ATTACHMENT 1

Braidwood Units 1 and 2 – Responses to NRC Request for Additional Information (RAI) Regarding Ultimate Heat Sink Temperature Increase License Amendment Request (License Nos. NPF-72 and NPF-77)

April 2015

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SCVB-RAI-2

Section 3.6.1 identifies changes to the Mass and Energy (M&E) release by correcting the Steam Generator tube material density and specific heat as discussed in Westinghouse Nuclear Safety Advisory Letter (NSAL)-14-2. However, the LAR does not mention M&E release corrections discussed in NSAL-06-6 and NSAL-11-5. Describe changes in the following containment analyses results using the corrected methodology that incorporates corrections listed in the above three NSALs, (a) containment peak pressure, (b) containment peak gas temperature for Environment Equipment Qualification (EEQ), (c) containment peak wall temperature, (d) containment sump peak water temperature, (e) pump Net Positive Suction Head (NPSH) Available (NPSHA) for the pumps that draw water from the containment sump during recirculation mode of safety injection and containment cooling, and (f) containment minimum pressure analysis for Emergency Core Cooling System performance capability.

Westinghouse Response

(Impact on Containment Pressure)

All issues related to the subject Nuclear Safety Advisory Letters (NSALs) (NSAL-06-6, NSAL-11-5, and NSAL-14-2) were explicitly addressed (and subsequent corrections in the loss-of-coolant (LOCA) mass and energy (M&E) release analysis were made) in support of the Ultimate Heat Sink (UHS) analysis. Sensitivity studies for the individual effects of each of the issues identified in the NSALs are not available; although the following discussion outlines the issues for the NSALs and potential impacts.

NSAL-06-6 "LOCA Mass and Energy Release Analysis"

Westinghouse has identified several procedural issues with past guidance provided for the performance of loss-of-coolant accident (LOCA) mass and energy (M&E) release analyses (Reference 1) by Westinghouse.

Eight issues have been identified that could potentially affect the LOCA M&E (WCAP-10325-P-A; Reference 1) results which are used to evaluate containment integrity.

1) Area of the Downcomer in the REFLOOD Code
Westinghouse-designed reactors can be divided into downflow and upflow barrel baffle designs. The upflow modelling for Braidwood Units 1 and 2 was correct, therefore no impact.

2) Area of the Upper Plenum in the FROTH Code
The FROTH computer program is run in conjunction with the REFLOOD computer program and calculates the LOCA M&E releases for the post-reflood period until the steam generator (SG) secondary side pressure(s) is calculated to equilibrate at the containment design pressure. During this time period, the two-phase mixture levels in the core, upper plenum, hot leg and SG inlet plenum are the principle parameters of interest. Due to a misinterpretation of a database parameter, the cross-sectional area of the upper plenum (AUPP) was being over predicted, which resulted in a reduction in entrainment to the SGs and thus less steam production. Correction of the upper plenum area results in increased M&E releases, and therefore a pressure penalty, i.e.,

an increase in the calculated containment pressure, for the double-ended pump suction (DEPS) break case.

3) Review of Other FROTH Inputs
A review of the FROTH code user controlled input variables showed that the SG inlet plenum area (ASGP), which is used to calculate the void fraction in the SG inlet plenum, was based on a value that was generally too small. The ASGP modelling for Braidwood Units 1 and 2 was conservatively calculated, therefore, there is not an adverse impact.

4) WCAP-10325-P-A Model Features
The Westinghouse LOCA M&E release model (Reference 1) was approved in February 1987. Westinghouse identified the need to clarify two model features; the assumptions placed on the SG exit steam enthalpy during the post-reflood period and the assumed power level used in the LOCA M&E analysis. This issue has been resolved through discussions with the NRC. Their acceptance is documented in Reference 2. The NRC staff found that the assumption used for the SG exit enthalpy during the post-reflood period is conservative. The NRC staff also found that Westinghouse's understanding of performing analyses at the licensed core power, regardless of power level, is an acceptable method, as long as the plant-specific calorimetric uncertainty is considered. The issue has no impact on the analysis results.

5) Main Feedwater Addition Following a Reactor Trip
The Westinghouse methods account for the addition of main feedwater (MFW) to the SG secondaries following a LOCA in the time frame from reactor trip until MFW isolation is calculated to occur. Double ended hot leg (DEHL) breaks are not affected and depending upon the time at which peak containment pressure is calculated, a penalty to peak pressure may occur for the DEPS minimum ECCS case.

6) Auxiliary Feedwater System Purge and Unisolatable Volumes
After isolation of the MFW, a volume of hot MFW will reside in the main feed lines between the auxiliary feedwater (AFW) injection point and the SG secondary side. Once AFW flow is initiated, the hot MFW water will be pushed into the SG secondary side. As the SGs are calculated to depressurize, there may be additional volume trapped between the AFW injection point and the MFW isolation valve that will flash and be pushed into the SG secondary. Addition of these effects to the LOCA M&E release calculation has been shown to increase the total energy release to containment and results in a penalty to the calculated containment pressure.

7) AFW Flow for FROTH Code
In some cases it was found that the actual flow used in the analysis was a pump flow (total AFW flow available) and not flow per SG. This resulted in SG AFW flows that were high. AFW flow is only modeled in the FROTH code which calculates the transient from end of reflood until the SG secondary side pressure has been calculated to have depressurized to the containment design pressure. The effect on analysis results for reductions in AFW flow during this short period is a pressure penalty in peak containment pressure.

8) Possibility of Asymmetric AFW Flow
LOCA analyses are performed assuming that off-site power is lost coincident with the event, and with the limiting single failure of one diesel generator to start. If a plant design does not start the diesel-driven AFW pump on the loss-of-offsite power or a safety injection (SI) signal, the typical

design will have one motor-driven AFW pump in operation which generally will not feed all SGs. Thus, one or more SGs may not receive any AFW flow. If the broken loop SG is the SG with no AFW flow, there will be an effect on the calculated LOCA M&E release. The AFW modelling for Braidwood Units 1 and 2 was conservatively calculated, therefore a no adverse impact.

<u>NSAL-11-5 "Westinghouse LOCA Mass and Energy Release Calculations Issues"</u>

This NSAL is applicable to LOCA M&E release calculations performed for Westinghouse-designed pressurized water reactors (PWRs) utilizing the methodology documented in WCAP-10325-P-A (Reference 1).

The six issues listed below can potentially impact the plant specific LOCA M&E release calculation results which are used as input to the containment integrity response analyses. The six issues, which include generic errors that can impact the plant specific analyses differently, are:

1) The reactor vessel modeling did not include all the appropriate vessel metal mass available from the component drawings. Generic sensitivity results for the correction of this effect to the LOCA M&E release calculation has showed no significant increase in the total energy release to containment and no significant increase in the calculated containment pressure.

2) The reactor vessel modelling did not include all the appropriate vessel metal mass in the reactor vessel barrel/baffle region. The effect on analysis results for correction in vessel metal mass resulted in a pressure penalty in peak containment pressure.

3) The reactor coolant pump (RCP) homologous curve input incorrectly included an absolute zero point coordinate. This generic sensitivity studies showed no significant increase in the calculated peak containment pressure during the blowdown phase; only the reflood phase portion is affected and would result in a containment pressure penalty.
4) The RCP homologous curve input incorrectly contained a sign error in a coordinate value. This resulted in a mis-prediction of the RCP hydraulic loss; when corrected a small benefit (for the Braidwood analysis) was realized.

5) The LOCA M&E release analysis initializes at a non-conservative (low) steam generator (SG) secondary pressure condition (reflecting a steamline pressure and not the appropriate higher tube region bundle pressure). The effect of this correction would be pressure penalty (increase) on the peak containment pressure.

6) An error found in the EPITOME computer code (WCAP-10325-P-A methodology only) that is used to determine the M&E release rate during the long-term (i.e., post-reflood) SG depressurization phase of the LOCA transient. The error results in an underestimated energy release in the long-term, post-reflood phase of the transient. The impact of this single issue is expected to be a penalty to the calculated containment pressure.

NSAL-14-2 "Westinghouse Loss-of Coolant Accident Mass and Energy Release Calculation Issue for Steam Generator Tube Material Properties"

The LOCA M&E release analyses are sensitive to energy stored in the reactor coolant system (RCS) metal, including the SG tubes. Recently, it was determined that the input modification program database and the input modification program preprocessor were incorrectly using the density for stainless steel in determining the mass of the SG tubes and the specific heat (C_p) of stainless steel for the stored metal energy. The increase in the LOCA M&E release associated with this issue affects the plant specific containment LOCA blowdown and post-blowdown transient conditions relative to a penalty on calculated peak containment pressure.

SCVB-RAI-2
(Impact on Containment Temperature)

Describe changes in the following containment analyses results using the corrected methodology that incorporates corrections listed in the above three NSALs, ... (b) containment peak gas temperature for Environment Equipment Qualification (EEQ), (c) containment peak wall temperature, (d) containment sump peak water temperature, .. and (f) containment minimum pressure analysis for Emergency Core Cooling System performance capability.

Westinghouse Response

The residual impact as a result of an energy release (due to the subject NSAL(s)) will be an increase in pressure and temperature. Temperature is a function of pressure; therefore, if there is a resultant pressure increase, temperature will increase for the containment peak gas temperature, containment peak wall temperature, and containment sump peak water temperature, Items (b)(c)and (d) of SCVB-RAI-2

Item (f) of SCVB-RAI-2,

The subject NSAL(s) issues are specific to the LOCA M&E release methodology (WCAP-10325-P-A) utilized for containment integrity analyses and do not affect the LOCA emergency core cooling system (ECCS) analysis and does not fall within the reporting requirement of 10 CFR 50.46.

References:

1. WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version," May 1983.
2. Letter from Herbert N. Berkow (NRC) to Mr. James A. Gresham (Westinghouse): Acceptance of Clarifications of Topical Report WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version" (TAC No. MC7980); October 18, 2005. [ADAMS Accession No. ML052660242]

SCVB-RAI-3

Section 3.6.1 states:

"However, the SX temperature change coupled with the other changes described above resulted in peak pressure values similar to the current design analysis."

For each of the "other changes", provide a summary whether the change resulted in an increase or decrease in released mass, and an increase or decrease in the released energy.

Westinghouse Response

Text taken from page 17 of 25 of RS-14-193 (Reference 1),

"Other changes were made to the LOCA containment analysis not related to the UHS temperature change. The analysis incorporated updated containment spray and safety injected flow rates. Additional changes to correct open items were 1) correction for the SG tube material to have the correct density and specific heat (Reference NSAL 14-2) 2) correction to the SATAN78 power shape selection option to select a chopped cosine power shape and 3) incorporation of an NRC approved model (Reference 3) for drift flux and break flow with inertia."

The underlined text reflects the "other changes" made to the LOCA containment analysis not related to the UHS temperature change.

-Updated containment spray: The containment spray was modelled at a reduced flow rate from 3285 gpm to 3113 gpm; the change will not directly affect the mass and energy release, but will result in reduced containment spray performance (heat removal) and a pressure penalty.

-Reduction in ECCS flow rates: A reduction in emergency core cooling system (ECCS) flow will result in a pressure penalty due to a reduction in the condensation of steam with the cold injection water; increased steam release / energy release.

-Correction for the SG tube material to have the correct density and specific heat (Reference NSAL 14-2): The correction for SG tube material will result in an increase in the LOCA M&E release during the blowdown and post-blowdown transients, thus resulting in a penalty on the peak containment pressure.

Incorporation of an NRC approved model (Reference 2) for drift flux and break flow with inertia: The SATAN78 input model was revised to include and credit the effects (potential benefits) for drift flux and break flow with inertia consistent with the NRC's SER approval for WCAP-10325-P-A (Reference 2) LOCA M&E release methodology evaluation model. The result of applying the drift flux model is a small benefit in blowdown M&E releases. In addition, WCAP-10325-P-A evaluation methodology has been approved to incorporate improved fluid momentum flux terms, which can be applied to the break flow model, which also can result in a small reduction in mass and energy release.

