ATTACHMENT

NINE MILE POINT UNIT TWO

SECONDARY CONTAINMENT DRAWDOWN ANALYSIS

AND

LOCA RADIOLOGICAL ANALYSIS

AUGUST 1988

1.0 SECONDARY CONTAINMENT DRAW-DOWN ANALYSIS

1.1 Introduction

Following a postulated loss-of-coolant accident (LOCA), the secondary containment at Nine Mile Point Unit 2 is maintained at a negative pressure of at least 0.25 inch water gauge with respect to the surrounding outdoor atmosphere by the standby gas treatment system (SGTS) and the secondary containment unit coolers. Maintaining the negative pressure provides an essentially leak-tight barrier such that potential releases of radioactivity are controlled, filtered, and monitored prior to discharge to the environment.

For a relatively short period immediately following a LOCA, the secondary containment may not be maintained at a negative pressure of 0.25 inch water gauge. Due to time delays in actuating the SGTS and unit coolers, the pressure in the secondary containment rises due to the effects of assumed building inleakage and the accumulation of heat within the building. The pressure rises to a peak value and then descends as the SGTS becomes operational to draw air out of the building, and the unit coolers remove heat from the building atmosphere. When the pressure returns to the negative 0.25 inch water gauge or lower, the SGTS pressure control system maintains the negative pressure by controlling the SGTS flow.

1.2 Original Draw-Down Analysis

The original analysis is described in Section 6.2.3.3 of the Unit 2 Final Safety Analysis Report (FSAR). The analysis was performed in accordance with the assumptions recommended in the Standard Review Plan 6.2.3 (NUREG-0800) "Secondary Containment Functional Design."

A. Draw-down Time

The original analysis concludes that the secondary containment pressure rises to a peak value of positive 0.04 inch water gauge at 42 seconds after a LOCA and then decreases to negative 0.25 inch water gauge at 129 seconds. The time necessary to re-establish the negative pressure of 0.25 inch water gauge in the secondary containment following a LOCA is referred to as the draw-down time. Figure 1 shows the pressure response calculated by the original analysis. This figure represents the same information as Figure 6.2-76 of the FSAR.

The LOCA radiological analysis corresponding to the 129-second draw-down time is presented in Section 15.6.5 of the FSAR. The analysis assumes that all activity leaking from the primary containment into the secondary containment during the first 129 seconds will be instantaneously released to the environment without credit for filtration, mixing, or for elevated release through the plant stack.

B. Differential Temperature

The difference between the reactor building air temperature and the service water pump discharge temperature is hereafter referred to as the differential temperature. The analysis assumed the initial design maximum air temperature of 104°F for the reactor building environment and a service water pump discharge header temperature of 81°F. These assumptions formed the basis for the maximum 23°F differential temperature that was included in the analysis for heat removal capability of the unit coolers. However, to establish the most limiting draw-down time, the minimum differential temperature should have been used since it results in a lower heat removal capability for the unit coolers and, thus, a longer draw-down time.

Maintaining the analyzed 23°F differential temperature imposes a limitation on plant conditions. For example, the reactor building is normally maintained at temperatures below 85°F. Deliberate heating of the reactor building is required in the summer during periods of high service water temperature to maintain a 23°F differential temperature. Therefore, the draw-down analysis has been revised to reduce the differential temperature requirement and, thus, minimize the impact on plant operation.

C. Single Failure Assumption

The Standard Review Plan 6.2.3 states that the draw-down analysis should be based on the assumptions of a loss-of-offsite power (LOOP) and a single active failure of the component which produces the most severe design conditions. The original analysis assumed a LOOP and a single active failure of one emergency diesel generator. These assumptions render one division of the unit coolers and one division of the SGTS inoperable for the secondary containment draw-down.

An evaluation of a number of accident scenarios determined that the postulation of no LOOP and loss of the Division II 600 volt bus produces the most limiting draw-down time. The loss of the Division II 600 volt bus renders one division of the unit coolers and one division of the SGTS inoperable, while leaving all divisional emergency core cooling equipment operable and, thus, a source of heat generation. The loss of Division II bus produces a higher ratio of heat load to heat removal capacity than the loss of its equivalent Division I bus. Also, by postulating that offsite power is available, a number of non-essential components remain operable and generate heat. This scenario produces the most limiting draw-down time because, although not consistent with the recommendation in the Standard Review Plan, it results in the most limiting combination of heat loads and heat removal capabilities within the secondary containment.

1.3 Revised Draw-Down Analysis

For the reasons discussed above, the secondary containment draw-down analysis has been revised for Nine Mile Point Unit 2. The revision incorporates a more conservative single failure assumption than the Standard Review Plan and extends the draw-down time to six minutes. The six-minute draw-down time is required to minimize the differential temperature imposed on the plant.

The draw-down time is largely influenced by the difference between the building inleakage rate and the SGTS exhaust flow rate, hereafter referred to as differential flow, and the heat removal rates for given heat loads and environmental conditions. The heat removal rate is directly proportional to the differential temperature between the reactor building air and the service water pump discharge header. Therefore, the differential temperature for a given draw-down time is inversely proportional to the differential flow.

As discussed in Section 1.2, the objective of the revised analysis is to reduce the differential temperature of 23°F to a less restrictive value. An extended draw-down of 6 minutes was selected for this purpose. Differential temperatures were calculated using the 6-minute draw-down time, with the revised assumptions shown on Table 1, and a range of differential flows. Figure 2 shows a curve which has been developed from these calculations to show the relationship between differential temperature and differential flow. For each differential flow shown in Figure 2, the corresponding differential temperature provides assurance that the pressure in the secondary containment can be returned to negative 0.25 inch water gauge within the 6-minute draw-down time.

The as-tested value (see Section 1.4) for SGTS exhaust flow rate is about 3700 CFM and for reactor building inleakage about 1500 CFM. Thus, the as-tested differential flow is about 2200 CFM which corresponds to a differential temperature of about 7.5°F, as indicated in Figure 2.

Table 1 provides a comparison of the assumptions used in the original and revised draw-down analyses. Each of the items is discussed below:

A. The revised analysis assumes an accident scenario which consists of: 1) a design basis LOCA, 2) no loss of offsite power, and 3) a loss of Division II 600 volt bus. As discussed earlier, this accident scenario produces the most limiting design conditions with respect to the draw-down time.

B. The original analysis calculated the 129-second draw-down time based on the given inputs of a 23°F temperature differential and a differential flow of 810 CFM. The revised analysis assumes a 6-minute draw-down time and calculates the differential temperatures required for various differential flow conditions. Utilizing a draw-down time of 6 minutes in lieu of 129 seconds results in a lower differential temperature requirement. The

increase in the draw-down time has been evaluated with respect to its effect on control room and offsite doses. This evaluation is discussed in Section 2.0 of this attachment.

C. The original analysis was based on a simplified single volume model. The revised analysis uses a more realistic multiple volume model to predict the heat load and cooling capacity distribution within the reactor building.

D. With the assumption that offsite power is available, non-essential lights would remain on and produce a considerable heat load. A modification was performed such that the non-essential lights will automatically trip off upon a LOCA event. Therefore, the revised analysis assumes reduced heat loads from non-essential lights in the reactor building.

E. To accelerate the heat removal process, the reactor building unit coolers were modified to automatically initiate upon receipt of a LOCA signal. The revised analysis assumes that the unit coolers are fully functional to remove heat within 60 seconds after a LOCA. The 60-second assumption accounts for time delays for the fan to reach full speed and the service water to reach full flow capacity to the unit coolers.

F. The revised analysis does not include heat loads from the spent fuel pool and its associated cleanup and cooling equipment. Therefore, the revised analysis is valid only for the first plant operating cycle when there is no spent fuel. Section 1.5 of this attachment discusses plans for incorporating spent fuel pool heat loads in the draw-down analysis.

