ATTACHMENT 6 to TXX-05127

Mark up of FSAR

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Regulatory Guide 1.4

Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors.

Discussion

The analysis of the radiological consequences of the loss-of-coolant accident presented in Section 15.6.5 ~~complies with the requirements of Revision 2 (6/74) of this regulatory guide except that only gamma radiation contribution is taken into account in the determination of whole body exposures.~~ follows the guidance provided in Regulatory Guide 1.195 instead of Regulatory Guide 1.4.

Regulatory Guide 1.5

Assumptions Used for Evaluating Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors.

Discussion

This regulatory guide is not applicable to the CPSES which has pressurized water reactor steam supply systems.

Regulatory Guide 1.6

Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems

Discussion

The CPSES design complies with the requirements of Safety Guide 6 (3/10/71). For details see Section 8.3.

Regulatory Guide 1.7

Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident

Discussion

The CPSES design of the hydrogen recombiners and Hydrogen Purge System meet the requirements of Revision 2 (11/78) of this regulatory guide as discussed in Section 6.2.5, with the following exceptions and justifications.

Part C.4

The CPSES design takes exception to this regulatory position by de-classifying portions of the Hydrogen Purge System (HPS) which contain filters to seismic category II. This means that the filter unit is not required to be functional after a seismic event but it must remain in place. This is consistent with the accident scenarios postulated at CPSES. A LOCA is not postulated to occur coincidentally with a seismic event.

Refer to Appendix 1A(N).

Regulatory Guide 1.23

Onsite Meteorological Programs

Discussion

The meteorological monitoring program at CPSES complies with the requirements and those applicable recommendations of the Second Proposed Revision 1 to Regulatory Guide 1.23 (April, 1986) as discussed in Section 2.3. Refer to the CPSES Offsite Dose Calculation Manual (ODCM), Section I, 4.3.3.6 and Bases 3/4.3.3.6 for an exception.

Refer to Section 2.3 for a description of the design and siting of the primary meteorological tower. The quality assurance program for meterological monitoring is identified in FSAR Table 17A-1 and Section 17.2.

Regulatory Guide 1.24

Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure

Discussion

The analysis of the radiological consequences of the radioactive gas storage tank failure accident presented in Section 15.7.1 complies with the requirements of Safety Guide 24 (3/23/72) except that only gamma radiation contribution is taken into account in the determination of whole body exposures and the dose calculation methodology is consistent with Regulatory Guide 1.195.

Regulatory Guide 1.25

Assumptions used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors

Discussion

The analysis of the radiological consequences of the fuel handling accident ~~inside the Fuel Building~~ presented in Section 15.7.4 ~~complies with the requirements of Safety Guide 25 (3/23/72) except as follows:~~ follows the guidance provided in Regulatory Guide 1.195 instead of that in Regulatory Guide 1.25.

~~1.~~ ~~No iodine adsorber efficiency has been considered.~~

~~2.~~ ~~Only gamma radiation contribution is taken into account in the determination of whole body exposures.~~

Regulatory Guide 1.26

Quality Group Classifications and Standards for Water-, Steam-, and

Design Basis Tornado for Nuclear Power Plants

Discussion

The CPSES is designed to conform to the requirements of this regulatory guide, dated April 1974, except that it is designed to withstand the effects of a Design Basis Tornado having a maximum wind speed of 360 mph which is made up of a rotational speed of 300 mph and a translational speed of 60 mph. A simultaneous pressure drop of 3.0 psi at the rate of 1.0 psi per second is considered.

The Design Basis Tornado for CPSES was determined prior to the issuance of this Regulatory Guide and was approved for use by the Atomic Energy Commission's Safety Evaluation Report Dated September 3, 1974.

Also refer to Section 3.3.

Regulatory Guide 1.77

Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors

Discussion

The analysis of the radiological consequences of a control rod ejection accident presented in Section 15.4.8 ~~complies with the requirements of the Appendix B assumptions of this regulatory guide, dated May 1974, except only gamma radiation contribution is taken into account in the determination of whole body exposures.~~ follows the guidance provided in Regulatory Guide 1.195 instead of that in Regulatory Guide 1.77.

Regulatory Guide 1.78

Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release

Discussion

The CPSES design meets the intent of this regulatory guide, dated June 1974, as discussed in Sections 2.2 and 6.4.

Regulatory Guide 1.79

Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors

Discussion

The Initial Test Program, as described in Section 14.2, is in compliance with the provisions of Revision 1 (9/75) of this regulatory guide with the following exception.

This regulatory guide is not applicable to CPSES.

Regulatory Guide 1.146

Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plant

Discussion

Effective June 1, 1983, CPSES Quality Assurance Audit Personnel are in compliance with regulatory guide 1.146, August 1980.

Regulatory Guide 1.148

Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants

Discussion

CPSES is not committed to this Regulatory Guide. The operability program for active valves is covered in Section 3.9N and 3.9B.

Regulatory Guide 1.150

Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations.

Discussion

Refer to Appendix 1A(N).

Regulatory Guide 1.155

Station Blackout

Discussion

CPSES complies with the guidance of Regulatory Guide 1.155 (August 1988) as described in Appendix 8B.

Regulatory Guide 1.163

Performance-Based Containment Leak-Test Program

Discussion

Effective by August 12, 1996, CPSES complies with Regulatory Guide 1.163, September 1995.

INSERT A

Insert A (after page 1A(B)-61)

Regulatory Guide 1.195

Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light- Water Nuclear Power Reactors, May 2003.

Discussion

The radiological consequences analysis for the design basis accidents follow the guidance provided in Regulatory Guide 1.195 and CPSES adopts the dose limits defined by Regulatory Guide 1.195. The CPSES design for skin dose calculations DCFs uses DOE/EH-0070 instead of Figure 12 recommended in Regulatory Guide 1.195.

Regulatory Guide 1.196

Control Room Habitability at Light-Water Nuclear Power Reactors, May 2003

Discussion

The CPSES control room habitability program follows the guidance provided in Regulatory Guide 1.196 with the following exceptions:

1. An acceptable alternative to the specific TS change identified in RG 1.196 is used which includes CRE integrity testing and periodic assessments. A Control Room Integrity Program is implemented.

2. CPSES currently is using RG 1.52, Revision 1 for the control room design. CPSES used Revision 2 for testing only.

3. CPSES uses RG 1.140 as information only for non-safety related air filtration systems.

Regulatory Guide 1.197

Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors, May 2003

Discussion

CPSES uses RG 1.197 dated May 2003 for testing of the CRE. CPSES has successfully completed comparison testing as described in section 1.2 of RG 1.197. Individual component tests will be performed in the future to determine the total CRE leakage as necessary. If future design changes warrant another ASTM E741 test, they will be performed with the exceptions as defined in NEI 99-03 Revision 1, Appendix EE.

6.4 HABITABILITY SYSTEMS

6.4.1 DESIGN BASES

6.4.1.1 Control Room Envelope

The Control Room pressurized envelope, as defined in Section 6.4.2.1, includes the Control Room and all areas adjacent to the Control Room on elevation 831' 6" of the Electrical and Control Building containing plant information and equipment that may be needed during an emergency including kitchen, sanitary facilities, and computer rooms.

Control Room design is based upon the safe occupation of the Control Room envelope during normal operation and for a period of not less than 30 days after a loss-of-coolant accident (LOCA). Habitability systems ensure that the personnel ~~occupying the~~ Control Room during these times will not be ~~exposed~~ to radiation ~~doses~~ exceeding 5 rem whole body gamma dose, ~~30~~ rem thyroid dose, and ~~30~~ rem beta skin dose. The allowable unprotected beta skin dose may be increased to 75 rem when special protective clothing and eye protection is used. The Control Room is designed in accordance with NRC General Design Criterion (GDC) 19 [1]. The Control Room envelope contains adequate medical supplies and the necessary kitchen and sanitary facilities to sustain plant personnel for a period of 30 days following a DBA. The necessary food and water for five plant personnel for five days will be permanently stored in the Control Room. 50

6.4.1.2 Radiation and Toxic Gas Protection

Control Room shielding is designed to limit the dose from external sources to a level compatible with the dose criteria given in Subsection 6.4.1.1 based on the inventories given in Table 6.4-1.

The Control Room HVAC system is designed to maintain a positive pressure with respect to the environs during normal and emergency modes of operation.

Airborne radioiodine is limited to levels compatible with the dose criteria given in Subsection 6.4.1.1, based on the radioiodine activities given in Table 6.5-6, and a Containment leak rate of 0.1- percent for the first 24 hr following an accident and one-half this value for the balance of the accident. Refer to Subsection 15.6.5.4 for an analysis of the inhalation dose to the Control Room operators. In the event of a toxic gas release, the control room may be manually isolated from the outside environment, by entering a complete recirculation mode of operation. For CPSES, the probability of simultaneous occurrence of a toxic gas release and radiological release caused by a loss of coolant accident (LOCA) is assumed to be extremely low. Therefore, the event of concurrent releases is not considered in the design basis. See Section 2.2.3 for a discussion of toxic gas releases and analyses.

Airborne radioactive material in the Control Room atmosphere is controlled after an accident by the emergency recirculation filtration units and emergency pressurization

6.4.2.5 Shielding Design

The shielding design for the Control Room is based on the requirements specified in 10 CFR Part 50, Appendix A, GDC 19. The Control Room is designed to provide radiation protection for personnel occupancy under accident conditions so that no individual will receive exposures in excess of 5-rem whole-body gamma dose, ~~30~~-rem thyroid dose and 75-rem unprotected beta skin dose (with special protective clothing and eye protection). 50

To achieve this goal, the shielding design of the Control Room considers airborne contaminants within the Control Room and other DBA sources of radiation. Specifically, these other sources are fission products released to the reactor Containment atmosphere, airborne radioactive contaminants surrounding the Control Room, and sources of radiation caused by potentially contaminated equipment in the vicinity of the Control Room. They are considered to be dominant sources of radiation, and they are among the principal parameters for the shielding design of the Control Room.

Shield thicknesses of structural concrete provided for the Control Room are shown on Figure 12.3-14. The 2 ft-0 in. structural shielding walls surrounding the Control Room, combined with the roof and floor slabs above the Control Room, provide more radiation protection for personnel in the Control Room. In addition to shield thicknesses, distances that separate dominant radiation sources from the Control Room are included on the scaled layout and arrangement drawings of the facility in Section 1.2. Radioactive decay for each isotope of the DBA source is taken into consideration in the analysis of the dose to Control Room occupants shown in Subsection 15.6.5.4. A layout drawing of the Control Room and associated structures is presented on Figure 1.2-33.

DBA sources of radiation surrounding the Control Room and shielding related considerations are presented in Section 12.2, Section 12.3, and Subsection 15.6.5.4. A plan view drawing of the Control Room and associated structures identifying distances and shield thicknesses is shown in Figure 12.3-14.

6.4.3 SYSTEM OPERATIONAL PROCEDURES

The following modes of operation characterize the Control Room Air-Conditioning System:

1. Normal operation

2. Emergency recirculation

3. Emergency ventilation

4. Isolation (emergency recirculation without pressurization)

The Control Room Air-Conditioning System is automatically switched to the emergency recirculation mode upon receipt of signals as outlined in Section 9.4. In addition, the system may be manually switched to the emergency ventilation mode of operation by the operator from the Control Room ventilation panels. This feature enables the removal of and consistent with NRC Regulatory Guide 1.196 [11]

6.4.4.3 Evaluation of Heating, Ventilation, Air-Conditioning, and Filtration System

The HVAC and Filtration System readiness is ensured by the periodic testing program described in Section 6.4.5. Safe operation is ensured by having redundant equipment for the Control Room HVAC and Filtration System. A complete safety evaluation is given in Section 9.4.

6.4.5 TESTING AND INSPECTION

Preoperational tests are conducted on the Control Room HVAC and Filtration System to ensure that all equipment satisfies the design criteria during all modes of operation. Tests are also performed, as described in Section 9.4, to ensure overall system performance. The leakage tests will be conducted by closing all the access points to the Control Room.