References:

1. RS-14-193, "Request for a License Amendment to Braidwood Station, Units 1 and 2, Technical Specification 3.7.9, "Ultimate Heat Sink", August 19, 2014.
2. WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version," May 1983.

SCVB-RAI-4

Section 3.6.1 states:

"The Braidwood Station Units were reanalyzed to assess an increase in the water temperature of the UHS to 104°F"

(a) Describe the methodology used for performing the proposed short and long term M&E release reanalysis and how does it differ from the current licensing basis analysis methodology.

(b) Describe the methodology used for performing the proposed short and long term containment pressure analysis for peak pressure, containment peak temperature analysis for EEQ, sump water temperature analysis for pump NPSH. How does the methodology for the proposed analysis differ from the current licensing basis analysis methodology?

(c) Provide a comparison of the inputs and assumptions in the proposed analysis that were changed from the current analysis. Provide justification for those inputs and assumptions in which the conservatism in the proposed analysis is reduced.

(d) Provide the resulting graphs for the most limiting LOCA peak pressure analysis for the double ended hot leg break and double ended pump suction break for both units.

Westinghouse Response – Item (a)

The methodology used for performing short-term mass and energy release analysis and containment response is described in WCAP-8264-P-A, Revision 1 and WCAP-8077, documented in References 1 and 2, respectively. A change (increase) in the UHS temperature and associated impact to performance of the Reactor Containment Fan Coolers (RCFCs) is a post-blowdown phase issue in the LOCA analysis. The short-term LOCA M&E release analysis supports the subcompartment analyses, which are performed to ensure that the subcompartment walls can maintain their structural integrity during the short pressure pulse (generally less than three seconds) following a high energy pipe rupture. Due to the short duration of the short-term LOCA M&E releases and subsequent subcompartment analysis these analyses are not affected by the changes in UHS temperature and associated impact to RCFC performance. Therefore, no evaluation of short-term LOCA containment analysis impact is required.

The long-term LOCA mass and energy releases are analyzed to approximately 10^6 seconds 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 Environmental Qualification (EQ) acceptance limits. For the Braidwood UHS increase, Westinghouse generated the mass and energy releases using the March 1979 model, described in References 3, 4, 5, 6, and 7. The Nuclear Regulatory Commission (NRC) review and approval letter (Reference 6) is included with Reference 3. This is not a first time application of this methodology to the Braidwood Units 1 & 2, and was utilized and approved previously for the power uprate program and the measurement uncertainty recapture (MUR).

Westinghouse Response – Item (b)

The Braidwood Units 1 & 2 containment systems are designed such that for all loss-of-coolant accident (LOCA) break sizes, up to and including the double-ended severance of a main reactor coolant pipe, the containment peak pressure remains below the design pressure. This section details the containment response subsequent to a hypothetical LOCA assuming measurement uncertainty recapture (MUR) conditions and an increase in the UHS temperature to 104°F. The containment response analysis uses the long term mass and energy release data generated based on the methodology outline in the Response for Item (a) above.

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 assure that the containment pressure remains below its design pressure at the uprated conditions. In support of equipment design and licensing criteria (e.g. 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 included sump temperature. The containment sump temperature transient presented was taken from the limiting double-ended pump suction break case analyzed for containment integrity with respect to containment pressure response.

Transient phenomena within the reactor coolant system (RCS) affect containment conditions by means of convective mass and energy transport through the pipe break.

Calculation of containment pressure and temperature is accomplished by use of the digital computer code COCO (Reference 8). COCO is a mathematical model of a generalized containment; the proper selection of various options in the code allows the creation of a specific model for a particular design. The values used in the specific model for different aspects of the containment are derived from plant-specific input data. The COCO code has been used and found acceptable to calculate containment pressure transients for dry containment plants. This is not a first time application of this methodology to the Braidwood Units 1 & 2, and was utilized and approved previously for the power uprate program and the MUR (Reference 10).

Westinghouse Response – Item (c)

LOCA M&E Release Changes

1) Revised Steam Generator Tube Material Properties:
   It was determined that the input modification program database and the input modification program pre-processor used to develop SATAN78 SG tube input were using the density for stainless steel in determining the mass of the SG tubes and the specific heat (C_p) of stainless steel for the stored metal energy. Density and heat capacity adjustments were made to correctly reflect the INCONEL® 690 (Unit 1) and 600 (Unit 2) alloy. The density change was made to correct the mass of the SG tubes, and the Cp adjustment will correct the initial energy in the primary (tubes). (Reference 9)

INCONEL is a registered trademark of the Special Metals Corporation group of companies.

2) Revised Pumped Safety Injection and Recirculation flow rates

SCVB-RAI-4 – Table 1

Minimum Safety Injection		
Injection Flow		
	Current	Revised
RCS Pressure	Flow	Flow
(psia)	(gpm)	(gpm)
15	5686.1	4637
20	5685.4	4576
40	5609.8	4334
60	5117.1	4070
80	4532.4	3776
100	3666.7	3448
120	1968.2	3065
300	893.7	914
Cold Leg Recirculation Flow		
RCS Pressure		
(psia)	(gpm)	(gpm)
15	994.1	3208

3) Incorporation of an NRC approved model (Reference 3) for drift flux and break flow with inertia: The SATAN78 input model was revised to include and credit the effects (potential benefits) for drift flux and break flow with inertia consistent with the NRC's SER approval for WCAP-10325-P-A (Reference 3) LOCA M&E release methodology evaluation model. The result of applying the drift flux model is a small benefit in blowdown M&E releases. In addition, WCAP-10325-P-A evaluation methodology has been approved to incorporate improved fluid momentum flux terms, which can be applied to the break flow model, which also can result in a small reduction in mass and energy release.

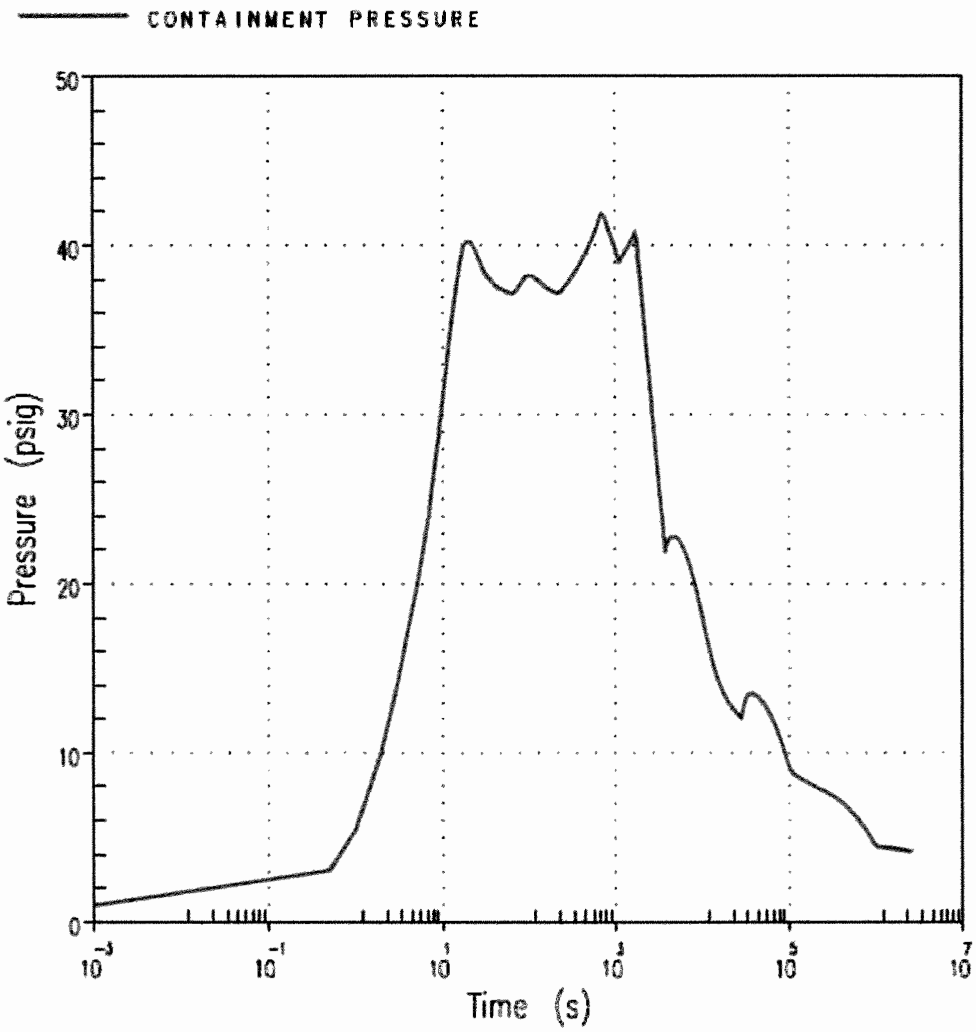
4) The SATAN78 input model was revised to model a chopped cosine power shape so that the exit temperature from the upper core node correctly equals the core outlet temperature, which is associated with the NRC approved LOCA M&E release methodology, i.e., WCAP-10325-P-A.

Containment Response Changes

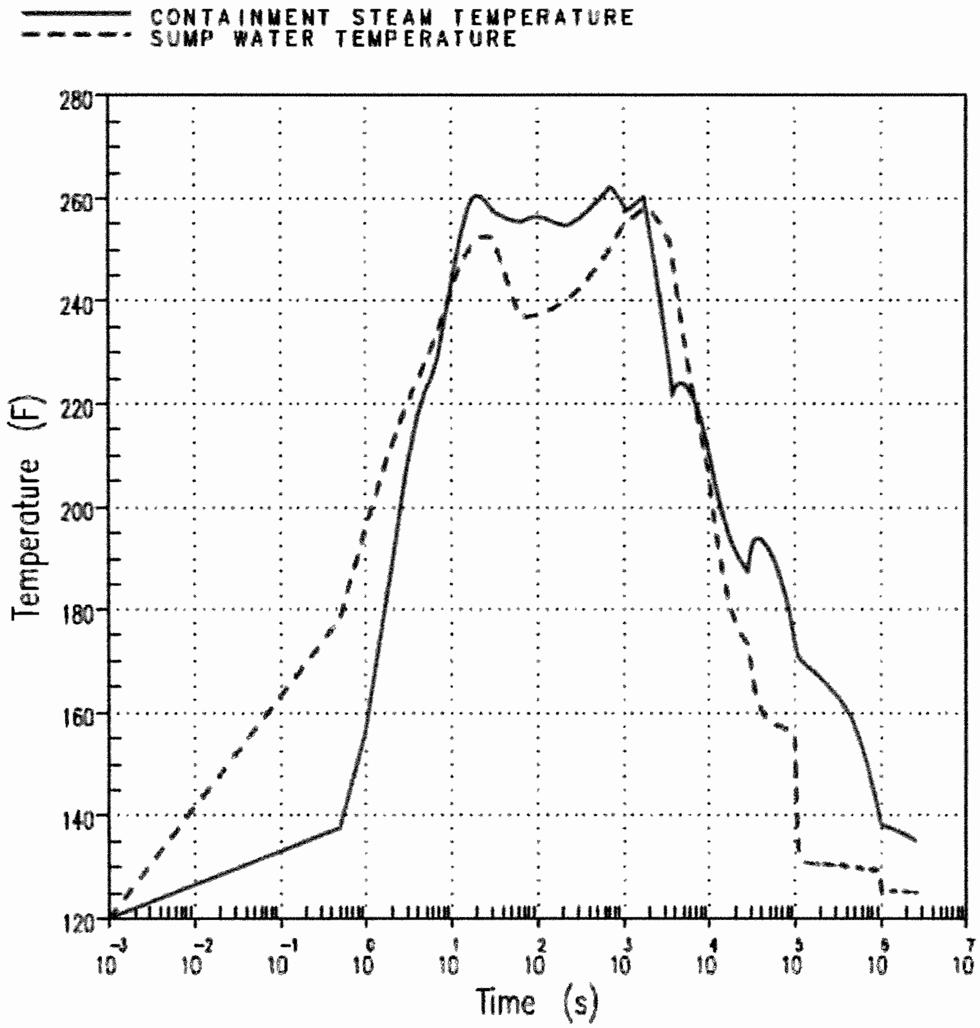
1) Service Water Temperature (°F): Revised from 100 to 104
The service water temperature change affected the analysis by changing the heat removal performance of the RCFCs; and also by the reduced cooling performance of the component water (CCW) heat exchangers (HXs) due to the delta T change in service water temperature.

