SCE&G
A SCANA Company

South Carolina Electric & Gas Company
P.O. Box 88
Jenkinsville, SC 29065
(803) 345-4040

Ollie S. Bradham
Vice President
Nuclear Operations

September 30, 1988

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Virgil C. Summer Nuclear Station
Docket No. 50/395
Operating License No. NPF-12
Technical Specification Change
VANTAGE 5 Fuel

Gentlemen:

On May 20, 1988, South Carolina Electric & Gas Company (SCE&G) requested a revision to the Technical Specifications for the Virgil C. Summer Nuclear Station (VCSNS) in support of refueling and operating with VANTAGE 5 fuel. This submittal contained a Radiological Impact Assessment in which it concluded that the transition from current fuel to VANTAGE 5, with its extended burnup characteristics, would have a small impact on thyroid and whole body doses.

To supplement the previous Radiological Impact Assessment, SCE&G has performed a detailed evaluation of the environment consequences for the Final Safety Analysis Report (FSAR) Chapter 15 accidents impacted by the fuel change. This letter forwards the results of the dose calculations. This evaluation used the reactor coolant and core source terms for VANTAGE 5 fuel, previously supplied on August 31, 1988, in combination with the current NRC accepted methodology for dose evaluations as described in the FSAR.

Tables 1, 2 and 3 present the limiting analysis results from the FSAR and this evaluation for those transients impacted by the fuel change. Due to the revised source term methodology, the transition to VANTAGE 5 fuel generally results in a small decrease in gamma and beta doses and a small increase in thyroid doses. In all cases, the dose results are well within applicable NRC acceptance criteria.

In addition to the limiting analysis results presented in Tables 1, 2 and 3 more realistic calculations have been performed consistent with the current dose consequence presentation given in the VCSNS FSAR. These results are summarized in the FSAR markups attached.

A001
1/1

This letter completes the SCE&G assessment of environmental consequences from Chapter 15 events with VANTAGE 5 fuel. If there should be any questions, please do not hesitate to call.

Very truly yours,

O. S. Bradham

MDB/OSB:lcd
Attachments

c: D. A. Nauman/J. G. Connelly, Jr./O. W. Dixon, Jr./T. C. Nichols, Jr.
E. C. Roberts
W. A. Williams, Jr.
J. N. Grace
J. J. Hayes, Jr.
General Managers
C. A. Price/R. M. Campbell, Jr.
R. B. Clary
K. E. Nodland
J. C. Snelson
G. O. Percival
R. L. Prevatte
J. B Knotts, Jr.
H. G. Shealy
NSRC
RTS (TSP 880013)
NPCF
File (813.20)

TABLE 1
CURRENT CHAPTER 15 DOSES FOR LOPAR FUEL

FSAR SECTION	TITLE	SITE BOUNDARY RESULTS (Rem)			LOW POPULATION ZONE RESULTS (Rem)		
		Gamma	Beta	Thyroid	Gamma	Beta	Thyroid
15.2.9	Loss of Offsite Power	9.83E-4	2.04E-3	2.77E-2	2.27E-4	4.71E-4	3.82E-3
15.3.7	Instrument Line Break	3.18E-2	3.66E-2	6.72E-1	1.85E-3	2.13E-3	3.90E-2
15.4.1	Loss of Coolant Accident	4.45	2.99	1.52E+2	6.87E-1	4.04E-1	2.5E+1
15.4.2	Steam Line Break	1.46E-2	8.75E-3	9.70	3.26E-3	2.53E-3	2.05
15.4.3	Steam Generator Tube Rupture	1.40E-1	5.99E-2	3.67E-1	3.54E-2	1.52E-2	2.89E-1
15.4.4	Locked Rotor	7.87E-1	1.11	6.99	1.83E-1	2.57E-1	1.62
15.4.5	Fuel Handling Accident						
	Inside Containment	1.22	1.62	1.35E+2	N/A	N/A	N/A
	Outside Containment	1.22	1.62	6.75			
15.4.6	Rod Ejection	1.82E-1	9.98E-2	5.01E+1	2.64E-2	1.47E-2	1.39E+1

TABLE 2
REVISED CHAPTER 15 DOSES FOR VANTAGE 5 FUEL

FSAR SECTION	TITLE	SITE BOUNDARY RESULTS (Rem)			LOW POPULATION ZONE RESULTS (Rem)		
		Gamma	Beta	Thyroid	Gamma	Beta	Thyroid
15.2.9	Loss of Offsite Power	6.65E-4	1.30E-3	2.87E-2	1.53E-4	3.00E-4	3.95E-3
15.3.7	Instrument Line Break	2.85E-2	3.20E-2	7.60E-1	1.66E-3	1.86E-3	4.41E-2
15.4.1	Loss of Coolant Accident	2.78	2.16	1.74E+2	3.57E-1	3.07E-1	2.89E+1
15.4.2	Steam Line Break	1.23E-2	8.04E-3	1.27E+1	2.70E-3	1.78E-3	2.76
15.4.3	Steam Generator Tube Rupture	1.88E-1	2.17E-1	4.13E-1	4.71E-2	5.46E-2	3.19E-1
15.4.4	Locked Rotor	5.63E-1	8.27E-1	8.01	1.31E-1	1.92E-1	1.85
15.4.5	Fuel Handling Accident						
	Inside Containment	1.40	1.65	1.53E+2	N/A	N/A	N/A
	Outside Containment	1.40	1.65	7.66			
15.4.6	Rod Ejection	1.56E-1	7.20E-2	5.28+1	2.36E-2	1.15E-2	1.46E+1

Table 3
CONTROL ROOM DOSES FOLLOWING A LOCA

	Doses (Rem)	
	Current FSAR	VANTAGE 5
Gamma	2.26	1.70
Beta	9.29	6.30
Thyroid	30.00	30.00

ATTACHMENT

This attachment contains marked-up pages for the environmental consequence sections of Chapter 15 of the FSAR as summarized below.

ACCIDENT	ITEM	FSAR PAGE NO
Loss of Offsite Power	Section 15.2.9.4 Table 15.2-5 to 15.2-8	15.2-33 15.2-55 to 15.2-58
Instrument Line Break	Section 15.3.7 Table 15.3-6 Table 15.3-7	15.3-14 15.3-22 15.3-23
Loss of Coolant Accident	Section 15.4.1.4.3 Tables 15.4-10 to 15.4-18	15.4-14 15.4-75 to 15.4-79, 15.4-81 15.4-83 to 15.4-85
Steam Line Break	Section 15.4.2.1.4 Tables 15.4-23 to 15.4-27 Figures 15.4-61 to 15.4-63	15.4-22 15.4-91 to 15.4-95 -----
Steam Generator Tube Rupture	Section 15.4.3.4 Tables 15.4-29 to 15.4-33 Figures 15.4-66 to 15.4-68	15.4-34 15.4-97 to 15.4-101 -----
Locked Rotor	Section 15.4.4.4 Tables 15.4-34a to 15.4-34e Figures 15.4-77a to 15.4-77c	15.4-39, 15.4-40 15.4-103, 15.4-105 to 15.4-108 -----
Fuel Handling Accident	Section 15.4.5.1 Tables 15.4-35 to 15.4-37 Tables 15.4-39 to 15.4-41 Table 15.4-50 Table 15.4-51	15.4-41, 15.4-47, 15.4-48 15.4-109 to 15.4-111 15.4-114 to 15.4-116 15.4-127 15.4-128
Rod Ejection	Section 15.4.6.4.4 Tables 15.4-44 to 15.4-46	15.4-61 15.4-121 to 15.4-123

6. Defective fuel is equal to one percent.

7. No noble gas is dissolved in steam generator water.

8. The iodine partition factor in the steam generators is 0.01.

9. During the postulated accident, iodine carryover from the primary side is uniformly mixed with the water in the steam generators and is diluted by the incoming feedwater.

10. The steam release for cooling down the plant is equally contributed by all steam generators.

11. The 0-2 and 2-8 hour atmospheric diffusion factors, given in Appendix 15A, and the 0-8 hour breathing rate of 3.47×10^{-4} m^3/sec are applicable.

12. Dose model used to evaluate the environmental consequence of this accident is given in Appendix 15A.

Steam releases to the atmosphere for the loss of offsite power are given by Table 15.2-5.

Using the previously listed assumptions, isotopic releases to the environment are summarized by Tables 15.2-7 and 15.2-8 for realistic and conservative assumptions, respectively.

Gamma, beta, and thyroid doses in the first two hours of the loss of offsite power to plant auxiliaries for the realistic analysis at the site boundary are ~~2.38×10^{-9}~~ 7.15×10^{-9} rem, ~~3.24×10^{-9}~~ 1.40×10^{-8} rem and ~~9.98×10^{-8}~~ 1.19×10^{-7} rem, respectively. The corresponding doses at the low population zone are ~~1.05×10^{-9}~~ 3.19×10^{-9} rem, ~~1.44×10^{-9}~~ 6.33×10^{-9} rem and ~~2.98×10^{-8}~~ 3.58×10^{-8} rem respectively.

The gamma, beta, and thyroid doses in the first two hours of the loss of offsite power to plant auxiliaries for the conservative analysis at the site boundary are ~~9.83×10^{-4}~~ 6.65×10^{-4} rem, ~~2.04×10^{-3}~~ 1.30×10^{-3} rem and ~~2.77×10^{-2}~~ 2.87×10^{-2} rem, respectively. Corresponding doses at the low population zone are ~~2.27×10^{-4}~~ 1.53×10^{-4} rem, ~~4.71×10^{-4}~~ 3.00×10^{-4} rem and ~~3.82×10^{-3}~~ 3.95×10^{-3} rem, respectively, for the duration of the accident.

The doses for this accident are well within the limits defined in 10 CFR 100 (25 Rem, whole body and 300 Rem, thyroid).

15.2.10 EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS

15.2.10.1 Identification of Causes and Accident Description

Addition of excessive feedwater causes an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the reactor coolant system (RCS). The overpower - overtemperature protection (neutron overpower, overtemperature and overpower ΔT trips) prevents any power increase which could lead to a DNBR less than 1.30.

TABLE 15.2-5

PARAMETERS USED IN LOSS OF OFFSITE POWER ANALYSIS

	Realistic Analysis	Conservative Analysis
Core thermal power	29[illegible] MWt	2900 MWt
Steam generator tube leak rate prior to and during accident	100 lbs/day(1)	1.0 gpm
Fuel defects	0.12%(1)	1%
Iodine partition factor in steam generators prior to and during accident	0.01	0.01
Blowdown rate per steam generator prior to accident	~~[illegible]~~ 41 gpm	15 gpm
Duration of plant cooldown by secondary system after accident	8 hr	8 hr
Steam release from three steam generators	447,900 lbs (0-2 hr) 757,700 lbs (2-8 hr)	629,800 lbs (0-2 hr) 757,700 lbs (2-8 hr)
Feedwater flow to three steam generators	629,800 lbs (0-2 hr) 841,800 lbs (2-8 hr)	429,800 lbs (0-2 hr) 841,800 lbs (2-8 hr)
Meteorology	Annual average	Accident

(1) American National Standards Institute, "Source Term Specification," ANSI N237, Revision 2.

TABLE 15.2-6

SECONDARY SYSTEM EQUILIBRIUM CONCENTRATION[1]

Isotopes	Lp=0.01 gpm[2]	Lp=0.1 gpm	Lp=1.0 gpm
I-131	2.65 ~~2.41~~x10^{-1} μCi/lb	2.65 ~~2.41~~x10^{0} μCi/lb	2.65 ~~2.41~~ x 10^{1} μCi/lb
I-132	5.24 ~~1.99~~x10^{-2}	5.24 ~~1.99~~x10^{-1}	5.24 ~~1.99~~ x 10^{0}
I-133	2.95 ~~2.97~~x10^{-1}	2.95 ~~2.97~~x10^{0}	2.95 ~~2.97~~ x 10^{1}
I-134	4.77 ~~5.47~~x10^{-3}	4.77 ~~5.47~~x10^{-2}	4.77 ~~5.47~~ x 10^{-1}
I-135	9.41 ~~1.01~~x$10^{-~~1~~2}$	9.41 ~~1.01~~x$10^{~~0~~-1}$	9.41 ~~1.01~~ x $10^{~~1~~0}$

(1) Using primary coolant parameters and activities in Tables 11.1-1 and 11.1-2.

(2) Lp = primary to secondary leakage rate.

TABLE 15.2-7

LOSS OF OFFSITE POWER ACCIDENT
ISOTOPIC RELEASE TO ENVIRONMENT
REALISTIC ANALYSIS

Isotope	Activity Released to Environment by Accident (Ci) (0-2 hr)	(2-8 hr)
I-131	4.80 ~~4.24~~ x 10^{-5}	7.91 ~~6.95~~ x 10^{-5}
I-132	2.93 ~~2.49~~ x 10^{-5}	5.82 ~~3.76~~ x 10^{-5}
I-133	7.23 ~~5.37~~ x 10^{-5}	1.23 ~~2.17~~ x 10^{-4}
I-134	4.94 ~~3.38~~ x 10^{-6}	1.13 ~~1.68~~ x 10^{-5}
I-135	3.44 ~~2.32~~ x 10^{-5}	6.18 ~~4.15~~ x 10^{-5}
Xe-131m	1.00 ~~6.81~~ x 10^{-3}	3.00 ~~2.06~~ x 10^{-3}
Xe-133	1.18 ~~1.95~~ x 10^{-1}	3.55 ~~5.78~~ x 10^{-1}
Xe-133m	7.72 ~~3.79~~ x 10^{-3}	2.32 ~~1.14~~ x 10^{-2}
Xe-135	3.64 ~~1.18~~ x 10^{-3}	1.09 ~~3.55~~ x 10^{-2}
Xe-135m	2.50 ~~5.67~~ x 10^{-4}	7.50 ~~1.70~~ x 10^{-4}
Xe-137	0 ~~3.79 x 10^{-5}~~	0 ~~1.14 x 10^{-4}~~
Xe-138	3.26 ~~1.89~~ x 10^{-4}	9.78 ~~5.68~~ x 10^{-4}
Kr-83m	0 ~~8.67 x 10^{-5}~~	0 ~~2.60 x 10^{-4}~~
Kr-85	4.14 ~~2.56~~ x 10^{-3}	1.24 ~~2.61~~ x 10^{-2}
Kr-85m	9.10 ~~4.18~~ x 10^{-4}	2.73 ~~1.25~~ x 10^{-3}
Kr-87	5.44 ~~2.46~~ x 10^{-4}	1.63 ~~7.38~~ x 10^{-3}
Kr-88	1.63 ~~2.95~~ x 10^{-3}	4.90 ~~2.38~~ x 10^{-3}
Kr-89	0 ~~2.19 x 10^{-5}~~	0 ~~6.58 x 10^{-5}~~

TABLE 15.2-8

LOSS OF OFFSITE POWER ACCIDENT
ISOTOPIC RELEASE TO ENVIRONMENT
CONSERVATIVE ANALYSIS(1)

Isotope	Activity Released to Environment by Accident (Ci) (0-2 hr)	Activity Released to Environment by Accident (Ci) (2-8 hr)
I-131	1.01 ~~[illegible]~~ $\times 10^{-1}$	1.35 ~~[illegible]~~ $\times 10^{-1}$
I-132	2.78 ~~[illegible]~~ $\times 10^{-2}$	5.00 ~~[illegible]~~ $\times 10^{-2}$
I-133	1.17 ~~[illegible]~~ $\times 10^{-1}$	1.65 ~~[illegible]~~ $\times 10^{-1}$
I-134	3.87 ~~[illegible]~~ $\times 10^{-3}$	8.58 ~~[illegible]~~ $\times 10^{-3}$
I-135	4.12 ~~[illegible]~~ $\times 10^{-2}$	6.38 ~~[illegible]~~ $\times 10^{-2}$
Xe-131m	7.29 ~~[illegible]~~ $\times 10^{-1}$	2.19 ~~[illegible]~~ $\times 10^{0}$
Xe-133	8.62 ~~[illegible]~~ $\times 10^{1}$	2.59 ~~[illegible]~~ $\times 10^{2}$
Xe-133m	5.63 ~~[illegible]~~ $\times 10^{0}$	1.69 ~~[illegible]~~ $\times 10^{1}$
Xe-135	2.65 ~~[illegible]~~ $\times 10^{0}$	7.96 ~~[illegible]~~ $\times 10^{0}$
Xe-135m	1.82 ~~[illegible]~~ $\times 10^{-1}$	5.50 ~~[illegible]~~ $\times 10^{-1}$
Xe-138	2.39 ~~[illegible]~~ $\times 10^{-1}$	7.20 ~~[illegible]~~ $\times 10^{-1}$
Kr-83m	---	---
Kr-85	3.02 ~~[illegible]~~ $\times 10^{0}$	9.05 ~~[illegible]~~ $\times 10^{0}$
Kr-85m	6.63 ~~[illegible]~~ $\times 10^{-1}$	1.99 ~~[illegible]~~ $\times 10^{0}$
Kr-87	3.98 ~~[illegible]~~ $\times 10^{-1}$	1.19 ~~[illegible]~~ $\times 10^{0}$
Kr-88	1.19 ~~[illegible]~~ $\times 10^{0}$	3.57 ~~[illegible]~~ $\times 10^{0}$
Kr-89	---	---

(1) Primary to secondary leakage = 1.0 gpm

3. The iodine partition factor for activity released from the break is 0.1.

4. The concentration of radioactive nuclides in the reactor coolant is listed in Table 11.1-2 for the conservative case and in Table 11.1-5 for the realistic case.

