Nebraska Public Power District
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NLS2023051
October 10, 2023

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Response to Nuclear Regulatory Commission Request for Additional Information Regarding Application to Revise Technical Specifications to Adopt TSTF-551, "Revise Secondary Containment Surveillance Requirements"
Cooper Nuclear Station, Docket No. 50-298, License No. DPR-46

References:
1. Email from Thomas Wengert, U.S. Nuclear Regulatory Commission, to Linda Dewhirst, Nebraska Public Power District, dated September 14, 2023, "Cooper - Final RAI RE: LAR to Adopt TSTF-551, Revision 3 (EPID L-2023-LLA-0068)"
2. Letter from Khalil Dia., Nebraska Public Power District, to the U.S. Nuclear Regulatory Commission, dated May 3, 2023, "Application to Revise Technical Specifications to Adopt TSTF-551, "Revise Secondary Containment Surveillance Requirements""

Dear Sir or Madam:

The purpose of this letter is for the Nebraska Public Power District (NPPD) to respond to the Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI) (Reference 1) related to the Cooper Nuclear Station (CNS) "Application to Revise Technical Specifications to Adopt TSTF-551, "Revise Secondary Containment Surveillance Requirements,"" (Reference 2).

The response to the RAI is provided in the attachment to this letter. The enclosure contains relevant portions of supporting analyses that were requested.

NPPD has reviewed the information supporting a finding of no significant hazards consideration and the environmental evaluation that were previously provided to the NRC in Reference 2. The additional information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration and does not affect the conclusion of the environmental evaluation.

This letter does not contain any new regulatory commitments.

If you have any questions concerning this matter, please contact Linda Dewhirst, Regulatory Affairs and Compliance Manager, at (402) 825-5416.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 10/10/2023
(Date)

Sincerely,

Khalil Dia
Site Vice President

/dv

Attachment: Response to Request for Additional Information

Enclosure: Supporting Analyses

cc: Regional Administrator w/ attachment and enclosure
USNRC - Region IV

Cooper Project Manager w/ attachment and enclosure
USNRC - NRR Plant Licensing Branch IV

Senior Resident Inspector w/ attachment and enclosure
USNRC - CNS

NPG Distribution w/ attachment and enclosure

CNS Records w/ attachment and enclosure

Attachment

Response to Request for Additional Information

Cooper Nuclear Station, Docket No. 50-298, License No. DPR-46

The Nuclear Regulatory Commission Request for Additional Information (RAI) regarding the License Amendment Request to revise Technical Specifications to adopt Technical Specifications Task Force (TSTF) Traveler TSTF-551, "Application to Revise Technical Specifications to Adopt TSTF-551, "Revise Secondary Containment Surveillance Requirements"" is shown in italics. The Nebraska Public Power District (NPPD) response to the request is shown in normal font.

<u>*Question 1*</u>

TSTF-551, Revision 3 states, in part:

> *For the secondary containment to be considered operable, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained by a single operating SGT [standby gas treatment] subsystem. ...As long as [an] SGT subsystem can draw the required vacuum on the secondary containment when needed (as demonstrated by SR 3.6.4.1.4 or SR 3.6.4.1.5), the secondary containment can perform its safety function.*

SR 3.6.4.1.4 in NUREG-1433, "Standard Technical Specifications - General Electric Plants (BWR [Boiling Water Reactor]/4)," Revision 5 (ML21272A357), verifies that secondary containment can be drawn down to the necessary vacuum within a specified time period using one SGT subsystem. The Cooper facility's TSs do not include this SR.

