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NRC-98-4

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1-920-433-5544 fax

January 23, 1998

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

Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Request for Additional Information Regarding Proposed Amendment 148a

References:
1) Letter from C. R. Steinhardt (WPSC) to NRC Document Control Desk, "Proposed Amendment 148a to the KNPP Operating License," dated September 25, 1997.
2) Letter from L. N. Tran (NRC) to M. L. Marchi (WPSC), dated November 25, 1997.
3) Wisconsin Public Service Corporation Topical Report WPSRSEM-NP-A, "Reload Safety Evaluation Methods for Application to Kewaunee," Rev. 2, October 1988.
4) Northern States Power Company Topical Report NAD-8102-A4, "Reload Safety Evaluation Methods for Application to PI Units," Rev. 4, June 1986.
5) R. J. Laufer (NRC) to M. L. Marchi (WPSC), dated March 20, 1995.
6) E. R. Mathews (WPSC) to J. G. Keppler (NRC), dated May 7, 1980.

In Reference 1, Wisconsin Public Service Corporation (WPSC) requested a license amendment to allow a relaxation of the main steam line break analyses assumption for closure time of the Main Steam Isolation Valves. On November 25, 1997, the Nuclear Regulatory Commission staff requested additional information (Reference 2). WPSC's response to the two NRC questions is provided as Attachment 1.

Should you require additional information, please contact me or a member of my staff.

Sinccrely,

M L Marchi

Mark L. Marchi
Manager-Nuclear Business Group

A001

JTH/smm
Attach.
cc: US NRC - Region III
US NRC Senior Resident Inspector

ATTACHMENT 1

Letter from M. L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

January 23, 1998

Response to NRC Questions Concerning Proposed Amendment 148a

Response to NRC Questions Concerning Proposed Amendment 148a

1. *Your submittal indicated that a new containment peak pressure analysis has been performed for the Main Steam Line Break (MSLB) using a 5-second Main Steam Isolation Valve (MSIV) closure time with a 2-second allowance for instrumentation delays. The new peak containment pressure is calculated to be 60.7 psia at 96.7 seconds for a break of 1.4 ft^2 at zero power. This pressure is equal to the containment design pressure of 46 psig. In the containment peak pressure analysis with the MSIV closure time of 10 seconds (current FSAR), the containment peak pressure was 60.3 psia at 477 seconds for a 0.5 ft^2 break at zero power (FSAR Table 14.2-2). The 0.5 ft^2 break produced a greater containment pressure than the 1.4 ft^2 break. Your submittal indicates that the new analyses have considerably refined the original licensing analyses. Please confirm that the models used for the new mass and energy analyses and containment response analyses are conservative models. Specifically, if your analyses or portions thereof were performed in-house, rather than by your NSSS vendor or AE using their approved methods, please confirm that the methods and assumptions used conform to Standard Review Plan criteria and industry practice as recommended in ANSI/ANS-56.4-1983 "American National Standard - Pressure and Temperature Transient Analysis for Light Water Reactor Containments." In addition, please provide a discussion of the extent to which your mass and energy release analyses and containment pressure response analyses for MSLB containment peak pressure conform to the recommendations and guidance of the cited ANS Standard. Deviations from the industry practices should be identified and justified.*

WPSC Response

WPSC main steam line break (MSLB) containment response analyses are performed in-house using codes previously accepted by the NRC (References 3 and 4). The WPSC MSLB containment response methods were developed under the guidance of Dr. R. C. Kern of Nuclear Engineering Technology Corporation. The methods have been applied successfully to support Technical Specification Amendment 116 (Ref. 5) and the response to NRC Bulletin 80-04 (Ref. 6) and have been incorporated into the Updated Safety Analysis Report.