2) Containment Spray Pump Flow Rate (gpm): Revised from 3285 to 3113.

Westinghouse Response – Item (d)



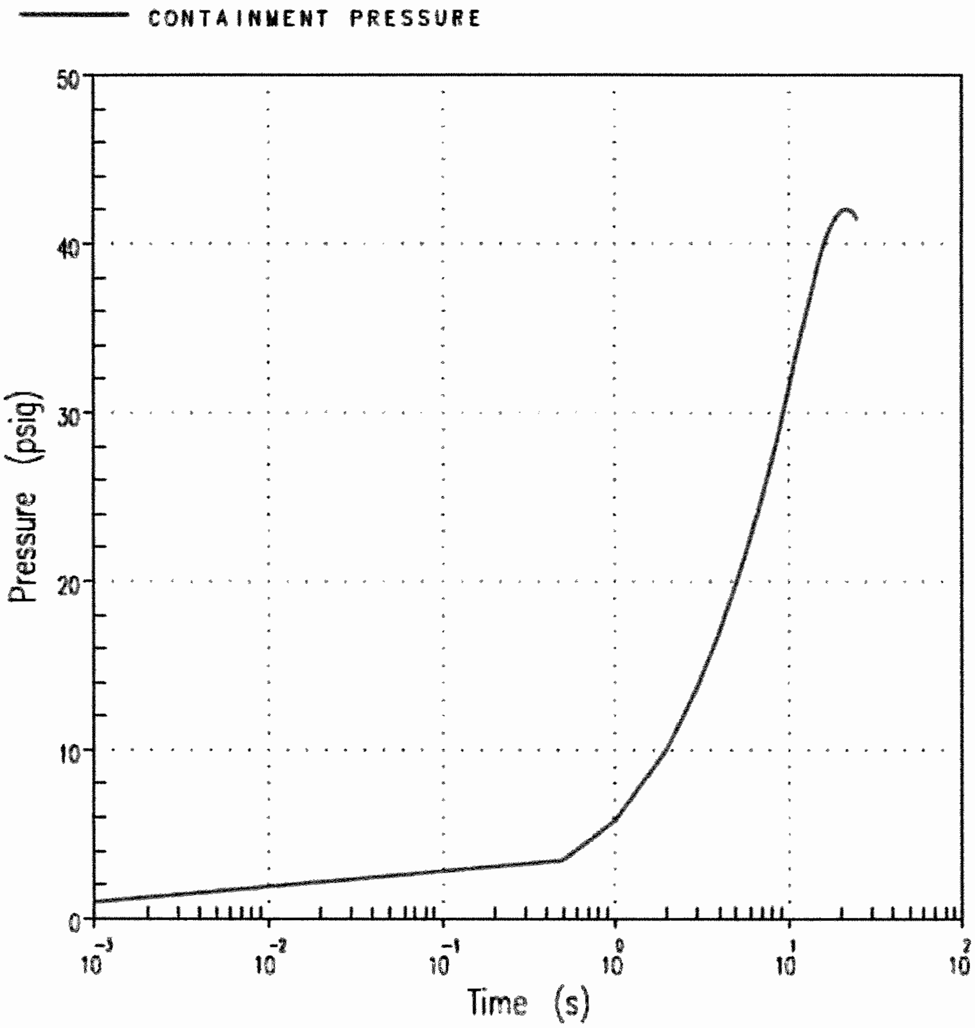
SCVB-RAI-4 – Figure 1: Braidwood Unit 1 UHS Temperature Increase Analysis Double Ended Pump Suction Break with Minimum ECCS Flows Containment Pressure Transient



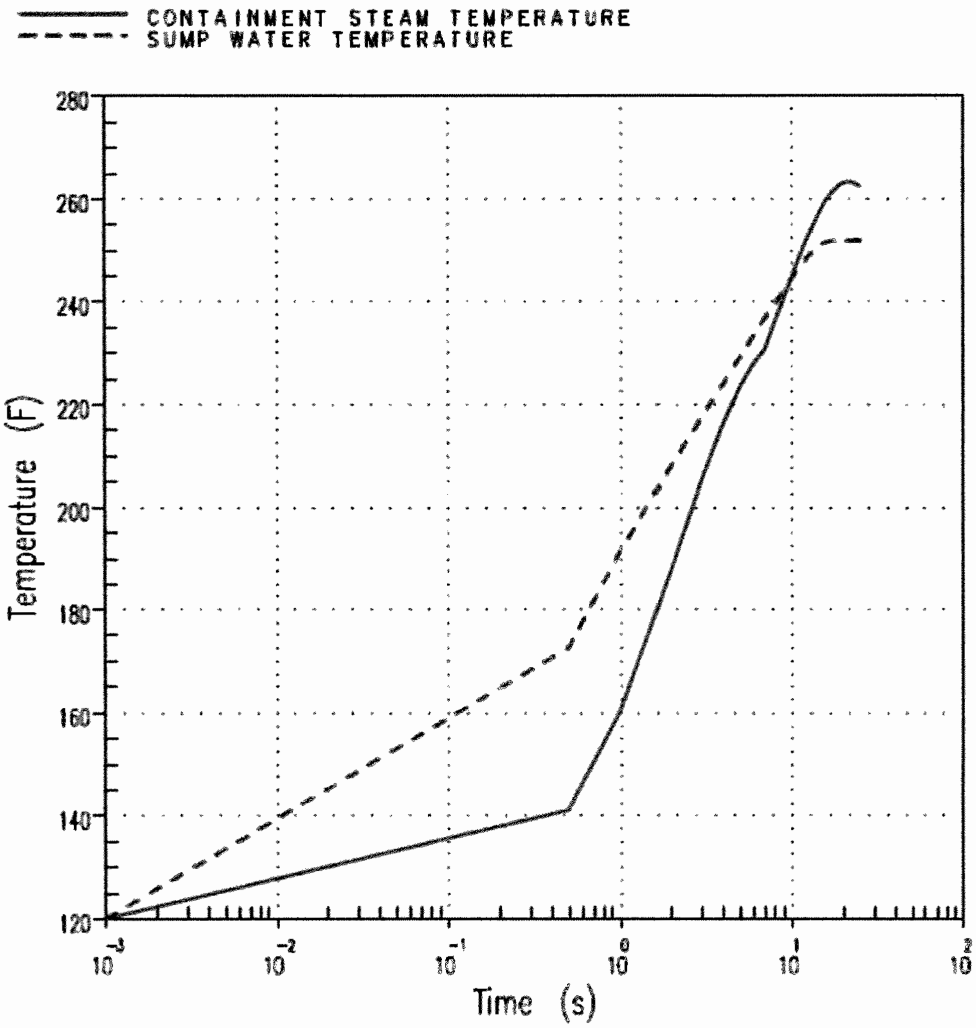
SCVB-RAI-4 – Figure 2: Braidwood Unit 1 UHS Temperature Increase Analysis