Control Room pressure will be established by controlling the outside air intake flow of the emergency pressurization units until the design pressure is achieved. Should the outside makeup airflow through the emergency pressurization unit exceed the maximum allowable flow of approximately 800 scfm, a survey shall be conducted to locate points of excessive leakage and attempt to seal them. Tests shall be repeated as often as necessary until the above criteria are established.

Control Room pressure is measured by the permanently installed differential pressure transmitters.

The result of the Control Room leak test is considered acceptable if the emergency pressurization airflow does not exceed 800 scfm with the Control Room envelope being maintained at 0.125-in. wg.

Planned leakage tests will be performed to verify that adequate pressurization of the Control Room envelope is maintained during Emergency Recirculation mode. The Control Room envelope will be maintained at 0.125-in. wg positive pressure relative to the outside atmosphere with a maximum makeup airflow of approximately 800 scfm.

In-place testing of air cleaning components will be performed in accordance with test methods and acceptance criteria described in ANSI N510 [7].

Control Room equipment will also be tested in accordance with the methods described in ANSI N509 [6].

Testing requirements, acceptability and frequencies for ESF components are established in the Technical Specifications.

6.4.6 INSTRUMENTATION REQUIREMENTS

Sufficient indications are provided in the Control Room for the operator to monitor HVAC system performance. Annunciators indicate HVAC system or component

The Control Room Emergency Filtration System and the Control Room Envelope are tested as part of the Control Room Integrity Program as established in the Technical Specifications.

malfunctions. See Sections 7.3 and 9.4 for more detailed discussions of the instrumentation.

Fire protection and alarm devices are annunciated in the Control Room as described in Subsection 9.5.1.

Ionization and radiation sensors detect smoke and unsafe levels of radiation in the incoming air. The system automatically switches to emergency recirculation if a radiation detector fails. Subsequent to automatic initiation of the emergency recirculation mode, the operator can regain manual control over the system from the Control Room ventilation panels. A Control Room vertical panel mounted selector switch allows the operator to select normal operation or emergency ventilation modes as required for the operation conditions.

REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 19, Control Room.

2. NRC Regulatory Guide 1.4, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors. Revision 2, June 1974, United States Nuclear Regulatory Commission.

3. NRC Regulatory Guide 1.52, Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants, Revision 1, July 1976, United States Nuclear Regulatory Commission.

4. NRC Regulatory Guide 1.78, Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, June 1974, United States Nuclear Regulatory Commission.

5. Deleted.

6. ANSI N509, Nuclear Power Plant Air Cleaning Units and Components.

7. ANSI N510, Field Testing of Nuclear Air Cleaning Systems.

8. Leakage Characteristics of Openings for Reactor Housing Components, NAA-SR-MEMO-5137, Atomics International, Div. of North American Aviation, Inc., June 20, 1960.

9. Deleted.

10. ANSI/ASHRAE 15-78, Safety Code for Mechanical Refrigeration.

11. NRC Regulatory Guide 1.196, Control Room Habitability at Light-Water Nuclear Power Reactors, May 2003.

CPSES/FSAR

TABLE 6.4-1

TO CONTAINMENT

NOBLE GAS AND HALOGEN INVENTORIES RELEASED TO CONTAINMENT AS THE RESULT OF A MAXIMUM CREDIBLE ACCIDENT

Noble Gases			Halogens		
Curies Released From Core			Curies Released From Core		
Nuclide		(100% Core Inventory)	Nuclide		(50% Core Inventory)[a]
Kr-83m	1.3	1.2×10^{7}	Br-82		1.5×10^{5}
Kr-85m	3.0	2.7×10^{7}	Br-83		6.0×10^{6}
Kr-85	7.3	6.6×10^{5}	Br-84		1.0×10^{7}
Kr-87	5.4	4.9×10^{7}			
Kr-88	7.7	7.0×10^{7}	I-130		8.8×10^{5}
Kr-89		8.7×10^{7}	I-131	5.4	4.9×10^{7}
			I-132	7.9	7.2×10^{7}
Xe-131m	7.7	7.0×10^{5}	I-133	1.1	1.0×10^{8}
Xe-133m	3.2	2.9×10^{7}	I-134	1.2	1.1×10^{8}
Xe-133	2.2	1.9×10^{8}	I-135	1.0	$9.4 \times 10^{\cancel{7}\,8}$
Xe-135m	4.4	4.0×10^{7}			
Xe-135	4.6	4.2×10^{7}			Kilograms Released
Xe-138	1.8	1.6×10^{8}	Nuclide		From Core
			I-127		1.4
			I-129		5.7[b]

Note: Per R.G. 1.195

a. ~~H~~alf of the released halogens iodines in the core are assumed plated out with the remainder airborne and available for release from containment immediately. to be available for release to the containment atmosphere.

b. Equivalent to 1.007 Ci

TABLE 6.4-3

LIMITATIONS OF CONTROL ROOM ENVIRONMENT

Ambient pressure, in. wg	+0.125*
Ambient temperature, FDB	75 ±5
Ambient relative humidity, percent	35 to 50
Noxious substances	Not Applicable
Maximum Radiation doses to operators:	
Whole Body Gamma Dose	5 rem
Beta Skin Dose	75 rem**
Thyroid Inhalation Dose	50 ~~30~~ rem ***

* Except during detection of smoke or toxic gas, at which time the operator may manually initiate control room isolation mode and maintain the control room at atmospheric pressure.

** The 75 rem limit for beta skin dose is allowed due to the use of protective clothing and eye protection. The dose limit is 50 rem if credit is not taken for protective clothing.

*** Regulatory Guide 1.195 Control Room Thyroid limit.

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TABLE 6.4-4

(Sheet 2)

POTENTIAL LEAK PATHS AND THEIR APPROPRIATE LEAKAGE CHARACTERISTICS*

Components	Description of Components	Leak Rate (ft^3/min)
Unidentified Leakage	Fan margin	464
TOTAL		800

* The criteria used to establish leakages is based on a pressure differential of 1/4 in.wg.as specified in NRC Regulatory Guide 1.78[4].

i.

Note: The Control Room is pressurized to 1/8 in. wg. in the Emergency Recirculation mode of operation. The leakrate will be ≤800 ft^3/min at 1/8 in. wg. positive pressure.

2. Total in-leakage is confirmed to be consistent with NRC Regulatory Guide 1.196 [11].

h = drop fall height

E = total collection efficiency for a single drop

F = spray flow rate

d = drop diameter

V = containment spray volume

modeled using a value of 10/m (this value is reduced to 1.0/m after the airborne activity is reduced by a factor of 50).

This model embodies assumptions equivalent to most assumptions applied to the washout of elemental iodine. The drop collection efficiency is ~~evaluated using E = 0.0015; this minimum value underpredicts the initial washout by several orders of magnitude [1].~~ The particulate iodine removal coefficient is listed in Table 6.5-5.

. The most conservative

For each region calculated, elemental iodine removal coefficients are above 10 hr^{-1}, respectively. The removal coefficients used in LOCA offsite dose calculation (see Section 15.6.5.3) have been limited to a maximum value of 10 hr^{-1} for elemental iodine [1] ~~and the most conservative~~ particulate iodine removal

of 11.4

coefficient ~~1.07~~ hr^{-1} is also taken into account on the LOCA offsite dose calculation until the inventory is reduced by a factor of 50, after which a value of 1.14 hr -1 is used.

4. Single Failure Analysis

The containment spray chemical additive subsystem is designed to function reliably in the event of a LOCA. Redundancy requirements are met by providing two electrically separated motor-operated chemical additive tank stop valves in parallel such that, in case of single active failure, there still remains a full 100-percent capacity to reduce the elemental iodine concentration inside the Containment. In addition, an air-operated normally open valve is provided in series with each of the motor-operated valves to ensure that the line may be closed when the chemical additive tank is empty and thus prevent air from being entrained in the suction of the containment spray pumps. There is no credit for the air-operated valve (passive, fails open) in the pH analysis. A single failure analysis is presented in Table 6.5-4 for the chemical additive subsystem.

Reproductibility of subsystem design parameters is ensured by using performance tested chemical eductors and spray nozzles, as well as constant speed containment spray pumps. Furthermore, the only moving parts in the system are those associated with the pumps, valves, and motors. System component parameters are verified by testing.

TABLE 6.5-5

CONTAINMENT SPRAY SYSTEM CHARACTERISTICS

	Regions					
	A	B	C	D	E	Total
Total volume, ft^3	2,309,461	167,645	72,686	124,910	356,333	3,031,035
Sprayed containment volumes, ft^3	1,664,901	35,040	2,640	3,364	0.0	1,705,945
Unsprayed containment volumes ft^3	644,560	132,605	70,046	121,546	356,333	1,325,090
Spray drops fall height, ft	126[(a)]	38	23	20	-	-
Number of nozzles per train [(b)]	274 (272-Unit 1 train A) (273-Unit 1 train B)	67 [36](c) (64-Unit 2 train A) (67-Unit 2 train B)	14 [5](c) (14-Unit 2 train A) (13-Unit 2 train B)	27 [10](c) (23-Unit 2 train A) (24-Unit 2 train B)	-	382 (380-Unit 1 train A) (381-Unit 1 train B) (375-Unit 2 train A) (378-Unit 2 train B)
Elemental iodine removal coefficient, hr^{-1} (d)	~~25.92~~ 60.4	~~83.7~~ 106.9	~~62.1~~ 119.4	~~90.72~~ 162.9		
Particulate iodine removal coefficient, hr^{-1}	~~1.07~~ 11.4	~~3.45~~ 38.7	~~2.57~~ 63.2	~~3.76~~ 76.3	-	-

(a) Average value

(b) 15.2 gpm/nozzle

(c) Unobstructed spray nozzle (train A)

(d) Maximum Xe allowed to be used in analysis is 10 hr^{-1}.

TO OUTSIDE ATMOSPHERE FROM THE CONTAINMENT

TABLE 6.5-6

RADIOACTIVE IODINE ISOTOPES AVAILABLE FOR RELEASE FOLLOWING A LOCA

Isotope	Activity* (Ci) (x10[7])	
I-131	~~2.45~~	2.7
I-132	~~3.6~~	4.0
I-133	~~5.0~~	5.5
I-134	~~5.5~~	6.1
I-135	~~4.7~~	5.2

1.195

* Based on NRC Regulatory Guide ~~1.4~~ assumptions, ~~this is the activity available for release from containment atmosphere.~~

half of the iodine activity released from the core to the containment is available for release to the outside atmosphere.

