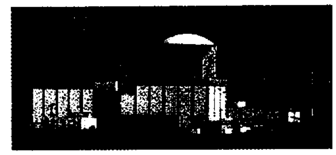
NRC-01-038

NMC
Committed to Nuclear Excellence

Kewaunee Nuclear Power Plant
N490, State Highway 42
Kewaunee, WI 54216-9511
920-388-2560

Operated by
Nuclear Management Company, LLC



April 13, 2001

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

Ladies and Gentlemen:

DOCKET 50-305
OPERATING LICENSE DPR-43
KEWAUNEE NUCLEAR POWER PLANT
NUCLEAR MANAGEMENT COMPANY, LLC. RESPONSE TO NRC 'S REQUEST FOR ADDITIONAL INFORMATION ON WISCONSIN PUBLIC SERVICE CORPORATION RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT, WPSRSEM-NP, REVISION 3

References: 1) Letter from Kenneth H. Weinhauer (NMC) to Document Control Desk (NRC), dated October 12, 2000, Wisconsin Public Service Corporation Reload Safety Evaluation Methods Topical Report, WPSRSEM-NP, Revision 3

2) Letter from John G. Lamb (NRC) to Mark Reddemann (NMC) dated January 23, 2001, Kewaunee Nuclear Power Plant - Request For Additional Information Related To Reload Safety Evaluation Methods Topical Report, WPSRSEM-NP, Revision 3 (TAC NO. MB0306)

3) Letter from John G. Lamb (NRC) to Mark Reddemann (NMC) dated February 1, 2001, Kewaunee Nuclear Power Plant - Request For Additional Information Related To Reload Safety Evaluation Methods Topical Report, WPSRSEM-NP, Revision 3 (TAC NO. MB0306)

4) Letter from Mark E. Reddemann (NMC) to Document Control Desk (NRC), dated February 7, 2001, "Response to NRC 's Request for Additional Information on Wisconsin Public Service Corporation Reload Safety Evaluation Methods Topical Report, WPSRSEM-NP, Revision 3."

5) Letter from John G. Lamb (NRC) to Mark Reddemann (NMC) dated March 23, 2001, Kewaunee Nuclear Power Plant - Request For Additional Information Related To Reload Safety Evaluation Methods Topical Report, WPSRSEM-NP, Revision 3 (TAC NO. MB0306)

A001

In reference 1, Nuclear Management Company, LLC. (NMC) submitted a request for approval of the Kewaunee Nuclear Power Plant (KNPP) Reload Safety Evaluations Methods Topical Report, WPSRSEM-NP, Revision 3. In reference 2, 3, and 5 the NRC staff requested additional information concerning this topical report. This letter is NMC's response to the NRC's request for additional information associated with reference letter 5.

Attachment 1 to this letter contains NMC's response. This letter contains answers to all the RAI questions of reference 5 except for a copy of WCAP-15427, "Development and Qualification of a GOTHIC Containment Evaluation Model for the Kewaunee Nuclear Power Plant." NMC has not received the applicable information from Westinghouse to allow us to send this information. When this information is received, WCAP-15427 will be submitted.

If you should have any questions concerning this matter, please contact John Holly (920) 388-8296 or Jerry Riste (920) 388-8424 of my staff.

In accordance with the requirements of 10 CFR 50.30(b), this submittal has been signed and notarized. A complete copy of this submittal has been transmitted to the State of Wisconsin as required by 10 CFR 50.91(b)(1).

Sincerely,

Mark E. Reddemann

Mark E. Reddemann
Site Vice President

GOR/JTH/smm

Attach.

cc - US NRC, Region III
US NRC Senior Resident Inspector
Electric Division, PSCW

Subscribed and Sworn to
Before Me This 13th Day
of April, 2001

Paula C. Maree

Notary Public, State of Wisconsin

My Commission Expires:
October 24, 2004

ATTACHMENT 1

Letter from Mark E. Reddemann (NMC)

To

Document Control Desk (NRC)

Dated

April 13, 2001

Response to NRC Questions Concerning Reload Safety Evaluation Methods

1.0 *Please provide one copy of Reference A 12, WCAP-15427—NMC has not received the applicable information from Westinghouse to allow us to send the information. When the information is received, WCAP-15427 will be submitted.*

2.0 *To the extent that the above report does not contain this information, provide for the GOTHIC model:*

2.1 *Details of how the Kewaunee containment will be modeled for a LOCA and a main steam line break*

The GOTHIC containment model described in WCAP-15427 was created for the purpose of benchmark comparison analyses to the COCO LOCA, the CONTEMPT LOCA and the CONTEMPT Main Steam Line Break (MSLB) containment models. The COCO and CONTEMPT models are the models used for the current containment integrity safety analyses. The GOTHIC containment model for KNPP LOCA and MSLB containment safety analyses will be substantially the same as the GOTHIC containment model developed for the benchmark comparison analyses. Any model refinements will be justified based on changed plant equipment performance, engineering judgement and best modeling practices given the GOTHIC code's capabilities.

The KNPP safety analysis GOTHIC containment model, like the COCO and CONTEMPT safety analysis models, will be consistent with the conservative modeling assumptions required for containment integrity safety analyses.

Following is a detailed description of how the KNPP containment will be modeled in GOTHIC. This description applies to both the LOCA and the MSLB containment analyses. The MSLB containment model, however, does not include containment sump recirculation and the residual heat removal system model; so the discussions of these systems do not apply to MSLB.

Containment Response Analysis

Accident Description

The Kewaunee Nuclear Power Plant (KNPP) containment system is designed so that for design basis LOCA and MSLB accidents, up to and including the double-ended severance of a reactor coolant or main steam system, pipe, the containment peak pressure remains below the design pressure. The containment response analysis uses the mass and energy release data from the LOCA and MSLB mass and energy release calculations.

The containment response analysis demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a design basis LOCA and MSLB accident inside containment. Equipment design and licensing criteria (e.g., qualified operating life), with respect to post-accident environmental conditions, are also verified through the containment pressure and temperature transient response analyses.

Input Parameters and Assumptions

An analysis of containment response to a design basis LOCA or MSLB accident must start with knowledge of the initial conditions in the containment. The pressure, temperature, and humidity of the containment atmosphere prior to the postulated accident are specified in the analysis and are shown in Table 1.

Also, values for the initial temperature of the service water (SW) and refueling water storage tank (RWST) are assumed, along with containment spray (CS) pump flow rate and containment fan coil unit (CFCU) heat removal performance. All of these values are chosen conservatively and are shown in Table 1. Long-term sump recirculation is addressed via residual heat removal system (RHR) heat exchanger (HX) performance. The primary function of the RHR system is to remove heat from the core by way of low-head safety injection. Table 1 provides the RHR system parameters assumed in the analysis.

A series of analyses, using different break sizes and locations, was performed for the LOCA containment response. The mass and energy releases for the minimum and maximum safeguards cases for a double-ended pump suction (DEPS) break and the releases from the blowdown of a double-end hot leg (DEHL) break are calculated by Westinghouse.

For the maximum safeguards (DEPS for LOCA) case, a failure of a containment spray pump was assumed as the single failure. This leaves one containment spray pump and four CFCUs available as active heat removal systems. Table 2 provides the performance data for one spray pump in operation. (Note: For the maximum safeguards case, a limiting assumption was made concerning the modeling of the recirculation system, i.e., heat exchangers. Minimum safeguards data were conservatively used to model the RHR HXs, i.e., one RHR HX was credited for residual heat removal. Emergency safeguards equipment data are given in Table 1.)

The minimum safeguards case was based upon a diesel train failure. This leaves one containment spray pump and two CFCUs available as active heat removal systems. Due to the duration of the DEHL transient (i.e., blowdown only), no containment safeguards equipment is modeled.

The calculations for all of the DEPS cases were performed for 1 million seconds (approximately 11.6 days). The DEHL cases were terminated soon after the end of the blowdown.

The following are the major assumptions made in the containment analysis.

The mass and energy released to the containment are based on LOCA and MSLB mass and energy release calculations.

Homogeneous mixing is assumed. The steam-air mixture and the water phases each have uniform properties. More specifically, thermal equilibrium between the air and the steam is assumed. However, this does not imply thermal equilibrium between the steam-air mixture and the water phase.

Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermodynamic properties.

For the blowdown portion of the LOCA and MSLB analysis, the discharge flow separates into steam and water phases at the breakpoint. The saturated water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the containment. For the post-blowdown portion of the LOCA analysis, steam and water releases are input separately.

The saturation temperature at the partial pressure of the steam is used for heat transfer to the heat sinks and the fan coolers.

The containment air-steam-water mixture is separated into a water (pool) phase and a steam-air phase. Sufficient relationships to describe the transient are provided by the equations of conservation of mass and energy as applied to each system, together with appropriate boundary conditions. As thermodynamic equations of state and conditions may vary during the transient, the equations have been derived for all possible cases of superheated or saturated steam and subcooled or saturated water. Switching between states is handled automatically by the GOTHIC code.