1.4 Differential Temperature Requirements For Plant Operation

A secondary containment draw-down surveillance test, N2-OSP-GTS-R001, was performed on July 17, 1987, and again on August 16, 1987. The test results indicate that the SGTS initiates the draw-down at a maximum exhaust rate of about 3700 CFM and that the building inleakage rate is about 1500 CFM when the reactor building is at a negative pressure of 0.25 inch water gauge. These as-tested values result in a differential flow of about 2200 CFM. From the correlation given in Figure 2, this differential flow requires a minimum differential temperature of about 7.5°F to provide assurance that the 6 minute draw-down time is maintained.

To account for any degradation of the secondary containment pressure boundary that may occur over an 18-month cycle, a ten percent margin for differential flow is included and shown in Figure 2. The 18-month test frequency for the inleakage and SGTS exhaust rate is specified in the Technical Specifications to provide adequate assurance that the differential flow will remain unaffected.

In addition, the effective reactor building inleakage rate increases with decreasing outdoor air temperature. At low outdoor temperature conditions, the colder air outside the reactor building is more dense than the warmer air inside the building. The denser air

increases the differential pressure at lower elevations of the building and, hence, increases the effective inleakage rate of the reactor building. Calculations show that at an extreme outdoor air temperature of -40°F, the building inleakage rate varies from 1500 CFM at the top of the building to about 2900 CFM at the bottom of the building when the reactor building indoor air temperature is maintained at 90°F.

A higher differential temperature is required to account for the increased inleakage rate resulting from low outdoor air temperature. Corrected differential temperatures, as a function of outdoor air temperatures ranging from -40°F to 100°F, have been calculated and are plotted in Figure 3. This curve is based on the current test values of 1500 CFM for building inleakage rate and 3700 CFM for SGTS exhaust rate. A ten percent margin for the differential flow is also included, as previously discussed.

The curve in Figure 3 is currently imposed as a plant operating limit. As described in the Licensee Event Report 87-040, Supplement 1, an administrative procedure has been implemented to verify that the required differential between the reactor building air temperature and the service water pump discharge header temperature is being maintained, whenever the reactor coolant is greater than or equal to 200°F and the reactor is in operational modes 1 (RUN), 2 (STARTUP), or 3 (HOT SHUTDOWN).

As an example, for an extreme outdoor air temperature of -40°F, a differential temperature of about 18°F is required. Under this condition, assuming the reactor building is maintained at the minimum design value of 70°F and service water is at 40°F, a differential temperature of 30°F is available. Therefore, at the low outdoor air temperature, sufficient margin exists between the actual and required differential temperatures.

Again, as an example, during the summer with an expected minimum outdoor air temperature of approximately 50°F, a differential temperature of about 13°F is required. Under this condition, assuming the service water could reach as high as 82°F, heating the reactor building to about 95°F is required to satisfy the differential temperature requirement.

In conclusion, a required differential temperature between the reactor building air temperature and service water pump discharge header temperature has been established as a plant operating condition. The required differential temperature varies from 18°F to 10°F, for outdoor air temperature ranging from -40°F to 100°F. For outdoor air temperatures over 100°F, a minimum differential temperature of 10°F is imposed as a plant operating condition.

1.5 Action Plan For Final Resolution

The revised analysis does not include heat loads from the spent fuel pool, since there is no spent fuel in the pool for the first plant operating cycle. Preliminary results indicate that the heat loads from the spent fuel pool and its associated cleanup and cooling

equipment will add up to 10°F to the existing differential temperature requirement of 10°F to 18°F. The higher differential temperature requirement would impose additional impact on plant operation.

Final resolution is currently in progress to further reduce the differential temperature. The computer code used in the analysis is being upgraded to incorporate the evaporative heat loads from the spent fuel pool. Furthermore, several concepts involving hardware modifications are being evaluated for feasibility. A revised analysis incorporating spent fuel pool heat loads will be implemented prior to the end of the first refueling outage.

2.0 LOCA RADIOLOGICAL ANALYSIS

2.1 Introduction

The loss-of-coolant accident (LOCA) radiological analysis has been revised for Nine Mile Point Unit 2. The original analysis is described in Section 15.6.5 of the FSAR and the calculated results are evaluated against the dose guidelines of 10 CFR 100 for the offsite and 10 CFR 50, Appendix A, General Design Criterion (GDC) 19 for the control room.

The revised analysis utilizes the same assumptions and methodology as the original analysis, with the following exceptions:

1) extended secondary containment draw-down time of 6 minutes,

2) revised, but more conservative, dispersion factors for release from the SGTS building to the control room, and

3) revised assumptions for the bypass leakage calculation:

   - as-built pipe lengths for the main steam, feedwater, drywell purge, and wetwell purge lines,

   - more realistic temperature profiles for the reactor water cleanup and feedwater lines with respect to the bypass leakage, and

   - more realistic fractional leakage rate for the nitrogen addition system line.

Each of these revisions is discussed below.

2.2 Specific Changes

Table 2 summarizes the changes to the LOCA radiological analysis.

Changes in the assumptions for the pipe lengths, temperature profiles, and fractional leakage rate affect the bypass leakage rates and iodine concentration ratios. Table 3 provides a detailed list of input parameters and changes to the LOCA radiological analysis. This table utilizes the format of Table 15.6-13 of the FSAR. As in the original analysis, the revised analysis is based on the conservative isentropic case.

Each change and its estimated effects on dose consequences is discussed below:

1. Secondary Containment Draw-Down Time

   The secondary containment draw-down analysis has been revised and the draw-down time increased from 129 seconds to 6 minutes. The revised radiological analysis assumes that during the first 6 minutes following a LOCA, leakage from the primary containment, traversing incore probes (via the nitrogen purge), and engineered safety feature components are released directly to the environment. No credit is taken for mixing, filtration, or elevated release.

   The Exclusion Area Boundary (EAB) doses are calculated based on activity released during the first two hours following an accident. Since the activity released during the 6 minute draw-down is unfiltered, the increase in the draw-down time contributes significantly to the EAB thyroid dose. The delay times for many bypass leakage releases are long enough such that they do not contribute to the 2-hour EAB doses. The increase in the draw-down time from 129 seconds to 6 minutes (about 200% increase) results in a comparable increase in the EAB thyroid dose from 90.2 to 232.0 rem (about 160% increase).

   The increase in the draw-down time has an insignificant effect on the Low Population Zone (LPZ) and the control room doses. The doses to both the LPZ and control room are calculated based on activities released during the 30-day accident duration. Thus, bypass leakage releases dominate the LPZ and control room doses.

2. Dispersion Factor

   As shown in Table 2, the dispersion factor from the SGTS building to the control room has been changed as a result of finalizing the meteorological calculation. This dispersion factor is used in the radiological analysis to calculate control room doses from activity released from the SGTS building. The release points for bypass leakage paths from the SGTS building are identified in Table 3.

   The dispersion factor is specified as a function of time, up to 30 days from an accident. The dispersion factor is increased by about 22% for the first 24 hours, by about 19% from 24 hours to 96 hours, and by about 16% from 96 hours to 720 hours (30 days).

3. Pipe Length, Temperature Profile, and Fractional Leakage Rate

   Leakage through the main steam, feedwater, reactor water cleanup, drywell purge, wetwell purge, and nitrogen addition system lines is considered bypass leakage and is included in the LOCA radiological analysis. As shown in Table 2, the pipe lengths, temperature profiles, and fractional leakage rates for the bypass leakage calculation have been changed for all these lines.

