USAR APPENDIX K

CONTAINMENT PRESSURE RESPONSE TO LOCA

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APPENDIX K – CONTAINMENT PRESSURE RESPONSE TO LOCA

The uncontrolled release of pressurized high-temperature reactor coolant, termed a loss-of-coolant accident (LOCA), will result in release of steam and water into the containment. This, in turn, will result in increases in the global containment pressure and temperature. This appendix discusses containment issues relative to a postulated LOCA that must be considered for the Prairie Island Nuclear Power Plants.

The long-term LOCA mass and energy releases are analyzed and are utilized as input to the containment integrity analysis, which demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a hypothetical large-break LOCA. The containment safeguards systems must be capable of limiting the peak containment pressure to less than the design pressure and to limit the temperature excursion to less than the acceptance limits.

This analysis bounds both Units 1 and 2 with consideration of the two different steam generator designs, either 400V+ or 422V+ fuel, and corrections due to NSAL-11-05 (Reference 9). 01435911

K.1 Long-Term LOCA Mass and Energy Release to the Containment

The mass and energy release rates described in this section form the input to further computations to evaluate the containment conditions following the postulated accident. The long-term LOCA mass and energy releases for the hypothetical double-ended pump suction (DEPS) rupture with minimum safeguards and maximum safeguards and double-ended hot-leg (DEHL) rupture break cases are discussed in this section. These three LOCA cases are used for the long-term containment integrity analyses

K.1.1 Initial Conditions and Assumptions

The mass and energy release analysis is sensitive to the assumed characteristics of various plant systems in addition to other key modeling assumptions. Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. Some of the most critical items are the RCS initial conditions, core decay heat, safety injection flow, and primary and secondary metal mass and steam generator heat release modeling. Tables K-1 through K-3 present key data assumed in the analysis. The initial conditions and assumptions used in this analysis produced bounding mass and energy release to the containment at MUR conditions.

A total core power of 1683 MWt including calorimetric uncertainty was used in the analysis. RCS operating temperatures that bound the highest average coolant temperature range were used as bounding analysis conditions. Additionally, an allowance to account for instrument error and deadband is reflected in the initial RCS temperatures.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure. The RCS has a higher fluid density at the higher pressure (assuming a constant temperature) and subsequently has a higher RCS mass available for releases. Thus, 2,250 psia plus uncertainty was selected for the initial pressure as the limiting case for the long-term mass and energy release calculations.

The core stored energy selected to bound fuel products that will be used at Prairie Island is 5.77 full-power seconds (FPS). The margins in the core stored energy address the thermal fuel model and associated manufacturing uncertainties and the time in the fuel cycle for maximum fuel densification.

Margin in RCS volume of 3 percent (which is composed of 1.6-percent allowance for thermal expansion and 1.4-percent allowance for uncertainty) was modeled.

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A uniform steam generator tube plugging level of 0 percent was modeled. This assumption maximizes the reactor coolant volume and fluid release by virtue of consideration of the RCS fluid in all steam generator tubes. After blowdown, the steam generators are active heat sources since significant energy remains in the secondary metal and water mass. The 0-percent tube plugging assumption maximizes the heat transfer area and, therefore, the transfer of secondary energy across the steam generator tubes. Additionally, this assumption reduces the reactor coolant loop resistance, which reduces the ΔP upstream of the break for the pump suction breaks and increases break flow.

The secondary-to-primary heat transfer is maximized by assuming conservative heat transfer coefficients. Conservative energy transfer is ensured by maximizing the initial internal energy of the inventory in the steam generator secondary side. The internal energy is based on full-power operation plus uncertainties.

The mass and energy release calculation considered configurations/failures to conservatively bound respective alignments of the safety injection system. The cases include: a minimum safeguards case (one high-head safety injection [HHSI] and one low-head safety injection [LHSI] pumps) (see Table K-2); and a maximum safeguards case, (two HHSI and two LHSI pumps) (see Table K-3). In addition, during reflood the containment backpressure is assumed to be equal to the containment design pressure. This assumption was shown in WCAP-10325-P-A (Reference 1) to be conservative for the generation of mass and energy releases.

In summary, the following assumptions were employed to ensure that the mass and energy releases are conservatively calculated, thereby maximizing energy release to containment:

- Maximum expected operating temperature of the RCS (Tave = 560°F)
- Allowance for RCS temperature uncertainty (+4.0°F)
- Margin in RCS volume of 3 percent (which is composed of 1.6-percent allowance for thermal expansion, and 1.4-percent allowance for uncertainty)
- Core rated power of 1,677 MWt
- Allowance for calorimetric error (+0.36 percent of power)
- Conservative heat transfer coefficients (that is, steam generator primary/secondary heat transfer, and RCS metal heat transfer)
- Allowance in core stored energy for effect of fuel densification
- A margin in core stored energy (+15 percent to account for manufacturing tolerances)
- An allowance for RCS initial pressure uncertainty (+40. psi)
- A maximum containment backpressure equal to design pressure (46.0 psig)
- Steam generator tube plugging level (0-percent uniform)
  - Maximizes reactor coolant volume and fluid release
  - Maximizes heat transfer area across the steam generator tubes
  - Reduces coolant loop resistance, which increases break flow

The release of mass and energy from the RCS in the event of a LOCA is based on these conditions and assumptions.

All other inputs used in the analysis are from References 7 and 8.

K.1.2 LOCA Mass and Energy Release Phases

The mass and energy releases following a postulated LOCA continue over a time period, which is divided into four phases.

- Blowdown – the period of time from accident initiation to the time that the RCS and containment reach an equilibrium state.
- Refill – the period of time while the lower plenum is filled by accumulator and Emergency Core Cooling System (ECCS) water. At the end of blowdown, an amount of water remains in the cold legs, downcomer, and lower plenum. It is assumed that this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Thus, the refill period is conservatively neglected in the mass and energy release calculation.
- Reflood – begins when the water from the lower plenum enters the core and ends when the core is completely quenched.
- Post-reflood (FROTH) – describes the period following the reflood phase. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators prior to exiting the break as steam. After the broken loop steam generator cools, the break flow becomes two phase.

K.1.3 Description of Methodology

The evaluation model used for the long-term LOCA mass and energy release calculations is the March 1979 model described in WCAP-10325-P-A (Reference 1). The methodology employs the following codes to model the phenomena:

- SATAN78 – Blowdown Phase
- REFLOOD10325 – Refill, Reflood and Post-Reflood Phases
- EPITOME – Steam Generator Equilibriation

K.1.4 Break Size and Location

Generic studies have been performed with respect to the effect of postulated break size on the LOCA mass and energy releases. The double-ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the RCS loop can be postulated for a pipe rupture for mass and energy release purposes:

- Hot leg (between vessel and steam generator)
- Cold leg (between pump and vessel)
- Pump suction (between steam generator and pump)

The break locations analyzed for this program are the double-ended pump suction (DEPS) rupture (10.46 ft^2) and the double-ended hot leg (DEHL) rupture (9.154 ft^2).