A review of the applicable sections of ANSI/ANS 56.4-1983 was performed to confirm that the WPSC containment response analysis methods and assumptions conform to the guidance and recommendations of the standard (Attachment 2). In addition, a review of the following Standard Review Plan sections was performed:

15.1.5 Steam system piping failures inside and outside of containment (PWR)

6.2.1 Containment functional design

6.2.1.4 Mass and energy release analysis for postulated secondary system pipe ruptures

Based on the review of the WPSC MSLB containment response methods and assumptions against ANSI/ANS 56.4-1983 and the applicable NRC Standard Review Plan sections, several deviations from the standards were identified and evaluated through sensitivity analyses as detailed in Attachment 2. All deviations were shown to have a relatively small impact on the containment pressure response and thus the deviations are considered justified. The net effect of the deviations would be a slight reduction in the peak containment pressure calculated by the WPSC model.

It is concluded that the WPSC MSLB containment response and mass and energy release methods and assumptions are, with the minor deviations as discussed above, consistent with the recommendations and criteria of the standards. The WPSC MSLB model is therefore conservatively calculating the containment pressure and temperature response during a main steam line break accident.

2. *In your MSLB analysis, the initial steam generator water level is assumed at 50% of the narrow-range-level span. Please provide the basis of this assumption and show that the assumption is conservative relative to the plant operational restrictions. Also, describe the SG water level assumed in the original MSLB analysis and the basis of that assumption. Confirm that the SG initial water level of 67 ft at the SG wide-range level as indicated in Figure 3 of your September 25, 1997, submittal is consistent with 50% of the narrow-range-level span.*

<u>WPSC Response</u>

a) The basis for the assumption of steam generator water level at 50% of narrow range (NR) span is to bound plant operation. Normal steam generator levels in automatic control are 44% of NR span for power levels above 20% power. At less than 20% power in automatic control, the level is slowly ramped down from 44% to 33%. At hot shutdown, the operators manually control the steam generator levels, typically at about 40% NR span. Witb approval of the proposed changc, the levels at hot shutdown would be typically controlled in a range around 45% of NR span and administratively controlled to achieve a value less than 50% of NR span. Additionally, a turbine trip and a main feedwater isolation will occur at 67% of NR span.

   In the original FSAR main steam line break containment response analysis, the steam release calculation used extremely simple assumptions. The analysis assumed initial hot shutdown conditions at the time steam pressure is highest and there is the greatest inventory of water in the steam generator. From the FSAR MSLB description, the initial fiuid inventory was 153,600 lbm (3250 ft^3) which corresponded to 33% of NR level. The basis for these assumptions appears to be the Kewaunee steam generator design, the fluid thermodynamics at hot zero power conditions, and the normal "programmed" steam generator level (33%) for hot shutdown. However, if levels are not maintained above the programmed level of 33%, the steam generator level transients during plant startup and generator synchronization to the grid can challenge the reactor trips on low steam generator levels. Therefore, steam generator water levels are manually controlled at higher values to provide margin and avoid unnecessary reactor trips.

b) The parameter plotted as steam generator (SG) wide range level in Figure 3 of our September 25, 1997, submittal is a non-physical SG wide range level parameter calculated by the WPSC main steam line break (MSLB) model. The trend of the model generated non-physical wide range level parameter is the same as the trend of the actual SG wide range level. An actual SG wide range level versus time plot is presented in Figure 1 of this attachment. As shown in Figure 1, the initial actual SG wide range level assumed in the analysis is 41.9 feet. This initial wide range level corresponds to 50 % level on the narrow range level indication. Figure 2 of this attachment shows the Kewaunee SG narrow and wide range level tap locations and the initial MSLB assumed water level.