Passive Heat Removal

The significant heat removal source during the early portion of the transient is the containment structural heat sinks. Provision is made in the containment pressure response analysis for heat transfer through, and heat storage in, both interior and exterior walls. Every wall is divided into a large number of nodes. For each node, a conservation of energy equation expressed in finite-difference form accounts for heat conduction into and out of the node and temperature rise of the node. Table 3 is the summary of the containment structural heat sinks used in the analysis. The thermal properties of each heat sink material are shown in Table 4.

Active Heat Removal

For a large break, the engineered safety features are quickly brought into operation. Because of the brief period of time required to depressurize the RCS or the main steam system, the containment safeguards are not a major influence on the blowdown peak pressure. However, they reduce the containment pressure after the blowdown and maintain a low, long-term pressure and a low, long-term temperature.

Refueling Water Storage Tank, Injection

During the injection phase of post-accident operation, the low-head safety injection pumps water from the RWST tank into the reactor vessel. Since this water enters the vessel at RWST temperature, which is less than the temperature of the water in the vessel, it is modeled as absorbing heat from the core until the saturation temperature is reached. Safety injection and CS can be operated for a limited time, depending on the RWST capacity.

Residual Heat Removal, Sump Recirculation

After the supply of refueling water is exhausted, the recirculation system is operated to provide long-term cooling of the core. In this operation, water is drawn from the sump, cooled in an RHR exchanger, then pumped back into the reactor vessel to remove core residual heat and energy stored in the vessel metal. The heat is removed from the RHR HX by the component cooling water (CCW). The RHR HXs and CCW HXs are coupled in a closed-loop system, where the ultimate heat sink is the service water cooling to the CCW HX.

Containment Spray

Containment spray is an active heat removal mechanism that is used for rapid pressure reduction and for containment iodine removal. During the injection phase of operation, the CS pumps draw water from the RWST and spray it into the containment through nozzles mounted high above the operating deck. As the spray droplets fall, they absorb heat from the containment atmosphere. Since the water comes from the RWST, the entire heat capacity of the spray from the RWST temperature to the temperature of the containment atmosphere is available for energy absorption. During the recirculation phase, credit is not taken for the sprays.

When a spray droplet enters the hot, saturated, steam-air containment environment, the vapor pressure of the water at its surface is much less than the partial pressure of the steam in the atmosphere. Hence, there will be diffusion of steam to the drop surface and condensation on the droplet. This mass flow will carry energy to the droplet. Simultaneously, the temperature difference between the atmosphere and the droplet will cause the droplet temperature and vapor pressure to rise. The vapor pressure of the droplet will eventually become equal to the partial pressure of the steam, and the condensation will cease. The temperature of the droplet will essentially equal the temperature of the steam-air mixture.

Containment Fan Coil Unit

The CFCUs are another means of heat removal. Each CFCU has a fan that draws in the containment atmosphere from the volume adjacent to the CFCU. Since the CFCUs do not use water from the RWST, the mode of operation remains the same both before and after the low-head safety injection change to the recirculation mode. The steam/air mixture is routed through the enclosed CFCU unit, past essential service water cooling coils. The fan then discharges the air through ducting containing a check damper. The discharged air is directed out emergency discharge dampers immediately adjacent to the CFCU. See Table 5 for CFCUs heat removal capability assumed for the containment response analyses.

Table 1

Containment Response Analysis Parameters

Service water temperature (°F)	80
RWST water temperature (°F)	120
Initial containment temperature (°F)	120
Initial containment pressure (psia)	16.85
Initial relative humidity (%)	17.7
Net free volume (ft3)	1.32×10^6
Containment Fan Coil Unit	
Total	4
Analysis maximum	4
Analysis minimum	2
Containment high setpoint (psig)	5.00
Delay time (sec) With offsite power Without offsite power	 75.0 85.0
Containment Spray Pumps	
Total	2
Analysis maximum	1
Analysis minimum	1
Spray flow rate (gpm) Injection phase (per pump) Recirculation phase (total)	 Table 2 Not Modeled
Containment high-high setpoint (psig)	23.0
Delay time (sec) With offsite power (delay after high-high setpoint) Without offsite power (delay after high-high setpoint)	 106.0 135.0
Recirculation switchover begins, sec Minimum safeguards Maximum safeguards	 3992.0 1223.0
Containment spray termination time, (sec) Minimum safeguards Maximum safeguards	 3802.0 1223.0

Table 1 (Cont.) Containment Response Analysis Parameters	
Residual heat removal system	
RHR heat exchangers	
Modeled in analysis	1
UA Btu/hr-F	0.62×10^6
RHR flows through RHR HX – gpm	
Minimum safeguards	
Time (sec)	Flow (gpm)
3992.0	0.0
6171.0	155.3
6201.0	1397.2
5.0E+5	1397.2
Maximum safeguards	
Time (sec)	Flow (gpm)
1223.0	0.0
1413.0	500.0
1443.0	1397.2
5.0E+5	1397.2
CCW flow through RHR HX – gpm	1250.0
Component cooling water heat exchangers	
Modeled in analysis	1
UA Btu/hr-F	1.01×10^6
CCW heat exchanger flows – gpm	
CCW flow	2500.0
Service water flow	2000.0
Additional heat loads, Btu/hr	4.6×10^6

Table 2 Containment Spray Performance		
Containment Pressure (psig)	With 1 Pump (gpm)	With 2 Pumps (gpm)
0	1487.0	2974.0
10	1452.0	2904.0
20	1414.0	2828.0
30	1370.0	2740.0
40	1328.0	2656.0
46	1300.0	2600.0
50	1280.0	2560.0

Table 3
Kewaunee Structural Heat Sinks for Containment Integrity Analysis[1, 2, & 3]

Sink	SurfacesDescription	Material	Total Exposed Area (ft^2)	Thickness (in)
1	Containment cylinder – Coating #4	Carbon steel	41,300	1.5
2	Containment dome - Coating #4	Carbon steel	17,300	0.75
3	Reactor vessel liner – Coating #4	Carbon steel - Concrete backup	1260 1260	0.25 12.00
4	Refueling canal	Stainless steel - Concrete backup	1100 1100	0.1875 12.0
5	Refueling canal	Stainless steel - Concrete backup	5500 5500	0.25 12.0
6	Misc. supports – Coating #4	Carbon steel	4055	0.168
7	Misc. supports – Coating #4	Carbon steel	16,925	0.25
8	Misc. supports – Coating #4	Carbon steel	28,500	0.375
9	Crane – Coating #5	Carbon steel	2000	0.75
10	Crane – Coating #5	Carbon steel	500	1.0
11	Hand rails – Coating #4	Carbon steel	1695	0.0725
12	Grating – Coating #4	Carbon steel	12,400	0.045
13	Exposed conduit and cable trays – Coating #4	Carbon steel	2000	0.05
14	Ductwork – Coating #4	Carbon steel	18,000	0.035
15	Walls 1' to 1.9' – exposed 2 sides – Coating #2	Concrete	2806	6.0
16	Floors 12.0 in and greater – Coating #2	Concrete	12,896	12.0
17	Walls 4' to 7' 4" – exposed 2 sides – Coating #2	Concrete	18,588	24.0
18	Floor (in contact with sump) – Coating #2	Concrete	1088	12.0
19	Walls 2' to 3' 2" – exposed 2 sides – Coating #2	Concrete	28,898	12.0
20	Floors 4 in to 10 in – Coating #2	Concrete	6810	4.0

Table 3 (Cont'd)

Kewaunee Structural Heat Sinks for Containment Integrity Analysis[1, 2, & 3]

Notes:

1) The 2200 ft^2 surface area of the accumulator tanks will not be used as a heat sink. Steel would soak during the first few minutes of a transient and the accumulators will not be empty during the first 60 seconds.

2) Using 12 mil paint thickness from CONTEMPT model

3) There is an air annulus between the concrete containment cylinder and dome and steel shell.