Double Ended Pump Suction Break with Minimum ECCS Flows

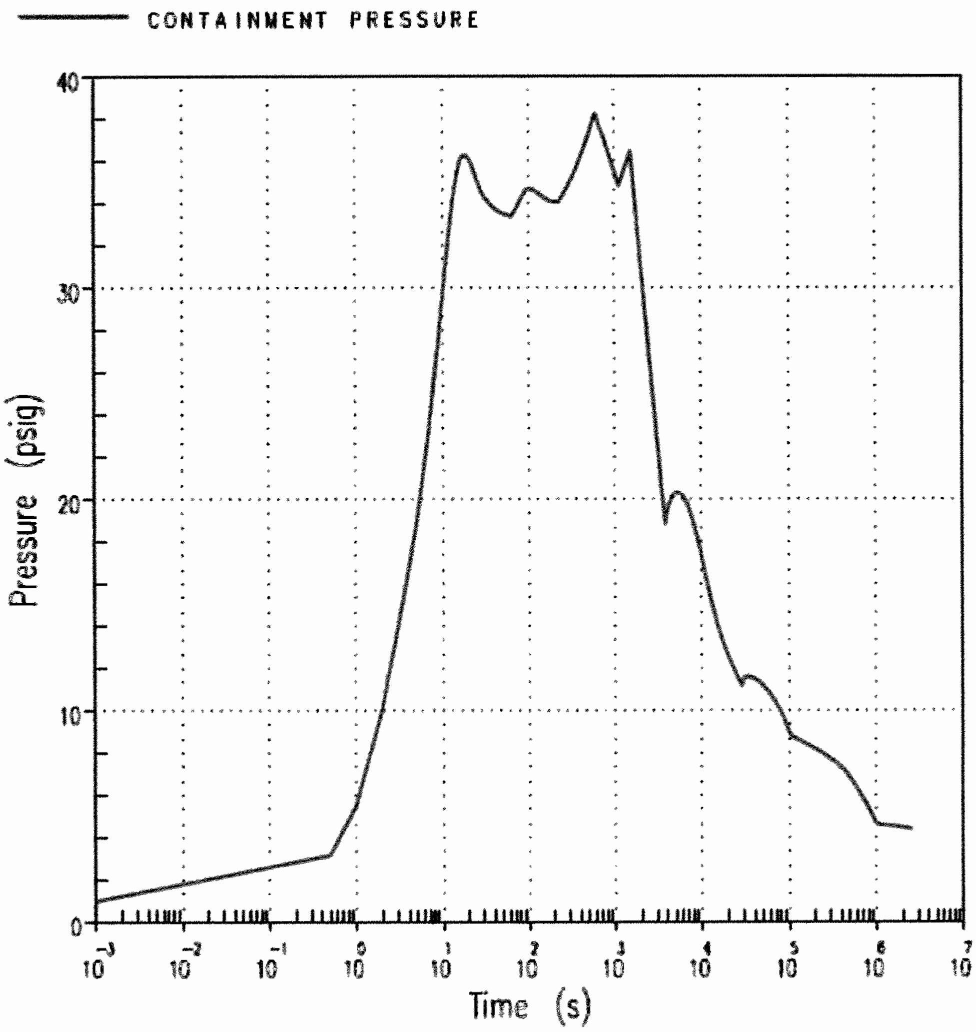
Containment Temperature Transient



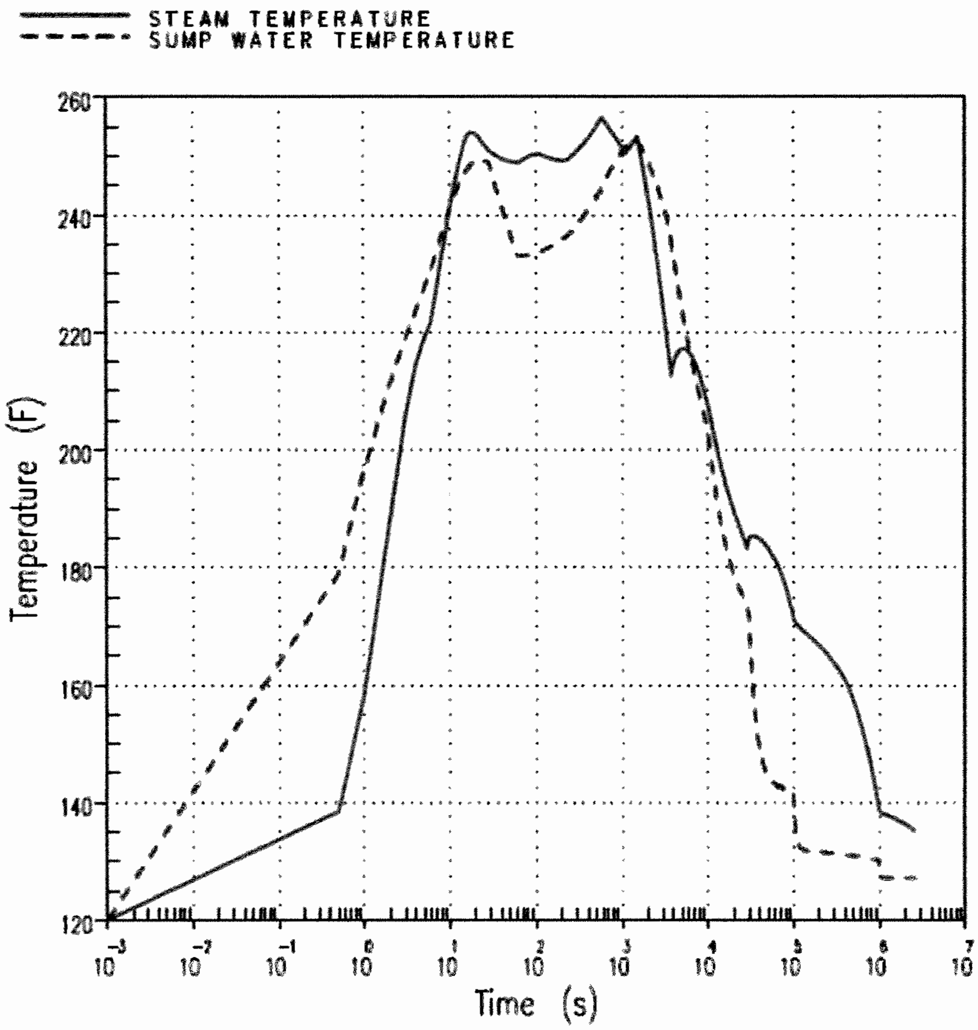
SCVB-RAI-4 – Figure 3: Braidwood Unit 1 UHS Temperature Increase Analysis
Double Ended Hot Leg Break with Minimum ECCS Flows
Containment Pressure Transient



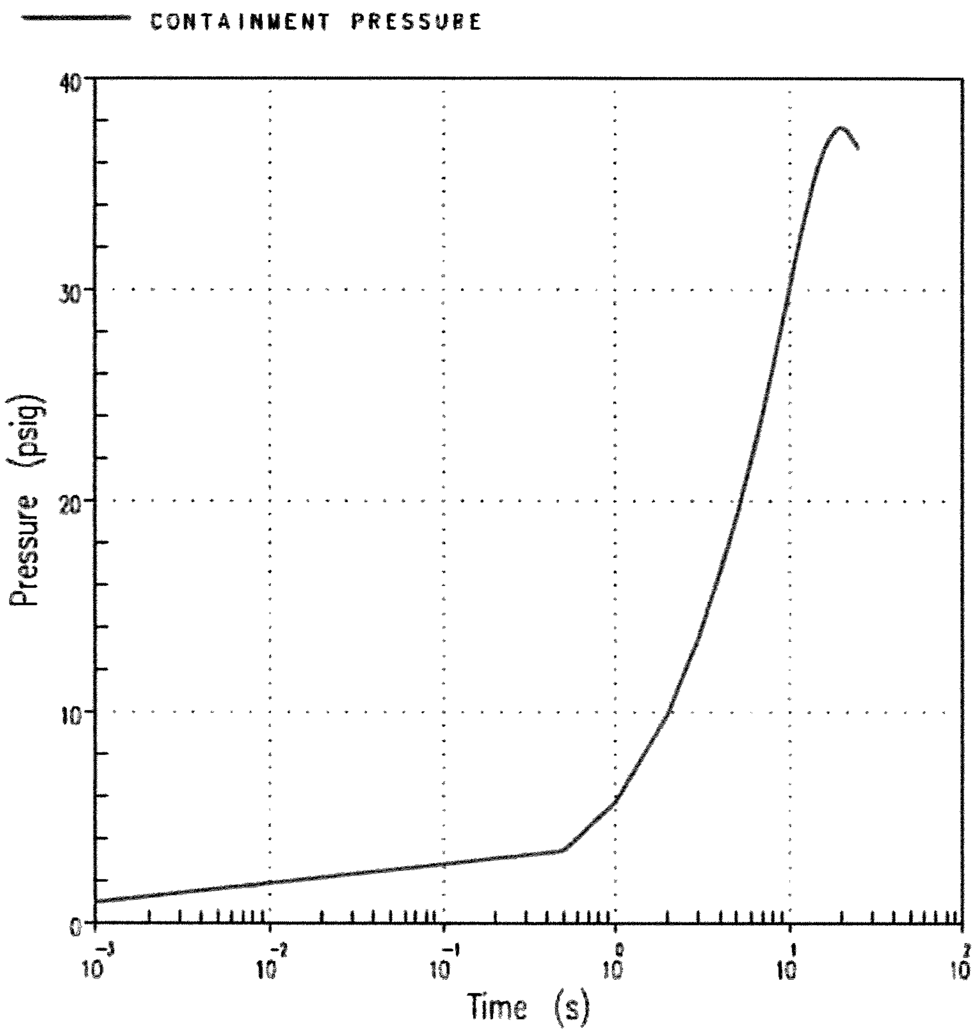
SCVB-RAI-4 – Figure 4: Braidwood Unit 1 UHS Temperature Increase Analysis
Double Ended Hot Leg Break with Minimum ECCS Flows
Containment Temperature Transient



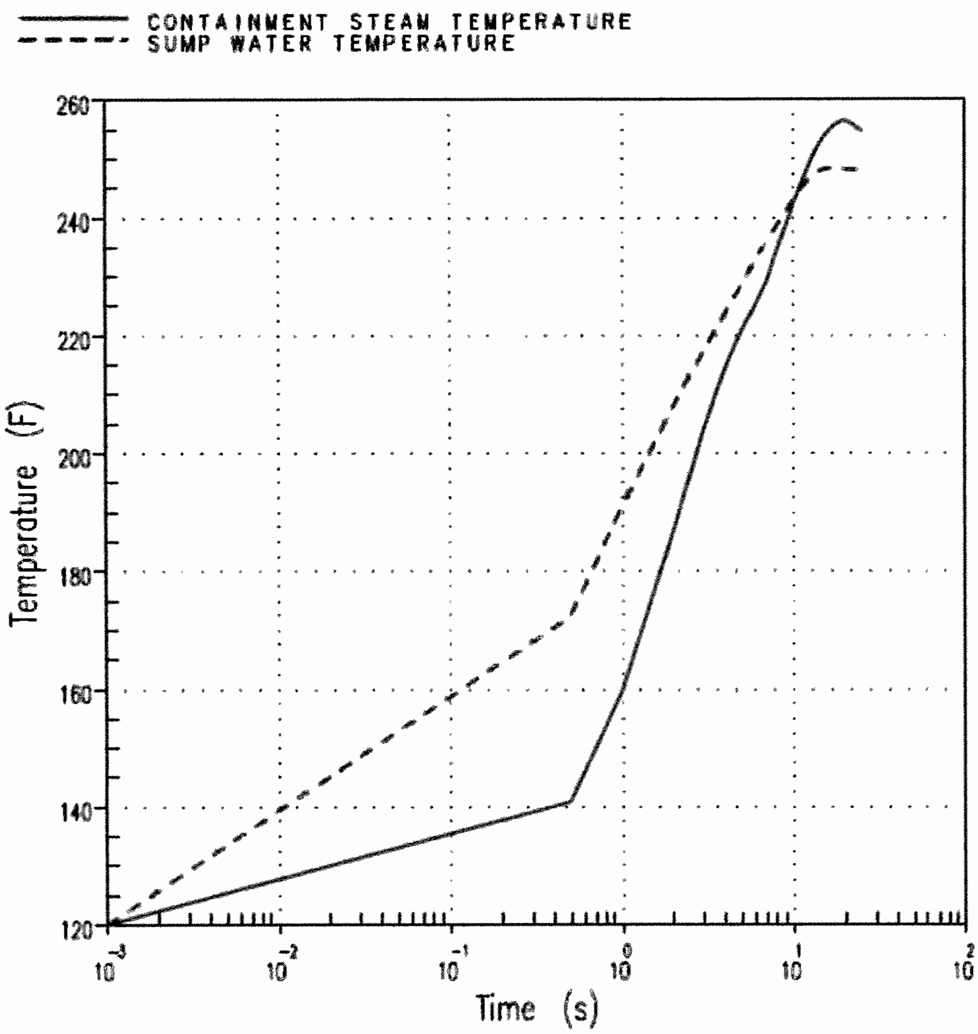
SCVB-RAI-4 – Figure 5: Braidwood Unit 2 UHS Temperature Increase Analysis
Double Ended Pump Suction Break with Minimum ECCS Flows
Containment Pressure Transient



SCVB-RAI-4 – Figure 6: Braidwood Unit 2 UHS Temperature Increase Analysis Double Ended Pump Suction Break with Minimum ECCS Flows Containment Temperature Transient



SCVB-RAI-4 – Figure 7: Braidwood Unit 2 UHS Temperature Increase Analysis Double Ended Hot Leg with Minimum ECCS Flows Containment Pressure Transient



SCVB-RAI-4 – Figure 8: Braidwood 2 Unit UHS Temperature Increase Analysis

Double Ended Hot Leg Break with Minimum ECCS Flows

Containment Temperature Transient

References:

1. "Westinghouse Mass and Energy Release Data For Containment Design," WCAP-8264-P-A, Rev. 1, August 1975 (Proprietary), WCAP-8312-A, Revision 2 Non-Proprietary).
2. WCAP-8077, "Ice Condenser Containment Pressure Transient Analysis Method," March 1973.
3. WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version", May, 1983.
4. 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].
5. EPRI 294-2, "Mixing of Emergency Core Cooling Water with Steam; 1/3-Scale Test and Summary," Final Report, June 1975. [Alternate name WCAP-8423]
6. Letter from Mr. Charles E Rossi (NRC) to Mr. William J. Johnson (W), "Acceptance for Referencing of Licensing Topical Report WCAP-10325, 'Westinghouse LOCA Mass and Energy Release Model for Containment Design (Proprietary) - March 1979 Version'," February 17, 1987 (A copy can be found in Reference 3 cited above).
7. Letter from Herbert N. Berkow (NRC) to Mr. James A. Gresham (Westinghouse): Acceptance of Clarifications of Topical Report WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version" (TAC No. MC7980); October 18, 2005. [ADAMS Accession No. ML052660242]
8. "Containment Pressure Analysis Code (COCO)," WCAP-8327, July 1974 (Proprietary), WCAP-8326, July 1974 (Non-Proprietary).
9. NSAL-14-2, "Westinghouse Loss-of-Coolant Accident Mass and Energy Release Calculation Issue for Steam Generator Tube Material properties," March 31, 2014.
10. Letter from Joel S. Wiebe, Senior Project Manager (NRC) to Michael J. Pacilio, Exelon Generation Company, LLC, Braidwood Station, Units 1 and 2, and Byron Station, Unit Nos. 1 and 2 – Issuance of Amendments Regarding Measurement Uncertainty Recapture Power Uprate (TAC Nos. MF2418, MF2419, MF2420, and MF2421," February 7, 2014. (Accession No. ML13281A000)

SCVB-RAI-5

Refer to Sections 3.6.2.1 and 3.6.2.2; what methodology is used for the proposed MSLB analysis compared to the current analysis, and is the methodology up to date with all errors corrected?