The DEHL rupture has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be the highest for this break location, the amount of energy released from the steam generator secondary is minimal because the majority of the fluid that exits the core vents directly to containment bypassing the steam generators. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction or cold-leg break locations where the core exit mixture must pass through the steam generators before venting through the break. Therefore, only the mass and energy releases for the hot-leg break blowdown phase are calculated and presented in the LOCA mass and energy section of the report.

The cold-leg break location is less limiting in terms of the overall containment energy releases. The cold-leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. The cold-leg break is bounded by other breaks and no further evaluation is necessary.

The pump suction break combines the effects of the relatively high-core flooding rate, as in the hot-leg break, and the addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy rates during the post-blowdown period by including all of the available energy of the RCS in calculating the releases to containment.

Thus, the DEHL and DEPS cases are used to analyze long-term LOCA containment integrity.

K.1.5 Application of Single-Failure Criterion

An analysis of the effects of the single-failure criterion has been performed on the mass and energy release rates for each break analyzed. An inherent assumption in the generation of the mass and energy release is that offsite power is lost. Loss of offsite power results in the actuation of the emergency diesel generators, required to power the safety injection system. This is not an issue for the blowdown period, which is limited by the DEHL break.

Two cases have been analyzed to determine bounding mass and energy releases while addressing a single failure. The first case assumes minimum safeguards safety injection (SI) flow based on the postulated single failure of one emergency diesel generator, which results in the loss of one train of safeguards equipment. The second case assumes maximum safeguards SI flow consistent with the approved methodology described in WCAP-10325-P-A (Reference 1). In the maximum safeguards case, a single failure of the containment heat removal equipment is assumed in the containment response analysis. The analysis of the minimum and maximum safeguards cases for the DEPS break provides confidence that a bounding peak containment pressure is identified with consideration of credible single failures.

K.1.6 Decay Heat Model

On November 2, 1978, the Nuclear Power Plant Standards Committee (NUPPSCO) of the ANS-approved ANS Standard 5.1 (Reference 2) for the determination of decay heat. This standard was used in the mass and energy release model for Prairie Island. Table K-18 lists the decay heat curve used in the Prairie Island mass and energy release analysis.

Significant assumptions in the generation of the decay heat curve for use in the LOCA mass and energy releases analysis include the following:

- The decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.
- The decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.
- The fission rate is constant over the operating history of maximum power level.
- The factor accounting for neutron capture in fission products has been taken from *American National Standard for Decay Heat Power in Light Water Reactors* (Reference 2).
- The fuel has been assumed to be at full power for 10^8 seconds.
- The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.
- Two sigma uncertainty (two times the standard deviation) has been applied to the fission product decay.

Based upon NRC staff review, (Safety Evaluation Report [SER] of the March 1979 evaluation model [Reference 1]), use of the ANS Standard-5.1, November 1979 decay heat model was approved for the calculation of mass and energy releases to the containment following a LOCA.

K.1.7 Steam Generator Equilibration and Depressurization

Steam generator equilibration and depressurization is the process by which secondary-side energy is removed from the steam generators in stages. The froth part of the REFLOOD10325 computer code calculates the heat removal from the secondary mass until the secondary temperature is the T_{sat} at the containment design pressure. After the froth calculations, the EPITOME code continues the calculation for steam generator cooldown removing steam generator secondary energy at different rates (that is, first- and second-stage rates). The first-stage rate is applied until the steam generator reaches T_{sat} at the user-specified intermediate equilibration pressure, when the secondary pressure is assumed to reach the actual containment pressure. Then the second-stage rate is used until the final depressurization, when the secondary reaches the reference temperature of T_{sat} at 14.7 psia, or 212°F. The heat removal of the broken loop and intact loop steam generators are calculated separately.

K.1.8 Sources of Mass and Energy

The sources of mass considered in the LOCA mass and energy release analysis are the RCS, accumulators, and pumped SI.

The energy inventories considered in the LOCA mass and energy release analysis are from the following energy sources:

- RCS water
- Accumulator water
- Pumped SI water
- Decay heat
- Core-stored energy
- RCS metal (includes steam generator tubes)
- Steam generator metal (includes transition cone, shell, wrapper, and other internals)
- Steam generator secondary energy (includes fluid mass and steam mass)
- Secondary transfer of energy (feedwater into and steam out of the steam generator secondary)

The analysis used the following energy reference points:

- Available energy: 212°F; 14.7 psia (energy available that could be released)
- Total energy content: 32°F; 14.7 psia (total internal energy of the RCS)
- The mass and energy inventories are presented at the following times, as appropriate:
- Time zero (initial conditions)
- End of blowdown time
- End of refill time
- End of reflood time
- Time of broken loop steam generator equilibration to pressure setpoint
- Time of intact loop steam generator equilibration to pressure setpoint
- Time of full depressurization (3,600 seconds)

In the mass and energy release data presented, no Zirc-water reaction heat was considered because the cladding temperature does not rise high enough for the rate of the Zirc-water reaction heat to proceed.

K.1.9 Mass and Energy Release Analysis

K.1.9.1 Blowdown Mass and Energy Release

The SATAN78 code is used for computing the blowdown transient. The code utilizes the control volume (element) approach with the capability for modeling a large variety of thermal fluid system configurations. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. A point kinetics model is used with weighted feedback effects. The major feedback effects include moderator density, moderator temperature, and Doppler broadening. A critical flow calculation for subcooled (modified Zaloudek), two-phase (Moody), or superheated break flow is incorporated into the analysis. The methodology for the use of this model is described in WCAP-10325-P-A (Reference 1).

K.1.9.2 Reflood Mass and Energy Release Data

The WREFLOOD code is used for computing the reflood transient. The WREFLOOD code consists of two basic hydraulic models: one for the contents of the reactor vessel, and one for the coolant loops. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena, such as pumped SI and accumulators, reactor coolant pump (RCP) performance, and steam generator releases are included as auxiliary equations that interact with the basic models as required. The WREFLOOD code permits the capability to calculate variations during the core reflooding transient of basic parameters, such as core flooding rate, core and downcomer water levels, fluid thermo-dynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flow rates through the primary system. The code permits hydraulic modeling of the two flow paths available for discharging steam and entrained water from the core to the break; that is, the path through the broken loop and the path through the unbroken loops.