The NRC staff noted that the SGT system description, as described in the Cooper updated safety analysis report (USAR), assumes that both trains of SGT actuate in situations when the SGT system is required. For example, the loss of coolant accident (LOCA) design basis accident analysis in Revision 30 of the Cooper USAR in Section XIV-6.3.8.3.1.c (ML21130A101) states, in part:

> *Both SGT trains start upon a Group 6 PCIS [Primary Containment Isolation System] signal (assumed from time = 0 seconds to 1 hour), with heater power to one train assumed to have failed. At the 1-hour point, the SGT train with the failed heater is manually secured. Table XIV-6-7 identifies the SGT flow and iodine removal efficiencies that are assumed. This includes a -1% [percent] correction to account for SGT filter bypass.*

Furthermore, Cooper USAR Table XIV-6-7, "SGT System Flows and Iodine Removal Efficiencies," shows that the from time = 0 hours to 1 hour, both SGT trains are running with 1492 cubic feet per minute (cfm) of flow.

Additionally, the description of the SGT system in Cooper USAR Revision 30, Section V-3.3.4 (ML21130A089) states, in part:

> *... Upon receipt of any of the initiation signals, both fans start, all SGT system isolation valves open and each fan draws air from the isolated reactor building at a flow rate of approximately 1780 cfm. When the required negative pressure is reached, a single train of Standby Gas Treatment is capable of maintaining this negative pressure. ...*

Based on the description in the USAR, it appears that both SGT subsystems are required to draw the adequate secondary containment vacuum.

Verify, and provide a supporting analysis, that one SGT subsystem is capable of establishing the required secondary containment vacuum, as indicated in the proposed SR 3.6.4.1.1 NOTE.

NPPD Response

Calculation NEDC 07-056, Rev. 2, "Reactor Building Post-DBA Pressure Analysis at Cooper Nuclear Station," determines the pressure response of the Secondary Containment following a design basis Loss of Coolant Accident (LOCA) for a variety of scenarios and conditions. This calculation evaluates design basis single active failures, including failure of one Standby Gas Treatment (SGT) subsystem, and demonstrates that one SGT subsystem is capable of establishing the required Secondary Containment vacuum within 640 seconds. The required Secondary Containment vacuum is that vacuum assumed in the Cooper Nuclear Station accident analysis approved in Amendment 234, "Application of the Alternative Source Term for Loss-of-Coolant Accident Dose Consequences" (ML092310349). The maximum period of time to establish the required Secondary Containment vacuum determined in the calculation is 657 seconds and occurs in the beyond design basis scenario that assumes multiple failures (one SGT subsystem and the Reactor Building Heating and Ventilation System air-operated inlet valve), an initial Reactor Building pressure equal to atmospheric (0.0 inches water gauge), and 99th percentile 1-hour average wind speeds.

The Cooper Nuclear Station (CNS) LOCA dose analysis is documented in calculation NEDC 07-082, Rev. 5, "Radiological Dose Analysis for a Loss of Coolant Accident (LOCA) at Cooper Nuclear Station." This calculation includes two assumptions regarding operation of the SGT System, both of which are used to conservatively maximize the Control Room occupant, Low Population Zone (LPZ), and Exclusion Area Boundary (EAB) doses.

The first assumption from calculation NEDC 07-082 that relates to the SGT System is concerned with the period of time that Secondary Containment is pressurized greater than atmosphere, which is referred to as the period of positive pressurization. During this period of positive pressurization, it is assumed that the Primary Containment (drywell) releases directly to the environment as a ground-level release, in accordance with Regulatory Guide 1.183, Appendix A, RP 4.2. A time of 657 seconds is used as the period of positive pressurization, which is the largest period of positive pressurization determined in calculation NEDC 07-056 and occurs with a single SGT subsystem operating along with other limiting conditions. This period of positive pressurization is discussed in Updated Safety Analysis Report (USAR) Sections V-3.4 and XIV-6.3.8.3.1.a. Maximizing the period of positive pressurization results in a larger release from the Primary Containment directly to the environment and conservatively maximizes dose.

The second assumption is concerned with maximizing fission product release from Secondary Containment to the environment through the SGT System. To maximize dose, the SGT System flow is maximized by assuming both trains of SGT are in operation for the first one hour of the event.

