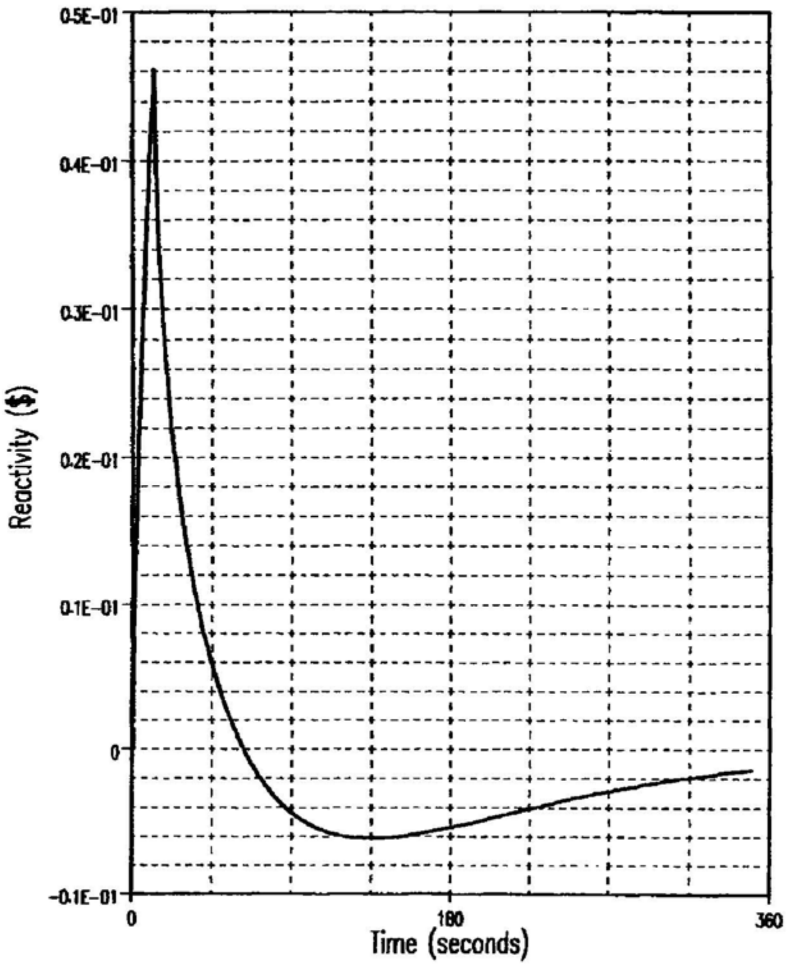


AMSAC/DSS:
UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER
NUCLEAR POWER VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14828.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-28 REV. 31*

01183316

FIGURE 14.8-28

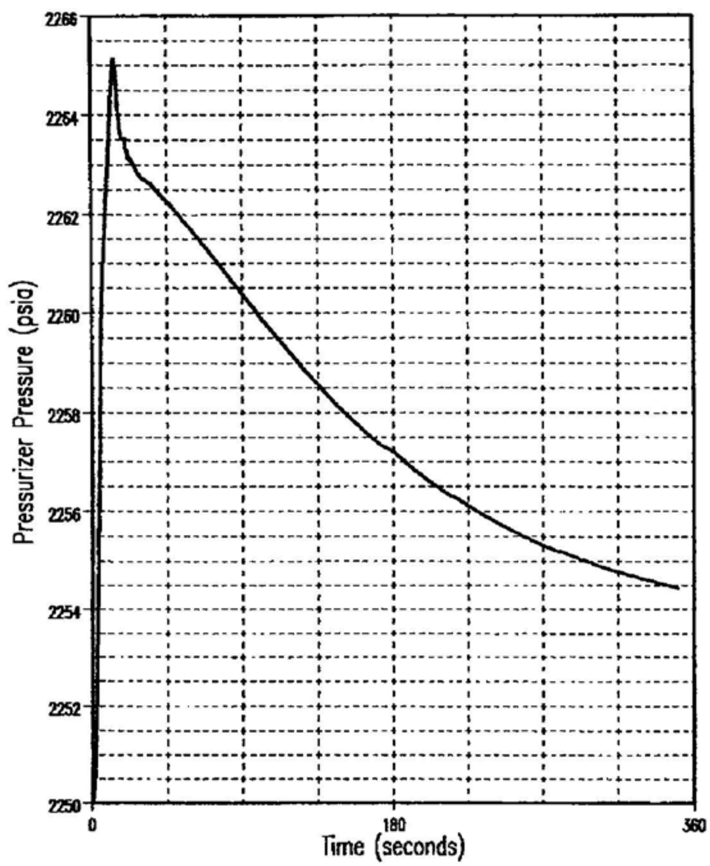


AMSAC/DSS:
UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER
CORE REACTIVITY VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14829.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	FIGURE 14.8-29 REV. 31

