Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-14-199

December 17, 2014

10 CFR 50.4
10 CFR 50.34(b)

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

Watts Bar Nuclear Plant, Unit 2
Construction Permit No. CPPR-92
NRC Docket No. 50-391

Subject: **Watts Bar Nuclear Plant Unit 2 - Response to Request for Additional Information Regarding NRC Review of Final Safety Analysis Report Amendment 112 for Chapter 6**

References:
1. TVA letter to NRC, "Watts Bar Nuclear Plant (WBN) - Unit 2 - Final Safety Analysis Report (FSAR), Amendment 112," dated May 30, 2014
2. NRC E-mail to TVA, "Watts Bar, Unit 2 - Final RAIs Regarding Review of SAR Amendment 112 for Chapter 6," dated October 9, 2014 (ADAMS Accession No. ML14286A024)

In Reference 1, the Tennessee Valley Authority (TVA) submitted Amendment 112 to the Watts Bar Nuclear Plant (WBN) Unit 2 Final Safety Analysis Report (FSAR). In reference 2, the Nuclear Regulatory Commission (NRC) submitted a Request for Additional Information (RAI) related to Chapter 6 of the Reference 1 submittal. Enclosure 1 to this letter provides responses to the NRC's request for additional information. Enclosure 2 provides marked-up changes to the Unit 2 FSAR that support TVA's responses in Enclosure 1. These changes will be incorporated in the next amendment to the Unit 2 FSAR.

Two new regulatory commitments made in this letter are listed in Enclosure 3.

If you have any questions or comments, please contact Gordon Arent at (423) 365-2004.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 17th day of December 2014.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosures:

1. Request for Additional Information Watts Bar Nuclear Plant, Unit 2 Chapter 6 Amendment 112
2. Proposed FSAR Updates for Watts Bar Unit 2
3. List of Commitments

cc (Enclosure):

U. S. Nuclear Regulatory Commission, Region II
NRC Resident Inspector Watts Bar Nuclear Plant Unit 1
NRC Resident Inspector Watts Bar Nuclear Plant Unit 2
NRR Project Manager - Watts Bar Nuclear Plant

ENCLOSURE 1

REQUEST FOR ADDITIONAL INFORMATION
WATTS BAR NUCLEAR PLANT, UNIT 2
CHAPTER 6 AMENDMENT 112

The following Nuclear Regulatory Commission (NRC) Requests for Additional Information (RAI) come from the NRC staff in the Containment and Ventilation Branch (SCVB) of the Office of Nuclear Reactor Regulation (NRR).

SCVB-RAI-1

Section 6.3.2.14 states:

> *"Recirculation operation gives the limiting net positive suction head requirement, and the net positive suction head available is determined from the containment pressure, vapor pressure of liquid in the sump, containment sump level relative to the pump elevation and the pressure drop in the suction piping from the sump to the pumps."*

From the above statement it is not clear as to what containment pressure is used in determining the limiting Net Positive Suction Head Available (NPSHA) at the pump inlet. Please revise the FSAR by stating that a containment pressure of zero pound-per-square-inch gage was used in calculating the most limiting (minimum) NPSHA.

TVA Response

The Final Safety Analysis Report (FSAR) Section 6.3.2.14 "Net Positive Suction Head" will be revised to state that a containment pressure of zero pounds per square inch gauge (psig) is used in calculating the most limiting (minimum) net positive suction head available. A mark-up of the proposed FSAR changes are provided in Enclosure 2 of this letter.

SCVB-RAI-2

Section 6.3.2.14, under heading "Residual Heat Removal Pumps" states:

> *"No credit is taken for water level above the RHR sump strainer assembly, and no credit is taken for containment over pressure."*

Explain what is meant by "containment over pressure". In case it implies the pressure developed inside the containment above the normal operating pressure during an accident or an abnormal event, please replace "containment over pressure" with "containment accident pressure". Refer to SECY-11-0014, second paragraph under the heading "Background" for the rationale for terminology correction.

TVA Response:

The wording "containment over pressure" will be revised to "containment accident pressure" in Section 6.3.2.14 (see Enclosure 2 of this letter).

SCVB-RAI-3

In FSAR Section 6.2.1.3.3, under heading "Containment Pressure Calculation", the ice temperature used in the containment pressure calculation as per assumptions (2) and (10) is 15°F. Surveillance Requirement (SR) 3.6.11.1 states: "Verify maximum ice bed temperature is ≤ 27°F". The ice bed temperature is a key parameter for the ice condenser performance for pressure suppression, i.e., assuming a lower ice bed temperature for the long term pressure response is less conservative than using a higher temperature. For a conservative containment pressure calculation, please justify using a non-conservative assumption of ice temperature of 15°F instead of 27°F.

TVA Response:

It is acknowledged that the Technical Specification (TS) initial ice bed temperature range surveillance is higher than the value used in the ice condenser containment integrity analyses. A low initial temperature has historically been used because of the density effect on air mass. To respond to SCVB RAI-3, Westinghouse performed a sensitivity analysis on the Watts Bar Nuclear Plant (WBN) Unit 2 analysis of record while assuming the TS Surveillance Requirement (SR) 3.6.11.1 maximum ice bed temperature of 27°F. This sensitivity analysis identified an increase in calculated pressure. Investigation has determined that the direction of conservatism for the initial temperature is also influenced by the timing of ice bed melt, which is affected by initial ice mass and mass and energy release.

For the current WBN Unit 2 ice mass, a 27°F initial condition for the ice condenser volume produces a higher containment peak pressure than 15°F. The analysis was rerun with different initial ice masses until the results of the analysis closely matched the current FSAR analysis. These studies determined that the initial ice mass needed to be increased from 2.26 million pounds to 2.33 million pounds. This is an increase of 70,000 pounds of ice.

FSAR Section 6.2 will be updated to address these revised analyses, including conforming changes to Section 6.7, as shown in Enclosure 2 to this letter. The Technical Specifications will be revised by January 30, 2015 to reflect the higher ice mass specified by the analysis described above.

SCVB-RAI-4

FSAR Section 6.2.1.1.1, fourth paragraph, item (1) states:

"The design basis blowdown energy of 314.9 x 10^6 Btu and mass of 498.1x 10^3 lb put into the containment. (See Section 6.2.1.3.6)"

The blowdown energy reported in Amendment 111 in same section of FSAR was 317.3 x 10^6 Btu and mass of 502.7x 10^3 lb. Explain the reasons of the reduction in the LOCA blowdown mass and energy. FSAR Section 6.2.1.3.6 does not explain the reasons of the mass and energy reduction.

TVA Response:

The subject integrated blowdown mass and energy data provided in the Amendment 111 contains data that was incorrectly applied. The correct Amendment 111 integrated blowdown energy and mass is 315.7 x 10^6 Btu and 499.8 x 10^3 lb, respectively.

The appropriate data for Amendment 112 is correct as presented, that is "The design basis blowdown energy of 314.9 x 10^6 Btu and mass of 498.1x 10^3 lb put into the containment."

The reason for the reduction in the integrated mass and energy is due to a shorter blowdown transient time from 27.2 seconds to 26.8 seconds for the Amendment 112 reanalysis. The blowdown phase transient was directly affected by the corrections relative to the metal heat capacity calculation with respect to the steam generator tube (primary metal) and subsequent secondary side metal as a result of Nuclear Safety Advisory Letter (NSAL) 14-2, "Westinghouse Loss-of-Coolant Accident Mass and Energy Release Calculation Issue for Steam Generator Tube Material Properties," March 31, 2014. In addition, to mitigate the impact from the work performed in response to RAI-3 above, a reduction in the conservative value for core stored energy was utilized. These changes also have an effect on downstream input for the blowdown calculation and transient time.

For an ice condenser containment design containment pressure analysis the blowdown integrated mass and energy release is used to initialize the LOTIC1 post-blowdown containment pressure response. Since the peak containment pressure occurs after the ice bed has melted out, the total integrated release (which is greater for the Amendment 112 analysis) is of more importance. The total integrated energy release (calculated based upon the WCAP-10325-P-A, Reference 1, evaluation model) has increased from 938.84 x 10^6 Btu (Amendment 111) to 949.05 x 10^6 million Btu for the Amendment 112 reanalysis.

SCVB-RAI-5

FSAR Section 6.2.1.3.6 "Mass and Energy Release Data" refers to Reference 20, WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version," for the evaluation model used for the long term LOCA mass and energy release calculations. Westinghouse has issued Nuclear Safety Advisory Letters (NSALs)-06-6, -11-5, and -14-2 reporting errors in the WCAP-10325-P-A methodology and requires containment analyses should be corrected. These specific NSALs have been addressed by other licensees in recent license amendments. Describe changes in the following containment analyses results using the corrected WCAP-10325-P-A methodology that incorporates corrections listed in the above NSALs: (a) containment peak pressure, (b) containment peak gas temperature for Environment Equipment Qualification (EEQ), (c) containment peak wall temperature, (d) containment sump peak water temperature, (e) pump Net Positive Suction Head Available (NPSHA) for the pumps that draw water from the containment sump during recirculation mode of safety injection and containment cooling, and (f) containment minimum pressure analysis for Emergency Core Cooling System (ECCS) performance capability. Also add statement in the FSAR stating corrected version of WCAP-10325-P which removed errors reported in NSALs-06-6, -11-5, and -14-2 was used for the containment LOCA M&E release analysis.