Using the previously listed assumptions, isotopic releases to the environment are determined to be those listed in Tables 15.3-6 and 15.3-7 for the realistic and conservative cases, respectively.

Gamma, beta and thyroid doses at the site boundary for the realistic case are 7.30×10^{-6} Rem, 6.61×10^{-6} Rem and 5.90×10^{-4} Rem, respectively. Corresponding doses at the low population zone are 8.20×10^{-7} Rem, 7.65×10^{-7} Rem and 6.63×10^{-5} Rem, respectively.

Gamma, beta and thyroid doses at the site boundary for the conservative case are 3.18×10^{-2} Rem, 3.66×10^{-2} Rem and 6.72×10^{-1} Rem, respectively. Corresponding doses at the low population zone are 1.85×10^{-3} Rem, 2.13×10^{-3} Rem and 3.90×10^{-2} Rem, respectively.

Doses resulting from this accident are well within the limits defined in 10 CFR 20 (25 Rem whole body and 300 Rem thyroid).

15.3.8 REFERENCES

1. Esposito, V. J., Kesavan, K. and Maul B. A., "WFLASH - A FORTRAN-IV Computer Program for Simulation of Transients in a Multi-Loop PWR," WCAP-8200, Revision 2 (Proprietary) and WCAP-8261, Revision 1 (Non-Proprietary), July, 1974.

2. Porsching, T. A., Murphy, J. H., Redfield, J. A., and Davis V. C., "FLASH-4: A Fully Implicit FORTRAN-IV Program for the Digital Simulation of Transients in a Reactor Plant," WAPD-TM-84; Bettis Atomic Power Laboratory, March, 1969.

3. Bordelon, F. M., et al., "LOCTA-IV Program: Loss of Coolant Transient Analysis," WCAP-8301 (Proprietary) and WCAP-8305 (Non-Proprietary), June, 1974.

4. Hellman, J. M., "Fuel Densification Experimental Results and Model for Reactor Application," WCAP-8218-P-A (Proprietary) and WCAP-8219-A (Non-Proprietary), March, 1975.

5. Altamore, S. and Barry, R. F., "The TURTLE 24.0 Diffusion Depletion Code," WCAP-7213-P-A (Proprietary) and WCAP-7758-A (Non-Proprietary), January, 1975.

6. Barry, R. F., "LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094," WCAP-3269-26, September, 1963.

TABLE 15.3-6

CHEMICAL AND VOLUME CONTROL SYSTEM
LETDOWN LINE RUPTURE - ISOTOPIC RELEASE
TO THE ENVIRONMENT - REALISTIC CASE

Isotope	Activity Released (Ci)
I-131	2.78 ~~2.46~~ x 10^{-1}
I-132	2.88 ~~9.93~~ x ~~10^{-2}~~ 10^{-1}
I-133	4.57 ~~3.48~~ x 10^{-1}
I-134	6.67 ~~4.47~~ x 10^{-2}
I-135	2.59 ~~1.74~~ x 10^{-1}
Xe-131m	2.18 ~~1.49~~ x ~~10^{-1}~~ 10^{0}
Xe-133	2.59 ~~4.23~~ x ~~10^{1}~~ 10^{2}
Xe-133m	1.69 ~~8.28~~ x ~~10^{-1}~~ 10^{1}
Xe-135	7.96 ~~2.57~~ x 10^{0}
Xe-135m	5.47 ~~1.24~~ x 10^{-1}
~~Xe-137~~	~~8.28 x 10^{-2}~~
Xe-138	7.16 ~~4.14~~ x 10^{-1}
~~Kr-83m~~	~~1.94 x 10^{-1}~~
Kr-85	9.06 ~~5.55~~ x ~~10^{-2}~~ 10^{0}
Kr-85m	1.99 ~~9.12~~ x ~~10^{-1}~~ 10^{0}
Kr-87	1.19 ~~5.37~~ x ~~10^{-1}~~ 10^{0}
Kr-88	3.58 ~~1.74~~ x 10^{0}
~~Kr-89~~	~~4.80 x 10^{-2}~~

TABLE 15.3-7

CHEMICAL AND VOLUME CONTROL SYSTEM
LETDOWN LINE RUPTURE - ISOTOPIC RELEASE
TO THE ENVIRONMENT - CONSERVATIVE CASE

Isotope	Activity Released (Ci)
I-131	2.32 ~~2.07~~ x 10^{0}
I-132	2.40 ~~2.71~~ x 10~~$^{-1}$~~ ^{0}
I-133	3.81 ~~3.48~~ x 10^{0}
I-134	5.56 ~~4.80~~ x 10^{-1}
I-135	2.16 ~~1.91~~ x 10^{0}
Xe-131m	1.82 ~~1.66~~ x 10^{1}
Xe-133	2.16 ~~2.58~~ x 10^{3}
Xe-133m	1.41 ~~3.30~~ x 10~~^{1}~~ ^{2}
Xe-135	6.63 ~~6.54~~ x 10^{1}
Xe-135m	4.56 ~~4.14~~ x 10^{0}
Xe-138	5.97 ~~6.63~~ x 10^{0}
Kr-85	7.55 ~~6.07~~ x 10^{1}
Kr-85m	1.66 ~~1.91~~ x 10^{1}
Kr-[illegible]	9.95 ~~1.08~~ x 10~~^{1}~~ ^{0}
Kr-88	2.98 ~~3.24~~ x 10^{1}

will be bypassed around the Control Room Emergency Filter Plenum. For the emergency mode of operation, the control room recirculation air flow will be routed through the Control Room Emergency Filter Plenum.

In the purge mode of operation, the system will supply 100 percent outside air to the control room. The purge air inlet cover plate will be removed from the outside air intake plenum and the system relief dampers to allow an outside air flow of 21,270 cfm into the control room. The recirculation flow will be terminated for the duration of the purge mode.

Each control room air intake is provided with two isolation valves in series. One of the two valves restricts the outside air flow to a maximum of 1000 cfm flow for both normal and emergency modes. Upon receipt of an engineered safety features actuation signal, the control room ventilation system switches to the emergency mode.

For the purpose of this analysis, the maximum allowable air intake value was determined for the limiting total integrated dose to control room personnel. The results are presented in Table 15.4-18. The maximum allowable air intake value was determined to be 2760 cfm. The system operating flow of 1000 cfm provides adequate margin for the protection of control room personnel as specified under General Design Criterion 19 of 10 CFR 50, Appendix A.

2423

15.4.2 MAJOR SECONDARY SYSTEM PIPE RUPTURE

15.4.2.1 Major Rupture of a Main Steam Line

15.4.2.1.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam line would result in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam line rupture is a potential problem mainly because of the high power peaking factors which exist assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid injection delivered by the safety injection system.

The limiting main steam line break was selected based upon the sensitivity studies performed in "Reactor Core Response to Excessive Secondary Steam Releases," WCAP-9226, January, 1978.

The analysis of a main steam line rupture is performed to demonstrate that the following criterion is satisfied:

TABLE 15.4-10

RADIOACTIVITY RELEASE FROM THE RECIRCULATION LOOPS

(Emergency Core Cooling and Reactor Building Spray Systems)

Isotope	Release Rate curies/min
I-131	1.9 ~~1.7~~ x 10^{-2}
I-132	2.7 ~~2.6~~ x 10^{-2}
I-133	3.8 ~~3.9~~ x 10^{-2}
I-134	4.1 ~~4.5~~ x 10^{-2}
I-135	3.5 x 10^{-2}

This table is based on 50 percent of core iodine inventory in the sump, 71,454 ft^3 of water in the sump, 5860 cc/hr leakage and a 100 decontamination factor of iodine between liquid and airborne phases.

TABLE 15.4-11

PRIMARY COOLANT INVENTORY

Isotope	Activity (Ci)
I-131	5.29 ~~6.75~~ x 10^2
I-132	5.48 ~~1.76~~ x 10^2
I-133	8.69 ~~2.94~~ x 10^2
I-134	1.27 ~~1.10~~ x 10^2
I-135	4.91 ~~4.35~~ x 10^2
Kr-85	1.73 ~~1.57~~ x 10^3
Kr-85m	3.78 ~~4.35~~ x 10^2
Kr-87	2.27 ~~2.46~~ x 10^2
Kr-88	6.80 ~~7.37~~ x 10^2
Xe-131m	4.16 ~~3.78~~ x 10^2
Xe-133	4.91 ~~5.90~~ x 10^4
Xe-133m	3.21 ~~2.56~~ x ~~10^2~~ 10^3
Xe-135	1.51 ~~1.49~~ x 10^3
Xe-135m	1.04 ~~9.45~~ x ~~10^3~~ 10^2
Xe-138	1.36 ~~1.51~~ x 10^2

TABLE 15.4-12

FISSION PRODUCT ACTIVITY AIRBORNE IN CONTAINMENT
REALISTIC CASE - LOSS OF COOLANT ACCIDENT

Isotope	Activity (Ci)
I-131	2.65 ~~2.37~~ x 10^2
I-132	2.74 ~~8.80~~ x 10~~^{1}~~ ^{2}
I-133	4.35 ~~3.97~~ x 10^2
I-134	6.35 ~~5.50~~ x 10^1
I-135	2.45 ~~2.18~~ x 10^2
Kr-85	1.72 ~~1.37~~ x 10^3
Kr-85m	3.78 ~~4.35~~ x 10^2
Kr-87	2.27 ~~2.46~~ x 10^2
Kr-88	6.80 ~~7.37~~ x 10^2
Xe-131m	4.16 ~~3.78~~ x 10^2
Xe-133	4.91 ~~5.90~~ x 10^4
Xe-133m	3.21 ~~7.36~~ x 10~~[illegible]~~ ^{3}
Xe-135	1.51 ~~1.49~~ x 10^3
Xe-135m	1.04 ~~9.45~~ x 10~~^{1}~~ ^{2}
Xe-138	1.36 ~~1.51~~ x 10^2

TABLE 15.4-13

FISSION PRODUCT ACTIVITY AIRBORNE IN CONTAINMENT

CONSERVATIVE CASE - LOSS OF COOLANT ACCIDENT

Isotope	Activity (Ci)
I-131	5.46 ~~3.84~~ x 10^5
I-132	8.40 ~~6.40~~ x 10^4
I-133	3.63 ~~2.84~~ x 10^5
I-134	8.08 ~~6.85~~ x 10^4
I-135	1.90 ~~1.47~~ x 10^5
Kr-83m	1.27 ~~1.44~~ x 10^4
Kr-85	1.36 ~~1.59~~ x 10^5
Kr-85m	4.20 ~~5.21~~ x 10^4
Kr-87	4.07 ~~5.40~~ x 10^4
Kr-88	9.01 ~~1.14~~ x $10^{\cancel{5}\,4}$
Kr-89	1.50 ~~2.04~~ x 10^4
Xe-131m	9.18 ~~2.08~~ x 10^3
Xe-133	1.65 ~~1.44~~ x 10^6
Xe-133m	1.70 ~~2.39~~ x $10^{\cancel{4}\,5}$
Xe-135	1.02 ~~1.07~~ x 10^5
Xe-135m	1.61 ~~1.76~~ x $10^{\cancel{4}\,5}$
Xe-138	7.02 ~~6.03~~ x $10^{\cancel{4}\,5}$

TABLE 15.4-14

FISSION PRODUCT ACTIVITY AIRBORNE IN CONTAINMENT
REGULATORY GUIDE 1.4 CASE - LOSS OF COOLANT ACCIDENT

Isotope	Activity (Ci)
I-131	1.95 ~~1.79~~ x 10^{7}
I-132	2.8 ~~2.73~~ x 10^{7}
I-133	3.95 ~~4.00~~ x 10^{7}
I-134	4.25 ~~4.70~~ x 10^{7}
I-135	3.65 x 10^{7}
Kr-83m	9.1 ~~1.33~~ x 10~~7~~ 6
Kr-85	6.4 ~~8.13~~ x 10^{5}
Kr-85m	2.0 ~~3.20~~ x 10^{7}
Kr-87	3.7 ~~6.17~~ x 10^{7}
Kr-88	5.3 ~~8.79~~ x 10^{7}
Kr-89	1.5 ~~1.14~~ x 10~~8~~ 7
Xe-131m	5.4 ~~5.43~~ x 10^{5}
Xe-133	1.5 ~~1.65~~ x 10^{8}
Xe-133m	2.3 ~~4.20~~ x 10~~8~~ 7
Xe-135	3.3 ~~4.52~~ x 10^{7}
Xe-135m	3.1 ~~4.46~~ x 10^{7}
Xe-138	1.3 ~~1.46~~ x 10^{8}

TABLE 15.4-15 (Continued)

PARAMETERS USED TO EVALUATE OFFSITE DOSES FOR THE LOSS OF COOLANT ACCIDENT

Parameter	Realistic Analysis	Conservative Analysis	Regulatory Guide 1.4 Analysis
Form of Iodine Activity in Containment Available for Release			
Elemental Iodine	91%	91%	91%
Organic Iodine	4%	4%	4%
Particulate Iodine	5%	5%	5%
Number of Spray Pumps Operating	1 of 2	1 of 2	1 of 2
Spray Removal Coefficient for Iodine			
Elemental	12.55 hr^{-1}	12.55 hr^{-1}	12.55 hr^{-1}
Particulate	0.507 hr^{-1}	0.507 hr^{-1}	0.507 hr^{-1}
Effective Decontamination Factor of Spray on Elemental Iodine	100	100	100
Containment Free Volume	1.84 x 10^6 ft^3	1.84 x 10^6 ft^3	1.84 x 10^6 ft^3
Containment Leak Rate	0.2% per day (0-24 hr) 0.1% per day (1-30 days)	0.2% per day (0.24 hr) 0.1% per day (1-30 days)	0.2% per day (0-24 hr) 0.1% per day (1-30 days)
Containment Recirculation Flow	54,200 ~~60,270~~ cfm	54,200 ~~60,270~~ cfm	54,200 ~~60,270~~ cfm

TABLE 15.4-16

OFFSITE DOSES FROM LOSS OF COOLANT ACCIDENT

	Thyroid Dose (Rem)	
	Site Boundary (0-2 hours) 1609 meters	Low Population Zone (0-30 days) 4827 meters
Realistic Analysis	1.78 ~~1.55~~ x 10^{-3}	3.00 ~~2.63~~ x 10^{-4}
Conservative Analysis	2.87 ~~1.99~~ x 10^{0}	5.06 ~~3.50~~ x 10^{-1}
Regulatory Guide 1.4 Analysis	1.74 ~~1.52~~ x 10^{2}	2.89 ~~2.50~~ x 10^{1}
10 CFR 100 Guidelines	300	300

	Gamma and Beta Doses (Rem)			
	Site Boundary (0-2 hours) 1609 meters		Low Population Zone (0-30 days) 4827 meters	
	Gamma Dose	Beta Skin Dose	Gamma Dose	Beta Skin Dose
Realistic Analysis	5.59 ~~5.99~~ x 10^{-5}	1.14 ~~1.32~~ x 10^{-4}	1.33 ~~2.37~~ x 10^{-5}	3.69 ~~4.29~~ x 10^{-5}
Conservative Analysis	~~6.21~~ 6.80 x 10^{-3}	~~6.55~~ 8.35 x 10^{-3}	~~1.17~~ 1.63 x 10^{-3}	~~1.66~~ 1.84 x 10^{-3}
Regulatory Guide 1.4 Analysis	~~4.45~~ 2.78 x 10^{0}	~~2.99~~ 2.16 x 10^{0}	~~6.87~~ 3.57 x 10^{-1}	3.07 ~~4.04~~ x 10^{-1}
10 CFR 100 Guidelines	25[(1)]		25[(1)]	

---

(1) Whole body dose.

