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Subject: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Response to NRC Request for Additional Information Regarding ANO's November 3, 1999, Containment Uprate License Amendment Request

Gentlemen:

In a letter dated November 3, 1999 (2CAN119903), Entergy Operations, Inc. submitted a license amendment request for Arkansas Nuclear One, Unit 2 (ANO-2) regarding increasing the design pressure of the containment building from 54 to 59 psig. During a telephone conference call between members of the NRC and ANO staffs on March 8, 2000, the NRC requested additional information in regard to the November 3, 1999 letter. The ANO staff's responses to the questions are provided in Attachment 1 to this letter.

Attachment 2 of this letter contains two detailed sketches depicting the main steam isolation and containment spray actuation signals used to isolate main feedwater and main steam. The NRC requested a more detailed version of the sketches originally provided in Enclosure 3, pages 12 and 13, of the November 3, 1999 letter.

Additionally, Attachment 3 contains a listing of typographical corrections from the November 3, 1999 letter. In addition to the listing of changes, corrected Safety Analysis Report pages are enclosed.

Should you have any questions or comments, please contact me.

Very truly yours,

Jimmy D. Vandergrift
Director, Nuclear Safety Assurance

JDV/dwb
Attachments (3)

A001

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ANO Responses to NRC Staff Questions

NRC Question #1

The NRC posed the following four-part question in regard to the proposed change to Technical Specification 3/4.6.2, "Depressurization, Cooling and pH Control Systems" (see page 11 of 18 of the attachment to ANO's November 3, 1999, letter).

a) What conservatism or underlying assumptions was built into the original calculations?
b) With the excess conservatism built into the original analysis, what was the basis for choosing 6.3% as an allowable pump degradation?
c) What percent allowable degradation did your reanalysis show, and is the code allowable 10% operability requirement or your re-analyzed condition bounding?
d) If your reanalysis shows degradation below the code operability requirement, why not establish a second set of reference values per IWP-3112 to show operability?

ANO Response

Background

There are, and have been, no operability concerns with the current 6.3% degradation value in Technical Specification Surveillance 4.6.2.1.b. The containment spray pumps have shown only a small amount of degradation. Degradation of the pumps is well within the existing 6.3% allowable degradation. This change is not needed to support the containment uprate to 59 psig. In fact, the analysis supporting the uprated containment provides more margin for operation of the containment spray pumps. The containment uprate license amendment was viewed as an appropriate opportunity to provide additional pump degradation margin. This change would also make the allowable containment spray pump degradation consistent with the ASME Code. As described below in the answer to part c of the NRC's question, the code allowable 10% degradation becomes the limiting degradation, not degradation limited by the analysis.

Combined answer to NRC question 1, parts a and b

The 6.3% allowable pump degradation was developed in an original (circa 1978) Bechtel Power Corporation calculation. The calculation is a hand calculation using outputs from a 1970s vintage "system resistance" calculation. The calculation made use of system curves hand drawn onto pump curves and adjusted pump curves utilizing the pump affinity laws. "Case 1b" of the hand calculation was for one spray pump train at 2000 gpm taking suction from the refueling water tank. In case 1b, containment spray pump "B" was the limiting pump at 6.3% degradation. In short, the 1978 analysis is less sophisticated than the analytical tools available today, but is conservative.

Answer to NRC question 1, part c

The analysis for allowable pump degradation demonstrates that the pumps could degrade by 11.7%. The 10% code allowable degradation is bounding.

Answer to NRC question 1, part d

The reanalysis doesn't show actual pump degradation below the code operability limit. It shows that pump degradation to below the code limit (-10%) would still provide acceptable flow. Actual pump performance is at a level well above the more restrictive 6.3% degradation allowed by Technical Specification 4.6.2.1.b. Since the reference values discussed in IWP-3112 are based on actual pump performance curves, they will not be affected by a change to the flow assumed in the safety analysis.