TVA Response:

Westinghouse NSALs-06-6, -11-5, and -14-2 were issued to utilities for the long-term loss of coolant accident (LOCA) mass and energy (M&E) release analyses of record. These NSALs are applicable to long-term LOCA M&E release calculations performed for Westinghouse-designed pressurized water reactors utilizing the methodology documented in WCAP-10325-P-A and WCAP-8264-P-A, Revision 1 (References 1 and 2, respectively).

The WCAP-10325-P-A (Reference 1) evaluation model is used for the long-term LOCA M&E release calculation for containment integrity analyses for the Watts Bar Nuclear Plant. The containment analysis scope impacted by WCAP-10325-P-A M&E release is in FSAR Section 6.2.1.3.3, "Long-Term Containment Pressure Analysis." The resultant impact of the NSALs on the containment peak pressure was an increase in the calculated peak pressure from 12.6 psig to a value of 12.86 psig. It should be noted that the new analysis performed in response to question 3 above results in a lower peak containment pressure of 12.40 psig. There was no change to the associated peak containment temperature; and likewise there was no adverse impact on the peak sump temperature during the recirculation phase calculated for the LOCA analysis. In addition, neither the pump NPSHA, nor the containment minimum pressure analysis for ECCS performance capability is impacted by the NSALs.

The text for FSAR Section 6.2.1.3.6, "Mass and Energy Release Data," will be modified to acknowledge that Nuclear Safety Advisory Letters (NSALs)-06-6, 11-5, and 14-2 have been addressed in the M&E release calculations.

References:

1. WCAP-10325-P-A, May 1983 (Proprietary) and WCAP-10326-A (Non-Proprietary), "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version."

2. WCAP-8264-P-A, Revision 1, August 1975 (Proprietary) and WCAP-8312-A, Revision 2, (Non-Proprietary) "Topical Report Westinghouse Mass and Energy Release Data for Containment Design."

Enclosure 2

PROPOSED FSAR UPDATES FOR WATTS BAR UNIT 2

subdivided into six subcompartments to allow for maldistribution of break flow to the ice bed.

The conditions in these compartments are obtained as a function of time by the use of fundamental equations solved through numerical techniques. These equations are solved for three distinct phases in time. Each phase corresponds to a distinct physical characteristic of the problem. Each of these phases has a unique set of simplifying assumptions based on test results from the ice condenser test facility. These phases are the blowdown period, the depressurization period, and the long term.

The most significant simplification of the problem is the assumption that the total pressure in the containment is uniform. This assumption is justified by the fact that after the initial blowdown of the reactor coolant system, the remaining mass and energy released from this system into the containment are small and very slowly changing. The resulting flow rates between the control volumes will also be relatively small. These small flow rates then are unable to maintain significant pressure differences between the compartments.

In the control volumes, which are always assumed to be saturated, steam and air are assumed to be uniformly mixed and at the control volume temperature. The air is considered a perfect gas, and the thermodynamic properties of steam are taken from the ASME steam table.

For the purpose of calculation, the condensation of steam is assumed to take place in a condensing node located between the two control volumes in the ice storage compartment.

Replace with "27"

Containment Pressure Calculation

The following are the major input assumptions used in the LOTIC analysis for the pump suction pipe rupture case with the steam generators considered as an active heat source for the Watts Bar Nuclear Plant containment:

Replace with "2.33"

(1) Minimum safeguards are employed in all calculations, e.g., one of two spray pumps and one of two spray heat exchangers; one of two RHR pumps and one of two RHR heat exchangers providing flow to the core; one of two safety injection pumps and one of two centrifugal charging pumps; and one of two air return fans.

(2) 2.26 x 10^6 lbs. of ice initially in the ice condenser which is at 15°F.

(3) The blowdown, reflood, and post reflood mass and energy releases described in Section 6.2.1.3.6 were used.

(4) Blowdown and post-blowdown ice condenser drain temperatures of 190°F and 130°F are used[5].

(5) Nitrogen from the accumulators in the amount of 2955.68 lbs. included in the calculations.

(6) Essential raw cooling water temperature of 88°F is used on the spray heat exchanger and the component cooling heat exchanger. Note: The containment analysis was run at an ERCW temperature of 88°F although the containment spray, component cooling, and residual heat removal heat exchanger UA values are based on an ERCW temperature of 85°F to provide additional conservatism.

(7) The air return fan is effective 10 minutes after the transient is initiated. The actual air return fan initiation can take place in 9 ± 1 minutes, with initiation as early as 8 minutes not adversely affecting the analysis results.

(8) No maldistribution of steam flow to the ice bed is assumed.

Replace with "27"

(9) No ice condenser bypass is assumed. (This assumption depletes the ice in the shortest time and is thus conservative.)

(10) The initial conditions in the containment are a temperature of 120°F in the lower and dead-ended volumes, 110°F in the upper volume, and 15°F in the ice condenser. All volumes are at a pressure of 0.3 psig and a 10-percent relative humidity, except the ice condenser which is at 100–percent relative humidity.

(11) A containment spray pump flow of 4000 gpm is used in the upper compartment. The analyzed diesel loading sequence for the containment sprays to energize and come up to full flow and head in 234 seconds is tabulated in Table 6.2.1-25.

(12) A residual spray (2000 gpm design, 1475 gpm analytical) is used. The residual heat removal pump and spray pump take suction from the sump during recirculation.

During the recirculation phase of a LOCA mass and energy release transient, a portion of the RHR pump flow can be diverted to the RHR sprays. The minimum time before RHR spray can be placed in service, as indicated in the Watts Bar Nuclear Plant System Description N3-72-4001, R19, Containment Heat Removal Spray System, is at least 1 hour after LOCA initiation to ensure adequate RHR flow to the core to remove the initial decay heat. Based on the preceding criteria, the RHR spray initiation was modeled at 4346.7 seconds into the LOCA containment response transient.

A discussion of the core cooling capability of the emergency core cooling system is given in Section 6.3.1 for this mode of operation.

(13) Containment structural heat sink data is found in Table 6.2.1-1. (Note: The dead-ended compartment structural heat sinks were conservatively neglected.)

With these assumptions, the heat removal capability of the containment is sufficient to absorb the energy releases and still keep the maximum calculated pressure well below design.

The following plots are provided:

Figure 6.2.1-1, Containment Pressure Versus Time

Figure 6.2.1-2a, Upper Compartment Temperature Versus Time

Figure 6.2.1-2b, Lower Compartment Temperature Versus Time

Figure 6.2.1-3, Active and Inactive Sump Temperature Transients

Figure 6.2.1-4, Melted Ice Mass Transient

Figure 6.2.1-4a, Comparison of Containment Pressure Versus Ice Melt

Tables 6.2.1-3 and 6.2.1-4 give energy accountings at various points in the transient.

As can be seen from Figure 6.2.1-1 the maximum calculated Containment pressure is 12.86 psig, occurring at approximately 4348 seconds.

Replace with "12.40" Replace with "4346"

Structural Heat Removal

Provision is made in the containment pressure analysis for heat storage in interior and exterior walls. Each wall is divided into a number of nodes. For each node, a conservation of energy equation expressed in finite difference forms accounts for transient conduction into and out of the node and temperature rise of the node. Table 6.2.1-1 is a summary of the containment structural heat sinks used in the analysis. The material property data used is found in Table 6.2.1-5.

The heat transfer coefficient to the containment structures is based primarily on the work of Tagami, Reference [21]. An explanation of the manner of application is given in Reference [3].

When applying the Tagami correlations a conservative limit was placed on the lower compartment stagnant heat transfer coefficients. They were limited to 72 Btu/hr-ft^2. This corresponds to a steam-air ratio of 1.4 according to the Tagami correlation. The imposition of this limitation is to restrict the use of the Tagami correlation within the test range of steam-air ratios where the correlation was derived.

6.2.1.3.4 Short-Term Blowdown Analysis

TMD Code - Short-Term Analysis

(1) Introduction

Insert the following:
"A corrected version of WCAP-10325-P-A computer codes and input, which removed errors reported in References 29, 30 and 31, was used for the containment LOCA M&E release analysis. The NSAL corrections are corrections to calculations in support of the approved methodology, and not a change in methodology."

pressure is below design pressure.

The 1/2 inch diameter break is not sufficient to open the ice condenser inlet doors. For this break, the upper compartment spray is sufficient to condense the break steam flow.

In conclusion, it is apparent that there is a substantial margin between the design deck leakage area of 5 ft^2 and that which can be tolerated without exceeding containment design pressure. A preoperational visual inspection is performed to ensure that the seals between the upper and lower containment have been properly installed.

6.2.1.3.6 Mass and Energy Release Data

Long-Term Loss-of-Coolant Accident Mass and Energy Releases

The evaluation model used for the long-term LOCA mass and energy release calculations is the March 1979 model described in Reference 20. This evaluation model has been reviewed and approved by the NRC.