Figure 1

MAIN STEAM LINE BREAK

CONTAINMENT RESPONSE

SG WIDE RANGE LEVEL vs TIME

slb14myy0

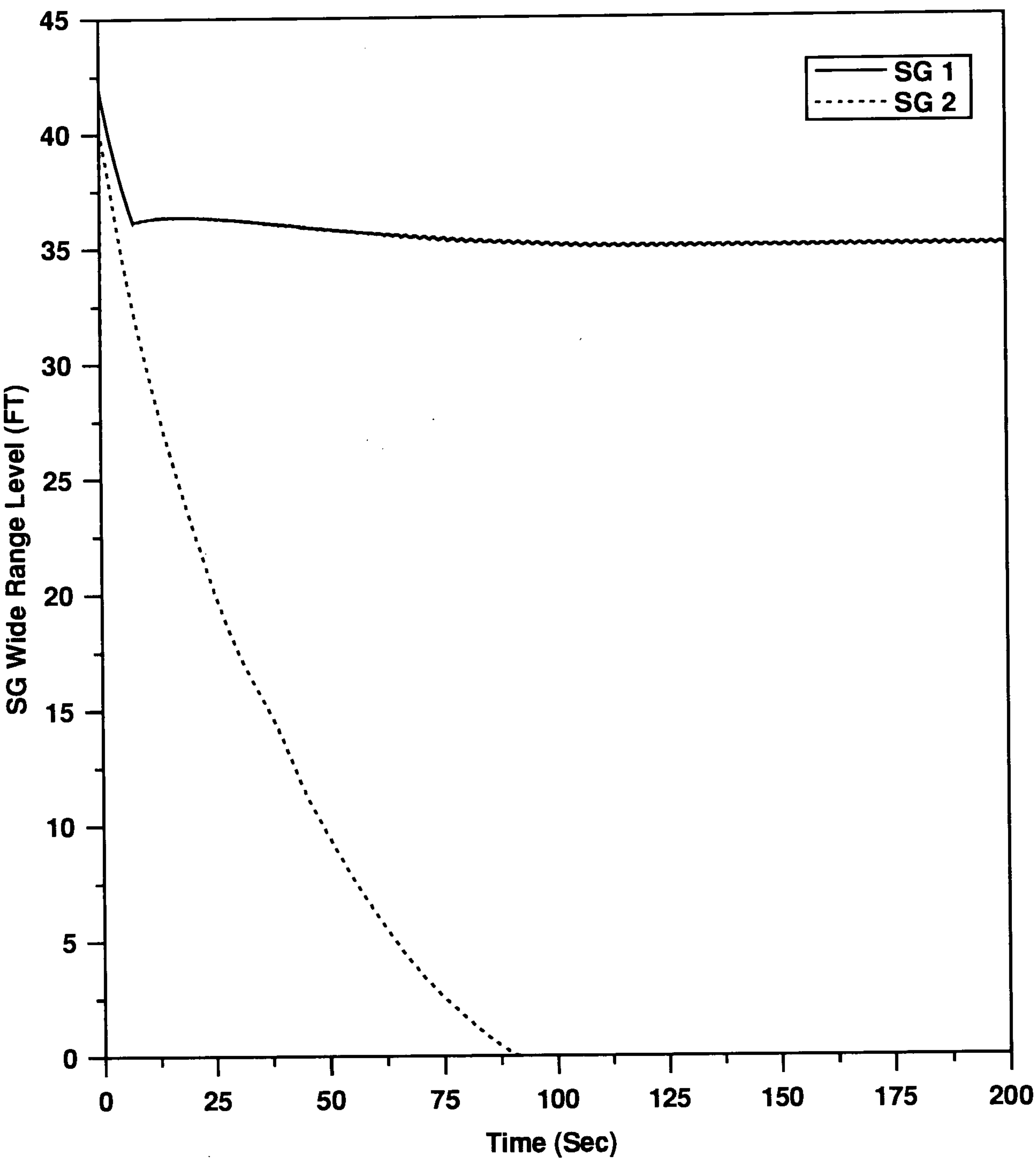
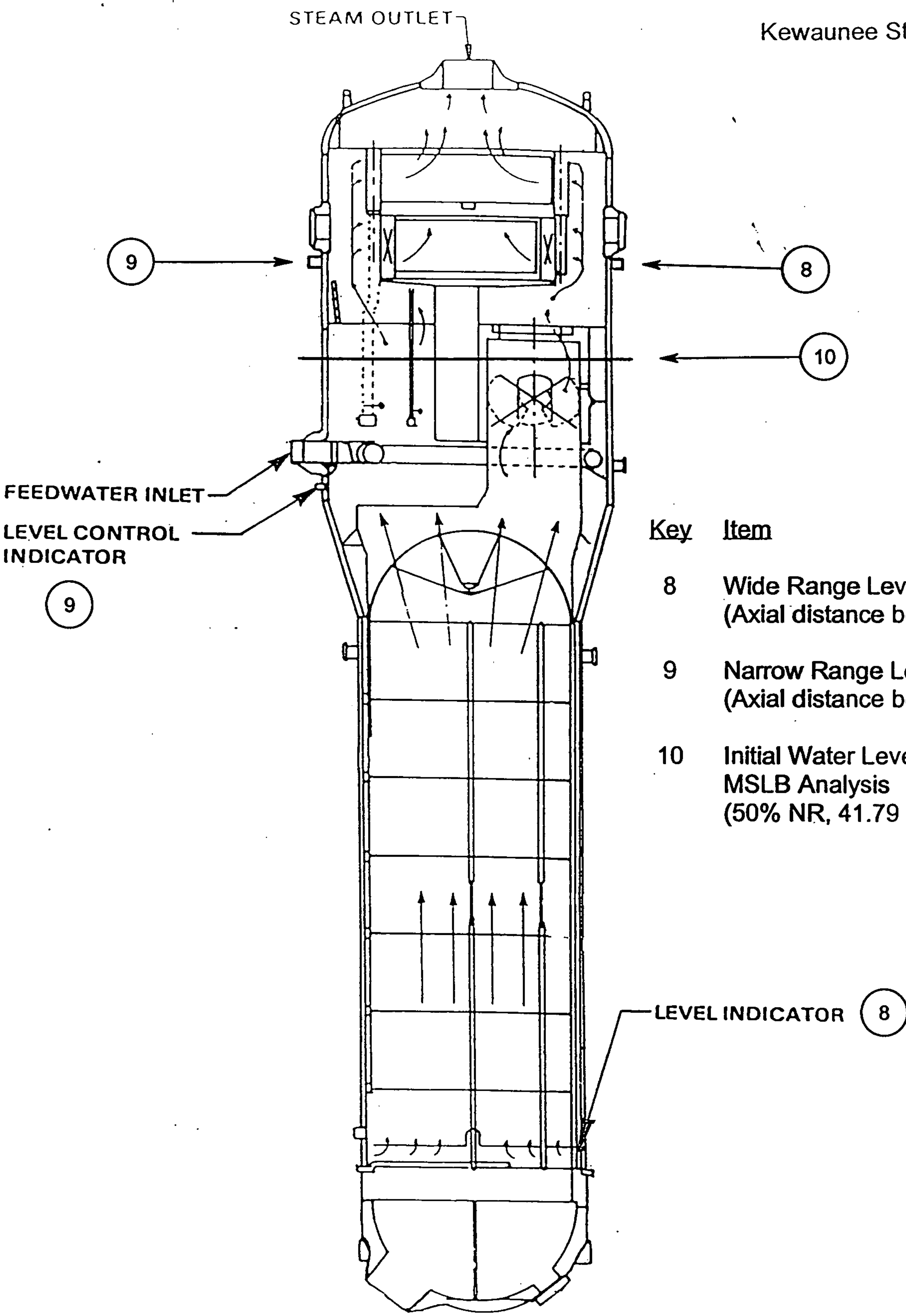


