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SUBJECT: Forwards updated info re secondary containment bypass leakage discussed during 850615 meeting w/NRC.Info intended to assist in closing of SER Open Item 6.Info will be incorporated into Amend 21 to FSAR.

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July 26, 1985
(NMP2L 0453)

Mr. Walter Butler, Chief
Licensing Branch No. 2
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Butler:

Re: Nine Mile Point Unit 2
Docket No. 50-410

Enclosed please find updated information regarding secondary containment bypass leakage. This information was discussed during a meeting with your staff on June 15, 1985. This submittal is intended to assist in closing SER Open Item #6. The information will be incorporated into Amendment 21 of the FSAR.

Very truly yours,

CVMangan

C. V. Mangan
Vice President
Nuclear Engineering & Licensing

JM:mbg

xc: R. A. Gramm, NRC Resident Inspector
Project File (2)

B001
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Railcar entrance to the reactor building railroad airlock is through an interlocking double door airlock system. The railroad airlock is completely within and along the northeast side of the reactor building at el 261 ft. One of the interlocked doors is the exterior railcar door at the north end of the railroad airlock, and the other is the interior railcar door at the south end of the railroad airlock. A smaller door for personnel ingress and egress is incorporated into the design of the interior railcar door. All three doors must be closed before any one of them can be opened.

The reactor building pressure control function automatically maintains a uniform subatmospheric pressure of 0.25 in W.G. by monitoring the differential pressure between the reactor building interior and the external atmospheric pressure. The differential pressure is monitored by a differential pressure transmitter. The signal that indicates the differential pressure also controls the position of the recirculation dampers in the HVRS supply fan units. In the event of reactor building isolation, the reactor building pressure control instrumentation regulates the reactor building pressure by controlling the SGTS recirculation flow.

The reactor building pressure control instrumentation is designed to eliminate fluctuations in reactor building pressure caused by such factors as wind gusts. Reactor building pressure is indicated and recorded and loss of negative pressure is alarmed in the main control room.

6.2.3.2.3 Bypass Leakage Paths

Table 6.2-56 presents a tabulation of all primary containment process piping penetrations including the potential reactor building bypass leakage paths. The potential bypass leakage paths are routed through the reactor building and terminate in the radwaste, standby gas
21 treatment, or turbine generator buildings. No guard pipes are used on penetrations and therefore guard pipes cannot constitute a bypass leakage path. All process lines that rely on a closed system within the primary containment as a leakage boundary terminate within the reactor building; therefore, these lines are not considered potential bypass leakage paths.

Bypass leakage is included in the radiological evaluation of design basis events. This is discussed in Section 15.6.5.5. Tables 6.2-55a and 6.2.55b show the bypass leakage paths considered. They include four main steam lines, two main

steam drain lines, one reactor water cleanup line, one
feedwater line, four post-accident sampling lines, four
primary containment purge lines, and four drywell floor and 21
equipment vent and drain lines.

The analysis used to predict the bypass leakage rates is 21
discussed in Section 6.2.3.2.4. Single failure is included
in the analysis in that failure of one division of
electrical power is assumed. This results in all motor-
operated containment isolation valves on that division
failing as is (assumed open), thereby reducing the
restrictions to bypass leakage. This is the worst single
failure to consider for this evaluation.

All leakage is conservatively assumed to be across isolation
valve seats and to remain within the system piping until
released to the environment. Any leakage escaping across
outboard isolation valve stem packing would be released to
the secondary containment or main steam tunnel. Any leakage
into the secondary containment would be processed by the 21
standby gas treatment system. Contaminants leaked into the
main steam tunnel will be transported to the environment
more slowly due to the much larger cross-sectional area of
the tunnel and the resulting slower average velocities.

No credit is taken for a reduction in bypass leakage due to
water inboard of or trapped between isolation valves. The
isolation valves are assumed to leak containment atmosphere
instantaneously following the accident. No credit is taken
for the time required to initially pressurize the volume
between the isolation valves.

Leakage transport time to the environment is based on 1/2 of
the available horizontal and vertically downward flow piping 21
located between the outboard isolation valve and the
environment.

Further conservatism is added to the analysis by the
assumption that all isolation valves in these paths, except
the main steam isolation valves (MSIV)and feedwater check 21
valves, leak at a rate equal to the maximum permissible,
ASME Section XI, Subsection IWV-3426, recommended acceptance
level of 7.5 scf/day per inch of nominal valve diameter at
functional pressure. The MSIVs are assumed to leak at
6 SCFH, nearly three times the valve design limit. Leakage
across check valves, except the feedwater check valves, is 21
assumed to be at twice the recommended rate of 7.5 scf/day
per inch of nominal valve diameter, as provided for by ASME
Section XI, Subsection IWV-3426. Leakage across the
feedwater valves is assumed to be 12 scfh. 21

21 Several process lines are excluded from consideration as bypass leakage paths due to nitrogen seals inherent in the system design and installation arrangement. These are discussed below.

21 Nitrogen Seals

Some systems rely on pressurized nitrogen contained within the system to eliminate bypass leakage. These nitrogen seals consist of pressurized nitrogen accumulator tanks which apply backpressure to containment isolation valves. A typical nitrogen seal is shown on Figure 6.2-88. As the pressurized accumulators are initially at a pressure greater than the maximum postulated containment pressure, all leakage is considered to be into the primary containment as long as the accumulator tank is at a pressure greater than that of the primary containment.

The process lines that eliminate bypass leakage by the use of nitrogen or water seals are discussed below and include condensate makeup and drawoff (CNS), reactor core isolation
21 cooling (RCIC), high pressure core spray (HPCS), feedwater (FWS), and instrument air systems (IAS).

CNS

While not directly connected to the primary containment, the condensate makeup and drawoff system is used as the alternate fill source to the RHR, HPCS, LPCS, and RWCU systems. Each condensate fill connection to these systems is isolated by means of a normally closed globe valve. The main supply line into the secondary containment contains a check valve at the low point which, in case of a pipe break outside the containment, is sealed by a 70-ft leg of water. Although the condensate makeup and drawoff system is not of seismic design, any line break within the reactor building would provide a preferential flow path, for containment atmosphere leakage, into the reactor building atmosphere. Under this condition gaseous leakage would be collected by the SGTS and thus not be classified as bypass leakage.

21

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RCIC

The RCIC path from the primary containment to the condensate storage building is protected from bypass leakage. When RCIC is taking suction from the condensate storage tank (2CNS-TK1A), the tank static head pressure maintains a 23-psig water seal at valve 2ICS*V28 and/or 2ICS*MOV136 (Figure 6.2-81). Also, the piping arrangement as shown in Figure 6.2-81 provides a loop seal with a high point at 2ICS*MOV136. Thus, any containment atmosphere leakage through this valve during the period that containment pressures exceed water seal pressure would be trapped at this high point. If a LOCA and an SSE take place simultaneously and a condensate line break occurs, 2ICS*MOV129 on the condensate tank line will shut automatically, creating an additional barrier to bypass leakage.

HPCS

The arrangement of the HPCS suction line from condensate storage tank 2CNS-TK1B provides enough static head pressure to keep a 75-ft (32 psig) water seal at the line low point (valve 2CSH*MOV101) in Figure 6.2-83. Further, the piping arrangement as shown in Figure 6.2-83 provides two intermediate loop seals with high points at valves 2CSH*MOV118 and 2CSH*V59, ensuring that any containment atmosphere leakage occurring during the 20 min that containment pressures exceed water seal pressure would be trapped between these high points. If a LOCA and an SSE take place simultaneously and a condensate line break occurs, 2CSH*MOV101 on the condensate storage tank line will shut automatically, creating an additional barrier against bypass leakage.

FWS

For loss-of-coolant accidents not involving a feedwater line break, sufficient water exists 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. See Figure 6.2-84.

For a break in feedwater piping inside containment, bypass leakage through this piping is included in the analysis of Section 15.6. However, as discussed below, a water seal, restored after the break, will effectively prevent escape of containment atmosphere to the environment after 10 min. 21

In considering a break in the feedwater piping within the
21 primary containment, credit can be given to the piping
arrangement which provides low stress levels along with pipe
whip restraints. Consequently, it can be stated that the
containment penetration is a break exclusion area. Assuming
a break in the feedwater line at the end of the break
exclusion region inside the primary containment (see
Section 3.6A and Figure 3.6A-20), sufficient water will
remain in the line, even after flashing due to initial
depressurization, to maintain a vertical water seal on the
feedwater isolation valves (Figure 6.2-84). Water losses
due to long-term containment pressure reduction and the
associated water vaporization and the backleakage through
the two isolation check valves for 30 days will be
replenished by reactor water leaking from the break. Within
10 min after the break the ECCS injection water will reflood
the reactor to above the level of the feedwater sparger. At
that point water would flood back into the feedwater piping
and then into the intact containment penetration piping
(Figure 6.2-84). This would more than make up for any
losses due to leakage out the containment isolation valves.
Thus a continuous water seal is provided to prevent any
bypass leakage through the feedwater lines after the initial
21 10-min refilling period. Notwithstanding the above, bypass
leakage through a ruptured feedwater line is included in the
21 radiological analysis for the entire 30-day period to ensure
conservative analysis results.

In addition to the two isolation check valves, each feedwater line has a remote-manual gate valve outboard of the isolation valves that may be shut subsequent to a LOCA anytime the operators determine that feedwater flow is unnecessary or unavailable. The gate valve provides further back leakage control. However, this valve is assumed to remain open for the purpose of evaluating bypass leakage.

<u>IAS</u>

Instrument air system lines, serving pneumatically operated valves inside the primary containment, are supplied with nitrogen from the nitrogen inerting system while the plant is operating (Figures 6.2-85 and 6.2-86). Closed outboard penetration isolation valves are maintained pressurized (nitrogen seal) by the nitrogen inerting system at pressures exceeding the maximum containment pressure. If nitrogen supply to the reactor building is lost, pressure on the isolation valves would be maintained by accumulator tanks which would remain pressurized. In the long term, accumulator tanks serving the ADS accumulators and valves may require repressurization due to ADS valve actuation

losses and leakage. Post-accident makeup N_2 will be provided by two bottled nitrogen connections located outside the reactor building (see Section 1.10, Item II.K.3.28).