01183316

FIGURE 14.8-29

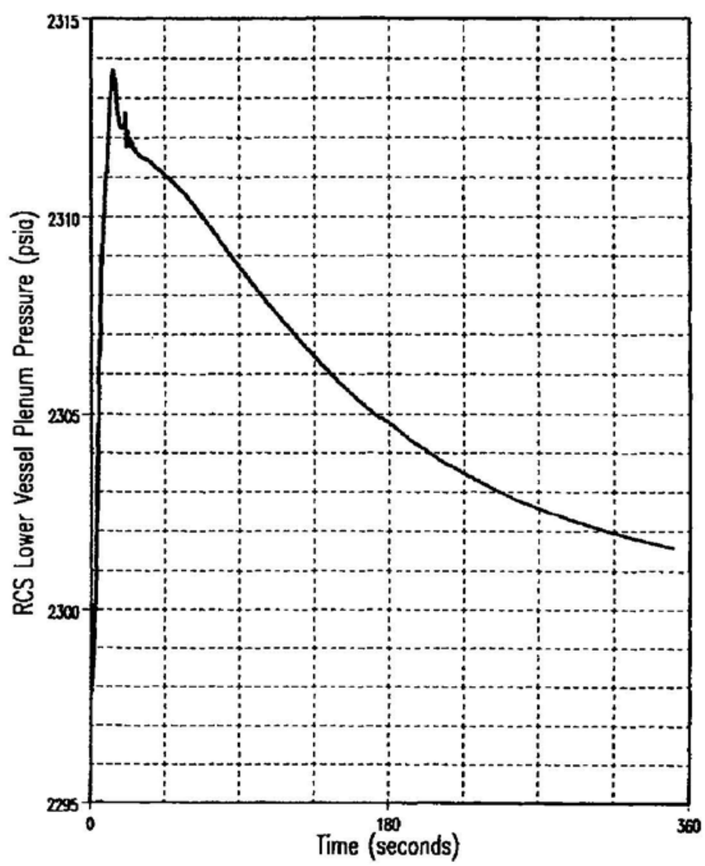


01183316

AMSAC/DSS:
UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER
PRESSURIZER PRESSURE VERSUS TIME

DWN	KJF	DATE	2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE:	NONE
CHECKED		CAD FILE	U14830.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-30 REV. 31*	