The time history of conditions within an ice condenser containment during a postulated loss-of-coolant accident (LOCA) can be divided into two periods:

1. The initial reactor coolant blowdown, which for the largest assumed pipe break occurs within approximately 30 seconds.

2. The post blowdown phase of the accident which begins following the blowdown and extends several hours after the start of the accident.

LOCA Mass and Energy Release Phases

The containment system receives mass and energy releases following a postulated rupture in the RCS. These releases continue over a time period, the LOCA analysis calculational model is typically divided into four phases:

1. Blowdown - the period of time from accident initiation (when the reactor is at steady-state operation) to the time that the RCS and containment reach an equilibrium state at containment design pressure.

2. Refill - the period of time when the reactor vessel lower plenum is being filled by accumulator and Emergency Core Cooling System (ECCS) water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, it is assumed that this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Therefore, the refill period is conservatively neglected in the mass and energy release calculation.

(22) "Topical Report Westinghouse ECCS Evaluation Model 1981 Version," WCAP-9220-P-A, Revision 1, February 1982 (Proprietary), WCAP-9221-A, Revision I, February 1982 (Non-Proprietary).

(23) Docket No. 50-315, "Amendment No. 126, Facility Operating License No. DPR-58 (TAC No. 71062), D. C. Cook Nuclear Plant Unit 1," June 9, 1989.

(24) "Mixing of Emergency Core Cooling Water with Steam: 1/3-Scale Test and Summary," WCAP-8423, June 1975 (Proprietary).

(25) ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactor," August 29, 1979.

(26) W. H. McAdams, Heat Transmission, McGraw-Hill 3rd edition, 1954, p. 172.

(27) "Answers to AEC Questions on Report WCAP-8282," WCAP-8282-AD1, May 1974 (Proprietary).

(28) "Long Term Ice Condenser Containment Code - LOTIC Code," WCAP-8354-P-A-S1, April 1976 (Proprietary), WCAP-8355-A-S1, April 1976 (Non–proprietary).

Add the following references:

(29) NSAL-06-6, "LOCA Mass and Energy Release Analysis," June 6, 2006.

(30) NSAL-11-5, "Westinghouse LOCA Mass and Energy Release Calculation Issues," July 25, 2011.

(31) NSAL-14-2, "Westinghouse Loss-of-Coolant Accident Mass and Energy Release Calculation Issue for Steam Generator Tube Material Properties," March 31, 2014.

Table 6.2.1-3 Energy Balances

Sink	Approx. End of Blowdown (t = 10.0 sec)	Approx. End of Reflood (t=243.4 sec)
	(Millions of BTUs)	
Ice Heat Removal*	190.811	246.406
Structural Heat Sinks*	17.686	57.676
RHR Heat Exchanger Heat Removal*	0.0	0.0
Spray Heat Exchanger Heat Removal*	0.0	0.0
Energy Content of Sump**	173.409	229.192
Ice Melted (millions of lbm)	0.6160	0.8386

*Integrated Heat Rates

**Energy Content of Sump Includes Active and Inactive Regions

Replace with the following:

	Approx. End of Blowdown (t = 10.0 sec)	Approx. End of Reflood (t = 239.8 sec)
	(Millions of BTUs)	
Ice Heat Removal*	192.652	245.358
Structural Heat Sinks*	16.805	55.090
RHR Heat Exchanger Heat Removal*	0.0	0.0
Spray Heat Exchanger Heat Removal*	0.0	0.0
Energy Content of Sump**	176.159	230.606
Ice Melted (millions of lbm)	0.6334	0.8497

Table 6.2.1-4 Energy Balances

	Ice Meltout Time Approx. (t=2659.8 sec)	Approx. Time of Peak Pressure (t=4348.6 sec)
	(Millions of BTUs)	
Ice Heat Removal*	601.367	601.389
Structural Heat Sinks*	68.084	100.780
RHR Heat Exchanger Heat Removal*	18.566	38.875
Spray Heat Exchanger Heat Removal*	0.0	32.176
Energy Content of Sump**	597.741	595.857
Ice Melted (millions of lbs)	2.2599	2.2600

*Integrated Heat Rates

**Energy Content of Sump Includes Active and Anactive Regions

Replace with the following:

	Approx. Ice Meltout Time (t = 2873.8 sec)	Approx. Time of Peak Pressure (t=4346.0 sec)
	(Millions of BTUs)	
Ice Heat Removal*	606.794	606.794
Structural Heat Sinks*	69.283	95.687
RHR Heat Exchanger Heat Removal*	21.093	38.362
Spray Heat Exchanger Heat Removal*	3.058	31.405
Energy Content of Sump**	606.616	600.618
Ice Melted (millions of lbm)	2.330	2.330

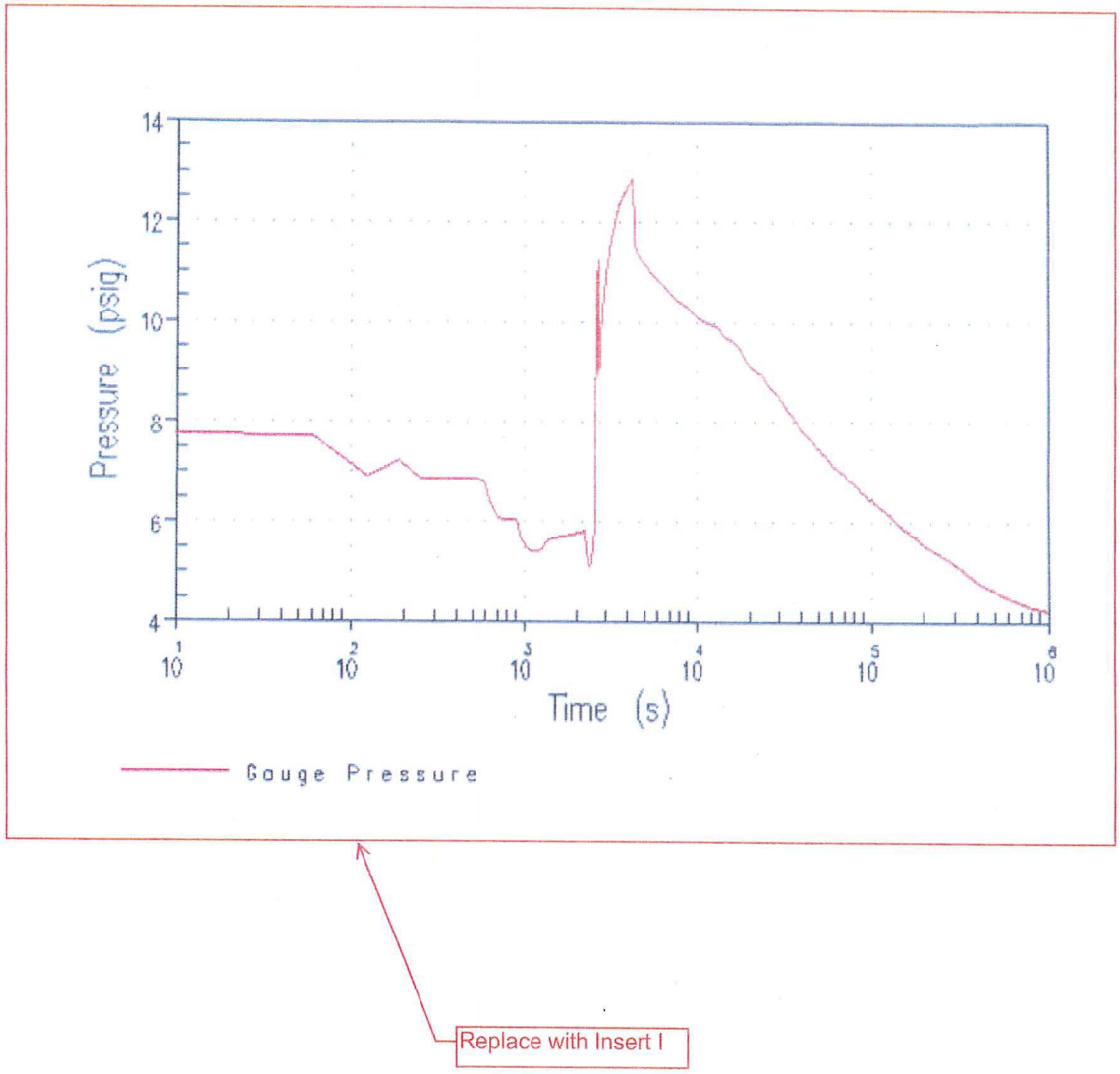
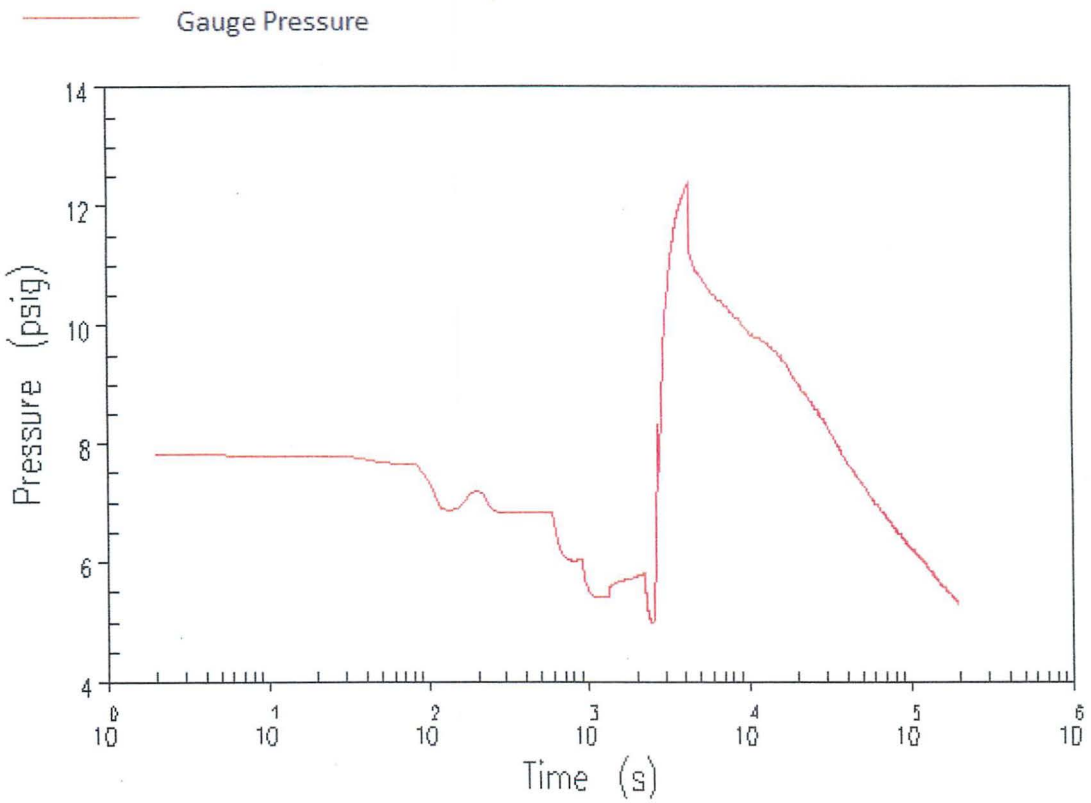