6.2.3.2.4 Bypass Leakage Rates

Bypass leakage rates as a function of time after the postulated LOCA are predicted for each path by two methods, assuming isothermal flow and isentropic flow. Table 6.2-55a lists the bypass paths considered and their contributions 21
to the total bypass leakage, assuming isothermal flow determined with the following equation:

$$\dot{m} = K \left\{ (P_u^2 - P_D^2)/RT_u \right\}^{1/2} \qquad (6.2\text{-}12)$$

Where:

P_u = Upstream absolute pressure (post-LOCA pressure/temperature profile per Section 6.2.1)

P_D = Downstream absolute pressure

T_u = Upstream absolute temperature

R = Gas constant

K = Constant (determined from the technical specification of allowable leak rate) 21

$\dot{m}$ = Mass flow rate

To quantify the sensitivity of the bypass leakage analysis to the flow model assumption, the bypass calculation was repeated considering the leakage flow to be characterized as isentropic flow through an orifice. Table 6.2-55b summarizes the isentropic flow results determined with the following equation:

$$\dot{m} = A \left\{ 2\, g_c \left(\frac{\gamma}{\gamma-1}\right) \left(\frac{P^2{}_u}{RT_u}\right) \left(\frac{P_D}{P_u}\right)^{\frac{2}{\gamma}} \left[1-\left(\frac{P_D}{P_u}\right)^{\frac{\gamma-1}{\gamma}}\right] \right\}^{1/2} \qquad (6.2\text{-}13)$$

Where:

P_u = Upstream absolute pressure (post-LOCA pressure/temperature profile per Section 6.2.1)

P_D = Downstream absolute pressure

T_u = Upstream absolute temperature

R = Gas constant

γ = Specific heat ratio

g_c = Conversion constant

A = Orifice flow area (to be determined from the
21 technical specification of allowable leak rate)

$\dot{m}$ = Mass flow rate

The isentropic flow is generally 20 to 35 percent higher than the isothermal flow.

In each case the fractional flow rate is evaluated using the following equation:

$$f = \frac{\dot{m}}{\rho V} \qquad (6.2\text{-}14)$$

Where:

$\dot{m}$ = Mass flow rate

ρ = Density of containment air and steam mixture (P/RT)

V = Containment volume

f = Fractional flow rate

The containment bypass leak rate for various paths is calculated based on two closed valves in series or one closed and one open valve depending upon the direct consequence of the postulated failure of one emergency diesel generator.

The failure of one diesel generator, combined with a loss of offsite power, would be the worst single failure.

6.2.3.2.5 Iodine Plateout Considerations

The radiological consequences arising from bypass leakage are provided in Section 15.6.5. The analyses include credits for elemental iodine deposition on the walls of the piping between the outboard isolation valve and the release point. Details of the iodine deposition analysis can be found in Section 15.6.5.5.3.

6.2.3.2.6 Activity Transport Delay Considerations

The leakage of activity from the primary containment to the environment is through a portion of piping downstream of the outer isolation valve. Because of the very low leakage rates, there is a considerable transport delay time between the outer isolation valve and the release point. Therefore, the analyses include the credit of the delay time in the dose calculations. This is further explained in Section 15.6.5.5.3. 21

TABLE 6.2-55a 21

EVALUATION OF POTENTIAL BYPASS LEAKAGE PATHS (ISOTHERMAL FLOW MODEL) 21

Line Description	Termination Region	Bypass Leakage Barrier	Leak Rate(3) Tech Spec SCFH(1,4)	Leak Rate(3) Fraction/Day(2)	Containment Bypass Leak Rate (Fraction/Day)(5) 0-2 hr	0-8 hr	8-24 hr	1-4 day	4-30 day	
4 Main steam lines	Turbine Bldg	2-21" valves in each line	6	0.139×10^{-3}	0.540×10^{-3}	0.492×10^{-3}	0.478×10^{-3}	0.432×10^{-3}	0.305×10^{-3}	21
Main steam drain line (inboard)	Turbine Bldg	1-6" valve	1.875	0.435×10^{-4}	0.633×10^{-4}	0.572×10^{-4}	0.555×10^{-4}	0.497×10^{-4}	0.344×10^{-4}	21
Main steam drain line (outboard)	Turbine Bldg	1-2" valve	0.625	0.145×10^{-4}	0.211×10^{-4}	0.191×10^{-4}	0.185×10^{-4}	0.166×10^{-4}	0.115×10^{-4}	21
4 Post accident sampling lines	Radwaste Tunnel	1-3/4" valve in each line	0.2344	0.543×10^{-5}	0.317×10^{-4}	0.286×10^{-4}	0.277×10^{-4}	0.249×10^{-4}	0.172×10^{-4}	21
Drywell equipment drain line	Radwaste Tunnel	1-4" valve	1.25	0.290×10^{-4}	0.422×10^{-4}	0.381×10^{-4}	0.370×10^{-4}	0.331×10^{-4}	0.229×10^{-4}	21
Drywell equipment vent line	Radwaste Tunnel	1-2" valve	0.625	0.145×10^{-4}	0.211×10^{-4}	0.191×10^{-4}	0.185×10^{-4}	0.166×10^{-4}	0.115×10^{-4}	21
Drywell floor drain line	Radwaste Tunnel	1-6" valve	1.875	0.435×10^{-4}	0.633×10^{-4}	0.572×10^{-4}	0.555×10^{-4}	0.497×10^{-4}	0.344×10^{-4}	21
Drywell floor vent line	Radwaste Tunnel	1-3" valve	0.9375	0.217×10^{-4}	0.317×10^{-4}	0.286×10^{-4}	0.277×10^{-4}	0.249×10^{-4}	0.172×10^{-4}	21
RWCU line	Turbine Bldg	1-8" valve	2.5	0.579×10^{-4}	0.845×10^{-4}	0.763×10^{-4}	0.739×10^{-4}	0.663×10^{-4}	0.459×10^{-4}	21
Feedwater line	Turbine Bldg	2-24" check valves	12	0.278×10^{-3}	0.270×10^{-3}	0.246×10^{-3}	0.239×10^{-3}	0.216×10^{-3}	0.153×10^{-3}	21
CPS supply line to drywell	Standby Gas Trtmt Area	2-14" valves	4.38	0.102×10^{-3}	0.985×10^{-4}	0.898×10^{-4}	0.873×10^{-4}	0.789×10^{-4}	0.557×10^{-4}	21
CPS supply line to drywell	Standby Gas Trtmt Area	2-2" valves	0.625	0.145×10^{-4}	0.141×10^{-4}	0.128×10^{-4}	0.125×10^{-4}	0.113×10^{-4}	0.795×10^{-5}	21

TABLE 6.2-55a (Cont) 21

Line Description	Termination Region	Bypass Leakage Barrier	Leak Rate(3) Tech Spec SCFH(1,4)	Fraction/ Day(2)	Containment Bypass Leak Rate (Fraction/Day)(5) 0-2 hr	0-8 hr	8-24 hr	1-4 day	4-30 day
CPS supply line to supp. chamber(6)	Standby Gas Trtmt Area	2-12" valves	3.75	0.523×10^{-4}	0.507×10^{-4}	0.462×10^{-4}	0.450×10^{-4}	0.406×10^{-4}	0.287×10^{-4}
CPS supply line to supp. chamber(6)	Standby Gas Trtmt Area	2-2" valves	0.625	0.871×10^{-5}	0.845×10^{-5}	0.770×10^{-5}	0.749×10^{-5}	0.677×10^{-5}	0.478×10^{-5}

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(1) Std Conditions: 14.7 psia and 68°F
(2) Fraction/Day is defined as fraction of drywell volume leakage/day per line under test conditions.
(3) Test Conditions: Air medium; 40 psig and 80°F
(4) The leak rate is based on ASME Section XI (Subsection IWV-3426) applied to each valve, except for main steam and feedwater lines.
(5) Fraction/day is defined as fraction of drywell volume leakage/day under LOCA conditions.
(6) Leak rate is defined as a fraction of entire primary containment volume under LOCA conditions. 21

TABLE 6.2-55b

EVALUATION OF POTENTIAL BYPASS LEAKAGE PATHS (ISENTROPIC FLOW MODEL)

Line Description	Termination Region	Bypass Leakage Barrier	Leak Rate[3] Tech Spec SCFH[1,4]	Leak Rate[3] Fraction/Day[2]	Containment Bypass Leak Rate (Fraction/Day)[5] 0-2 hr	0-8 hr	8-24 hr	1-4 day	4-30 day	21
4 Main steam lines	Turbine Bldg	2-21" valves in each line	6	0.139×10^{-3}	0.651×10^{-3}	0.606×10^{-3}	0.591×10^{-3}	0.550×10^{-3}	0.411×10^{-3}	21
Main steam drain line (inboard)	Turbine Bldg	1-6" valve	1.875	0.435×10^{-4}	0.663×10^{-4}	0.615×10^{-4}	0.597×10^{-4}	0.564×10^{-4}	0.442×10^{-4}	21
Main steam drain line (outboard)	Turbine Bldg	1-2" valve	0.625	0.145×10^{-4}	0.221×10^{-4}	0.205×10^{-4}	0.199×10^{-4}	0.188×10^{-4}	0.147×10^{-4}	
4 Post accident sampling lines	Radwaste Tunnel	1-3/4" valve in each line	0.2344	0.543×10^{-5}	0.331×10^{-4}	0.307×10^{-4}	0.298×10^{-4}	0.282×10^{-4}	0.221×10^{-4}	21
Drywell equipment drain line	Radwaste Tunnel	1-4" valve	1.25	0.290×10^{-4}	0.442×10^{-4}	0.410×10^{-4}	0.398×10^{-4}	0.376×10^{-4}	0.294×10^{-4}	
Drywell equipment vent line	Radwaste Tunnel	1-2" valve	0.625	0.145×10^{-4}	0.221×10^{-4}	0.205×10^{-4}	0.199×10^{-4}	0.188×10^{-4}	0.147×10^{-4}	
Drywell floor drain line	Radwaste Tunnel	1-6" valve	1.875	0.435×10^{-4}	0.663×10^{-4}	0.615×10^{-4}	0.597×10^{-4}	0.564×10^{-4}	0.442×10^{-4}	21
Drywell floor vent line	Radwaste Tunnel	1-3" valve	0.9375	0.217×10^{-4}	0.331×10^{-4}	0.307×10^{-4}	0.298×10^{-4}	0.282×10^{-4}	0.221×10^{-4}	
RWCU line	Turbine Bldg	1-8" valve	2.5	0.579×10^{-4}	0.883×10^{-4}	0.820×10^{-4}	0.796×10^{-4}	0.751×10^{-4}	0.589×10^{-4}	21
Feedwater line	Turbine Bldg	2-24" check valves	12	0.278×10^{-3}	0.326×10^{-3}	0.303×10^{-3}	0.296×10^{-3}	0.275×10^{-3}	0.206×10^{-3}	
CPS supply line to drywell	Standby Gas Trtmt Area	2-14" valves	4.38	0.102×10^{-3}	0.119×10^{-3}	0.111×10^{-3}	0.108×10^{-3}	0.100×10^{-3}	0.751×10^{-4}	21
CPS supply line to drywell	Standby Gas Trtmt Area	2-2" valves	0.625	0.145×10^{-4}	0.170×10^{-4}	0.158×10^{-4}	0.154×10^{-4}	0.143×10^{-4}	0.107×10^{-4}	

TABLE 6.2-55b (Cont)