FIGURE 14.8-30

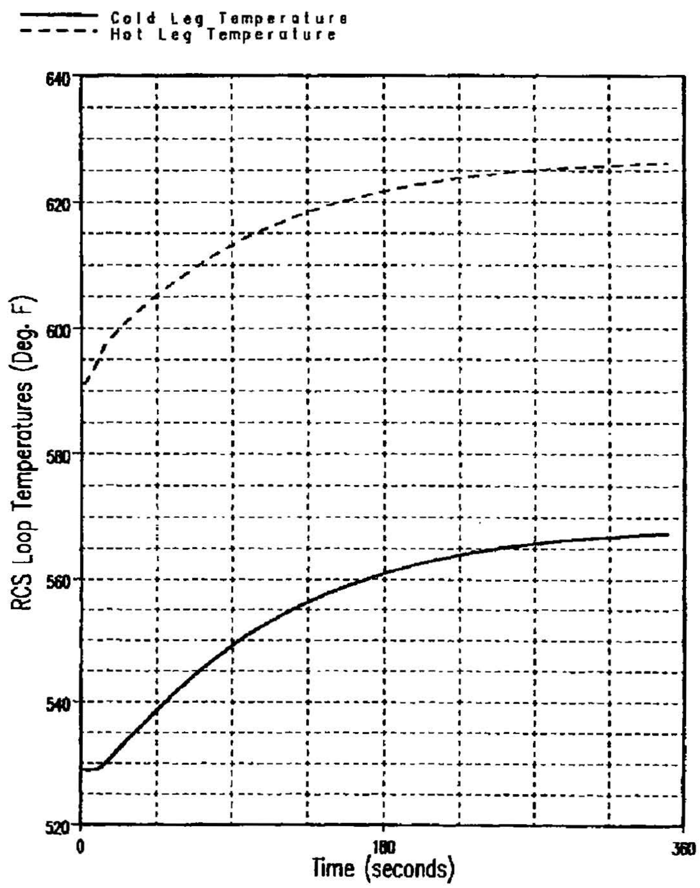


AMSAC/DSS:
UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER
RCS PRESSURE VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14831.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	FIGURE 14.8-31 REV. 31

01183316

FIGURE 14.8-31

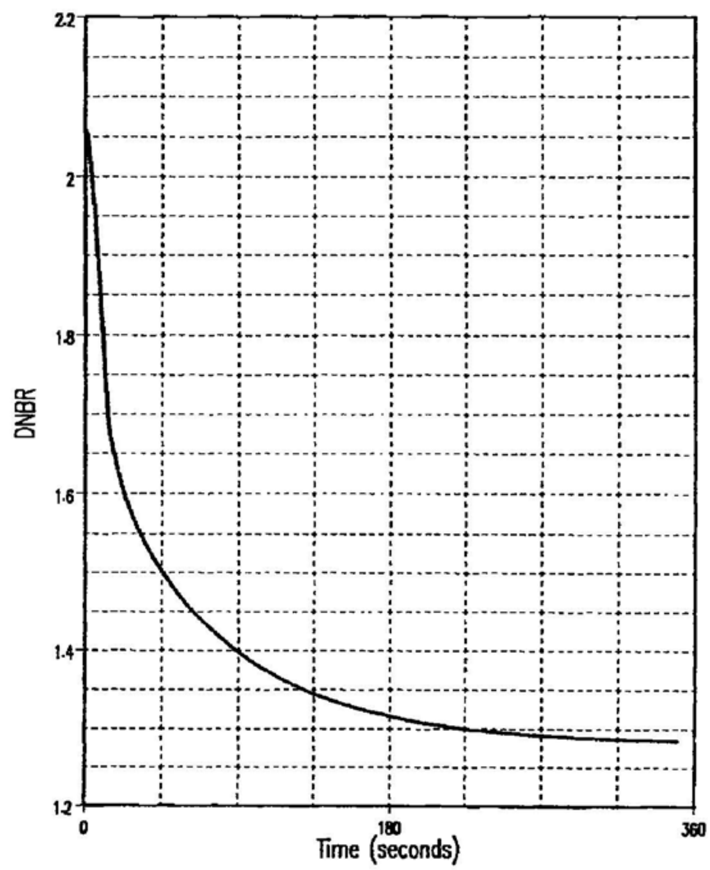


AMSAC/DSS:
UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER
RCS LOOP TEMPERATURES VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14832.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-32 REV. 31*	