Westinghouse Response

The analysis methodology is the same as utilized in the previous analysis of the SLB M&E releases inside containment as documented in the current analysis of Byron and Braidwood Units 1 and Units 2. The analysis of the SLB M&E releases inside containment is based on the approved methods documented in WCAP-8822 and its supplements (Reference 1). Similarly, the analysis of the SLB containment response is based on the methods documented in WCAP-8327 (Reference 2). The same single failures as currently existing in the current analysis (UFSAR Section 6.2.1.4.3) have been assumed:

Main Steam Isolation Valve (MSIV) Failure

The main steamline isolation function is accomplished via the MSIV in each of the three steamlines. Each valve closes on an isolation signal to terminate steam flow from the associated steam generator. The postulated steamline break upstream of this valve creates a situation in which the steam generator on the faulted loop cannot be isolated, even when the MSIV successfully closes. The break location allows a continued blowdown from the faulted-loop steam generator until it is empty and all sources of main and auxiliary feedwater addition are terminated. If the faulted-loop MSIV fails to close, blowdown from more than one steam generator is prevented by the closure of the corresponding MSIV for each intact-loop steam generator. Therefore, there is no failure of a single MSIV that could cause continued blowdown from multiple steam generators. In addition to the continued steam generator blowdown, the steam in the unisolable section of the steamline is considered.

Main Feedwater Isolation Valve (FIV) Failure

If the FIV in the main feedwater line to the faulted steam generator is assumed to fail in the open position, isolation is provided via closure of the backup valve in the main feedwater system. The additional inventory between the FIV and the backup valve in the faulted loop would be available to be released through the break.

Containment Safeguards Equipment Failure

Failure of a diesel generator would result in the loss of one containment safeguards train resulting in minimum heat removal capability. However, a diesel generator failure is only considered for cases that assume a loss of offsite power. For cases that assume offsite power is available, the limiting single failure in the containment safeguards system for SLB has been determined to be the failure of one reactor containment fan cooler (RCFC) train that results in the loss of 2 RCFC units. Failure of one containment spray pump was shown, by plant-specific analyses, to be less limiting than the failure of one RCFC train.

Westinghouse is not aware of errors associated with the analysis method for the SLB M&E releases or the SLB containment response analysis method.

1) WCAP-8822, "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822-S1-P-A, "Mass and Energy Releases Following a Steam Line Rupture, Supplement 1 – Calculation of Steam Superheat in Mass/Energy Releases Following a Steamline Rupture," September 1986; WCAP-8822-S2-P-A, "Mass and Energy Releases Following a Steam Line Rupture, Supplement 2 – Impact of Steam Superheat in Mass/Energy Releases Following a Steamline Rupture for Dry and Subatmospheric Containment Designs," September 1986.

2) WCAP-8327, "Containment Pressure Analysis Code (COCO)," July 1974.

SCVB-RAI-6

Refer to Section 3.6.2.1; for both units, provide the following information:

(a) Describe the MSLB cases analyzed for containment peak temperature, and provide their comparison with the cases analyzed in the current analysis.

(b) If other than the currently analyzed cases were selected, provide basis for their selection.

(c) Provide a comparison of the inputs and assumptions in the proposed analysis that were changed from the current analysis. Provide justification for those inputs and assumptions in which the conservatism in the proposed analysis is reduced.

(d) Explain why the peak containment temperatures are less in the proposed analysis than in the current analysis.

(e) Provide the graph of the most limiting MSLB peak temperature profile case.

Westinghouse Response – SCVB-RAI-6 – Item (a)

The case definitions for both Units 1 and Units 2 are the same as documented for the current licensing bases. This analysis consists of four power levels (100%, 70%, 30%, and 0%) with the following break sizes:

- 1.1 ft^2 (Units 1)/1.4 ft^2 (Units 2) full double-ended rupture (DER) at the outlet nozzle of the steam generator (SG); the break area is defined by the cross-sectional area of the integral flow restrictor in the nozzle.
- 1.0 ft^2 small DER at the outlet nozzle of the steam generator.
- Split breaks downstream of the outlet nozzle of the steam generator, of varying areas, dependent on the initial power.

The steamline break spectrum for Byron/Braidwood Units 1 is shown in Table 1 and the break spectrum for Byron/Braidwood Units 2 is shown in Table 2. A comparison of the peak temperature results is presented in Table 3 through Table 6. The limiting peak temperature values for the Current Analysis are documented in Table 6.2-1 and Table 6.2-1a of the UFSAR.

SCVB-RAI-6 – Table 1
Units 1 Steamline Break Inside Containment Case Definition

Case Number	Break Type	Break Size	Initial Power Level	Single Failure	Offsite Power Available	
1	Full DER	1.1 ft^2	100%	RCFC	Yes	No
2	Full DER	1.1 ft^2	70%	RCFC	Yes	No
3	Full DER	1.1 ft^2	30%	RCFC	Yes	No
4	Full DER	1.1 ft^2	0%	RCFC	Yes	No
5	Full DER	1.1 ft^2	100%	FIV	Yes	No
6	Full DER	1.1 ft^2	70%	FIV	Yes	No
7	Full DER	1.1 ft^2	30%	FIV	Yes	No
8	Full DER	1.1 ft^2	0%	FIV	Yes	No
9	Full DER	1.1 ft^2	100%	MSIV	Yes	No
10	Full DER	1.1 ft^2	70%	MSIV	Yes	No
11	Full DER	1.1 ft^2	30%	MSIV	Yes	No
12	Full DER	1.1 ft^2	0%	MSIV	Yes	No
13	Small DER	1.0 ft^2	100%	RCFC	Yes	No
14	Small DER	1.0 ft^2	70%	RCFC	Yes	No
15	Small DER	1.0 ft^2	30%	RCFC	Yes	No
16	Small DER	1.0 ft^2	0%	RCFC	Yes	No
17	Small DER	1.0 ft^2	100%	FIV	Yes	No
18	Small DER	1.0 ft^2	70%	FIV	Yes	No
19	Small DER	1.0 ft^2	30%	FIV	Yes	No
20	Small DER	1.0 ft^2	0%	FIV	Yes	No
21	Small DER	1.0 ft^2	100%	MSIV	Yes	No
22	Small DER	1.0 ft^2	70%	MSIV	Yes	No
23	Small DER	1.0 ft^2	30%	MSIV	Yes	No
24	Small DER	1.0 ft^2	0%	MSIV	Yes	No
25	Split	1.00 ft^2	100%	RCFC	Yes	No
26	Split	0.96 ft^2	70%	RCFC	Yes	No
27	Split	0.90 ft^2	30%	RCFC	Yes	No
28	Split	0.64 ft^2	0%	RCFC	Yes	No
29	Split	1.00 ft^2	100%	FIV	Yes	No
30	Split	0.96 ft^2	70%	FIV	Yes	No
31	Split	0.90 ft^2	30%	FIV	Yes	No
32	Split	0.64 ft^2	0%	FIV	Yes	No
33	Split	1.00 ft^2	100%	MSIV	Yes	No
34	Split	0.96 ft^2	70%	MSIV	Yes	No
35	Split	0.90 ft^2	30%	MSIV	Yes	No
36	Split	0.64 ft^2	0%	MSIV	Yes	No

SCVB-RAI-6 – Table 2
Units 2 Steamline Break Inside Containment Case Definition

Case Number	Break Type	Break Size	Initial Power Level	Single Failure	Offsite Power Available	
1	Full DER	1.4 ft^2	100%	RCFC	Yes	No
2	Full DER	1.4 ft^2	70%	RCFC	Yes	No
3	Full DER	1.4 ft^2	30%	RCFC	Yes	No
4	Full DER	1.4 ft^2	0%	RCFC	Yes	No
5	Full DER	1.4 ft^2	100%	FIV	Yes	No
6	Full DER	1.4 ft^2	70%	FIV	Yes	No
7	Full DER	1.4 ft^2	30%	FIV	Yes	No
8	Full DER	1.4 ft^2	0%	FIV	Yes	No
9	Full DER	1.4 ft^2	100%	MSIV	Yes	No
10	Full DER	1.4 ft^2	70%	MSIV	Yes	No
11	Full DER	1.4 ft^2	30%	MSIV	Yes	No
12	Full DER	1.4 ft^2	0%	MSIV	Yes	No
13	Small DER	1.0 ft^2	100%	RCFC	Yes	No
14	Small DER	1.0 ft^2	70%	RCFC	Yes	No
15	Small DER	1.0 ft^2	30%	RCFC	Yes	No
16	Small DER	1.0 ft^2	0%	RCFC	Yes	No
17	Small DER	1.0 ft^2	100%	FIV	Yes	No
18	Small DER	1.0 ft^2	70%	FIV	Yes	No
19	Small DER	1.0 ft^2	30%	FIV	Yes	No
20	Small DER	1.0 ft^2	0%	FIV	Yes	No
21	Small DER	1.0 ft^2	100%	MSIV	Yes	No
22	Small DER	1.0 ft^2	70%	MSIV	Yes	No
23	Small DER	1.0 ft^2	30%	MSIV	Yes	No
24	Small DER	1.0 ft^2	0%	MSIV	Yes	No
25	Split	0.81 ft^2	100%	RCFC	Yes	No
26	Split	0.82 ft^2	70%	RCFC	Yes	No
27	Split	0.83 ft^2	30%	RCFC	Yes	No
28	Split	0.62 ft^2	0%	RCFC	Yes	No
29	Split	0.81 ft^2	100%	FIV	Yes	No
30	Split	0.82 ft^2	70%	FIV	Yes	No
31	Split	0.83 ft^2	30%	FIV	Yes	No
32	Split	0.62 ft^2	0%	FIV	Yes	No
33	Split	0.81 ft^2	100%	MSIV	Yes	No
34	Split	0.82 ft^2	70%	MSIV	Yes	No
35	Split	0.83 ft^2	30%	MSIV	Yes	No
36	Split	0.62 ft^2	0%	MSIV	Yes	No

SCVB-RAI-6 – Table 3
Units 1 Steamline Break Inside Containment Cases
Peak Temperature Results (No LOOP)

Case	Peak Temperature (Proposed Analysis)	Peak Temperature (Current Analysis)
	(°F)	(°F)
Case 1	245.2	NL*
Case 2	245.8	NL*
Case 3	245.8	278
Case 4	248.0	255
Case 5	247.7	NL*
Case 6	245.9	NL*
Case 7	243.9	NL*
Case 8	245.5	NL*
Case 9	248.5	NL*
Case 10	248.0	NL*
Case 11	248.8	NL*
Case 12	248.5	NL*
Case 13	307.4	307
Case 14	305.8	305
Case 15	303.8	302
Case 16	294.5	299
Case 17	298.8	NL*
Case 18	297.5	NL*
Case 19	294.5	NL*
Case 20	286.3	NL*
Case 21	322.1	327

Case	Peak Temperature (Proposed Analysis)	Peak Temperature (Current Analysis)
	(°F)	(°F)
Case 22	321.3	NL*
Case 23	317.5	NL*
Case 24	308.9	NL*
Case 25	318.4	318
Case 26	314.9	313
Case 27	309.4	NL*
Case 28	287.7	NL*
Case 29	309.6	NL*
Case 30	305.7	NL*
Case 31	299.8	NL*
Case 32	276.7	NL*
Case 33	310.0	NL*
Case 34	306.1	NL*
Case 35	300.2	NL*
Case 36	276.9	NL*

NL* not limiting as reported in UFSAR Table 6.2-1

SCVB-RAI-6 – Table 4
Units 1 Steamline Break Inside Containment Cases
Peak Temperature Results (LOOP)

Case	Peak Temperature (Proposed Analysis)	Peak Temperature (Current Analysis)
	(°F)	(°F)
Case 1	237.2	284
Case 2	238.4	NL*
Case 3	239.7	278
Case 4	234.8	NL*
Case 5	240.2	284
Case 6	241.7	NL*
Case 7	243.2	NL*
Case 8	234.9	NL*
Case 9	248.0	NL*
Case 10	246.1	NL*
Case 11	245.2	NL*
Case 12	242.7	NL*
Case 13	311.6	NL*
Case 14	310.3	NL*
Case 15	307.7	NL*
Case 16	300.4	NL*
Case 17	311.8	NL*
Case 18	310.3	NL*
Case 19	307.7	NL*
Case 20	300.4	NL*
Case 21	328.2	334