Paint Coating Systems:

Coating #1: Plastite 9028 surfacer – flush: Phenoline 305 Primer – 4 mils; Phenoline 305 Finish – 4 mils

Coating #2: Plastite 9028 Amine-Epoxy Filler – flush: Plastite 9009 Primer – 6 mils; Phenoline 300 Finish – 8 mils

Coating #3: Carbozinc 11 Primer – 3 mils; Phenoline 305 Finish – 4 mils

Coating #4: Carbozinc 11 Primer – 3 mils; Phenoline 305 Finish – 8 mils

Coating #5: Carbozinc 11 Primer – 3 mils
1 mil = 1/1000 inch

Table 4 Thermophysical Properties of Containment Heat Sinks		
Material	Conductivity (Btu/hr-ft-°F)	Volumetric Heat Capacity (Btu/ft^3-°F)
Carbon steel	26.0	56.4
Stainless steel	8.0	56.4
Concrete	0.80	28.8
Phenoline 300 finish	0.083	28.8
Phenoline 305 finish	0.083	28.8
Phenoline 305 primer	0.083	28.8
Carbozinc 11 primer	0.9	28.8

Table 5 Containment Fan Coil Unit Performance	
Containment Temperature (F)	Heat Removal Rate (Btu/sec) per Containment Fan Coil Unit
100	833.3
136	2944.4
205	8750.0
244	13111.1
270	15833.3
300	15833.3

2.2 <u>Assumptions used for licensing basis calculations including:</u>

a) containment initial conditions of temperature, pressure, and humidity (reference response 2.1 and WCAP 15427 page 4)

b) heat transfer correlations (reference WCAP 15427 page 2)

c) modeling of the velocity of the containment atmosphere and its effect on heat transfer

Both the current containment design basis accident (DBA) evaluation models (COCO and CONTEMPT) and the proposed, benchmarked GOTHIC containment DBA evaluation model represent the containment as a single, lumped parameter volume. All three models use the Tagami correlation for condensation heat and mass transfer during blowdown, and the Uchida correlation post-blowdown. Therefore, the containment atmosphere velocity component is not included in the heat and mass transfer correlations for any of the models. A more mechanistic heat and mass transfer correlation would include the containment atmosphere velocity component. This would produce a more realistic calculation of the condensation heat and mass transfer.

d) timing of ESF equipment (reference response 2.1 and WCAP15427 page 5)

e) assumptions of spray behavior (droplet size distribution, spray effectiveness)

A small, constant diameter spray drop diameter was input to the benchmarked GOTHIC containment DBA evaluation model to simulate the instantaneous equilibration assumption in the COCO model and the 100% spray efficiency assumption in the CONTEMPT model. (reference response 2.1 and WCAP 15427 page 3 and page 8)

f) interactions of the containment atmosphere with the sump following a LOCA

The COCO and CONTEMPT containment DBA evaluation models do not allow interfacial heat and mass transfer between the sump and the containment atmosphere. Interfacial heat and mass transfer were prevented in the benchmarked GOTHIC containment DBA evaluation model by using an interface area input value of 0.0 ft^2. (reference WCAP 15427 page 3)

g) modeling of RHR heat exchanger (reference WCAP 15427 page 7, 8)

h) modeling of fan coolers (reference response 2.1 and WCAP 15427 page 5)

i) modeling of containment structures (reference response 2.1 and WCAP15427 pages 16,17)

j) any other modeling assumptions which have a significant effect on the calculated pressure and temperature of the containment

The purpose of the LOCA and MSLB containment response comparison was to develop and benchmark a GOTHIC containment DBA evaluation model that would closely match the results from the accepted CONTEMPT and COCO containment DBA evaluation models. Therefore, the GOTHIC containment DBA evaluation model used the same (or approximated the same) modeling assumptions as the base containment DBA evaluation models. For example, the RHR system model that was created for the GOTHIC containment DBA evaluation model for the LOCA case matches as closely as possible the flow rate and heat transfer (UA) inputs that are specified in the COCO containment DBA evaluation model. Similarly, input for the heat sinks, initial conditions, timing of ESF equipment and fan cooler heat removal in the GOTHIC containment DBA evaluation model was the same as in the corresponding CONTEMPT (MSLB) and/or COCO (LOCA) containment DBA evaluation models

2.3 Describe or provide a reference on how calculations of the mass and energy release (including entrainment) to the containment are performed.

Westinghouse performs the mass and energy release calculations for LOCA using Westinghouse approved methodologies. KNPP performs the mass and energy release calculations for the MSLB accident. This response will focus on the KNPP MSLB methods.

KNPP has had a history of development of MSLB safety analysis methods of which the MSLB mass and energy release calculations are an integral part.

In 1977 KNPP engineers were trained in transient and accident analysis methods by Westinghouse and Nuclear Associates International (Dr. R.C. Kern).

In 1978 and 1979 KNPP developed safety analysis capabilities, benchmarked the safety analysis methods to the KNPP FSAR Chapter 14 accident analyses, and submitted topical reports on core analysis, reload safety evaluation, and plant safety analysis methods (reference 2-1).

In 1980 KNPP applied MSLB safety analysis methods to support a response to I.E.Buletin 80-04. I.E.Bulletin 80-04 was concerned with the containment pressure response during a MSLB accident with continued auxiliary feedwater addition (reference 2-2).

In 1988 KNPP submitted revision 2 to the RSEM topical report (reference 2-3).

In 1994, with assistance from Dr. R.C.Kern, KNPP further developed MSLB containment response methods using the DYNODE, RETRAN and CONTEMPT computer codes and WCAP 8860. WCAP 8860 is a Westinghouse report describing mass and energy release from a MSLB. The MSLB safety analysis methods' improvements included development of water entrainment calculation methods using DYNODE and RETRAN, development of the CONTEMPT containment model (which at that time was used for LOCA containment integrity safety analyses) for MSLB containment analysis, and the improvement in MSLB methods to align with the WCAP 8860 and ANSI/ANS 56.4-1983 standards. The upgraded MSLB methods included analyses for a spectrum of power levels, break sizes and break locations plus single failure criteria analyses.

In 1994 and 1995 KNPP applied the upgraded MSLB methods in support of proposed technical specification 131, elimination of high concentration boric acid in the Boric Acid Storage Tanks (BAST). This proposed amendment was subsequently approved by NRC staff (reference 2-4). USAR chapter 14 section 14.2.5 (MSLB accident analysis) was revised in USAR revision 13 to reflect the BAST design change and the upgraded MSLB safety analyses.

In 1997 KNPP applied the MSLB methods to support tech spec amendment 148a which revised the bases for the main steam isolation valve (MSIV) closure time. In the request for additional information regarding proposed amendment 148a (reference 2-5) there is presented a discussion of the extent to which the KNPP MSLB mass and energy release analyses and the containment pressure response analyses for MSLB conform to the recommendations and guidance of ANSI/ANS-56.4-1983. Deviations from the industry practices were identified and justified. The TS change for MSIV closure time was approved (reference 2-6).

In 1998 plant safety analyses (including MSLB analyses) were performed to address increased steam generator tube plugging, fuel design change, and a new departure form nucleate boiling (DNBR) correlation. These analyses are documented in the current KNPP USAR.

Following is a description of the MSLB mass and energy calculation methodology. This methodology is documented in the current USAR section 14.2.5 (reference 2-7).

<u>MSLB Analysis – Containment Response</u>

There are four major factors that influence the release of mass and energy following a steam line break. These are the initial steam generator fluid inventory, primary to secondary heat transfer, protective system operation, and the state of the secondary fluid blowdown. The following is a list of those plant variables that determine the influence of each of these factors.

- Plant power level
- Main feedwater system design
- Auxiliary feedwater system design
- Break type, area, location
- Availability of offsite power
- Steam generator design
- Steam system failures
- Steam generator reverse heat transfer and RCS metal heat capacity

All of these variables are considered in the analyses and are conservatively selected based on the KNPP design.

Steam line break analyses cases are described based on a specific set of five parameters in the following manner:

Power level: 0, 30, 70, and 102 percent for the rated power level

Break size and location:

— Location is downstream of the steam generator outlet nozzle flow restrictor

Break size:

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

Single failures: There are three single failures:

— One feedwater-regulating valve fails to isolate. This is denoted as R.

— One MSIV fails to isolate. This is denoted as M.

— One containment safeguards train (one containment safeguard train is: one internal containment spray train and two containment fan cooler units) fails to activate. This is denoted as N.

Offsite power: Cases with and without the availability of offsite power are considered.

Entrainment: The quality of steam exiting the break is explicitly modeled and is dependent on break size and power level.

Based on the above parameters, steam line break analysis cases are designated as follows:
Break size (Units of ft^2)

Single failure

R – Feedwater regulating valve failure

M – MSIV failure

N – Containment safeguards system failure

Offsite power

Y – Yes

N – No

Entrainment

Y – Yes

Power level

0 – 0 percent

3 – 30 percent

7 – 70 percent

2 – 102 percent

For identification purposes, a six-number/letter identification tag represents the cases. For example:

14NYY2 represents the steam line break case with:

14 = 1.4 ft^2 break

N = single active failure is one containment safeguards train

Y = offsite power is available

Y = entrainment is modeled

2 = initial power level is 102 percent

Further descriptions of the methods for steam line break analysis are the following:

The main feedwater flow is calculated using the following assumptions:

— The feedwater pumps are running at full speed at the start of the transient and are tripped off on the safety injection signal. A conservative flow coastdown is modeled.

— The condensate pumps are running at full speed throughout the transient.

— The regulating valve for the unfaulted loop remains at its initial position until the time at which it strokes to its fully-closed position at a rate of 5 percent/second following an isolation signal. At that time, the valve is closed instantaneously.