A complete thermal equilibrium mixing condition for the steam and ECCS injection water during the reflood phase has been assumed for each loop receiving ECCS water. This is consistent with the use and application of the M&E release evaluation model (Reference 1) in recent analyses, for example, D. C. Cook Docket (Reference 3). Even though the WCAP-10325-P-A (Reference 1) model credits steam/water mixing only in the intact loop and not in the broken loop, the justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (Reference 3). Moreover, this assumption is supported by test data and is further discussed below.

The model assumes a complete mixing condition (that is, thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of 2 distinct physical processes. The first is a two-phase interaction with steam condensation by cold ECCS water. The second is a single-phase mixing of condensate and ECCS water. Since the steam release is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that must be considered. (Any spillage directly heats only the sump.)

The most applicable steam/water mixing test data have been reviewed for validation of the containment integrity reflood steam/water mixing model. These data were generated in 1/3-scale tests (Reference 4) and are the largest scale data available and, thus, most clearly simulate the flow regimes and gravitational effects that would occur in a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

A group of 1/3-scale tests corresponds directly to containment integrity reflood conditions. The injection flowrates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in WCAP-10325-P-A (Reference 1). For all of these tests, the data clearly indicate the occurrence of very effective mixing with rapid steam condensation.

The mixing model used in the containment integrity reflood calculation is, therefore, wholly supported by the 1/3-scale steam/water mixing data.

Additionally, the following justification is also noted. The post-blowdown limiting break for the containment integrity peak pressure analysis is the pump suction double-ended rupture break. For this break, there are two flow paths available in the RCS by which M&E can be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the RCP. Steam that is not condensed by ECCS injection in the intact RCS loops passes around the downcomer and through the broken-loop cold leg and pump-in venting to containment. This steam also encounters ECCS injection water as it passes through the broken-loop cold leg, complete mixing occurs, and a portion of it is condensed. It is this portion of steam, which is condensed, that is credited in this analysis. Based upon the postulated break location and the actual physical presence of the ECCS injection nozzle, this assumption is justified. A description of the test and the test results are contained in References 1 and 4.

K.1.9.3 Post-Reflood Mass and Energy Release Data

The FROTH code (Reference 5) computes the post-reflood transient. The code calculates the heat release rates resulting from a two-phase mixture present in the steam generator tubes. The mass and energy releases that occur during this phase are typically superheated due to the depressurization and equilibration of the broken loop and intact loop steam generators.

During the post-reflood phase of the transient, the RCS pressure has equilibrated with the containment pressure. The steam generators are pressurized and contain a secondary inventory at an temperature that is much higher than the primary side. Therefore, there is significant heat transfer from the secondary system to the primary system. Steam is produced in the core due to core decay heat. For a pump suction break, a two-phase fluid exits the core, flows through the hot legs, and becomes superheated as it passes through the steam generator. Once the broken loop cools, the break flow becomes two phase. During the calculation, ECCS injection is addressed for both the injection phase and the recirculation phase. The calculation stops when the secondary side equilibrates to the saturation temperature (T_{sat}) at the containment design pressure, after this point the EPITOME code completes the steam generator depressurization.

This model is described in WCAP-10325-P-A (Reference 1). The mass and energy release rates are calculated by FROTH and EPITOME until the time of steam generator depressurization. After steam generator depressurization (14.7 psia), the mass and energy release available to containment is generated directly from core boil-off/decay heat.

K.2 LOCA Containment Response Analysis

The Prairie Island Nuclear Power Plant containment system is designed so that for all LOCA break sizes, up to and including the double-ended severance of a reactor coolant pipe, the containment peak pressure remains below the design pressure. This section details the containment response subsequent to a hypothetical LOCA. The containment response analysis uses the long-term LOCA mass and energy release data from subsection K.1 and the GOTHIC evaluation model described in reference 8.

The containment response analysis demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a LOCA inside containment. The impact of LOCA mass and energy releases on the containment pressure is addressed to ensure that the containment pressure remains below its design pressure at the licensed core power conditions. In support of equipment design and licensing criteria (for example, qualified operating life), with respect to post-accident environmental conditions, long-term containment pressure and temperature transients are generated to conservatively bound the potential post-LOCA containment conditions.

K.2.1 Accident Description

A break in the primary RCS piping causes a loss-of-coolant, which results in a rapid release of mass and energy to the containment atmosphere. Typically the blowdown phase for the large LOCA events (DEHL, cold leg, or pump suction pipe breaks) is over in less than 30 seconds. This large and rapid release of high-energy, two-phase fluid causes a rapid increase in the containment pressure, which results in the actuation of the emergency fan cooler and containment spray systems.

The RCS accumulators begin to refill the lower plenum and downcomer of the reactor vessel with water after the end of blowdown. The reflood phase begins after the vessel fluid level reaches the bottom of the fuel. During this phase, the core is quenched with water from both the accumulators and pumped SI. The quenching process creates a large amount of steam and entrained water that is released to containment through the break. This two-phase mixture would have to pass through the steam generators and also absorb energy from the secondary side coolant if the break were located in the cold leg or pump suction piping.

The LOCA mass and energy release decreases with time as the system cools. Core decay heat is removed by nucleate boiling after the reflood phase is complete. The long term core fluid level is maintained by pumping water into the vessel from the sump recirculation system. The containment heat removal systems continue to condense steam and slowly reduce the containment pressure and temperature over time.

K.2.2 Input Parameters and Assumptions

A series of analyses, using bounding break sizes and locations, was performed for the LOCA containment response. Subsection K.1 documented the mass and energy releases for the DEPS and DEHL breaks. The DEPS break cases were run with both minimum and maximum safeguards. The minimum safeguards case assumes a diesel train failure. This assumption leaves one of two containment spray pumps and two of four containment fan coil units (CFCUs) available as active heat removal systems. One Residual Heat Removal (RHR) pump and heat exchanger is modeled in the minimum SI case. For the maximum safeguards DEPS case, one train of sprays and one train of fan coolers were also modeled along with two RHR pumps and heat exchangers.

The containment initial conditions (pressure, temperature, and humidity) assumed for the M&E analyses are shown in Table K-24. Also, values for the initial temperature of the service water (SW) and refueling water storage tank (RWST) is presented. All of these values are chosen conservatively.

Table K-24 also includes the containment cooling system assumptions used in the M & E analysis. These include the containment spray (CS) pump flow rate, the CFCU heat removal performance curves, and the RHR System parameters. The CFCU heat removal performance curves assume that two CFCUs are operating at 90% rated capacity up to 240F air temperature. From 240.1F up to 270F, it is assumed that the two CFCUs are operating at 50% rated capacity (or one CFCU at 100% capacity) (Reference 11). This reduction accounts for the possibility of two phase flow developing in the CFCU piping. See reference 8 for the assumptions used in the GOTHIC recirculation model. 01496292

K.2.3 Description of the Prairie Island GOTHIC Containment Model

Calculation of the containment pressure and temperature is accomplished by use of the digital computer code GOTHIC. The Prairie Island Evaluation Model is retrievable through the analysis of record as documented in Reference 8. 01435911

K.2.4 Acceptance Criteria

The containment response for design-basis containment integrity is an American Nuclear Society (ANS) Condition IV event, an infrequent fault. The containment analysis methodology satisfies GDC Criterion 10 with consideration of GDC Criterion 41 as described in Section 1.5 of the USAR. To accomplish this, the peak containment pressure is verified to be less than the design limit of 46 psig for the worst case LOCA event. In addition, the peak liner temperature is verified less than 268°F.