Line Description	Termination Region	Bypass Leakage Barrier	Leak Rate[3] Tech Spec SCFH[1,4]	Leak Rate[3] Fraction/Day[2]	Containment Bypass Leak Rate (Fraction/Day)[5] 0-2 hr	0-8 hr	8-24 hr	1-4 day	4-30 day
CPS supply line[6] to supp. chamber	Standby Gas Trtmt Area	2-12" valves	3.75	0.523×10^{-4}	0.612×10^{-4}	0.570×10^{-4}	0.556×10^{-4}	0.517×10^{-4}	0.387×10^{-4}
CPS supply line[6] to supp. chamber	Standby Gas Trtmt Area	2-2" valves	0.625	0.871×10^{-5}	0.102×10^{-4}	0.950×10^{-5}	0.926×10^{-5}	0.861×10^{-5}	0.644×10^{-5}

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(1) Std Conditions: 14.7 psia and 68°F
(2) Fraction/Day is defined as fraction of drywell volume leakage/day per line under test conditions.
(3) Test Conditions: Air medium; 40 psig and 80°F
(4) The leak rate is based on ASME Section XI (Subsection IWV-3426) applied to each valve, except for main steam lines.
(5) Fraction/Day is defined as fraction of drywell volume leakage/day under LOCA conditions.
(6) Leak rate is defined as a fraction of entire primary containment volume under LOCA conditions. 21

TABLE 6.2-56 (Cont)

Pene-tration No.	System Designation	GDC or Reg. Guide	ESF System	Fluid	Size (in)	FSAR Arrange-ment Figure(1)	Location of Valve Inside/ Outside Primary Contain-ment	Length of Pipe - Con-tainment to Outside Isolation Valve	Type Test (1)	Potential Bypass Leakage Path	Valve(9) Number SWEC	Number GE	Type	Oper-ator	Actuator Mode Primary	Actuator Mode Secondary	Position Normal (3)	Position Shutdown	Position Post-Accident	Position Power Failure(10)	Isola-tion Signal (4)	Closure Time (5,6)	Power Source (7)	Notes	
Z-4A	Feedwater line A to RPV	55	No	Water	24	6.2-70 Sh. 3	Outside	2'-1"	C	Yes	2FWS*AOV23A	B22-F032A	Swing Check	AOV	Process	Spring (test only)	Open	Closed	Closed	N/A	Reverse flow	The time it takes for one valve volume to pass through the valve	N/A	11,32	21
							Inside		C		2FWS*V12A	B22-F010A	Swing Check	N/A	Process	N/A	Open	Closed	Closed	N/A	Reverse flow				
							Outside	16'-4"	C		2FWS*MOV21A	B22-F065A	Gate	MOV	Elec.	Manual	Open	Closed	Closed	FAI	RM	N/A	Div I		
				Water	8	6.2-70 Sh. 3	Outside	57'-8"	C		2WCS*MOV200	G33-F040	Globe	MOV	Elec.	Manual	Open	Open	Closed	FAI	RM	N/A	Div I		
Z-4B	Feedwater line B to RPV	55	No	Water	24	6.2-70 Sh. 3	Inside		C	Yes(36)	2FWS*V12B	B22-F010B	Swing Check	N/A	Process	N/A	Open	Closed	Closed	N/A	Reverse flow	The time it takes for one valve volume to pass through the valve	N/A	11,32	21
							Outside	2'-1"	C		2FWS*AOV23B	B22-F032B	Swing Check	AOV	Process	Spring (test only)	Open	Closed	Closed	N/A	Reverse flow				
							Outside	16'-4"	C		2FWS*MOV21B	B22-F065B	Gate	MOV	Elec.	Manual	Open	Closed	Closed	FAI	RM	N/A	Div II		
				Water	8	6.2-70 Sh. 3	Outside	65'-8"	C		2WCS*MOV200	G33-F040	Globe	MOV	Elec.	Manual	Open	Open	Closed	FAI	RM	N/A	Div I		
Z-5A	RHS Pump A suction from suppression pool	56	Yes	Water	24	6.2-70 Sh. 4	Outside	5'-6"	C	No(29)	2RHS*MOV1A	E12-F004A	Tricen-tric butter-fly	MOV	Elec.	Manual	Open	Closed	Open	FAI	RM	45	Div I	13,35	
Z-5B	RHS Pump B suction from suppression pool	56	Yes	Water	24	6.2-70 Sh. 4	Outside	20'-9"	C	No(29)	2RHS*MOV1B	E12-F004B	Tricen-tric butter-fly	MOV	Elec.	Manual	Open	Closed	Open	FAI	RM	45	Div II	13,35	
Z-5C	RHS Pump C suction from suppression pool	56	Yes	Water	24	6.2-70 Sh. 4	Outside	9'-9"	C	No(29)	2RHS*MOV1C	E12-F004C	Tricen-tric butter-fly	MOV	Elec.	Manual	Open	Closed	Open	FAI	RM	45	Div II	13, 35	
Z-6A	RHS test line Loop B to sup-pression pool	56	Yes	Water	18	6.2-70 Sh. 6	Outside	19'-3"	C	No(29)	2RHS*MOV30B	E12-F201B	Tricen-tric butter-fly	MOV	Elec.	Manual	Open	Closed	Open	FAI	RM	85	Div I	15, 35	

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TABLE 6.2-56 (Cont)

Pene-tration No.	System Designation	GDC or Reg. Guide	ESF System	Fluid	Size (in)	FSAR Arrange-ment Figure(1)	Location of Valve Inside/Outside Primary Contain-ment	Length of Pipe - Con-tainment to Outside Isolation Valve	Type Test (2)	Potential Bypass Leakage Path	Valve(8) Number SWEC	Valve(8) Number GE	Type	Oper-ator	Actuator Mode Primary	Actuator Mode Secondary	Position Normal (3)	Position Shutdown	Position Post-Accident	Position Power Failure(10)	Isola-tion Signal (4)	Closure Time (5,6)	Power Source (7)	Notes	
Z-46B	Capped spare				4				A																
Z-46C	Fire protection water for con-tainment hose reel standpipe						See Note 20			No(31)															
Z-46D	Capped spare				4				A																
Z-47	CCP return from drywell space cooler	57	No(31)	Water	8	6.2-70 Sh. 28	Inside		C	No(31)	2CCP*SOV122	-	Gate	MOV	Elec.	Manual	Open	Open	Closed	FAI	B,F,RM	38	Div II		
							Outside	7'-3"	C		2CCP*SOV124	-	Gate	MOV	Elec.	Manual	Open	Open	Closed	FAI	B,F,RM	36	Div I		
Z-48	Purge exhaust from drywell	56	No	Air	14	6.2-70 Sh. 29	Inside	-	C	No(31)	2CPS*AOV108	-	Butter-fly	AOV	Pneu-matic	Manual	Closed	Closed	Closed	Closed	B,F,Y,RM	5	Div II		
							Outside	7'-3"	C		2CPS*AOV110	-	Butter-fly	AOV	Pneu-matic	Manual	Closed	Closed	Closed	Closed	B,F,Y,RM	5	Div I		
Z-49	Purge inlet to drywell	56	No	Air/N_2	14	6.2-70 Sh. 29	Inside	-	C	Yes	2CPS*AOV106	-	Butter-fly	AOV	Pneu-matic	Manual	Closed	Closed	Closed	Closed	B,F,Y,RM	5	Div II	21	21
							Outside	4'-0"	C		2CPS*AOV104	-	Butter-fly	AOV	Pneu-matic	Manual	Closed	Closed	Closed	Closed	B,F,Y,RM	5	Div I		
Z-50	Purge inlet to wetwell	56	No	Air/N_2	12	6.2-70 Sh. 29	Inside	-	C	Yes	2CPS*AOV107	-	Butter-fly	AOV	Pneu-matic	Manual	Closed	Closed	Closed	Closed	B,F,Y,RM	5	Div II		21
							Outside	4'-3"	C		2CPS*AOV105	-	Butter-fly	AOV	Pneu-matic	Manual	Closed	Closed	Closed	Closed	B,F,Y,RM	5	Div I		
Z-51	Purge exhaust from wetwell	56	No	Air	12	6.2-70 Sh. 29	Inside	-	C	No(31)	2CPS*AOV109	-	Butter-fly	AOV	Pneu-matic	Manual	Closed	Closed	Closed	Closed	B,F,Y,RM	5	Div II		
							Outside	6'-6"	C		2CPS*AOV111	-	Butter-fly	AOV	Pneu-matic	Manual	Closed	Closed	Closed	Closed	B,F,Y,RM	5	Div I		
Z-52A	Capped spare				1				A																
Z-52B	Capped spare				1				A																
Z-53A	Instrument air to ADS valve accumulators	56	No	N_2	1 1/2	6.2-70 Sh. 30	Outside	1'-0"	C	Yes(30)	2IAS*SOV164		Globe	SOV	Elec.	N/A	Open	Open	Open	Closed	B,F,RM	N/A	Div I		
							Inside		C		2IAS*V448		Check	N/A	Process	N/A	Open	Open	Open	N/A	Reverse flow	N/A	N/A		