01183316

FIGURE 14.8-32

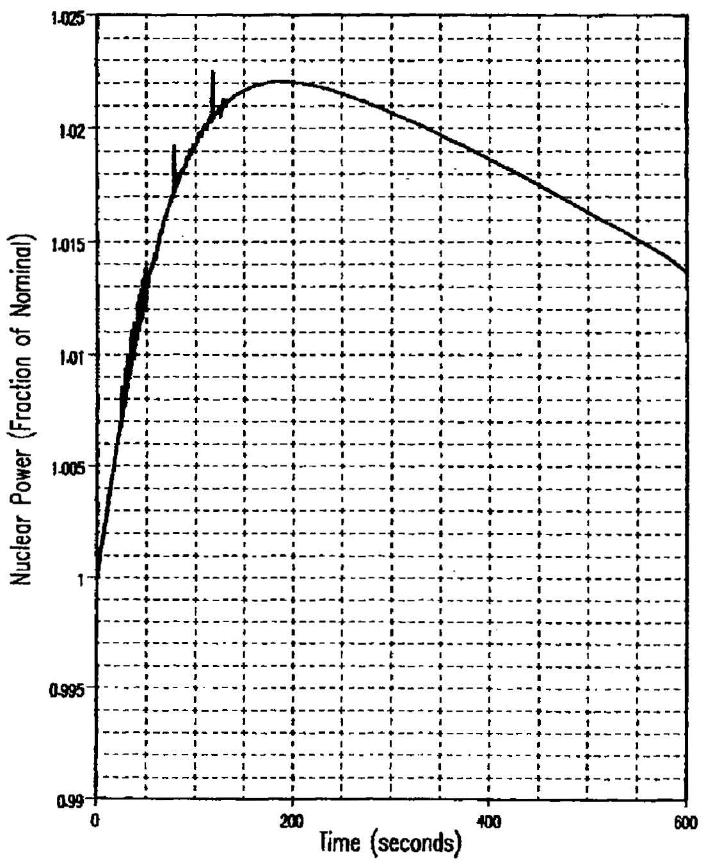


AMSAC/DSS:
UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER
DNBR VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14833.DGN		FIGURE 14.8-33 REV. 31

01183316

FIGURE 14.8-33

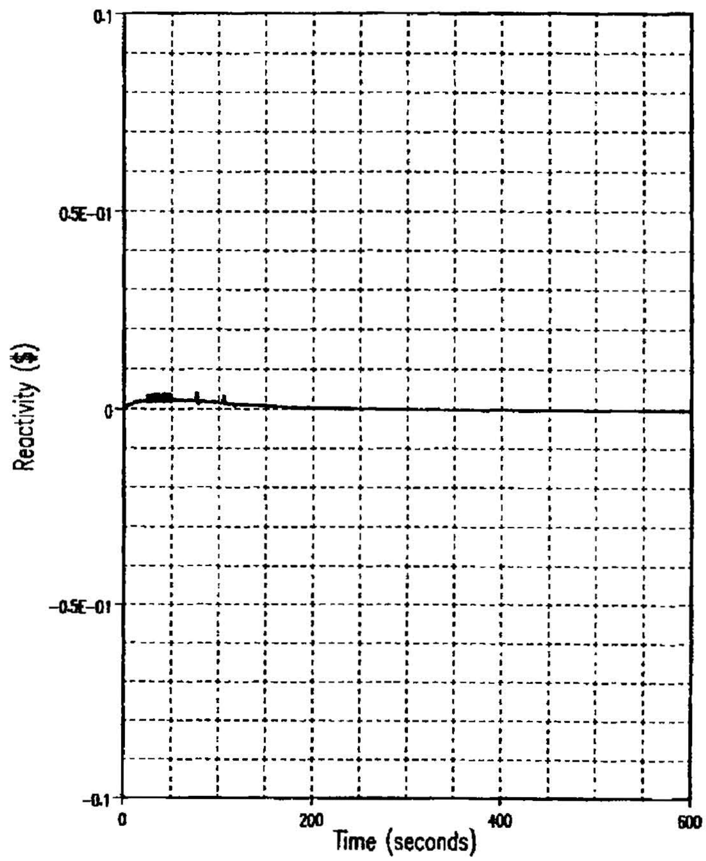


AMSAC/DSS:
UNCONTROLLED BORON DILUTION – NUCLEAR POWER VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED	CAD FILE U14834.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-34 REV. 31*	