Figure 6.2.1-1 Containment Pressure Versus Time

Insert I


Gauge Pressure
14
12
10
8
6
4
Pressure (psig)
0
10
1
10
2
10
3
10
4
10
5
10
6
10
Time (s)

Replace with Insert J

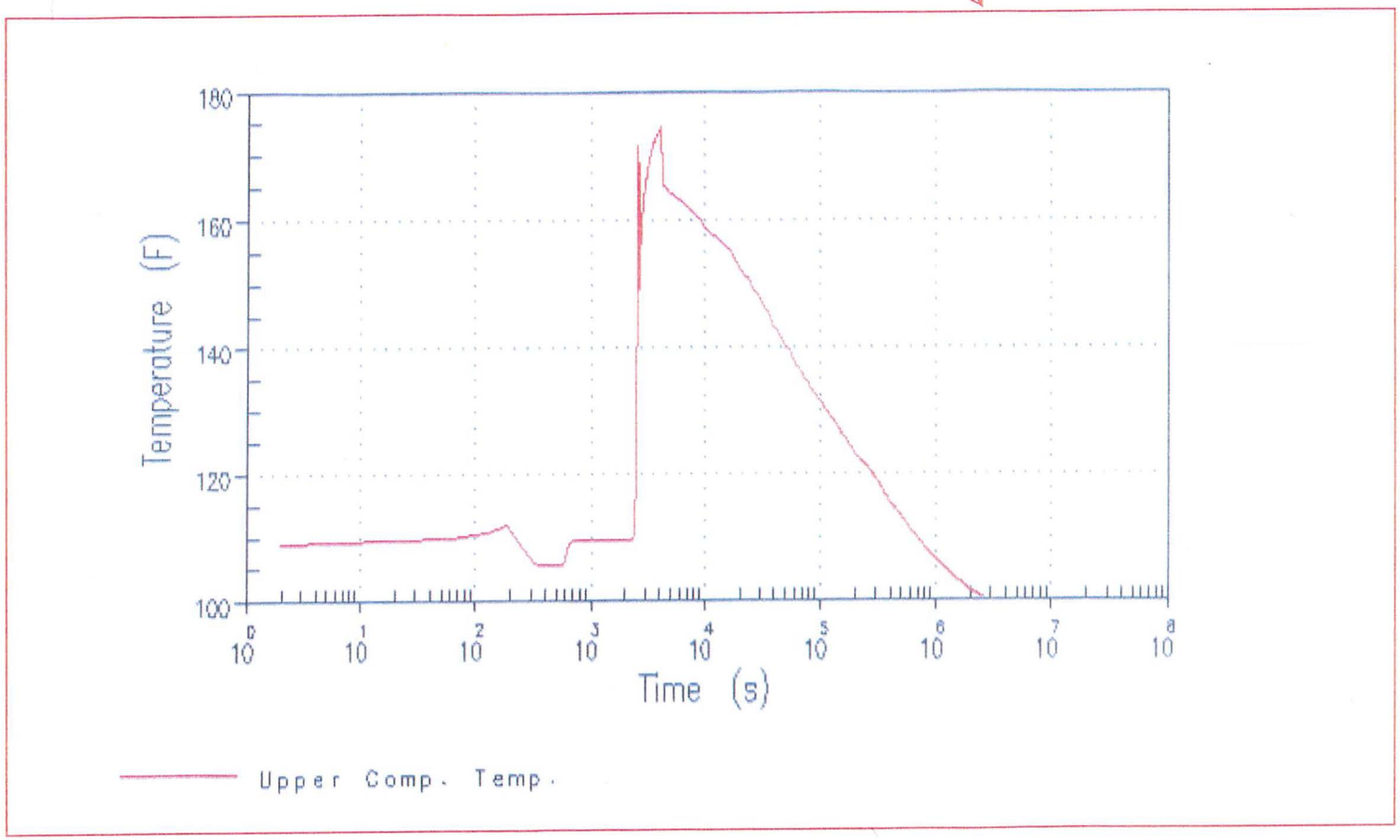


Figure 6.2.1-2a Upper Compartment Temperature Versus Time

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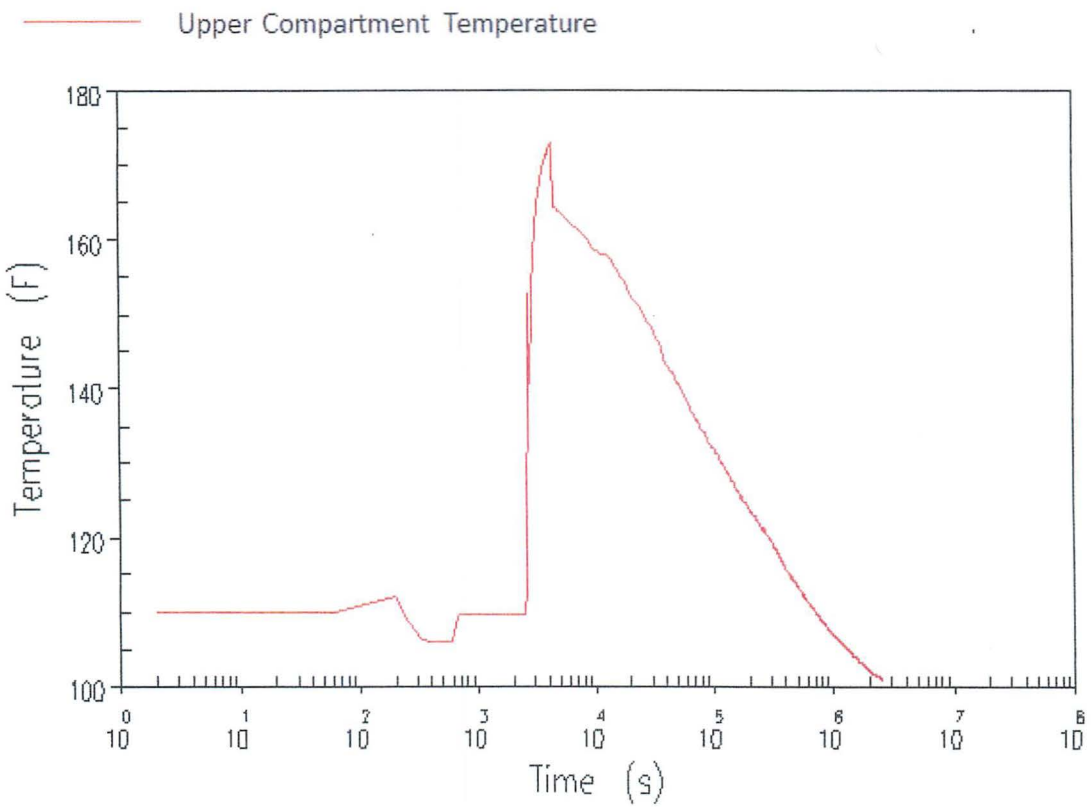