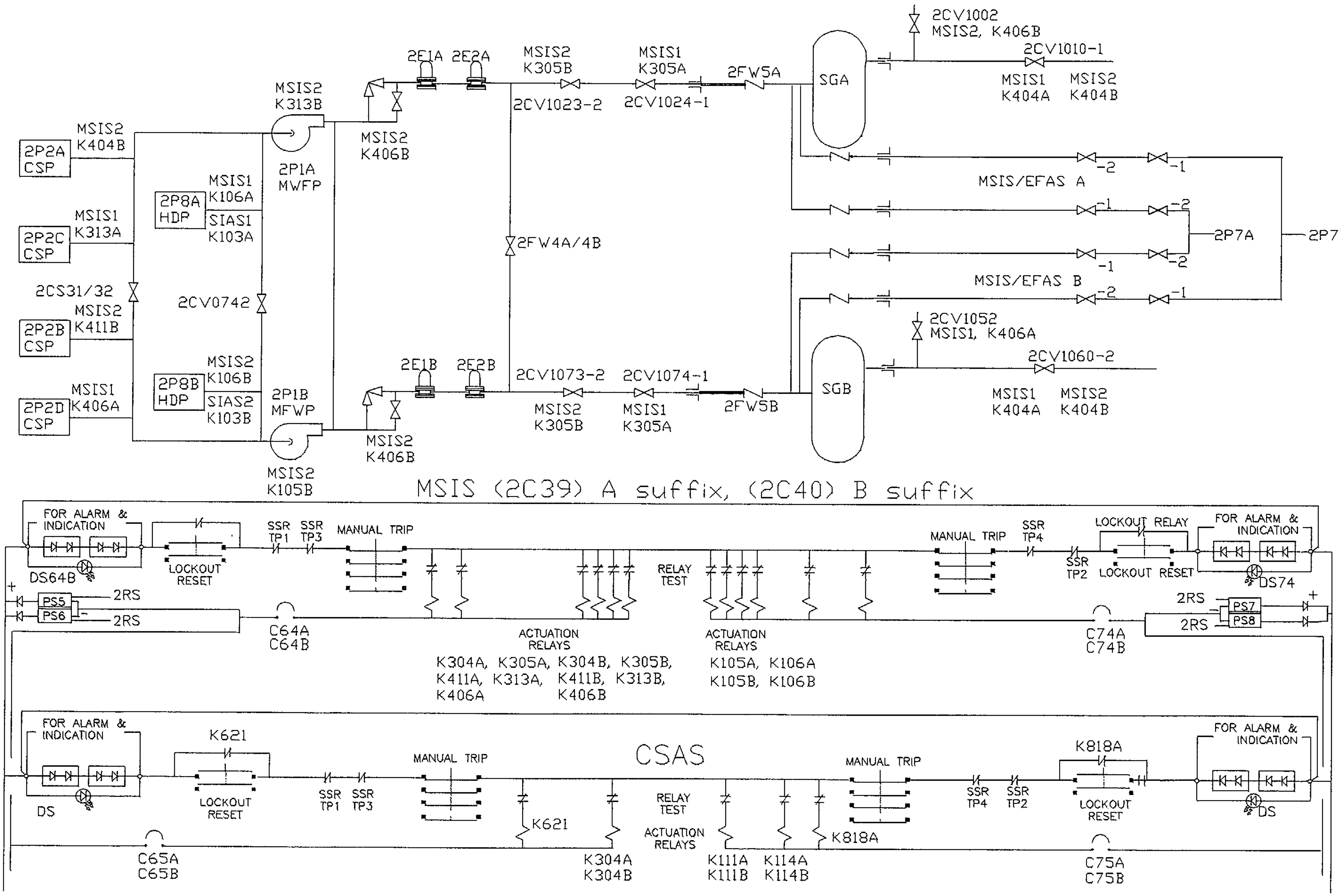
NRC Question #2

In the November 3, 1999, letter, Entergy Operations, Inc. requested NRC concurrence with the conclusion that the existing containment wide range pressure transmitters satisfy the intent of NRC Regulatory Guide 1.97 in regard to their calibrated range (see page 9 of 11 of Enclosure 5 to ANO's November 3, 1999, letter). The NRC staff requests confirmation that these transmitters are used only for indication and recording purposes and are not used for any automatic protective functions.