Case	Peak Temperature (Proposed Analysis)	Peak Temperature (Current Analysis)
	(°F)	(°F)
Case 22	327.0	NL*
Case 23	324.1	NL*
Case 24	316.3	NL*
Case 25	327.7	327
Case 26	324.3	NL*
Case 27	319.0	318
Case 28	293.1	NL*
Case 29	327.7	NL*
Case 30	324.3	NL*
Case 31	319.0	NL*
Case 32	293.1	NL*
Case 33	329.0	NL*
Case 34	325.5	NL*
Case 35	320.0	NL*
Case 36	293.6	NL*

NL* not limiting as reported in UFSAR Table 6.2-1

SCVB-RAI-6 – Table 5
Units 2 Steamline Break Inside Containment Cases
Peak Temperature Results (No LOOP)

Case	Peak Temperature (Proposed Analysis)	Peak Temperature (Current Analysis)
	(°F)	(°F)
Case 1	236.8	NL*
Case 2	242.0	NL*
Case 3	247.1	282.4
Case 4	250.8	280.9
Case 5	241.5	281.7
Case 6	244.6	NL*
Case 7	245.3	NL*
Case 8	248.5	NL*
Case 9	248.2	NL*
Case 10	248.0	NL*
Case 11	250.2	NL*
Case 12	251.7	NL*
Case 13	304.1	NL*
Case 14	303.4	305.6
Case 15	302.8	304.0
Case 16	294.4	299.1
Case 17	295.2	NL*
Case 18	294.8	NL*
Case 19	293.6	NL*
Case 20	287.1	NL*
Case 21	318.9	326.3

Case	Peak Temperature (Proposed Analysis)	Peak Temperature (Current Analysis)
	(°F)	(°F)
Case 22	318.2	NL*
Case 23	316.4	NL*
Case 24	309.1	NL*
Case 25	307.3	307.0
Case 26	306.5	NL*
Case 27	305.5	NL*
Case 28	286.8	NL*
Case 29	298.2	NL*
Case 30	297.0	NL*
Case 31	296.0	NL*
Case 32	276.6	NL*
Case 33	298.5	NL*
Case 34	297.5	NL*
Case 35	296.3	NL*
Case 36	276.4	NL*

NL* not limiting as reported in UFSAR Table 6.2-1a

SCVB-RAI-6 – Table 6
Units 2 Steamline Break Inside Containment Cases
Peak Temperature Results (LOOP)

Case	Peak Temperature (Proposed Analysis)	Peak Temperature (Current Analysis)
	(°F)	(°F)
Case 1	234.1	281.7
Case 2	236.5	NL*
Case 3	239.9	282.4
Case 4	238.1	NL*
Case 5	237.0	NL*
Case 6	239.8	NL*
Case 7	243.6	NL*
Case 8	238.1	NL*
Case 9	248.2	NL*
Case 10	248.0	NL*
Case 11	248.7	NL*
Case 12	246.8	NL*
Case 13	306.3	NL*
Case 14	306.4	NL*
Case 15	306.4	310.6
Case 16	300.7	NL*
Case 17	306.8	NL*
Case 18	306.3	NL*
Case 19	306.5	NL*
Case 20	300.6	NL*
Case 21	323.0	330.8

Case	Peak Temperature (Proposed Analysis)	Peak Temperature (Current Analysis)
	(°F)	(°F)
Case 22	323.0	NL*
Case 23	322.7	NL*
Case 24	316.3	NL*
Case 25	315.2	322.6
Case 26	315.1	NL*
Case 27	314.9	317.7
Case 28	292.6	NL*
Case 29	315.4	NL*
Case 30	315.0	NL*
Case 31	315.2	NL*
Case 32	292.5	NL*
Case 33	316.0	NL*
Case 34	316.0	NL*
Case 35	315.8	NL*
Case 36	293.1	NL*

NL* not limiting as reported in UFSAR Table 6.2-1a

Westinghouse Response – SCVB-RAI-6 – Item (b)

The break spectrum for the proposed analyzed cases for both Units 1 and Units 2 is the same as documented for the current licensing bases. The only differences are the range of power levels is 100%, 70%, 30%, and 0% of the NSSS current power uprate to 3672 Mwt (this is discussed in paragraph d. in UFSAR Section 6.2.1.4.1), and the split break areas have been defined (using the method in Section 2.1 of WCAP-8822) based on the current power uprate to 3672 Mwt.

Westinghouse Response – SCVB-RAI-6 – Item (c)

The following changes to the analysis are due to the increase in the UHS temperature.

SLB M&E releases: The auxiliary feedwater (AFW) enthalpy input to LOFTRAN was revised to a plant-specific value of 80.2 Btu/lbm (based on 104°F) from the generic bounding value of 91.12 Btu/lbm (based on 120°F) in the current analysis. This input represents a value in the less limiting direction but is justified since it is plant specific to the Byron and Braidwood units.

SLB containment response: The RCFC heat removal has been adjusted based on the 104°F UHS temperature as shown in the table below. The heat removal capacity has been reduced in the proposed analysis compared to the current analysis.

SCVB-RAI-6 – Table 7
RCFC Heat Removal Rate, per fan cooler

Containment Temperature (°F)	Proposed Analysis Heat Removal (Btu/sec)	Current Analysis Heat Removal (Btu/sec)
100	0	0
110	600.25	893.54
120	1732.94	1946.27
130	3042.98	3181.61
160	8098.00	8057.82
190	14509.36	14535.02
220	21558.78	21896.92
271	33702.04	34613.85
300	40508.12	41813.19
350	52193.28	54225.24

The following changes to the analysis were also included in the models to reflect updated inputs.

SLB M&E releases: Revised minimum safety injection flowrates from one charging pump were assumed in the calculation of the M&E releases as shown in the table below. The minimum flowrates increased slightly over the range of RCS pressures 2250 psia to 300 psia attributed to the use of updated piping flow software used to calculate the safety injection pump flowrate. SLB M&E releases are not sensitive to small changes in the safety injection flowrates since the intent is to provide boron to the core rather than water to the RCS. Since minimum boron concentrations are used in the SLB safety analysis, and the boron worth for reactivity changes is also minimized, the effect of small changes are insignificant. The dominant parameter for SLB M&E releases is the moderator temperature, steamline isolation, and the dryout of the SG. There would be no changes to the conclusions of the calculations for small changes in safety injection flowrates.

SCVB-RAI-6 – Table 8
Minimum Flow from One High-Head Safety Injection Pump

RCS Pressure (psia)	Proposed Analysis Flowrate (gpm)	Current Analysis Flowrate (gpm)
15	407.0	420.6
100	402.0	412.9
300	395.0	390.3
400	385.0	380.3
500	375.0	369.7
600	364.0	359.0
700	354.3	348.3
800	343.3	336.8
900	331.0	324.6
1000	319.6	312.3
1100	307.3	299.8
1200	294.7	286.4
1300	281.0	272.8
1400	239.2	219.5
1500	224.0	203.5
1600	207.0	184.6
1700	189.0	165.2
1800	169.6	144.6
1900	150.0	123.1
2000	128.5	99.7
2100	105.9	74.9
2200	63.0	35.1
2250	33.7	0.0
2282	0.0	0.0

SLB containment response: The containment spray pump flowrates were revised from 3285 gpm to 3113 gpm for Train A and from 3795 gpm to 3475 gpm for Train B. When there is no assumed loss of offsite power, both trains are available and the containment response analysis assumes 6588 gpm. When there is an assumed LOOP, the minimum value from a single train, 3113 gpm, is assumed in the containment response analysis. The spray flowrate reduction reduces the containment heat removal capability in the proposed analysis compared to the current analysis.

Westinghouse Response – SCVB-RAI-6 – Item (d)

There are differing reasons for the decrease in the containment temperatures depending on the SLB size and how the analysis models have been developed.

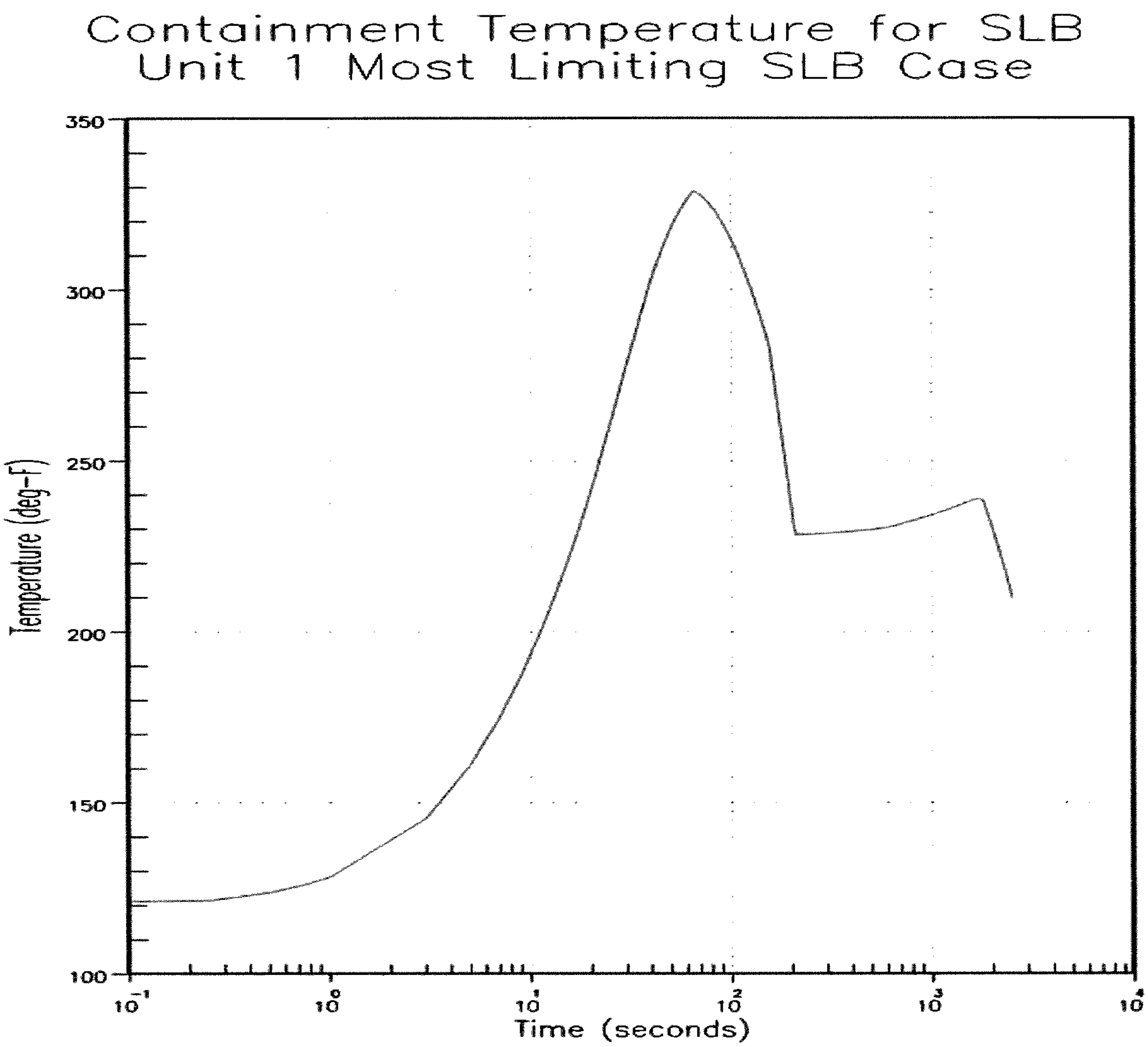
In the 2000 power uprate analysis, credit was taken for the reduced cross-sectional flow area of the MSIV in the faulted loop compared to the cross-sectional flow area of the steam piping. This produced a change in the SLB M&E model related to the blowdown from the main steam piping. The result is that the peak temperatures for the large double-ended rupture (DER) cases shifted from a blowdown peak very early in the transient to a later peak and a reduction of 30 to 40°F in the peak temperature responses for these SLB cases. However, the peak temperatures associated with the large DER are not limiting compared to the transient temperature responses of the small DER cases and the split-break SLB cases, which assume no revaporization of the condensed liquid on the containment surfaces. This is reflected in the temperature differences for Cases 1 through 12 in Tables 3, 4, 5 and 6 of the response to SCVB-RAI-6 (a).