TABLE 15.4-17

PARAMETERS USED IN ANALYSIS OF CONTROL ROOM DOSE FOLLOWING A LOSS OF COOLANT ACCIDENT

	Parameters
Control Room Free Volume	226,040 ft^3
Filtered Recirculation Flow	19,143 ~~21,270~~ cfm
Recirculation Filter Efficiencies	95% for all species of iodine
Maximum Control Room Filtered Air Infiltration Rate	2413 ~~2750~~ cfm*
Control Room Unfiltered Air Infiltration Rate	10 cfm
Maximum Control Room Outleakage	Equal to total inleakage (2423 ~~2160~~ cfm)
Meteorology	0-8 hrs: 9.35×10^{-4} sec/m^3 8-24 hrs: 6.63×10^{-4} sec/m^3 1-4 days: 3.95×10^{-4} sec/m^3 4-30 days: 2.45×10^{-4} sec/m^3
Percent of Time Operator Is in Control Room Following Accident	0 - 24 hrs 100% 1 - 4 days 60% 4 - 30 days 40%
Duration of Accident	30 days
Breathing Rate of Operators in Control Room	3.47×10^{-4} m^3/sec
Activity Release Assumptions	Table 15.4-15
Method of Dose Calculation	Appendix 15A

* Actual system capacity 2500 cfm (single train)

TABLE 15.4-18

CONTROL ROOM DOSES FOLLOWING A LOCA

	Doses (Rem)		
	Thyroid	Gamma	Beta Skin
Realistic Analysis	2.6 ~~2.54~~ $\times 10^{-4}$	2.2 ~~2.55~~ $\times 10^{-4}$	8.8 ~~9.91~~ $\times 10^{-4}$
Conservative Analysis	4.5 ~~3.25~~ $\times 10^{-1}$	8.9 ~~8.08~~ $\times 10^{-3}$	4.6 ~~4.71~~ $\times 10^{-2}$
Ultra-Conservative Analysis	3.0 3.08 $\times 10^{1}$	1.7 ~~2.26~~ $\times 10^{0}$	6.3 ~~9.29~~ $\times 10^{0}$

8. No condenser air removal system release and no steam generator blowdown during the accident.

9. No noble gas is dissolved in the steam generator water.

10. The iodine partition factor in the unfaulted steam generators, 0.01, is determined as follows:

$$\frac{\text{amount of iodine/unit mass steam}}{\text{amount of iodine/unit mass liquid}}$$

11. During the postulated accident, iodine carryover from the primary side in the two unfaulted steam generators is diluted in the incoming feedwater.

12. In the faulted steam generator, all water boils off and is released through the break immediately after the accident. The partition factor for iodine released is assumed to be 1.0. After this initial release, further iodine is released due to primary to secondary leakage in the faulted steam generator. A partition factor of 1.0 is also assumed for this release.

13. The primary pressure remains constant at 2235 psig for 0-2 hours and then decreases linearly to atmospheric during the period 2-8 hours.

14. The 0-2 hour and 2-8 hour atmospheric diffusion factors given in Appendix 15A and the 0-8 hour breathing rate of 3.47×10^{-4} m^3/sec are used.

15. The dose model used to evaluate this accident is given in Appendix 15A.

Steam releases to the atmosphere in the first two hours of the steam line break are given in Table ~~15.4-13~~ 15.4-23. Isotopic releases to the environment using these assumptions are summarized by Tables 15.4-24 through 15.4-27.

The gamma, beta and thyroid doses for the steam line break accident, based upon the realistic analysis, are ~~5.36×10^{-9}~~ 2.34×10^{-6} Rem, ~~5.47×10^{-9}~~ 2.05×10^{-6} Rem and ~~4.46×10^{-5}~~ 7.14×10^{-4} Rem, respectively, at the site boundary. Corresponding doses at the low population zone are ~~1.75×10^{-9}~~ 4.42×10^{-7} Rem, ~~1.85×10^{-9}~~ 4.37×10^{-7} Rem and ~~8.16×10^{-7}~~ 6.87×10^{-5} Rem, respectively.

Gamma, beta and thyroid doses at the site boundary and at the low population zone for the steam line break accident, based upon the conservative analysis, are given by Figures 15.4-61 through 15.4-63 as a function of primary to secondary leak rate. The doses resulting from this accident are well within the limits defined by 10 CFR 100 (25 Rem, whole body; 300 Rem, thyroid) for the range of credible steam generator tube leakage.

TABLE 15.4-23 (Continued)

PARAMETERS USED IN STEAM LINE BREAK ANALYSES

Parameter	Realistic Analysis	Conservative Analysis
Blowdown rate per steam generator prior to accident	41 gpm	15 gpm
Initial steam and water release from faulted steam generator	165,000 lb (0-30 minutes)	165,000 lb (0-30 minutes)
Long term steam release from faulted steam generator	12 lb (0-8 hours)	1,300 lb (0-8 hours)
Steam release from two unfaulted steam generators	332,700 lb (0-2 hours) 665,400 lb (2-8 hours)	332,700 lb (0-2 hours) 665,400 lb (2-8 hours)
Feedwater flow to two unfaulted steam generators	453,900 lb (0-2 hours) 721,500 lb (2-8 hours)	453,900 lb (0-2 hours) 721,500 lb (2-8 hours)
Meteorology	Annual average	Accident

TABLE 15.4-24

STEAM LINE BREAK ISOTOPIC RELEASE TO ENVIRONMENT CONSERVATIVE ANALYSIS(1)

Isotope	Activity Released to Environment by Accident (Ci) (0-2 hr)	(2-8 hr)
I-131	4.90 ~~4.45~~ x 10^-~~2~~1	1.37 ~~1.24~~ x 10^-~~1~~0
I-132	7.93 ~~4.77~~ x 10^-~~3~~2	2.12 ~~4.38~~ x 10^-~~3~~1
I-133	3.5[illegible] ~~5.63~~ x 10^-~~2~~1	9.16 ~~2.05~~ x 10^-~~2~~1
I-134	6.90 ~~1.81~~ x 10^-~~3~~2	2.04 ~~2.71~~ x 10^-~~3~~1
I-135	1.76 ~~2.05~~ x 10^-~~2~~1	4.80 ~~1.10~~ x 10^-~~2~~1
Xe-131m	3.87 ~~9.09~~ x 10^{-3}	1.16 ~~2.75~~ x 10^{-2}
Xe-133	6.95 ~~1.42~~ x 10^~~0~~-1	2.08 ~~4.27~~ x 10^{0}
Xe-133m	7.15 ~~1.81~~ x 10^{-2}	2.14 ~~5.44~~ x 10^-~~2~~1
Xe-135	4.28 ~~3.59~~ x 10^{-2}	1.29 ~~1.08~~ x 10^{-1}
Xe-135m	6.77 ~~2.27~~ x 10^-~~3~~2	2.03 ~~6.82~~ x 10^-~~3~~1
Xe-138	2.96 ~~3.64~~ x 10^-~~3~~1	8.86 ~~1.09~~ x 10^-~~2~~1
Kr-83m	5.34 ~~0.0~~ x 10^{-3}	1.60 ~~0.0~~ x 10^{-2}
Kr-85	5.72 ~~3.77~~ x 10^{-2}	1.71 ~~1.13~~ x 10^{-1}
Kr-85m	1.76 ~~1.06~~ x 10^{-2}	5.29 ~~3.15~~ x 10^{-2}
Kr-87	1.71 ~~5.91~~ x 10^-~~3~~2	5.15 ~~1.77~~ x 10^{-2}
Kr-88	3.79 ~~1.77~~ x 10^{-2}	1.13 ~~5.32~~ x 10^-~~2~~1
Kr-89	6.31 ~~0.0~~ x 10^{-3}	1.90 ~~0.0~~ x 10^{-2}

(1) Primary to Secondary Leakage = 0.01 gpm.

TABLE 15.4-25

STEAM LINE BREAK ISOTOPIC RELEASE
TO ENVIRONMENT CONSERVATIVE ANALYSIS(1)

Isotope	Activity Released to Environment by Accident (Ci) (0-2 hr)	(2-8 hr)
I-131	4.90 ~~4.45~~ x $10^{-\cancel{1}\,0}$	1.37 ~~1.24~~ x $10^{\cancel{-1}+1}$
I-132	7.93 ~~4.77~~ x $10^{-\cancel{2}\,1}$	2.12 ~~4.38~~ x $10^{-\cancel{1}\,0}$
I-133	3.55 ~~5.63~~ x $10^{-\cancel{1}\,0}$	9.16 ~~2.05~~ x $10^{-\cancel{1}\,0}$
I-134	6.90 ~~1.81~~ x $10^{-\cancel{2}\,1}$	2.04 ~~2.71~~ x $10^{-\cancel{1}\,0}$
I-135	1.76 ~~2.05~~ x $10^{\cancel{-1}\,0}$	4.80 ~~1.10~~ x $10^{-\cancel{1}\,0}$
Xe-131m	3.87 ~~9.09~~ x 10^{-2}	1.16 ~~2.73~~ x 10^{-1}
Xe-133	6.95 ~~1.42~~ x $10^{\cancel{1}\,0}$	2.08 ~~6.27~~ x 10^{1}
Xe-133m	7.15 ~~1.91~~ x 10^{-1}	2.14 ~~5.44~~ x $10^{\cancel{-1}\,0}$
Xe-135	4.28 ~~3.59~~ x 10^{-1}	1.29 ~~1.08~~ x 10^{0}
Xe-135m	6.77 ~~2.27~~ x $10^{-\cancel{2}\,1}$	2.03 ~~6.82~~ x $10^{-\cancel{1}\,0}$
Xe-138	2.96 ~~3.64~~ x $10^{-\cancel{1}\,0}$	8.86 ~~1.09~~ x $10^{-\cancel{1}\,0}$
Kr-83m	5.34 ~~0.0~~ x 10^{-2}	1.60 ~~0.0~~ x 10^{-1}
Kr-85	5.72 ~~3.77~~ x 10^{-1}	1.71 ~~1.15~~ x 10^{0}
Kr-85m	1.76 ~~1.04~~ x 10^{-1}	5.29 ~~3.15~~ x 10^{-1}
Kr-87	1.71 ~~5.91~~ x $10^{-\cancel{2}\,1}$	5.15 ~~1.77~~ x 10^{-1}
Kr-88	3.79 ~~1.77~~ x 10^{-1}	1.13 ~~5.32~~ x $10^{-\cancel{1}\,0}$
Kr-89	6.31 ~~0.0~~ x 10^{-2}	1.90 ~~0.0~~ x 10^{-1}

---

(1) Primary to Secondary Leakage = 0.1 gpm.

TABLE 15.4-26

STEAM LINE BREAK ISOTOPIC RELEASE TO ENVIRONMENT CONSERVATIVE ANALYSIS[1]

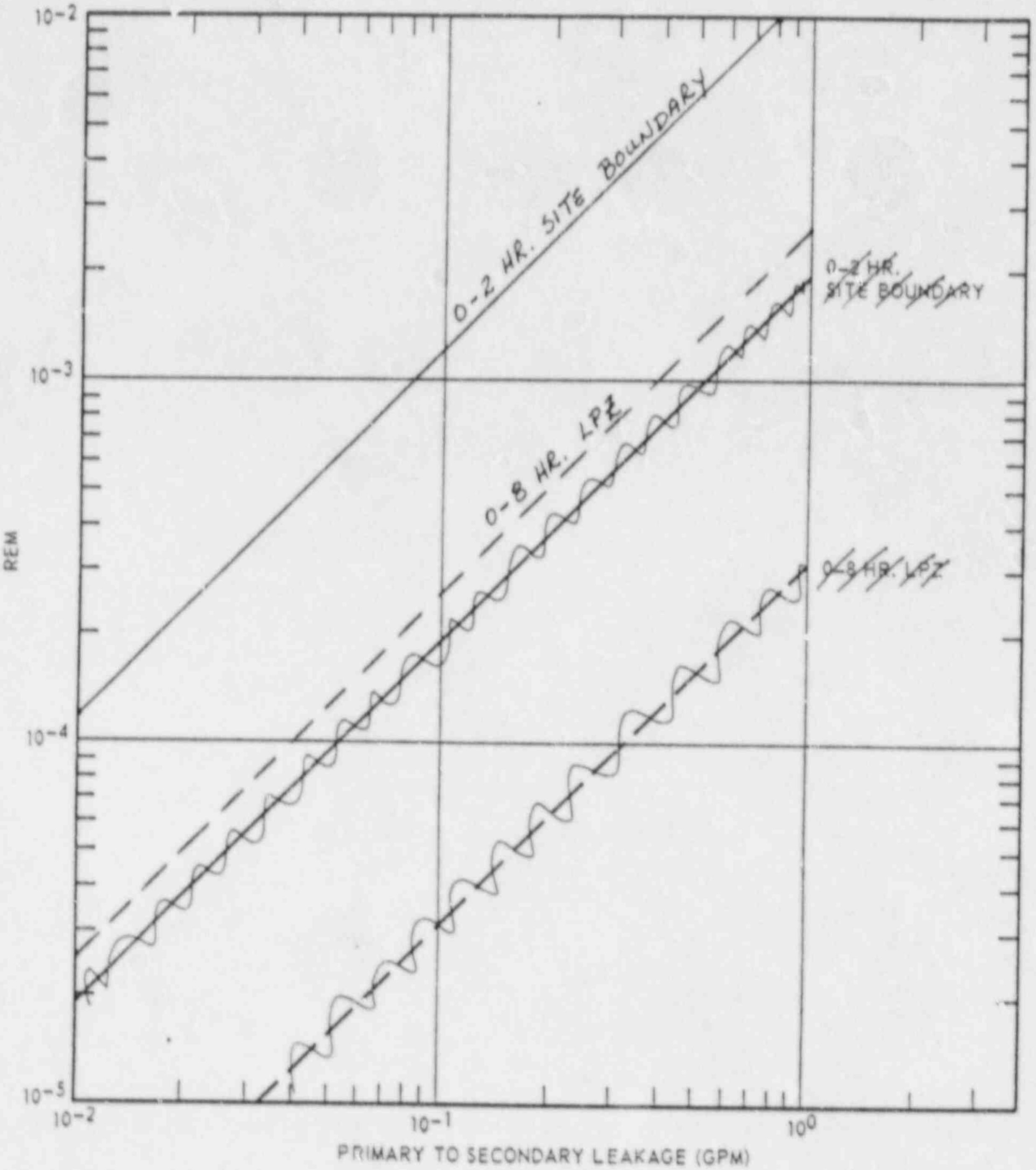
Isotope	Activity Released to Environment by Accident (Ci) (0-2 hr)	Activity Released to Environment by Accident (Ci) (2-8 hr)
I-131	4.90 ~~4.45~~ x $10^{\not{0}+1}$	1.37 ~~1.24~~ x $10^{\not{0}+2}$
I-132	7.93 ~~4.77~~ x $10^{\not{-1}\,0}$	2.12 ~~4.38~~ x $10^{\not{-1}+1}$
I-133	3.55 ~~5.63~~ x $10^{\not{0}+1}$	9.16 ~~2.05~~ x $10^{\not{0}+1}$
I-134	6.90 ~~1.81~~ x $10^{\not{-1}\,0}$	2.04 ~~2.71~~ x $10^{\not{-1}+1}$
I-135	1.76 ~~2.05~~ x $10^{\not{0}+1}$	4.80 ~~1.10~~ x $10^{\not{0}+1}$
Xe-131m	3.87 ~~9.09~~ x 10^{-1}	1.16 ~~2.73~~ x 10^{0}
Xe-133	6.95 ~~1.42~~ x $10^{\not{1}+1}$	2.08 ~~4.27~~ x 10^{2}
Xe-133m	7.15 ~~1.81~~ x 10^{0}	2.14 ~~5.44~~ x $10^{\not{0}+1}$
Xe-135	4.28 ~~3.59~~ x 10^{0}	1.29 ~~1.08~~ x 10^{1}
Xe-135m	6.77 ~~2.27~~ x $10^{\not{-1}\,0}$	2.03 ~~6.82~~ x $10^{\not{-1}+1}$
Xe-138	2.96 ~~3.64~~ x $10^{\not{-1}+1}$	8.86 ~~1.09~~ x $10^{\not{0}+1}$
Kr-83m	5.34 ~~0.0~~ x 10^{-1}	1.60 ~~0.0~~ x 10^{0}
Kr-85	5.72 ~~3.77~~ x 10^{0}	1.71 ~~1.13~~ x 10^{1}
Kr-85m	1.76 ~~1.04~~ x 10^{0}	5.29 ~~3.15~~ x 10^{0}
Kr-87	1.71 ~~5.91~~ x $10^{\not{-1}\,0}$	5.15 ~~1.77~~ x 10^{0}
Kr-88	3.79 ~~1.77~~ x 10^{0}	1.13 ~~5.32~~ x $10^{\not{0}+1}$
Kr-89	6.31 ~~0.0~~ x 10^{-1}	1.90 ~~0.0~~ x 10^{0}

(1) Primary to Secondary Leakage = 1.0 gpm.