Pipe Lengths

The pipe lengths for the bypass leakage calculation have been changed to reflect as-built configurations of the main steam, feedwater, drywell purge, and wetwell purge lines. The revised pipe lengths are longer than the values previously used in the calculation. The main steam line length includes the portion of the pipe between the two isolation valves, which was previously not considered. Similarly, the revised feedwater line length now includes the piping between the two isolation check valves and the break exclusion portion of the pipe. The drywell and wetwell purge lines consider the actual line lengths rather than the radial distance between the outboard isolation valves and release points.

Any portion of the pipe which provides an upward flow motion is not included in the pipe length determination in order to compensate for the effect of natural convection. Therefore, the revised pipe lengths are still conservatively determined for the bypass leakage calculation.

The models for predicting the delay time and iodine deposition are discussed in Section 15.6.5 of the FSAR. The delay time is the time it takes the leakage through the isolation valve to fill the volume of piping between the valve and the release point. One-half of the actual calculated delay time is conservatively used in the radiological analysis. Any release of activity to the environment is assumed to occur after the delay time.

The analysis also takes credit for deposition of elemental and particulate iodines on the inside surface of the piping between the isolation valve and the release point. No credit is taken for the deposition of organic iodine. The rate of deposition is an exponential function of the deposition surface area. The iodine deposition is calculated in terms of the ratio of iodine concentration at the release point to the concentration at the isolation valve. The revised delay times and iodine concentration ratios are shown in Table 3 for the affected lines.

The changes in the pipe lengths and the resultant changes in iodine deposition reduce the consequences of the bypass leakages. These changes have an insignificant effect on the EAB doses, but substantially reduce the LPZ and control room thyroid doses.

Temperature Profile

As described in Section 6.2.3.3 of the FSAR, for loss-of-coolant accidents not involving a feedwater line break, sufficient water is contained in the vertical feedwater piping between the containment penetration and the reactor vessel to prevent bypass leakage for at least 30 days after the accident. The original radiological analysis assumed that bypass leakage occurs non-mechanistically through the feedwater line, despite the

existence of sufficient water seals to prevent such leakage. The temperature of the piping was assumed to decrease from 425°F to 120°F over the 30-day accident for the feedwater line.

The revised radiological analysis assumes a realistic but still conservative scenario for the bypass leakages. Realistically, the feedwater line becomes a bypass leakage path only if this line is ruptured inside the primary containment. Immediately following the assumed rupture, rapid depressurization causes the water in the line to flash into steam. The fluid inside the line would be at a saturated condition governed by the drywell pressure. At the peak drywell pressure of 45 pounds per inch, gauge (psig), the corresponding saturation temperature is 293°F.

The pipe metal temperature reaches equilibrium with the fluid temperature in several minutes. Since the delay time for the feedwater line is more than 9 hours, the iodine deposition calculation is not affected by the initial time period when the line temperature is greater than 293°F. An assumption of 293°F is conservative for the scenario for the following reasons:

A. The 293°F represents the saturation temperature of steam at the maximum drywell pressure of 45 psig. No credit is taken for temperature decreases associated with decreasing drywell pressure during the 30-day period.

B. The environmental temperature outside the drywell is significantly lower than 293°F. No credit is taken for the effect of lower temperature outside the drywell.

Likewise, the bypass leakage path for the RWCU line is postulated by ruptures of feedwater lines in the turbine building. The temperature profile for the RWCU bypass leakage is also assumed to be a constant temperature of 293°F.

Therefore, a realistic scenario for bypass leakage in the feedwater and RWCU lines is assumed in the revised radiological analysis. The temperature of the pipe and the fluid within the piping is assumed to be constant at 293°F over the 30-day accident period.

The iodine deposition rate varies exponentially with the temperature of the piping. Therefore, the change in the temperature profile affects the iodine deposition rate. Calculations indicate that the iodine deposition rate reaches its maximum value at the new pipe temperature of 293°F such that all elemental and particulate iodines are essentially deposited on the wall of the piping.

In conclusion, the changes in the temperature profiles result in higher iodine deposition rates, which reduce the consequences of bypass leakage from the feedwater and RWCU lines.

Fractional Leakage Rate

In the original analysis, for conservatism, the leakage rate for the nitrogen addition system line was calculated as a fraction of the wetwell volume per day. As discussed below, the revised leakage rate is more realistically calculated as a fraction of the drywell volume per day.

The nitrogen system bypass leakage path consists of 3 instrument air lines, 2 containment purge lines, and 1 nitrogen addition line. These lines penetrate the primary containment and are joined into one common line outside the containment leading to the nitrogen tank. The pipe length for delay time and iodine deposition is conservatively calculated from the common line to the release point, ignoring the lengths of the individual lines. Thus, the above lines become potential bypass paths.

The leakage from the two containment purge lines is from the wetwell atmosphere; whereas the leakage from the other four lines is from the drywell. Therefore, it is more realistic to base the fractional rate on leakage from the drywell instead of the wetwell. Thus, leakages are assumed to be from the drywell atmosphere. The change results in a slight reduction in the bypass leakage rates for the nitrogen addition system line since the volume of the drywell is larger than the wetwell.

Use of the drywell as the source volume is conservative. The initiating pipe break for the design basis LOCA occurs in the drywell. The activity released from the break will initially be confined to the drywell atmosphere and eventually mix with the wetwell atmosphere. Thus, the activity concentration is expected to be higher in the drywell than the wetwell. Since four of the six lines originate in the drywell, the change to the drywell as a source volume is a reasonable approach. Overall, the change in the basis for the fractional leakage rate slightly reduces the dose consequences.

2.3 Results

The calculated doses from the revised radiological analysis are presented in Table 4. Regulatory limits and the calculated doses from the original analysis are also included in Table 2 for comparison.

In general, the whole body and beta doses for the exclusion area boundary (EAB), the low population zone (LPZ), and the control room increase slightly.

The thyroid dose at the EAB increases from 90.2 to 232.0 rem. This is primarily due to the increase in the draw-down time from 129 seconds to 6 minutes.

The control room and LPZ thyroid doses decrease by about 47% (29.5 to 15.5 rem) and 25% (74.4 to 56.1 rem), respectively. The decrease is primarily due to the effects of the longer pipe lengths, reduced temperature profiles, and revised fractional leakage rate of certain bypass leakage paths.

The revised LOCA radiological consequences remain within the regulatory limits of 10 CFR 100 for the offsite doses and GDC-19 for the control room doses.