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Pene-tration No.	System Designation	GDC or Reg. Guide	ESF System	Fluid	Size (in)	FSAR Arrange-ment Figure(1)	Location of Valve Inside/Outside Primary Contain-ment	Length of Pipe - Con-tainment to Outside Isolation Valve	Type Test (1)	Potential Bypass Leakage Path	Valve(9) Number SWEC	Number GE	Type	Oper-ator	Actuator Mode Primary	Actuator Mode Secondary	Position Normal (3)	Position Shutdown	Position Post-Accident	Position Power Failure(10)	Isola-tion Signal (4)	Closure Time (5,6)	Power Source (7)	Notes
Z-53B	Instrument air to ADS valve accumulators	56	No	N_2	1 1/2	6.2-70 Sh. 30	Outside Inside	1'-0"	C C	Yes(30)	2IAS*SOV165 2IAS*V449	- -	Globe Check	SOV N/A	Elec. Process	N/A N/A	Open Open	Open Open	Open Open	Closed N/A	B,F,RM Reverse flow	N/A N/A	Div II N/A	
Z-53C	Instrument air to SRV accumu-lator tank	56	No	N_2	1 1/2	6.2-70 Sh. 30	Outside Inside	1'-0"	C C	Yes(30)	2IAS*SOV166 2IAS*SOV184	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Open Open	Closed Closed	Closed Closed	B,F,RM B,F,RM	N/A N/A	Div I Div II	
Z-54A	Capped spare				3				A															
Z-55A	Hydrogen recom-biner 1A supply to wetwell	56	Yes	Air	3	6.2-70 Sh. 31	Inside Outside	2'-0"	A, C A, C	No(31)	2HCS*MOV4A 2HCS*MOV1A	- -	Globe Globe	MOV MOV	Elec. Elec.	Manual Manual	Closed Closed	Closed Closed	Open Open	FAI FAI	B,F,RM B,F,RM	19 19	Div I Div I	12, 22
Z-55B	Hydrogen recom-biner 1B supply to wetwell	56	Yes	Air	3	6.2-70 Sh. 31	Inside Outside	2'-0"	A, C A, C	No(31)	2HCS*MOV4B 2HCS*MOV1B	- -	Globe Globe	MOV MOV	Elec. Elec.	Manual Manual	Closed Closed	Closed Closed	Open Open	FAI FAI	B,F,RM B,F,RM	19 19	Div II Div II	12, 22
Z-56A	Hydrogen recom-biner 1A return from drywell	56	Yes	Air	3	6.2-70 Sh. 31	Inside Outside	2'-0"	A, C A, C	No(31)	2HCS*MOV6A 2HCS*MOV3A	- -	Globe Globe	MOV MOV	Elec. Elec.	Manual Manual	Closed Closed	Closed Closed	Open Open	FAI FAI	B,F,RM B,F,RM	19 19	Div I Div I	12, 22
Z-56B	Hydrogen recom-biner 1B return from drywell	56	Yes	Air	3	6.2-70 Sh. 31	Inside Outside	2'-0"	A, C A, C	No(31)	2HCS*MOV6B 2HCS*MOV3B	- -	Globe Globe	MOV MOV	Elec. Elec.	Manual Manual	Closed Closed	Closed Closed	Open Open	FAI FAI	B,F,RM B,F,RM	19 19	Div II Div II	12, 22
Z-57A	Hydrogen recom-biner 1A return from wetwell	56	Yes	Air	3	6.2-70 Sh. 31	Inside Outside	2'-0"	A, C A, C	No(31)	2HCS*MOV5A 2HCS*MOV2A	- -	Globe Globe	MOV MOV	Elec. Elec.	Manual Manual	Closed Closed	Closed Closed	Open Open	FAI FAI	B,F,RM B,F,RM	19 19	Div I Div I	12, 22
Z-57B	Hydrogen recom-biner 1B return from wetwell	56	Yes	Air	3	6.2-70 Sh. 31	Inside Outside	2'-0"	A, C A, C	No(31)	2HCS*MOV5B 2HCS*MOV2B	- -	Globe Globe	MOV MOV	Elec. Elec.	Manual Manual	Closed Closed	Closed Closed	Open Open	FAI FAI	B,F,RM B,F,RM	19 19	Div II Div II	12, 22
Z-58	Containment purge to dry-well	56	No	Air	2	6.2-70 Sh. 29	Inside Outside	3'-4"	C C	Yes	2CPS*SOV122 2CPS*SOV120	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Closed Closed	Closed Closed	Closed Closed	Closed Closed	B,F,Y RM B,F,Y RM	N/A N/A	Div II Div I	21

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TABLE 6.2-56 (Cont)

Pene-tration No.	System Designation	GDC or Reg. Guide	ESF System	Fluid	Size (in)	FSAR Arrange-ment Figure(1)	Location of Valve Inside/ Outside Primary Contain-ment	Length of Pipe - Con-tainment to Outside Isolation Valve	Type Test (2)	Potential Bypass Leakage Path	Valve(9) Number SWEC	Number GE	Type	Oper-ator	Actuator Mode Primary	Actuator Mode Secondary	Position Normal (3)	Position Shutdown	Position Post-Accident	Position Power Failure(10)	Isola-tion Signal (4)	Closure Time (5,6)	Power Source (7)	Notes
Z-59	Containment purge to wet-well	56	No	Air	2	6.2-70 Sh. 29	Inside		C	Yes	2CPS*SOV121	-	Globe	SOV	Elec.	N/A	Closed	Closed	Closed	Closed	B,F,Z, PS	N/A	Div II	21
							Outside	18'-6"	C		2CPS*SOV119	-	Globe	SOV	Elec.	N/A	Closed	Closed	Closed	Closed	B,F,Z, PS	N/A	Div I	
Z-60A	CMS from dry-well	56	No	Air	3/4	6.2-70 Sh. 32	Inside Outside	1'-2"	C C	No(31)	2CMS*SOV51A 2CMS*SOV50A	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Open	Closed Closed	B,F,RM B,F,PS	N/A N/A	Div II Div I	
Z-60B	CMS from dry-well	56	Yes	Air	3/4	6.2-70 Sh. 32	Inside Outside	1'-2"	C C	Yes(33)	2CMS*SOV24A 2CMS*SOV24C	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Open Open	Closed Closed	B,F,RM B,F,PS	N/A N/A	Div I Div I	
Z-60C	CMS to dry-well	56	No	Air	3/4	6.2-70 Sh. 32	Inside Outside	0'-3"	C C	No(31)	2CMS*SOV63A 2CMS*SOV62A	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,PS B,F,RS	N/A N/A	Div II Div I	
Z-60D	CMS to dry-well	56	Yes	Air	3/4	6.2-70 Sh. 32	Inside Outside	0'-4"	C C	Yes(33)	2CMS*SOV33A 2CMS*SOV32A	- -	Globe Globe	SOV SOV	Elec. Elec.	Elec. N/A	Open Open	Closed Closed	Open Open	Closed Closed	B,F,PS B,F,RS	N/A N/A	Div I Div I	
Z-60E	CMS from dry-well	56	No	Air	3/4	6.2-70 Sh. 32	Inside Outside	0'-7"	C C	No(31)	2CMS*SOV61B 2CMS*SOV60B	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,RM B,F,RM	N/A N/A	Div II Div I	
Z-60F	CMS from dry-well	56	Yes	Air	3/4	6.2-70 Sh. 32	Inside Outside	0'-7"	C C	Yes(33)	2CMS*SOV24B 2CMS*SOV24D	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Open Open	Closed Closed	B,F,RM B,F,PS	N/A N/A	Div II Div II	
Z-60G	CMS to drywell	56	No	Air	3/4	6.2-70 Sh. 32	Inside Outside	0'-7"	C C	No(31)	2CMS*SOV63B 2CMS*SOV62B	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,RS B,F,PS	N/A N/A	Div II Div I	
Z-60H	CMS to drywell	56	Yes	Air	3/4	6.2-70 Sh. 32	Inside Outside	1'-0"	C C	Yes(33)	2CMS*SOV33B 2CMS*SOV32B	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Open Open	Closed Closed	B,F,RS B,F,RS	N/A N/A	Div II Div II	
Z-61A	Capped spare				3/4				A															
Z-61B	CMS from wet-well	56	Yes	Air	3/4	6.2-70 Sh. 32	Inside Outside	15'-0"	C C	No(31)	2CMS*SOV26A 2CMS*SOV26C	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Open Open	Closed Closed	B,F,PS B,F,RM	N/A N/A	Div I Div I	
Z-61C	CMS to wetwell	56	Yes	Air	3/4	6.2-70 Sh. 32	Inside Outside	18'-3"	C C	No(31)	2CMS*SOV34A 2CMS*SOV35A	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Open Open	Closed Closed	B,F,PS B,F,RS	N/A N/A	Div I Div I	
Z-61D	Capped spare				3/4				A															
Z-61E	CMS from wet-well	56	Yes	Air	3/4	6.2-70 Sh. 32	Inside Outside	0'-8"	C C	No(31)	2CMS*SOV26B 2CMS*SOV26D	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Open Open	Closed Closed	B,F,RM B,F,RS	N/A N/A	Div II Div II	

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15.6.5 Loss-of-Coolant Accidents (LOCA) (Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary) Inside Primary Containment

This event involves the postulation of a spectrum of piping breaks inside containment varying in size, type, and location. The break type includes steam and/or liquid process system lines. This event is also assumed to be coincident with a safe shutdown earthquake (SSE).

The event has been analyzed quantitatively in Sections 6.2, 6.3, 7.1, 7.3, and 8.3. Therefore, the following discussion provides information not presented in the subject sections. All other information is covered by cross-referencing.

The postulated event represents the envelope evaluation for liquid or steam line failures inside containment.

15.6.5.1 Identification of Causes and Frequency Classification

15.6.5.1.1 Identification of Causes

There are no realistic, identifiable events which would result in a pipe break inside the containment of the magnitude required to cause a LOCA coincident with SSE plus SACF criteria requirements. The subject piping is designed to high quality and strict industry code and standard criteria and severe seismic and environmental conditions. However, since such an accident provides an upper limit estimate to the resultant effects for this category of pipe breaks, it is evaluated without the causes being identified.

15.6.5.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.5.2 Sequence of Events and Systems Operation

15.6.5.2.1 Sequence of Events

The sequence of events associated with this accident is shown in Table 6.3-2 for core system performance and Table 6.2-8 for barrier (containment) performance.

Following the pipe break and scram, the low-low water level or high drywell pressure signal initiates HPCS and RCIC systems at time 0, plus approximately 30 sec.

The low-low-low water level or high drywell pressure signal initiates MSIV closure, and both the LPCS and LPCI systems at time 0, plus approximately 40 sec.

Since automatic actuation and operation of the ECCS is a system design basis, no operator actions are required for the accident.

15.6.5.2.2 Systems Operations

Accidents that could result in the release of radioactive fission products directly into the containment are the results of postulated nuclear system primary coolant pressure boundary pipe breaks. Possibilities for all pipe break sizes and locations are examined in Sections 6.2 and 6.3, including the severance of small process system lines, the main steam lines upstream of the flow restrictors, and the recirculation loop pipelines. The most severe nuclear system effects and the greatest release of radioactive material to the containment result from a complete circumferential break of one of the two recirculation loop pipelines. The minimum required functions of any reactor and plant protection system are discussed in Sections 6.2, 6.3, 7.3, 7.6, and 8.3, and Appendix 15A.

15.6.5.2.3 Effect of Single Failures and Operator Errors

Single failures and operator errors have been considered in the analysis of the entire spectrum of primary system breaks. The consequences of a LOCA with considerations for single failures are shown to be fully accommodated without the loss of any required safety function (see Appendix 15A for further details).

15.6.5.3 Core and System Performance

15.6.5.3.1 Mathematical Model

The analytical methods and associated assumptions which are used in evaluating the consequences of this accident provide conservative assessment of the expected consequences of this very improbable event. The details of these calculations, their justification, and bases for the models are developed in Sections 6.3, 7.3, 7.6, 8.3, and Appendix 15A.

15.6.5.3.2 Input Parameters and Initial Conditions

Input parameters and initial conditions used for the analysis of this event are given in Table 6.3-1.

15.6.5.3.3 Results

Results of this event are given in detail in Section 6.3. The temperature and pressure transients resulting as a consequence of this accident are insufficient to cause perforation of the fuel cladding. Therefore, no fuel damage results from this accident. Post-accident tracking instrumentation and control is assured. Continued long-term core cooling is demonstrated. Radiological input is minimized and within limits. Continued operator control and surveillance is examined and guaranteed.

15.6.5.3.4 Consideration of Uncertainties

This event was conservatively analyzed (see Sections 6.3, 7.3, 7.6, 8.3, and Appendix 15A for details).