TABLE 15.4-27

STEAM LINE BREAK ISOTOPIC RELEASE
TO ENVIRONMENT REALISTIC ANALYSIS

Isotope	Activity Released to Environment by Accident (Ci) (0-2 hr)	(2-8 hr)
I-131	2.13 ~~1.90~~ x 10^{-3}	1.33 ~~1.15~~ x 10^{-3}
I-132	1.23 ~~1.14~~ x 10^{-3}	1.36 ~~4.03~~ x $10^{-\cancel{4}3}$
I-133	3.19 ~~2.37~~ x 10^{-3}	2.18 ~~1.66~~ x 10^{-3}
I-134	1.95 ~~1.29~~ x 10^{-4}	3.12 ~~2.09~~ x 10^{-4}
I-135	1.50 ~~1.01~~ x 10^{-3}	1.23 ~~8.26~~ x $10^{-\cancel{4}3}$
Xe-131m	1.00 ~~6.80~~ x $10^{-\cancel{5}3}$	3.00 ~~2.04~~ x $10^{-\cancel{4}3}$
Xe-133	1.18 ~~1.93~~ x $10^{-\cancel{2}1}$	3.55 ~~5.78~~ x $10^{-\cancel{2}1}$
Xe-133m	7.93 ~~3.70~~ x $10^{-\cancel{4}3}$	2.30 ~~1.11~~ x $10^{-\cancel{3}2}$
Xe-135	3.63 ~~1.10~~ x $10^{-\cancel{2}3}$	1.09 ~~3.53~~ x $10^{-\cancel{3}2}$
Xe-135m	2.49 ~~5.67~~ x $10^{-\cancel{5}4}$	7.47 ~~1.70~~ x 10^{-4}
Xe-137	0 ~~3.70 x 10^{-5}~~	0 ~~1.14 x 10^{-4}~~
Xe-138	3.27 ~~1.89~~ x 10^{-4}	9.60 ~~5.68~~ x 10^{-4}
Kr-83m	0 ~~3.67 x 10^{-5}~~	0 ~~2.68 x 10^{-4}~~
Kr-85	4.13 ~~2.55~~ x $10^{-\cancel{5}3}$	1.24 ~~7.60~~ x $10^{-\cancel{5}2}$
Kr-85m	9.08 ~~4.16~~ x 10^{-4}	2.73 ~~1.25~~ x 10^{-3}
Kr-87	5.44 ~~2.46~~ x 10^{-4}	1.63 ~~7.38~~ x $10^{-\cancel{4}3}$
Kr-88	1.63 ~~7.94~~ x $10^{-\cancel{4}3}$	4.90 ~~2.38~~ x 10^{-3}
Kr-89	0 ~~2.19 x 10^{-5}~~	0 ~~6.60 x 10^{-5}~~

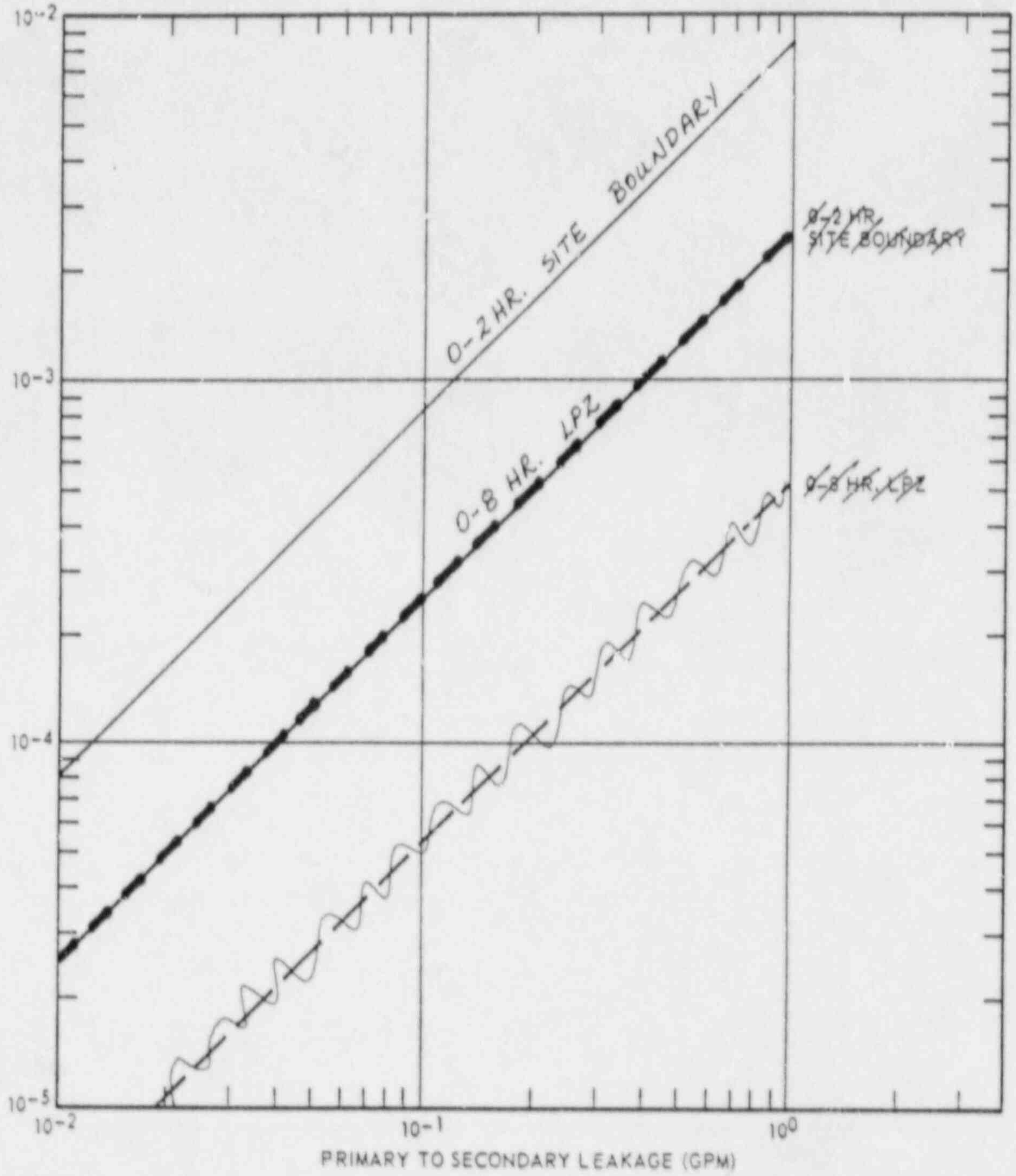


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Steam Line Break Whole Body Gamma Dose

CONSERVATIVE CASE

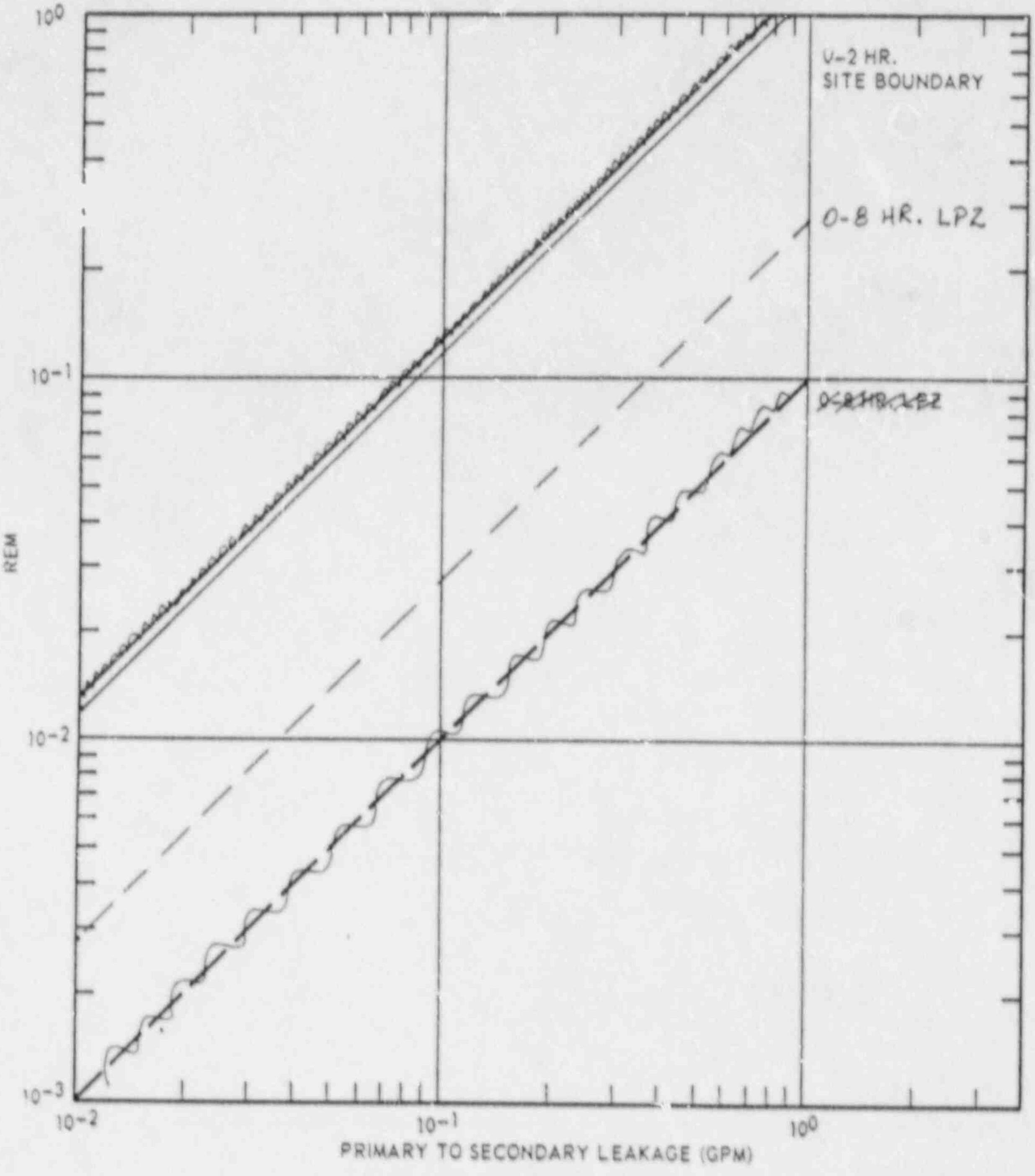
Figure 15.4-61



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Steam Line Break Surface Body
Beta Dose
CONSERVATIVE CASE

Figure 15.4-62



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Steam Line Break
Thyroid Dose
CONSERVATIVE CASE
Figure 15.4-63

The following information was required for the Steam Generator Tube Rupture Accident dose calculation:

1. The approximate mass of metal in contact with the RCS water is 1.83 x 10^6 lb.

2. The secondary side volume in one steam generator is 5947 ft^3.

3. The air ejector flow rate is

   Main condenser: 576 lb/hr, air/vapor mixture
   Auxiliary condensers: 215 lb/hr, air/vapor mixture

4. The normal operation letdown rate is 60 gpm.

5. The total RCS volume at hot conditions is 9410 ft^3, including the pressurizer. The liquid volume in the RCS during normal operation is 8850 ft^3 at hot conditions.

6. The volume fraction of liquid in the steam generators at normal operation is 0.3.

7. Emergency feedwater system initiation time is less than 1 minute. The flow rate to each steam generator is 190 gpm.

8. Figure 15.4-83 shows the liquid volume fraction in the faulted and non-faulted steam generators as a function of time after the tube rupture.

9. Pressure as a function of time in the primary system and in the faulted and non-faulted steam generators is shown in Figure 15.4-84.

The steam releases to the atmosphere for the postulated steam generator tube rupture are given in Table 15.4-29. Isotopic releases to the environment based upon these assumptions are summarized in Tables 15.4-30 through 15.4-33.

Gamma, beta and thyroid doses at the site boundary in the first two hours of the postulated steam generator tube rupture accident and based upon the realistic analysis are ~~[illegible]~~ 1.78E-4 Rem, ~~[illegible]~~ 2.06E-4 Rem and ~~[illegible]~~ 1.09E-4 Rem, respectively. Corresponding doses at the low population zone are ~~[illegible]~~ 2.00E-5 Rem, ~~[illegible]~~ 2.32E-5 Rem and ~~[illegible]~~ 1.22E-5 Rem, respectively, for the duration of the accident.

Gamma, beta and thyroid doses at the site boundary and low population zone resulting from the postulated steam generator tube rupture accident and based upon the conservative analysis as a function of primary to secondary leak rate are given by Figures 15.4-67 through 15.4-68. The doses from this accident are well within the limits defined in 10 CFR 100 (25 Rem, whole body, 300 Rem, thyroid) for the range of credible steam generator tube leakage.

TABLE 15.4-29

PARAMETERS USED IN STEAM GENERATOR TUBE RUPTURE ANALYSES

	Realistic Analysis	Conservative Analysis
Core thermal power	2900 MWt	2900 MWt
Steam generator tube leak rate prior to and during accident	100 lbs/day[1]	0.01 to 1.0 gpm
Offsite power	Available	Lost
Fuel defects	0.12%[1]	1%
Iodine partition factors in steam generators prior to and during accident	0.01	0.01
Blowdown rate per steam generator prior to accident	41 gpm	15 gpm
Time to isolate defective steam generator	30 min	30 min
Duration of plant cooldown by secondary system after accident	8 hr	8 hr
Steam release from defective steam generator	48,000 lbs (0-30 min)	48,000 lbs (0-30 min)
Steam release from 2 unaffected steam generators	316,000 lbs (0-2 hr) 835,000 lbs (2-8 hr)	316,000 lbs (0-2 hr) 835,000 lbs (2-8 hr)
Feedwater flow to 2 unaffected steam generators	346,000 lbs (0-2 hr) 883,000 lbs (2-8 hr)	346,000 lbs (0-2 hr) 883,000 lbs (2-8 hr)
Reactor coolant released to the defective steam generator	125,000 lbs	125,000 lbs
Meteorology	Annual average	Accident

---

(1) American National Standards Institute, "Source Term Specification," N237, Draft Revision 2.

TABLE 15.4-30

STEAM GENERATOR TUBE RUPTURE ISOTOPIC RELEASE TO ENVIRONMENT CONSERVATIVE ANALYSIS[1]

Isotope	Activity Released to Environment by Accident (Ci) (0-2 hr)	Activity Released to Environment by Accident (Ci) (2-8 hr)
I-131	3.56 ~~3.18~~ x 10^{-1}	2.70 2.42 x 10^{-2}
I-132	3.69 ~~1.18~~ x 10^{-1}	2.67 8.56 x 10^{-3} 2
I-133	5.66 ~~5.32~~ x 10^{-1}	4.36 3.99 x 10^{-2}
I-134	8.50 ~~2.38~~ x 10^{-2}	6.12 5.30 x 10^{-3}
I-135	3.31 ~~2.91~~ x 10^{-1}	2.42 2.15 x 10^{-2}
Xe-131m	1.25 ~~1.13~~ x 10^{4}	2.00 1.82 x 10^{-2}
Xe-133	1.48 ~~1.77~~ x 10^{4}	2.36 2.83 x 10^{0}
Xe-133m	9.65 ~~2.28~~ x 10^{2}	1.65 3.63 x 10^{-2} 1
Xe-135	4.54 ~~4.48~~ x 10^{2}	7.27 7.17 x 10^{-2}
Xe-135m	3.12 ~~2.84~~ x 10^{1}	5.00 4.54 x 10^{-3}
Xe-138	4.09 ~~4.54~~ x 10^{1}	6.54 7.26 x 10^{-3}
Kr-83m	---	---
Kr-85	5.16 ~~4.71~~ x 10^{2}	8.27 7.54 x 10^{-2}
Kr-85m	1.14 ~~1.30~~ x 10^{2}	1.82 2.09 x 10^{-2}
Kr-87	6.81 ~~7.37~~ x 10^{1}	1.09 1.18 x 10^{-2}
Kr-88	2.04 ~~2.21~~ x 10^{2}	3.27 3.54 x 10^{-2}
Kr-89	---	---

(1) Primary to secondary leakage = 0.01 gpm.

TABLE 15.4-31

STEAM GENERATOR TUBE RUPTURE ISOTOPIC RELEASE TO ENVIRONMENT CONSERVATIVE ANALYSIS[(1)]

Isotope	Activity Released to Environment by Accident (Ci) (0-2 hr)	(2-8 hr)
I-131	4.40 ~~3.93~~ x 10^{-1}	2.70 ~~2.42~~ x 10^{-1}
I-132	4.49 ~~1.44~~ x 10^{-1}	2.67 ~~8.56~~ x $10^{-\cancel{2}1}$
I-133	7.19 ~~6.54~~ x 10^{-1}	4.36 ~~3.39~~ x 10^{-1}
I-134	1.03 ~~8.78~~ x $10^{-\cancel{2}1}$	6.12 ~~5.27~~ x 10^{-2}
I-135	4.04 ~~2.37~~ x 10^{-1}	2.42 ~~2.18~~ x 10^{-1}
Xe-131m	1.25 ~~1.13~~ x 10^{2}	2.00 ~~1.52~~ x 10^{-1}
Xe-133	1.48 ~~1.77~~ x 10^{4}	2.36 ~~2.03~~ x 10^{1}
Xe-133m	9.65 ~~2.38~~ x 10^{2}	1.55 ~~3.36~~ x $10^{\cancel{+1}0}$
Xe-135	4.54 ~~4.48~~ x 10^{2}	7.27 ~~7.17~~ x 10^{-1}
Xe-135m	3.12 ~~2.84~~ x 10^{1}	5.00 ~~4.54~~ x 10^{-2}
Xe-138	4.09 ~~4.34~~ x 10^{1}	6.54 ~~7.26~~ x 10^{-2}
Kr-83m	---	---
Kr-85	5.16 ~~4.71~~ x 10^{2}	8.27 ~~7.54~~ x 10^{-1}
Kr-85m	1.14 ~~1.30~~ x 10^{2}	1.82 ~~2.09~~ x 10^{-1}
Kr-87	6.81 ~~7.37~~ x 10^{1}	1.09 ~~1.18~~ x 10^{-1}
Kr-88	2.04 ~~2.21~~ x 10^{2}	3.27 ~~3.34~~ x 10^{-1}
Kr-89	---	---

(1) Primary to secondary leakage = 0.1 gpm.