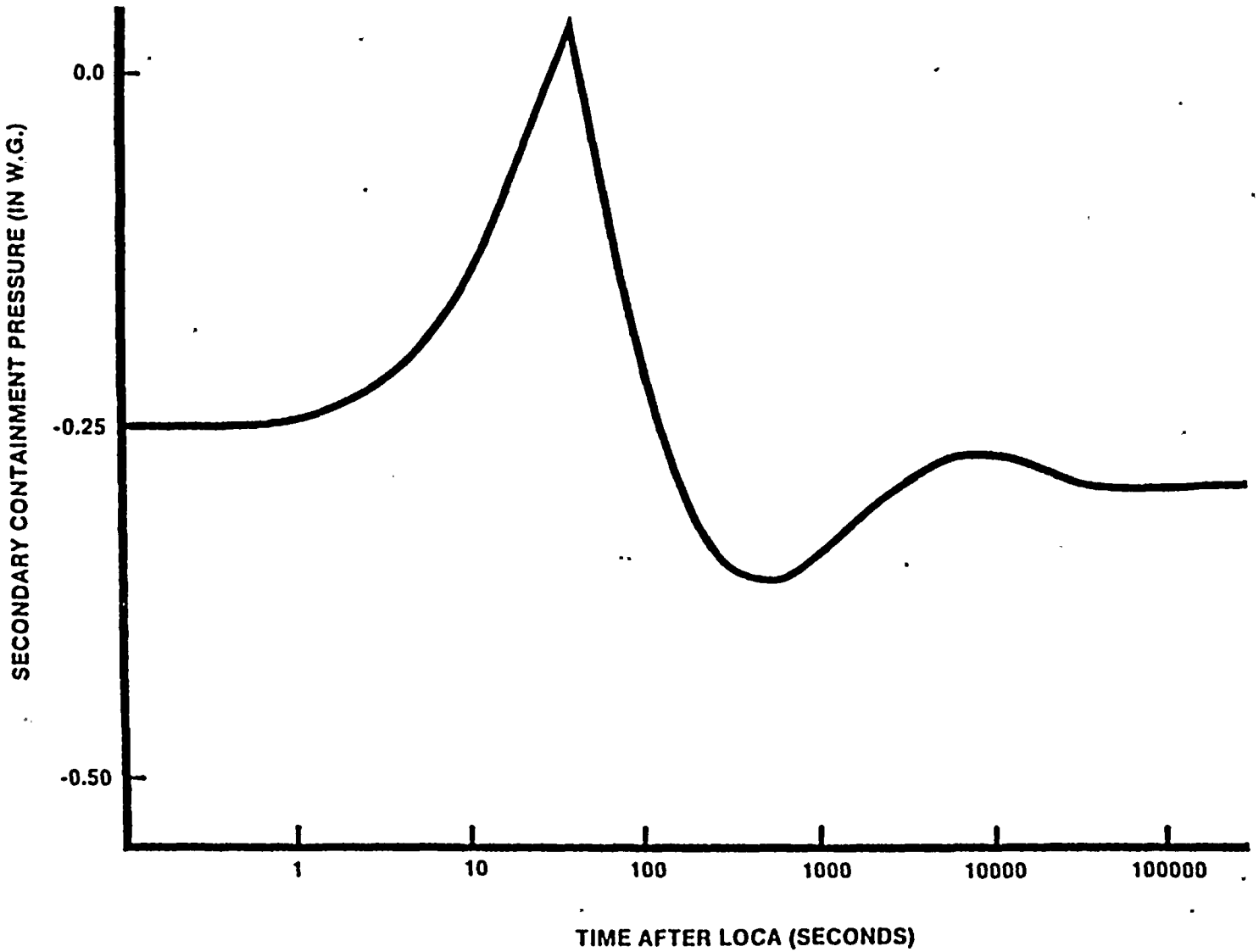

FIGURE 1 - Secondary Containment Pressure Response From the Original Draw-Down Analysis (Excerpt from Figure 6.2-76 of the FSAR).

FIGURE 2 - Differential Temperature as a Function of Differential Flow for a Draw Down Time of 6 Minutes

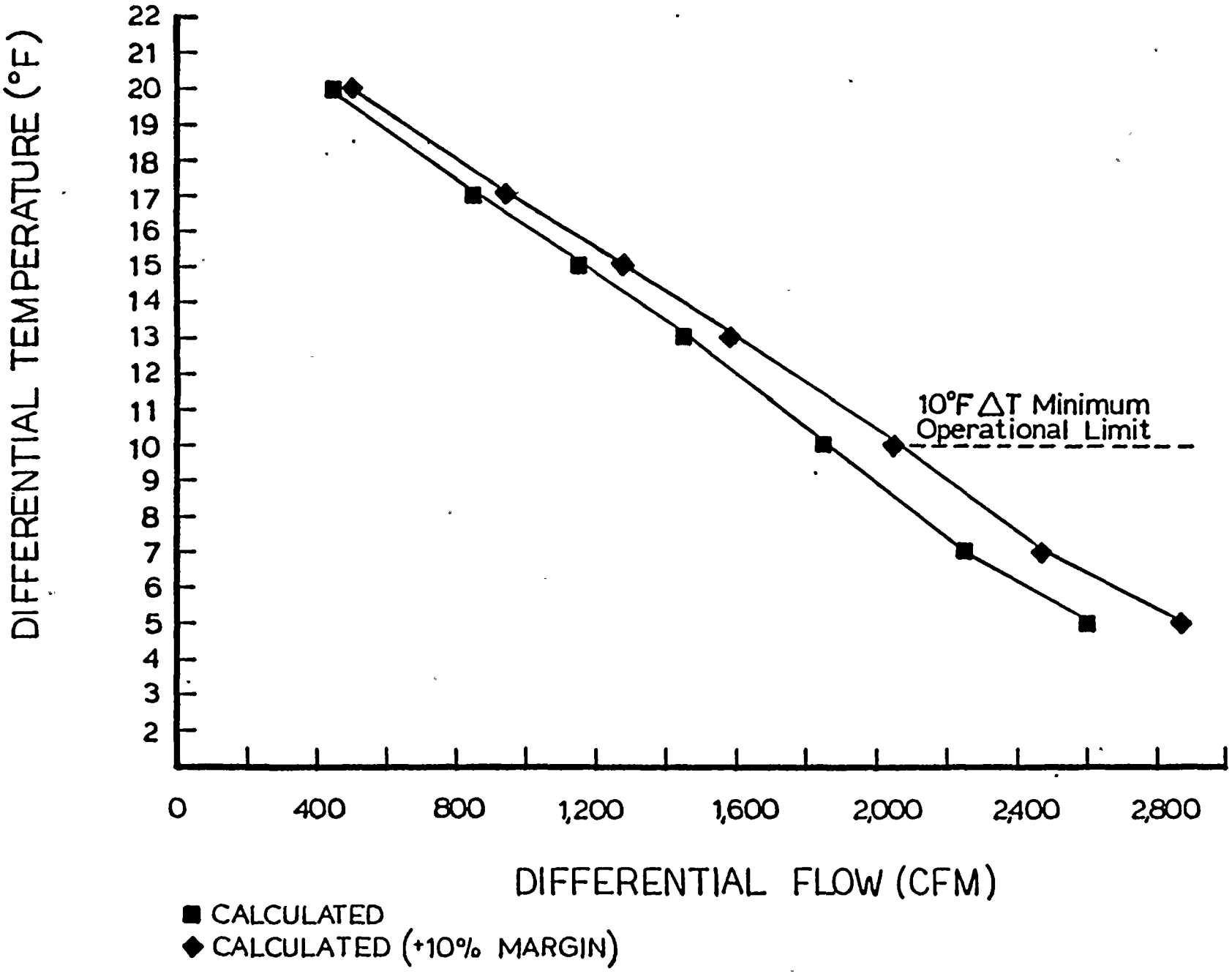

FIGURE 3 - Differential Temperature as a Function of Outdoor Air Temperature for Reactor Building Temperature of 100°F or Lower