01183316

FIGURE 14.8-34

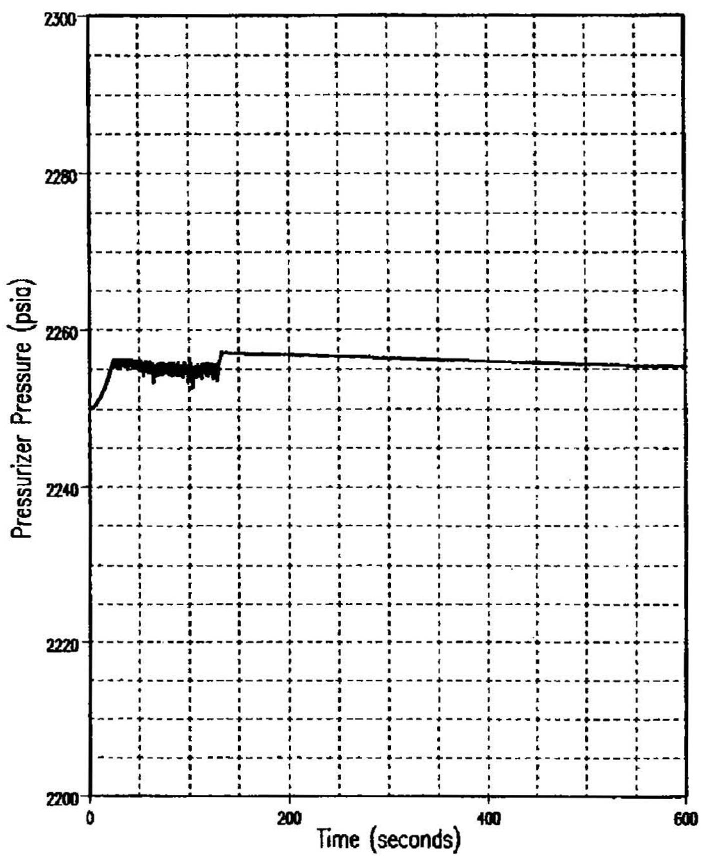


AMSAC/DSS:
UNCONTROLLED BORON DILUTION – CORE REACTIVITY VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE U14835.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-35 REV. 31*

01183316

FIGURE 14.8-35

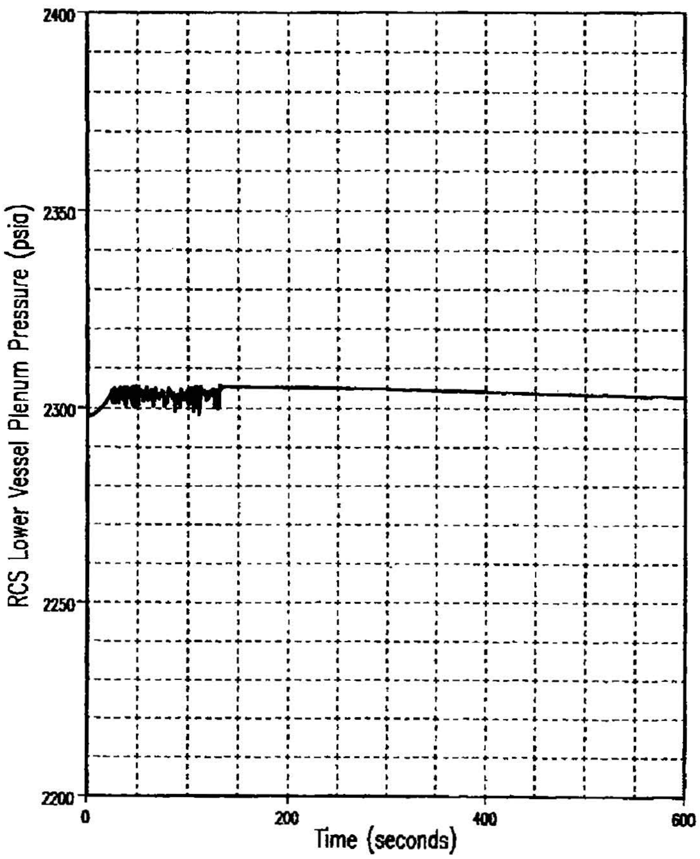


AMSAC/DSS:
UNCONTROLLED BORON DILUTION – PRESSURIZER PRESSURE VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE UI4836.DGN		*FIGURE 14.8-36 REV. 31*