— The behavior of the regulating valve for the faulted loop is assumed to begin opening at t = 0.0 second at an 8 percent/second rate until the time the isolation signal occurs. It is held at that position until the time at which it strokes to its fully-closed position at a rate of 5 percent/second following the isolation signal. At that time, the valve is closed instantaneously. For cases with a regulating valve failure, isolation is produced by closure of the feedwater isolation valve. The assumption used for the isolation valve is that it begins to close, at the time of the isolation signal, from full open at a rate of 1.18 percent/second. The initial opening of the regulating valve is the same as for the case without a regulating valve closure failure.

The auxiliary feedwater flow split between the two steam generators is modeled. The auxiliary feedwater is initiated, prior to the time for the activation signal, at full capacity and using a conservatively high enthalpy. All three auxiliary feedwater pumps are assumed to be operating.

The core physics parameters are based on a bounding set corresponding to EOC conditions and minimum Technical Specification shutdown requirements. The scram worth includes having the most reactive rod stuck out.

The dynamic reactor coolant pump model is used, which includes the gravity head and pump heat effects.

Conservative setpoints and time delays are used throughout.

No credit is taken for charging flow.

No credit is taken for steam generator tube plugging.

The following considerations are made in modeling the steam lines:

— The pressure-balancing line is modeled to allow communication between the steam lines in an unrestricted manner.

— Main steam isolation for the unfaulted loop is assumed to occur instantaneously at the time required for the nonreturn check valve to close in the faulted loop, which is 5 seconds after the break occurs.

— The MSIV failure is modeled as a failure of the nonreturn check valve in the faulted loop. Steam flow from the unfaulted loop continues until the MSIV in the unfaulted main steam line closes. A closure assumption of 5 seconds is used for the MSIV. The time from the event initiation until MSIV closure signal receipt, plus signal instrumentation delays as applicable to the accident sequence analyzed, is added to the 5-second MSIV closure time assumption. At the time of the MSIV closure, the entire faulted and unfaulted loop steam lines from the MSIV to the turbine and the pressure-balancing line are added to the total fluid mass and energy input to containment.

The turbine is tripped at t = 0.0 seconds for 0-percent power cases, and prior to or at the actual time of reactor trip for at-power cases. These are conservative assumptions that maximize the available steam for blowdowns.

A constant containment backpressure of 14.7 psia is conservatively assumed in all cases.

A conservatively high RCS flow rate is assumed.

Steam generator fluid inventory is maximized. Initial steam generator water level is 49 percent. (44.0 percent with 5.0-percent uncertainty) narrow-range level for all cases)

Entrainment analysis methods are used to obtain the time-dependent quality of the failed steam line break flow, which is power-level and break-size dependent. The quality of the unfaulted steam line break flow is conservatively assumed to be 1.0.

Following is a discussion on the development of entrainment data for Main Steam Line Break (MSLB) Safety Analysis:

The MSLB containment pressure and temperature safety analyses use an entrainment model. The entrainment model that is used for the DYNODE MSLB mass and energy release calculations is based on RETRAN SG modeling and experimental data (reference 2-8).

Entrainment refers to the moisture carryover or quality of the steam/water mixture effluent from a MSLB. At certain break sizes the high steam flow rates result in the inability of the steam generator (SG) moisture separators to remove entrained liquid from the steam break flow. Liquid water is swept out (entrained) of the broken steam line along with the exiting steam.

The amount of entrainment for a given MSLB accident scenario is calculated using the DYNODE and RETRAN codes in an iterative manner. The DYNODE model, at a given power level and break size, is run to obtain fill junction data (reactor coolant system, main and auxiliary feedwater system, and main steam system boundary conditions) for input to the RETRAN SG model. RETRAN is then run with the DYNODE fill junction data to calculate an entrainment versus time curve.

The entrainment data is put back into DYNODE, which calculates a new set of fill junction data for RETRAN. RETRAN is run with the new fill junction data and the resulting entrainment versus time curve is compared to the previous curve. If the two curves agree to within established acceptance criteria then a self-consistent set of DYNODE and RETRAN cases have been found and the final RETRAN entrainment curve for that break size and power level is acceptable. If the two curves do not agree to within the established criteria then further DYNODE/RETRAN iterations are required until the entrainment curves are converged to within the acceptance criteria.

2.3.1 *A comparison of RETRAN-3D steam generator model with experimental data*

Steam Generator blowdown quality uncertainty factors were determined using a RETRAN-3D model of the Combustion Engineering (CE) experiments documented in reference 2-8. The RETRAN-3D calculated blowdown average quality results are compared with the CE measured results in Table 6.

The model underpredicts the average quality for the small break runs (runs 114 through 116) and overpredicts the quality for the large break runs (runs 109 through 112). There does not appear to be any trend with respect to initial water level. The model tended to overpredict the total mass and energy releases for those runs in which entrainment was observed.

Figures 1 through 4 compare the pressure and temperature responses for runs 109,112,116,and 119 respectively. These comparisons show good overall agreement of the calculated results with the measurements.

TABLE 6
RETRAN 3D Predicted vs. CE Experiment Measured Average Blowdown Quality (Xave)

Test Run	RETRAN 3D Predicted Xave	Measured Xave	Measured/Predicted
109	68.5	65.3	0.953
110	22.2	25.8	1.162
111	32.1	36.4	1.134
112	42.9	48.1	1.121
114	28.1	21.6	0.769
115	34.9	26.7	0.765
116	51.0	41.3	0.810
117	100.0	58.9	0.589
118	100.0	69.6	0.696
119	100.0	77.6	0.776