Figure 2

Kewaunee Steam Generator



Key	Item
8	Wide Range Level Taps (Axial distance between taps = 47.9 ft)
9	Narrow Range Level Taps (Axial distance between taps = 12.0 ft)
10	Initial Water Level MSLB Analysis (50% NR, 41.79 ft WR)

ATTACHMENT 2

Letter from M. L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

January 23, 1998

Review of the WPSC MSLB Containment Response
Methods and Assumptions Against Applicable
Sections of ANSI/ANS 56.4-1983

3. Mass and Energy Releases

3.3 PWR Secondary System Releases

3.3.1 Energy Sources

3.3.1.1 Reactor Coolant System Water and Metal

The Reactor Coolant System fluid and heat structure input data are based on nominal, cold, unpressurized dimensions. For the MSLB analysis, RCS fluid and heat structure thermodynamic properties are calculated in the model consistent with the operating conditions. However, the RCS fluid volume is not incrcased due to thermal expansion of metal structures. Thus, a sensitivity analysis was performed in whicb the RCS fluid volumes were increased by 5% to account for metal operating conditions and manufacturing tolerances and uncertainties. The effect on containment pressure was to increase the peak containment pressure by 0.09 psi.

Steam generator tube plugging (SGTP) is assumed to be at 0%. Current Kewaunee steam generators are at a plugging level of 26.0%. Therefore, 0% SGTP is a conservative assumption with respect to RCS fluid inventory and metal stored energy.

The initial metal heat structure temperatures are the same as the temperatures of the water with which thcy are in contact. This initial metal temperature combined with the additional metal structure due to the 0% SGTP assumption maximize the metal heat structure energy.

A deviation of the WPSC model assumption for RCS fluid volumes is identified for this item. However, it is demonstrated through sensitivity analysis that the effect of this deviation on containment pressure response is less than 0.1 psi. Therefore, the deviation is justified.

3.3.1.2 Steam Generator Secondary Water and Metal

The steam generator fiuid and heat structure input data are based on nominal, cold, unpressurized dimensions. For the MSLB analysis, SG fiuid and heat structure thermodynamic properties are calculated in the model consistent with the operating conditions. A conservatively high initial SG liquid inventory, which is the key parameter, is used.

Nominal SG inventory is used to compute the liquid entrainment in the steam release, which is conservative relative to the inventory that is used in the blowdown calculation.

The initial metal heat structure temperatures are the same as the temperatures of the water with which they are in contact. This initial metal temperature combined with the additional metal structure due to the 0% SGTP assumption maximize the metal heat structure energy.

3.3.1.3 Core Stored Energy

The core stored energy and the steady state core temperature distribution are consistent with the operating conditions and the time of fuel cycle life. A conservatively low gap heat transfer coefficient is used to compute the initial fuel temperature. The low gap heat transfer coefficient maximizes the initial core stored energy.

3.3.1.4 Fission Heat

The fission heat is conservatively calculated. The core cooldown reactivity is maximized and the trip reactivity insertion rate and shutdown margin are minimized. These assumptions are therefore in conformance with the recommendations and guidance of the ANS Standard. Boron transport effects in the Safety Injection (SI) system are neglected in the model. A sensitivity study was performed in which all the water in the SI line was conservatively assumed to be at 0 ppm boron concentration. The effect on containment peak pressure was to increase it by 0.015 psi. Adjusting the boron reactivity effects for reduced concentration in the SI line causes a negligible impact on containment pressure. The model assumptions relative to boron injection are therefore justified.

3.3.1.5 Decay of Actinides
and
3.3.1.6 Fission Product Decay

100% of the 1971 proposed ANS decay heat standard is used. A sensitivity analysis assuming 120% of the 1971 ANS decay heat standard was performed. The effect on containment pressure was to increase peak containment pressure by 0.019 psi.

The WPS model deviates from the standard. Sensitivity analysis on the WPS decay heat assumption has shown that it has minimal impact on contaiument pressure results. Thus, the deviation is justified.

3.3.1.7 Main Steam Lines

The flow of steam from the unaffected steam generator and steam lines to the containment prior to isolation is included. The steam in any steam line which can not be isolated from the primary containment is assumed to be released. Flows to containment are maximized and the delay in isolation is conservatively long. Turbine stop valve delay and closure is conservatively short to maximize steam flows to the containment.

The break is assumed to be instantaneous to maximize the release to containment.

3.3.1.8 Main Feedwater Lines

Main feedwater flow to the SG's is included until the flow is calculated to terminate. The dynamics of the flow to the affected and unaffected SG's are appropriately calculated. All flows are upper bound values. Flow rates consider the effects of pump suction and discharge pressures. Sigual delays and valve closure times are conservatively long.

The unisolated feedwater line is included as part of the initial steam generator inventory to model flashing and its release to containment.

3.3.1.9 Auxiliary Feedwater System

Auxiliary feedwater (AFW) flow to the steam generators is included in the analysis. The dynamics of the AFW flow includes the variations of the pressures in the affected and unaffected steam generators. Flow to the affected SG is maximized since: all three AFW pumps are assumed to be running; desigu pump performance data is used for each AFW pump, recirculation flow losses are not included, and delay times for AFW pnmp start are minimized. For the analysis, there is no termination of AFW flow during the time interval of interest in the transient either by automatic isolation or by operator action.