Upper Compartment Temperature
180
160
140
120
100
Temperature (F)
0
10
1
10
2
10
3
10
4
10
5
10
6
10
7
10
8
10
Time (s)

Replace with Insert K

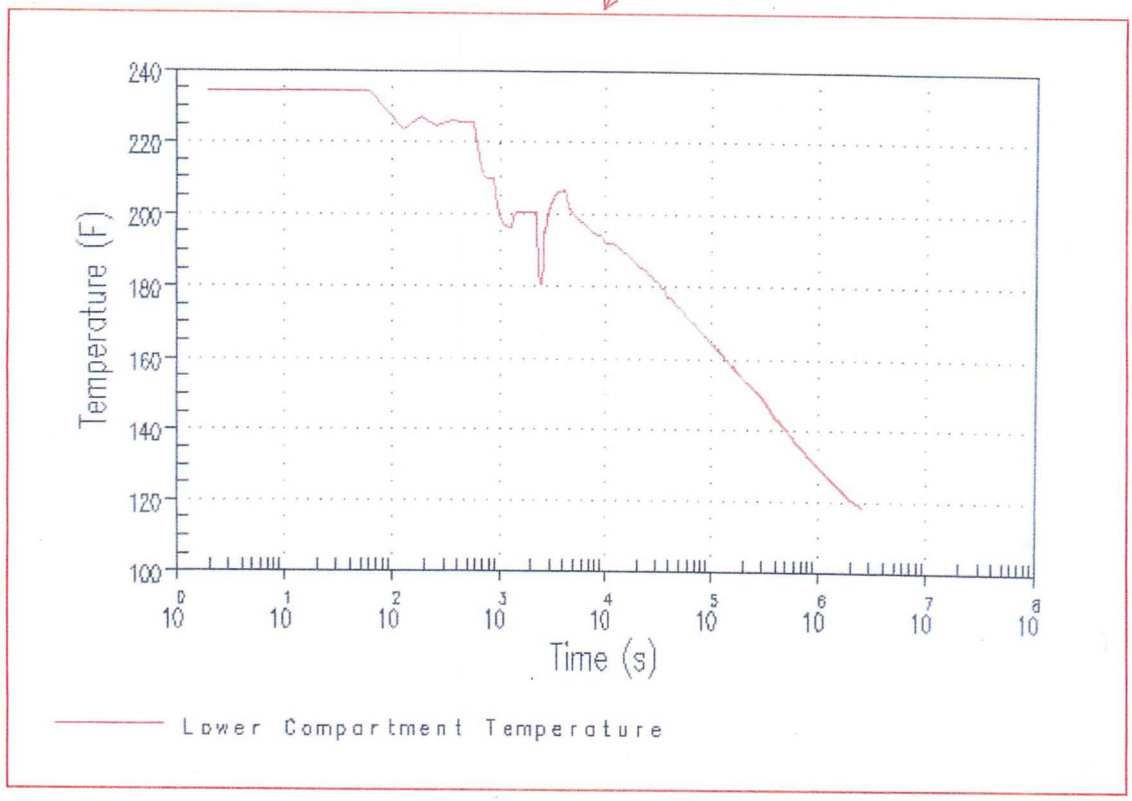
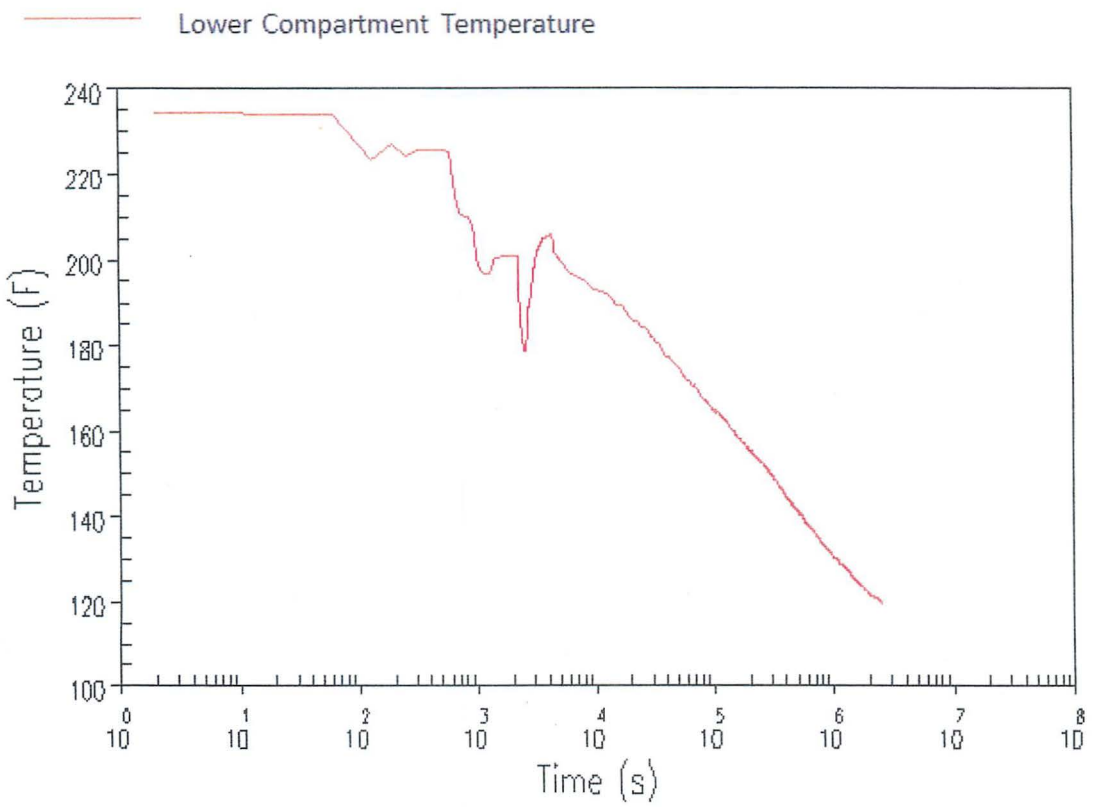


Figure 6.2.1-2b Lower Compartment Temperature Versus Time

Insert K


Lower Compartment Temperature
240
220
200
180
160
140
120
100
Temperature (F)
0
1
2
3
4
5
6
7
8
10
Time (s)