01183316

FIGURE 14.8-36

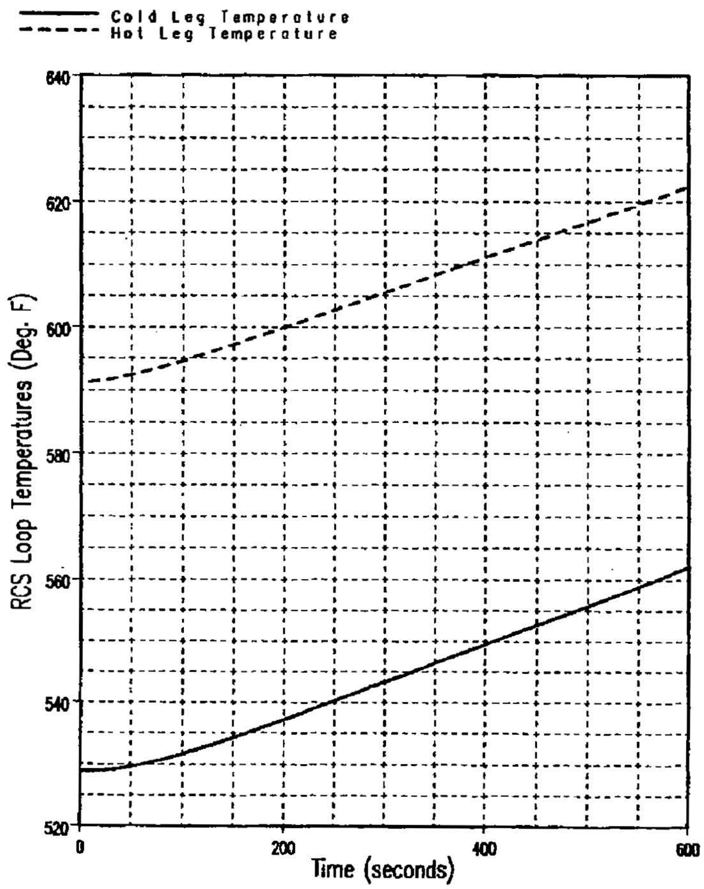


AMSAC/DSS:
UNCONTROLLED BORON DILUTION – RCS PRESSURE VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14837.DGN		*FIGURE 14.8-37 REV. 31*

01183316

FIGURE 14.8-37



AMSAC/DSS:
UNCONTROLLED BORON DILUTION – RCS LOOP TEMPERATURES VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14838.DGN		*FIGURE 14.8-38 REV. 31*

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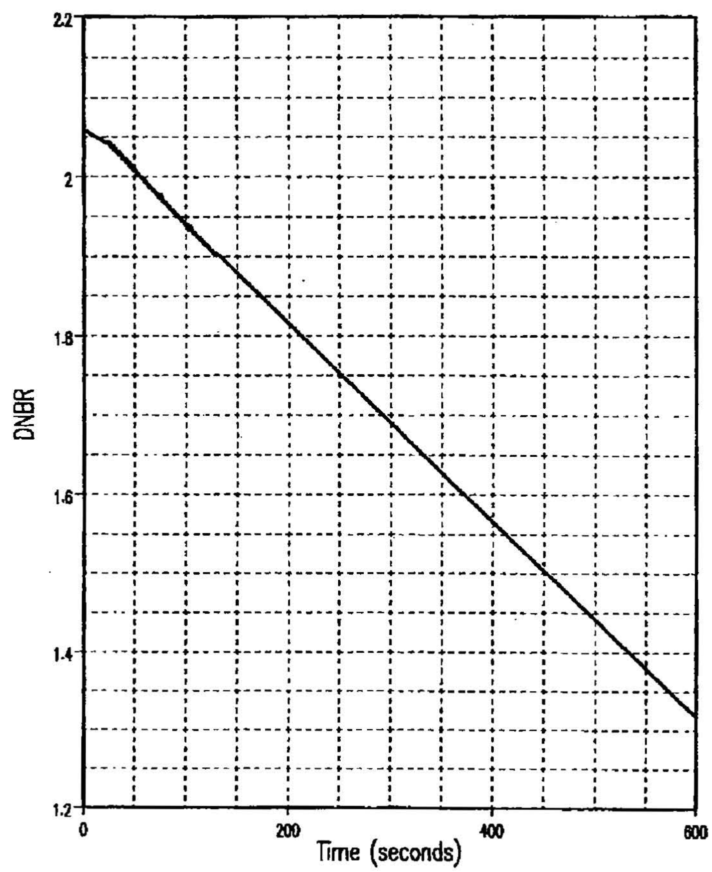
FIGURE 14.8-38



AMSAC/DSS:
UNCONTROLLED BORON DILUTION - DNBR VERSUS TIME

DWN KJF	DATE 2-08-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE NONE	
CHECKED	CAD FILE U14839.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 14.8-39 REV. 31*	