TABLE 6.5-7

ESF FILTRATION UNITS EMPLOYED DURING DESIGN BASIS ACCIDENTS

Design Basis Accident		Filtration Units Employed	Reference Section
1.	Loss-of-coolant accident	Control Room Emergency pressurization units	15.6.5.4(4)(b)
		Emergency filtration units	15.6.5.4(4)(b)
		Primary plant exhaust unit	15.6.5.4(2)(f)
2.	Waste gas system failure	Control Room Emergency Pressurization Unit and Emergency Filtration Units	15.7.1
3.	Steam generator tube rupture	Control Room Emergency Pressurization Unit and Emergency Filtration Units	15.6.3
4.	Fuel handling accident	Control Room Emergency Pressurization Unit and Emergency Filtration Units	15.7.4
5.	Control rod ejection accident	Control Room Emergency Pressurization Unit and Emergency Filtration Units	15.4.8
6.	CVCS letdown line break	Control Room Emergency Pressurization Unit and Emergency Filtration Units	15.6.2
7.	Main Steam line	Control Room Emergency Pressurization Unit and Emergency Filtration Units	15.1.5
8.	Liquid waste tank failure	Control Room Emergency Pressurization Unit and Emergency Filtration Units	15.7.2
9.	Locked Rotor	Control Room Emergency Pressurization Unit and Emergency Filtration Units	15.3.3

15.0 ACCIDENT ANALYSIS

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List of Tables (Continued)

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List of Tables (Continued)

6

(Sheet 1 of 2)
CORE AND GAP ACTIVITIES

TABLE 15.0-7

CORE AND GAP ACTIVITIES BASED ON FULL POWER OPERATION FOR 650 DAYS[a]

Isotope	Core Activity (Curies)	Percent Activity in Gap	Gap Activity (Curies)
I-131	8.80×10^7	0.960	8.45×10^5
I-132	1.34×10^8	0.105	1.41×10^5
I-133	1.97×10^8	0.316	6.24×10^5
I-134	2.31×10^8	0.0652	1.50×10^5
I-135	1.79×10^8	0.180	3.22×10^5
Xe-131m	6.68×10^5	1.17	7.81×10^3
Xe-133	2.03×10^8	0.778	1.58×10^6
Xe-133m	5.16×10^6	0.510	2.63×10^4
Xe-135	5.55×10^7	0.210	1.17×10^5
Xe-135m	5.46×10^7	0.0355	1.94×10^4
Xe-138	1.79×10^8	0.0370	6.63×10^4
Kr-83m	1.64×10^7	0.0964	1.58×10^4
Kr-85	9.99×10^5	18.3	1.83×10^5
Kr-85m	3.95×10^7	0.145	5.72×10^4
Kr-87	7.59×10^7	0.0783	5.94×10^4
Kr-88	1.08×10^8	0.116	1.25×10^5
Kr-89	1.40×10^8	0.0161	2.26×10^4

a. This table is used to estimate activities. The activities used in the licensing basis calculations are shown in sheet 2.

for Locked Rotor and Fuel Handling Accidents

(Sheet 2)

CPSES/FSAR

CORE AND GAP ACTIVITIES

(15.0-7)

TABLE 15.6-8

CORE INVENTORY AND DESIGN BASES GAP ACTIVITIES~~(a)~~ (b)

Isotope	Curies in Core (x10^7)~~(b)~~ (c)	Percent of Activity in Gap	Curies in Gap (x10^7)	INSERT A	INSERT B
Kr-83m	1.32 ~~1.2~~	~~10~~ 5	0.066 ~~0.12~~	10	0.132
Kr-85	0.00726 ~~0.066~~	~~30~~ 10	0.00726 ~~0.0198~~	10	0.00726
Kr-85m	2.97 ~~2.7~~	~~10~~ 5	0.1485 ~~0.27~~	10	0.297
Kr-87	5.39 ~~4.9~~	~~10~~ 5	0.2695 ~~0.49~~	10	0.539
Kr-88	7.7 ~~7.0~~	~~10~~ 5	0.385 ~~0.70~~	10	0.770
Kr-89	8.7	10	0.87		
Xe-131m	0.077 ~~0.07~~	~~10~~ 5	0.00385 ~~0.007~~	10	0.0077
Xe-133	21.8 ~~19.0~~	~~10~~ 5	1.09 ~~1.90~~	10	2.18
Xe-133m	3.20 ~~2.9~~	~~10~~ 5	0.16 ~~0.29~~	10	0.32
Xe-135	4.60 ~~4.2~~	~~10~~ 5	0.23 ~~0.42~~	10	0.46
Xe-135m	4.40 ~~4.0~~	~~10~~ 5	0.22 ~~0.40~~	10	0.44
Xe-138	18.0 ~~16.0~~	~~10~~ 5	0.90 ~~1.60~~	10	1.80
I-131	10.8 ~~9.8~~	~~10~~ 8	0.864 ~~0.98~~	10	1.08
I-132	15.8 ~~14.4~~	~~10~~ 5	0.79 ~~1.44~~	10	1.58
I-133	22.0 ~~20.0~~	~~10~~ 5	1.10 ~~2.00~~	10	2.20
I-134	24.2 ~~22.0~~	~~10~~ 5	1.21 ~~2.20~~	10	2.42
I-135	20.7 ~~18.8~~	~~10~~ 5	1.035 ~~1.88~~	10	2.07

b ~~a.~~ Noble Gas and Iodine inventories used for design basis accident analysis.

c ~~b.~~ Three region equilibrium cycle core at end of life. The three regions have operated at a specific power of 40.03 MW/MTU for 300, 600, and 900 EFPD, respectively.

INSERT A Percent of Activity in Gap for Rod Ejection Accident

INSERT B Curies in Gap (x 10^7)

15.1.5.3 Environmental Consequences

The postulated accidents involving release of steam from the secondary system do not result in a release of radioactivity unless there is leakage from the RCS to the secondary system in the steam generators. A conservative analysis of the potential offsite doses resulting from this accident is presented considering equilibrium operation based upon Technical Specification limits of primary coolant activity concentration and primary to secondary leakage.

The radiological consequences analysis for a postulated rupture of a main steam line assumes a leak rate to the faulted steam generator during the accident that results in calculated consequences approaching a small fraction (10%) of the 10CFR100 guideline values for the accident-initiated iodine spike. This leak rate provides a maximum primary-to-secondary leak rate limit against which the predicted end-of-cycle leakage is compared.

The following assumptions and parameters are used to calculate the activity releases and offsite doses for a postulated main steam line break.

1. Offsite power is lost; thus, the main steam condenser is not available for steam dump.

2. Air ejector release and steam generator blowdown do not occur during the accident.

3. After 8 hours, steam and activity release to the environment are terminated due to primary and secondary side pressure equalization.

from the intact SGs is terminated at 8 hours

4. The release of activity that leaks into the faulted SG continues until the RCS is cooled below 212 degrees F (20 hours).

5 ~~4.~~ The elemental iodine partition factor in the affected steam generator throughout the accident is 1.0.

6 ~~5.~~ The elemental iodine partition factor in the unaffected steam generators throughout the accident is 0.01.

7 ~~6.~~ All releases are assumed to be at ground level.

8 ~~7.~~ Two separate cases are analyzed:

a. An iodine spike occurs coincident with the steam line break. The iodine appearance rates from the fuel are increased to 500 times the equilibrium appearance rates associated with a primary coolant concentration of 0.45 uCi/gm Dose Equivalent I-131. The spike appearance rates are given in Table 15.1-3.

1.0

b. A transient has occurred prior to the steam line break. The transient causes an iodine spike that raises the primary coolant activity to 60 uCi/gm Dose Equivalent I- 131. The preaccident spike concentrations are given in Table 15.1-3.

8. Other assumptions are in Table 15.1-3.

0.001 be 1.2 rem and 0.002 rem

0.61

Based on the foregoing model, the thyroid doses and whole body doses for the pre-accident iodine spike case were calculated to 44.0 rem and 0.10 rem, respectively, at the EAB and ~~23.0~~ rem and 0.04 rem, respectively, at the LPZ. These doses are below the applicable limit values of 300 rem thyroid and 25 rem whole body set forth in 10CFR100. The thyroid doses and whole body doses for the accident-initiated iodine spike case were calculated to be 26.9 rem and 0.15 rem, respectively, at the EAB and 21.5 rem and 0.10 rem, respectively, at the LPZ. These doses are below a "small fraction" of the 10CFR100 values; or 30 rem thyroid and 2.5 rem whole body.

2.5 0.004 1.4 rem and 0.004

15.1.5.4 Conclusions

The analysis has shown that the criteria stated earlier in Section 15.1.5.1 are satisfied.

Although DNB and possible clad perforation following a steam line rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis shows that no DNB or clad perforation occurs for any main steam line rupture assuming the most reactive RCCA stuck in its fully withdrawn position. The radiological consequences are within the guideline values of 10CFR100 for the pre-accident iodine spike case and within a "small fraction" (10%) of the 10CFR100 guideline values for the accident-initiated iodine spike case.

REFERENCES

1. "Westinghouse Anticipated Transients Without Trip Anaylsis," WCAP-8330, August 1974.

2. International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, Volume 3 No. 1-4, 1979.

3. Not Used.

CPSES/FSAR

TABLE 15.1-3

(Sheet 1 of 3)

PARAMETERS FOR POSTULATED MAIN STEAM LINE BREAK ACCIDENT ANALYSIS

1. Data and assumptions used to estimate radioactive source from postulated accidents

a.	Power level (MWt)	3565 1.0
b.	Primary coolant activity concentration prior to accident	0.45 uCi/gm Dose Equivalent I-131
c.	Secondary side coolant activity concentration prior to accident	0.1 uCi/gm Dose Equivalent I-131
d.	Primary to Secondary leakage in affected steam generator (~~gpm~~) gpd	27.79 500
e.	Primary to Secondary leakage in unaffected steam generators (~~gpd~~) gpm	450 0.6528
f.	Offsite power lost	lost
g.	Failed fuel	none

2. Data and assumptions used to estimate activity released

a.	Iodine partition factor for initial and long term steam release from affected steam generator	1.0
b.	Iodine partition factor in unaffected steam generators prior to and during accident	0.01
c.	Initial steam release from affected steam generator (~~lb~~) lbm (0 to 15 min) 5	176,000 117,500

Note: The entire secondary side liquid inventory in the affected steam generator is assumed to be released.

CPSES/FSAR

TABLE 15.1-3

(Sheet 2)

PARAMETERS FOR POSTULATED MAIN STEAM LINE BREAK ACCIDENT ANALYSIS

lbm

9

d.	Long term steam release from affected steam generator ~~(lb)~~ ~~(15 min to 8 hr)~~ 5 minutes to 20 hours		0 (Units 1 and 2)
e.	steam release from three unaffected steam generators		
	~~(lb)~~ (0 to 2 hr)		438,000
	(2 to 8 hr)		~~873,000~~ 960,000
3.	Dispersion data		
	a.	EAB and LPZ distances	2080m and 4 miles
	b.	x/Q	@ EAB (0 – 2 hr) 1.6×10^{-4} sec/m^3 @ LPZ (0 – 8 hr) ~~2.3~~ 2.4 $\times 10^{-5}$ sec/m^3 @ LPZ (8-24 hr) 1.6×10^{-5} sec/m^3
4.	Dose data		
	a.	Method of dose calculations	See Appendix 15B
	b.	Dose conversion assumptions	See Appendix 15B
	c.	Primary and secondary side equilibrium activities	See Table 15.1-~~4A~~ 4
	d.	Iodine spike appearance rates	I-131: ~~103.2~~ 229 Ci/min I-132: ~~304.3~~ 730 Ci/min I-133: ~~185.0~~ 428 Ci/min I-134: ~~128.6~~ 318 Ci/min I-135: ~~140.7~~ 326 Ci/min
	e.	Pre-accident spike iodine concentrations	I-131: ~~46.71~~ 46.8 uCi/gm I-132: ~~46.71~~ 46.8 uCi/gm I-133: ~~70.20~~ 72.0 uCi/gm I-134: ~~9.53~~ 9.6 uCi/gm I-135: ~~38.37~~ 38.4 uCi/gm

TABLE 15.1-3

(Sheet 3)

PARAMETERS FOR POSTULATED MAIN STEAM LINE BREAK ACCIDENT ANALYSIS

f. Doses (rem)	thyroid	whole body (gamma dose)
@EAB (0-2 hr)		
concurrent iodine spike	~~26.9~~ 1.4	~~0.15~~ 0.004
preaccident iodine spike	~~44.0~~ 1.2	~~0.10~~ 0.002
@LPZ (0-30 days)		
concurrent iodine spike	~~21.5~~ 2.5	~~0.10~~ 0.004
preaccident iodine spike	~~23.0~~ 0.61	~~0.04~~ 0.001

CPSES/FSAR

TABLE 15.1-4

PRIMARY AND SECONDARY SIDE EQUILIBRIUM ACTIVITIES (ACCIDENT ANALYSIS)