Replace with Insert L

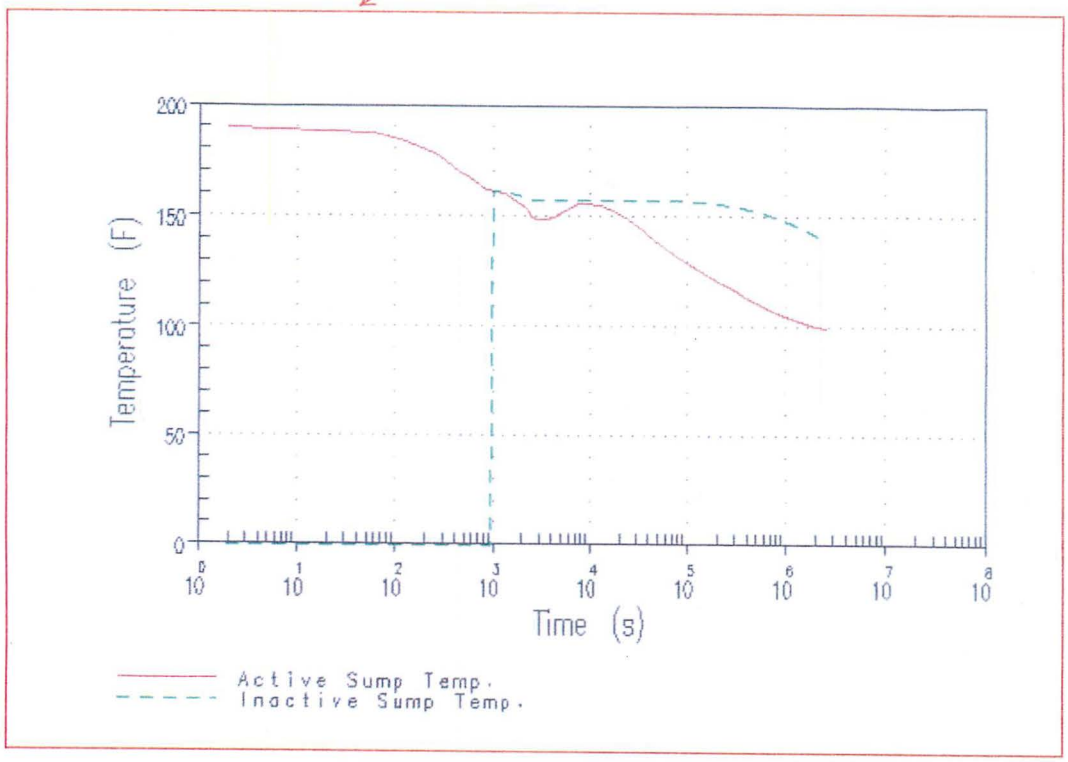


Figure 6.2.1-3 Active and Inactive Sump Temperature Transients

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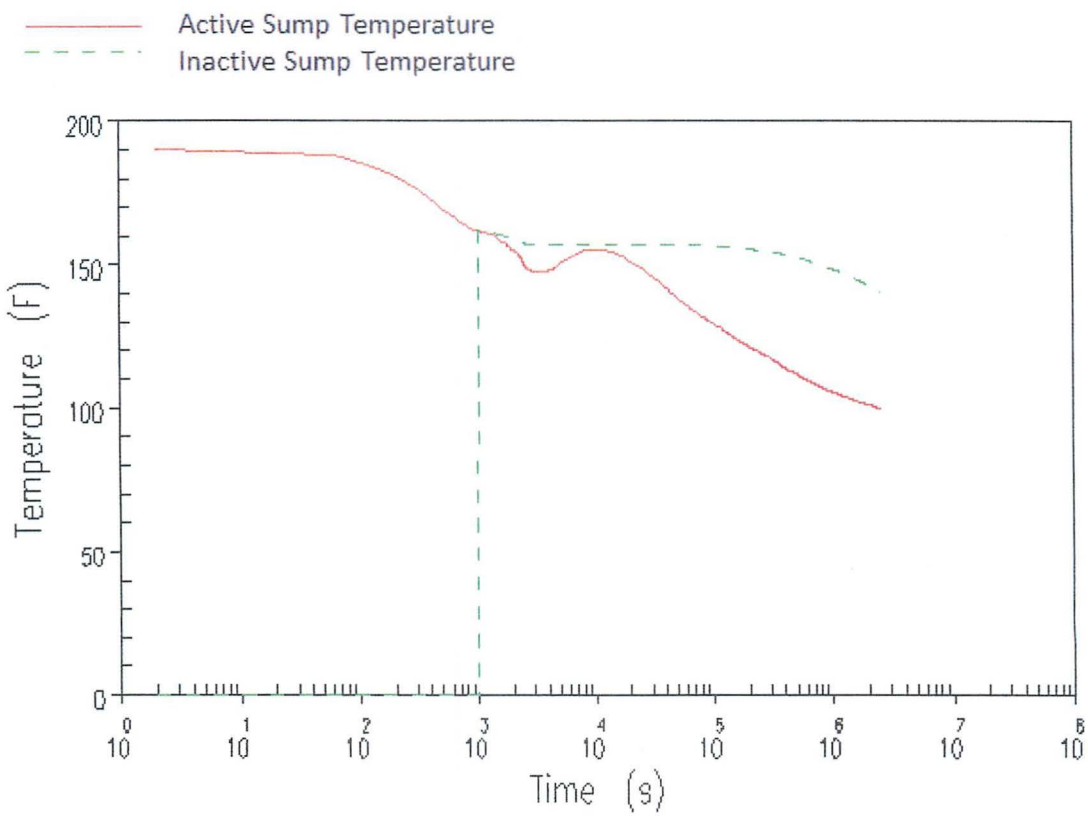

Active Sump Temperature
Inactive Sump Temperature
200
150
100
50
0
Temperature (F)
0
10
1
10
2
10
3
10
4
10
5
10
6
10
7
10
8
10
Time (s)

Replace with Insert M

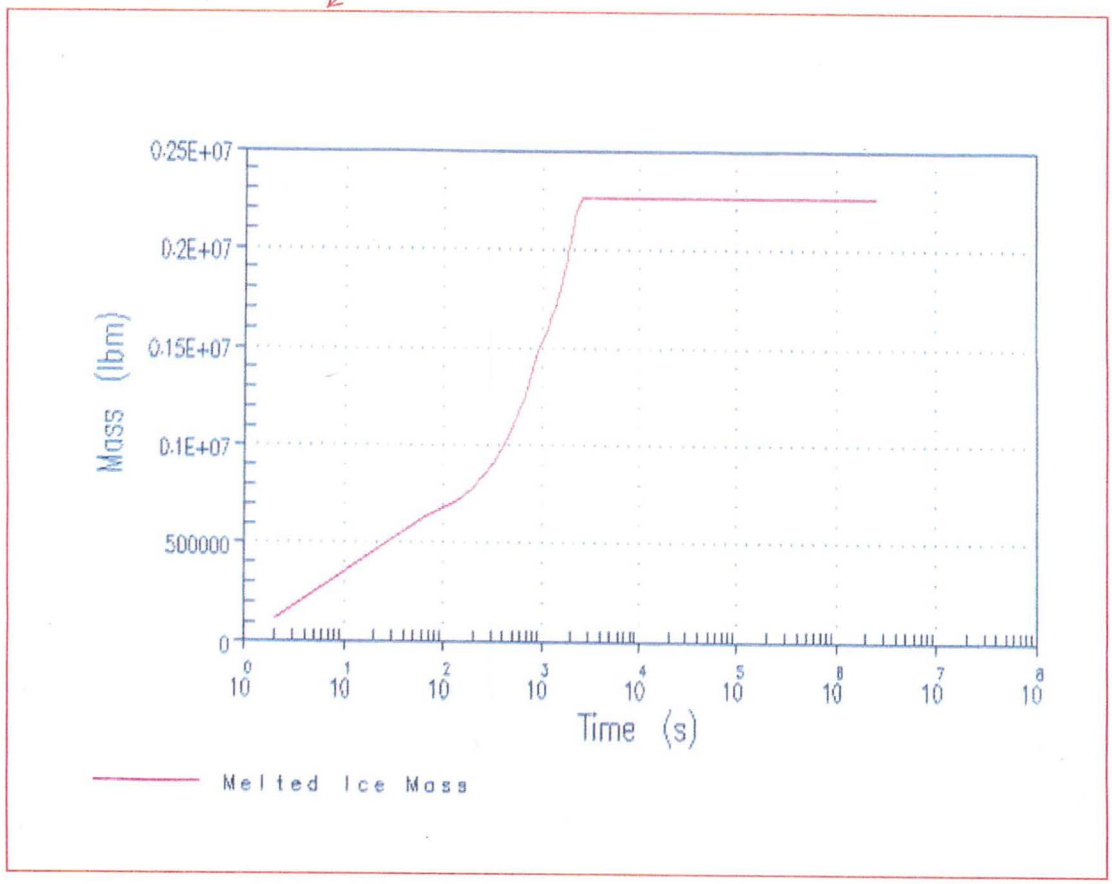
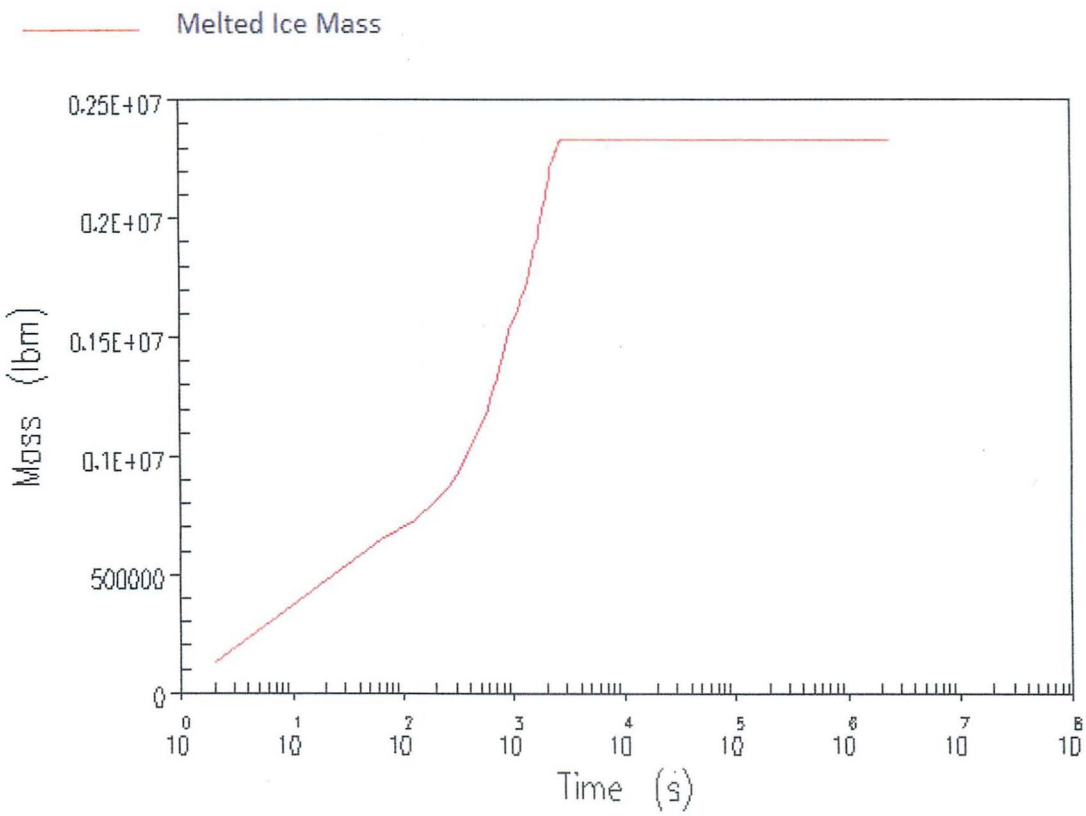