15.6.5.4 Barrier Performance

The design basis for the containment is to maintain its integrity, and experience acceptable stresses after the instantaneous rupture of the largest single primary system piping within the structure, while also accommodating the dynamic effects of the pipe break at the same time an SSE is also occurring. Therefore, any postulated LOCA does not result in exceeding the containment design limit. For details and results of the analyses, see Sections 3.8, 3.9, and 6.2.

15.6.5.5 Radiological Consequences

Three separate radiological analyses are provided for this accident: 18

1. The first two, isentropic and isothermal (see Section 6.2.3.2), present two different approaches to the design basis analysis. The results of both approaches are used in determining the adequacy of the plant design to meet 10CFR100 and General Design Criterion 19 guidelines. 18

2. The third is based on assumptions considered to provide a realistic estimate of radiological consequences. This analysis is referred to as the realistic analysis. 18

15.6.5.5.1 Design Basis Analyses 18

The methods, assumptions, and conditions used to evaluate this accident are in accordance with those guidelines set forth in the NRC Standard Review Plan 15.6.5, Revision 2,

and Regulatory Guides 1.3 and 1.7. Specific values of parameters used in this evaluation are presented in Table 15.6-13.

15.6.5.5.2 Fission Product Release from Fuel

It is assumed that the reactor is operating at a power level of 3,489 MWt for 1,000 days prior to the accident. The airborne source immediately available for release from primary containment contains 100 percent of the core noble gas inventory and 25 percent of the core halogen inventory. The suppression pool source contains no noble gases and 50 percent of the core halogen inventory. While not specifically stated in Regulatory Guide 1.3 or Standard Review Plan 15.6.5, the assumed release of 100 percent of the core noble gas activity and 50 percent of the halogen activity implies fuel damage approaching melt conditions. Even though this condition is inconsistent with operation of the ECCS system (Section 6.3), it is assumed applicable for the evaluation of this accident. The airborne activity available for release from the primary containment at T=0 hr post-LOCA is presented in Table 15.6-14.

15.6.5.5.3 Fission Product Transport to the Environment

The transport pathways consist of leakage from the primary containment to the environment through several different mechanisms. Where applicable, the SGTS filter efficiency for halogen removal is assessed as 99 percent. The mechanisms for leakage from the primary containment are discussed in the following paragraphs:

1. Containment leakage - The technical specification leak rate of the primary containment and its penetrations (excluding the bypass leakage paths) is 1.1 percent per day for the duration of the accident. Additional primary containment leakage
21 occurs through one of the traversing incore probes (TIP) at 0.21 percent per day into secondary containment. All this leakage is to the reactor building, and from there to the environment via a 3,500 cfm SGTS. Credit is taken for 50 percent mixing within the reactor building. This leakage is the same for the isothermal and isentropic approaches to the design basis analysis.

2. Leakage from ESF components outside primary containment - 1 gpm total leakage of suppression pool fluid into the reactor building is assumed to occur for the duration of the accident. Ten

percent of the iodines in this leakage become airborne and available for release through the SGTS. This leakage is the same for the isothermal and isentropic approaches to the design basis analysis.

3. Reactor building pressurization - During the time when the pressure in the reactor building is greater than negative one-quarter inch water gauge compared to the environment leakage from the primary containment, TIP, and ESF systems travels directly to the environment. No credit is taken for mixing or filtration. This leakage is the same for the isothermal and isentropic approaches to the design basis analysis. 21

4. Bypass leakage - The piping paths listed below provide potential routes for post-LOCA primary containment atmosphere to bypass the reactor building and the standby gas treatment system and be released directly to the environment.

   1. Main steam lines (4).

   2. Feedwater line.

   3. Post-accident sampling lines (4).

   4. Main steam drain lines (2).

   5. Reactor water cleanup line.

   6. Drywell equipment drain and vent lines (2).

   7. Drywell floor drain and vent lines (2).

   8. Primary containment purge lines (4). 21

   Section 6.2.3 describes in detail the two methods used to determine the leak rates through the isolation valve(s) for each path. These two methodologies, one considering an isothermal flow process and the other considering an isentropic flow process, define the two separate approaches to the flow design basis analysis.

   Using the leak rate data from Tables 6.2-55a and 6.2-55b, a prerelease holdup time is calculated for each bypass leakage path using the slug-flow method. The slug-flow method assumes that the

leakage front occupies the full cross-sectional area of the pipe as it travels through the line. As time increases and the isolation valves leak at progressively lower rates due to a lowering of the differential pressure across the valves, the slugs move more slowly down the pipe.

The holdup time is equal to the time it takes the
leakage through the valve to fill the volume of
piping between the isolation valve and the release
21 point. One-half of the actual calculated holdup
times are used.

21 Credit is taken for the deposition of elemental and
particulate iodines (which constitute 91 percent
21 and 5 percent, respectively, of the total iodines
released from the core per Regulatory Guide 1.3) on
the walls of the piping between the isolation valve
and the release point for each bypass leakage path.
In addition, 4 percent of the iodines released are
21 assumed to be in organic form. The elemental and
particulate iodine plate out mechanism described
below is derived from information given in
references 6 through 11 (Section 15.6.7). The
21 ratio of the elemental and particulate iodine
concentration at the release point to the
concentration at the isolation valve is:

$$\frac{C_{RP}}{C_{IV}} = \exp - \left[\frac{A_d V_d}{A_c V_B}\right]$$

where:

C_{RP}, C_{IV} = Elemental and particulate iodine concentration at the release point and the isolation valve, respectively (uCi/cc)

A_d = Pipe deposition area

V_d = Deposition velocity for elemental iodines = $9.0 \times 10^{-8} \exp [8100/RT_s]$ cm/sec

R = Gas constant = 1.987 cal./mole-°K

T_s = Absolute temperature (°K)

21 A_c = Flow cross-sectional area

V_B = Flow velocity 21

The change in C_{RP}/C_{IV} due to the change in temperature and flow velocity with time is taken into account conservatively. The temperature of the air/steam mixture and the piping for the main steam line, FW line, and RWCU line iodine deposition analysis is assumed to be decreasing at a rate of 100°F/day for first two days. For all other iodine deposition analyses, the temperature of the air/steam mixture and piping is assumed to be 120°F and 104°F for the duration of the accident. These temperature profiles are assumed to be the same for the isothermal and isentropic flow approaches. 21 21

No reduction due to deposition is taken for the remaining 4 percent of the core iodines released that are not in elemental or particulate form. Therefore, the ratio of the total iodine concentration at the release point to the total iodine concentration at the isolation valve is: 21

$$\frac{I_{RP}}{I_{IV}} = \left(\frac{C_{RP}}{C_{IV}}\right) * 0.96 + 1.0 * 0.04$$

As shown in Section 6.2.3, the flow rates for each bypass leakage path are different for the isentropic and isothermal cases. Therefore, the iodine deposition factors are also different.

All pertinent data used to determine the radiological consequences of a design basis LOCA are presented in Table 15.6-13.

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Fission product releases to the environment based on the foregoing assumptions for both the isothermal and isentropic approaches are given in Tables 15.6-15a and 15.6-15b. 18

15.6.5.5.4 Results

The calculated exposures for the design basis analysis are presented in Tables 15.6-16a and 15.6-16b and are within the guidelines of 10CFR100 and General Design Criterion 19. 18

15.6.5.5.5 Realistic Analysis

The realistic analysis is based on a realistic, but still conservative, assessment of this accident. Specific values of parameters used in the evaluation are presented in Table 15.6-13.

Fission Product Release from Fuel

Since this accident does not result in any fuel damage, the only activity released to the primary containment is that activity contained in the reactor coolant, plus any additional activity which may be released as a consequence of reactor scram and vessel depressurization.

While there are various activation and corrosion products contained in the reactor coolant, the products of primary

importance are the iodine isotopes I-131 to I-135. The design reactor coolant iodine activities are normalized to the maximum technical specification[5] limits. The coolant concentrations and the normalized concentrations for these isotopes are presented in Table 15.6-17.

Considering that approximately 40 percent of the released liquid flashes to steam, it is conservatively assumed that 40 percent of the released iodine activity is airborne initially.

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As a consequence of reactor scram and depressurization, additional iodine activity is released from those rods which experienced cladding perforation during normal operation. The reactor coolant iodine activities that are normalized to the maximum technical specification[4] limits are multiplied by 500 to account for iodine spiking[5].

While no measurements have been obtained during a pressure transient as rapid as the LOCA, it is difficult to predict the actual release rate from the fuel as a consequence of iodine spiking. It is, therefore, arbitrarily assumed that 100 percent of the spiking source term is released during the time period that 40 percent of the discharged coolant is flashing to steam. The initial airborne iodine concentrations resulting are presented in Table 15.6-18.

Since it is also assumed that plate-out and condensation remove 50 percent of the airborne iodine activity, the resultant activity airborne in the primary containment from spiking is presented in Table 15.6-18.

Fission Product Transport to the Environment

A large reactor coolant pipe fails and instantaneously releases the entire mass of coolant in the reactor vessel and recirculation system to the primary containment. The activity airborne in the primary containment and available for release leaks to the reactor building at a constant rate for 30 days. From 0 to 90 sec post-LOCA, the reactor building
13 ing cannot maintain a pressure less than -1/4 in. W.G. During this period, the activity leaking from the primary
13 containment is assumed to be released directly to the environment. The standby gas treatment system (SGTS) begins
13 operation within 25 sec after the LOCA signal. After 90 sec, the reactor building returns to a pressure less than -1/4 in. W.G. and for the remaining 30 days the activity airborne in the reactor building is removed and filtered by the SGTS and exhausted through the main stack.

The leak rate from the primary containment to the reactor building is 1.1 percent/day, where 50 percent mixing is assumed to occur. Release from the reactor building to the environment through a 99 percent iodine-efficient SGTS is at a rate of 3,500 cfm. The integrated isotopic activity released to the environment is presented in Table 15.6-19.

Results

The calculated radiological exposures for this event are presented in Table 15.6-20 and, as shown, are a small fraction of 10CFR100.

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15.6.6.3.3 Consideration of Uncertainties

This event was conservatively analyzed and uncertainties were adequately considered (Section 6.3).

15.6.6.4 Barrier Performance

Accidents that result in the release of radioactive materials outside the containment are the results of postulated breaches in the RCPB or the steam power-conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the main steam lines as described in Section 15.6.4. The feedwater system piping break is less severe than the main steam line break. Results of analysis of this event can be found in Sections 6.2.3 or 6.2.4.

15.6.6.5 Radiological Consequences

15.6.6.5.1 Design Basis Analysis

The NRC provides no specific regulatory guidelines for the evaluation of this accident; therefore, no specific design basis analysis is presented. However, the radiological consequences of this event are enveloped by the results of analyses for the main steam line break (presented in Section 15.6.4.5). This is considered justified since the feedwater line check valves isolate the reactor from the downstream side of the break at time ϕ+ seconds after accident initiation. Therefore, there is no reactor coolant backflow via the feedwater lines connected to the RPV. The only contribution is from main steam, which must first pass through the turbines, condenser, and other condensate and feedwater system components, which reduce isotopic concentrations due to decay and demineralization. In the main steam line break analysis, a coincident iodine spike is assumed based on a compound spiking sequence giving 4 uCi/gm dose equivalent I-131, as described in Section 15.6.4.5.2.