Nuclide	Primary Coolant Concentration (uCi/gm)[a]	Secondary Liquid Concentration (uCi/gm)[b]	~~Secondary Steam Concentration (uCi/gm)~~
Kr-83m	~~1.10 x 10^{-1}~~ 5.06e-01	0.00 x 10^{-0}	~~3.69 x 10^{-6}~~
Kr-85m	~~4.80 x 10^{-1}~~ 2.20e+00	0.00 x 10^{-0}	~~1.60 x 10^{-5}~~
Kr-85	~~1.75 x 10^{+0}~~ 8.03e+00	0.00 x 10^{-0}	~~5.85 x 10^{-5}~~
Kr-87	~~3.12 x 10^{-1}~~ 1.43e+00	0.00 x 10^{-0}	~~1.04 x 10^{-5}~~
Kr-88	~~8.64 x 10^{-1}~~ 3.96e+00	0.00 x 10^{-0}	~~2.89 x 10^{-5}~~
~~Kr-89~~	~~2.64 x 10^{-3}~~	~~0.00 x 10^{-0}~~	~~8.82 x 10^{-7}~~
Xe-131m	~~5.28 x 10^{-1}~~ 2.42e+00	0.00 x 10^{-0}	~~1.76 x 10^{-6}~~
Xe-133m	~~4.08 x 10^{+0}~~ 1.87e+01	0.00 x 10^{-0}	~~1.36 x 10^{-4}~~
Xe-133	~~6.48 x 10^{+1}~~ 2.97e+02	0.00 x 10^{-0}	~~2.16 x 10^{-3}~~
Xe-135m	~~1.15 x 10^{-1}~~ 5.28e-01	0.00 x 10^{-0}	~~3.85 x 10^{-6}~~
Xe-135	~~1.73 x 10^{+0}~~ 7.92e+00	0.00 x 10^{-0}	~~5.77 x 10^{-5}~~
~~Xe-137~~	~~4.08 x 10^{-2}~~	~~0.00 x 10^{-0}~~	~~1.36 x 10^{-6}~~
Xe-138	~~1.54 x 10^{-1}~~ 7.04e-01	0.00 x 10^{-0}	~~5.13 x 10^{-6}~~
I-131	6.72 x 10^{-1} 7.8e-01	~~7.34 x 10^{-2}~~ 8.276e-02	~~7.34 x 10^{-4}~~
I-132	6.72 x 10^{-1} 7.8e-01	~~2.53 x 10^{-2}~~ 2.853e-02	~~2.53 x 10^{-4}~~
I-133	1.01 x 10^{+0} 1.2e+00	~~8.43 x 10^{-2}~~ 9.505e-02	~~8.43 x 10^{-4}~~
I-134	1.37 x 10^{-1} 1.6e-01	~~1.90 x 10^{-3}~~ 2.14e-03	~~1.90 x 10^{-5}~~
I-135	5.52 x 10^{-1} 6.4e-01	~~2.99 x 10^{-2}~~ 3.371e-02	~~2.99 x 10^{-4}~~

a. Based on a primary coolant specific activity of 1.0 uCi/gm Dose Equivalent I-131.

Noble gases are based on operation with 1% fuel defects.

~~Note: The primary coolant TS limit has been changed (reduced) to 0.45 uCi/gm Dose Equivalent I-131 (DEI-131). However, the applicable accident analyses have been demonstrated to be conservative using 1.0 uCi/gm DEI-131. The only exception is the Main Steam Line Break analysis that supports the implementation of the Steam Generator Alternate Repair Criteria (SG ARC). The SG ARC analysis assumes a DEI-131 value of 0.45 uCi/gm. This assumption maximizes the SG ARC benefit. Therefore, with the exception of the Main Steam Line Break analysis for the SG ARC, all reported doses assume a DEI-131 limit value of 1.0 uCi/gm.~~

b. Based on a secondary side specific activity of 0.10 uCi/gm Dose Equivalent I-131.

~~TABLE 15.1-4A~~

~~PRIMARY AND SECONDARY SIDE EQUILIBRIUM ACTIVITIES (STEAMLINE BREAK ANALYSIS)~~[c]

~~Nuclide~~	~~Primary Coolant Concentration (uCi/gm)~~[a]	~~Secondary Liquid Concentration (uCi/gm)~~[b]	~~Secondary Steam Concentration (uCi/gm)~~
~~Kr-83m~~	~~1.10×10^{-1}~~	~~$0.00 \times 10^{+0}$~~	~~$0.00 \times 10^{+0}$~~
~~Kr-85m~~	~~4.80×10^{-1}~~	~~$0.00 \times 10^{+0}$~~	~~$0.00 \times 10^{+0}$~~
~~Kr-85~~	~~$1.75 \times 10^{+0}$~~	~~$0.00 \times 10^{+0}$~~	~~$0.00 \times 10^{+0}$~~
~~Kr-87~~	~~3.12×10^{-1}~~	~~$0.00 \times 10^{+0}$~~	~~$0.00 \times 10^{+0}$~~
~~Kr-88~~	~~8.64×10^{-1}~~	~~$0.00 \times 10^{+0}$~~	~~$0.00 \times 10^{+0}$~~
~~Kr-89~~	~~2.64×10^{-2}~~	~~$0.00 \times 10^{+0}$~~	~~$0.00 \times 10^{+0}$~~
~~Xe-131m~~	~~5.28×10^{-1}~~	~~$0.00 \times 10^{+0}$~~	~~$0.00 \times 10^{+0}$~~
~~Xe-133m~~	~~$4.08 \times 10^{+0}$~~	~~$0.00 \times 10^{+0}$~~	~~$0.00 \times 10^{+0}$~~
~~Xe-133~~	~~$6.48 \times 10^{+1}$~~	~~$0.00 \times 10^{+0}$~~	~~$0.00 \times 10^{+0}$~~
~~Xe-135m~~	~~1.15×10^{-1}~~	~~$0.00 \times 10^{+0}$~~	~~$0.00 \times 10^{+0}$~~
~~Xe-135~~	~~$1.73 \times 10^{+0}$~~	~~$0.00 \times 10^{+0}$~~	~~$0.00 \times 10^{+0}$~~
~~Xe-137~~	~~4.08×10^{-2}~~	~~$0.00 \times 10^{+0}$~~	~~$0.00 \times 10^{+0}$~~
~~Xe-138~~	~~1.54×10^{-1}~~	~~$0.00 \times 10^{+0}$~~	~~$0.00 \times 10^{+0}$~~
~~I-131~~	~~3.505×10^{-1}~~	~~8.276×10^{-2}~~	~~8.276×10^{-4}~~
~~I-132~~	~~3.505×10^{-1}~~	~~2.853×10^{-2}~~	~~2.853×10^{-4}~~
~~I-133~~	~~5.27×10^{-1}~~	~~9.505×10^{-2}~~	~~9.505×10^{-4}~~
~~I-134~~	~~7.13×10^{-2}~~	~~2.14×10^{-3}~~	~~2.14×10^{-5}~~
~~I-135~~	~~2.88×10^{-1}~~	~~3.371×10^{-2}~~	~~3.371×10^{-4}~~

~~a. Based on a primary coolant specific activity of 0.45 uCi/gm Dose Equivalent I-131.~~

~~b. Based on a secondary side specific activity of 0.10 uCi/gm Dose Equivalent I-131.~~

~~c. Calculated using thyroid dose conversion factors from Reference 2.~~

Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (clad temperature). The Baker-Just parabolic rate equation shown below is used to define the rate of the zirconium-steam reaction.

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp(-45{,}500/1.986T)$$

where

w = amount reacted (mg/cm^2)

t = time (sec)

T = temperature (°F)

The heat of reaction is 1510 cal/gm.

The effect of the zirconium-steam reaction is included in the calculation of the "hot spot" clad temperature transient.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Evaluation of the Radiological Consequences

An evaluation of the number of rods in DNB was performed in order to assess the radiological consequences of this event. Analysis assumptions were made to maximize the heat flux and thus minimize the DNB, consistent with the current FSAR. The number of rods experiencing DNB is determined from a pin census showing the relative peaking factors of each rod.

INSERT LOCKED ROTOR

15.3.3.3 Conclusions

As noted in Appendices 4A and 4B for Units 1 and 2, respectively, the following conclusions are made:

1. Because the peak RCS pressure reached during any of the transients is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered.

2. Because the peak clad surface temperature calculated for the hot spot during the worst transient remains considerably less than 2700°F the core will remain in place and intact with no loss of core cooling capability.

shown in sheet 2 of Table 15.0-7

LOCKED ROTOR

The doses resulting from a locked rotor accident are based on a conservative fission product release to the reactor coolant of the gap activity from 10% of the fuel rods in the core. The method follows Regulatory Guide 1.195. Prior to the accident, it is assumed that the plant has been operating with simultaneous fuel defects and steam generator tube leakage for a sufficient amount of time to establish equilibrium levels of activity in the primary and secondary coolants equal to the Table 15.1-4 values.

Following the postulated locked rotor accident, the activity released from the pellet-gap due to the fuel failure is assumed to be instantaneously released to the primary coolant and is assumed to be immediately available for release from the RCS.

The following assumptions are used in the analysis:

1. The initial Reactor Coolant System (RCS) activity is 1.0 μCi/gm DE I-131.
2. The initial RCS noble gas activity is based on 1% fuel defects.
3. The initial secondary activity is 0.1 μCi/gm DE I-131.
4. A radial peaking factor (FΔh) of 1.65 is applied in the calculation of gap activities released (per Regulatory Guide 1.195).
5. 10% of rods are assumed to undergo clad damage sufficient to release all of their gap activity.
6. The primary to secondary leak rate is 1.0 gpm plant total.

A summary of the parameters used in the analysis is given in Table 15.3-1.

Based on the foregoing model, the thyroid and whole body doses are conservatively calculated to be 1.5 rem and 0.12 rem, respectively, for the exclusion area boundary (EAB), and 2.5 rem and 0.037 rem respectively, for the low population zone (LPZ). These doses are well below the values of 30 rem thyroid and 2.5 rem whole body set forth in Regulatory Guide 1.195.

~~3. The number rods experiencing DNB remains less than the number of rods assumed to experience DNB for the Rod Ejection Event (see section 15.4.8), therefore, the conclusions of the radiological analysis for that event (i.e., the radiological doses are below the dose valves set forth in 10CFR100) apply to this event.~~

15.3.4 REACTOR COOLANT PUMP SHAFT BREAK

15.3.4.1 Identification of Causes and Accident Description

The accident is postulated as an instantaneous failure of a reactor coolant pump shaft, such as discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the reactor coolant pump rotor seizure event. With a failed shaft, the impeller could conceivably be free to spin in the reverse direction instead of being fixed in position. The effect of such reverse spinning is a slight decrease in the end point (steady state) core flow.

The analysis presented in Section 15.3.3 represents the limiting condition, assuming a locked rotor for forward flow but a free-spinning shaft for reverse flow in the affected loop.

15.3.4.2 Conclusions

The conclusions of Section 15.3.3.3 apply for the Reactor Coolant Pump Shaft Break event.

REFERENCES

1. Not Used.

2. Baldwin, M. S., Merrian, M. M., Schenkel, H. S. and Van De Walle, D. J., "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," WCAP-8424, Revision 1, June 1975.

Insert Table 15.3-1 after this page

TABLE 15.3-1

PARAMETERS FOR POSTULATED LOCKED ROTOR ACCIDENT ANALYSIS

1.	Data and assumptions used to estimate radioactive source from postulated accidents		
	a. Power level		3565
	b. Primary coolant activity concentration prior to accident		1.0 μCi/gm Dose Equivalent I-131
	c. Secondary side coolant activity concentration prior to accident		0.1 μCi/gm Dose Equivalent I-131
	d. Primary to secondary leakage		1 gpm plant total
	e. Offsite power		lost
	f. Failed fuel		10%
	g. Radial Peaking Factor		1.65
2.	Data assumptions used to estimate activity released		
	a. Iodine partition factor for steam release from secondary side		0.01
	b. Steam release from secondary side		450,000 (0 – 2 hours) 1,002,000 (2 – 8 hours)
3.	Dispersion data		
	a. EAB and LPZ distances		2080 m and 4 miles
	b X/Q		@ EAB (0 -2 hours) 1.6×10^{-4} sec/m^3 @ LPZ) (0 – 8 hours) 2.4×10^{-5} sec/m^3
4.	Dose Data		
	a. Method of dose calculations		See Appendix 15B
	b. Dose conversion assumptions		See Appendix 15B
	c. Doses	thyroid	whole body (gamma dose)
	@EAB (0 -2 hours)	1.5 rem	0.12 rem
	@LPZ (0 – 8 hours)	2.5 rem	0.037 rem

15.4.8.3 Environmental Consequences

The ejection of an RCCA constitutes a break in the RCS, located in the reactor pressure vessel head. The effects and consequences of loss of coolant accidents are discussed in Section 15.6.5. Following the RCCA ejection, the operator would follow the same emergency instructions as for any other loss of coolant accident to recover from the event.

Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than 10 percent of the rods entered DNB.

Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at beginning-of-life, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits [2]. Since the severity of the present analysis does not exceed the "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since the core is under-moderated, and bowing will tend to increase the under-moderation at the hot spot. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

shown in sheet 2 of Table 15.0-7

Radiological Consequences

The doses resulting from a rod ejection accident are based on a conservative fission product release to the reactor coolant of the gap activity from 10 percent of the fuel rods in the core plus the core activity from the assumed 0.25 percent core melt. The method of analysis complies with the requirements of $\boxed{\text{Position 3}}$ ~~Appendix B~~ of Regulatory Guide $\boxed{1.195.}$ ~~1.77 except as noted in Appendix 1A(B).~~

Following a postulated rod ejection accident, two activity release paths contribute to the total radiological consequences of the accident. The first release path is via Containment leakage resulting from release of activity from the primary coolant to the Containment. The second path is the contribution of contaminated steam in the secondary system dumped through the relief valves, since offsite power is assumed to be lost.

Prior to the accident, it is assumed that the plant has been operating with simultaneous fuel defects and steam generator tube leakage for a sufficient period of time to establish equilibrium levels of activity in the primary and secondary coolants equal to the Table 15.1-4 ~~limits.~~ $\boxed{\text{values.}}$

Following a postulated rod ejection accident, the activity released from the pellet-clad gap due to failure of a portion of the fuel rods and the melted fuel is assumed to be instantaneously released to the primary coolant. The activity released to the primary coolant is assumed to be uniformly mixed throughout the coolant instantaneously. Thus, the total activity of the primary coolant is assumed to be immediately available for release from the RCS. Of the activity released to the Containment from the coolant through the rupture in the reactor vessel head, 100 percent is assumed to be mixed instantaneously throughout the Containment. Fifty (50) percent of the iodine activity released from the melted fuel is assumed to immediately plate out on Containment surfaces. The remaining activity is available for leakage from the Containment at the design leak rate of 0.10 percent of Containment volume per day for the first 24 hours, and at a rate of 0.05 percent of Containment volume per day for the duration of the accident. The only removal processes considered in the Containment for the activity remaining after the above plate out are radioactive decay and leakage from the Containment.

The model for the activity available for release to the atmosphere from the relief valves assumes that the release consists of the activity in the secondary coolant prior to the accident plus that fraction of the activity leaking from the primary coolant through the steam generator tubes following the accident. The leakage of primary coolant to the secondary side of the steam generator is assumed to continue at its initial rate, which is assumed to be the same rate as the leakage prior to the accident, until the pressures in the primary system and the secondary system are equalized. No mass transfer from the primary system to the secondary system through steam generator tube leakage is assumed thereafter. With the assumption of coincident loss of offsite power, activity is assumed to be released to the atmosphere from a steam dump through the relief valves ~~for 400 seconds.~~ $\boxed{\text{8 hours.}}$

A summary of parameters used in the analysis is given in Table 15.4-4.

Fuel melting, limited to less than the innermost 10 percent of the fuel pellet at the hotspot, is included in the design criteria to ensure that fuel dispersion into the coolant does not occur [1]. Even though centerline melting in a small fraction of the core is not expected, a conservative upper limit of fission product release from the core as a result of a rod ejection accident can be estimated.

The Regulatory Guide ~~1.77~~ 1.195 limit of fission product release from the core for this very conservative case is determined using the following assumptions:

1. 100 percent of the noble gases and iodines in the clad gaps of those fuel rods experiencing clad damage (assumed to be 10 percent of the rods in the core) are released to the reactor coolant. The affected fuel rods are assumed to be operating at 1.65 times the core average and thus have 1.65 times the average fisson product inventory. The gap activities are presented in Table 15.6-8.

2. 25 percent of the iodines and 100 percent of the noble gases in the fuel that melts are assumed to be released to the reactor coolant and available for release via the containment building. 50 percent of the iodines and 100 percent of the noble gases in the fuel that melts are assumed to be available for release from the secondary systems via primary-to-secondary leakage. The core activities are presented in Sheet 2 of Table 15.0-7.

3. The fraction of fuel melting is conservatively assumed to be 0.25 percent of the core, as determined by the following method [1]:

   a. A conservative upper limit of 50 percent of the rods experiencing clad damage also may experience centerline melting (a total of 5 percent of the rods in the core).

   b. Of rods experiencing centerline melting, only a conservative maximum of the innermost 10 percent of the volume actually melts (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 experiences melting due to the power distribution (0.5 of the 0.5 percent of the core = 0.25 percent of the core).

4. Prior to the accident, the plant is assumed to be operating at full power with coincident fuel defects and steam generator tube leakage. The steam generator tube leak rate is assumed to be 1.0 gpm. The initial primary coolant and secondary liquid ~~and steam~~ activity concentrations are given in Table 15.1-4.

5. Instantaneous mixing in the Containment of all activity released from the coolant is assumed.

6. Fifty (50) percent of the iodine activity released from the melted fuel to the Containment atmosphere immediately plates out on Containment surfaces.

7. Primary and secondary system pressures are equalized after ~~1100 seconds;~~ 8 hours, thus terminating primary to secondary leakage in the steam generators.

8. For the case of loss of offsite power, ~~73,200~~ 1,452,000 lb of steam are discharged from the secondary system through the relief valves, during the ~~first 400 seconds~~ following the accident. Steam dump is terminated after ~~400 seconds.~~ 8 hours

9. All releases to the atmosphere are assumed to be at ground level.

10. Other assumptions are detailed in Table 15.4-4.

Based on the foregoing model, the thyroid and whole body doses from the ~~primary side~~ containment leakage for this accident are conservatively calculated to be ~~12.4 rem and 4.2 x 10^{-2} rem~~ 14.0 rem and 0.059 rem, respectively, for the exclusion area boundary (EAB), and ~~8.49 rem and 1.55 x 10^{-2} rem~~ 20.0 rem and 0.031 rem respectively, for the low population zone (LPZ). The thyroid and whole body doses from the secondary side releases for this accident are conservatively calculated to be ~~3.1 x 10^{-2} rem and~~ 2.3 rem and ~~2.9 x 10^{-2}~~ 0.29 rem, respectively, for the EAB, and ~~4.4 x 10^{-3} rem and 4.19 x 10^{-3} rem,~~ 3.8 rem and 0.091rem, respectively, for the LPZ. The doses at these distances are below the dose values of 25 rem whole body and 300 rem thyroid set forth in 10CFR100.

15.4.8.4 Conclusions

As noted in Appendices 4A and 4B for Units 1 and 2, respectively, even on a pessimistic basis, the analyses indicate that the described fuel and clad limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the RCS. The analyses have demonstrated that upper limit in fission product release as a result of a number of fuel rods entering DNB amounts to 10 percent.

REFERENCES

1. "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.

2. Risher, D. H., Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1-A, January 1975.

3. Bishop, A. A., Sandburg, R. O. and Tong, L. S., "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," ASME 65-HT-31, August 1965.

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TABLE 15.4-4

(Sheet 1 of 4)

PARAMETERS FOR POSTULATED ROD EJECTION ACCIDENT ANALYSIS

1.	Data and assumptions used to estimate radioactive source from postulated accidents		
	a.	Power level (MWt)	3565
	b.	Primary Side equilibrium activity concentration	1.0 uCi/gm Dose Equivalent I-131
	c.	Secondary Side equilibrium activity concentration	0.1 uCi/gm Dose Equivalent I-131
	d.	Steam generator tube leak rate prior to and during steam dump (gpm)	1
	e.	Failed fuel	10 percent of fuel rods in core
	f.	Radial peaking factor	1.65
	g. ~~f.~~	Activity released to reactor coolant from failed fuel and available for release (i.e., gap fraction)	
		Noble gases	10 percent ~~of gap inventory~~ of core inventory
		Iodines	10 percent ~~of gap inventory~~ of core inventory
	h. ~~g.~~	Melted fuel	0.25 percent of core
	i. ~~h.~~	Activity released to reactor coolant from melted fuel	
		Noble gases inventory	0.25 percent of core
		Iodines	0.0625 percent of core inventory (available for release via the containment building

~~* Note: The primary coolant TS limit has been changed (reduced) to 0.45 uCi/gm Dose Equivalent I-131). However, the applicable accident analyses have been demonstrated to be conservative using 1.0 uCi/gm DEI-131. The only exception is the Main Steam Line Break analysis that supports the implementation of the Steam Generator Alternate Repair Criteria (SG ARC). The SG ARC analysis assumes a DEI-131 value of 0.45 uCi/gm. This assumption maximizes the SG ARC benefit. Therefore, with the exception of the Main Steam Line Break analysis for the SG ARC, all reported doses assume a DEI-131 limit value of 1.0 uCi/gm.~~

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TABLE 15.4-4

(Sheet 2)

PARAMETERS FOR POSTULATED ROD EJECTION ACCIDENT ANALYSIS

		0.125 percent of core inventory available for release from the secondary system primary to-seocndary leakage)
j ~~i.~~	Iodine fractions (organic, elemental, and particulate)	Regulatory Guide ~~1.4~~ 1.195
2.	Data and assumptions used to estimate activity released	
a.	Plate out of iodine activity	50 percent released to Containment (applied only to fuel melt releases)
b.	Containment leak rate	0.1 percent of containment volume per day (~~∅~~ 0≤t≤24 hr) 0.05 percent of containment volume per day (24 < t < 720 hr)
c.	Iodine partition factor in steam generators prior to and during accident	0.01
d.	Offsite power	lost
e.	Steam dump from relief valves (lb)	~~73,200~~ 450,000 (0 - 2 Hours) 1,002,000 (2 - 8 Hours)
f.	Duration of dump from relief valves ~~(sec)~~ hour	~~400~~ 8
g.	Time between accident and equalization of primary and secondary system pressures ~~(sec)~~ hour	~~1100~~ 8
h.	Steam dump to condenser (lb)	0.0
3.	Dispersion data	
a.	EAB and LPZ distances	2080m and 4 miles

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TABLE 15.4-4

(Sheet 3)

PARAMETERS FOR POSTULATED ROD EJECTION ACCIDENT ANALYSIS

b.	X/Q	@EAB	1.6×10^{-4} sec/m^3 (0-2 hr)
		@LPZ	2.4 ~~2.3~~ $\times 10^{-5}$ sec/m^3 (0-8 hr)
		@LPZ	2.2×10^{-6} sec/m^3 (8-24 hr)
		@LPZ	1.1×10^{-6} sec/m^3 (1-4 days)
		@LPZ	6.6×10^{-7} sec/m^3 (4-30 days)

4. Dose data

a.	Method of dose calculation	See Appendix 15B
b.	Dose conversion assumptions	See Appendix 15B
c.	Doses	@EAB (0-2 hr)
		thyroid dose
	Containment leakage	= 12.4 Rem
	Primary to secondary leakage	= 3.1×10^{-2} rem
	Total thyroid dose	= 12.4 rem
		14.0 rem 2.3 rem
		Whole body dose (gamma dose)
		0.059 0.29
	Containment leakage	= 4.2×10^{-2} rem
	Primary to secondary leakage	= 2.9×10^{-2} rem
	Total whole body dose	= 7.1×10^{-2} rem
		8 hours
		@LPZ (0-30 days)

CPSES/FSAR

TABLE 15.4-4

(Sheet 4)

PARAMETERS FOR POSTULATED ROD EJECTION ACCIDENT ANALYSIS

	thyroid dose
Containment leakage	$\boxed{20.0}$ = ~~8.49~~ rem
Primary to secondary leakage	$\boxed{3.8}$ = ~~4.40 x 10^-3^~~ rem
~~Total thyroid dose~~	~~= 8.49 rem~~

	Whole body dose (gamma dose)
Containment leakage	$\boxed{0.031}$ = ~~1.55 x 10^-2^~~ rem
Primary to secondary leakage	$\boxed{0.091}$ = ~~4.19 x 10^-3^~~ rem
~~Total whole body dose~~	~~= 2.0 x 10^-2^ rem~~

15.6.1.3 Conclusions

As noted in Appendices 4A and 4B for Units 1 and 2, respectively, the results of the analysis show that the pressurizer low pressure and the Overtemperature N-16. Reactor Protection System signals provide adequate protection against the RCS depressurization event.