Figure 6.2.1-4 Melted Ice Mass Transient

Insert M


Melted Ice Mass
0.25E+07
0.2E+07
0.15E+07
0.1E+07
500000
0
Mass (lbm)
10^0
10^1
10^2
10^3
10^4
10^5
10^6
10^7
10^8
Time (s)

Replace with Insert N

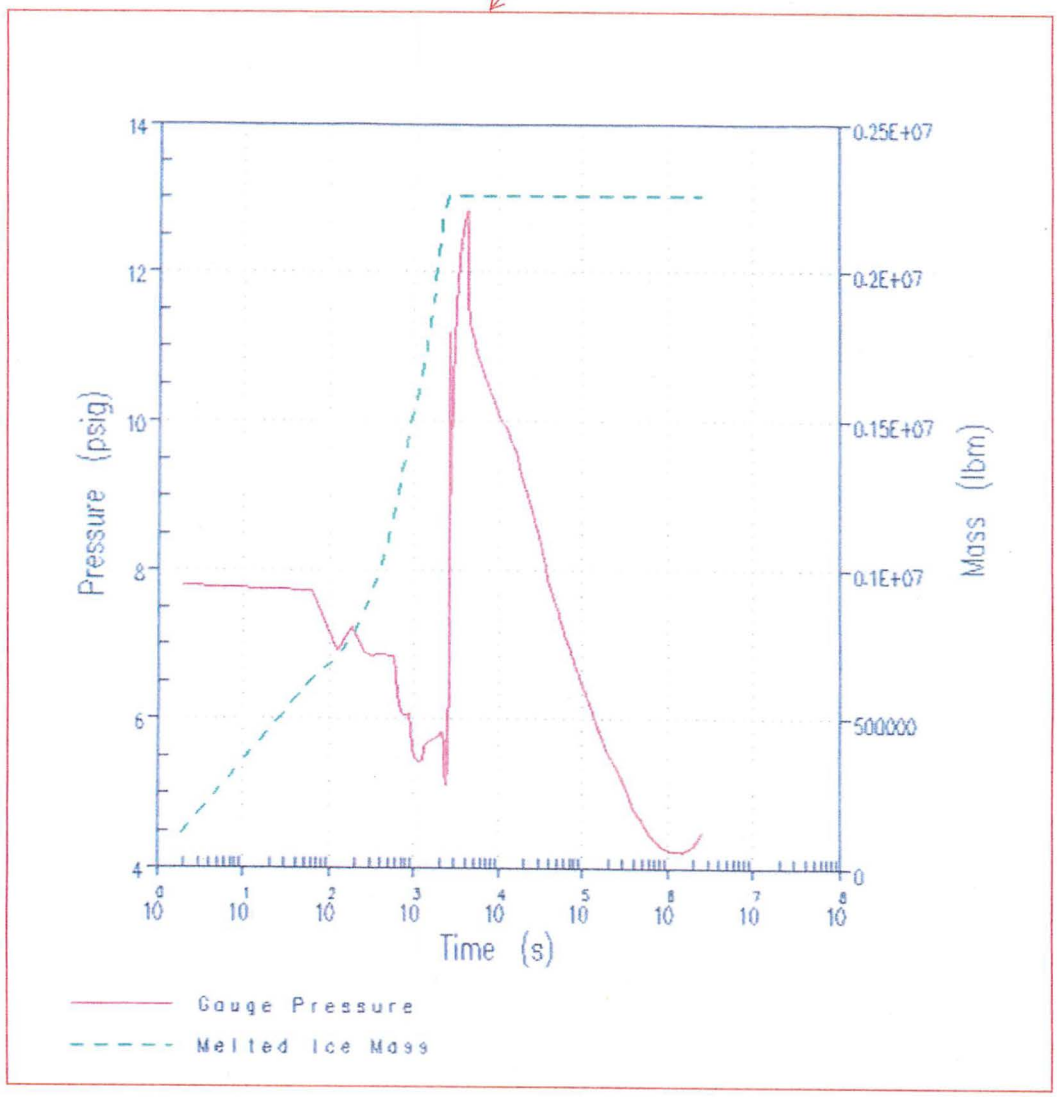
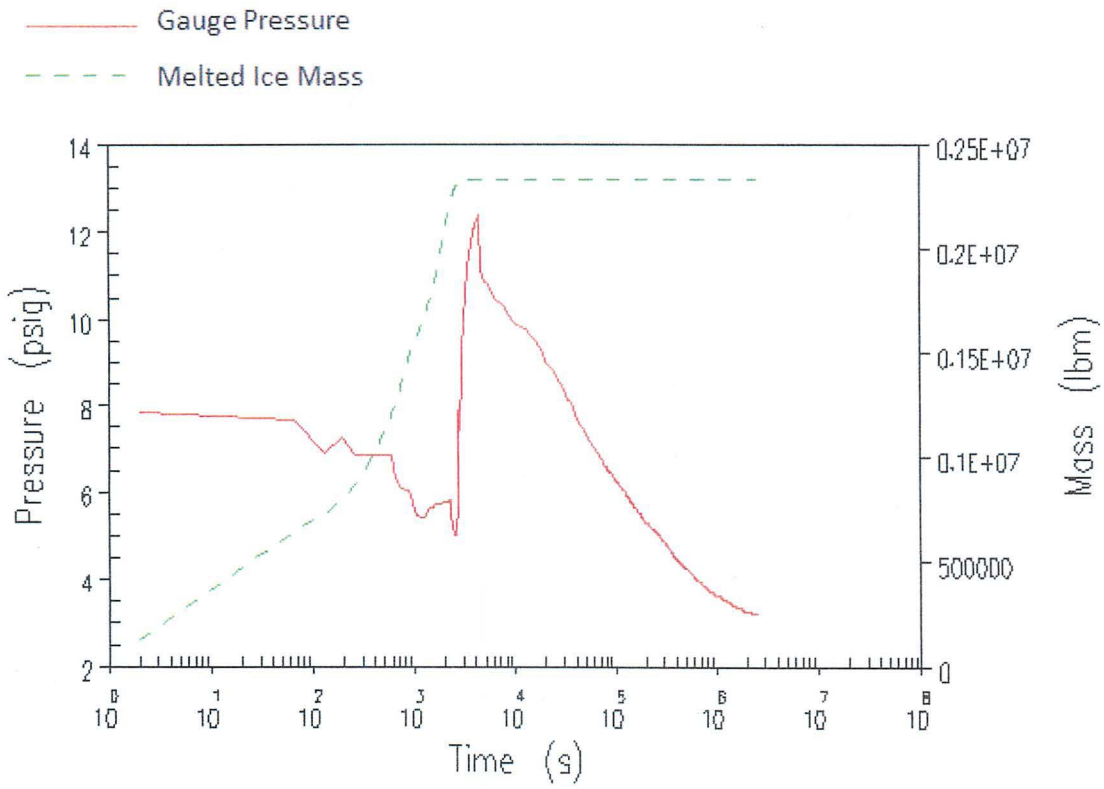


Figure 6.2.1-4a Comparison of Containment Pressure Versus Ice Melt

Insert N


Gauge Pressure
Melted Ice Mass
Pressure (psig)
Mass (lbm)
Time (s)
14
12
10
8
6
4
2
0.25E+07
0.2E+07
0.15E+07
0.1E+07
500000
0
10^0
10^1
10^2
10^3
10^4
10^5
10^6
10^7
10^8

is provided with redundant ESF grade air cleanup and filtration systems as described in Section 6.5.1.

With these design ground rules, continued function of the ECCS meets minimum core cooling requirements. A single passive failure analysis is presented in Table 6.3-9. It demonstrates that the ECCS can sustain a single passive failure during the long term phase and still retain an intact flow path to the core to supply sufficient flow to maintain the core covered and affect the removal of decay heat. An event resulting in maximum leakage would have an insignificant impact on ECCS capability since a 100% capacity redundant train is available to assure the ECCS capability and since the maximum leakage represents less than 0.5% of system flow capacity.

6.3.2.12 Protection Provisions

The provisions taken to protect the system from damage that might result from dynamic effects on piping systems are discussed in Section 3.6. The provisions taken to protect the system from missiles are discussed in Section 3.5. The provisions to protect the system from seismic damage are discussed in Sections 3.7, 3.9, and 3.10. Thermal stresses on the reactor coolant system are discussed in Section 5.2.

6.3.2.13 Provisions for Performance Testing

The provisions incorporated to facilitate performance testing of components are discussed in Section 6.3.4.