K.2.5 Analysis Results

The sequence of events for the DEHL, DEPS-MIN, and DEPS-MAX cases are shown in Tables K-25 through K-27. The containment response calculations for the DEPS cases were performed for 172800 seconds (2 days). Since the steam generator secondary side energy is effectively isolated for hot leg breaks, the containment response calculation for the DEHL case is shown for the blowdown phase only (approximately 20 seconds).

The containment pressure, steam temperature, water (sump) temperature, and containment liner temperature profiles from each of the LOCA cases are shown in Figure K-13 through Figure K-16. Table K-28 summarizes the LOCA containment response results for the three cases studied.

K.2.5.1 Double-Ended Hot Leg Break

This analysis assumes a loss-of-offsite power coincident with a double-ended rupture of the RCS piping between the reactor vessel outlet nozzle and the steam generator inlet (that is, a break in the RCS hot leg). The associated single-failure assumption is the failure of a diesel to start, resulting in one train of safety injection and containment safeguards equipment being available. This combination results in a minimum set of safeguards being available. Furthermore, loss-of-offsite power delays the actuation times of the safeguards equipment due to the required diesel-startup time after receipt of the SI signal.

The Figures K-13 through K-16 show the containment pressure, atmosphere temperature, and sump temperature and liner temperature transients. The peak pressure for this case was 43.1 psig at 16 seconds. 01435911

K.2.5.2 Double-Ended Pump Suction Break with Minimum Safeguards

This analysis assumes a loss-of-offsite power coincident with a double-ended rupture of the RCS piping between the steam generator outlet and the RCS pump inlet (suction). The associated single-failure assumption is the failure of a diesel to start, resulting in one train of SI pumps and containment safeguards equipment being available. This combination results in a minimum set of safeguards being available. Furthermore, loss-of-offsite power delays the actuation times of the safeguards equipment due to the required diesel-startup time after receipt of the SI signal.

Figures K-13 through K-16 show the containment pressure, temperature, sump temperature and liner temperature transients. The peak pressure of 45.3 psig occurs after reflood at 3600 seconds. 01435911

K.2.5.3 Double-Ended Pump Suction Break with Maximum Safeguards

This analysis assumes a loss-of-offsite power coincident with a double-ended rupture of the RCS piping between the steam generator outlet and the RCS pump inlet (suction). The associated single-failure assumption is the failure of a train of containment safeguards being available. Furthermore, loss-of-offsite power delays the actuation times of the safeguards equipment due to the required diesel-startup time after receipt of the SI signal.

The DEPS break with maximum safeguards has a transient history similar to the minimum safeguards case discussed in subsection K.2. Figures K-13 through K-16 provides the containment pressure, steam temperature, sump temperature and liner temperature. Table K-28 shows that a peak pressure of 41.4 psig at 3602 seconds was calculated. 01435911

K.2.6 Conclusions

The LOCA containment response analyses have been performed for Prairie Island. The analyses produced long-term pressure and temperature profiles for each case. The peak containment pressure was less than 46 psig for all cases. The containment liner temperature was less than 268°F in all cases. Based on the results, all applicable containment integrity acceptance criteria for Prairie Island have been met.

References

1. WCAP-10325-P-A (Proprietary), WCAP-10326-A (Nonproprietary), Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version, May 1983.

2. ANSI/ANS-5.1 1979, American National Standard for Decay Heat Power in Light Water Reactors, August 1979.

3. Docket No. 50-315, “Amendment No. 126, To Facility Operating License No. DPR-58 (TAC No. 71062), for D. C. Cook Nuclear Plant Unit 1,” June 9, 1989. [Adams Accession Number ML021050051] 01435911

4. WCAP-8423, EPRI 294-2, Mixing of Emergency Core Cooling Water with Steam; 1/3-Scale Test and Summary, Final Report, June 1975.

5. WCAP-8264-P-A, Rev. 1, August 1975 (Proprietary), WCAP-8312-A (Nonproprietary), Westinghouse Mass and Energy Release Data for Containment Design.

6. DELETED

7. WCAP 16206-P, Safety Analysis Transition Program Engineering Report for the Prairie Island Nuclear Power Plant, Volume 1 Engineering Analysis, February 2004.

8. WCAP-16219-P Development and Qualification of a GOTHIC Containment Evaluation Model for the Prairie Island Nuclear Generating Plants, Ofstun R., April 2004

9. NSAL-11-5, “Westinghouse LOCA Mass and Energy Release Calculation Issues”, July 2011. 01435911

10. NRC/RCPL-05-138, Letter from Herbert N. Berkow (NRC) to James A. Gresham (Westinghouse) Subject- Acceptance of Clarifications of Topical Report WCAP-10325-P-A “Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version “(TAC No MC7980), October 18, 2005. [Adams Accession Number ML052660242]

11. CN-CRA-03-85, Rev. 0, Development and Benchmark Testing of a Single Volume GOTHIC Containment Model to Perform the RSG MSLB DBA Analyses for Prairie Island Units 1 and 2, November 2003. 01496292

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Table K-1
System Parameters Initial Conditions

Parameters	Value
Core Thermal Power (MWt)	1683
RCS Flow Rate (gpm / loop)	89,000
Vessel Outlet Temperature (°F)	592.6 (2)
Core Inlet Temperature (°F)	527.4
Initial Steam Generator Steam Pressure (psia)	796
Steam Generator Design (1)	Framatome 56/19
Steam Generator Tube Plugging (%)	0
Initial Steam Generator Secondary Side Mass (lbm)	130711
Assumed Maximum Containment Backpressure (psia)	60.7
Accumulator Water volume (ft^3) per accumulator N_2 cover gas pressure (psig) Temperature (°F)	 1,270 710 120

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Note:
Steam generator secondary side mass include appropriate uncertainty and/or allowance (See WCAP 16206-P).

(1) The Framatome Model 56/19 SGs bound the Westinghouse Model 51 Steam Generators.

(2) Reference 7 lists this as 599.8°F which is actually the core exit temperature. The vessel outlet temperature is slightly lower due to bypass flow documented in Westinghouse document PCWG-06-44.