TABLE 15.4-32

STEAM GENERATOR TUBE RUPTURE ISOTOPIC RELEASE TO ENVIRONMENT CONSERVATIVE ANALYSIS[(1)]

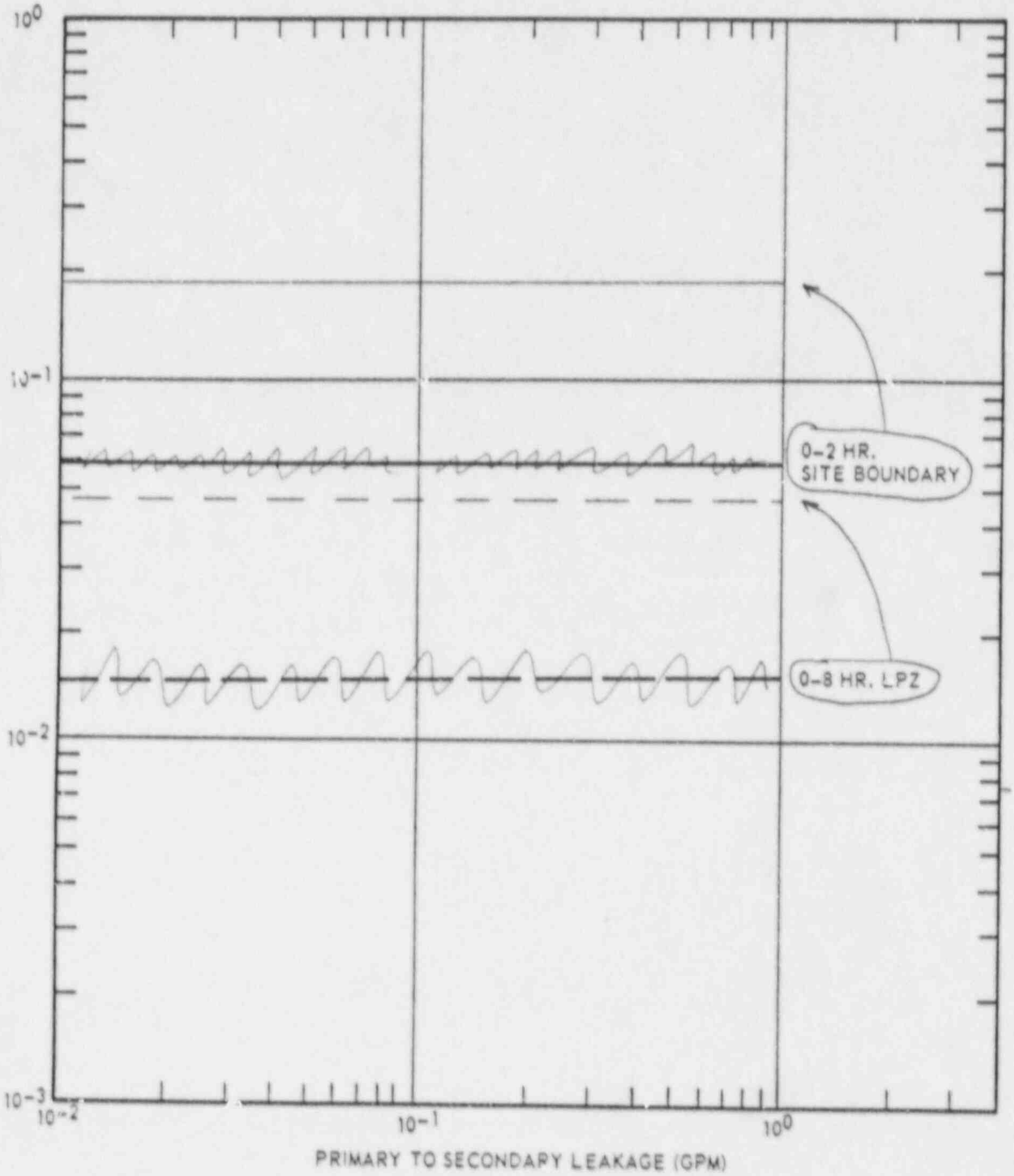
	Activity Released to Environment by Accident (Ci)	
Isotope	(0-2 hr)	(2-8 hr)
I-131	1.27 ~~1.14~~ x 10^{0}	2.70 ~~2.42~~ x 10^{0}
I-132	1.25 ~~4.02~~ x 10^{0} (~~+1~~)	2.67 ~~8.56~~ x 10^{0} (~~+1~~)
I-133	2.05 ~~1.88~~ x 10^{0}	4.36 ~~3.99~~ x 10^{0}
I-134	2.87 ~~2.48~~ x 10^{-1}	6.12 ~~5.30~~ x 10^{-1}
I-135	1.14 ~~1.01~~ x 10^{0}	2.42 ~~2.15~~ x 10^{0}
Xe-131m	1.26 ~~1.14~~ x 10^{2}	2.00 ~~1.82~~ x 10^{0}
Xe-133	1.49 ~~1.77~~ x 10^{4}	2.36 ~~2.83~~ x 10^{2}
Xe-133m	9.70 ~~3.21~~ x 10^{2}	1.55 ~~3.03~~ x 10^{1} (~~0~~)
Xe-135	4.56 ~~4.50~~ x 10^{2}	7.27 ~~7.17~~ x 10^{0}
Xe-135m	3.14 ~~2.28~~ x 10^{1}	5.00 ~~4.54~~ x 10^{-1}
Xe-138	4.11 ~~4.56~~ x 10^{1}	6.54 ~~7.28~~ x 10^{-1}
Kr-83m	---	---
Kr-85	5.19 ~~4.74~~ x 10^{2}	8.27 ~~7.54~~ x 10^{0}
Kr-85m	1.15 ~~1.31~~ x 10^{2}	1.82 ~~2.09~~ x 10^{0}
Kr-87	6.85 ~~7.61~~ x 10^{1}	1.09 ~~1.18~~ x 10^{0}
Kr-88	2.05 ~~2.22~~ x 10^{2}	3.27 ~~3.54~~ x 10^{0}
Kr-89	---	---

(1) Primary to secondary leakage = 1.0 gpm.

TABLE 15.4-33

STEAM GENERATOR TUBE RUPTURE ISOTOPIC RELEASE TO ENVIRONMENT REALISTIC ANALYSIS

Isotope	Activity Released to Environment by Accident (Ci) (0-2 hr)	Activity Released to Environment by Accident (Ci) (2-8 hr)
I-131	4.15 ~~3.58~~ $\times 10^{-2}$	6.81 ~~7.79~~ $\times 10^{-5}$
I-132	4.32 ~~1.48~~ $\times 10^{-2}$	5.49 ~~4.54~~ $\times 10^{-5}$
I-133	6.87 ~~5.18~~ $\times 10^{-2}$	1.33 ~~9.89~~ $\times$ ~~10^{-5}~~ 10^{-4}
I-134	9.94 ~~6.67~~ $\times 10^{-3}$	9.46 ~~6.46~~ $\times 10^{-6}$
I-135	3.88 ~~2.80~~ $\times 10^{-2}$	6.39 ~~4.20~~ $\times 10^{-5}$
Xe-131m	1.50 ~~1.02~~ $\times$ ~~10^{0}~~ 10^{1}	2.00 ~~7.36~~ $\times$ ~~10^{-4}~~ 10^{-3}
Xe-133	1.78 ~~2.09~~ $\times$ ~~10^{2}~~ 10^{3}	2.37 ~~3.35~~ $\times$ ~~10^{-2}~~ 10^{-1}
Xe-133m	1.16 ~~5.67~~ $\times$ ~~10^{1}~~ 10^{2}	1.52 ~~7.37~~ $\times$ ~~10^{-4}~~ 10^{-2}
Xe-135	5.45 ~~1.70~~ $\times 10^{+1}$	7.27 ~~2.35~~ $\times 10^{-3}$
Xe-135m	3.74 ~~8.50~~ $\times$ ~~10^{+1}~~ 10^{0}	4.98 ~~1.13~~ $\times 10^{-4}$
~~Xe-137~~	~~5.62 $\times 10^{-1}$~~	~~3.57 $\times 10^{-5}$~~
Xe-138	4.90 ~~2.03~~ $\times 10^{0}$	6.53 ~~3.78~~ $\times 10^{-4}$
~~Kr-83m~~	~~1.20 $\times 10^{0}$~~	~~1.73 $\times 10^{-4}$~~
Kr-85	6.20 ~~3.00~~ $\times 10^{+1}$	8.27 ~~5.07~~ $\times$ ~~10^{-4}~~ 10^{-3}
Kr-85m	1.36 ~~6.24~~ $\times$ ~~10^{0}~~ 10^{1}	1.82 ~~8.32~~ $\times$ ~~10^{-4}~~ 10^{-3}
Kr-87	8.16 ~~3.40~~ $\times 10^{0}$	1.09 ~~4.32~~ $\times$ ~~10^{-4}~~ 10^{-3}
Kr-88	2.45 ~~1.19~~ $\times 10^{1}$	2.27 ~~1.59~~ $\times 10^{-3}$
~~Kr-89~~	~~5.27 $\times 10^{-1}$~~	~~4.08 $\times 10^{-5}$~~

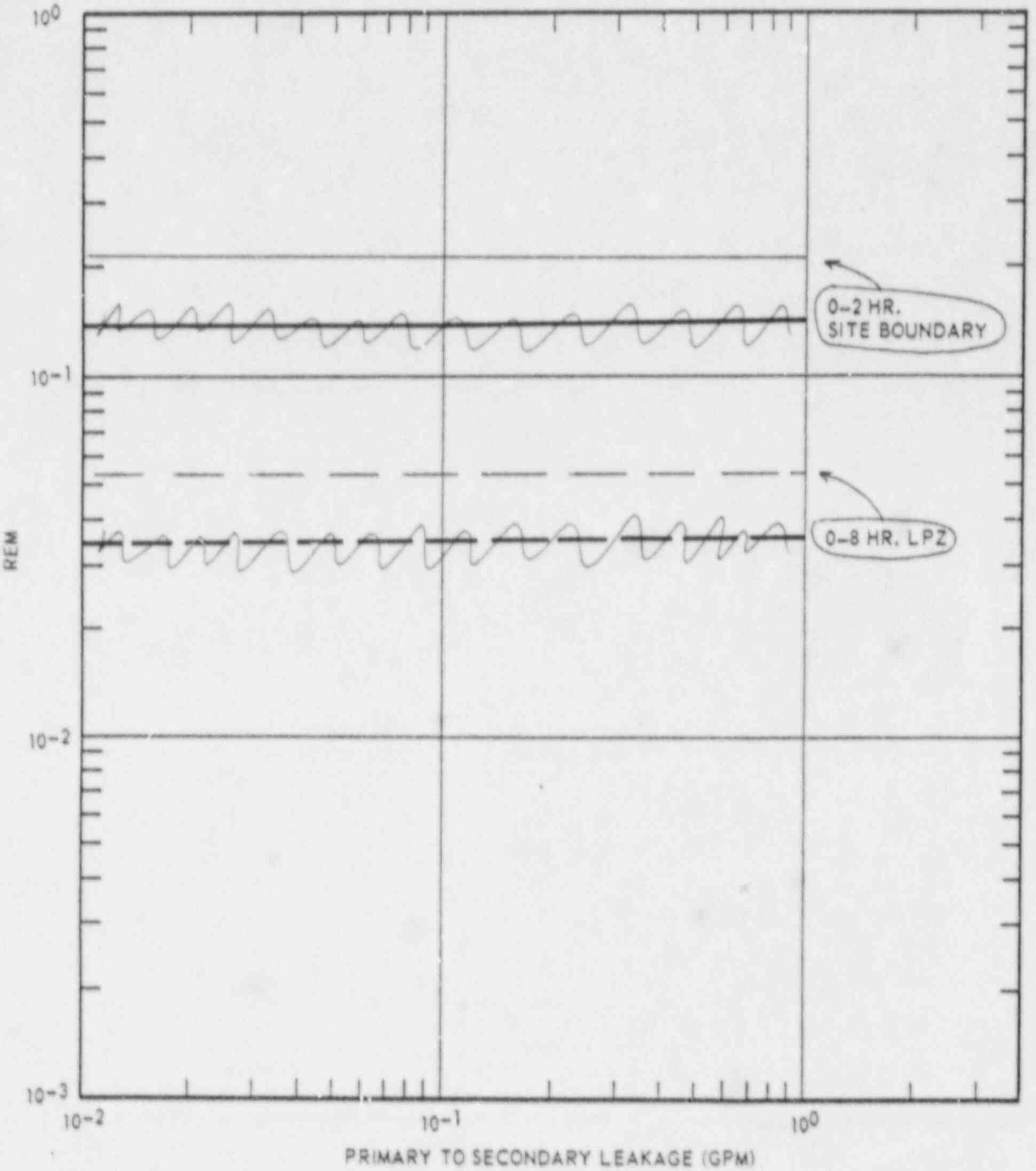


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Steam Generator Tube Rupture
Surface Body Gamma Dose

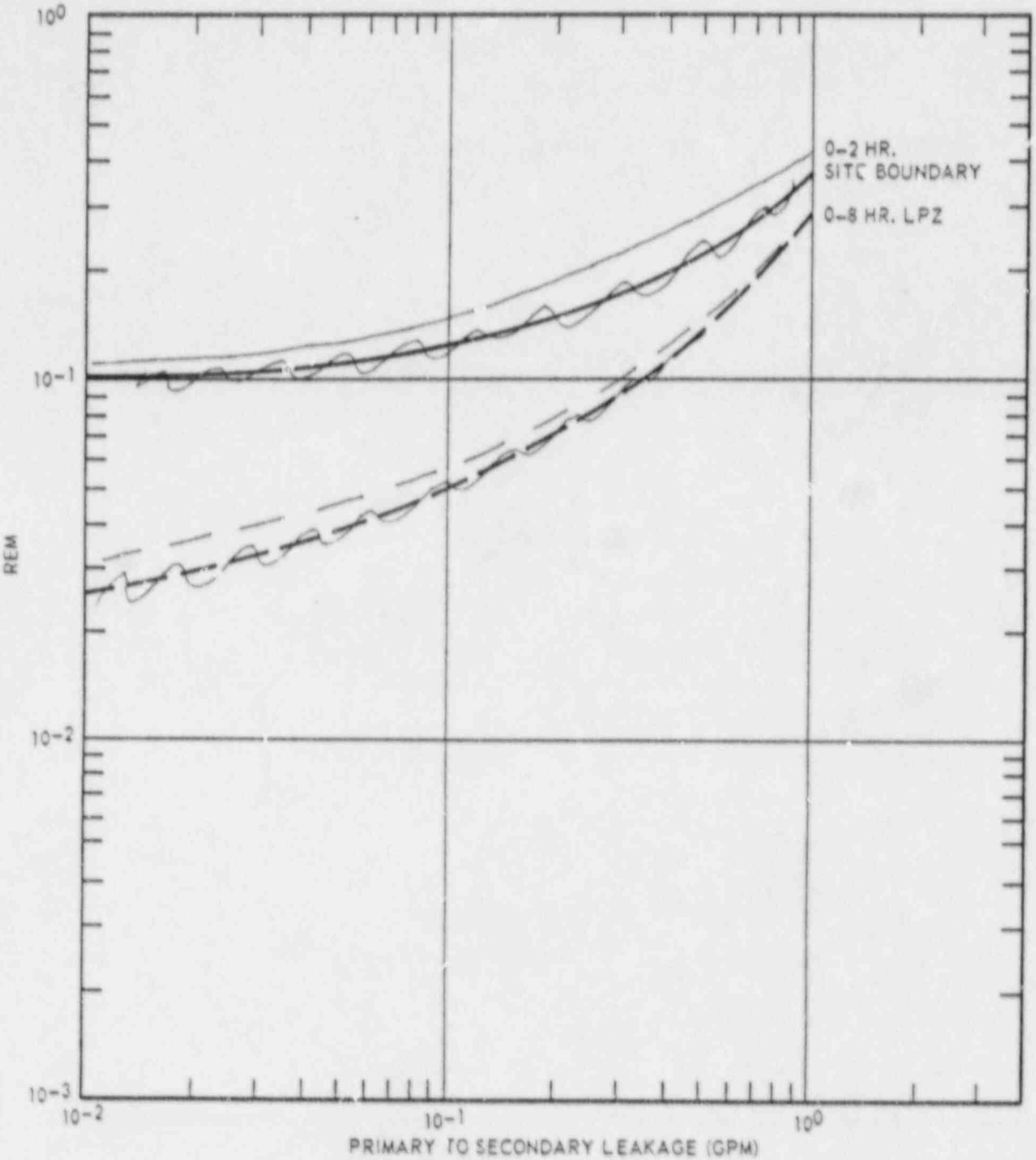
CONSERVATIVE CASE

Figure 15.4-66



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Steam Generator Tube Rupture
Whole Body ~~Gamma~~ Beta Dose
CONSERVATIVE CASE
Figure 15.4-67



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Steam Generator Tube Rupture
Thyroid Dose

CONSERVATIVE CASE

Figure 15.4-68

4. The iodine partition factor in the steam generators, 0.01, is determined as follows:

$$\frac{\text{amount of iodine/unit mass steam}}{\text{amount of iodine/unit mass liquid}}$$

5. No noble gas is dissolved or contained in the steam generator water; i.e., all noble gas leaked to the secondary system is continuously released with steam from the steam generators through the condenser air removal system.