6.3.2.14 Net Positive Suction Head

The ECCS is designed so that adequate net positive suction head is provided to system pumps. See Table 6.3-12, Available NPSH During ECCS Operation, for a complete listing of available NPSH for all pumps in ECCS while operating. Adequate net positive suction head is shown to be available for all pumps as follows:

(1) Residual Heat Removal Pumps

The net positive suction head of the RHR pumps is evaluated for normal plant shutdown operation, and for both the injection and recirculation modes of operation for the design basis accident. Recirculation operation gives the limiting net positive suction head requirement, and the net positive suction head available is determined from the containment pressure, vapor pressure of liquid in the sump, containment sump level relative to the pump elevation and the pressure drop in the suction piping from the sump to the pumps. No credit is taken for water level above the RHR sump strainer assembly, ~~and~~ no credit is taken for containment ~~over~~ pressure. The net positive suction head evaluation is based on all pumps ~~operating~~ at the maximum design basis accident flow rates. The RHR pump head capacity curves are given in Figure 6.3-2.

accident

(2) Safety Injection and Centrifugal Charging Pumps

A containment pressure of zero psig (building pressure of 14.3 psia) is used in calculating the most limiting (minimum) NPSHA.

(4) Basket Loading

The ice baskets are capable of being loaded by a pneumatic ice distribution system. The baskets contain a minimum of 2.26 x 10^6 pounds of ice.

(5) External Basket Design

2.33

The baskets are designed to minimize any external protrusions which would interfere with lifting, weighing, removal and insertion.

(6) Basket Coupling

Baskets are capable of being coupled together in 48-foot columns.

(7) Basket Couplings and Stiffening Rings

Couplings or rings are located at 6-ft intervals along the basket and have internal inserts to support the ice from falling down to the bottom of the ice column during and after a DBA and/or SSE.

Design and Test Loads

The minimum test and basic design loads are given in Table 6.7-2.

6.7.4.2 System Design

The ice condenser is an insulated cold storage room in which ice is maintained in an array of vertical cylindrical columns. The columns are formed by perforated metal baskets with the space between columns forming the flow channels for steam and air. The ice condenser is contained in the annulus formed by the containment vessel wall and the crane wall circumferentially over a 300° arc.

The ice columns are composed of four baskets approximately 12 feet long each, filled with flake ice. The baskets are formed from a 14 gage (.075) perforated sheet metal, as shown in Figure 6.7-8. The perforations are 1.0 in. x 1.0 in. holes, spaced on a 1.25-inch center. The radius at the junction of the perforation is 1/16 inch. The ice basket material is made from ASTM-569 and/or A1011 which is a commercial quality, low carbon steel. The basket component parts are corrosion protected by a hot dip galvanized process. The perforated basket assembly has an open area of approximately 64% to provide the necessary surface area for heat transfer between the steam/air mixture and the ice to limit the containment pressure within design limits. The basket heat transfer performance was confirmed by the autoclave test.

Interconnection couplings and stiffening rings are located at the bottom and 6-ft. levels, respectively, of each basket section. The bottom coupling and stiffening ring are cylindrical in shape and approximately 3 inches high with a rolled internal lip and/or welded bottom ring. The lip/ring provides stiffening to the basket and a stop for the cruciforms at 6 feet intervals. These cruciforms prevent the ice in the basket from displacing axially in the event of loss of ice caused by sublimation or partial melt down

6.7.6 Refrigeration System

6.7.6.1 Design Basis

Function

The refrigeration system serves to cool down the ice condenser from ambient conditions of the reactor containment and to maintain the desired equilibrium temperature in the ice compartment. It also provides the coolant supply for ice machines A, B, and C during ice loading. The refrigeration system additionally includes a defrost capability for critical surfaces within the ice compartment.

During a postulated loss-of-coolant accident the refrigeration system is not required to provide any heat removal function. However, the refrigeration system components which are physically located within the containment must be structurally secured (not become missiles) and the component materials must be compatible with the post-LOCA environment.

Design Conditions

(1) Operating Conditions

See individual component sections:

(A) Floor cooling - Section 6.7.1

(B) Air handling units (AHUs) - Section 6.7.7

(2) Performance Requirements

(A) The mandatory design parameters that relate to refrigeration performance are:

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(i)	Maximum total weight of ice in columns	3.0×10^6 lbs
(ii)	Minimum total weight of ice in columns	2.26×10^6 lbs
(iii)	Nominal ice condenser cooling air temperature	10°F - 15°F

(B) The design must also provide a sufficiently well-insulated ice condenser annulus such that, with a complete loss of all refrigeration capacity, sufficient time exists for an orderly reactor shutdown prior to ice melting. A design objective is that the insulation of the cavity is adequate to prevent ice melting for approximately 7 days in the unlikely event of a complete loss of refrigeration capability.

(C) The not-directly-safety-related design-objective parameters are:

(i) Ice Sublimation

Ice sublimation and mass transfer is reduced to the lowest possible limits by maintaining essentially isothermal conditions

Table 6.7-18 Refrigeration System Parameters Continued
(Sheet 2 of 2)

2.4	Refrigeration Medium (glycol) - UCAR Thermofluid 17 or equal			
	Concentration, ethylene glycol in water - 50 weight % or 47.8 volume %			
	At temperature:	-5°F	0°F	100°F
	Specific gravity	1.083	1.082	1.056
	Absolute viscosity (centipoises)	25.0	20.5	2.3
	Kinematic viscosity (centistokes)	23.1	18.9	2.18
3.0	Ice Condenser (per one containment unit)			
3.1	Ice Bed			
	Amount of ice initially stored per unit, maximum			3.0×10^6lbs
	Minimum amount of ice			2.26×10^6lbs
	Ice displacement per year, design objective			2%
	Design predicted ice displacement per year to wall panels for normal operation			<0.3%
	Ice melt during maximum LOCA, calculated, approx.			See Section 6.2.1
	Temperature of ice & static air			15°F to 20°F nominal
	Pressure at lower doors due to cold head, nominal			1 psf
	Inlet opening pressure			1 psf
3.2	Air Handling Units - 30 dual packages installed per Containment			
	Refrigeration requirements per containment, calculated, nominal			51.5 tons
	Gross capacity per dual package rated			2.5 tons
	Glycol entering temperature, approx.			-5°F
	Glycol exit temperature, approx.			1°F
	Glycol flow per air handler (1/2 package)			6 gpm nominal
	Total glycol flow, 30 x 2 x 6			360 gpm nominal
	Glycol pressure drop, estimated			50 feet
	Air blower head			2' H_2O
	Air entering temperature, estimated			15°F
	Air exit temperature			10°F nominal
	*Maximum ice weight not to exceed 3.0×10^6 lbs. [20]			

2.33

closed position upon loss of necessary flow head, and has a leakage area at 15 psig differential pressure of not more than 5.6 square inches. The position of the damper is monitored in the main control room.

Simultaneously with the return of air from the upper compartment to the lower compartment, post severe accident hydrogen mixing capability is provided by the air return fan system in the following regions of the containment: containment dome, each of the four steam generator enclosures, pressurizer enclosure, upper reactor cavity, each of the four accumulator rooms, and the instrument room. These regions are served by separate hydrogen collection headers which terminate on the suction side of each of the two air return fans. A schematic of this system is shown in Figure 9.4-28. The minimum design airflow from each region is sufficient to limit the local concentration of hydrogen to not more than the allowable volume percent range as specified in Section 6.2.5.2. Minimum design flow rates are shown in Figure 9.4-28.

Replace with "12.40"

The header systems are airflow-balanced prior to initial plant operation to assure that the actual airflows are at least equal to the minimum design flow when either or both fans are in operation.

Replace with "4346"

6.8.3 Safety Evaluation

The design bases of the fans are to reduce containment pressure after blowdown from a LOCA or other high energy line break, prevent excessive hydrogen concentrations in pocketed areas, and circulate air through the ice condenser. The containment air return fans turn on 9 ± 1 minute after Phase B containment isolation signal. Peak containment pressure, about 12.86 psig, is attained at approximately 4348 seconds. The fans provide a continuous mixing of containment compartment atmosphere for the long-term post-blowdown environment. Mixing of the compartment atmospheres helps to bring fission products in contact with the ice bed and/or the upper compartment spray for removal from the containment atmosphere. The fans also aid in mixing the containment atmosphere to preclude hydrogen pocketing, which is assumed to be produced as a result of the severe accident.

Each fan located in the lower compartment, when operating alone, transfers 40,000 cfm from the upper compartment into the lower compartment and circulates 1,690 cfm from the enclosed areas in the lower compartment through the hydrogen collector duct headers to prevent excessive localized hydrogen buildup following a DBA. A back-draft damper, normally closed, is located upstream of each deck fan to prevent reverse flow during the initial LOCA or other high energy line blowdown.

The air return fans have sufficient head to overcome the compartment differentials that occur after the reactor coolant system blowdown. The fan head is sufficient to overcome the density effects of steam generation and resistance to airflow through the ice condenser and other system losses. After complete ice bed melt out, each fan has sufficient head to deliver 41,690 cfm with the containment pressurized to the design pressure rating.

The fans are designed to withstand the post DBA environment and were shown to survive the beyond-design-basis accident containment environment (Section 6.2.5).

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
DOCKET NO. 50-391

LIST OF COMMITMENTS

1. The Watts Bar Nuclear Plant Unit 2 FSAR will be updated by January 16, 2015 to address the changes as shown in Enclosure 2 of this letter.

2. The Watts Bar Nuclear Plant Unit 2 Technical Specifications will be updated by January 30, 2015 to reflect the higher ice mass described in Enclosure 2 of this letter.