Table K-2
SI Flow Minimum Safeguards
(For M&E Calculations)

RCS Pressure (psig)	Total Flow (lbm/sec)
Injection Mode (reflood phase)	
0	308.80
20	290.20
40	270.00
60	246.90
80	220.00
100	187.70
120	141.00
140	84.80
160	84.40
180	83.90

Recirculation Sequence		
	Vessel Injection (lbm/sec)	
Time (sec) [(1)]	**from RWST**	**from Sump**
1200	87.9	0
1440	87.9	207.3
2520	0	207.3
10000	0	207.3
50000	0	207.3
82500	0	207.3
118500	0	207.3
1000000	0	207.3

(1) The times for initiation of SI Recirculation are based on reasonable estimates for the shortest time before initiation of recirculation.

Table K-3
SI Flow Maximum Safeguards
(For M&E Calculations)

RCS Pressure (psig)	Total Flow (lbm/sec)
Injection Mode (reflood phase)	
0	753.80
20	713.40
40	669.70
60	625.30
80	577.30
100	522.60
120	460.60
140	376.80
160	225.30
180	161.60

Recirculation Sequence		
Time (sec) (1)	Vessel Injection (lbm/sec)	
	from RWST	from Sump
600	376.9	0
840	376.9	277.9
1440	0	277.9
1680	0	555.8
10000	0	555.8
100000	0	555.8
1000000	0	555.8

(1) The times for initiation of SI Recirculation are based on reasonable estimates for the shortest time before initiation of recirculation.

**TABLES K-4 THROUGH K-19
AND TABLE K-23
HAVE BEEN
DELETED.**

Table K-24

LOCA Containment Response Analysis Parameters
Page 1 of 2

Service Water Temperature (°F)				95
RWST Water Temperature (°F)				120
Initial Containment Temperature (°F)				120
Initial Containment Pressure (psia)				16.7
Initial Relative Humidity (%)				30
Net Free Volume (ft^3)				1.32×10^6
CFCU				
Total				4
Analysis Maximum				2
Analysis Minimum				2
Containment High Setpoint (psig)				5.00
Delay Time (sec)				60
Heat Removal as a Function of Steam Temperature	Vap Sat Temp (°F)	Heat Removal per FCU[1] (Btu/hr x 10^6)	# of FCU times efficiency	Heat Removal in Analysis (Btu/s)
	0.0	0	2 * 0.9 = 1.8	0.0
	100.0	1	2 * 0.9 = 1.8	500
	120.0	4	2 * 0.9 = 1.8	2000
	140.0	9	2 * 0.9 = 1.8	4500
	160.0	12.5	2 * 0.9 = 1.8	6250
	180.0	16.5	2 * 0.9 = 1.8	8250
	200.0	22	2 * 0.9 = 1.8	11000
	220.0	28	2 * 0.9 = 1.8	14000
	240.0	35.5	2 * 0.9 = 1.8	17750
	240.1	35.5	2 * 0.5 = 1	9861
	260.0	45	2 * 0.5 = 1	12500
	265.0	47.5	2 * 0.5 = 1	13194
	270.0	49	2 * 0.5 = 1	13611
	270.1	0	0	0.0
	300.0	0	0	0.0
Containment Spray Pumps				
Total				2
Analysis Maximum				1
Analysis Minimum				1
Flow Rate (gpm)	Injection Phase (per pump)			1200
	Recirculation Phase			0
Containment High-High Setpoint (psig)				24.0
Delay time (sec)				72

Note 1:Heat removal per FCU does not necessarily reflect the performance of the currently installed FCUs but rather is the FCU performance assumed in this analysis.

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Table K-24

LOCA Containment Response Analysis Parameters
Page 2 of 2

Recirculation Switchover, Full Flow Established, (sec) Minimum Safeguards Maximum Safeguards	 1440 1680
CS Termination Time, (sec)	1440
RHR System	
RHR Heat Exchangers	
Modeled in Analysis Minimum SI Maximum SI	 1 2
RHR Flows through RHR Heat Exchangers	
Minimum Safeguards Time (sec) 0.0 1439 1440 172,800	 Flow (lbm/s) 0.0 0.0 207.3 207.3
Maximum Safeguards Time (sec) 0.0 839 840 1679 1680 172,800	 Flow (lbm/s) 0.0 0.0 277.9 277.9 555.8 555.8
CCW Flow per RHR Heat Exchanger – lbm/s	305.5
OCW Heat Exchangers	
Modeled in Analysis Minimum SI Maximum SI	 1 2
CCW Flow (lbm/s)	305.5
Service Water Flow per CCW Heat Exchanger (lbm/s)	275
Additional Heat Loads per CCW Heat Exchanger, Btu/hr	500,000

Table K-25

DEHL Break Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power Are Assumed
0.35	Containment High Pressure Setpoint of 5.0 psig Reached
0.38	Compensated Pressurizer Pressure for Turbine Trip – 1,850 psia Reached
3.51	Containment High-High Pressure Setpoint of 23.0 psig Reached
3.57	Low-Pressureizer Pressure SI Setpoint – 1,700 psia Reached
6.65	Broken Loop Accumulator Begins Injecting Water
6.70	Intact Loop Accumulator Begins Injecting Water
15.04	Peak Containment Steam Temperature Occurs (265.3°F)
16.04	Peak Containment Pressure Occurs (43.07 psig)
16.8	End of Blowdown Phase
16.8	Accumulator Mass Adjustment for Refill Period
16.8	Transient Modeling Terminated

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Table K-26

DEPS Break Sequence of Events (Minimum Safeguards)

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power Are Assumed
0.34	Containment High Pressure Setpoint of 5.0 psig Reached
0.44	Compensated Pressurizer Pressure for Turbine Trip – 1,850 psia Reached
3.64	Low-Pressurizer Pressure SI Setpoint – 1,700 psia Reached
3.66	Containment High-High Pressure Setpoint of 24.0 psig Reached
6.92	Broken Loop Accumulator Begins Injecting Water
7.03	Intact Loop Accumulator Begins Injecting Water
14.4	End of Blowdown Phase
14.4	Accumulator Mass Adjustment for Refill Period
19.8	Pumped SI Begins after 20-Second Diesel Delay
38.4	Intact Loop Accumulator Water Injection Ends
39.3	Broken Loop Accumulator Water Injection Ends
60.34	Emergency CFCUs Heat Removal Begins
75.74	CS Pump (from RWST) Begins
205.02	End of Reflood Phase
1,200	Recirculation Sequence Begins – Low-Head Pump Stopped and HHSI continues to pump from RWST
1,381.6	Mass & Energy Release Assumption: Broken Loop Steam Generator Equilibration
1,440	Containment Spray Terminated
1,440	LH Flow from Sump Begins
1,620.7	Mass & Energy Release Assumption: Intact Loop Steam Generator Equilibration
2,520	HHSI Pump Stopped
3,600	Mass & Energy Release Assumption: Both Steam Generators Equilibrate to 14.7 psia
3,600	Peak Containment Pressure Occurs
86,400	Containment Steam Temperature at 24 hours = 196°
172,800	Transient Modeling Terminated