**RETRAN-3D CE Blowdown Test
Run #109**

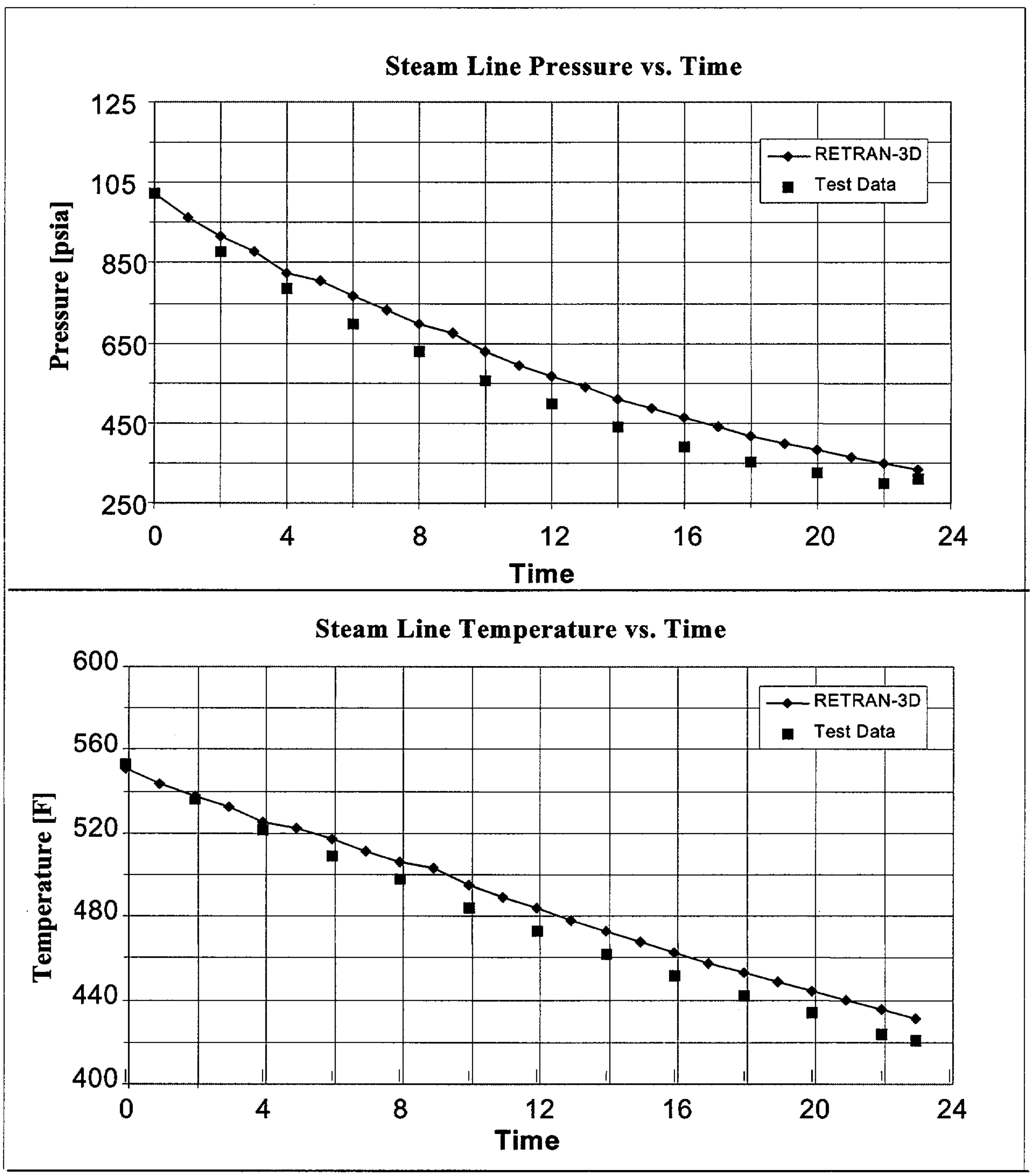


FIGURE 1

RETRAN-3D CE Blowdown Test
Run #112

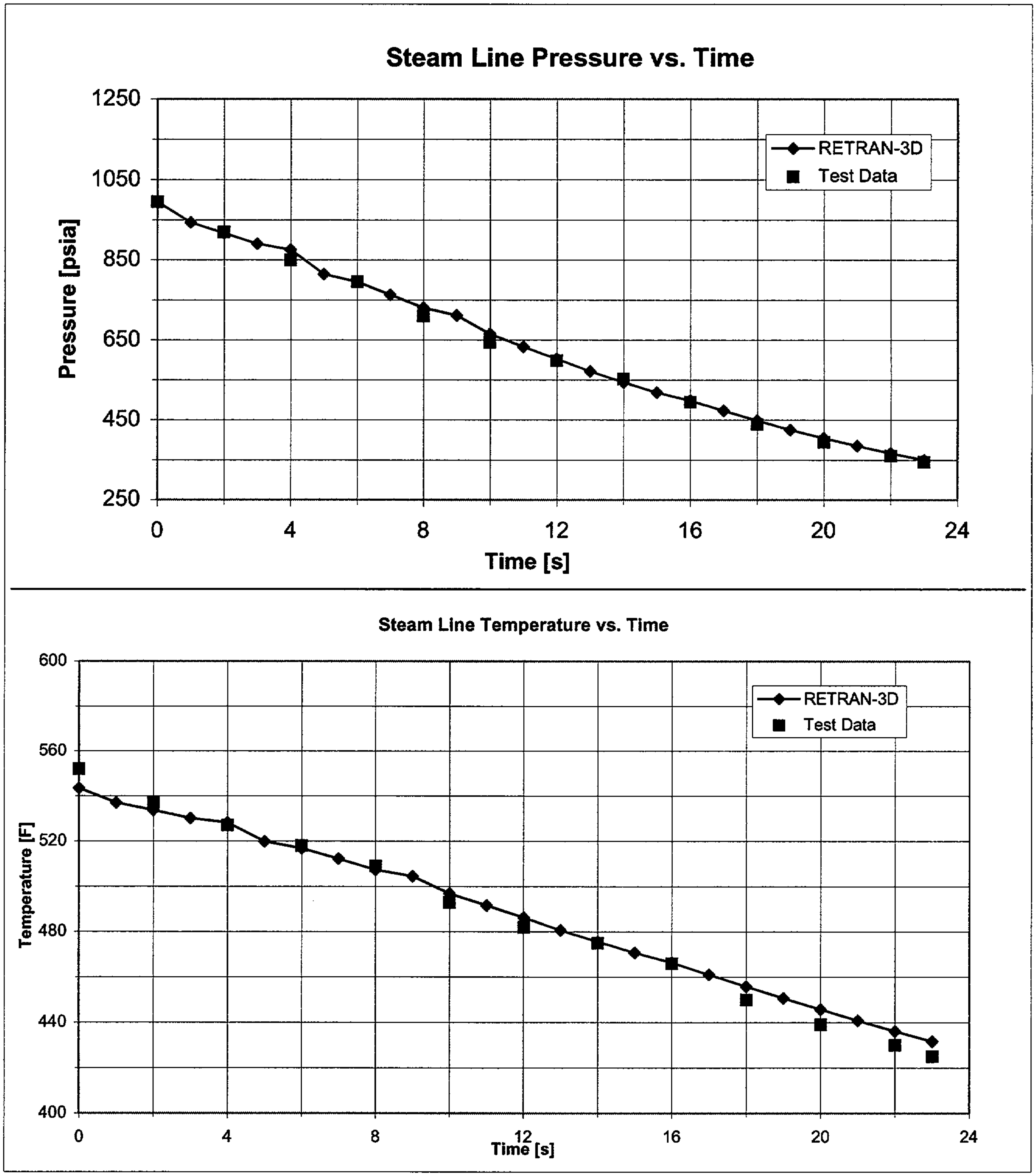


FIGURE 2

RETRAN-3D CE Blowdown Test
Run #116

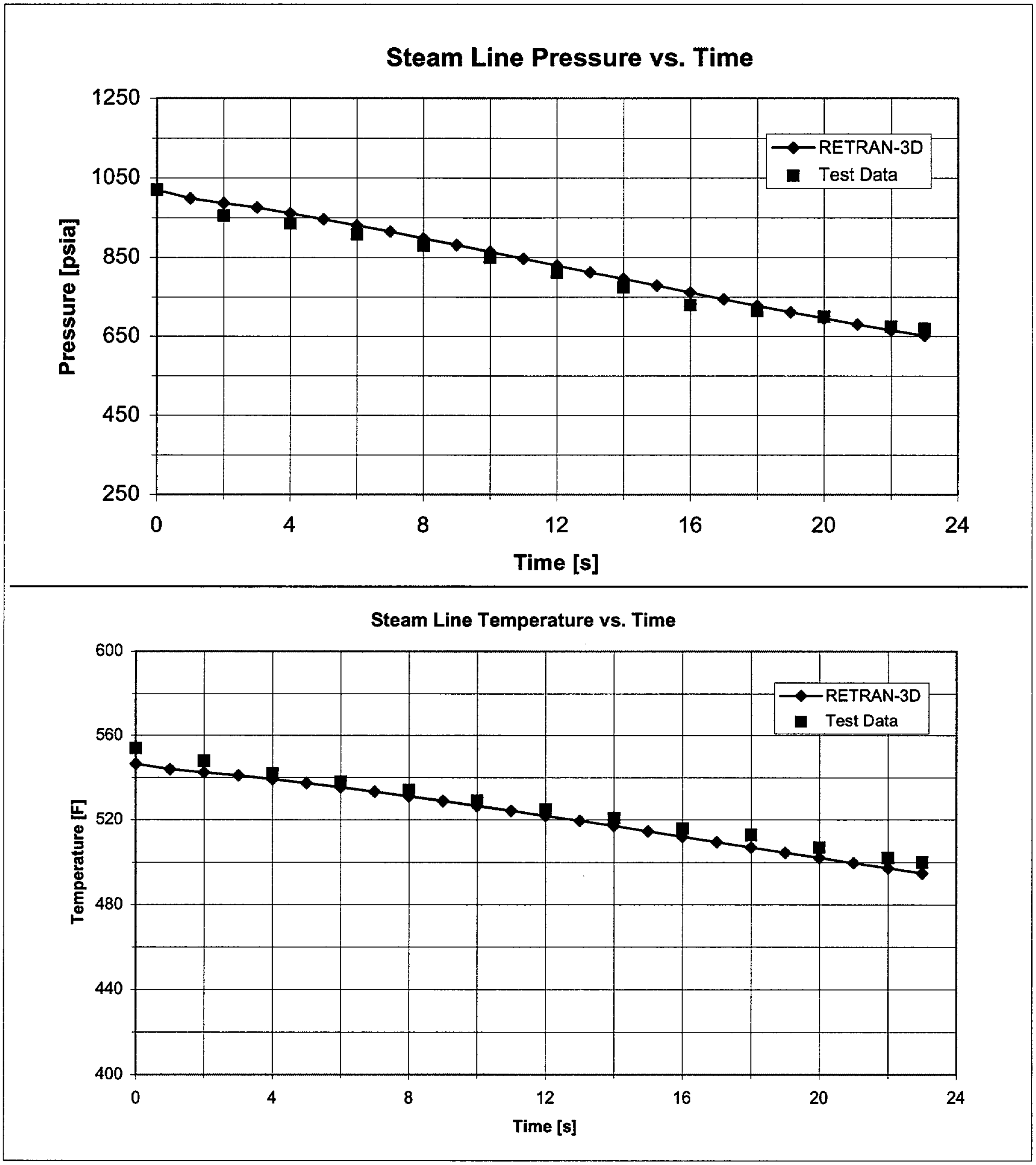


FIGURE 3

RETRAN-3D CE Blowdown Test
Run #119

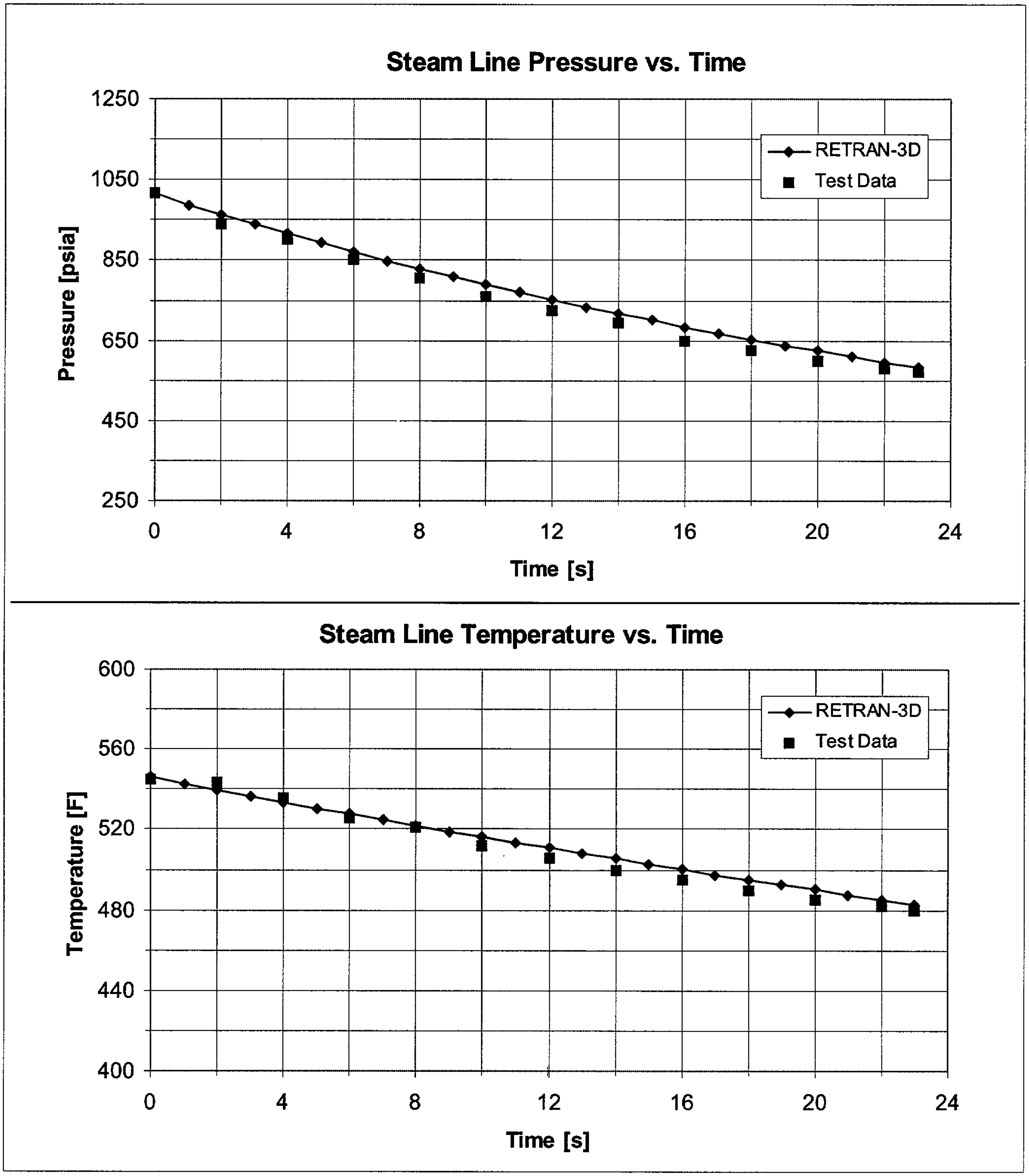


FIGURE 4