As stated in USAR Section XIV-6.3.8.3.1, the assumptions and initial conditions listed in Sections XIV-6.3.8.3.1.a through XIV-6.3.8.3.1.c are used in calculating the fission products released to the environs from Secondary Containment. These assumptions and initial conditions are taken from the CNS LOCA dose analysis, calculation NEDC 07-082.

The description of the SGT System in USAR Section V-3.3.4 details the operation of the SGT System upon receipt of an initiation signal, per the system design capabilities. USAR Section V-3.4 discusses the pressurization analysis, calculation NEDC 07-056, and the bounding 657-second period of positive pressurization. As previously noted, the 657-second period of positive pressurization is obtained with only a single SGT subsystem in operation.

Relevant portions of NEDC 07-082 and NEDC 07-056 are included in the enclosure to this letter.

Enclosure

Supporting Analyses

Relevant portions of NEDC 07-082 (2 pages) and NEDC 07-056 (7 pages) are enclosed.

DOCUMENT (SET) START

ATTACHMENT 9.2 **ENGINEERING CALCULATION COVER PAGE**

Sheet 1 of 3

(19) RISK Significant ℗3

CALCULATION COVER PAGE	(5) **CALCULATION NO: NEDC 07-082** (6) **REVISION/Change Notice No: 5**	(20) **Effective Date:** 1/31/23 (2) **Page 1 of 56**
(1) EC #: 22-001	(7) Title: **Radiological Dose Analysis for a Loss of Coolant Accident (LOCA) at Cooper Nuclear Station**	
(3) Design Basis Calc: ☒ YES ☐ NO	(9) System(s)/Structure: MS, PC, SC, CS, RHR, HPCI, ADS, SLC, ALT	(10) Discipline: Mechanical
(11) Safety Class: ☒ Quality Related ☐ Non-Quality Related	(12) Component/Equipment/Structure: MS-AOV-AO80A/B/C/D MS-AOV-AO86A/B/C/D	
(18) Proprietary: ☐ YES ☒ NO		
(4) Superseded: ☐ YES ☒ NO	(21) Technical Conscience Review Board: ☐ YES ☒ NO	

(14) Keywords (Description/Topical Codes): LOCA, Dose, AST, Control Room, EAB, LPZ, RG 1.183, Amendment 234, Amendment 242, RADTRAD

(8) Calculation Description:

The purpose of this calculation is to determine the dose to the control room occupant and to a person at the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) at the Cooper Nuclear Station site following a design basis Loss of Coolant Accident (LOCA). The analysis is performed using an Alternative Source Term (AST) in accordance with the guidance provided by the NRC in Regulatory Guide 1.183 (July 2000) and as allowed by 10CFR50.67.

Revision 5 incorporates the extended period of positive Reactor Building pressurization that occurs immediately following LOCA initiation, per NEDC 07-056, Rev. 2. CR-CNS-2022-06539 is addressed to align the calculated ESF leakage source term with the methodology presented.

(13) Conclusion/Recommendations:

The results are tabulated in Section 6 of this calculation for each of the three (3) receptor locations:

1. Control Room
2. Low Population Zone (LPZ)
3. Exclusion Area Boundary (EAB)

All calculated doses were found to be below the stipulated limits. It is therefore concluded that the regulatory dose limits will not be exceeded following a postulated design basis LOCA at Cooper Nuclear Station.

REVIEWS

(15) Name/Signature/Date	(16) Name/Signature/Date	(17) Name/Signature/Date
1/19/23 For Jan Bostelman per email	1/19/23 For Luke Jensen per email	Steve Nelson 1-23-23
1/19/23 Benjamin S. Swoboda	☒ **Design Verifier** ☐ **Technical Reviewer** ☐ Comments Attached	**Supervisor/Approval** ☐ Comments Attached
Responsible Engineer		

source term used for this accident is based on GE Hitachi GNF2 fuel with 24-month cycles, bounding core inventory source term [DI 10]. The core inventory source term was calculated using the isotope generation and depletion code ORIGEN2 version 2.1, incorporating the BWR extended burnup library BWRUE.LIB.