3.3.2 Initial Conditions

3.3.2.1 Time of Fuel Cycle Life

The assumed time of fuel cycle life is end of cycle. End of cycle core conditions maximize the containment pressure response.

3.3.2.2 Power Level

A spectrum of power levels from no load to 102% of rated power are analyzed.

3.3.2.3 Core Inlet Temperature

The initial core inlet temperature is the normal operating temperature for the power level being analyzed, adjusted upward for uncertainties.

3.3.2.4 Reactor Coolant System Pressure

The initial pressurizer pressure is set low (nominal -30 psi) to minimize MDNBR. A sensitivity study was performed in which initial pressurizer pressure was set to 2280 psia (nominal +30 psi). The effect on containment pressure was to increase peak containment pressure by 0.015 psi.

A deviation is identified for RCS pressure. However, through sensitivity analysis performed on the WPS model, the deviation has been shown to have negligible impact on containment pressure results and is justified.

3.3.2.5 Steam Generator Pressure
The SG initial pressure is consistent with the initial power level plus uncertainties. The WPS model assnmes 0% steam generator tube plugging (SGTP), and thus the model SG pressure conservatively bounds actual steam generator pressures which are reduced because of the reduced RCS to SG heat transfer capability due to the current SGTP levels being at 26.0%.

3.3.2.6 Reactor Coolant System Pressurizer Level
The initial pressurizer level is the nominal operating level, consistent with the initial power level, plus uncertainties to maximize the initial level.

3.3.2.7 Steam Generator Water Level
All analyses conservatively assume an initial SG water level of 50% of narrow range level span. This initial SG water level conservatively bounds the maximum expected level, consistent with the initial power level.

3.3.2.8 Core Parameters
Initial core parameters are chosen to maximize the containment pressure response.

3.3.2.9 Control Element Assembly (CEA) Position
The trip reactivity insertion and shutdown margin in the WPS model account for the effect of having the highest worth control element assembly stuck out of the core. Technical Specifications do not permit operation with a stuck out CEA.

3.3.2.10 Boron Concentration
The initial core boron concentration is 0 ppm, consistent with end of cycle operation, to maximize containment pressure response.

3.3.3 Single Failures

3.3.3.1 Single Active Failures
The most restrictive single active failure is considered in the WPS main steam line break containment response analysis methods. One train of containment heat removal systems, main steam isolation valve, and feedwater regulating valve are the single active failures considered. In addition, in all cases only one train of safety injection is assumed to be available. The loss of non-emergency electric power is also analyzed with safeguards timing delayed to account for the diesel generator startup time.

3.3.3.2 Single Passive Failures
Passive failures need not be considered consistent with the ANS standard.

3.3.4 Modeling

3.3.4.1 Nodalization
The steam generator model used for mass and energy release is a one node model created from more detailed calculations using a 3-node (downcomer, riser, and steam dome) model. The one node model is used since it yields conservative mass and energy blowdown results. The steam generator model used for entrainment calculations is modeled in greater detail so that the quality of steam at the break point is not under-predicted. The steam quality results are incorporated into the steam generator mass and energy release analyses. The nodalization of the RCS is consistent with the nodalization used for safety analysis of USAR Chapter 14 non-LOCA transients.

No credit is taken for SG tube uncovery in the affected SG to maximize the energy transferred from the RCS.

Sufficient detail is provided in the remaining system and component models to ensure that mass and energy releases to containment are not under-predicted.

3.3.4.2 Thermodynamic Conditions
The thermodynamic state conditions are determined in accordance with item 3.2.4.2 of the ANS Standard.

3.3.4.3 Pump Considerations
The reactor coolant pumps are modeled along with their heat addition to the RCS and are delivering conservatively high RCS flows

corresponding to 0% SGTP. The loss of power cases assume conservative rates of flow coast down.

The main feedwater, condensate, and heater drain pumps are conservatively modeled to maximize feedwater flow delivery to the SG's.