2.3.2 *A drawing showing the RETRAN 3-D steam generator noding used for the experiment(s) and the Kewaunee steam generator*

Figure 5 and 6 show the RETRAN-3D model steam generator noding used for the experiments and for KNPP steam generator respectively.

Figure 5 – CE Test Rig RETRAN Noding Diagram
(not to scale)

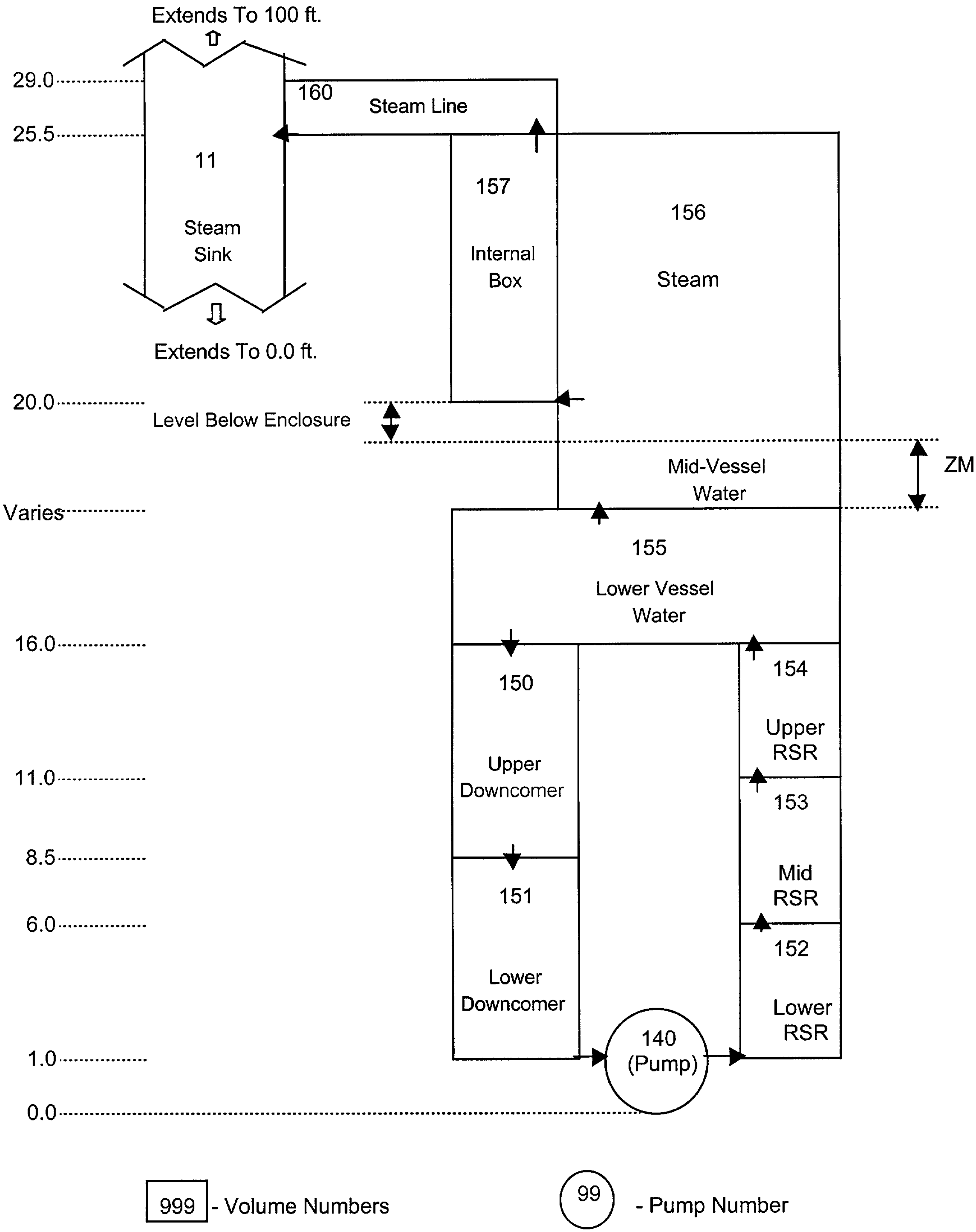
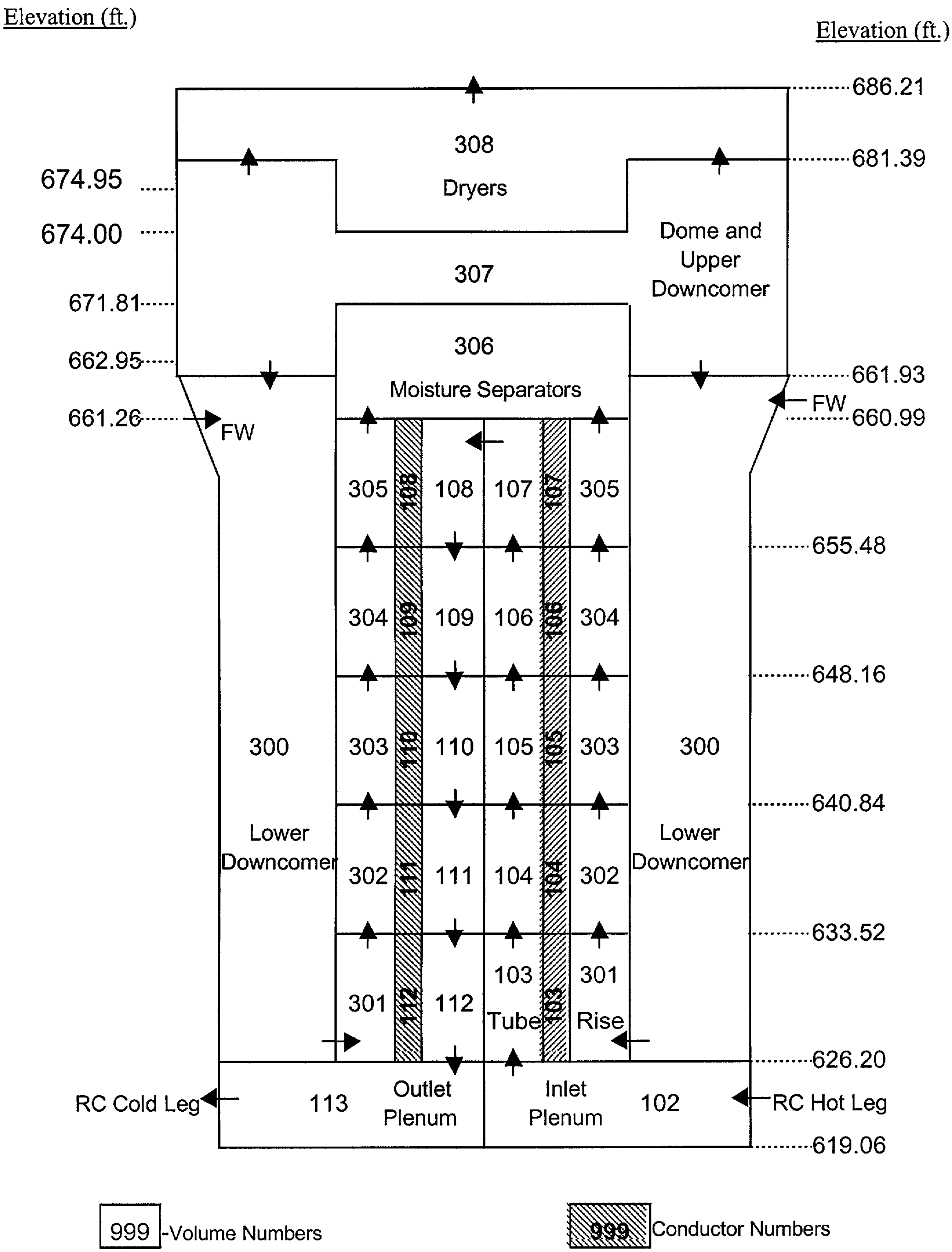