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Table K-27

DEPS Break Sequence of Events (Maximum Safeguards)

Time (sec)	Event Description
0	Break Occurs, Reactor Trip and Loss of Offsite Power Are Assumed
0.34	Containment High Pressure Setpoint of 5.0 psig Reached
0.44	Compensated Pressurizer Pressure for Turbine Trip – 1,850 psia Reached
3.64	Low-Pressureizer Pressure SI Setpoint – 1,700 psia Reached
3.66	Containment High-High Pressure Setpoint of 24.0 psig Reached
6.92	Broken Loop Accumulator Begins Injecting Water
7.03	Intact Loop Accumulator Begins Injecting Water
14.4	End of Blowdown Phase
14.4	Accumulator Mass Adjustment for Refill Period
19.8	Pumped SI Begins after 20-Second Diesel Delay
38.6	Intact Loop Accumulator Water Injection Ends
39.5	Broken Loop Accumulator Water Injection Ends
60.32	Emergency CFCUs Heat Removal Begins
75.73	CS Pump (from RWST) Begins
165.32	End of Reflood Phase
600	Recirculation Sequence Begins – Train 1 of SI Pumps Stopped – Train 2 continues pumping from RWST
840	Recirc Flow from Sump Begins from Train 1
1032.71	Mass & Energy Release Assumption: Broken Loop Steam Generator Equilibration
1,440	Containment Spray (from RWST) Terminated
1,440	2nd Train of SI Pumps from RWST Stopped
1,474.51	Mass & Energy Release Assumption: Intact Loop Steam Generator Equilibration
1,680	Recirc Flow from Sump Begins from Train 2
3602	Peak Containment Pressure Occurs 41.42 psig
86,400	Containment Gas Temperature at 24 hours = 188.4°F
172,800	Transient Modeling Terminated

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Table K-28

LOCA Containment Integrity Response Results (Loss-of-Offsite Power Assumed)

Case	Peak Press (psig)	Peak Gas Temp. (°F)	Peak Liner Temp (°F)	Pressure (psig) @ 24 hours	Temperature (°F) @ 24 hours
DEHL	43.1@ 16.1 sec	265.3 @ 15.0 sec	196.2 @ 16.8 sec	Not Applicable	Not Applicable
DEPS Min SI	45.3 @ 3600 sec	268.0 @ 1622 sec	264.8 @ 3602 sec	11.4 @ 86,400 sec	193 @ 86,400 sec
DEPS Max SI	41.4 @ 3602 sec	261.3 @ 795.2 sec	259.2 @ 3602.2 sec	10.8 @ 86,400 sec	188.4 @ 86,400 sec

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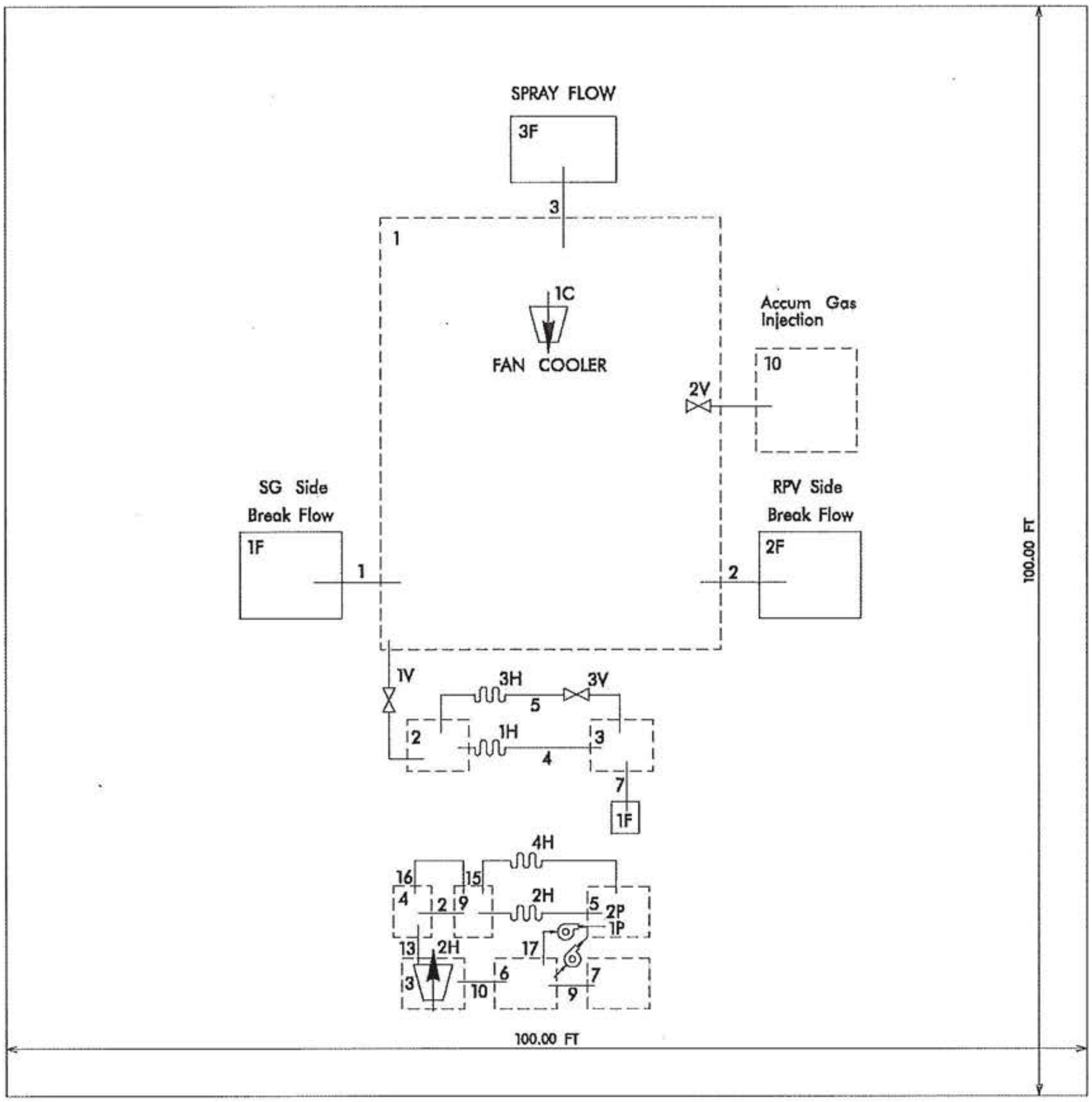