15.6.7 References

1. Moody, F. J. Maximum Two-Phase Vessel Blowdown From Pipes. ASME Paper Number 65-WA/HT-1, March 15, 1965.

2. Nguyen, D. et. al. Radiological Accident Evaluation - The CONCACO3 Code, NEDO-21143-1, December 1981.

3. DRAGON 4 Code, Dose and Radioactivity From Nuclear Facility Gaseous Outflows, NU-115, Version 4, Level 1, April 1982.

4. USNRC Standard Technical Specifications for General Electric Boiling Water Reactors. NUREG-0123, Rev. 2, Washington, D.C., August 1979.

5. USNRC Standard Review Plan, Radiological Consequences of a Small Line Carrying Primary Coolant Outside Containment, 15.6.2, Rev. 2, July 1981.

6. Report of the Special Committee on Source Terms, The American Nuclear Society, September 1984.

7. Genco, J. M., et al. Fission Product Deposition and Its Enhancement Under Reactor Accident Conditions: Deposition on Primary - System Surfaces, March 1969.

8. Kress, T. S. and Wright, A. L. Status of the Validation of the TRAP-MELT Computer Code for the Accident Source
18 Term Reassessment Study (ASTRS), Oak Ridge National Laboratory (Draft).

9. NUREG/CR-2713, BMI 2091, Vapor Deposition Velocity Measurements and Correlations for I_2 and CsI, May 1982.

10. NUREG/CR-0009, Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels, October 1978.

11. EPRI NP-876, Surface Effects in the Transport of Airborne Radioiodine at Light Water Nuclear Power Plants, September 1978.

TABLE 15.6-13

LOSS-OF-COOLANT ACCIDENT - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions				Realistic Basis Assumptions
	Isentropic Case		Isothermal Case		
1. Data and assumptions used to estimate radioactive source from postulated accidents					
a. Power level	3,489 MWt		3,489 MWt		3,489 MWt
b. Release of activity to containment air	100% core noble gas inventory 25% core halogen inventory		100% core noble gas inventory 25% core halogen inventory		100% of iodines in coolant flashing to steam
c. Release of activity to suppression pool	50% core halogen inventory		50% core halogen inventory		N/A
d. Iodine fractions					
(1) Organic	0.04		0.04		0.0
(2) Elemental	0.91		0.91		1.0
(3) Particulate	0.05		0.05		0.0
e. Computer code used(1)	Dragon 4		Dragon 4		Dragon 4
f. Single active failure	Loss of diesel		Loss of diesel		N/A
2. Data and assumptions used to estimate activity released					
a. Total mass of coolant released	N/A		N/A		2.72+8g
b. Four main steam lines					
Bypass leakage rates (fractions of drywell volume per day) (main steam tunnel release)	0-2 hr	0.0(2)	0-2 hr	0.0(2)	0.0
	0-8 hr	0.0	0-8 hr	0.0	0.0
	8-24 hr	0.0	8-24 hr	0.0	0.0
	24-24.37 hr	0.0	24-31.74 hr	0.0	0.0
	24.37-96 hr	5.48-4	31.74-96 hr	4.28-4	0.0
	96-720	4.12-4	96-720 hr	3.05-4	0.0

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TABLE 15.6-13 (Cont)

	Design Basis Assumptions: Isentropic Case		Design Basis Assumptions: Isothermal Case		Realistic Basis Assumptions	
Iodine concentration ratios	0-8 hr	0.0	0-8 hr	0.0	0.0	
	8-24 hr	0.0	8-24 hr	0.0	0.0	
	24-24.37 hr	0.0	24-31.74 hr	0.0	0.0	
	24.37-96 hr	0.273	31.74-96 hr	0.255	0.0	
	96-720 hr	0.04	96-720 hr	0.047	0.0	
Pipe inside diameter (actual/design basis)	25.23 in/25.23 in		25.23 in/25.23 in		N/A	
Pipe length (actual/design basis)	508 ft/312 ft		508 ft/312 ft		N/A	
Deposition surface (actual/design basis)	3,355 ft^2/2,060 ft^2		3,355 ft^2/2,060 ft^2		N/A	
Temperature transient - pipe inside surface	0-1 day 450°F		0-1 day 450°F		N/A	
	1-2 day 450-350°F		1-2 day 450-350°F			
	2-3 day 350-250°F		2-3 day 350-250°F			
	3-4 day 250-120°F		3-4 day 350-120°F			
	4-30 day 120°F		4-30 day 120°F			
c. Inboard main steam drain line						
Bypass leakage rates (fraction of drywell volume per day) (main steam tunnel release)	0-2 hr	0.0(2)	0-2 hr	0.0(2)	0.0	
	0-5.12 hr	0.0	0-5.51 hr	0.0	0.0	21
	5.12-8 hr	5.93-5	5.51-8 hr	5.44-5	0.0	
	8-24 hr	5.97-5	8-24 hr	5.55-5	0.0	
	24-96 hr	5.64-5	24-96 hr	4.97-5	0.0	
	96-720 hr	4.42-5	96-720 hr	3.44-5	0.0	
Iodine concentration ratios	0-5.12 hr	0.0	0-5.51 hr	0.0	0.0	
	5.12-720 hr	0.04	5.51-720 hr	0.04	0.0	
Pipe inside diameter (actual/design basis)	5.761 in/5.761 in		5.761 in/5.761 in		N/A	
Pipe length (actual/design basis)	84.3 ft/84.0 ft		84.3 ft/84.0 ft		N/A	
Deposition surface (actual/design basis)	127 ft^2/127 ft^2		127 ft^2/127 ft^2		N/A	
Temperature transient - pipe inside surface	0-720 hr 120°F		0-720 hr 120°F		N/A	
d. Four post accident sampling lines						
Bypass leakage rates (fraction of drywell volume per day) (radwaste tunnel release)	0-2 hr	3.31-5(2)	0-2 hr	3.17-5(2)	0.0	
	0-8 hr	3.07-5	0-8 hr	2.86-5	0.0	
	8-24 hr	2.98-5	8-24 hr	2.77-5	0.0	

TABLE 15.6-13 (Cont)

	Design Basis Assumptions Isentropic Case		Isothermal Case		Realistic Basis Assumptions
	24-96 hr	2.82-5	24-96 hr	2.48-5	0.0
	96-720 hr	2.21-5	96-720 hr	1.72-5	0.0
Iodine concentration ratio	All times 0.04		All times 0.04		0.0
Pipe inside diameter (actual/design basis)	0.18 in/0.18 in		0.18 in/0.18 in		N/A
Pipe length (actual/design basis)	935 ft/0 ft		935 ft/0 ft		N/A
Deposition surface (actual/design basis)	44 ft^2/1.0 ft^2		44 ft^2/1.0 ft^2		N/A
Temperature transient - pipe inside surfaces	0-720 hr 120°F		0-720 hr 120°F		N/A
e. One feedwater line					
Bypass leakage rates	0-2 hr	0.0[2]	0-2 hr	0.0[2]	0.0
(fraction of drywell volume per day)[3]	0-4.89 hr	0.0	0-6.20 hr	0.0	0.0
(main steam tunnel release)	4.89-8 hr	2.93-4	6.20-8	2.35-4	0.0
	8-24 hr	2.96-4	8-24 hr	2.39-4	0.0
	24-96 hr	2.75-4	24-96 hr	2.16-4	0.0
	96-720 hr	2.06-4	96-720 hr	1.53-4	0.0
Iodine concentration ratio	0-4.89 hr	0.0	0-6.20 hr	0.0	0.0
	4.89-8 hr	0.424	6.20-8 hr	0.419	0.0
	8-24 hr	0.228	8-24 hr	0.221	0.0
	24-96 hr	0.112	24-96 hr	0.087	0.0
	96-720 hr	0.04	96-720 hr	0.04	0.0
Pipe inside diameter (actual/design basis)	19.876 in/19.876 in		19.876 in/19.876 in		N/A
Pipe length (actual/design basis)	50 ft/50 ft		50 ft/50 ft		N/A
Deposition surface (actual/design basis)	260 ft^2/260 ft^2		260 ft^2/260 ft^2		N/A
Temperature transient - pipe inside surface	0-24 hr 425-325°F		0-24 hr 425-325°F		N/A
	24-48 hr 325-225°F		24-48 hr 325-225°F		N/A
	48-72 hr 225-120°F		48-72 hr 225-120°F		N/A
	72-720 hr 120°F		72-720 hr 120°F		
f. Outboard main steam drain line					
Bypass leakage rates	0-2 hr	2.21-5[2]	0-2 hr	2.11-5[2]	0.0
(fraction of drywell volume per day)	0-8 hr	2.05-5	0-8 hr	1.91-5	0.0

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TABLE 15.6-13 (Cont)

	Design Basis Assumptions: Isentropic Case		Design Basis Assumptions: Isothermal Case		Realistic Basis Assumptions	
(main steam tunnel release)	8-24 hr	1.99-5	8-24 hr	1.85-5	0.0	
	24-96 hr	1.88-5	24-96 hr	1.66-5	0.0	
	96-720 hr	1.47-5	96-720 hr	1.15-5	0.0	
Iodine concentration ratio	0-2 hr	0.05	0-2 hr	0.05		
	0-8 hr	0.05	0-8 hr	0.05	0.0	
	8-24 hr	0.041	8-24 hr	0.04		
	24-96 hr	0.04	24-96 hr	0.04		
	96-720 hr	0.04	96-720 hr	0.04		
Pipe inside diameter (actual/design basis)	1.687 in/1.687 in		1.687 in/1.687 in		N/A	
Pipe length (actual/design basis)	17.46 ft/0 ft		17.46 ft/0 ft		N/A	
Deposition surface (actual/design basis)	7.71 ft^2/1.0 ft^2		7.71 ft^2/1.0 ft^2		N/A	
Temperature transient - pipe inside surface	0-720 hr	120°F	0-720 hr	120°F	N/A	
g. Reactor water cleanup line						
Bypass leakage rates	0-2 hr	0.0(2)	0-2 hr	0.0(2)	0.0	21
(fraction of drywell volume per day)(3)	0-8 hr	0.0	0-8 hr	0.0	0.0	
(main steam tunnel release)	8-10.03 hr	0.0	8-10.80 hr	0.0	0.0	
	10.03-24 hr	7.97-5	10.80-24 hr	7.41-5	0.0	
	24-96 hr	7.51-5	24-96 hr	6.63-5	0.0	
	96-720 hr	5.89-5	96-720 hr	4.59-5	0.0	
Iodine concentration ratios	0-8 hr	0.0	0-8 hr	0.0	0.0	
	8-10.03 hr	0.0	8-10.80 hr	0.0	0.0	
	10.03-24 hr	0.218	10.80-24 hr	0.213	0.0	
	24-96 hr	0.064	24-96 hr	0.059	0.0	
	96-720 hr	0.04	96-720 hr	0.04	0.0	
Pipe inside (actual/design basis)	7.187 in/6.813 in		7.187 in/6.813 in		N/A	
Pipe length (actual/design basis)	599 ft/250 ft		599 ft/250 ft		N/A	
Deposition surface (actual/design basis)	1,127 ft^2/446 ft^2		168.7 ft^2/466 ft^2		N/A	
Temperature transient - pipe inside surface	0-24 hr	551-450°F	0-24 hr	551-450°F	N/A	
	24-48 hr	450-350°F	24-48 hr	450-350°F	N/A	
	48-72 hr	350-250°F	48-72 hr	350-250°F	N/A	
	72-96 hr	250-120°F	72-96 hr	250-120°F		