Figure 6 – Kewaunee RETRAN – 3D Steam Generator Noding Diagram
(not to scale)



2.3.3 A list of assumptions used in the modeling of the mass and energy release from the Kewaunee steam generator during a main steam line break

Reactor Coolant System Water and Metal

RCS fluid and heat structure thermodynamic properties are calculated in the model consistent with the operating conditions. RCS fluid volumes are increased by 5% to account for metal operating conditions and manufacturing tolerances and uncertainties

Steam generator tube plugging (SGTP) is assumed to be at 0%. 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 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.

Steam Generator Secondary Water and Metal

SG fluid 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.

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.

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

Decay of Actinides and Fission Product Decay

120% of the 1971 ANS decay heat standard is used.

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.

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. Signal 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.

Auxiliary Feedwater System

Auxiliary feedwater (AFW) flow to the steam generators is included in the analysis. The dynamics of the AFW flow include 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; design pump performance data is used for each AFW pump, recirculation flow losses are not included, and delay times for AFW pump 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.

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.

Power Level

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

Core Inlet Temperature

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

Reactor Coolant System Pressure

The initial pressurizer pressure was set to 2280 psia (nominal +30 psi).

Steam Generator Pressure

The SG initial pressure is consistent with the initial power level plus uncertainties. The model assumes 0% steam generator tube plugging (SGTP).

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.

Steam Generator Water Level

All analyses conservatively assume an initial SG water level of 49% of narrow range level span. This initial SG water level conservatively bounds the maximum expected level, consistent with the initial power level.

Core Parameters

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

Control Element Assembly (CEA) Position

The trip reactivity insertion and shutdown margin in the model account for the effect of having the highest worth control element assembly stuck out of the core.

Boron Concentration

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

Single Active Failures

The most restrictive single active failure is considered in the 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.

Single Passive Failures

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

Nodalization

The steam generator model NSSS simulation 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.

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.

Break Sizes

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

Break Flow Model

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

Primary Containment Backpressure

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

Heat Transfer Correlations

The heat transfer correlations of the 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.

Core to Reactor Coolant

Dittus-Boelter

Reactor Metal to Reactor Coolant

Dittus-Boelter

Unaffected SG Tubes and Reactor Coolant

Dittus-Boelter

Unaffected SG Coolant and Tubes

Thom

Unaffected SG Coolant and Metal

Subcooled	Dittus-Boelter
Saturated	Thom

Reactor Coolant to Affected SG Tubes

Dittus-Boelter

Affected SG Tubes to SG Coolant

Thom

Affected SG Metal to SG Coolant

Subcooled	Dittus-Boelter
Saturated	Thom

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.

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

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.

2.3.4 A description of any factor added to the quality of the steam generator blowdown to account for uncertainty in the mass and energy release calculation

The factor (multiplier) that is applied to the quality of the steam generator blowdown to account for uncertainty is calculated as follows using the measured to calculated results presented in Table 6 (see response to RAI 2.3.1).

Measured/predicted average quality Xave=0.877
Measured/predicted Sx=0.201
Multiplier=0.877*(1+2.911*0.201)=1.390

Where 2.911 is the one sided tolerance limit for a sample size of 10. The factor (multipler) is applied to the RETRAN-3D calculated blowdown quality values to obtain the 95/95 blowdown quality values that will be used for the MSLB safety analyses.

2.3.5 *A discussion of why a comparison using RETRAN-3D with data from an experimental facility which differs in scale and configuration from the Kewaunee steam generator provides confidence in the ability of RETRAN-3D to predict Kewaunee steam generator blowdown behavior following a main steam line break. (For instance: Are all the phenomena one would expect in the Kewaunee steam generator blowdown also included in the experiment? Would differences in the geometry between the experiment and the Kewaunee steam generator affect the confidence in RETRAN-3D to predict the behavior of the Kewaunee steam generator blowdown if there were good agreement between RETRAN-3D and the experiment?.*

NMC Response:

The comparison of the RETRAN-3D model results to the measured results for the blowdown of the experimental test rig (Ref. 2-8) establishes confidence in the ability of the RETRAN-3D model to predict the liquid entrainment phenomenon for the blowdown of the experimental test rig (See RAI 2.3.1). In addition, a statistical comparison of these results yields the applicable 95/95 uncertainty factor (See RAI 2.3.4).

It will be shown below that although the scale and configuration of the Kewaunee Nuclear Power Plant steam generators (KNPP SGs) differ from that of the experimental test rig, the parameters to which entrainment is most sensitive are similar. The degree of similarity provides confidence that the experimental test rig experiences all of the relevant physical phenomena during blowdown as the KNPP SGs would experience during blowdown. Likewise, the differences in the RETRAN-3D models for the experimental test rig and the KNPP SGs are justified to account for the differences in the scale and configuration of the physical systems and do not adversely impact the parameters to which entrainment is most sensitive. This provides confidence that the uncertainty factor derived from the RETRAN-3D experimental test rig model is applicable to the RETRAN-3D KNPP SG model.

The two key physical parameters that influence the entrainment phenomenon are the ratio of the break area to the vessel volume (A_{break}/V_{vessel}) and the ratio of the break area to the vessel free surface area ($A_{break}/A_{free\ surface}$). The ratio A_{break}/V_{vessel} is important in that it impacts the depressurization profile of the vessel (vessel pressure vs. time) and hence directly impacts the internal forces which provide the primary mechanisms that drive the fluid toward the break. This ratio also influences the flooding phenomenon that takes place in the upper regions of the vessel during blowdown events that experience appreciable entrainment. The flooding phenomenon greatly affects the level of entrainment and is not influenced significantly by the geometric details in that region. In a blowdown of a KNPP SG during a main steam line break the flooding of the upper vessel region is an expected phenomenon. The ratio $A_{break}/A_{free\ surface}$ is important in that it impacts the availability of water to separate from the predominately liquid mixture regions and enter the predominately vapor mixture regions and to rise upward and ultimately flow out of the break.

The table below compares these two physical parameters for the experimental test rig and the KNPP SGs. In addition, the measured and predicted steam dome average and minimum quality ranges are also compared. The comparisons demonstrate that the two key physical parameters for the KNPP SGs are substantially within the range established by the experiments. Thus, the experimental facility is shown to scale appropriately up to the KNPP SGs for the aspects of the physical systems that significantly affect entrainment. Further, the quality comparisons demonstrate the expected flooding phenomenon of the upper vessel region and that the entrainment results of the KNPP SGs applications are substantially within the ranges of the experimental data.