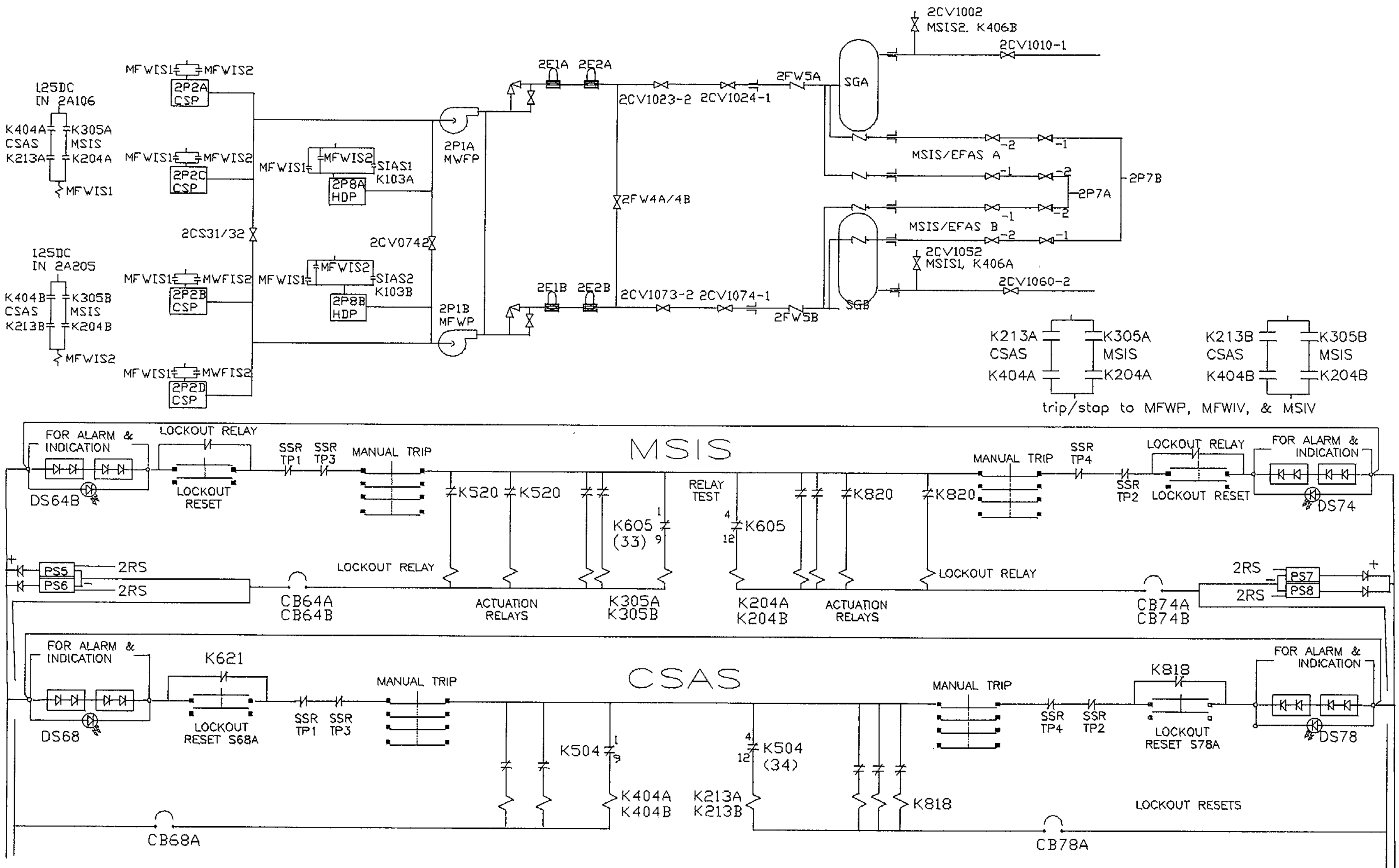
ANO Response

The containment wide range pressure transmitters are, in fact, used only for indication and recording purposes and are not used for any automatic protective functions.