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FIGURE 14.8-39

Figure 14.8-40 DELETED
REV 31

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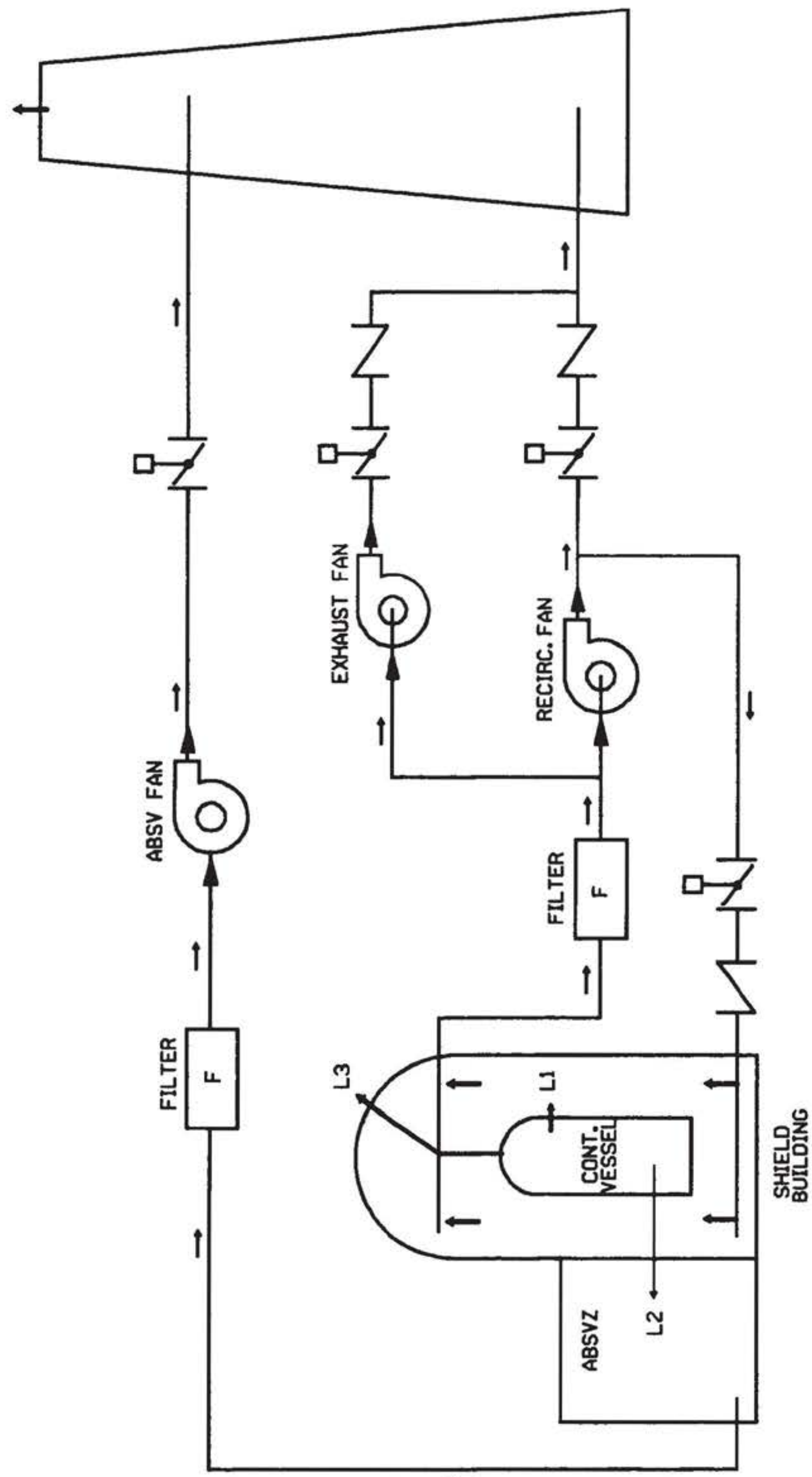

NOTE: ANALYTICAL FLOW MODEL FOR PRAIRIE ISLAND BASIS OFFSITE DOSE ANALYSIS

OFF-SITE DOSE ANALYSIS MODEL

DWN T. MILLER	DATE 6-23-99	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U14901.DGN		**FIGURE 14.9-1 REV. 18**