One reason for the reduction in the limiting peak containment temperature as presented in Section 3.6.2.1 of the LAR is the delay time assumed for start of the reactor containment fan coolers (RCFCs). For all MSLB cases analyzed for the UHS temperature increase, the RCFCs were modeled to actuate on the High-1 containment pressure signal plus a delay of ~27 seconds. For the small DER cases and the split-break cases, this resulted in an earlier actuation of the RCFCs and a reduction of 8 to 14°F in the most limiting peak temperature responses compared to the current analysis. For the large DER cases, the peak temperature is impacted more by the updated mass & energy releases than the actuation of the RCFCs. However, for the SLB cases that assume a loss of offsite power (LOOP), the containment temperatures have been recalculated in response to this request assuming actuation of the RCFCs consistent with the diesel start time, 65 seconds for Units 1 and 66.3 seconds for Units 2, and the current analysis of record as described in UFSAR Section 6.2.1.1.3. The containment temperatures in Table 4 and Table 6 reflect the longer time to actuate the RCFC consistent with the diesel start time. The following paragraph is a revision to the text in Section 3.6.2.1 of the LAR based on the analysis with the LOOP delay included.

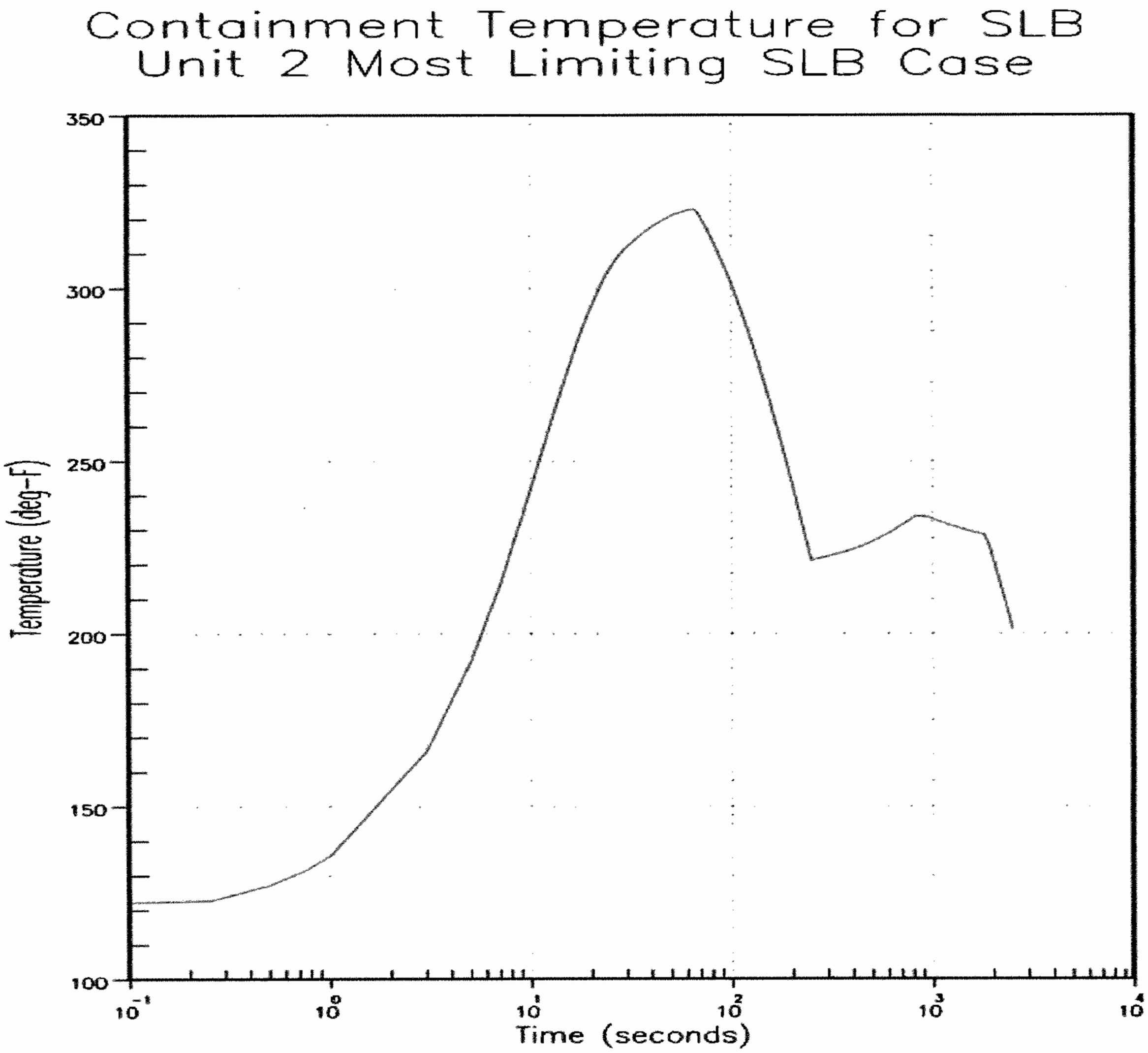
> "The peak temperature response of all the Unit 1 MSLB cases analyzed is 329.0°F. This is lower than the maximum Unit 1 temperature of 333.6°F from the current design basis analyses. The peak temperature response of all the Unit 2 MSLB cases analyzed is 323.0°F. This is lower than the maximum Unit 2 temperature of 326.3°F from the current design basis analyses. These peak temperatures were for a 1.0 ft^2 split break at 100% initial powers, with a single failure of the main steam isolation valve in the faulted loop for Unit 1, and a 1.0 ft^2 small double-ended rupture (DER) at 100% and 70% initial powers, with a single failure of the main steam isolation valve in the faulted loop for Unit 2. The reanalyzed limiting MSLBs assume a loss of offsite power and 0% revaporization of the condensed liquid on the containment surfaces."

The remaining differences in the paragraph above relate to a legacy issue that appears in the older current design basis analyses for the small DER cases. An overly conservative calculation of the SLB M&E releases from the faulted steam generator resulted in a double accounting of the fluid mass in the steam generator until the completion of steamline isolation. The model for the small DER cases has been corrected to reflect the appropriate steam flow from the faulted steam generator. The total break flow combines the faulted steam generator steam flow with the reverse flow from the main steam piping header until steamline isolation has been completed. The reduced M&E releases account for a reduction of 6 to 8°F in the containment temperature affecting Cases 21 through 24 in Tables 3, 4, 5 and 6 of the response to SCVB-RAI-6 (a).

Westinghouse Response – SCVB-RAI-6 – Item (e)



SCVB-RAI-6 – Figure 1: Units 1



SCVB-RAI-6 – Figure 2: Units 2

SCVB-RAI-7

Refer to Section 3.6.2.2; for both units, provide the following information:

(a) Describe the MSLB cases analyzed for containment peak pressure, and provide their comparison with the cases analyzed in the current analysis.

(b) Describe the cases in the proposed and the current analysis that resulted in the maximum peak pressures.

(c) If other than the currently analyzed cases were selected, provide basis for their selection.

(d) Provide a comparison of the inputs and assumptions in the proposed analysis that were changed from the current analysis. Provide justification for those inputs and assumptions in which the conservatism in the proposed analysis is reduced.

(e) Provide the graph of the most limiting MSLB peak pressure profile case.

Westinghouse Response – SCVB-RAI-7 – Item (a)

Refer to answer to item (a) of SCVB-RAI-6 for the description of the proposed analyzed cases, Tables 1 and 2. Only the most limiting of the SLB cases from those tables were analyzed with inputs adjusted to maximize the containment pressure transient. The peak pressure results are shown below in Table 1 for Units 1 and Table 2 for Units 2. The limiting peak pressure values for the Current Analysis are documented in Table 6.2-1 and Table 6.2.1a of the UFSAR.

SCVB-RAI-7 – Table 1
Units 1 Steamline Break Inside Containment
Maximum Initial Containment Pressure Cases
Peak Pressure Results (No LOOP)

Case	Peak Pressure (Proposed Analysis) (psig)	Peak Pressure (Current Analysis) (psig)
Case 4	32.23	37.7
Case 5	32.12	39.3
Case 9	32.58	NL*
Case 10	32.35	NL*
Case 11	32.75	NL*
Case 12	32.50	NL*
Case 15	31.65	NL*
Case 25	32.05	NL*
Case 26	32.29	NL*
Case 27	32.82	NL*
Case 28	32.38	NL*
Case 29	32.62	NL*
Case 30	31.79	NL*
Case 31	31.77	NL*
Case 33	33.67	NL*
Case 34	33.77	NL*
Case 35	34.48	NL*
Case 36	32.44	NL*

NL* not limiting as reported in UFSAR Table 6.2-1

SCVB-RAI-7 – Table 2
Units 2 Steamline Break Inside Containment
Maximum Initial Containment Pressure Cases
Peak Pressure Results (No LOOP)

Case	Peak Pressure (Proposed Analysis)	Peak Pressure (Current Analysis)
	(psig)	(psig)
Case 4	33.83	38.3
Case 8	32.53	NL*
Case 11	33.62	NL*
Case 12	34.32	NL*
Case 15	31.71	NL*
Case 24	32.00	NL*
Case 27	32.89	NL*
Case 28	33.54	NL*
Case 35	34.44	NL*
Case 36	33.47	NL*

NL* not limiting as reported in UFSAR Table 6.2-1a

Westinghouse Response – SCVB-RAI-7 – Item (b)

Proposed Analysis (Units 1)

The peak pressure response of all the SLB cases analyzed is 34.48 psig from Case 35 in Table 1 of SCVB-RAI-6, which is well below the 50 psig design pressure acceptance criterion. This break is a 0.90 ft^2 split break at 30% of 3672 Mwt initial power, with a single failure of the main steam isolation valve in the faulted loop, and with offsite power available.

Current Analysis (Units 1)

The peak pressure response of all the SLB cases analyzed is 39.30 psig (UFSAR Table 6.2-1), which is well below the 50 psig design pressure acceptance criterion. This break is a full double-ended break at 95.1% of 3672 Mwt initial power, with a single failure of the feedwater isolation valve in the faulted loop, and with offsite power available.

Proposed Analysis (Units 2)

The peak pressure response of all the SLB cases analyzed is 34.44 psig from Case 35 in Table 2 of SCVB-RAI-6, which is well below the 50 psig design pressure acceptance criterion. This break is a 0.83 ft^2 split break at 30% of 3672 Mwt initial power, with a single failure of the main steam isolation valve in the faulted loop, and with offsite power available.

Current Analysis (Units 2)

The peak pressure response of all the SLB cases analyzed is 38.30 psig (UFSAR Table 6.2-1a), which is well below the 50 psig design pressure acceptance criterion. This break is a full double-ended break at 0% initial power, with a single failure in the containment safeguards equipment, and with offsite power available.

<u>Westinghouse Response – SCVB-RAI-7 – Item (c)</u>

The break spectrum for the proposed analyzed cases for both Units 1 and Units 2 is the same as documented for the current licensing bases. The only differences are the range of power levels is 100%, 70%, 30%, and 0% of the NSSS current power uprate to 3672 Mwt (this is discussed in paragraph d. in UFSAR Section 6.2.1.4.1), and the split break areas have been defined (using the method in Section 2.1 of WCAP-8822) based on the current power uprate to 3672 Mwt.

Westinghouse Response – SCVB-RAI-7 – Item (d)

The following changes to the analysis are due to the increase in the UHS temperature.

SLB M&E releases: The auxiliary feedwater (AFW) enthalpy input to LOFTRAN was revised to a plant-specific value of 80.2 Btu/lbm (based on 104°F) from the generic bounding value of 91.12 Btu/lbm (based on 120°F) in the current analysis. This input represents a value in the less limiting direction but is justified since it is plant specific to the Byron and Braidwood units.

SLB containment response: The RCFC heat removal has been adjusted based on the 104°F UHS temperature as shown in the table below. The heat removal capacity has been reduced in the proposed analysis compared to the current analysis.