Sketch One
Existing Main Feedwater and Main Steam Isolation



Sketch Two
MSIS and CSAS Isolation of MFW and MS
(contacts shown for power operation)



*Errata for Enclosure 2 of November 3, 1999, Letter

Page 3.8-19	The last paragraph on the page (i.e., the last two lines) should be deleted due to a pagination error in the original submittal.
Page 6.2-6	Change "in" to "was" on the last line of the page. The sentence should read, "Frothing analysis was performed internally."
Page 6.2-12	A comma should be inserted following the word "starts" in the last sentence of the third bullet (third paragraph). In the third sentence of the next to last paragraph, the acronym for "MFWIVs" should be "MFIVs."
Page 6.2-70	This page should be deleted in its entirety.
Page 6.2-80	The page number on this page is incorrect which resulted in section 6.2.2.2.1 incorrectly following section 6.2.4.2. The correct page number should be 6.2-59.
Page 6.7-12	The enthalpy value for time 151.50 secs (top table, fourth line of data) is missing the last significant digit. A zero should be added. The number should be 1213.30.
Page 6.7-34	Note 1 has improper subject-verb agreement. The sentence should read "... 0% Power cases which were slot breaks of 1.94 ft^2."
Page 8.3-37	The last sentence of the added text has incorrect subject-verb agreement. The 's' should be removed from the word "follows." The sentence should read, "As a result, the historical analyses that follow..."
Page 9.4-40	A line of text is missing at the top of the page. The following line should be inserted at the bottom of the page: "105 psig. The conducted environmental tests used the 80 psig base. This has been accepted as an"

* The eight (8) corrected Safety Analysis Report pages are provided in the pages that follow.

Flued head design was used for all containment penetration connections, except for 2P-53 which has closed flanges on both ends to allow for temporary access to containment. 2P-53 does not require a flued head as process piping does not pass through this penetration.

Penetration connections are butt welds without backing bars. All welds were radiographed. Table 3.8-1 gives pipe sizes and materials for all penetrations. Table 3.8-2 shows results of radiographic examinations and corrective measures for defects.

The Topical Report BC-TOP-1, Revision 1, Reference 6, constitutes the basic approach used in the design of the liner plate.

There is a minor difference in the design of the Unit 2 liner plate from that presented in the topical report. The 1/4-inch liner plate material is ASTM A-516, Grade 60, with a specified yield stress of 32,000 psi, instead of ASTM A-442, which has a specified yield stress of 30,000 psi.

3.8.1.4.3 Computer Programs Used in the Analysis

Computer programs used in the original analysis of the containment are presented in Table 3.8-3.

3.8.1.5 Structural Acceptance Criteria

The fundamental acceptance criterion for the completed containment is the successful completion of the initial and uprate structural integrity tests which measure responses within the limits predicted by analyses. The limits are predicted based on test load combinations and code allowable values for stress, strain, or gross deformation for the range of material properties and construction tolerances. In this way the margins of safety associated with the design and construction of the containment are, as a minimum, the accepted margins associated with nationally recognized codes of practice.

The Structural Integrity Test (SIT) ~~is planned to~~ yields information on both the overall response of the containment and the response of localized areas, such as major penetrations or buttresses, which are important to its design functions.

The design and analysis methods, as well as the type of construction and construction materials, were chosen to allow assessment of the structure's capability throughout its service life. Additionally, surveillance testing provided further assurances of the structure's continuing ability to meet its design functions.

~~Table 3.8-4 shows the values of calculated stresses and stains at critical sections of the containment, allowable values of stresses and strains. These values indicate the margin of safety provided in the design.~~ During the 2R14 Steam Generator Replacement Outage the containment design pressure will be increased from 54 psig to 59 psig. An uprate Structural Integrity Test (SIT) will be performed to demonstrate the acceptability of the containment at the 59 psig pressure.

the single-wall-model temperature elevated which conservatively maximizes the thermal driving force for wall-to-coolant heat transfer for all the walls actually in the RCS when they are represented by a single wall model.