FIGURE 14.9-4, DELETED

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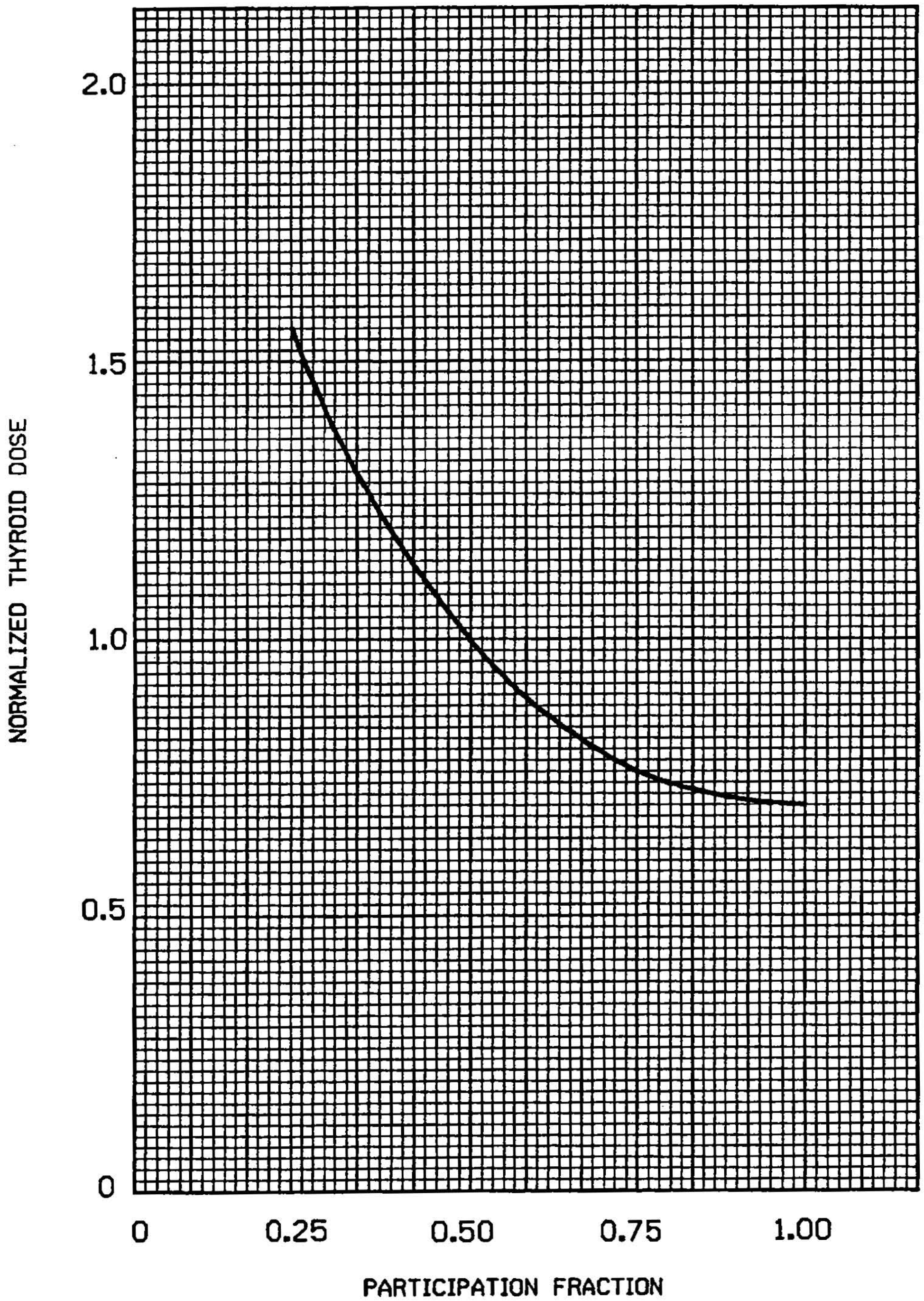

SENSITIVITY OF 2-HR THYROID DOSE TO
ANNULUS VOLUME PARTICIPATION FRACTION

DWN G.L.D	DATE 6-23-99	NORTHERN STATES POWER COMPANY	SCALE: NONE
CHECKED	CAD FILE U14905.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	FIGURE 14.9-5 REV. 18

Figure 14.10-1
has been
DELETED.

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APPENDIX 14A

OFFSITE DOSE CALCULATIONS FOR HIGH BURNUP FUEL

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