15.6.2 BREAK IN INSTRUMENT LINE OR OTHER LINES FROM REACTOR COOLANT PRESSURE BOUNDARY THAT PENETRATE CONTAINMENT

There are no instrument lines connected to the RCS that penetrate the Containment. However, the 3 inch CVCS letdown line and grab sample lines from the hot legs of reactor coolant loops 1 and 4, and from the steam and liquid space of the pressurizer, do penetrate the Containment. The grab sample lines are provided with normally closed isolation valves on both sides of the Containment wall and are designed in accordance with the requirements of General Design Criterion 55.

The most severe pipe rupture with regard to radioactivity release during normal plant operation occurs in the Chemical and Volume Control System (CVCS). This Condition III event would be a complete severance of the 3 inch letdown line just outside containment, between the outboard letdown isolation valve and letdown heat exchanger (see Figure 9.3-10), at rated power condition. The occurrence of a complete severance of the letdown line would result in a loss-of-reactor coolant at the rate of about 190 gpm, which is within the makeup capacity of any two of the three charging pumps (see Table 9.3-7, sheet 1). Two pressure switches located in the CVCS letdown line provide the means for detecting a rupture of the line. A low pressure signal, initiated by the rupture, will activate an alarm on the main control board, alerting the operator of the rupture, causing him to manually isolate the system. No credit is taken for Engineered Safety Features System actuation. Frequent operation of the automatic Reactor Makeup System will also provide the operator some indication of the loss-of-reactor coolant. No transient analysis is performed for this accident.

It is conservatively assumed that

The time required for the operator to identify the accident and initiate the closure of the letdown isolation valve is expected to be within 30 minutes after accident initiation including 10 seconds for the letdown isolation valve closure time. Reactor coolant is assumed to be released until the isolation valve is fully closed. ~~Calculations indicate that~~ 20.1 percent of the leaking coolant flashes to steam. All of the iodine in this steam is assumed to become airborne and is available for release to the atmosphere. All noble gases contained in the leaking primary coolant are available for release to the atmosphere. A ground level release was postulated with an atmospheric dilution factor of 1.6×10^{-4} sec/m^3 at the minimum EAB distance. The equilibrium concentration of radioactive nuclides in the reactor coolant is given in Table 15.1-4. Effects of a concurrent iodine spike are included in the analysis.

Based on the foregoing model, 0.039 the doses at the EAB are conservatively calculated to be 5.7 ~~3.8~~ rem to the thyroid and ~~2.15 x 10^-2^~~ rem to the whole body. The LPZ doses are 0.0058 conservatively calculated to be ~~0.55~~ 0.85 rem to the thyroid and ~~3.1 x 10^-3^~~ rem to the whole body. As expected, the radiological consequences resulting from the failure of the 3 inch CVCS letdown line do not exceed a small fraction of the dose values set forth in 10CFR100.

15.6.3 STEAM GENERATOR TUBE FAILURE

15.6.3.1 Identification of Causes and Accident Description

The accident examined is the complete severance of a single steam generator tube. This event is considered an ANS Condition IV event, a limiting fault (see Section 15.0.1) and is analyzed to demonstrate that the resulting radiological doses are within the guideline values of 10CFR100. Timely operator response is required to terminate the primary-to-secondary break flow and to ensure that the ruptured steam generator does not fill with water and flood the main steamlines. This criterion is important because the main steamlines and relief valves are not designed for liquid flow.

The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the RCS. In the event of a coincident loss of offsite power, or failure of the condenser dump system, discharge of activity to the atmosphere takes place via the steam generator safety and/or power operated relief valves.

In view of the fact that the steam generator tube material is Inconel-600 and is a highly ductile material, it is considered that the assumption of a complete severance of a tube is somewhat conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the Steam and Power Conversion System is subject to continual surveillance and an accumulation of minor leaks which exceeds the limits established in the Technical Specifications is not permitted during unit operation.

The reactor operators are expected to determine that a steam generator tube rupture has occurred, and to identify and isolate the ruptured steam generator on a restricted time scale in order to minimize the contamination of the secondary system and to ensure the termination of the radioactive release to the atmosphere from the ruptured steam generator. In the following discussions, the steam generator with the ruptured U-tube is referred to as the ruptured steam generator. The operator is then expected to carry out the appropriate recovery procedures on a restricted time scale in order to terminate the primary-to-secondary break flow before the water level in the ruptured steam generator rises into the main steam system piping. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily.

Assuming normal operation of the various plant control systems, the following sequence of events is initiated by a tube rupture:

The time dependent mass releases used to assess the radiological consequences of the postulated steam generator tube rupture are calculated from the RETRAN02 thermal-hydraulic analysis described above. Time-dependent values of the leakage rate into the ruptured steam generator and the flashing fraction were also used to assess the radiological consequences for the 0-2 hour time period following the event. Following the closure of the atmospheric relief valve block valve, the additional radiological dose is due to the leakage from the primary system into the intact steam generators and the initial concentration of radioactivity contained in the intact steam generators.

Two separate iodine spikes are considered:

Case I — A reactor transient has occurred prior to the tube rupture and raised the primary coolant iodine concentration to 60 uCi/gm Dose Equivalent Iodine-131 (DEQ I-131). The resulting preaccident isotopic iodine concentrations are shown in Table 15.6-3.

Case II — The reactor trip or primary system depressurization associated with the postulated accident creates an iodine spike in the primary system. The spike is assumed to increase the iodine appearance rate (inleakage from the defective fuel rods to the primary coolant) to ~~500~~ 335 times the equilibrium appearance rate. The concurrent iodine spike appearance rates are presented in Table 15.6-4.

The assumptions below are used to determine the initial primary and secondary activities and to calculate the activity released and the offsite doses for the postulated steam generator tube rupture accident.

1. The initial primary coolant iodine activity (i.e., prior to any iodine spike considerations) is assumed to be at 1.0 uCi/gm DEQ I-131.

~~NOTE: The primary coolant TS limit has been changed (reduced) to 0.45 uCi/gm Dose Equivalent I-131 (DEI-131). However, the applicable accident analyses have been demonstrated to be conservative using 1.0 uCi/gm DEI-131. The only exception is the Main Steam Line Break analysis that supports the implementation of the Steam Generator Alternate Repair Criteria (SG ARC). The SG ARC analysis assumes a DEI-131 value of 0.45 uCi/gm. This assumption maximizes the SG ARC benefit. Therefore, with the exception of the Main Steam Line Break analysis for the SG ARC, all reported doses assume a DEI-131 limit value of 1.0 uCi/gm.~~

2. The primary coolant activity has been leaking into the secondary side at one gpm for a period of time long enough to establish equilibrium activity concentrations is the steam generators.

3. All noble gas activity transported from the primary system to the secondary system and all noble gas and iodine activity initially in the steam region of the steam generators is immediately released to the environment. The initial iodine activity in the water region of the ruptured steam generator increases over time due to the unflashed portion of the leakage.

4. Due to the pressure differential between the primary and secondary sides, a fraction of the primary coolant that ~~leaked~~ is discharged to the defective steam generator flashes to steam. This flashed fraction does not mix with the steam generator water and, therefore, is not subjected to any iodine removal process in the steam generator. However, the flashed fraction experiences iodine removal in the condenser when that path is available.

5. Radioactive decay of parent iodines to noble gas products is not considered during the iodine spiking processes ~~and~~ or as unflashed iodine accumulates in the steam generators. ~~The radioactive decay of the parent iodines is conservatively assumed to not decrease the activity of the parent iodines.~~

6. A ground level release is assumed. No credit is taken for radioactive decay or cloud depletion due to ground deposition during plume transport.

7. Worst case, five percentile atmospheric dispersion factors are assumed.

8. A breathing rate of ~~3.47E-04~~ 3.5E-04 m3/sec is assumed.

9. Conservative iodine partition factors of 0.01 and 0.15 are used in the steam generator and condenser, respectively, to account for iodine removal effects within those components.

10. The dose contribution from the intact generator is based upon the effects of iodine spiking and a 1 gpm primary-to-secondary leak rate lasting for 8 hours. The leakage is assumed to ~~totally flash directly to the atmosphere.~~ mix with the secondary side water.

11. Other assumptions are detailed in Table 15.6-2.

Based on the foregoing model, the thyroid and whole body doses at the EAB are conservatively calculated to be ~~46.2~~ 22.0 rem and ~~0.12~~ 0.080 rem, respectively, for the preaccident iodine spike case and ~~17.5~~ 8.8 rem and ~~0.11~~ 0.077 rem, respectively, for the concurrent iodine spike case. The thyroid and whole body doses at the LPZ are conservatively calculated to be ~~7.8 rem and 1.99 x 10^{-2}~~ 3.4 rem and 0.013 rem, respectively, for the preaccident iodine spike case and ~~6.73~~ 1.4 rem and ~~3.53 x 10^{-2}~~ 0.012 rem, respectively, for the concurrent iodine spike case. As expected, the doses are well below the values of 300 rem to the thyroid and 25 rem to the whole body set forth in 10CFR100.

15.6.4 SPECTRUM OF BWR STEAM SYSTEM PIPING FAILURES OUTSIDE OF CONTAINMENT

This section is not applicable to the CPSES.

15.6.5 LOSS OF COOLANT ACCIDENTS RESULTING FROM A SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

15.6.5.1 Identification of Causes and Frequency Classification

A LOCA is the result of a pipe rupture of the RCS pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 square feet (ft^2) This event is considered an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis (see Section 15.0.1).

Large Break LOCA Evaluation Model

For the Unit 1 and Unit 2 large break LOCA analyses, the limiting PCT is less than the 10CFR50.46 Acceptance Criteria limit of 2200EF. The maximum local and total core metal-water reaction is well below the embrittlement limit of 17% and 1% respectively as required by 10CFR50.46. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

The containment pressure and temperature response to postulated LOCAs are presented in Section 6.2.1.5.

Small Break LOCA Evaluation Model

Based on the results of the LOCA sensitivity studies [14] the limiting small break was found to be less than a 10 inch diameter rupture of the RCS cold leg. For Units 1 and 2, a small break spectrum analysis showed that the limiting small break was 3 inch. The worst single failure assumed is one safety injection train.

Based on the results reported in Appendices 4A and 4B, the calculated PCT resulting from a small break LOCA is less than the limiting large break PCT and remains well below the Acceptance Criteria limits of 10CFR50.46.

15.6.5.4 Environmental Consequences

To demonstrate in a conservative manner that the operation of a nuclear power station does not present any undue radiological hazard to the general public, a hypothetical accident involving a gross release of fission products is evaluated. No mechanism for such a release has been postulated because it would require a number of simultaneous failures to occur in the engineered safety features. The core fission product inventory is assumed to be released into the containment as described in ~~TID-14844 [21].~~ Regulatory Guide 1.195 [34]. Numerical values for the total core fission product inventory of the isotopes considered in calculating the radiation doses are listed in Table 15.6-8.