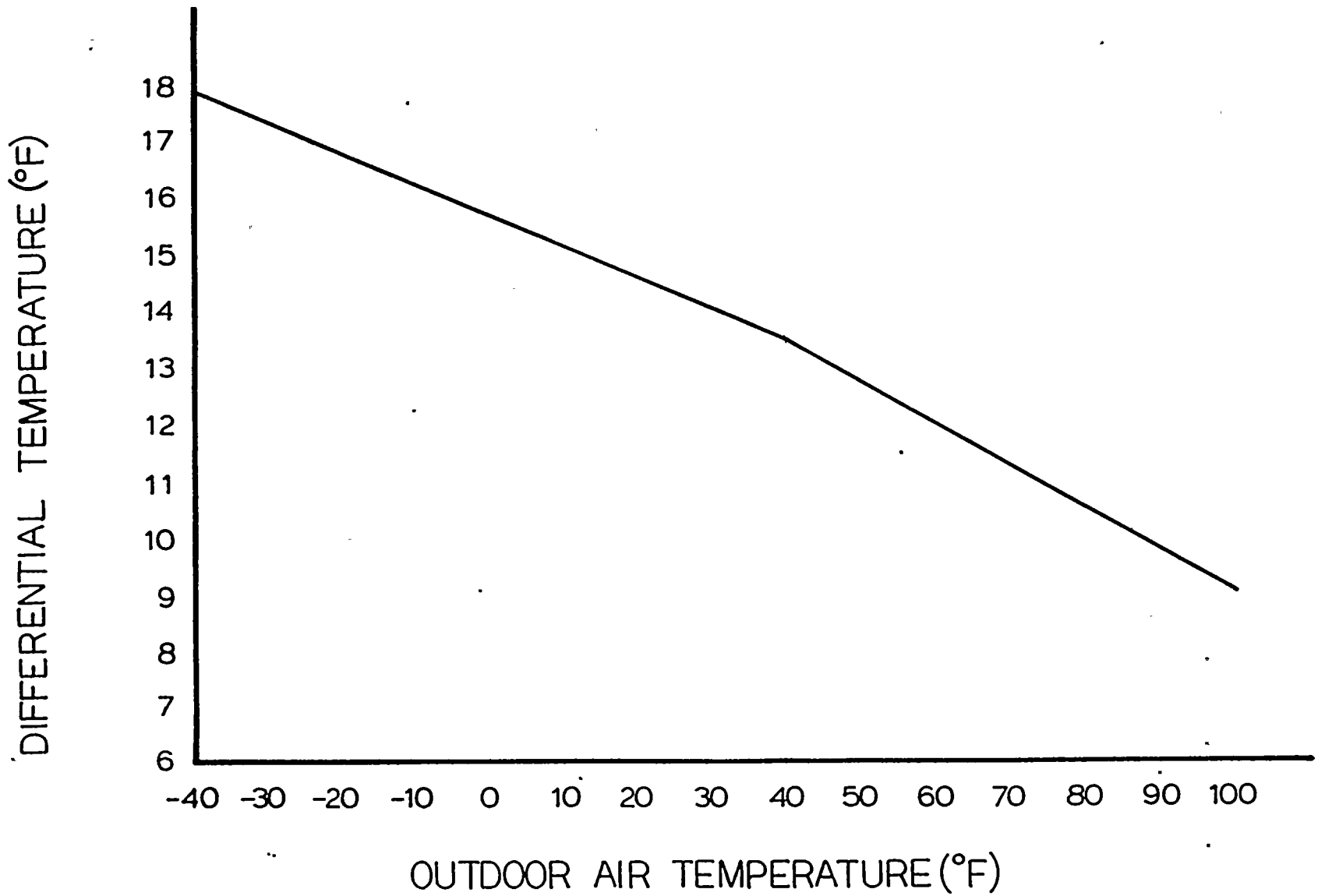

TABLE 1

ASSUMPTIONS AND RESULTS OF THE SECONDARY CONTAINMENT DRAW-DOWN ANALYSIS

	Original Analysis *	Revised Analysis
1. Accident Scenarios:		
Loss of Coolant Accident	Yes	Yes
Loss of Offsite Power	Yes	No
Div. I Diesel Failure	No	No
Div. II Diesel Failure	Yes	No
Div. II 600 Volt Bus Failure	No	Yes
Unit Coolers Available	Div. I only	Div. I + spares
2. Draw-Down Time	129 seconds	6 minutes
3. Computer Modeling	Single Volume	Multiple Volumes
4. Normal Lighting Heatload	No	No
5. Spent Fuel Pool Heatload	Yes	No
6. Unit Cooler Setpoint	85°F	Initiate on LOCA Signal
7. Differential Flow:		
SGTS Exhaust Rate	4000 CFM	Variable (see Figure 2)
Building Inleakage Rate	3190 CFM	Variable (see Figure 2)
Differential Flow	810 CFM	810 CFM to 2200 CFM
8. Differential Temperature		
Initial Reactor Building	104°F	Variable (see Figure 2)
Entering Service Water	81°F	Variable (see Figure 2)
Differential Temperature	23°F	7°F to 20°F

* Section 6.2.3.3 of the FSAR.

TABLE 2

SUMMARY CHANGES IN THE LOCA RADIOLOGICAL ANALYSIS

	Parameters	Original Analysis[1]	Revised Analysis
1.	Secondary Containment Draw-Down Time	2 minutes	6 minutes
2.	Dispersion Factor From SGTS Building to Control Room	0-8 hr, 1.75-3 8-24 hr, 1.34-3 24-96 hr, 4.57-4 96-720 hr, 1.35-4	0-8 hr, 2.15-3 8-24 hr, 1.64-3 24-96 hr, 5.45-4 96-720 hr, 1.56-4
3.	Pipe Length:		
	a) Main Steam Line	312 feet	392 feet
	b) Feedwater Line	50 feet	80 feet
	c) Drywell Purge Inlet Line	32 feet	54 feet
	d) Wetwell Purge Inlet Line	32 feet	42 feet
4.	Temperature Profile:		
	a) Reactor Water Cleanup	551°F to 120°F, Time Dependent	293°F, Constant
	b) Feedwater	425°F to 120°F, Time Dependent	293°F, Constant
5.	Fractional Leakage Rate For Nitrogen Addition System Line	Based on wetwell volume	Based on drywell volume

(1) Section 15.6.5 of the FSAR.

TABLE 3

PARAMETERS TABULATED FOR LOSS-OF-COOLANT RADIOLOGICAL ANALYSIS

	Design Basis Assumptions	
	Original Analysis	Revised Analysis
1. Data and assumptions used to estimate radioactive source from postulated accidents		
a. Power level	3,489 MWt (105%)	same
b. Release of activity to containment air	100% core noble gas inventory 25% core halogen inventory	same
c. Release of activity to suppression pool	50% core halogen inventory	same
d. Iodine fractions		
(1) Organic	0.04	same
(2) Elemental	0.91	
(3) Particulate	0.05	
e. Computer code[1]	Dragon	same
f. Single active failure	Loss of diesel, or failure of one MSIV	same
2. Data and assumptions used to estimate activity released		

TABLE 3 (Cont)

	Design Basis Assumptions			
	Original Analysis		Revised Analysis	
a. Four main steam lines				
Bypass leakage rates per line (fractions of drywell volume per day) (main steam tunnel release)	Two Valves Closed		Two Valves Closed(3)	
	0-2 hr	0.0(4)	0-2 hr	0.0(4)
	0-8 hr	0.0	0-32.34 hr	0.0
	8-24 hr	0.0	32.34-36 hr	1.45-4
	24-24.37 hr	0.0	36-40 hr	1.44-4
	24.37-96 hr	5.48-4	40-44 hr	1.43-4
	96-720	4.12-4	44-48 hr	1.42-4
			48-54 hr	1.40-4
			54-60 hr	1.39-4
			60-66 hr	1.37-4
			66-72 hr	1.35-4
			72-80 hr	1.33-4
			80-88 hr	1.31-4
			88-96 hr	1.28-4
			96-720 hr	1.10-4
			One Valve Closed	
			0-2 hr	0.0(4)
			0-23.56 hr	0.0
			23.56-24 hr	1.90-4
			24-28 hr	1.90-4
			28-32.34 hr	1.89-4
			32.34-36 hr	1.88-4
			36-40 hr	1.87-4
			40-44 hr	1.86-4
			44-48 hr	1.85-4
			48-54 hr	1.84-4
			54-60 hr	1.82-4
			60-66 hr	1.80-4
			66-72 hr	1.78-4
			72-80 hr	1.76-4
			80-88 hr	1.74-4
			88-96 hr	1.71-4
			96-720 hr	1.50-4

TABLE 3 (Cont)