TABLE 15.6-13 (Cont)

	Design Basis Assumptions				Realistic Basis Assumptions	
	Isentropic Case		Isothermal Case			
	96-720 hr 120°F		96-720 hr 120°F			
h. Drywell equipment drain (DER) line						
Bypass leakage rates	0-1.24 hr	0.0(2)	0-1.29 hr	0.0(2)	0.0	
(fraction of drywell volume per day)	1.24-2 hr	4.25-5(2)	1.29-2 hr	4.00-5(2)	0.0	
(radwaste reactor building	0-1.24	0.0	0-1.29 hr	0.0	0.0	
vent release)	1.24-8 hr	4.01-5	1.29-8 hr	3.70-5	0.0	
	8-24 hr	3.98-5	8-24 hr	3.70-5	0.0	
	24-96 hr	3.76-5	24-96 hr	3.31-5	0.0	
	96-720 hr	2.94-5	96-720 hr	2.29-5	0.0	
Iodine concentration ratio	0-1.24 hr	0.0	0-1.29 hr	0.0	0.0	
	1.24-720 hr	0.04	1.29-720 hr	0.04	0.0	
Pipe inside diameter (actual/design basis)	4.026 in/4.026 in		4.026 in/4.026 in		N/A	
Pipe length (actual/design basis)	75 ft/35 ft		75 ft/35 ft		N/A	
Deposition surface (actual/design basis)	79 ft²/37 ft²		79 ft²/37 ft²		N/A	
Temperature transient - pipe inside surface	0-720 hr 120°F		0-720 hr 120°F		N/A	21
i. Drywell equipment drain (DER) vent line						
Bypass leakage rates	0-0.96 hr	0.0(2)	0-1 hr	0.0(2)	0.0	
(fraction of drywell volume per day	0.96-2 hr	2.12-5(2)	1-2 hr	2.00-5(2)	0.0	
(radwaste reactor building	0-0.96 hr	0.0	0-1 hr	0.0	0.0	
vent release)	0.96-8 hr	2.01-5	1-8 hr	1.86-5	0.0	
	8-24 hr	1.99-5	8-24 hr	1.85-5	0.0	
	24-96 hr	1.88-5	24-96 hr	1.66-5	0.0	
	96-720 hr	1.47-5	96-720 hr	1.15-5	0.0	
Iodine concentration	0-0.96 hr	0.0	0-1.0 hr	0.0	0.0	
	0.96-720 hr	0.04	1.0-720 hr	0.04	0.0	
Pipe inside diameter (actual/design basis)	2.067 in/2.067 in		2.067 in/2.067 in		N/A	
Pipe length (actual/design basis)	350 ft/54 ft		350 ft/54 ft		N/A	
Deposition surface (actual/design basis)	189 ft²/29 ft²		189 ft²/29 ft²		N/A	
Temperature transient -	0-720 hr 120°F		0-720 hr 120°F		N/A	

TABLE 15.6-13 (Cont)

	Design Basis Assumptions: Isentropic Case		Design Basis Assumptions: Isothermal Case		Realistic Basis Assumptions
pipe inside surface					
j. Drywell floor drain (DFR) line					
Bypass leakage rates (fraction of drywell volume per day) (radwaste reactor building vent release)	0-2 hr	0.0(2)	0-2 hr	0.0(2)	0.0
	0-2.23 hr	0.0	0-2.36 hr	0.0	0.0
	2.23-8 hr	5.98-5	2.36-8 hr	5.51-5	0.0
	8-24 hr	5.97-5	8-24 hr	5.55-5	0.0
	24-96 hr	5.64-5	24-96 hr	4.97-5	0.0
	96-720 hr	4.42-5	96-720 hr	3.44-5	0.0
Iodine concentration ratio	0-2.23 hr	0.0	0-2.36 hr	0.0	0.0
	2.23-720 hr	0.04	2.36-720 hr	0.04	0.0
Pipe inside diameter (actual/design basis)	6.06 in/6.065 in		6.06 in/6.065 in		N/A
Pipe length (actual/design basis)	46.8 ft/37 ft		46.8 ft/37 ft		N/A
Deposition surface (actual/design basis)	74.25 ft^2/59 ft^2		74.25 ft^2/59 ft^2		N/A
Temperature transient - pipe inside surface	0-720 hr	120°F	0-720 hr	120°F	N/A 21
k. Drywell floor drain (DFR) vent line					
Bypass leakage rates (fraction of drywell volume per day) (radwaste reactor building vent release)	0-1.94 hr	0.0(2)	0-2 hr	0.0(2)	0.0
	1.94-2 hr	3.19-5(2)	0-2.04 hr	0.0	0.0
	0-1.94 hr	0.0	2.04-8 hr	2.76-5	0.0
	1.94-8 hr	2.99-5	8-24 hr	2.77-5	0.0
	8-24 hr	2.98-5	24-96 hr	2.49-5	0.0
	24-96 hr	2.82-5	96-720 hr	1.72-5	0.0
	96-720 hr	2.21-5			0.0
Iodine concentration ratio	0-1.94 hr	0.0	0-2.04 hr	0.0	0.0
	1.94-720 hr	0.04	2.04-720 hr	0.04	0.0
Pipe inside diameter (actual/design basis)	3.068 in/3.068 in		3.068 in/3.068 in		N/A
Pipe length (actual/design basis)	114 ft/65 ft		114 ft/65 ft		N/A
Deposition surface (actual/design basis)	91.5 ft^2/52 ft^2		91.5 ft^2/52 ft^2		N/A
Temperature transient - pipe inside surface	0-720 hr	120°F	0-720 hr	120°F	N/A

TABLE 15.6-13 (Cont)

	Design Basis Assumptions				Realistic Basis Assumptions	
	Isentropic Case		Isothermal Case			
l. Drywell purge inlet line						
Bypass leakage rates (fraction of drywell volume per day) (SGTS building release)	0-2 hr	0.0[2]	0-2 hr	0.0[2]	0.0	
	0-6.01 hr	0.0	0-7.55 hr	0.0	0.0	
	6.01-8 hr	1.07-4	7.55-8 hr	8.59-5	0.0	
	8-24 hr	1.08-4	8-24 hr	8.73-5	0.0	
	24-96 hr	1.00-4	24-96 hr	7.89-5	0.0	
	96-720 hr	7.51-5	96-720 hr	5.57-5	0.0	
Iodine concentration ratio	0-6.01 hr	0.0	0-7.55 hr	0.0	0.0	
	6.01-720 hr	0.04	7.55-720 hr	0.04	N/A	
Pipe inside diameter (actual/design basis)	13.25 in/13.25 in		13.25 in/13.25 in		N/A	
Pipe lengths (actual/design basis)	45 ft/32 ft		45 ft/32 ft		N/A	
Deposition surface (actual/design basis)	156 ft^2/111 ft^2		156 ft^2/111 ft^2		N/A	
Temperature transient - pipe inside surface	0-720 hr 104°F		0-720 hr 104°F		N/A	21
m. Wetwell purge inlet line						
Bypass leakage rate (fraction of wetwell volume per day) (SGTS building release)	0-2 hr	0.0[2]	0-2 hr	0.0[2]	0.0	
	0-5.73 hr	0.0	0-7.19 hr	0.0	0.0	
	5.73-8 hr	5.50-5	7.19-8 hr	4.42-5	0.0	
	8-24 hr	5.56-5	8-24 hr	4.50-5	0.0	
	24-96 hr	5.17-5	24-96 hr	4.06-5	0.0	
	96-720 hr	3.87-5	96-720 hr	2.87-5	0.0	
Iodine concentration ratio	0-5.73 hr	0.0	0-7.2 hr	0.0		
	5.73-720 hr	0.04	7.2-720 hr	0.04		
Piping inside diameter (actual/design basis)	12.0 in/12.0 in		12.0 in/12.0 in		N/A	
Pipe length (actual/design basis)	129 ft/32 ft		129 ft/32 ft		N/A	
Deposition surface (actual/design basis)	405 ft^2/100 ft^2		405 ft^2/100 ft^2		N/A	
Temperature transient - pipe inside surface	0-720 hr 104°F		0-720 hr 104°F		N/A	

TABLE 15.6-13 (Cont)

	Design Basis Assumptions				Realistic Basis Assumptions	
	Isentropic Case		Isothermal Case			
n. Drywell purge makeup line						
Bypass leakage rate	0-0.83 hr	0.0[2]	0.0-1.03 hr	0.0[2]	0.0	
(fraction of wetwell volume	0.83-2 hr	1.64-5[2]	1.03-2.0 hr	1.34-5[2]	0.0	
per day)	0-0.83 hr	0.00	0-1.03 hr	0.0	0.0	
(SGTS building release)	0.83-8 hr	1.55-5	1.03-8 hr	1.25-5	0.0	
	8-24 hr	1.54-5	8-24 hr	1.25-5	0.0	
	24-96 hr	1.43-5	24-96 hr	1.13-5	0.0	
	96-720 hr	1.07-5	96-720 hr	7.95-6	0.0	
Iodine concentration ratio	0-0.83 hr	0.0	0-1.03 hr	0.0	0.0	
	0.83-720 hr	0.04	1.03-720 hr	0.04	0.0	
Pipe inside diameter (actual/design basis)	1.939 in/1.939 in		1.939 in/1.939 in		N/A	
Pipe lengths (actual/design basis)	129 ft/41 ft		129 ft/41 ft		N/A	
Deposition surface (actual/design basis)	65.5 ft²/20.8 ft²		65.5 ft²/20.8 ft²		N/A	
Temperature transient - pipe inside surface	0-720 hr	104°F	0-720 hr	104°F	N/A	21
o. Wetwell purge makeup line						
Bypass leakage rates	0-0.83 hr	0.0[2]	0-1.03 hr	0.0[2]	0.0	
(fraction of wetwell volume	0.83-2.0 hr	9.88-6[2]	1.03-2.0 hr	8.04-6[2]	0.0	
per day)	0-0.83 hr	0.0	0-1.03 hr	0.0	0.0	
(SGTS building release)	0.83-8 hr	9.35-6	1.03-8 hr	7.52-6	0.0	
	8-24 hr	9.26-6	8-24 hr	7.49-6	0.0	
	24-96 hr	8.61-6	24-96 hr	6.77-6	0.0	
	96-720 hr	6.44-6	96-720 hr	4.78-6	0.0	
Iodine concentration ratio	0-0.83 hr	0	0-1.03	0.0	0.0	
	0.83-720 hr	0.04	1.03-720 hr	0.04	0.0	
Pipe inside diameter (actual/design basis)	1.939 in/1.939 in		1.939 in/1.939 in		N/A	
Pipe length (actual/design basis	104 ft/41 ft		104 ft/41 ft		N/A	
Deposition surface (actual/design basis)	52.8 ft²/20.8 ft²		52.8 ft²/20.8 ft²		N/A	
Temperature transient - pipe inside surface	0-720 hr	104°F	0-720 hr	104°F	N/A	