6. The blowdown rate from the steam generators is continuous at 15.0 gpm per steam generator.

The following conservative assumptions and parameters are used to calculate the activity releases and offsite doses for a steam line break:

1. Prior to the accident, an equilibrium activity of fission products exists in the primary and secondary systems due to primary to secondary leakage in the steam generators.

2. Offsite power is lost and the main condenser is not available for steam dump.

3. Eight hours after the accident the residual heat removal system starts operation to cool down the plant.

4. After eight hours following the accident, no steam and activity are released to the environment.

5. Primary to secondary leakage is evenly distributed in the steam generators and varied from 0.01 gpm to 1.0 gpm.

6. Defective fuel prior to the accident is one percent.

7. As a result of the accident, ~~ten~~ fifteen percent of the fuel rods in the core are considered to be failed and their gap activity is considered to be released to, and instantaneously mixed with, the reactor coolant. The gap activity consists of 10 percent of the total noble gases other than Kr-85, 30 percent of the Kr-85, and 10 percent of the total radioactive iodine in the damaged rods at the time of the accident.

8. No condenser air removal system release and no steam generator blowdown occurs during the accident.

9. No noble gas is dissolved in the steam generator water.

10. The iodine partition factor in the steam generators, 0.01, is determined as follows:

$$\frac{\text{amount of iodine/unit mass steam}}{\text{amount of iodine/unit mass liquid}}$$

11. During the postulated accident, iodine in the steam generators is diluted with the incoming feedwater.

12. The primary pressure remains constant at 2235 psig for 0-2 hours and then decreases linearly to atmospheric during the period 2-8 hours.

13. The 0-2 hour and 2-8 hour atmospheric diffusion factors given in Appendix 15A and the 0-8 hour breathing rate of 3.47×10^{-4} m^3/sec are used.

14. The dose model used to evaluate this accident is given in Appendix 15A.

Steam releases to the atmosphere for the reactor pump locked rotor accident are given in Table 15.4-34a. Assumptions for the realistic analysis are also presented in Table 15.4-34a. Isotopic releases to the environment using these assumptions are summarized by Tables 15.4-34b through 15.4-34e.

The gamma, beta and thyroid doses for the reactor coolant pump locked rotor accident, based upon the realistic analysis, are 2.38×10^{-9} Rem, 3.[illegible]4 $\times 10^{-9}$ Rem and 9.98×10^{-8} Rem, respectively, at the site boundary. Corresponding doses at the low population zone are 1.05×10^{-9} Rem, 1.44×10^{-9} Rem, and 2.98×10^{-8} Rem, respectively.

Gamma, beta and thyroid doses at the site boundary and at the low population zone for the reactor coolant pump locked rotor accident, based upon the conservative analysis, are given by Figures 15.4-77a through 15.4-77c as a function of primary to secondary leak rate. The doses resulting from this accident are well within the limits defined by 10 CFR 100 (25 Rem, whole body; 300 Rem, thyroid) for the range of credible steam generator tube leakage.

15.4.5 FUEL HANDLING ACCIDENTS

A fuel handling accident (FHA) during refueling could release a fraction of the fission product inventory in the plant to the environment. Two accident scenerios are considered: (1) a refueling accident occurring inside containment and (2) a refueling accident occurring outside containment.

15.4.5.1 Fuel Handling Accident Inside of Containment

The postulated fuel handling accident inside containment is the dropping of a spent fuel assembly onto the core during refueling which results in damage to the fuel assemblies. For this postulated accident, two analyses bases are evaluated: (1) a realistic case and (2) a conservative case. The conservative case analysis is based on Regulatory Guide 1.25 assumptions. The assumed analysis parameters and radiological consequences associated with these cases are discussed below.

TABLE 15.4-34a

PARAMETERS USED IN LOCKED ROTOR ACCIDENT ANALYSIS

	Realistic Analysis	Conservative Analysis
Core Thermal Power	2900 MWt	2900 MWt
Steam Generator Tube Leak Rate Prior to Accident and for First Eight Hours Following Accident	100 lb/day(1)	0.01 to 1.0 gpm
Offsite Power	Lost	Lost
Fuel Defects	0.12 percent	1 percent
Failed Fuel	0.0	10 percent
Activity Released to Reactor Coolant from Failed Fuel	0.0	10 percent of gap inventory
Percent of Activity in Damaged Rods in the Gap		
Noble Gases (except Kr-85)	-	10 percent
Kr-85	-	30 percent
Iodines	-	10 percent
Iodine Partition Factor for Steam Generators	0.01	0.01
Duration of Plant Cooldown by Secondary System after Accident	8 hours	8 hours
Blowdown Rate per Steam Generator prior to Accident	~~62~~ 41 gpm	~~62~~ 15 gpm

TABLE 15.4-34b

REACTOR COOLANT PUMP LOCKED ROTOR
ISOTOPIC RELEASE TO ENVIRONMENT REALISTIC ANALYSIS

	Activity Released to Environment by Accident (Ci)	
Isotope	(0-2 hr)	(~~0-8~~ 2-8 hr)
I-131	4.50 ~~5.24~~ x 10^{-5}	7.91 ~~1.12~~ x ~~10^{-4}~~ 10^{-5}
I-132	2.93 ~~2.49~~ x 10^{-5}	5.82 ~~6.29~~ x 10^{-5}
I-133	7.23 ~~5.37~~ x 10^{-5}	1.23 ~~1.45~~ x 10^{-4}
I-134	4.94 ~~3.38~~ x 10^{-6}	1.12 ~~1.11~~ x 10^{-5}
I-135	3.44 ~~2.32~~ x 10^{-5}	6.18 ~~6.47~~ x 10^{-5}
Xe-131m	1.00 ~~6.81~~ x ~~10^{-8}~~ 10^{-3}	3.00 ~~2.72~~ x ~~10^{-4}~~ 10^{-3}
Xe-133	1.18 ~~1.93~~ x ~~10^{-2}~~ 10^{-1}	3.55 ~~2.71~~ x ~~10^{-2}~~ 10^{-1}
Xe-133m	7.72 ~~3.79~~ x ~~10^{-4}~~ 10^{-3}	2.32 ~~1.52~~ x ~~10^{-3}~~ 10^{-2}
Xe-135	3.64 ~~1.18~~ x 10^{-3}	1.09 ~~4.74~~ x ~~10^{-3}~~ 10^{-2}
Xe-135m	2.50 ~~5.67~~ x ~~10^{-5}~~ 10^{-4}	7.50 ~~2.27~~ x 10^{-4}
Xe-137	0 ~~3.79 x 10^{-5}~~	0 ~~1.52 x 10^{-4}~~
Xe-138	3.26 ~~1.89~~ x 10^{-4}	9.78 ~~7.57~~ x 10^{-4}
Kr-83m	0 ~~8.67 x 10^{-5}~~	0 ~~3.47 x 10^{-4}~~
Kr-85	4.14 ~~2.54~~ x ~~10^{-5}~~ 10^{-3}	1.24 ~~1.02~~ x ~~10^{-4}~~ 10^{-2}
Kr-85m	9.10 ~~4.18~~ x 10^{-4}	2.73 ~~1.69~~ x 10^{-3}
Kr-87	5.44 ~~2.46~~ x 10^{-4}	1.63 ~~9.84~~ x ~~10^{-4}~~ 10^{-3}
Kr-88	1.43 ~~7.95~~ x ~~10^{-4}~~ 10^{-3}	4.90 ~~3.18~~ x 10^{-3}
Kr-89	0 ~~2.19 x 10^{-5}~~	0 ~~8.77 x 10^{-5}~~

TABLE 15.4-34c

REACTOR COOLANT PUMP LOCKED ROTOR
ISOTOPIC RELEASE TO ENVIRONMENT CONSERVATIVE ANALYSIS(1)

	Activity Released to Environment by Accident (Ci)	
Isotope	(0-2 hr)	(0-8 hr)
I-131	2.12 ~~1.78~~ x 10^{-1}	8.46 ~~7.11~~ x 10^{-1}
I-132	3.03 ~~2.70~~ x 10^{-1}	1.21 ~~1.08~~ x 10^{0}
I-133	4.29 ~~3.97~~ x 10^{-1}	1.71 ~~1.58~~ x 10^{0}
I-134	4.62 ~~4.66~~ x 10^{-1}	1.85 ~~1.86~~ x 10^{0}
I-135	3.96 ~~3.62~~ x 10^{-1}	1.59 ~~1.45~~ x 10^{0}
Xe-131m	1.58 ~~1.44~~ x 10^{-1}	6.32 ~~5.77~~ x 10^{-1}
Xe-133	4.20 ~~4.24~~ x 10^{1}	1.68 ~~1.70~~ x 10^{2}
Xe-133m	6.23 ~~1.06~~ x 10^{0}	2.54 ~~4.25~~ x ~~10^{0}~~ 10^{1}
Xe-135	9.97 ~~1.12~~ x ~~10^{1}~~ 10^{0}	3.59 ~~4.49~~ x 10^{1}
Xe-135m	9.40 ~~1.10~~ x ~~10^{1}~~ 10^{0}	3.76 ~~4.41~~ x 10^{1}
Xe-138	3.54 ~~3.62~~ x 10^{1}	1.42 ~~1.45~~ x 10^{2}
Kr-83m	2.46 ~~3.30~~ x 10^{0}	9.87 ~~1.32~~ x ~~10^{1}~~ 10^{0}
Kr-85	5.67 ~~6.42~~ x 10^{-1}	2.27 ~~2.57~~ x 10^{0}
Kr-85m	5.43 ~~7.92~~ x 10^{0}	2.18 ~~3.17~~ x 10^{1}
Kr-87	1.02 ~~1.53~~ x 10^{1}	4.02 ~~6.13~~ x 10^{1}
Kr-88	1.44 ~~2.18~~ x 10^{1}	5.75 ~~8.72~~ x 10^{1}
Kr-89	1.77 ~~2.82~~ x 10^{1}	7.05 ~~1.13~~ x ~~10^{2}~~ 10^{1}

Note:

(1) Primary to secondary leakage equal to 0.01 gpm.

TABLE 15.4-34d

REACTOR COOLANT PUMP LOCKED ROTOR
ISOTOPIC RELEASE TO ENVIRONMENT CONSERVATIVE ANALYSIS(1)

Isotope	Activity Released to Environment by Accident (Ci) (0-2 hr)	(0-8 hr)
I-131	2.12 ~~1.78~~ x 10^{0}	8.46 ~~7.11~~ x 10^{0}
I-132	3.03 ~~2.70~~ x 10^{0}	1.21 ~~1.08~~ x 10^{1}
I-133	4.29 ~~3.97~~ x 10^{0}	1.71 ~~1.58~~ x 10^{1}
I-134	4.62 ~~4.66~~ x 10^{0}	1.85 ~~1.60~~ x 10^{1}
I-135	3.96 ~~3.44~~ x 10^{0}	1.59 ~~1.45~~ x 10^{1}
Xe-131m	1.50 ~~1.46~~ x 10^{0}	6.32 ~~5.77~~ x 10^{0}
Xe-133	4.40 ~~4.24~~ x 10^{2}	1.68 ~~1.70~~ x 10^{3}
Xe-133m	6.33 ~~1.06~~ x 10^{1}	2.54 ~~4.25~~ x ~~10^{1}~~ 2
Xe-135	8.97 ~~1.12~~ x ~~10^{2}~~ 1	3.59 ~~4.49~~ x 10^{2}
Xe-135m	8.40 ~~1.18~~ x ~~10^{2}~~ 1	3.36 ~~4.01~~ x 10^{2}
Xe-138	3.54 ~~3.62~~ x 10^{2}	1.42 ~~1.45~~ x 10^{3}
Kr-83m	2.46 ~~2.30~~ x 10^{1}	9.87 ~~1.32~~ x ~~10^{2}~~ 1
Kr-85	5.67 ~~6.42~~ x 10^{0}	2.27 ~~2.57~~ x 10^{1}
Kr-85m	5.43 ~~7.02~~ x 10^{1}	2.18 ~~2.17~~ x 10^{2}
Kr-87	1.02 ~~1.53~~ x 10^{2}	4.02 ~~6.13~~ x 10^{2}
Kr-88	1.44 ~~2.18~~ x 10^{2}	5.75 ~~8.72~~ x 10^{2}
Kr-89	1.77 ~~2.02~~ x 10^{2}	7.05 ~~1.13~~ x ~~10^{3}~~ 2

Note:

(1) Primary to secondary leakage equal to 0.1 gpm.

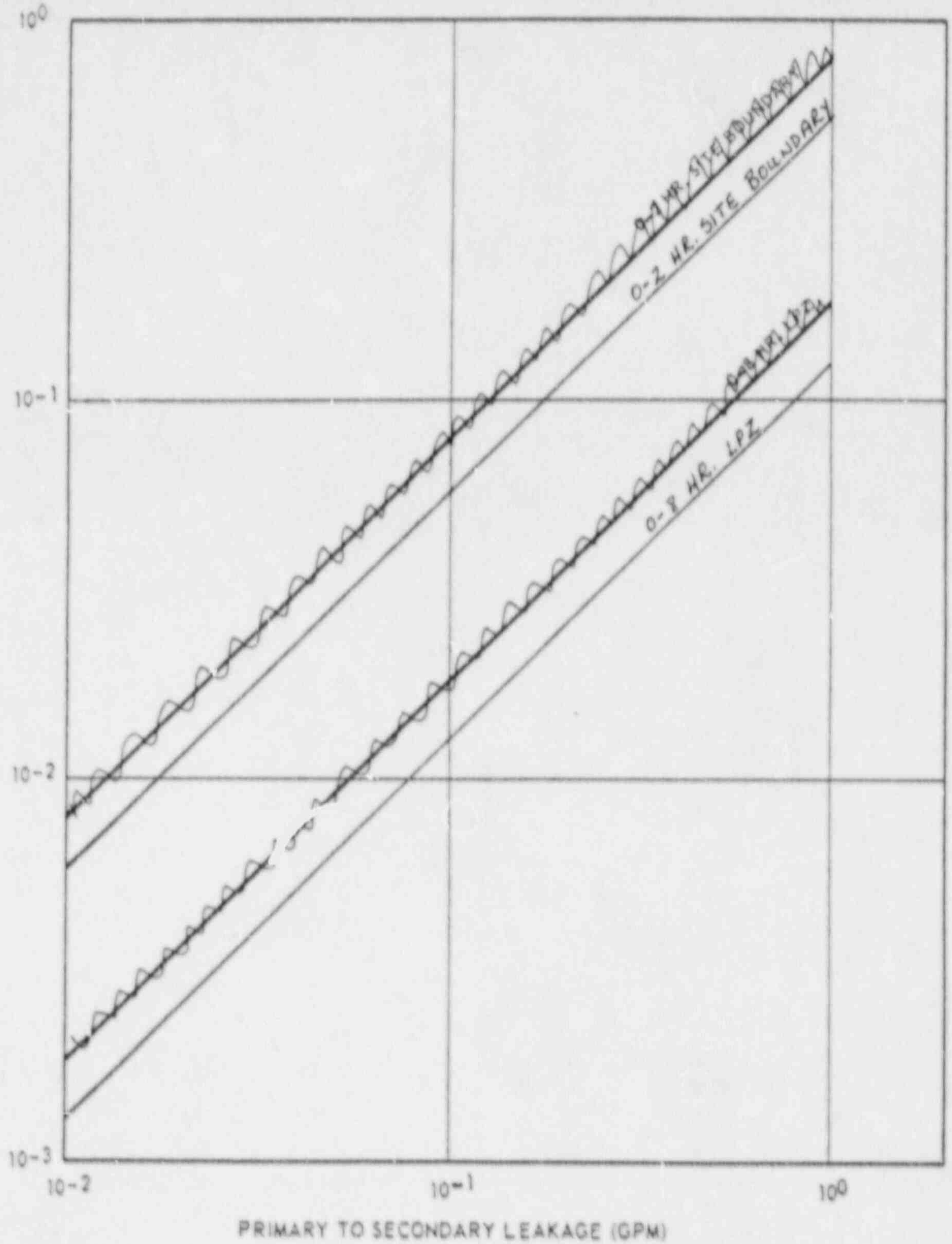
TABLE 15.4-34e

REACTOR COOLANT PUMP LOCKED ROTOR ISOTOPIC RELEASE TO ENVIRONMENT CONSERVATIVE ANALYSIS(1)

Isotope	Activity Released to Environment by Accident (Ci) (0-2 hr)	Activity Released to Environment by Accident (Ci) (0-8 hr)
I-131	2.12 ~~1.78~~ x 10^1	3.46 ~~7.11~~ x 10^1
I-132	3.03 ~~2.70~~ x 10^1	1.21 ~~1.08~~ x 10^2
I-133	4.29 ~~2.97~~ x 10^1	1.71 ~~1.58~~ x 10^2
I-134	4.62 ~~4.66~~ x 10^1	1.85 ~~1.06~~ x 10^2
I-135	3.96 ~~2.62~~ x 10^1	1.59 ~~1.45~~ x 10^2
Xe-131m	1.58 ~~1.04~~ x 10^1	6.32* ~~5.77~~ x 10^1
Xe-133	4.20 ~~4.24~~ x 10^3	1.68 ~~1.70~~ x 10^4
Xe-133m	6.33 ~~1.05~~ x 10^2	2.54 ~~4.25~~ x ~~10^2~~ 10^3
Xe-135	8.97 ~~1.12~~ x ~~10^3~~ 10^2	3.59 ~~4.49~~ x 10^3
Xe-135m	8.40 ~~1.10~~ x ~~10^3~~ 10^2	3.26 ~~4.41~~ x 10^3
Xe-138	3.54 ~~3.62~~ x 10^3	1.42 ~~1.49~~ x 10^4
Kr-83m	2.46 ~~3.30~~ x 10^2	9.87 ~~1.32~~ x ~~10^3~~ 10^2
Kr-85	5.67 ~~6.42~~ x 10^1	2.27 ~~2.57~~ x 10^2
Kr-85m	5.43 ~~2.92~~ x 10^2	2.18 ~~3.17~~ x 10^3
Kr-87	1.02 ~~1.53~~ x 10^3	4.02 ~~6.12~~ x 10^3
Kr-88	1.44 ~~2.18~~ x 10^3	5.75 ~~8.72~~ x 10^3
Kr-89	1.77 ~~2.82~~ x 10^3	7.05 ~~1.13~~ x ~~10^4~~ 10^3

Note:

(1) Primary to secondary leakage equal to 1.0 gpm.