	Design Basis Assumptions			
	Original Analysis		Revised Analysis	
	Two Valves Closed		Two Valves Closed	
Iodine concentration ratios	0-8 hr	0.0	0-32.34 hr	N/A
	8-24 hr	0.0	32.34-36 hr	0.350
	24-24.37 hr	0.0	36-40 hr	0.248
	24.37-96 hr	0.273	40-44 hr	0.196
	96-720 hr	0.04	44-48 hr	0.146
			48-54 hr	0.112
			54-60 hr	0.095
			60-66 hr	0.078
			66-72 hr	0.061
			72-80 hr	0.052
			80-88 hr	0.049
			88-96 hr	0.047
			96-720 hr	0.043
			One Valve Closed	
			0-23.56 hr	N/A
			23.56-24 hr	0.488
			24-28 hr	0.428
			28-32.34 hr	0.369
			32.34-36 hr	0.314
			36-40 hr	0.260
			40-44 hr	0.203
			44-48 hr	0.148
			48-54 hr	0.111
			54-60 hr	0.096
			60-66 hr	0.080
			66-72 hr	0.065
			72-80 hr	0.054
			80-88 hr	0.049
			88-96 hr	0.043
			96-720 hr	0.04
Pipe inside diameter (actual/design basis)	25.23 in/25.23 in		same	

TABLE 3 (Cont)

	Design Basis Assumptions		
	Original Analysis		Revised Analysis
Pipe Length (actual/design basis)	508 ft/312 ft		508 ft/392 ft
Deposition surface (actual/design basis)	3,355 ft^2/2,060 ft^2		3,355 ft^2/2,588 ft^2
Temperature transient - pipe inside surface	0-1 day 450°F 1-2 day 450-350°F 2-3 day 350-250°F 3-4 day 250-120°F 4-30 day 120°		same
b. Inboard main steam drain line(5)			
Bypass leakage rates (fraction of drywell volume per day) (main steam tunnel release)	0-2 hr 0-5.12 hr 5.12-8 hr 8-24 hr 24-96 hr 96-720 hr	0.0(4) 0.0 5.93-5 5.97-5 5.64-5 4.42-5	same
Iodine concentration ratios	0-5.12 hr 5.12-720 hr	0.0 0.04	same
Pipe inside diameter (actual/design basis)	5.761 in/5.761 in		same
Pipe length (actual/design basis)	84.3 ft/84.0 ft		same
Deposition surface (actual/design basis)	127 ft^2/127 ft^2		same
Temperature transient - pipe inside surface	0-720 hr 120°F		same
c. Four post accident sampling lines			
Bypass leakage rates (fraction of drywell volume per day) (radwaste tunnel release)	0-2 hr 0-8 hr 8-24 hr	3.31-5(4) 3.07-5 2.98-5	same

TABLE 3 (Cont)

	Design Basis Assumptions			
	Original Analysis		Revised Analysis	
	24-96 hr	2.82-5	same	
	96-720 hr	2.21-5		
Iodine concentration ratio	All times 0.04		same	
Pipe inside diameter (actual/design basis)	0.18 in/0.18 in		same	
Pipe length (actual/design basis)	935 ft/0 ft		same	
Deposition surface (actual/design basis)	44 ft^2/1.0 ft^2		same	
Temperature transient - pipe inside surfaces	0-720 hr 120°F		same	
d. One feedwater line				
Bypass leakage rates (fraction of drywell volume per day)(7) (main steam tunnel release)	0-2 hr	0.0(4)	0-2 hr	0.0(6)(4)
	0-4.89 hr	0.0	0-9.61 hr	0.0
	4.89-8 hr	2.93-4	9.61-24 hr	2.96-4
	8-24 hr	2.96-4	24-96 hr	2.75-4
	24-96 hr	2.75-4	96-720 hr	2.06-4
	96-720 hr	2.06-4		
Iodine concentration ratio	0-4.89 hr	0.0	0-9.61 hr	N/A
	4.89-8 hr	0.424	9.61-24 hr	0.044
	8-24 hr	0.228	24-720 hr	0.040
	24-96 hr	0.112		
	96-720 hr	0.04		
Pipe inside diameter (actual/design basis)	19.876 in/19.876 in		same	
Pipe length (actual/design basis)	50 ft/50 ft		80 ft/80 ft	
Deposition surface (actual/design basis)	260 ft^2/260 ft^2		416 ft^2/416 ft^2	

TABLE 3 (Cont)

	Design Basis Assumptions			
	Original Analysis		Revised Analysis	
Temperature transient - pipe inside surface	0-24 hr	425-325°F	0-720 hr	293°F
	24-48 hr	325-225°F		
	48-72 hr	225-120°F		
	72-720 hr	120°F		
e. Outboard main steam drain line				
Bypass leakage rates(6) (fraction of drywell volume per day) (main steam tunnel release)	0-2 hr	2.21-5(4)	same	
	0-8 hr	2.05-5		
	8-24 hr	1.99-5		
	24-96 hr	1.88-5		
	96-720 hr	1.47-5		
Iodine concentration ratio	0-2 hr	0.050	same	
	0-8 hr	0.050		
	8-24 hr	0.041		
	24-96 hr	0.040		
	96-720 hr	0.040		
Pipe inside diameter (actual/design basis)	1.687 in/1.687 in		same	
Pipe length (actual/design basis)	17.46 ft/0 ft		same	
Deposition surface (actual/design basis)	7.71 ft^2/1.0 ft^2		same	
Temperature transient - pipe inside surface	0-720 hr	120°F	same	
f. Reactor water cleanup line				
Bypass leakage rates(5) (fraction of drywell volume per day)(7) (main steam tunnel release)	0-2 hr	0.0(4)	0-2 hr	0.0(4)
	0-8 hr	0.0	0-13.09 hr	0.0
	8-10.03 hr	0.0	13.09-24 hr	7.96-5
	10.03-24 hr	7.97-5	24-96 hr	7.51-5
	24-96 hr	7.51-5	96-720 hr	5.89-5
	96-720 hr	5.89-5		

TABLE 3 (Cont)

	Design Basis Assumptions			
	Original Analysis		Revised Analysis	
Iodine concentration ratios	0-8 hr	N/A	0-13.09 hr	N/A
	8-10.03 hr	0.0	13.09-720 hr	0.040
	10.03-24 hr	0.218		
	24-96 hr	0.064		
	96-720 hr	0.04		
Pipe inside (actual/design basis)	7.187 in/6.813 in		same	
Pipe length (actual/design basis)	599 ft/250 ft		same	
Deposition surface (actual/design basis)	1,127 ft^2/446 ft^2		same	
Temperature transient - pipe inside surface	0-24 hr	551-450°F	0-720 hr	293°F
	24-48 hr	450-350°F		
	48-72 hr	350-250°F		
	72-96 hr	250-120°F		
	96-720 hr	120°F		
g. Drywell equipment drain (DER) line				
Bypass leakage rates(5) (fraction of drywell volume per day) (radwaste reactor building vent release)	0-1.24 hr	0.0(4)	same	
	1.24-2 hr	4.25-5(4)		
	0-1.24	0.0		
	1.24-8 hr	4.01-5		
	8-24 hr	3.98-5		
	24-96 hr	3.76-5		
	96-720 hr	2.94-5		
Iodine concentration ratio	0-1.24 hr	N/A	same	
	1.24-720 hr	0.04		
Pipe inside diameter (actual/design basis)	4.026 in/4.026 in		same	
Pipe length (actual/design basis)	75 ft/35 ft		same	

TABLE 3 (Cont)