9. The core-to-coolant heat transfer model considers extended nucleate boiling in the core. The Thom Nucleate Boiling correlation was used when the core surface is exposed to primary coolant. When nucleate boiling can no longer be supported, CEFLASH-4A considers a variety of transitional boiling or convective heat transfer correlation, depending on the localized conditions.

10. Heat Transfer across the steam generator tubes is modeled with the same heat transfer coefficient in both the forward and reverse directions.

11. Emergency feedwater flow is conservatively omitted since it would cool the secondary sides.

<u>Reflood and Post-Reflood Phases for Cold Leg Breaks</u>

Following the initial blowdown, the reactor is first refilled by the incoming safety injection flow, including SITs, and then reflooded as the core becomes quenched. The effect of the SGs on the mass and energy to containment is important for cold leg breaks after blowdown because the exiting steam passes through the SGs prior to exiting the RCS to the containment.

The next phase of the transient simulation is the reflood phase, which is defined as the time period during which the coolant accumulating in the reactor vessel increases from the bottom of the active core to two feet below the top of the active core. At this point, the core is considered to be quenched and the entrainment reduces significantly.

The reflood and post-reflood phases of the LOCA are simulated using the NRC-approved FLOODMOD2 methodology, Reference 84. The FLOOD3 computer code, Reference 86 is used to implement this methodology. The important features implemented by the FLOOD3 code for this analysis are summarized below.

1. Intact and broken loops were treated individually.
2. Specific volumes of the fluids in the primary loop were varied with time.
3. A uniform methodology for both Reflood and Post-Reflood time frames was used.
4. A rigorous heat transfer scheme was used for treating the transfer of energy from the loop and steam generator walls to the fluid in the primary loop.
5. Containment backpressure was bounded conservatively.
6. Safety injection flow rates were computed explicitly.
7. The fluid in the vessel was heated mechanistically.
8. SIT and safety injection enthalpies were input separately.
9. Frothing analysis was performed internally.

- Following the closure of the MSIVs, the flow of steam to the containment from the intact SG and isolated steam line cease. Although the intact SG repressurizes due to heat transfer from the RCS and eventually becomes an energy source, its affect on the containment response is small after the MSIVs close.

- The contribution of main feedwater flow is reduced significantly when the pumps trip off on the CSAS signal.

- The 0% power cases assume that feedwater flow to the steam generators is from the AFW system at the initiation of the steam line break. Once the transient starts, flow to the intact steam generator is diverted to the ruptured steam generator.

Similar to the LOCA, the MSLB containment analysis is performed in two parts. The SGNIII computer code was used to determine the mass and energy discharged into containment. These data were used to determine the containment response using the COPATTA computer code. This subsection provides an overview of the analyses and a summary of the important results.

6.2.1.1.3.2.1 **<u>Mass and Energy Analysis</u>**

6.2.1.1.3.2.1.1 **<u>Methodology</u>**

The SGNIII computer code, Reference 90, was used to determine the mass and energy data. SGNIII is a coupled primary and secondary model that calculates a time dependent mass and energy release.

The RELAP5 MOD3 code, Reference 91, was used to calculate the contribution of main feedwater, including flashing, to the affected and intact SGs. The code simulated the main feedwater trains including the main feedwater, condensate, and heater drain pumps, various valves and feedwater heaters. No credit was taken for closure of the main feedwater regulating valves. The primary inputs to the RELAP5 MOD3 code were the transient pressures of the intact and affected SGs. The time dependent RELAP5 MOD3 outputs of feedwater flowrate and its associated enthalpy (to each SG) were input directly to SGNIII.

The backup valves were built to the same code class as the piping in which they are installed (B31.1) but they have seismically qualified, Class 1E operators. Material traceability was required for the valve bodies. The valve actuation time is 18.5 seconds or less and they are located immediately upstream of the MFIVs. This assures that their performance meets or exceeds that of the existing valves which close within 25 seconds. The backup valves were installed during the first refueling outage. Annunciator alarms were added to notify operations should either backup valves' breaker open. See the FSAR analysis for additional details with respect to the reason for adding these valves.