SCVB-RAI-7 – Table 3
RCFC Heat Removal Rate, per Fan Cooler

Containment Temperature (°F)	Proposed Analysis Heat Removal (Btu/sec)	Current Analysis Heat Removal (Btu/sec)
100	0	0
110	600.25	893.54
120	1732.94	1946.27
130	3042.98	3181.61
160	8098.00	8057.82
190	14509.36	14535.02
220	21558.78	21896.92
271	33702.04	34613.85
300	40508.12	41813.19
350	52193.28	54225.24

The following changes to the analysis were also included in the models to reflect updated inputs.

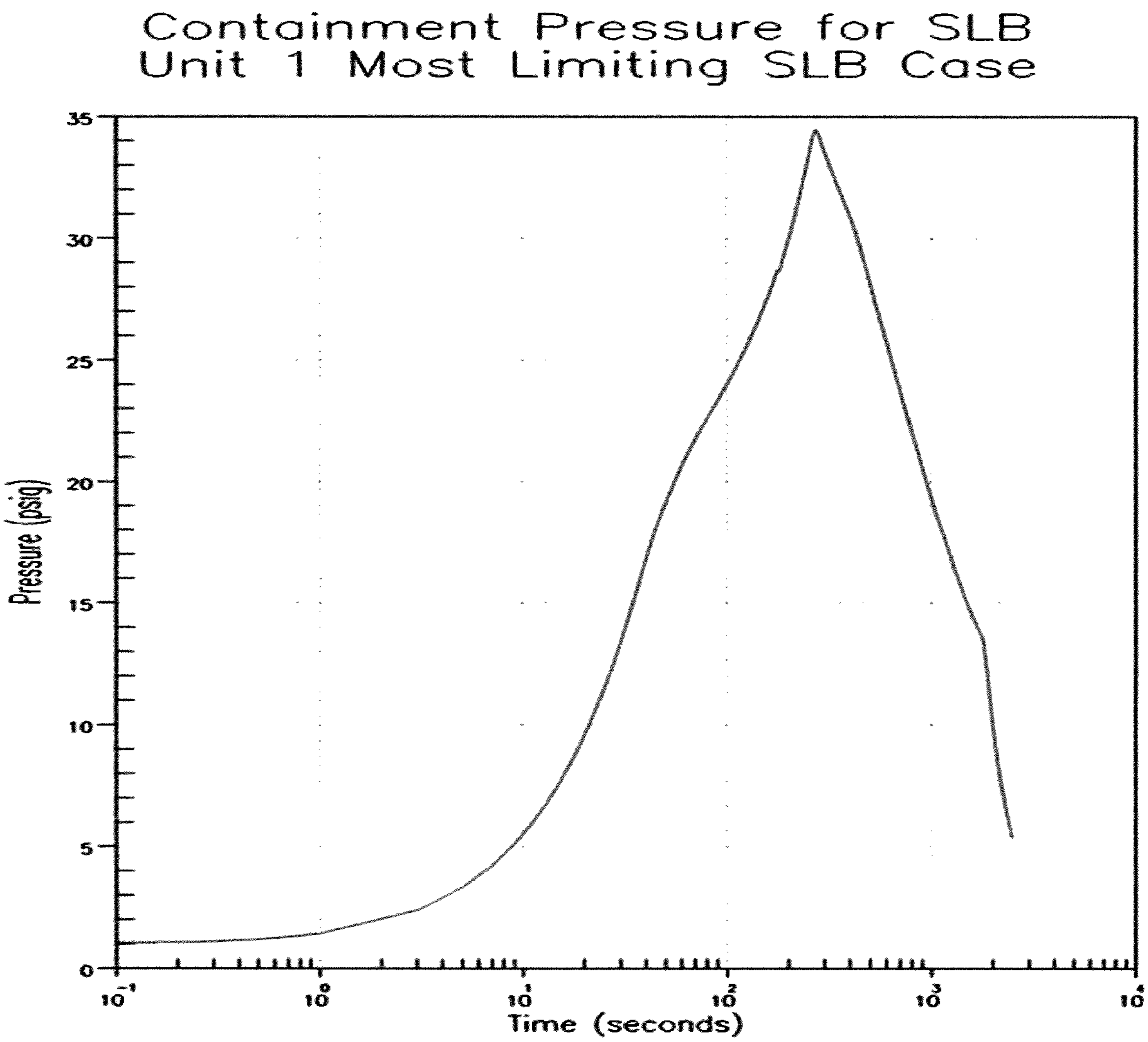
SLB M&E releases: Revised minimum safety injection flowrates from one charging pump were assumed in the calculation of the M&E releases as shown in the table below. The minimum flowrates increased slightly over the range of RCS pressures 2250 psia to 300 psia attributed to the use of updated piping flow software used to calculate the safety injection pump flowrate. SLB M&E releases are not sensitive to small changes in the safety injection flowrates since the intent is to provide boron to the core rather than water to the RCS. Since minimum boron concentrations are used in the SLB safety analysis, and the boron worth for reactivity changes is also minimized, the effect of small changes are insignificant. The dominant parameter for SLB M&E releases is the moderator temperature, steamline isolation, and the dryout of the SG. There would be no changes to the conclusions of the calculations for small changes in safety injection flowrates.

SCVB-RAI-7 – Table 4
Minimum Flow from One High-Head Safety Injection Pump

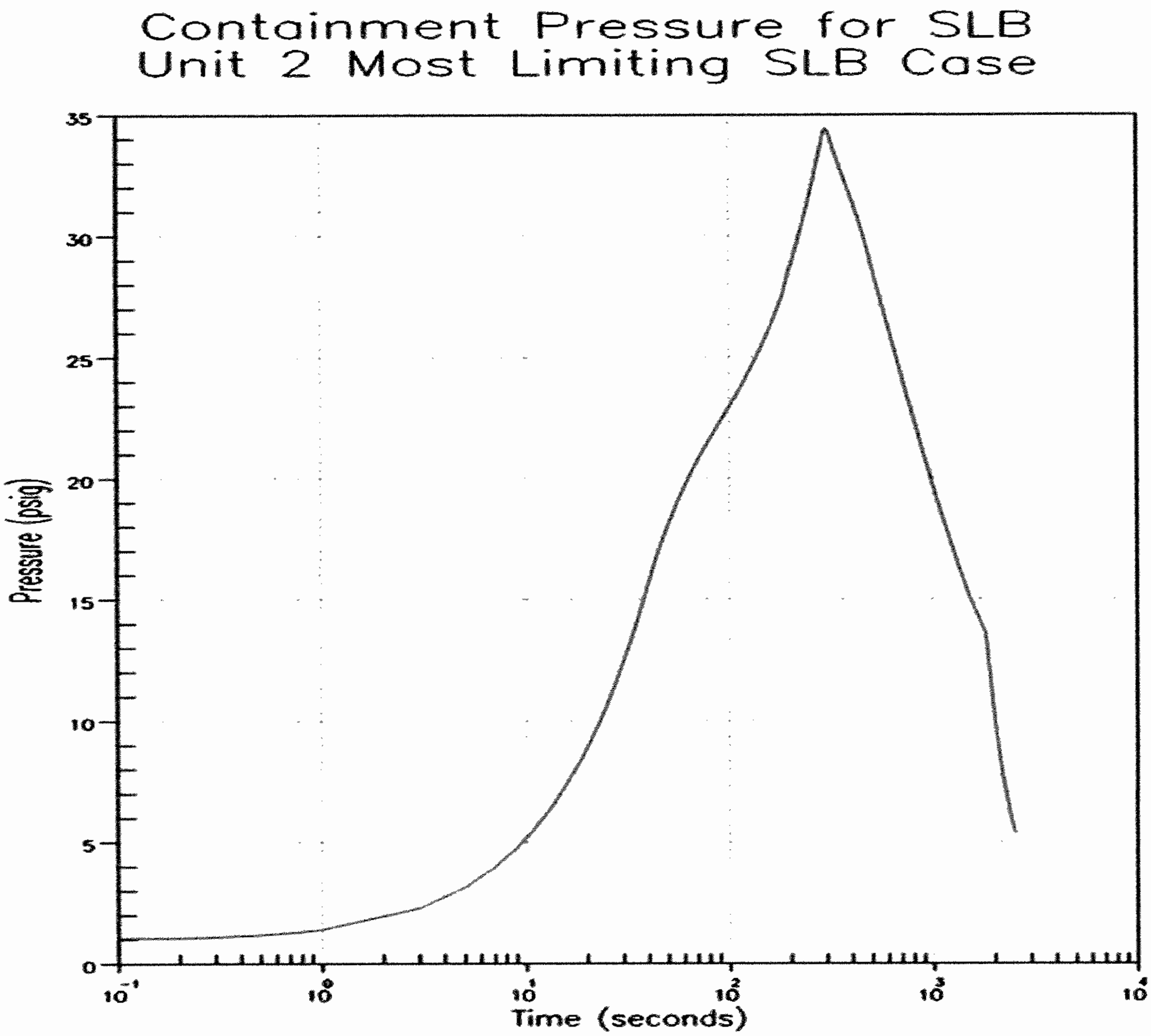
RCS Pressure (psia)	Proposed Analysis Flowrate (gpm)	Current Analysis Flowrate (gpm)
15	407.0	420.6
100	402.0	412.9
300	395.0	390.3
400	385.0	380.3
500	375.0	369.7
600	364.0	359.0
700	354.3	348.3
800	343.3	336.8
900	331.0	324.6
1000	319.6	312.3
1100	307.3	299.8
1200	294.7	286.4
1300	281.0	272.8
1400	239.2	219.5
1500	224.0	203.5
1600	207.0	184.6
1700	189.0	165.2
1800	169.6	144.6
1900	150.0	123.1
2000	128.5	99.7
2100	105.9	74.9
2200	63.0	35.1
2250	33.7	0.0
2282	0.0	0.0

SLB containment response: The containment spray pump flowrates were revised from 3285 gpm to 3113 gpm for Train A and from 3795 gpm to 3475 gpm for Train B. When there is no assumed loss of offsite power, both trains are available and the containment response analysis assumes 6588 gpm. When there is an assumed LOOP, the minimum value from a single train, 3113 gpm, is assumed in the containment response analysis. The spray flowrate reduction reduces the containment heat removal capability in the proposed analysis compared to the current analysis.

Westinghouse Response – SCVB-RAI-7 – Item (e)



SCVB-RAI-7 – Figure 1: Units 1



SCVB-RAI-7 – Figure 2: Units 2

SCVB-RAI-8

NUREG-0800, Standard Review Plan (SRP) Revision 2, July 1981, 6.2.1.5 provides NRC staff review guidance for the minimum containment pressure analysis for emergency core cooling system (ECCS) performance capability. Branch Technical Position CSB 6-1 provides guidance for complying with 10 CFR Part 50, Appendix K, paragraph I.D.2 when calculating the containment pressure response used for evaluating cooling effectiveness during the post-blowdown phase of a LOCA. The Branch Technical Position states that the minimum containment pressure should be calculated by including the effects of containment heat sinks and operation of all pressure-reducing systems. Provide the results of the re-analysis for the minimum containment pressure with the proposed change in the cooling water temperature, including the corrections and revisions to analyses described in your letter dated August 19, 2014.

Alternatively, provide justification why the proposed change in the cooling water temperature, including the corrections and revisions to analyses described in your letter dated August 19, 2014, does not impact the minimum containment pressure analysis.

Westinghouse Response

The Small Break Loss of Coolant Accident (SBLOCA) and Large Break LOCA (LBLOCA) analyses performed for Braidwood Units 1 and 2 model initial operating parameters intended to minimize transient containment backpressure. The Ultimate Heat Sink (UHS) temperature is neither explicitly modeled in the analyses nor does it adversely impact the conditions or systems modeled in the analyses. An increased UHS temperature would only potentially cause an increase in containment pressure, which would be a benefit to the analyses.

As such, the SBLOCA and LBLOCA Emergency Core Cooling System (ECCS) analyses are not impacted by the UHS temperature increase. The existing conservatively low containment backpressure curves remain applicable.