3.3.4.4 Break Flow

3.3.4.4.1 Break Sizes

A spectrum of break sizes is considered including a double-ended break of the main steam system piping (4.3 $ft.^2$ break) down to a small pipe break area of 0.1 ft^2. Split (or longitudinal) breaks are also considered. All breaks are defined by size, location, and area.

3.3.4.4.2 Break Flow Model

The break model is the Moody critical flow model which conforms to the recommendations and guidance in the ANS Standard.

3.3.4.5 Primary Containment Backpressure

The mass and energy release calculations assume a conservative, constant containment backpressure of 14.7 psia.

3.3.4.6 Heat Transfer Correlations

The heat transfer correlations of the WPS model are listed below.

These correlations, along with a conservatively large multiplier, are also used to calculate the reverse heat transfer in the unaffected SG to maximize the heat available for transfer to the secondary side of the affected SG.

3.3.4.6.1 Core to Reactor Coolant

Dittus-Boelter

3.3.4.6.2 Reactor Metal to Reactor Coolant

Dittus-Boelter

3.3.4.6.3 Unaffected SG Tubes and Reactor Coolant

Dittus-Boelter

3.3.4.6.4 Unaffected SG Coolant and Tubes

Thom

3.3.4.6.5 Unaffected SG Coolant and Metal

Subcooled	Dittus-Boelter
Saturated	Thom

3.3.4.6.6 Reactor Coolant to Affected SG Tubes

Dittus-Boelter

3.3.4.6.7 Affected SG Tubes to SG Coolant

Thom

3.3.4.6.8 Affected SG Metal to SG Coolant

Subcooled	Dittus-Boelter
Saturated	Thom

3.3.4.7 Core Modeling

Fission heat is calculated using a point kinetics model. Shutdown reactivities are assumed at their minimum values. Rod trip and insertion rate are biased toward minimizing trip reactivity worth and maximizing trip time delays.

Reactivity effects are consistent with end of cycle core physics parameters which leads to maximum containment pressures. Initial core stored energy and core thermal hydraulics are also conservatively assumed to maximize containment pressure.

3.3.4.8 Modeling of Metal Walls

Heat transfer from metal walls to coolant is calculated.

Conservative heat transfer coefficients are used. They are based on the Dittus-Boelter and Thom correlations and are consistent with the discussion in 3.3.4.6.

3.3.4.9 Modeling of Auxiliary Flows

Auxiliary feedwater flows are based on expected pump performance values. Uncertainties are applied to maximize flows and minimize delays. All three auxiliary feedwater pumps are assumed to be operating. Unequal flows due to differences in steam generator pressure are calculated by the model.

The safety injection system model is based on expected pump performance values. Uncertainties are applied in such a way as to minimize the SI flow. Only one SI pump is assumed in all cases.

3.3.4.10 Systems Interaction

The WPS MSLB analysis assumptions are selected to ensure the highest containment pressure and temperature have been determined:

-- The energy sources include reactor coolant, secondary coolant, metal, core power, and main and auxiliary feedwater flow.

-- Reactor trip is based on reactor protection system trips in the primary, secondary, and containment systems.

-- Steam flow to the turbine is minimized by fast closure and minimum delay of the turbine stop valves.

-- Main and auxiliary feedwater flow are conservatively calculated. Their dependence on other system parameters is modeled.

-- Containment pressure-initiated trips are included. The containment parameters and assumptions for containment heat removal are selected to maximize the peak containment temperature and pressure.

-- The steam generators are coupled hydraulically, via the main steam system pressure balancing line.

-- Selection of initial conditions for MSLB analysis considers the competing effects of input parameters.

4. Dry Primary Containment Pressure and Temperature Transient Analysis

4.2 Maximum Pressure and Temperature Analysis

4.2.1 Postulated Accidents

A spectrum of break areas, break locations, and power levels is considered to ensure that the maximum pressure and temperature transients are identified.

4.2.2 Duration of Analysis

The containment response is calculated for a sufficient amount of time to ensure the maximum pressure and temperature have been found. The containment pressure will return to <50% of design pressure within a 24-hour period provided the containment safeguards systems continue to function consistent with the single failure assumptions. For the case yielding the highest maximum pressure an extended analysis demonstrated that containment pressure was <50% of design pressure within 24 hours.