The radiological evaluation of this accident is divided into two parts: internal (thyroid) dose from inhalation of iodines in the leakage plume, and external (whole body) exposure as a result of immersion in the leakage plume.

The radiological consequences due to the release of core fission products during a postulated loss-of-coolant accident are evaluated in the following sections:

1. Radiological consequences of containment leakage

   The integrated thyroid doses and the integrated whole body doses are calculated using methods and assumptions in conformance with Regulatory Guide ~~1.4.~~ 1.195. The assumptions used in the analysis are listed below.

a. Twenty-five percent of the equilibrium radioactive iodine inventory developed from maximum full-power operation of the core is immediately available for leakage from the reactor containment. Of this 25 percent, 91 percent is in the form of elemental iodine, 5 percent is in the form of particulate iodine, and 4 percent is in the form of organic iodides.

b. All (i.e., 100 percent) of the equilibrium radioactive noble gas inventory developed from maximum full-power operation of the core is immediately available for leakage from the reactor containment.

c. The effects of radiological decay during holdup in the containment are taken into account.

d. The containment volume is divided into separate regions by concrete floors at different elevations (see Section 6.5.2). A radial gap between the concrete floors and the inner wall of the Containment Building permits a limited amount of convective mixing between these regions. The region not covered by containment spray is treated as a separate unsprayed volume which is assumed to mix with the volumes in the sprayed areas at a mixing rate of two turnovers per hour.

The Containment Spray System is actuated by a high containment pressure signal. For a discussion of the sequence of events of spray system operation, see Section 6.5.2. A sodium hydroxide spray is used to reduce the amount of fission product iodine available for release during the LOCA. The containment spray solution is assumed to interact with the elemental iodine and particulate iodine. The mathematical ~~model~~ models which calculate ~~calculates~~ the iodine spray removal ~~coefficient is~~ coefficients are presented in Section 6.5.2. For each region the calculated elemental iodine removal coefficients are above 10 hr^{-1}. The removal coefficient for elemental iodine used in the offsite dose calculation is limited to a ~~maximum value~~ of 10 hr^{-1} [22]. The calculated ~~A conservative~~ value of 11.4 ~~1.07 hr^{-1}~~ is used for the particulate iodine removal coefficient, ~~although higher removal coefficients have been calculated~~ until a DF of 50 is reached at which time the particular removal coefficient is reduced by a factor of 10. The elemental iodine removal effectiveness may be expected to diminish after the concentration in the containment atmosphere has been reduced by several orders of magnitude. The elemental iodine removal effectiveness of the spray system is conservatively assumed to cease after a decontamination of 100 in the containment atmosphere has been achieved.

e. The iodine and noble gases available for release to the environment are assumed to leak from the Containment at a maximum leak rate of 0.10 percent of the containment volume per day for the first 24 hours, and at 0.05 percent of the containment volume per day for the duration of the accident.

f. The duration of the accident is considered to be 30 days.

g. A ground-level release is assumed. Atmospheric dilution factors are discussed in Section 2.3 [23] and listed in Table 15.6-9.

h. No credit is taken for depletion of fission products in the plume due to ground deposition or radioactive decay in transit.

i. For the first 8 hours after the accident, the breathing rate of persons offsite is assumed to be ~~3.47~~ 3.5 x 10-4 cubic meters per second (m3/sec). Eight to 24 hours following the accident, the breathing rate is assumed to be ~~1.75~~ 1.8 x 10-4 m3/sec. From 24 hours through 30 days after the accident, the rate is assumed to be ~~2.32~~ 2.3 x 10-4 m3/sec.

j. The mathematical model and dose conversion factors presented in Appendix 15B are used for evaluating the radiological consequences of the LOCA

k. Other assumptions are detailed in Table 15.6-9.

Using the assumptions presented above and the mathematical models presented in Appendix 15B, the doses at the EAB were conservatively calculated to be ~~72.3~~ 49.2 rem to the thyroid and ~~0.91~~ 0.743 rem to the whole body; the doses at the LPZ were conservatively calculated to be ~~22.2~~ 19.9 rem to the thyroid and ~~0.29~~ 0.258 rem to the whole body.

2. Radiological consequences of engineered safety features equipment leakage outside containment.

Following a postulated LOCA, a potential source of fission product release is the leakage of water from engineered safety features (ESF) equipment located outside the containment. Such leakage could occur during the recirculation phase through components such as pump flanges, valves, and heat exchangers. The fission products could then be released from the water into the atmosphere resulting in offsite radiological consequences that contribute to the total dose from the LOCA.

An analysis of the offsite effects attributable to ESF equipment leakage is performed based on the following conservative assumptions:

a. 50 percent of the halogens originally present in the core are intimately mixed with the coolant water and are assumed to be available for release through ESF equipment outside containment (see Table 15.6-10).

~~b. All of the noble gases produced from the decay of halogens which remain in the leakage water are released to rooms housing the leaking components.~~

b. ~~c.~~ The leakage from all ESF components is conservatively assumed to start 10 minutes after the LOCA and continue for the duration of the accident at a rate of 2 gallons per minute.

c. ~~d.~~ An iodine partition factor of 0.1 is assumed. This factor is taken as the fraction of iodine in the leakage that becomes airborne.

d. ~~e.~~ Gaseous radioactivity released to rooms housing the leaking components is considered to be immediately swept away by the ventilation system and released to the atmosphere. (See Section 9.4.5 and Figures 1.2-16, 1.2-35 and 9.4-9).

e. ~~f.~~ An iodine adsorber efficiency of 95 percent is applied since the ventilation exhaust passes through HEPA filters and iodine adsorbers prior to release to the atmosphere. The iodine adsorbers are designed to the requirements of NRC Regulatory Guide 1.52 (See Appendix 1A(B)) as discussed in Section 9.4.3.

f. ~~g.~~ No credit is taken for an elevated release; all meteorological parameters are considered to be identical to those previously defined in this section.

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Based upon the foregoing model, the thyroid and whole body dose contributions due to ESF equipment leakage are conservatively calculated to be ~~16.6~~ 12.8 rem and ~~0.46~~ 0.0426 rem, respectively, for the EAB. The LPZ doses are conservatively calculated to be ~~14.1~~ 26.3 rem to the thyroid and ~~0.817~~ 0.0283 rem to the whole body.

4. ~~3.~~ Total dose due to a postulated LOCA

The total dose attributed to a postulated LOCA is the combined doses due to containment leakage ~~and~~ ESF equipment leakage, and releases through the containment pressure relief line. The combined EAB doses are ~~88.9~~ 63.0 rem to the thyroid and ~~1.4~~ 0.79 rem to the whole body. The combined LPZ doses are ~~36.3~~ 47.0 rem to the thyroid and ~~1.1~~ 0.29 rem to the whole body. As expected, the doses are below the values set forth in 10CFR100.

The dose to personnel engaging in mineral extraction operations within the exclusion area, in the event of a postulated LOCA, would be less than the dose values of 300 rem to the thyroid and 25 rem to the whole body set forth in 10CFR100.

5. ~~4.~~ Dose to the control room occupants

In the event of a Design Basis Accident (DBA), the safety injection actuation signal or a high radiation signal from the control room air intake monitors will initiate emergency recirculation and pressurization of the Control Room air conditioning system. Later, the emergency ventilation air makeup system can be brought into operation as described in Section 9.4.1.

The control room doses were analyzed for various design basis accidents. It was determined that the LOCA doses represent the limiting case. Therefore the methodology and the doses calculated for the LOCA are reported here.

The following assumptions are applied in the calculations of the dose to the control room occupants following the LOCA:

a. The basic assumptions presented in Items 1 and 2, above, are applied, except a constant breathing rate of ~~3.47~~ $\boxed{3.5}$ x 10^{-4} m^3/sec is assumed throughout the accident.

b. The control room pressurization (air intake) and recirculation iodine adsorbers are assigned a 99 percent decontamination efficiency for both elemental and organic iodines in accordance with Table 2 of Regulatory Guide 1.52 (See Appendix 1A(B)). The pressurization adsorbers are arranged in series with the recirculation adsorbers (see Figure 9.4-1) during the emergency pressurization mode, thus providing an equivalent decontaminating efficiency in excess of 99 percent for both elemental and organic iodines from the pressurization makeup air.

c. The control room air-conditioning system runs either in the emergency recirculation mode or the emergency ventilation mode during a LOCA.

d. During the emergency recirculation mode of operation, a constant air intake flow rate of 800 ft^3/min is assumed. This makes up for losses caused by leaks and maintains the control room atmosphere at a positive pressure of 0.125 inch water gauge relative to adjacent areas.

   Since both recirculation trains are actuated by the safety injection signal, the outside air intake flow rate during dual train operation is conservatively estimated to be 1600 cfm. If both trains are assumed to operate for one hour, the calculated thyroid dose would decrease, due to the additional iodine filtration. The calculated whole body gamma and beta skin doses would increase slightly due to the additional intake of outside air. In both cases, the calculated doses remain below the limits specified in 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 19. The reported control room doses are the higher values of the two cases.

e. During the emergency ventilation mode of operation, 3800 ft3/min of outdoor air is used to introduce fresh air into the control room.

f. The emergency ventilation mode of operation is under administrative control so that the dose to the control room occupants is minimized, and the need for air change is satisfied.

   The operating mode sequence used in this analysis is as follows:

A = projected area of Containment Building (3265 m^2) for the containment leakage source and 2088 m^2 for the ESF leakage source

Table 15.6-12 summarizes the X/Q values calculated utilizing this expression.

i. The CPSES CREFS design is zone isolation, with filtered recirculation air, and with a positive pressure. This design maximizes the iodine protection factors and minimizes the dose from iodine. The total unfiltered infiltration rate in the control room is conservatively assumed to be ~~12~~ 27 cfm, including 10 cfm due to ingress/egress ~~and~~ 2 cfm leakage from the ductwork passing through the control room pressure boundary, and 15 cfm from other sources. Filtered inleakage through the closed dampers due to the pressure differential is also included. The damper leakage air will be filtered by the recirculation filtration units.

Because the control room door ingress/egress is to a stairwell which is equivalent to a two-door vestibule, backflow will not occur with the CPSES CREFS design and the 10 cfm is not applicable per SRP 6.4. The ductwork has all welded joints which were leak tested prior to operation. Therefore, the assumed unfiltered inleakage from adjacent areas is conservative with respect to the SRP review criteria.

j. Habitability of the control room is based on the following occupancy factors:

Time Period	Occupancy Factor
0 to 24 hours	1.0
1 to 4 days	0.6
4 to 30 days	0.4

k. The air volume in the control room used to determine exposures to operators is 423,032 ft3.

l. The models for the major contributors to the control room dose are provided in Appendix 15B.

Using the above assumptions and procedures, the thyroid dose is conservatively calculated to be ~~27.4~~ 43 rem in the control room for the duration of the accident. The thyroid dose can be further reduced by the use of the available respiratory protection equipment. The total whole body gamma dose is conservatively calculated to be ~~3.76~~ 2.3 rem. This calculated dose includes whole body dose contributions from containment sources (both direct and scattered radiation), the external passing cloud, control room atmosphere, activity buildup on filters, and streaming through doors and penetrations. These calculated doses are less than the limiting values specified in ~~10CFR50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 19.~~ Regulatory Guide 1.195.

The skin dose received in the control room during the accident period is conservatively calculated to be ~~44.0~~ 16.0 rem. This calculated beta skin dose is less than the ~~75~~ 50 rem limit allowed by Regulatory Guide 1.195 if special protective clothing and eye protection ~~are~~ is not used. ~~Therefore,~~ However, special protective clothing and eye protection are provided for use, if required, to reduce the beta skin doses to the operators ~~to within acceptable limits in accordance with 10CFR50, Appendix A, GDC 19.~~

6. ~~5.~~ Environmental consequences of containment purging to control containment hydrogen concentration after a LOCA

Purging of the containment atmosphere provides a backup method for controlling potential hydrogen accumulation in the Containment following a postulated LOCA. The use of the hydrogen purge system (see Section 6.2.5.2.2) is precluded by redundant electric hydrogen recombiners located in the Containment Building (see Section 6.2.5.2.1). The electric hydrogen recombiners are the primary means of controlling post-LOCA hydrogen buildup. Thus, an analysis of the radiological consequences of containment purging is not provided.