PRARIE ISLAND
LONG–TERM LOCA GOTHIC CONTAINMENT MODEL
(SEE REFERENCE 8 FOR GOTHIC RECIRCULATION MODELING)

DWN KJF	DATE 3-23-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE UK01.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE K01 REV. 31*

FIGURE K-1

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Figures K-2 through K-12

have been

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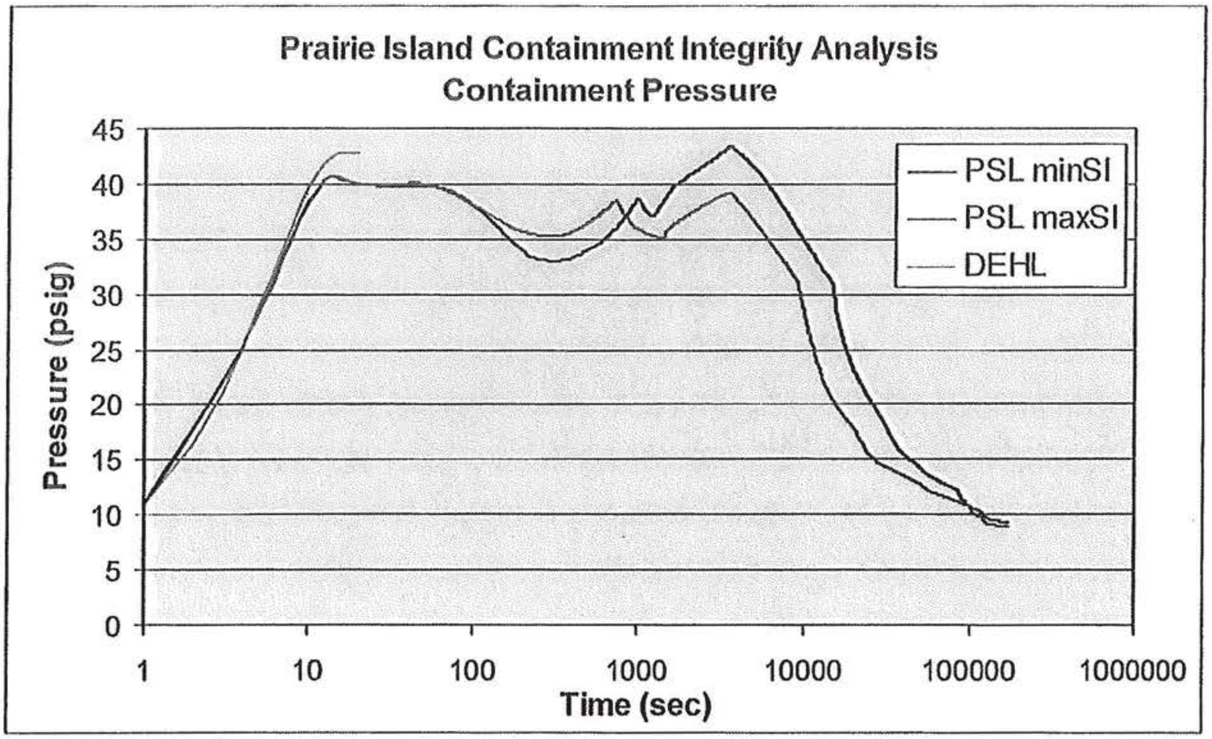
FIGURE K-13

LOCA CONTAINMENT INTEGRITY ANALYSIS: CONTAINMENT PRESSURE RESPONSE

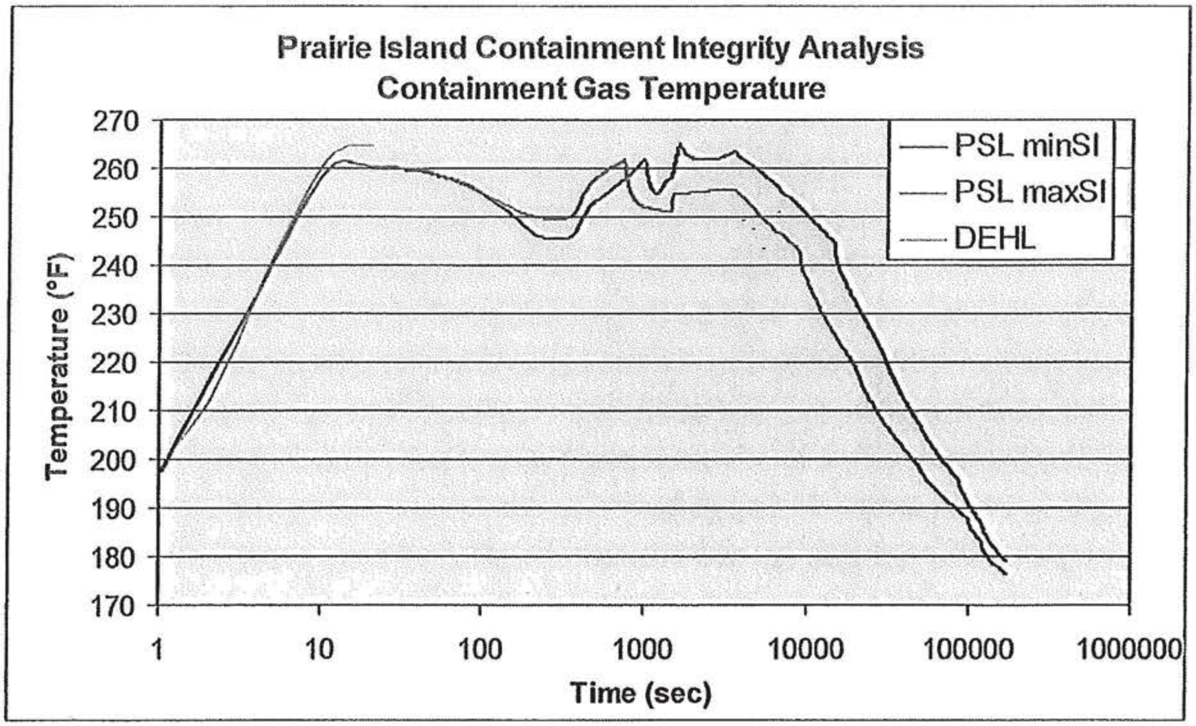


PRAIRIE ISLAND LOCA CONTAINMENT INTEGRITY ANALYSIS: CONTAINMENT PRESSURE RESPONSE			
DWN KJF	DATE 3-23-10	NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE UKI3.DGN		*FIGURE K13 REV. 31*

FIGURE K-13

FIGURE K-14

LOCA Containment Integrity Analysis: Containment Gas Temperature Response



PRAIRIE ISLAND
LOCA CONTAINMENT INTEGRITY ANALYSIS:
CONTAINMENT GAS TEMPERATURE RESPONSE

DWN KJF	DATE 3-23-10	NORTHERN STATES POWER COMPANY XcelEnergy	SCALE: NONE
CHECKED	CAD FILE UK14.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE K14 REV. 31*