	Design Basis Assumptions		
	Original Analysis		Revised Analysis
Deposition surface (actual/design basis)	79 ft^2/37 ft^2		same
Temperature transient - pipe inside surface	0-720 hr	120°F	same
h. Drywell equipment drain (DER) vent line			
Bypass leakage rates(5) (fraction of drywell volume per day (radwaste reactor building vent release)	0-0.96 hr	0.0(4)	same
	0.96-2 hr	2.12-5(4)	
	0-0.96 hr	0.0	
	0.96-8 hr	2.01-5	
	8-24 hr	1.99-5	
	24-96 hr	1.88-5	
	96-720 hr	1.47-5	
Iodine concentration	0-0.96 hr	N/A	same
	0.96-720 hr	0.04	
Pipe inside diameter (actual/design basis)	2.067 in/2.067 in		same
Pipe length (actual/design basis)	350 ft/54 ft		same
Deposition surface (actual/design basis)	189 ft^2/29 ft^2		same
Temperature transient - pipe inside surface	0-720 hr	120°F	same
i. Drywell floor drain (DFR) line			
Bypass leakage rates(5) (fraction of drywell volume per day) (radwaste reactor building vent release)	0-2 hr	0.0(4)	same
	0-2.23 hr	0.0	
	2.23-8 hr	5.98-5	
	8-24 hr	5.97-5	
	24-96 hr	5.64-5	
	96-720 hr	4.42-5	

TABLE 3 (Cont)

	Design Basis Assumptions		
	Original Analysis		Revised Analysis
Iodine concentration ratio	0-2.23 hr 2.23-720 hr	N/A 0.04	same
Pipe inside diameter (actual/design basis)	6.06 in/6.065 in		same
Pipe length (actual/design basis)	46.8 ft/37 ft		same
Deposition surface (actual/design basis)	74.25 ft^2/59 ft^2		same
Temperature transient - pipe inside surface	0-720 hr	120°F	same
j. Drywell floor drain (DFR) vent line			
Bypass leakage rates(5) (fraction of drywell volume per day) (radwaste reactor building vent release)	0-1.94 hr 1.94-2 hr 0-1.94 hr 1.94-8 hr 8-24 hr 24-96 hr 96-720 hr	0.0(4) 3.19-5(4) 0.0 2.99-5 2.98-5 2.82-5 2.21-5	same
Iodine concentration ratio	0-1.94 hr 1.94-720 hr	N/A 0.04	same
Pipe inside diameter (actual/design basis)	3.068 in/3.068 in		same
Pipe length (actual/design basis)	114 ft/65 ft		same
Deposition surface (actual/design basis)	91.5 ft^2/52 ft^2		same
Temperature transient - pipe inside surface	0-720 hr	120°F	same

TABLE 3 (Cont)

	Design Basis Assumptions			
	Original Analysis		Revised Analysis	
k. Drywell purge inlet line				
Bypass leakage rates[6] (fraction of drywell volume per day) (SGTS building release)	0-2 hr	0.0[4]	0-2 hr	0.0[4]
	0-6.01 hr	0.0	0-10.58 hr	0.0
	6.01-8 hr	1.07-4	10.58-24 hr	1.08-4
	8-24 hr	1.08-4	24-96 hr	1.00-4
	24-96 hr	1.00-4	96-720 hr	7.51-5
	96-720 hr	7.51-5		
Iodine concentration ratio	0-6.01 hr	N/A	0-10.58 hr	N/A
	6.01-720 hr	0.04	10.58-720 hr	0.040
Pipe inside diameter (actual/design basis)	13.25 in/13.25 in		same	
Pipe lengths (actual/design basis)	45 ft/32 ft		> 54 ft/54 ft	
Deposition surface (actual/design basis)	156 ft^2/111 ft^2		> 187 ft^2/187 ft^2	
Temperature transient - pipe inside surface	0-720 hr	104°F	same	
l. Wetwell purge inlet line				
Bypass leakage rate[6] (fraction of wetwell volume per day) (SGTS building release)	0-2 hr	0.0[4]	0-2 hr	0.0[4]
	0-5.73 hr	0.0	0-7.73 hr	0.0
	5.73-8 hr	5.50-5	7.73-8 hr	5.50-5
	8-24 hr	5.56-5	8-24 hr	5.56-5
	24-96 hr	5.17-5	24-96 hr	5.17-5
	96-720 hr	3.87-5	96-720 hr	3.87-5
Iodine concentration ratio	0-5.73 hr	0.0	0-7.73 hr	N/A
	5.73-720 hr	0.04	7.73-720 hr	0.040
Piping, inside diameter (actual/design basis)	12.0 in/12.0 in		same	

TABLE 3 (Cont)

	Design Basis Assumptions		
	Original Analysis		Revised Analysis
Pipe length (actual/design basis)	129 ft/32 ft		129 ft/42 ft
Deposition surface (actual/design basis)	405 ft^2/100 ft^2		405 ft^2/132 ft^2
Temperature transient - pipe inside surface	0-720 hr	104°F	same
m. Drywell purge makeup line			
Bypass leakage rate(6) (fraction of drywell volume per day) (SGTS building release)	0-0.83 hr	0.0(4)	same
	0.83-2 hr	1.64-5(4)	
	0-0.83 hr	0.00	
	0.83-8 hr	1.55-5	
	8-24 hr	1.54-5	
	24-96 hr	1.43-5	
	96-720 hr	1.07-5	
Iodine concentration ratio	0-0.83 hr	0.0	same
	0.83-720 hr	0.04	
Pipe inside diameter (actual/design basis)	1.939 in/1.939 in		same
Pipe lengths (actual/design basis)	129 ft/41 ft		same
Deposition surface (actual/design basis)	65.5 ft^2/20.8 ft^2		same
Temperature transient - pipe inside surface	0-720 hr	104°F	same
n. Wetwell purge makeup line			
Bypass leakage rates(6) (fraction of wetwell volume per day) (SGTS building release)	0-0.83 hr	0.0(4)	same
	0.83-2.0 hr	1.02-5(4)	
	0-0.83	0.0	
	0.83-8 hr	9.35-6	
	8-24 hr	9.26-6	

TABLE 3 (Cont)

	Design Basis Assumptions		
	Original Analysis		Revised Analysis
	24-96 hr	8.61-6	
	96-720 hr	6.44-6	
Iodine concentration ratio	0-0.83 hr	0	same
	0.83-720 hr	0.04	
Pipe inside diameter (actual/design basis)	1.939 in/1.939 in		same
Pipe length (actual/design basis)	104 ft/41 ft		same
Deposition surface (actual/design basis)	52.8 ft^2/20.8 ft^2		same
Temperature transient - pipe inside surface	0-720 hr	104°F	same
o. Drywell and wetwell purge makeup lines			
Bypass leakage rates(6) (fraction of primary containment per day) (SGTS building release)	0-2 hr	0.0(4)	same
	0-3 hr	0.0	
	3-13.6 hr	1.58-5	
	13.6-720 hr	0.0	
Iodine Concentration Ratio	0-3 hr	N/A	same
	3-13.6 hr	0.040	
	13.6-720 hr	N/A	
Pipe inside diameter (actual/design basis)	1.939 in/1.939 in		same
Pipe lengths (actual/design basis)	104 ft/41 ft		same
Deposition surface (actual/design basis)	52.8 ft^2/20.8 ft^2		same
Temperature transient-pipe inside surface	0-720 hr	104°F	same
p. 3 Instrument air lines 2 CPS lines 1 GSN line (total for 6 lines)			

TABLE 3 (Cont)