2.3. Leakage from the Drywell

Activity released from the reactor core during the blowdown phase of a LOCA will be mixed in the drywell atmosphere instantaneously and homogeneously in accordance with the guidance in Regulatory Guide 1.183 [1] Appendix A, Section 3. This analysis uses natural deposition in the drywell (Appendix A, RP 3.2), but does not include sprays (Appendix A, RP 3.3), or recirculation filters (Appendix A, RP 3.4) to further reduce the concentration of radionuclides. The Powers BWR natural deposition model [10] is used with the minimum deposition option set (10%) as recommended (Appendix A, RP 3.2).

As a result of the pressure buildup in the drywell, radionuclides may leak from the pressure vessel along various penetrations into the reactor building (secondary containment) [Section 2.3.1].

A CNS pressurization analysis, NEDC 07-056 [DI 32], indicates that the reactor building may have a positive pressure for 657 seconds in the worst evaluated case (low outside air temperature, failure of one SGTS train, failure of the HV intake air-operated damper, high wind conditions, and reactor building initially at atmospheric pressure). This worst evaluated case bounds the period of positive pressurization for the limiting design basis case (low outside air temperature and single failure of one SGT train). In this analysis it is assumed that the drywell releases directly to the environment during that period as a ground release in accordance with Regulatory Guide 1.183, App A, RP 4.2. To bound the design basis case, this analysis uses the 657-second release from the worst evaluated case. Potential leakage pathways from the drywell directly into the condenser, reactor building, or environment have also been evaluated. It was determined that there is no identifiable containment bypass leakage other than through the MISV's [DI 28].

2.3.1. Drywell to Reactor Building Release

The drywell leakage into the reactor building is set at 0.635%/day based on an assumed drywell pressure of 58 psig per CNS Technical Specifications [DI 5]. The Cooper LOCA containment analysis, that gives the worst-case leakage at 24 hours and beyond, shows that there is only one minute before the pressure within the drywell falls to 42.3 psia and the temperature falls below 300°F [DI 23]. After 24 hours, the drywell leakage into the reactor building is reduced because that document shows that under worst case cooling conditions, the long-term peak drywell pressure is about 22 psig (240°F) and occurs in approximately 8 hours post-accident. Table 2-3 details the temperatures and pressures used for this analysis. Note that the steam line wall temperature chosen is the design value; following a LOCA this temperature will decrease as soon as the MSIV's close. Similarly, the condenser temperature chosen is based on the design flows and temperatures entering the condenser during normal operation and will also decrease once MSIV's close.

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CALCULATION COVER PAGE	(5) **CALCULATION NO: NEDC 07-056** (6) **REVISION/Change Notice No: 2**	(20) **Effective Date:** 1/31/23 (2) **Page 1 of 115**
(1) EC #: 22-001	(7) Title: **Reactor Building Post-DBA Pressure Analysis at Cooper Nuclear Station**	
(3) Design Basis Calc: ☒ YES ☐ NO	(9) System(s)/Structure: SC, SGT, HV, SFP	(10) Discipline: Mechanical
(11) Safety Class: ☒ Quality Related ☐ Non-Quality Related	(12) Component/Equipment/Structure: HV-FAN- SF-R-1A-A/B HV-FAN- EF-R-1A/1B HV-AOV-257AV HV-AOV-259AV HV-AOV-261AV HV-MOV-258MV HV-MOV-260MV HV-MOV-272MV SGT-FAN- EF-R-1E/1F SGT-AOV-249AV/250AV	
(18) Proprietary: ☐ YES ☒ NO		
(4) Superseded: ☐ YES ☒ NO	(21) Technical Conscience Review Board: ☒ YES ☐ NO	

(14) Keywords (Description/Topical Codes):

DBA, LOCA, Reactor Building, Pressurization, Alion, ALION-CAL-NPPD-4012-01

(8) **Calculation Description:**

This calculation determines the Reactor Building pressure response following a design basis Loss of Coolant Accident (LOCA). A GOTHIC model was prepared starting with a model previously used for post-LOCA heatup analyses. Fan components and other elements were added to the original model to simulate the HV supply ventilation system and the Standby Gas Treatment (SGT) System. As a result of the accident, the HV fans trip off, the HV system isolation valves close, and the SGT System is started. Both LOOP and non-LOOP cases were evaluated in terms of event timing and varying heat loads. These events were simulated with the new model and the resulting Reactor Building pressure response was evaluated.