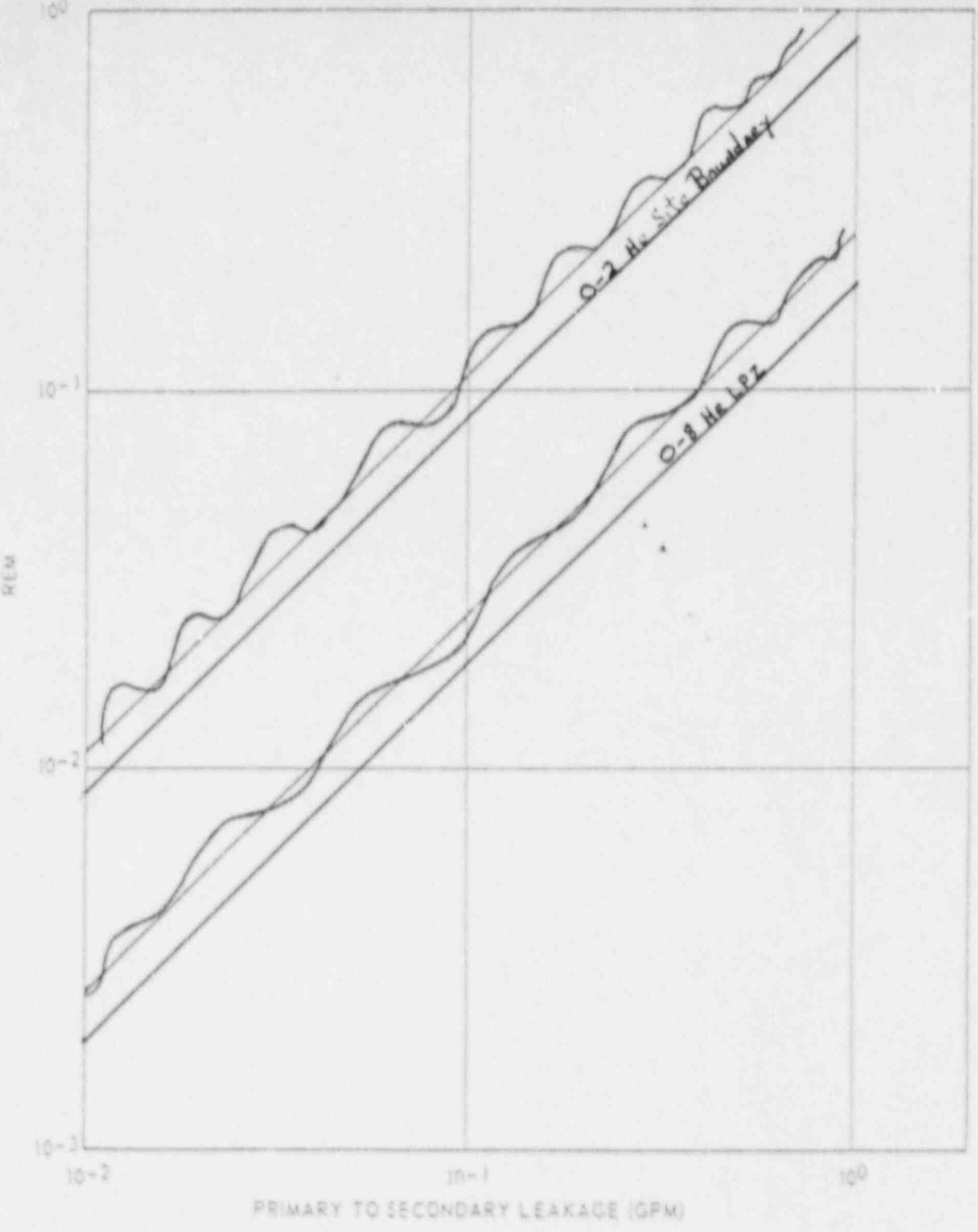


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Locked Rotor Accident Whole Body Gamma Dose

CONSERVATIVE CASE

Figure 15.4-77a

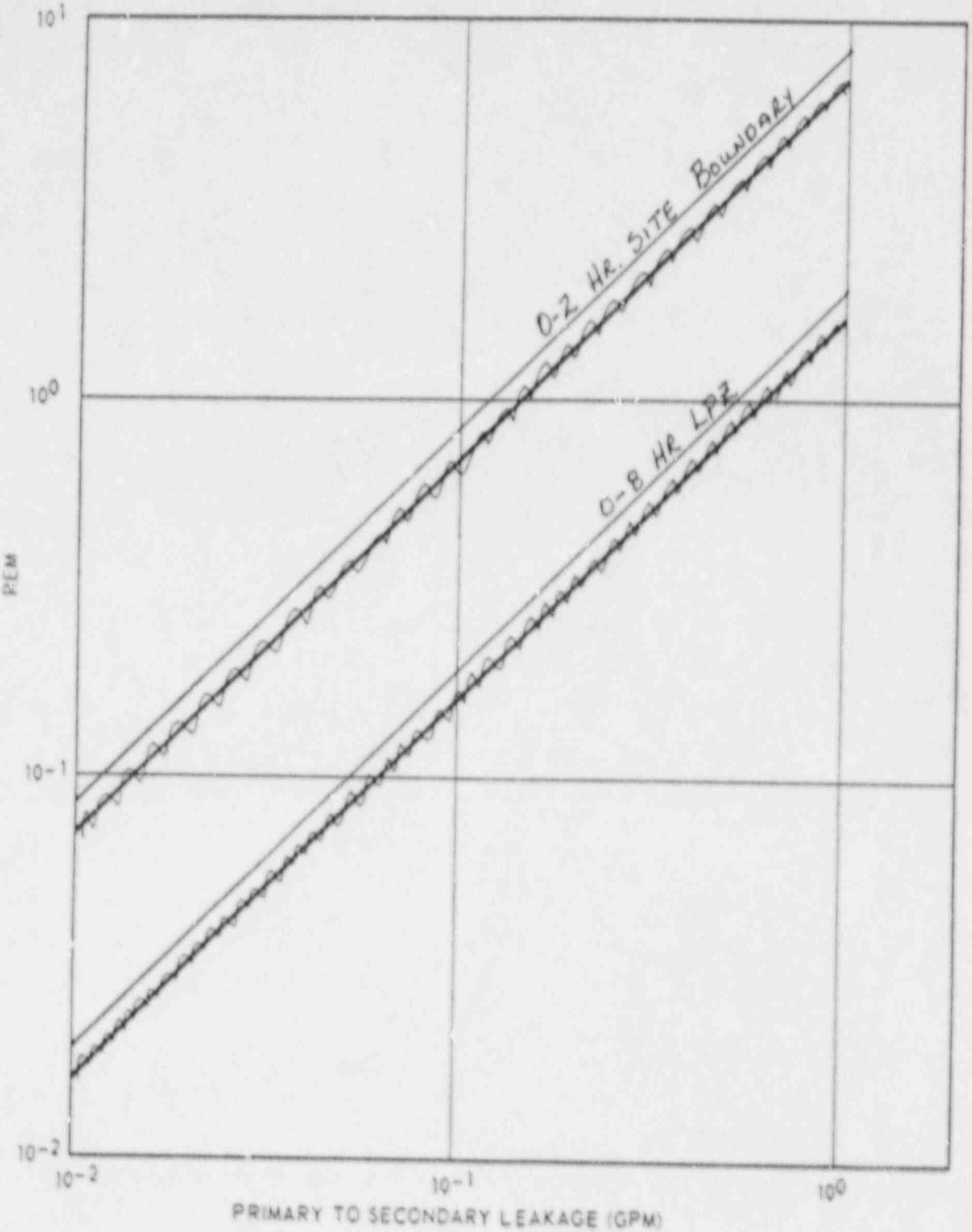


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Locked Rotor Accident Surface Body Beta Dose

Conservative Case

Figure 15.4-77b



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Locked Rotor Accident
Thyroid Dose
CONSERVATIVE CASE
Figure 15.4-77c

15.4.5.1.1 Identification of Causes and Accident Description

There are numerous administrative controls and physical limitations which are imposed to prevent a fuel handling accident from occurring during refueling operations. Nevertheless, an accident sequence has been postulated with the objective of assessing the potential risk to the public health and safety.

It is postulated that a spent fuel assembly is dropped onto the core during refueling resulting in breaching of the fuel rod cladding. As a result of the damage, a portion of the volatile fission gases are released to the water pool covering the core. Subsequently, a fraction of the water soluble gases are absorbed in the pool with the remainder being transported through the water and into the reactor building atmosphere. The escaped gases are assumed to be released instantaneously to the environment via the reactor building purge system and dispersed into the atmosphere.

15.4.5.1.2 Analysis of Effects and Consequences

15.4.5.1.2.1 Method of Analysis

The following assumptions are postulated in the calculation of the radiological consequences of a fuel handling accident inside containment:

Realistic Analyses

1. The accident occurs at 100 hours after reactor shutdown, which is the minimum time after shutdown that refueling operations could commence. Radioactive decay of the fission product inventory for this time period is taken into account.

2. All 264 pins in the dropped spent fuel assembly are damaged.

3. The assembly damaged is the highest-powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full-power operation at the end of core life immediately preceding shutdown. For the realistic analysis, nuclear characteristics of the highest-rated discharged assembly are given in Table ~~15.4-36.~~ 15.4-39 The model discussed in Section 15.1.7.2 is used to determine these fuel-pellet-cladding activities.

4. All activity in the clad gap of the damaged fuel, given in Table 15.4-36, is released to the reactor cavity pool.

5. The maximum fuel rod pressurization is 1200 psig.

6. The minimum water depth between the top of the damaged fuel rods and the reactor cavity pool surface is 23 feet.

7. Noble gases released to the reactor cavity pool are immediately released to the reactor building atmosphere.

The response time for the gas channel of RM-A4 to provide a closure signal directly to the interlocked reactor building purge isolation valves is based upon the following assumptions.

(a) Sample line length of 30 feet.
(b) Sample cavity of 0.04 ft^3.
(c) Sample flow of 1 cfm.
(d) Electronic and relay response of 0.56 seconds.

This results in a 13.2 second transient time for the sample from the duct to the detector. Adding the electronic and relay response produces a total of 13.8 seconds. Combining this with the closure time of the reactor building purge isolation valves (see Technical Specifications) of less than five seconds and adding an air flow time of 22.4 seconds from the fuel handling accident puff through the purge exhaust duct to the monitor sample point produces a total of less than 41.2 seconds from the occurrence of the postulated fuel handling accident puff inside containment until the reactor building purge is isolated.

15.4.5.4.2 Environmental Consequences of a Postulated Fuel Handling Accident Outside of Containment

Following a postulated fuel handling accident outside containment, a quantity of airborne radioactivity would be released to the environment via the fuel handling building charcoal exhaust system.

The dose received by an individual standing at the exclusion area boundary for the accident duration has been evaluated for both a conservative and a realistic case.

The bases for the conservative Regulatory Guide 1.25 evaluations are as follows:

1. The accident occurs 100 hours after plant shutdown, since plant technical specifications require the reactor to be subcritical 100 hours prior to the movement of spent fuel. Radioactive decay of the fission product inventory during the interval between shutdown and the start of refueling activities is taken into account.

2. The number of pins broken is a total of 314 pins. Being equivalent to 1.19 assemblies, this quantity of pins broken represents 50 pins broken in the impacted assembly as well as the 264 pins of the dropped assembly.

3. The assembly damaged is the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full power operation at the end of core life immediately preceding shutdown. ~~The power/temperature distribution and~~ Peaking factors used in both analyses are given in Table ~~15.4-35~~ 15.4-29. ~~Peak radial factors are given in FSAR Table 15.4-39~~.

4. The maximum fuel rod pressurization is 1200 psig.

5. The minimum water depth between the top of the damaged fuel rods and the spent fuel pool surface is 23 feet.

6. All activity in the clad gap, given in Table 15.4-37, is released to the spent fuel pool in the conservative analysis. This activity consists of 10 percent of the total noble gases other than Kr-85, 30 percent of the Kr-85, and 10 percent of the total radioactive iodine in the rods at the time of the accident. Activity available for release is given in Table 15.4-37.

7. Noble gases released to the spent fuel pool are immediately released to the atmosphere.

8. In the conservative analysis, the iodine gap inventory is composed of inorganic species (99.75%) and organic species (0.25%).

9. The spent fuel pool decontamination factor is 133 for inorganic iodines and 1 for all other radioisotopes.

10. No credit is taken for non-safety ventilation and purge systems.

11. No mixing of the pool releases with the fuel handling building atmosphere is assumed.

12. All iodine escaping from the spent fuel pool is immediately available for release to the environment.

13. No credit is taken for natural decay due either to holdup in the fuel handling building or after the activity has been released to the environment.

14. Isotopic data, breathing rate, and meteorology is given in Appendix 15A.

Assumptions used to evaluate the offsite dose of the realistic case are identical to those used for the conservative case with the following exceptions:

1. The number of pins broken in the realistic case is to 264. This is equivalent to one assembly. Gap activities are listed in Table 15.4-36.

2. Radial peaking factor is ~~1.55~~ 1.65 as listed in Table~~s~~ ~~15.4-35 and 15.4-39.~~ 15.4-39.

3. The effective decontamination factor for iodines in the spent fuel pool is 500 as given in Reference [41].

The activities released to the environment for the conservative and realistic models are given in Tables 15.4-36 and 15.4-37. The doses to the thyroid, whole body, and skin are given in Table 15.4-50.