4.2.3 Dry Primary Containment Analysis Model

4.2.3.1 Thermodynamic State Conditions

4.2.3.1.1 Dry Primary Containment Atmosphere Region

Evaporation/condensation between the containment atmosphere and pool is modeled.

No droplets are included in the model for the atmosphere.

Revaporization of the condensate on the metal heat structures and containment fan coil units (CFCU's) is limited to 8%.

Sensitivity analyses on the evaporation/condensation model were performed. The results of the sensitivity analyses showed that the peak containment pressure varies by approximately 0.04 psi depending on the evaporation/condensation model assumptions. This is a small effect. The current model assumptions are therefore justified.

In addition, the steam and non-condensible gases are homogeneously mixed and in thermal equilibrinm. Liquid water may exist in the containment atmosphere. The thermodynamic state conditions of the steam component and non-condensible component of the atmosphere region and the water in the sump region are modeled according to the guidance in the standard.

4.2.3.1.2 Dry Primary Containment Sump Region

Evaporation/condensation between the containment atmosphere and pool is modeled. Sensitivity analyses on the evaporation/condensation model were performed. The results of the sensitivity analyses showed that the peak containment pressure varies by approximately 0.04 psi depending on the evaporation/condensation model assumptions. This is a relatively small effect. The current model assumptions used are therefore justified.

4.2.3.2 Mass and Energy Transfer Mechanisms

4.2.3.2.1 Pipe Break Blowdown

The mass and energy release from the pipe break goes directly to the containment atmosphere region. Phase separation and flashing to the saturation temperature at the containment atmosphere steam partial pressure are modeled.

4.2.3.2.2 Energy Source Terms

Sensible heat terms and other exothermic reactions which could add significant additional energy to the containment system are considered.

4.2.3.2.3 Structural Heat Transfer

A lower bound estimate of the number and surface area of structural heat sinks is used in the analysis. All three modes of heat transfer are considered and those modes that are significant are modeled.

The heat transfer coefficient between the containment atmosphere and the metal heat structures is based on the model described in Reference 4 which predicts a conservatively high containment pressure response compared to the use of the Uchida correlation.

Sensitivity analyses were performed to determine the effect of using the Uchida heat transfer correlation in place of the Reference 4 methodology. The sensitivity analysis showed that using Uchida, peak containment pressure decreases by 0.7 psia. The WPS model assumption is conservative and justified.

The thermal resistance between steel cladding and concrete structures is not included since that resistance is small.

Therefore, the WPS model, although it deviates from the recommendation in the standard, is conservative.

4.2.3.2.4 Dry Primary Containment Spray System

Energy removal by the containment spray system is modeled. 100% efficiency for the condensation of steam by the spray water is assumed. A sensitivity analysis was performed to show that a reduced efficiency (95%) of spray water had minimal effect on peak containment pressure.

4.2.3.2.5 CHRS Energy Removal Terms

Credit is taken for containment heat removal systems. The systems modeled are the containment fan coil units (CFCU's) and the internal containment spray system (ICS).

The energy removal capabilities for these systems are based on design and/or system performance test data. Uncertainties are applied to minimize the systems' heat removal capabilities. In addition, conservative maximum timing delays are assumed for these heat removal systems.

4.2.3.2.6 Atmosphere Sump Interface

Mass and energy transfer across the atmosphere-sump interface need not be treated consistent with the ANS standard.

4.2.3.3 Modeling Considerations

Time step size and heat sink nodalization are selected to ensure a physically representative solution.

4.2.4 Initial Conditions

Initial conditions are chosen to yield a conservatively high peak containment pressure and temperature; upper bound initial pressure and temperature (16.85 psia, 120°F), lower bound initial relative humidity and net free volume (0.177 and 1.32E6 ft^3), and upper bound ambient temperature and pressure (120°F, 14.7 psia) are selected.

4.2.5 Single Failure Criteria

A single failure is assumed, consistent with the discussion provided in the response to item 3.3.3.1. The failure chosen results in the highest calculated containment pressure and temperature.