Revision 2 is performed to address the GOTHIC model sensitivity to assumed outside air temperatures, as noted in CR-CNS-2021-05402. This revision develops and evaluates new scenarios for design basis and beyond design basis conditions to determine the limiting Reactor Building pressurization effects. Additionally, the GOTHIC model inputs and assumptions are refined to produce more accurate results.

(13) **Conclusion/Recommendations:**

The results are provided in Section 6 for each of the cases evaluated. The worst-case conditions include failure of one SGT train and the design minimum outside air temperature. A time of 657 seconds is acceptable to use as the period of positive Reactor Building pressurization, as this is the maximum drawdown time of all evaluated scenarios.

REVIEWS

(15) Name/Signature/Date	(16) Name/Signature/Date	(17) Name/Signature/Date
12/1/22 Benjamin S. Swoboda **Responsible Engineer**	12/5/22 Jeff Maddox ☒ **Design Verifier** ☐ **Technical Reviewer** ☐ Comments Attached	Steve Nelson 1-23-23 **Supervisor/Approval** ☐ Comments Attached

6. Calculations

The GOTHIC files used in this analysis are identified in Table 4. The base model and the calibration models are described in the previous section. The remaining pressurization cases are described in the following paragraphs and in Table 5, which summarizes the event timing and environmental conditions for each case.

Table 4 - GOTHIC Model File Identification for the Pressurization Analysis

File Name	**Description**
cns_press_base_R2.GTH	This is the base model for the pressurization analysis. It was developed using the post-LOCA heatup model, *Case1R.GTH*, as a starting point. Normal operating heat loads are incorporated. It is used to calibrate the SGT System components.
cns_press_base_Tlo_R2.GTH	This is a modified version of the base model, *cns_press_base_R2.GTH*. Environmental conditions are set to design winter conditions. It is used to calibrate the HV System components at the minimum environmental temperature condition.
cns_press_base_Thi_R2.GTH	This is a modified version of the base model, *cns_press_base_R2.GTH*. Environmental conditions are set to design summer conditions. It is used to calibrate the HV System components at the maximum environmental temperature condition.
cns_press_case1_Thi_R2.GTH	This is the model for Case 1 at the maximum environmental temperature condition. It assumes that all of the HV isolation valves close normally. Single failure is one SGT train.
cns_press_case1_Tlo_R2.GTH	This is the model for Case 1 at the minimum environmental temperature condition. It assumes that all of the HV isolation valves close normally. Single failure is one SGT train.
cns_press_case2_Thi_R2.GTH	This is the model for Case 2 at the maximum environmental temperature condition. It assumes that the AOV in the HV System supply duct does not close. All other HV System isolation valves close normally and both SGT trains operate.
cns_press_case2_Tlo_R2.GTH	This is the model for Case 2 at the minimum environmental temperature condition. It assumes that the AOV in the HV System supply duct does not close. All other HV System isolation valves close normally and both SGT trains operate.
cns_press_case3_Tlo_R2.GTH	This is the model for Case 3. It is the same as Case 1_Tlo, except that a LOOP is assumed concurrent with LOCA initiation and a LF/LV condition is assumed for the SGT fan.