DELETE

TABLE 15.4-35

NUCLEAR CHARACTERISTICS OF HIGHEST RATED DISCHARGED ASSEMBLY

Core Power, MWt	2900
Number of Assemblies	157
Core Average Assembly Power at 102% of Full Power, MWt	18.84

Highest Power Discharged Assembly

Axial Peak to Average Ratio	1.55
Radial Peak to Average Ratio	1.55
Peak Linear Power Density, kW/ft	13.65
Maximum Centerline Fuel Temperature, °F	3639

Power-Tem[illegible]ture Distribution for Highest Rated Discharged Assembly

Fuel Temperature Range, °F	Percent of Assembly Volume Within Temperature Range	Percent of Assembly Power within Temperature Range
3600 - 3800	0.04	0.07
3400 - 3600	0.6	1.1
3200 - 3400	1.4	2.2
3000 - 3200	2.2	3.3
2800 - 3000	3.1	4.4
2600 - 2800	4.1	5.5
2400 - 2600	5.4	7.0
2200 - 2400	6.9	8.5
2000 - 2200	8.9	10.3
<2000	67.36	57.63

TABLE 15.4-36

REALISTIC CASE

ACTIVITIES IN HIGHEST RATED ASSEMBLY AT ~~TIME OF~~ 100 HOURS AFTER REACTOR SHUTDOWN

Isotope	Curies in Assembly ~~$(x10^5)$~~	Percent of Activity in Gap	Curies in Gap ~~$(x10^3)$~~
Kr-83m	~~1.26~~ 0	~~0.156~~ 0.14	~~0.196~~ 0
Kr-85	~~0.101~~ 6.7×10^3	~~23.3~~ 21.2	~~2.35~~ 1.4×10^3
Kr-85m	~~3.03~~ 0	~~0.234~~ 0.21	~~0.709~~ 0
Kr-87	~~5.82~~ 0	~~0.127~~ 0.11	~~0.737~~ 0
Kr-88	~~8.29~~ 0	~~0.187~~ 0.17	~~1.55~~ 0
Kr-89	~~10.7~~ 0	~~0.0260~~ 0.023	~~0.279~~ 0
Xe-131m	~~0.0512~~ 5.5×10^3	~~1.86~~ 1.7	~~0.0954~~ 9.4×10^1
Xe-133	~~15.6~~ 1.2×10^6	~~1.25~~ 1.1	~~19.4~~ 1.3×10^4
Xe-133m	~~0.396~~ 1.0×10^5	~~0.821~~ 0.74	~~0.325~~ 7.4×10^2
Xe-135	~~4.26~~ 2.1×10^3	~~0.34~~ 0.31	~~1.45~~ 6.5×10^0
Xe-135m	~~4.19~~ 6.3×10^0	~~0.0574~~ 0.52	~~0.24~~ 3.3×10^{-2}
Xe-138	~~13.7~~ 0	~~0.0599~~ 0.54	~~0.823~~ 0
I-131	~~6.75~~ 5.8×10^5	~~1.53~~ 1.4	~~10.4~~ 8.1×10^3
I-132	~~10.2~~ 4.9×10^5	~~0.17~~ 0.15	~~1.74~~ 7.4×10^2
I-133	~~15.1~~ 6.1×10^4	~~0.51~~ 0.46	~~7.72~~ 2.8×10^2
I-134	~~17.7~~ 0	~~0.105~~ 0.095	~~1.87~~ 0
I-135	~~13.7~~ 4.0×10^1	~~0.29~~ 0.26	~~3.99~~ 1.0×10^{-1}

TABLE 15.4-37

CONSERVATIVE CASE ACTIVITIES IN HIGHEST RATED ASSEMBLY AT ~~TIME OF~~ 100 HOURS AFTER REACTOR SHUTDOWN

Isotope	Curies in Assembly ~~$(x10^5)$~~	Percent Activity in Gap (1)	Curies in Gap ~~$(x10^3)$~~
Kr-83m	~~1.40~~ 0	10	~~14.0~~ 0
Kr-85	~~0.085~~ 6.7×10^3	30	~~2.6~~ 2.0×10^3
Kr-85m	~~3.36~~ 0	10	~~33.6~~ 0
Kr-87	~~6.48~~ 0	10	~~64.8~~ 0
Kr-88	~~9.24~~ 0	10	~~92.4~~ 0
Kr-89	~~1.20~~ 0	10	~~12.0~~ 0
Xe-131m	~~0.057~~ 5.5×10^3	10	~~0.57~~ 5.5×10^2
Xe-133	~~17.34~~ 1.2×10^6	10	~~173.4~~ 1.2×10^5
Xe-133m	~~0.44~~ 1.0×10^5	10	~~4.4~~ 1.0×10^4
Xe-135	~~4.75~~ 2.1×10^3	10	~~47.5~~ 2.1×10^2
Xe-135m	~~4.67~~ 6.3×10^0	10	~~46.7~~ 6.3×10^{-1}
Xe-138	~~15.34~~ 0	10	~~153.4~~ 0
I-131	~~7.52~~ 5.8×10^5	10	~~75.2~~ 5.8×10^4
I-132	~~11.45~~ 4.9×10^5	10	~~114.5~~ 4.9×10^4
I-133	~~16.82~~ 6.1×10^4	10	~~168.2~~ 6.1×10^3
I-134	~~19.76~~ 0	10	~~197.6~~ 0
I-135	~~15.34~~ 4.0×10^1	10	~~153.4~~ 4.0×10^0

NOTE:

(1) In accordance with Regulatory Guide 1.25.

TABLE 15.4-39

REALISTIC AND CONSERVATIVE CASE NUCLEAR CHARACTERISTICS OF HIGHEST RATED DISCHARGED ASSEMBLY

I. CONSERVATIVE CASE	
Core Power, MWt	2900
Number of Assemblies	157
Highest Power Discharged Assembly	
Axial Peak to Average Ratio(1)	1.65
Radial Peak to Average Ratio(1)	1.65
II. REALISTIC CASE	
Core Power, MWt	2900
Number of Assemblies	157
Highest Power Discharged Assembly	
Axial Peak to Average Ratio	~~1.55~~ 1.65
Radial Peak to Average Ratio	~~1.55~~ 1.65

NOTE:

(1) In accordance with Regulatory Guide 1.25 (see Appendix 3A).

TABLE 15.4-40

ACTIVITY RELEASES FROM A FUEL HANDLING ACCIDENT INTO BUILDING

Isotope	Activity Released Conservative Case (Curies)	Activity Released Realistic Case (Curies)
Kr-83m	0	0
Kr-85	2.39×10^3 ~~3,092~~	1.42×10^3 ~~2,797~~
Kr-85m	0 ~~5.09×10^{-3}~~	0 ~~1.08×10^{-4}~~
Kr-87	0	0
Kr-88	0 ~~1.50×10^{-6}~~	0 ~~2.51×10^{-8}~~
Kr-89	0	0
Xe-131m	6.55×10^2 ~~774~~	9.35×10^1 ~~130~~
Xe-133	1.43×10^5 ~~141,891~~	1.32×10^4 ~~15,875~~
Xe-133m	1.19×10^4 ~~2,190~~	7.40×10^2 ~~162~~
Xe-135	2.50×10^2 ~~272~~	6.51×10^0 ~~0.3~~
Xe-135m	7.50×10^{-1} ~~0.929~~	3.30×10^{-2} ~~4.78×10^{-3}~~
Xe-138	0	0
I-131	6.90×10^2 ~~625.24~~	1.62×10^1 ~~17.3~~
I-132	5.83×10^2 ~~1.67×10^{-10}~~	1.47×10^0 ~~5.08×10^{-13}~~
I-133	7.26×10^1 ~~73~~	5.61×10^{-1} ~~0.67~~
I-134	0	0
I-135	4.76×10^{-2} ~~6.16×10^{-2}~~	2.10×10^{-4} ~~3.2×10^{-4}~~

TABLE 15.4-41

OFFSITE DOSES DUE TO POSTULATED FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

Dose Type	Conservative (R.G. 1.25) Case (Rem)	Realistic Case (Rem)
Thyroid	1.53 ~~1.35~~ x 10^{2}	3.43 ~~3.67~~ x 10^{0}
Gamma	1.40 ~~1.22~~ x 10^{0}	1.13 ~~4.00~~ x 10^{-1}
Beta	1.65 ~~1.62~~ x 10^{0}	1.73 ~~2.30~~ x 10^{-1}

NOTE: Dose receptor point located 1 mile away at exclusion boundary.

TABLE 15.4-50

OFFSITE DOSES DUE TO FUEL HANDLING ACCIDENT OUTSIDE CONTAINMENT

Dose Type	Conservative (R.G. 1.25) Case (Rem)	Realistic Case (Rem)
Thyroid	7.66 ~~6.75~~ x 10^{0}	1.72 ~~1.83~~ x 10^{-1} [1]
Gamma	1.40 ~~1.22~~ x 10^{0}	1.13 ~~4.00~~ x 10^{-1}
Beta	1.65 ~~1.62~~ x 10^{0}	1.73 ~~2.30~~ x 10^{-1}

NOTE: Dose receptor point located 1 mile away at exclusion boundary.

TABLE 15.4-51

FUEL HANDLING ACCIDENT OUTSIDE CONTAINMENT - ISOTOPIC RELEASE TO ENVIRONMENT

Isotope	Conservative (R.G. 1.25) Case (Ci)	Realistic Case (Ci)
I-131	3.45 ~~3.13~~ x 10^{1}	8.10 ~~8.70~~ x 10^{-1}
I-132	2.92x10^{1} ~~0~~	7.35x10^{-2} ~~0~~
I-133	3.63 ~~3.65~~ x 10^{0}	2.81 ~~3.35~~ x 10^{-2}
I-134	0	0
I-135	2.38 ~~3.08~~ x 10^{-3}	1.10 ~~1.60~~ x 10^{-5}
Xe-131m	6.55 ~~7.74~~ x 10^{2}	9.35 ~~1.30~~ x 10^{2} 1
Xe-133	1.43 ~~1.42~~ x 10^{5}	1.32 ~~1.59~~ x 10^{4}
Xe-133m	1.19 ~~2.19~~ x 10^{3} 4	[illegible] ~~1.62~~ x 10^{2}
Xe-135	2.50 ~~2.72~~ x 10^{2}	[illegible] ~~8.30~~ x 10^{0}
Xe-135m	7.50 ~~9.29~~ x 10^{-1}	5.30 ~~4.78~~ x 10^{-1} 2
Xe-138	0	0
Kr-83m	0	0
Kr-85	2.39 ~~3.09~~ x 10^{3}	1.42 ~~2.80~~ x 10^{3}
Kr-85m	0 ~~5.89 x 10^{3}~~	0 ~~1.08 x 10^{-4}~~
Kr-87	0	0
Kr-88	0 ~~1.50 x 10^{-6}~~	0 ~~2.51 x 10^{-8}~~
Kr-89	0	0

AMENDMENT 4
AUGUST, 1988

2. It is assumed that 50 percent of the iodines and 100 percent of the noble gases in the fuel that melts are released to the reactor coolant. This is a very conservative assumption since only centerline melting could occur for a maximum time period of six seconds.

3. The fraction of fuel melting is conservatively assumed to be one quarter of one percent of the core, determined by the following method:

   a. A conservative upper limit of 50 percent of the rods experiencing clad damage may experience centerline melting (a total of five percent of the core).

   b. Of rods experiencing centerline melting, only a conservative maximum of the innermost ten percent of the rod volume will actually melt (equivalent to 0.5 percent of the core that could experience melting).

   c. A conservative maximum of 50 percent of the axial length of the rod will experience melting due to the power distribution (0.5 of the 0.5 percent of the core equals 0.25 percent of the core).

The remainder of the assumptions and parameters used to calculate the activity release from the plant and the subsequent offsite doses for the ultraconservative analysis are identical to those used for the conservative analysis.

15.4.6.4.4 Results

Isotopic releases to the containment are summarized in Tables 15.4-44 through 15.4-46.

For the realistic analysis, the gamma, beta and thyroid doses at the site boundary are ~~4.90×10^{-8}~~ (1.03×10^{-7}) Rem, ~~3.74×10^{-8}~~ (1.42×10^{-7}) Rem and ~~2.16×10^{-5}~~ (3.12×10^{-5}) Rem, respectively. Corresponding doses at the low population zone are ~~6.06×10^{-8}~~ (1.95×10^{-7}) Rem, ~~9.64×10^{-5}~~ (6.24×10^{-7}) Rem and ~~8.32×10^{-5}~~ (1.01×10^{-4}) Rem, respectively.

For the conservative analysis, the gamma, beta and thyroid doses at the site boundary (0-2 hours) are ~~1.55×10^{-1}~~ (1.36×10^{-1}) Rem, ~~8.16×10^{-2}~~ (6.18×10^{-2}) Rem and ~~4.56×10^{1} Rem~~ (4.70×10^{1}), respectively. Corresponding doses at the low population zone (0-30 days) are ~~2.29×10^{-2}~~ (2.07×10^{-2}) Rem, ~~1.23×10^{-2}~~ (9.95×10^{-3}) Rem and ~~1.23×10^{1}~~ (1.30×10^{1}) Rem, respectively.

For the ultraconservative analysis, the gamma, beta and thyroid doses at the site boundary (0-2 hours) are ~~1.82×10^{-1}~~ (1.56×10^{-1}) Rem, ~~9.98×10^{-2}~~ (7.20×10^{-2}) Rem and ~~5.01×10^{1}~~ (5.28×10^{1}) Rem, respectively. Corresponding doses at the low population zone (0-30 days) are ~~2.64×10^{-2}~~ (2.36×10^{-2}) Rem, ~~1.47×10^{-2}~~ (1.15×10^{-2}) Rem and ~~1.39×10^{1}~~ (1.46×10^{1}) Rem, respectively.

These doses are well within the limits defined in 10 CFR 100 (25 Rem, whole body; 300 Rem, thyroid) at the site boundary and low population zone for the two hour and thirty day periods, respectively, after the accident.

TABLE 15.4-44

CONTROL ROD EJECTION ACCIDENT
ISOTOPIC RELEASE TO CONTAINMENT REALISTIC ANALYSIS

Isotope	Activity Released by Accident (Ci)
I-131	6.35 ~~5.33~~ x 10^{1}
I-132	6.58 ~~2.20~~ x 10^{1}
I-133	1.04 ~~7.72~~ x $10^{\not{1}\,2}$
I-134	1.52 ~~9.90~~ x $10^{\not{0}\,1}$
I-135	5.89 ~~3.85~~ x 10^{1}
Xe-131m	4.99 ~~3.30~~ x $10^{\not{0}\,1}$
Xe-133	5.89 ~~9.33~~ x $10^{\not{2}\,3}$
Xe-133m	3.85 ~~1.83~~ x $10^{\not{1}\,2}$
Xe-135	1.81 ~~5.70~~ x $10^{\not{1}\,2}$
Xe-135m	1.25 ~~2.75~~ x $10^{\not{0}\,1}$
~~Xe-137~~	~~1.83 x 10^{0}~~
Xe-138	1.63 ~~9.17~~ x $10^{\not{0}\,1}$
~~Kr-83m~~	~~4.20 x 10^{0}~~
Kr-85	2.06 ~~1.23~~ x $10^{\not{0}\,2}$
Kr-85m	4.54 ~~2.02~~ x 10^{1}
Kr-87	2.72 ~~1.19~~ x 10^{1}
Kr-88	8.16 ~~3.85~~ x 10^{1}
~~Kr-89~~	~~1.06 x 10^{0}~~

TABLE 15.4-45

CONTROL ROD EJECTION ACCIDENT

ISOTOPIC RELEASE TO CONTAINMENT CONSERVATIVE ANALYSIS

Isotope	Activity Released by Accident (Ci)
I-131	7.80 ~~7.10~~ x 10^{5}
I-132	1.12 ~~1.09~~ x 10^{6}
I-133	1.58 ~~1.60~~ x 10^{6}
I-134	1.70 ~~1.88~~ x 10^{6}
I-135	1.46 x 10^{6}
Xe-131m	5.40 ~~5.80~~ x 10^{3}
Xe-133	1.50 ~~1.71~~ x 10^{6}
Xe-133m	2.30 ~~4.27~~ x ~~10^{4}~~ 10^{5}
Xe-135	3.30 ~~4.53~~ x 10^{5}
Xe-135m	3.10 ~~4.44~~ x 10^{5}
Xe-138	1.30 ~~1.46~~ x 10^{5}
Kr-83m	9.10 ~~1.33~~ x ~~10^{5}~~ 10^{4}
Kr-85	6.40 ~~9.65~~ x 10^{3}
Kr-85m	2.00 ~~3.20~~ x 10^{5}
Kr-87	3.70 ~~6.17~~ x 10^{5}
Kr-88	5.30 ~~8.80~~ x 10^{5}
Kr-89	6.50 ~~1.46~~ x ~~10^{6}~~ 10^{5}

TABLE 15.4-46

CONTROL ROD EJECTION ACCIDENT

ISOTOPIC RELEASE TO CONTAINMENT ULTRACONSERVATIVE ANALYSIS

Isotope	Activity Released by Accident (Ci)
I-131	8.78 ~~8.05~~ x 10^5
I-132	1.26 ~~1.23~~ x 10^6
I-133	1.78 ~~1.80~~ x 10^6
I-134	1.91 ~~2.12~~ x 10^6
I-135	1.64 x 10^6
Xe-131m	6.75 ~~7.16~~ x 10^3
Xe-133	1.88 ~~2.12~~ x 10^6
Xe-133m	2.88 ~~5.32~~ x ~~10^4~~ 10^5
Xe-135	4.13 ~~5.66~~ x 10^5
Xe-135m	3.88 ~~5.55~~ x 10^5
Xe-138	1.63 ~~1.83~~ x 10^6
Kr-83m	1.14 ~~1.66~~ x 10^5
Kr-85	8.00 ~~1.04~~ x ~~10^4~~ 10^3
Kr-85m	2.50 ~~4.00~~ x 10^5
Kr-87	4.63 ~~1.71~~ x 10^5
Kr-88	6.63 ~~1.10~~ x ~~10^6~~ 10^5
Kr-89	8.13 ~~1.43~~ x ~~10^6~~ 10^5