While the mass and energy release analysis was conducted separately from the containment response analysis, the COPATTA containment code was used to predict the times for the containment pressure to reach the CPH and CPHH setpoints.

6.2.2.2.1 Containment Spray System

A. General Description

A flow diagram of the CSS is shown in Figure 6.2-17.

The CSS consists of two separate loops of equal capacity and is independently capable of meeting CHRS requirements. Each loop consists of a containment spray pump, shutdown cooling heat exchanger, spray header, isolation valves, and the necessary piping, instrumentation and controls. The loops are supplied with borated water from a common Refueling Water Tank (RWT).

Upon system activation, the containment spray pumps are started and deliver boric acid to the respective spray headers. The spray headers are located at the highest possible level in the containment to maximize heat and iodine removal. Each header conforms to the shape of the containment dome. Figures 6.2-18 and 6.2-19 show details of the spray headers. The headers are located outside of and above the movable missile shield, and contain 131 spray nozzles each. During normal plant operation CSS piping is maintained full of water from the RWT to Elevation 505 feet, 0 inches (minimum) in the 6-inch diameter risers within containment. When low level is reached in the RWT, the Recirculation Actuation Signal (RAS) automatically transfers the containment spray pump suction to the containment sump by opening the recirculation line valves and closing the RWT outlet and pump minimum flow recirculation valves.

During the injection mode, prior to the start of recirculation, each spray train will deliver a minimum flow of 1875 gpm. At the start of recirculation, with suction from the containment sump, minimum spray flow increases to the nominal design minimum of 2,000 gpm.

Following the switchover of suction to the containment sump, the sump solution will contain boric acid and trisodium phosphate dodecahydrate (TSP-C). This mixture of boric acid and TSP-C will continue to remove post-accident energy and remove and retain fission product iodine as it is recirculated through the CSS. The TSP-C is stored in three containers constructed with wire mesh sides which allow the sprayed fluid to permeate the containers. These containers will become submerged in the sprayed fluid accumulating in the building allowing the TSP-C to dissolve. The TSP-C is used to raise the pH of the sump fluid to an equilibrium pH of 7.0 or greater. A pH of 7.0 or greater will assure that the iodine washed out of the reactor building atmosphere by the spraying action will not re-evolve from the liquid as it is sprayed back into the building.

In the recirculation mode, the spray water is cooled by the shutdown cooling heat exchangers prior to discharge into the containment. The shutdown cooling heat exchangers are cooled by the SWS which is described in Section 9.2.1.

Table 6.2-8B (continued)
MASS AND ENERGY RELEASE DATA
LIMITING CASE (LOCA)
(DEDLS Break With Loss of an Emergency Diesel Generator)

Reflood and Post-Reflood (Spillage to the sump and condensation)					
Time (sec)	Mass Rate (lbm/hr)	Energy Rate (Btu/hr)	Enthalpy (Btu/lbm)	Integral Mass (lbm)	Integral Energy (Btu)
150.20	3.985E+04	5.359E+07	1344.66	6.981E+04	8.638E+07
150.60	1.216E+05	1.523E+08	1253.14	6.982E+04	8.640E+07
151.00	7.171E+04	1.029E+08	1435.61	6.984E+04	8.642E+07
151.50	1.200E+05	1.456E+08	1213.30	6.986E+04	8.645E+07
152.00	6.707E+04	8.829E+07	1316.39	6.988E+04	8.647E+07
152.50	1.151E+05	1.530E+08	1328.37	6.990E+04	8.650E+07
152.51	0.000E+00	0.000E+00	1328.37	6.990E+04	8.650E+07