TABLE 15.6-13 (Cont)

		Design Basis Assumptions(5)	Realistic Basis Assumptions	
p.	Containment leakage rate (main stack release)	1.1% per day of primary containment volume for duration of accident	1.1% per day of primary containment volume for duration of accident	21
q.	Transfer incore probe leakage rate (main stack release)	0.21% per day of primary containment volume for duration of accident	N/A	
r.	Reactor building leak rate (main stack release)	3,500 cfm through standby gas treatment (SGTS)	3,500 cfm through SGTS	
s.	Percentage mixing in reactor building air	50%	50%	21
t.	Reactor building pressurization time (radwaste/reactor building vent release)	90 sec	90 sec	
u.	SGTS halogen filtration efficiency	99%	99%	
v.	ESF leakage to reactor building (main stack release)			
	(1) Leak rate	1 gpm	0.0	
	(2) Iodine partition factor (air/water)	0.1	0.0	
3.	All other pertinent data			
a.	Primary containment			
	(1) Drywell free air volume	2.85+5 ft^3	N/A	
	(2) Primary containment free air volume	4.73+5 ft^3	4.73+5 ft^3	
	(3) Suppression pool volume	1.45+5 ft^3	N/A	
b.	Reactor building			
	(1) Free air volume	3.88+6 ft^3	3.88+6 ft^3	

TABLE 15.6-13 (Cont)

		Design Basis Assumptions(5)	Realistic Basis Assumptions
c.	Control room		
	(1) Free air volume	2.096+5 ft³	3.81+05 ft³
	(2) Intake rate	1.00+3 cfm	1.50+3 cfm
	(3) Recirculation rate	7.50+2 cfm	7.50+2 cfm
	(4) Intake/recirculation halogen filtration efficiency	99%	99%
4.	Dispersion data (s/m³)		
a.	Stack		
	0-2 hr EAB	2.97-5	1.16-7
	0-8 hr LPZ	1.03-5	4.32-7
	8-24 hr LPZ	8.85-7	3.21-7
	24-96 hr LPZ	3.66-7	1.69-7
	96-720 hr LPZ	1.03-7	6.73-8
	0-8 hr control room	8.10-5	8.10-5
	8-24 hr control room	2.44-8	2.44-8
	24-96 hr control room	2.10-8	2.10-8
	96-720 hr control room	1.69-8	1.69-8
b.	Radwaste/reactor building vent(4)		
	0-2 hr EAB	1.90-4	2.19-5
	0-8 hr LPZ	1.78-5	6.48-6
	8-24 hr LPZ	1.19-5	N/A
	24-96 hr LPZ	4.93-6	N/A
	96-720 hr LPZ	1.40-6	N/A
	0-8 hr control room	2.13-4	2.13-4
	8-24 hr control room	1.66-4	N/A
	24-96 hr control room	9.88-5	N/A
	96-720 hr control room	4.70-5	N/A
c.	Main steam tunnel		
	0-2 hr EAB	1.90-4	N/A
	0-8 hr LPZ	1.78-5	N/A
	8-24 hr LPZ	1.19-5	N/A
	24-96 hr LPZ	4.93-6	N/A
	96-720 hr LPZ	1.40-6	N/A
	0-8 hr control room	1.29-3	N/A
	8-24 hr control room	9.90-4	N/A

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TABLE 15.6-13 (Cont)

		Design Basis Assumptions[5]	Realistic Basis Assumptions
	24-96 hr control room	3.37-4	N/A
	96-720 hr control room	9.92-5	N/A
d.	Radwaste tunnel (PASS area)		
	0-2 hr EAB	1.90-4	N/A
	0-8 hr LPZ	1.82-5	N/A
	8-24 hr LPZ	1.21-5	N/A
	24-96 hr LPZ	5.02-6	N/A
	96-720 hr LPZ	1.42-6	N/A
	0-8 hr control room	1.83-4	N/A
	8-24 hr control room	1.41-4	N/A
	24-96 hr control room	4.81-5	N/A
	96-720 hr control room	1.42-5	N/A
e:	SGTS building		
	0-2 hr EAB	1.90-4	N/A
	0-8 hr LPZ	1.78-5	N/A
	8-24 hr LPZ	1.19-5	N/A
	24-96 hr LPZ	4.93-6	N/A
	96-720 hr LPZ	1.40-6	N/A
	0-8 hr control room	1.75-3	N/A
	8-24 hr control room	1.34-3	N/A
	24-96 hr control room	4.57-4	N/A
	96-720 hr control room	1.35-4	N/A

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NOTE: 5.92-4 = 5.92×10^{-4}

(1) Dragon 4 Code, Dose and Radioactivity from Nuclear Facility Gaseous Outflows, NU-115, Version 4, Level 1, April 1982.

(2) EAB dose analysis only.

(3) See Section 6.2.3.

(4) These values are used for the 90-sec release directly to the environment during the period when reactor building pressure is above -0.25 in W.G.

(5) Common to isentropic and isothermal cases. 21

TABLE 15.6-14

LOSS-OF-COOLANT ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY AVAILABLE FOR RELEASE FROM PRIMARY
CONTAINMENT AT t = 0

(Ci)

Isotope	Activity
I-129	7.60-1
I-131	2.30+7
I-132	3.35+7
I-133	4.80+7
I-134	5.28+7
I-135	4.53+7
I-136	2.19+7
Br-83	2.73+6
Br-84	4.85+6
Br-85	5.80+6
Br-87	9.78+6
Kr-83m	1.10+7
Kr-85m	2.35+7
Kr-85	1.05+6
Kr-87	4.50+7
Kr-88	6.38+7
Kr-89	7.95+7
Xe-131m	5.51+5
Xe-133m	8.06+6
Xe-133	1.93+8
Xe-135m	3.63+7
Xe-135	2.49+7
Xe-137	1.69+8
Xe-138	1.61+8

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TABLE 15.6-15a

LOSS-OF-COOLANT ACCIDENT (DESIGN BASIS ANALYSIS) ACTIVITY RELEASE TO ENVIRONMENT* (ISOTHERMAL APPROACH)

(Ci)

Isotope	Activity	
I-129		
I-131		
I-132		
I-133		
I-134		
I-135		
I-136		
Br-83		
Br-84		
Br-85		
Br-87		
Kr-83m	(LATER)	21
Kr-85m		
Kr-85		
Kr-87		
Kr-88		
Kr-89		
Xe-131m		
Xe-133m		
Xe-133		
Xe-135m		
Xe-135		
Xe-137		
Xe-138		
Total		

*Total release for 30 days.

TABLE 15.6-15b

LOSS-OF-COOLANT ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT* (ISENTROPIC APPROACH)

(Ci)

Isotope	Activity
I-129	
I-131	
I-132	
I-133	
I-134	
I-135	
I-136	
Br-83	
Br-84	
Br-85	(LATER)
Br-87	
Kr-83m	
Kr-85m	
Kr-85	
Kr-87	
Kr-88	
Kr-89	
Xe-131m	
Xe-133m	
Xe-133	
Xe-135m	
Xe-135	
Xe-137	
Xe-138	
Total	

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*Total release for 30 days.

TABLE 15.6-16a

LOSS-OF-COOLANT ACCIDENT (DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS (ISOTHERMAL APPROACH)

	Whole-Body Dose (Rem)	Thyroid Dose (Rem)	Beta Dose (Rem)	
Exclusion area (2 hr)				21
Low-population zone (30 day)		(LATER)		
Control room (30 day)				

TABLE 15.6-16b

LOSS-OF-COOLANT ACCIDENT (DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS (ISENTROPIC APPROACH)

	Whole-Body Dose (rem)	Thyroid Dose (rem)	Beta Dose (rem)
Exclusion area (2 hr)			
Low-population zone (30-day)		(LATER)	
Control room (30-day)			

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TABLE 15.6-17

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS)
REACTOR COOLANT IODINE CONCENTRATIONS

Isotope	Design Reactor Coolant (uCi/gm)	Normalized Reactor Coolant (uCi/gm)
I-131	1.3-2	6.1-1
I-132	2.2-1	1.0+1
I-133	1.6-1	7.5+1
I-134	3.9-1	1.8+1
I-135	1.7-1	8.0+0

NOTE: $1.3\text{-}2 = 1.3 \times 10^{-2}$ 13

*Design reactor coolant iodines normalized to maximum technical specification limit of 4 uCi/gm.

TABLE 15.6-18

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY AIRBORNE IN CONTAINMENT DUE TO IODINE SPIKING

(Ci)

Isotope	Initial Airborne Activity	Airborne Activity Available for Release
I-131	8.33+4	1.67+4
I-132	1.41+6	2.82+5
I-133	1.02+6	2.04+5
I-134	2.50+6	5.00+5
I-135	1.90+6	2.18+5

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NOTE: 8.33+4 = 8.33×10^4 13

TABLE 15.6-19

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS) ACTIVITY RELEASE TO ENVIRONMENT

(Ci)

Isotope	Activity Released	
I-131	1.79+1	
I-132	4.21+0	
I-133	2.70+1	
I-134	2.87+0	
I-135	9.59+0	13
Xe-131m*	6.85+1	
Xe-133m*	4.25+2	
Xe-133*	1.28+4	
Xe-135m*	8.72+3	
Xe-135*	5.27+3	

NOTE: $1.79+1 = 1.79 \times 10^{1}$ 13

*The xenon isotopes were produced by the decay of the iodines released.

[illegible]

TABLE 15.6-20

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole-Body Dose (Rem)	Thyroid Dose (Rem)	Beta Dose (Rem)	
Exclusion area (2 hr)	1.53-4	1.41-2	3.54-5	
Low-population zone (30 d)	4.65-4	5.83-3	2.14-4	13
Control room (30 d)	3.75-4	1.48-3	4.42-3	

NOTE: 1.53-4 = 1.53×10^{-4} 13