Parameter	CE Test Rig (1)	KNPP OSG (2)	KNPP RSG (3)
A_{break}/V_{vessel} (10^{-4}/ft)	1.1 to 4.5 (4,5)	0.9 to 2.4 (6,7)	0.9 to 2.5 (6,8)
$A_{break}/A_{free\ surface}$ (10^{-2})	1.10 to 4.35 (4,9)	0.98 to 2.75 (6,10)	0.98 to 2.75 (6,11)
Ave $X_{steam\ dome}$ (12)	0.22 to 0.78 (13)	0.43 to 0.79 (14)	0.60 to 0.81 (14)
Min $X_{steam\ dome}$ (12)	0.13 to 1.00 (13)	0.10 to 0.26 (14)	0.10 to 0.42 (14)

Where:

(1) Combustion Engineering (CE) Experimental Test Rig
(2) KNPP original steam generators (OSG), Westinghouse Model 51
(3) KNPP replacement steam generators (OSG), Westinghouse Model 54F
(4) CE Test Rig Break Areas: 0.022 ft^2, 0.087 ft^2
(5) CE Test Rig Volume = 195 ft^3
(6) KNPP Steam Line Break Areas with Entrainment: 0.5 ft^2, 0.8 ft^2, 1.1 ft^2, 1.4 ft^2
(7) KNPP OSG Volume = 5758 ft^3
(8) KNPP RSG Volume = 5638 ft^3
(9) CE Test Rig Free Surface Area = 2 ft^2
(10) KNPP original steam generator (OSG) Free Surface Area = 51 ft^2
(11) KNPP replacement steam generator (RSG) Free Surface Area = 51 ft^2
(12) X= quality; qualities are for various blowdown times, break sizes, etc.
(13) CE Test Rig Experimental Data
(14) RETRAN-3D Calculated Results

Therefore, due to the similarity of the key parameters important to the entrainment phenomenon, the differences in scale and configuration between the experimental test rig and the KNPP SGs do not adversely affect the confidence that these two systems experience the same relevant physical phenomenon during blowdown. The differences in the RETRAN-3D models for the experimental test rig and the KNPP SGs are due primarily to the differences in the scale and configuration of the two physical systems. The RETRAN-3D models preserve the physical quantities used to derive the key parameters important to entrainment and so preserve the key parameters. Therefore, the uncertainty factor derived from the RETRAN-3D experimental test rig model is applicable to the RETRAN-3D KNPP SG model. The RETRAN-3D comparison to an experimental facility provides confidence that RETRAN-3D is capable of predicting KNPP SG blowdown behavior following a main steam line break.

2.4 List and describe the conservatisms in the above modeling

Conservatisms in the containment modeling are described in responses 2.1 and 2.2 above and in reference 2-5. In summary, the KNPP LOCA and MSLB containment safety analysis conservatisms include:

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%.

Dry Primary Containment Sump Region

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

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.

Energy Source Terms

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

Structural Heat Transfer

A lower bound estimate of the number and surface area of structural heat sinks is used in the analysis.

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

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.

Atmosphere Sump Interface

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

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, 120F), lower bound initial relative humidity and net free volume (0.177 and 1.32E6 ft^3), and upper bound ambient temperature and pressure (120F, 14.7 psia) are selected.

Single Failure Criteria

A single active failure is assumed, which results in the highest calculated containment pressure and temperature.

2.5 List any other licensing basis uses of the GOTHIC code beside LOCA and main steam line break analyses.

Currently there are no other licensing basis uses of GOTHIC.

For the steam generator replacement (SGR) project GOTHIC is applied to the MSLB accident outside of containment for the purpose of analyzing the auxiliary building compartment pressure and temperature response to various MSLB accident scenarios. The MSLB outside of containment analysis is performed to support equipment qualification evaluations. Duke Engineering Services has performed these analyses for KNPP using GOTHIC.

References:

2.1 Letter from E. W. James (WPSC) to Document Control Desk (NRC) dated January 1979 transmitting Wisconsin Public Service Corporation (WPSC), Kewaunee Nuclear Power Plant (KNPP), topical report entitled "Reload Safety Evaluation Methods for Application to Kewaunee"

2.2 Letter from E.R. Mathews (WPSC) to J.G.Keppler (NRC) Response to I.E.Bulletin 80-04" dated May 7,1980.

2.3 Letter from C.R. Steinhardt (WPSC) to Document Control Desk (NRC), dated November 9, 1988, transmitting WPSRSEM-NP-A, Revision 2.

2.4 Letter from R.L.Laufer (NRC) to M.L.Marchi (WPSC) dated 3-28-95 transmitting amendment number 116 to facility operating license DPR-43 KNPP TAC no. M91072.

2.5 Letter from M.L.Marchi (WPSC) to NRC Document Control Desk "Request for Additional Information Regarding Proposed Amendment 148a" dated Jan. 23 1998.

2.6 Letter from W.O.Long (NRC) to M.L.Marchi (WPSC) dated 4-15-98 transmitting "Basis Change to Technical Specification for facility operating license DPR-43 KNPP TAC no.M98638.

2.7 Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, Updated Safety Analysis Report, Revision 16

2.8 Kling, C.L., Peeler, G.B., Combustion Engineering, Inc., Combustion Division, Nuclear Power Department, report titled "Moisture Carryover During An NSSS Steam Line Break", CENPD-80, January 1973.

3. *Does the Kewaunee use of GOTHIC comply with the guidance of GL 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions," February 8, 1983] and GL 83-11 Supplement 1, "Licensee Qualification for Performing Safety Analyses?" [June 24, 1999]*

NMC Response

Generic Letter 83-11 and Generic Letter 83-11, Supplement 1 (Refs. 3-1, 3-2) provide guidance on licensee qualification for performing safety analyses and will be referred to below collectively as GL 83-11. The guidance provided in GL 83-11 is organized into the following areas: Eligibility, Application Procedures, Training and Qualification of Licensee Personnel, Comparison Calculations, and Quality Assurance and Change Control.

It is noted that the GOTHIC code has not been reviewed and approved generically by the U. S. Nuclear Regulatory Commission (NRC) and has not as yet been otherwise accepted as part of Nuclear Management Company (NMC) Kewaunee Nuclear Power Plant (KNPP) licensing basis. Thus, the GOTHIC code does not meet the eligibility requirements of GL 83-11 and in this regard cannot comply with GL 83-11. Based on a clarification of the request for additional information (RAI) during a telephone call between the NRC and the NMC on March 21, 2001, the response to the above RAI question is directed towards the NMC consistency with the guidance of GL 83-11 other than the guidelines on eligibility.

Eligibility

See discussion above.

Application Procedures

The NMC has extensive in-house application procedures for KNPP that ensure that the use of all approved methods is consistent with code and qualification bases. Specifically, procedures for containment analyses methods in the calculation of peak containment pressure and temperature response to mass and energy additions exist in addition to procedures for the calculation of the mass and energy releases. Similar procedures will be developed for all GOTHIC calculations.

Training and Qualification of Licensee Personnel

The NMC has established and implemented an effective training program to ensure that users of all approved codes are qualified in the proper application of approved codes and methods. Formal training in GOTHIC has been supplied by expert GOTHIC users from both Westinghouse Electric Corporation and Duke Engineering & Services. In addition, the NMC actively participates in the GOTHIC maintenance group and other GOTHIC-related industry events in order to maintain a proficiency in the use of GOTHIC and to keep abreast of any potential issues with regard to GOTHIC and its use.

Comparison Calculations

The NMC processes for verifying any methods or modeling modification include comparisons to plant data, analyses of record or other suitable data. Verification of the proper use of the GOTHIC code is demonstrated through comparison of the KNPP containment results for the GOTHIC models with the current KNPP containment analysis results. The verification has been documented in Reference 3-3.

Quality Assurance and Change Control

GOTHIC analyses will be performed in accordance with an approved Quality Assurance Program applicable to KNPP. The GOTHIC code and the GOTHIC input models are and will be controlled in accordance with directives and procedures governing controlled computer codes and controlled computer code input under the KNPP Operational Quality Assurance Program (OQAP, Ref. 3-4). The use of the GOTHIC code requires the tracking and assessment of all computer code errors in accordance with established procedures under the OQAP.

Therefore, the NMC use of the GOTHIC code in the analysis of KNPP is consistent with the guidance described in References 3-1 and 3-2.

References:

3-1: Eisenhut, D. G., U. S. Nuclear Regulator Commission, letter entitled "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions", Generic Letter No. 83-11, February 8, 1983.

3-2: Matthews, D. B., et. al., U. S. Nuclear Regulator Commission, letter entitled "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions", Generic Letter No. 83-11, Supplement 1, June 24, 1999.

3-3: Ofstun, R. P., Westinghouse Electric Company, "Development and Qualification of a GOTHIC Containment Evaluation Model for the Kewaunee Nuclear Power Plant", WCAP-15427, September 2000.

3-4: Kewaunee Nuclear Power Plant Operational Quality Assurance Program.