Reflood and Post-Reflood (Spillage to the sump and condensation)					
14.91	0.000e+00	0.000E+00	101.65	0.000E+00	0.000E+00
20.79	0.000E+00	0.000E+00	101.65	0.000E+00	0.000E+00
20.80	2.257E+07	2.293E+09	101.62	6.269E+01	6.371E+03
22.80	2.087E+07	2.145E+09	102.81	1.165E+04	1.198E+06
25.80	1.878E+07	1.962E+09	104.46	2.730E+04	2.833E+06
27.80	1.761E+07	1.858E+09	105.55	3.709E+04	3.865E+06
31.90	1.556E+07	1.677E+09	107.79	5.481E+04	5.776E+06
35.90	1.391E+07	1.530E+09	110.00	7.026E+04	7.476E+06
41.90	1.190E+07	1.350E+09	113.44	9.010E+04	9.726E+06
45.90	1.077E+07	1.247E+09	115.79	1.021E+05	1.111E+07
51.90	9.298E+06	1.113E+09	119.71	1.176E+05	1.297E+07
55.90	8.431E+06	1.034E+09	122.63	1.269E+05	1.412E+07
61.90	7.262E+06	9.265E+08	127.58	1.390E+05	1.566E+07
65.90	6.556E+06	8.614E+08	131.38	1.463E+05	1.662E+07
71.90	5.589E+06	7.716E+08	138.06	1.556E+05	1.790E+07
93.90	0.000E+00	0.000E+00	138.06	1.556E+05	1.790E+07
114.90	0.000E+00	0.000E+00	276.47	1.556E+05	1.790E+07
115.00	9.551E+04	2.640E+07	276.47	1.556E+05	1.790E+07
120.00	1.566E+05	4.329E+07	276.45	1.559E+05	1.796E+07
130.00	2.658E+05	7.349E+07	276.47	1.566E+05	1.817E+07
140.00	4.258E+05	1.177E+08	276.48	1.578E+05	1.849E+07
148.30	5.385E+05	1.489E+08	276.48	1.590E+05	1.884E+07
152.51	0.000E+00	0.000E+00	276.48	1.590E+05	1.884E+07

Table 6.2-9C

MAXIMUM CONTAINMENT PRESSURE AND TEMPERATURE RESULTS (MSLB)

Power[1]	Failure	Peak Pressure		Peak Temperature	
		psig	sec.	°F	sec
102%	Main feedwater pump to trip	55.3	141.8	410	43.1
94.9%	Main feedwater pump to trip	54.9	149.0	409	43.0
75%	Condensate pump to trip	54.0	157.8	407	42.8
50%	Main feedwater pump to trip	55.4	182.4	405	42.5
25%	Condensate pump to trip	55.3	228.0	403	42.2
0%	Containment Spray Train	57.7	196.6	398	45.5
0%[2]	1 Train of Containment Sprays and 1 Train of Containment Air Coolers (Tech. Spec. LCO case)	58.3	196.4	398	45.6

Note:

1 All cases were double-ended guillotine breaks except the 0% Power cases which were slot breaks of 1.94 ft^2.

2 This case does not represent the DBA. This case, however, does represent the limiting case for the Technical Specification (TS) Limiting Condition for Operation (LCO). In addition to the typical single failure peak containment pressure cases presented in the original FSAR, other cases have been assessed to determine the results of peak pressure conditions under the bounding TS LCO action statements (One CSS and one CCS available bounds two CSS available and two CCS out of service) for containment heat removal systems. The results of these additional analyses demonstrate that peak pressures are bounded by the containment design pressure of 59 psig.

Table 6.2-9D

SEQUENCE OF EVENTS
LIMITING CONTAINMENT PEAK PRESSURE ANALYSIS (MSLB)

(Slot break initiated from 0% power with failure of one containment spray train)

Time (sec)	Event Description
0.0	Start of Event
3.2	Containment Air Cooler Actuation Signal (CPH)
7.0	Containment Spray Actuation Signal (CPHH)
12.1	MSIV shuts
27.1	Backup MFIVs shut
33.2	Containment Air Coolers Start
45.6	Containment Spray Starts (time of peak containment temperature)
196.6	Time of Peak Containment Pressure
400.0	End of Analysis

In the event of an accident some systems are required to function under adverse environmental conditions to initiate or monitor engineered safety features. Typical components related to some of these systems are electrical penetration assemblies. During an accident event these assemblies are exposed to high levels of pressure and superheated containment conditions which they must be able to withstand without loss of function.