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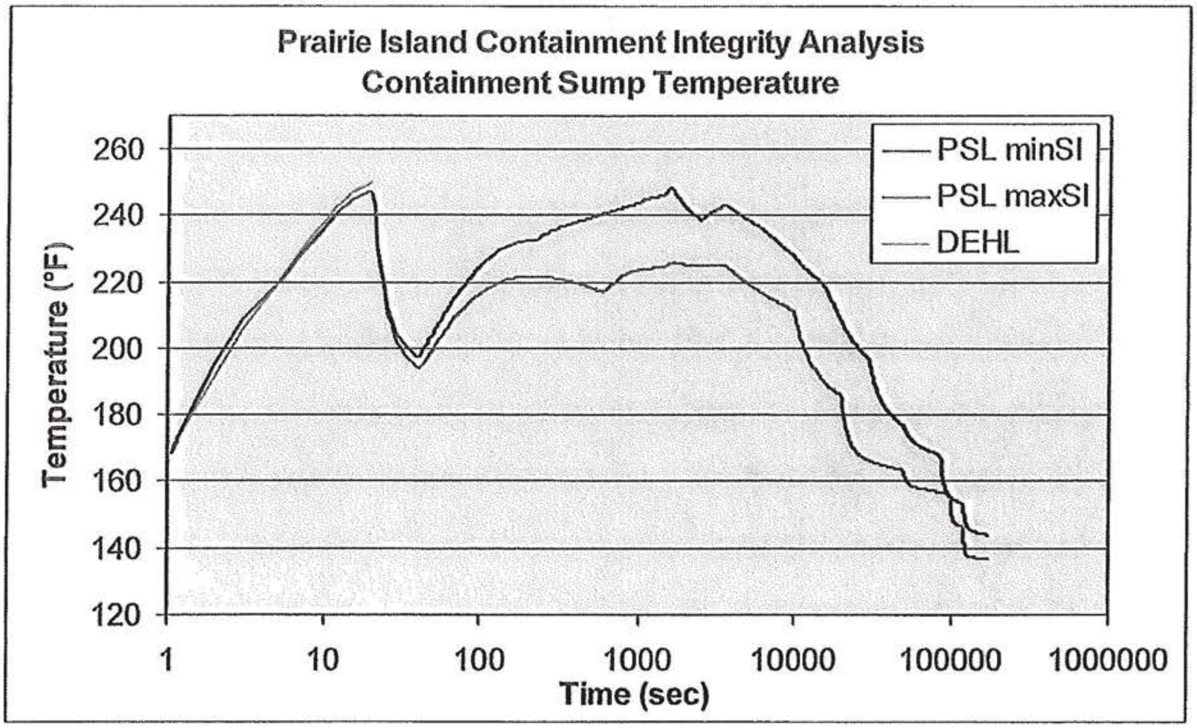
FIGURE K-15

LOCA Containment Integrity Analysis
Containment Sump Temperature



LOCA CONTAINMENT INTEGRITY ANALYSIS:
CONTAINMENT SUMP TEMPERATURE

DWN KJF	DATE 3-23-10	NORTHERN STATES POWER COMPANY XcelEnergy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE: NONE
CHECKED	CAD FILE UKI5.DGN		*FIGURE K15 REV. 31*

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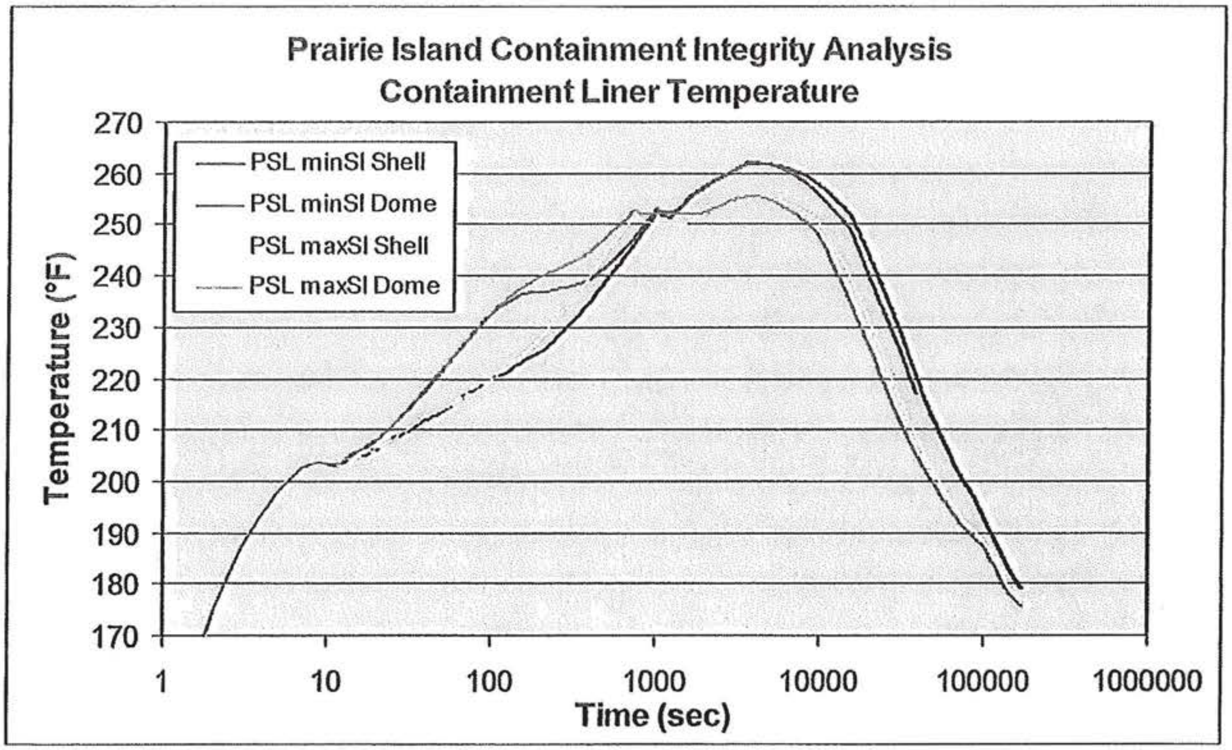
FIGURE K-16

LOCA Containment Integrity Analysis: Containment Liner Temperature



PRAIRIE ISLAND LOCA CONTAINMENT INTEGRITY ANALYSIS: CONTAINMENT LINER TEMPERATURE			
DWN KJF	DATE 3-23-10	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE
CHECKED	CAD FILE UK16.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE K16 REV. 31*