	Design Basis Assumptions			
	Original Analysis		Revised Analysis	
Bypass leakage rate[5] (SGTS building release)	0-1 hr	0.0	0-1 hr	0.0[4]
	1-8 hr	1.71-4	1-2 hr	1.16-4[4]
	8-24 hr	1.70-4	0-1 hr	0.0
	24-96 hr	1.60-4	1-8 hr	1.16-4
	96-720 hr	1.26-4	8-24 hr	1.15-4
	(Based on fraction of wetwell volume per day)		24-96 hr	1.08-4
			96-720 hr	8.48-5
			(Based on fraction of drywell volume per day)	
Iodine concentration factor	0-1 hr	0.0	same	
	1-720 hr	0.04		
Pipe inside diameter (actual/design basis)	2.469"/2.469"		same	
Pipe lengths (actual/design basis)	233'/233'			
Deposition surface (actual/design basis)	144 ft^2/144 ft^2		same	
Temperature transient - pipe inside surface	104°F		120°F	
q. 2 Instrument air lines to ADS accumulator (total for 2 lines)				
Bypass leakage rate[5] (fraction of primary containment volume per day) (SGTS building release)	0-0.64 hr	0.0[4]	same	
	0.64-2 hr	6.04-5[4]		
	0-0.64 hr	0.0		
	0.64-8 hr	6.06-5		
	8-24 hr	5.96-5		
	24-96 hr	5.64-5		
	96-720 hr	4.42-5		
Iodine concentration factor	0-0.64 hr	N/A	same	
	0.64-720 hr	0.04		

TABLE 3 (Cont)

		Design Basis Assumptions		
		Original Analysis		Revised Analysis
		Bypass Path Pen. Z53A	Bypass Path Pen. Z53B	
	Pipe inside diameter (actual/ design basis)	1.049"/1.049"	1.049"/1.049"	same
	Pipe lengths (actual/ design basis)	225'/225'	219'/219'	same
	Deposition surface (actual/ design basis	62 ft^2/62 ft^2	60 ft^2/60 ft^2	same
	Temperature transient - pipe inside surface	120°F	120°F	same
r.	Containment leakage rate (main stack release)	1.1% per day of primary containment volume for duration of accident		same
s.	Traversing incore probe leakage rate	0.21% per day of primary containment volume for duration of accident (main stack release from t = 129s to t = 720 hr) (radwaste/reactor building vent release from t = 0 to t = 129s)		0.21% per day of primary containment volume for duration of accident (main stack release from t = 6 minutes to t = 720 hr) (Radwaste/reactor building vent release from t = 0 to t = 6 minutes)
t.	Reactor building leak rate (main stack release)	3,500 cfm through standby gas treatment (SGTS)		same
u.	Percentage mixing in reactor building air	50%		same
v.	Reactor building pressurization time (radwaste/reactor building vent release)	129 sec		6 minutes
w.	SGTS halogen filtration efficiency	99%		same

TABLE 3 (Cont)

	Design Basis Assumptions	
	Original Analysis	Revised Analysis
x. ESF leakage to reactor building (main stack release)		
(1) Leak rate	1 gpm	same
(2) Iodine partition factor (air/water)	0.1	
3. All other pertinent data		
a. Primary containment		
(1) Drywell free air volume	2.85+5 ft^3	same
(2) Primary containment free air volume	4.73+5 ft^3	
(3) Suppression pool volume	1.45+5 ft^3	
b. Reactor building		
(1) Free air volume	3.88+6 ft^3	same
c. Control room		
(1) Free air volume	3.81+5 ft^3	same
(2) Intake rate	1.00+3 cfm	
(3) Recirculation rate	7.50+2 cfm	
(4) Intake/recirculation halogen filtration efficiency	99%	
4. Dispersion data (s/m^3)		
a. Stack		
0-2 hr EAB	2.97-5	same
0-8 hr LPZ	1.03-5	
8-24 hr LPZ	8.85-7	

TABLE 3 (Cont)

	Design Basis Assumptions	
	Original Analysis	Revised Analysis
24-96 hr LPZ	3.66-7	
96-720 hr LPZ	1.03-7	
0-8 hr control room	8.10-5	
8-24 hr control room	2.44-8	
24-96 hr control room	2.10-8	
96-720 hr control room	1.69-8	
b. Radwaste/reactor building vent[8]		
0-2 hr EAB	1.90-4	same
0-8 hr LPZ	1.78-5	
8-24 hr LPZ	1.19-5	
24-96 hr LPZ	4.93-6	
96-720 hr LPZ	1.40-6	
0-8 hr control room	2.13-4	
8-24 hr control room	1.66-4	
24-96 hr control room	9.88-5	
96-720 hr control room	4.70-5	
c. Main steam tunnel		
0-2 hr EAB	1.90-4	same
0-8 hr LPZ	1.78-5	
8-24 hr LPZ	1.19-5	same
24-96 hr LPZ	4.93-6	
96-720 hr LPZ	1.40-6	
0-8 hr control room	1.29-3	
8-24 hr control room	9.90-4	
24-96 hr control room	3.37-4	
96-720 hr control room	9.92-5	

TABLE 3 (Cont)

	Design Basis Assumptions	
	Original Analysis	Revised Analysis
d. Radwaste tunnel (PASS area)		
0-2 hr EAB	1.90-4	same
0-8 hr LPZ	1.82-5	
8-24 hr LPZ	1.21-5	
24-96 hr LPZ	5.02-6	
96-720 hr LPZ	1.46-6	
0-8 hr control room	1.83-4	
8-24 hr control room	1.41-4	
24-96 hr control room	4.81-5	
96-720 hr control room	1.42-5	
e. SGTS building		
0-2 hr EAB	1.90-4	1.90-4
0-8 hr LPZ	1.78-5	1.78-5
8-24 hr LPZ	1.19-5	1.19-5
24-96 hr LPZ	4.93-6	4.93-6
96-720 hr LPZ	1.40-6	1.40-6
0-8 hr control room	1.75-3	2.15-3
8-24 hr control room	1.34-3	1.64-3
24-96 hr control room	4.57-4	5.45-4
96-720 hr control room	1.35-4	1.56-4

NOTE: 5.92-4 = 5.92×10^{-4}

(1) Dragon Code, Dose and Radioactivity from Nuclear Facility Gaseous Outflows, NU-115.

(2) Both the loss of diesel and failure of one MSIV scenarios for single active failure are analyzed. The scenario that results in the highest calculated dose is used at each given receptor location (EAB, LPZ, and control room).

(3) The bypass leakage rates are specified on a per line basis. For the loss of diesel scenario, the two-valves-closed leak rates apply to each of the four main steam lines. For the MSIV failure scenario, the two-valves-closed leak rates apply to three of the main steam lines, and the one-valve-closed leak rates apply to one main steam line.

(4) EAB dose analysis only.

(5) The loss of diesel scenario leakage rates (only 1 isolation valve closed) are shown. The leak rate for the MSIV failure scenario (2 isolation valves closed) is lower.

(6) The leak rates for the loss of diesel and MSIV failure scenario are the same for these lines.

(7) See Section 6.2.3.

(8) These values are also used for the direct release to the environment during the period when reactor building pressure is above (-)0.25 inch water gage.

TABLE 4

LOSS-OF-COOLANT (DESIGN BASIS ANALYSIS)

RADIOLOGICAL CONSEQUENCES

	WHOLE BODY DOSE (rem)	THYROID DOSE (rem)	BETA DOSE (rem)
Original Analysis (FSAR Section 15.6.5)			
Exclusion Area (2 hr)	5.0	90.2	3.6
Low-Population Zone (30 day)	2.5	74.4	1.9
Control Room (30 day)	1.6	29.5	21.8
Revised Analysis			
Exclusion Area (2 hr)	6.3	232.0	4.2
Low-Population Zone (30 day)	2.6	56.1	1.9
Control Room (30 day)	1.6	15.5	22.9
Regulatory Limits			
10 CFR 100 - Exclusion Area (2 hr)	25	300	*
10 CFR 100 - Low Population Zone (30 day)	25	300	*
GDC 19 - Control Room (30 day)	5	30	30

* No limit specified.