The following paragraphs describe historical analyses pertaining to the MSLB. The MSLB analyses completed for the new Westinghouse steam generators along with power uprated conditions produce lower temperatures than that presented below due to the flow limiting devices installed in the outlet of the new steam generator nozzles (refer to SAR section 6.2). As a result, the historical analyses that follow are considered bounding for the new Westinghouse steam generators and power uprated conditions.

In order to examine the influence of accident events on safety-related component temperatures an analysis was conducted with the Bechtel COPATTA computer code. For the analysis, a typical containment penetration was modeled as a slab, with a 1/16-inch steel cover, 2-inch air gap to a simulated cable consisting of 0.1-inch insulation and 0.2-inch thick copper core. The outside surface is covered with organic paint of 0.006-inch thickness.

For reasons of high containment superheated temperature conditions during a Main Steam Line Break (MSLB) event (exceeding 400 °F), a transient temperature analysis was performed to determine the effect of superheat on safety-related equipment. The calculations were carried out using assumptions for containment initial conditions, heat removal systems, blowdown data and condensing heat transfer coefficients that maximize heat transfer to exposed components. A 60 percent break area MSLB with a realistic spray initiation time of 56 seconds produced a temperature of 410 °F at 46 seconds. The surface temperature of the electrical penetration reached a peak value of 255 °F which is well below the design temperature of 300 °F. This substantial difference in containment vapor temperature and component temperature is due to the fact that the energy transferred into the heat sinks is a function of the heat transfer mechanism. Energy transfer into heat sinks is a maximum as long as the heat sink surface temperature is lower than T(SAT) since condensation occurs. When the heat sink temperature equals T(SAT), energy transfer is restricted to convective heat transfer which is significantly lower than heat transfer by condensation. Therefore, this change in heat transfer mechanism is responsible for heat sink surface temperatures following containment saturation temperature and exceeds it only if superheated conditions prevail over a longer time period than those under consideration. This result implies that maximum heat sink temperatures are produced by an accident that furnished the highest saturation temperature and hence saturation pressure. This accident event is the DBA LOCA which is therefore the design basis for safety-related equipment qualification.

105 psig. The conducted environmental tests used the 80 psig base. This has been accepted as an adequate test basis since the fans and motors within the containment would not experience pressure greater than 59 psig under DBA conditions.

The normal purge supply system for containment accessibility consists of a centrifugal type fan, a hot water heating coil and roll type filter. The purge exhaust system consists of a vaneaxial fan, a roughing filter, a HEPA filter and a charcoal absorber. All components of the purge system, except interior ducts and two isolation valves, are located outside the containment. Ducts are provided inside the containment for adequate distribution. The normal purge system discharge to the atmosphere is monitored for radioactive material and alarmed to prevent release exceeding acceptable limits.

The containment is equipped with four cooling units to cool the Control Element Drive Mechanism (CEDM) shroud. The units are mounted on the removable missile shield at Elevation 426 feet, 6 inches and are ducted down to the shroud. Three units operate continuously during normal conditions with one unit as a standby. The cooling units consist of a fan-coil unit containing a low efficiency filter, a cooling coil and a centrifugal type fan. The units use chilled water and are on the main chilled water system.

The reactor cavity cooling system is designed to take air from the CCS ductwork and supply it around the reactor cavity area to maintain a maximum temperature of 110 °F. The system is equipped with two vaneaxial type fans. One fan operates continuously with the other as a standby.

Following the postulated DBA, a potentially major source of hydrogen production results from the decomposition of water by radiolysis. The elimination of the hydrogen in the containment is accomplished by two hydrogen recombiners (See Section 6.2.5.). Hydrogen samplers indicate the concentration of hydrogen in the containment.

A description of the major system components is given below:

Containment Purge Supply System

Fan (2VSF-2)

Type	Centrifugal
Capacity, cfm	40,000

Motor

Type	Induction
Horsepower rating, hp	60
Voltage, V	480
Phase	3
Enclosure	Open drip-proof
Insulation class	B