File Name	Description
cns_press_case4_Tlo_R2.GTH	This is the model for Case 4. It is the same as Case 1_Tlo, except with the 95th percentile 1-hour average wind speed.
cns_press_case5_Tlo_R2.GTH	This is the model for Case 5. It is the same as Case 1_Tlo, except with the 99th percentile 1-hour average wind speed.
cns_press_case6_Tlo_R2.GTH	This is the model for Case 6. It is the same as Case 1_Tlo, except that the Reactor Building pressure at the start of the event is set to atmospheric (0 in wg).
cns_press_case7_Tlo_R2.GTH	This is the model for Case 7. It is the same as Case 1_Tlo, except with the 99th percentile 1-hour average wind speed and the Reactor Building pressure at the start of the event is set to atmospheric (0 in wg).
cns_press_case8_Tlo_R2.GTH	This is the model for Case 8. It is the same as Case 1_Tlo, except that the AOV in the HV System supply duct does not close. This failure is in addition to one SGT train.
cns_press_case9_Tlo_R2.GTH	This is the model for Case 9. It is the same as Case 7, except that the AOV in the HV System supply duct does not close. This failure is in addition to one SGT train. This scenario represents the worst case, beyond design basis conditions.
cns_press_case10_Tlo_R2.GTH	This is the model for Case 10. It is the same as Case 1_Tlo, except that the HV supply temperature is set to 50°F, the outlet air temperature of the heating coil.
cns_press_case11_Tlo_R2.GTH	This is the model for Case 11. It is the same as Case 1_Tlo, except that the SGT and leakage flow paths are calibrated to be at the limits allowed by the Technical Specifications (1780 cfm and -0.25 in wg).

Table 5 - GOTHIC Model Event Timing and Variable Parameters for the Pressurization Analysis

Event Description	Time of Event (s)												
	Case 1_Thi	Case 1_Tlo	Case 2_Thi	Case 2_Tlo	Case 3	Case 4	Case 5	Case 6	Case 7	Case 8	Case 9	Case 10	Case 11
LOCA/LOOP	0	0	0	0	LOOP 0	0	0	0	0	0	0	0	0
Supply fan trips	0	0	0	0	0	0	0	0	0	0	0	0	0
Supply damper MOV starts to close (stroke time)	-	-	2 (90)	2 (90)	-	-	-	-	-	2 (90)	2 (90)	-	-
Supply damper AOV starts to close (stroke time)	2 (12)	2 (12)	-	-	2 (12)	2 (12)	2 (12)	2 (12)	2 (12)	-	-	2 (12)	2 (12)
Exhaust fan trips	0	0	0	0	0	0	0	0	0	0	0	0	0
Exhaust damper MOV starts to close (stroke time)	-	-	-	-	-	-	-	-	-	-	-	-	-
Exhaust damper AOV starts to close (stroke time)	0 (2.1)	0 (2.1)	0 (2.1)	0 (2.1)	2 (2.1)	0 (2.1)	0 (2.1)	0 (2.1)	0 (2.1)	0 (2.1)	0 (2.1)	0 (2.1)	0 (2.1)
Number of SGT Trains	1	1	2	2	1	1	1	1	1	1	1	1	1
SGT starts	2	2	2	2	16	2	2	2	2	2	2	2	2
SGT damper starts to open (stroke time)	2 (15)	2 (15)	2 (15)	2 (15)	16 (15)	2 (15)	2 (15)	2 (15)	2 (15)	2 (15)	2 (15)	2 (15)	2 (15)
Outside air temperature (°F)	97	-5	97	-5	-5	-5	-5	-5	-5	-5	-5	-5[(2)]	-5
Initial RB pressure (in wg)	-0.25	-0.25	-0.25	-0.25	-0.25	-0.25	-0.25	0	0	-0.25	0	-0.25	-0.25
Wind speed during event (mph)	0	0	0	0	0	16.6	27.0	0	27.0	0	27.0	0	0

Note (1): Only the controlling supply or exhaust damper was modeled for each case; '-' indicates the valve behavior was not modeled.
Note (2): Normal HV System supply air temperature set to 50°F for Case 10.

The completed pressurization model was used to calculate the Reactor Building pressure transients for 11 cases following a design basis accident. In all cases, the model is initialized with the normal operating heat loads and the HV System fans running. The fans are operated long enough to allow the HV System flows and Reactor Building temperatures to reach equilibrium. A trip is then initiated to shut down the HV System ventilation fans, start up the SGT System fan, and actuate the post-LOCA heat loads.

Upon the LOCA initiation trip, the HV System fans begin to coast down, the HV System isolation valves begin to close, the SGT isolation valve begins to open, and the SGT fan starts. The signal propagation time is conservatively accounted for in all scenarios (Assumption 4.17). For the LOOP case (Case 3), the SGT fan startup trip includes the signal propagation time plus an additional 14 second delay to allow for Diesel Generator startup (Assumption 4.17). Other differences between the various cases involve assumptions about which of the redundant HV System isolation valves are operational, how quickly they can operate, how many SGT trains are operational, and the local environmental conditions at the time of the event (i.e., temperature and wind speed).

Cases 1 through 3 are considered design basis cases. These cases assume a single failure and evaluate conditions within design and licensing basis limits (i.e., outside air temperature, initial Reactor Building pressure, Reactor Building leakage, and SGT flow). Cases 4 through 9 are beyond design basis (BDB) scenarios that evaluate windy conditions, multiple failures, higher Reactor Building initial pressure, and combinations of these conditions. Cases 10 and 11 are sensitivity studies that demonstrate margin associated with assumptions used to develop bounding conditions.

An important purpose for this analysis is to determine how long the containment pressure remains positive during the transient following a LOCA, because that information is required as input for the alternative source term radiological dose analysis in NEDC 07-082 (DI 51). Table 6 provides a summary of the time period that the containment pressure is above atmospheric pressure for each of the cases that were evaluated. Additional discussion of results for each case is provided in the following subsections.

Note that the times calculated by GOTHIC are conservatively rounded up to the next highest integer.

Table 6 - Summary of Reactor Building Pressurization Results

GOTHIC Case #	Time Period for $P_{RB} > P_{atm}$ (seconds)
1_Thi	2 – 333
1_Tlo	2 – 640
2_Thi	2 – 335
2_Tlo	2 – 635
3	2 – 598
4	2 – 641
5	2 – 644
6	2 – 646
7	2 – 650
8	2 – 647
9	2 – 657
10	2 – 428
11	2 – 501

From the above results, the limiting design basis scenario for Reactor Building pressurization is Case 1_Tlo, which postulates a single active failure of one SGT train. The limiting beyond design basis scenario is Case 9, which postulates active failures of one SGT train and the HV Supply isolation AOV (multiple failures), an initial Reactor Building pressure equal to atmospheric (0.0 in wg), and 99th percentile 1-hour average wind speeds. Both Cases 1_Tlo and 9 are performed at the design basis minimum outside air temperature of -5°F and no credit is taken for heating the HV supply air via AS heating coils. This assumption ensures that the resulting drawdown times are conservative over the entire range of design outside air temperatures and encompasses the potential loss of the non-Essential AS System heating. Note that normal cold weather operating conditions employ the AS-supplied heating coil in the HV System, which would reduce drawdown time by about 33% (Case 10). Additionally, the evaluated parameters for Reactor Building leakage resistance and SGT flow path resistance are conservative and bounding. The results from Cases 1_Tlo and 11 demonstrate that selected leakage and SGT flow path resistances conservatively increase the drawdown time by about 28%.

The above results also demonstrate that the limiting parameter for drawdown time is by far the outside air temperature. For Case 1, the drawdown time at the design maximum outside air temperature is nearly two times faster than the drawdown time at the design minimum temperature. The GOTHIC results show that Reactor Building pressurization and, consequently, the drawdown time, is primarily driven by room heat up and that inputs and assumptions which maximize room heat up will conservatively extend the drawdown time.

For input into the AST LOCA dose calculation, NEDC 07-082 (DI 51), a bounding time of 657 seconds for Reactor Building pressurization and subsequent drywell release to the environment is acceptable. This is the maximum period of positive pressurization for all design basis and beyond design basis conditions evaluated and contains a large amount of conservatism to assure that the Reactor Building will reach vacuum conditions within the assumed time frame.