RELOCATE to PAGE 15.6-20 as #3

3 ~~6.~~ Environmental consequences of releases through the containment pressure relief line in the event of a LOCA

An analysis of the radioactive effluents escaping the Containment to the environment after a LOCA, via the line through the controlled access area exhaust system, was performed using the following assumptions:

a. The maximum containment air/steam mass release to the environment was conservatively calculated assuming containment pressures for a large break LOCA, and critical flow via the 3-3/8 inch orifice plate at the inlet to the pressure relief system ductwork. No credit was taken for line losses in the ductwork or two butterfly valves.

b. Only reactor coolant activity is assumed to be released. ~~A spectrum of line breaks from 1 inch to 6 inches were analyzed.~~ It was conservatively assumed that all iodine and noble gas activity in the primary coolant was instantaneously released to the containment atmosphere.

c. ~~A preaccident iodine spike was considered in determining the primary reactor coolant activity. The corresponding reactor coolant iodine concentrations are listed in Table 15.6-3.~~ The primary coolant iodine activity is assumed to be 1.0 micro curie per gram dose equivalent I-131. The ~~noble gas~~ activity concentrations are presented in Table 15.1-4. The noble gas activity is based on operation with 1% fuel defects.

d. The containment pressure relief line isolation valve closure time including instrumentation delays will not exceed 5 seconds. ~~The radioactive fission products are assumed to be released from the Containment through the pressure relief line for a period of time equal to a typical small break LOCA reactor trip signal time (plus 5 seconds).~~ The analysis did not consider the reduction of mass flow during the valve closure time of five seconds; full flow was assumed until the lapse of five seconds.

e. No credit was taken for radioactive decay.

f. No credit was taken for an elevated release.

17a. Deleted

18. Deleted

19. Deleted

20. Deleted

20a. Deleted

21. DiNunno, J.J., Anderson, F. D., Baker, P.E. and Waterfield, R. L., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, U. S. Nuclear Regulatory Commission, Division of Licensing and Regulation, March 1962.

22. Postma, A. K., and Pasedag, W. F., "A Review of Mathematical Models for Predicting Spray Removal of Fission Products in Reactor Containment Vessels," WASH-1329, U. S. Nuclear Regulatory Commission, Accident Analysis Branch.

23. Meteorology and Atomic Energy," U. S. Nuclear Regulatory Commission, Division of Technical Information, 1968.

24. Murphy, K. G. and Campe, Dr. K. M., ANuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19,@ U. S. Nuclear Regulatory Commission

25. Deleted

26. Boatwright, W. J., Maier, S. M. and Lo, S. S., "Design Basis Analysis of a Postulated Steam Generator Tube Rupture Event for Comanche Peak Steam Electric Station, Unit 1", RXE-88-101, TU Electric, March, 1988.

27. Deleted

28. Deleted

29. Deleted

30. Deleted

31. Deleted

32. Deleted

33. Deleted

34. Regulatory Guide 1.195, "Methods and Assumptions For Evaluating Radiological Consequences Of Design Basis Accidents At Light-Water Nuclear Power Reactors," May 2003.

TABLE 15.6-2

(Sheet 1 of 2)

PARAMETERS FOR POSTULATED STEAM GENERATOR TUBE RUPTURE ACCIDENT ANALYSIS

		Parameter	Value
1.		Data and Assumptions Used to Estimate Radioactive Source Term For the Postulated Steam Generator Tube Rupture Accident	
	a.	Steam generator tube leak prior to and during accident (gpm)	1.0
	b.	Offsite power	lost
2.		Data and Assumptions Used to Estimate Activity Released to the Atmosphere	
	a.	Iodine partition factor in steam generators prior to and during accident	0.01
	b.	Iodine partition factor in condensers prior to accident	0.15
	c.	Duration of plant cooldown by secondary system after accident (hr)	8
3.		Dispersion Data	
	a.	Exclusion area boundary and low population zone distances	2080 m and 4 miles
	b.	x/Q (sec/m3)	
		0 - 2 hours	1.6×10^{-4}
		0 - 8 hours	$\boxed{2.4}$ ~~2.3~~ $\times 10^{-5}$
4.		Dose Data	
	a.	Method of dose calculation	See Appendix 15.B
	b.	Dose conversion assumptions	See Appendix 15.B

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TABLE 15.6-2

(Sheet 2)

PARAMETERS FOR POSTULATED STEAM GENERATOR TUBE RUPTURE ACCIDENT ANALYSIS

Parameter	Value
c. Doses (rem)	
With preaccident iodine spike	
@ exclusion area boundary	
(0 - 2 hours)	
Thyroid inhalation	~~46.2~~ 22.0
Whole body gamma	~~0.12~~ 0.080
@ low population zone	
(0 - 8 hours)	
Thyroid inhalation	~~7.8~~ 3.4
Whole body gamma	~~1.99 x 10-2~~ 0.013
With concurrent iodine spike	
@ exclusion area boundary	
(0 - 2 hours)	
Thyroid inhalation	~~17.5~~ 8.8
Whole body gamma	~~0.11~~ 0.077
@ low population zone	
(0 - 8 hours)	
Thyroid inhalation	~~6.73~~ 1.4
Whole body gamma	~~3.53 x 10-2~~ 0.012

TABLE 15.6-3

PREACCIDENT IODINE SPIKE CONCENTRATION IN THE PRIMARY COOLANT [(a)]

Spike Isotope	Preaccident Iodine Concentration (μCi/gm)
I-131	~~39.6~~ 46.8
I-132	~~39.6~~ 46.8
I-133	~~59.4~~ 72.0
I-134	~~8.1~~ 9.6
I-135	~~32.5~~ 38.4

(a) This concentration corresponds to 60 μCi/gm Dose Equivalent I-131, the maximum allowed at Rated Thermal Power by the Technical Specifications.

TABLE 15.6-4

IODINE APPEARANCE RATE TO THE REACTOR COOLANT AFTER THE ACCIDENT

Isotope	Iodine Appearance Rate[a] (Ci/~~sec~~ min)
I-131	~~1.46~~ 153
I-132	~~2.86~~ 489
I-133	~~3.34~~ 287
I-134	~~4.52~~ 213
I-135	~~3.20~~ 219

(a) Corresponds to ~~500~~ 335 times the equilibrium appearance rate based on operation with reactor coolant activity at 1.0 uCi/gm Dose Equivalent I-131.

~~NOTE: The primary coolant TS limit has been changed (reduced) to 0.45 uCi/gm Dose Equivalent I-131 (DEI-131). However, the applicable accident analyses have been demonstrated to be conservative using 1.0 uCi/gm DEI-131. The only exception is the Main Steam Line Break analysis that supports the implementation of the Steam Generator Alternate Repair Criteria (SG ARC). The SG ARC analysis assumes a DEI-131 value of 0.45 uCi/gm. This assumption maximizes the SG ARC benefit. Therefore, with the exception of the Main Steam Line Break analysis for the SG ARC, all reported doses assume a DEI-131 limit value of 1.0 uCi/gm.~~

TABLE 15.6-9

(Sheet 1 of 3)

PARAMETERS FOR POSTULATED LOCA ANALYSIS[(a)]

Data and assumptions used to estimate radioactive source from postulated accidents

a.	Power level (MWt)	3565
~~b.~~	~~Percent of fuel defected~~	~~1%~~
b ~~c.~~	Plateout of iodine activity released to containment	50%
c ~~d.~~	Activity released to containment and available for release to environment	
	Noble gases	100% of core inventory
	Iodines	25% of core inventory
d ~~e.~~	Iodine fractions (organic, elemental, and particulate)	Regulatory Guide ~~1.4, Regulatory Position 1.a~~ 1.195

2. Data and assumptions used to estimate activity released

a.	Containment volume	
	Directly sprayed volumes	1.706 ~~1.718~~ x 10^6 ft^3
	Unsprayed volumes affected by spray	1.325 ~~1.313~~ x 10^6 ft^3
	Total free volume	3.031 x 10^6 ft^3
b.	Mixing rate between sprayed and unsprayed volumes	2 turnovers per hour

(a) Per Regulatory Guide ~~1.4~~ 1.195

CPSES/FSAR

TABLE 15.6-9

(Sheet 2)

PARAMETERS FOR POSTULATED LOCA ANALYSIS[a]

c. Spray removal coefficient

Elemental iodine	10 hr^{-1}	
Organic iodines	0 hr^{-1}	
Particulte iodine	~~1.07~~ hr^{-1} 11.4 hr-1	for DF less than or equal to 50
	1.14 hr-1	for DF greater than 50

d. Effective decontamination factor of spray for elemental iodine — 100

e. Containment leak rate — 0.1% containment volume per day, (0≤t≤ 24 hrs) 0.05% of containment volume per day, (t > 24 hrs)

f. Duration of removal effectiveness

Elemental iodine	~~1.715~~ hr 2.518 hr	(DF of 100 attained)
Particulate iodine	~~8.45~~ hr 2.03 hr	(DF of 50 attained)
	4.0 hr	sprays assumed to be terminated

3. Dispersion data

a. Exclusion area boundary and LPZ distances — 2080 m and 4 miles

b. x/Q's (for time intervals of 2 hours, 8 hours, 24 hours, 4 days, 30 days)

@ EAB, onsite
5 percentile data
1.6×10^{-4} sec/m^3
(0-2 hrs)

@LPZ, onsite
5 percentile data
2.4e-05 ~~2.3 x 10^{-5}~~ sec/m^3
(0-8 hrs)
1.6e-05 ~~2.2 x 10^{-6}~~ sec/m^3
(8-24 hrs)
6.2e-06 ~~1.1 x 10^{-6}~~ sec/m^3
(1-4 days)
1.7e-06 ~~6.6 x 10^{-7}~~ sec/m^3
(4-30 days)

CPSES/FSAR

TABLE 15.6-9

(Sheet 3)

PARAMETERS FOR POSTULATED LOCA ANALYSIS[a]

4. Dose due to containment and ESF equipment leakage and the containment pressure relief line.

   a. Method of dose calculation — See Appendix 15B

   b. Dose conversion assumptions — See Appendix 15B

   c. Doses — @EAB, (0-2 hrs)
   thyroid = ~~88.9~~ rem 63.0
   whole body gamma = ~~1.4~~ rem 0.79

   @LPZ (0-30 days)
   thyroid = ~~36.3~~ rem 47.0
   whole body gamma = ~~1.1~~ rem 0.29

TABLE 15.6-10

ACTIVITY AVAILABLE FOR RELEASE VIA ESF COMPONENTS AT T=O FOLLOWING LOCA

ISOTOPE	ACTIVITY IN CURIES	
I-131	~~4.9×10^7~~	5.4×10^7
I-132	~~7.2×10^7~~	7.9×10^7
I-133	~~1.0×10^8~~	1.1×10^8
I-134	~~1.1×10^8~~	1.2×10^8
I-135	~~9.4×10^7~~	1.0×10^8

content of a single waste gas decay tank inventory of 200,000 Ci of noble gases (considered as dose equivalent Xe-133) is

alarm in the Control Room. The gaseous activity in the tanks then decays while the other tanks in the system are being filled with gaseous radioactivity. ~~#85#~~ The maximum activity that can be released as a result of a gas decay tank rupture is the activity stored in one gas decay tank immediately after it has been isolated from the GWPS.

The entire ~~contents of the tank are~~ assumed to be released to the Auxiliary Building, and all of the noble gases are assumed to leak from the building at ground level over a 2 hour period. Based on this model and an onsite 0 to 2 hr atmospheric dilution factor (1.6×10^{-4} sec/m^3), the whole body dose at the nearest point on the exclusion area boundary (EAB) is conservatively calculated to be ~~0.15 rem.~~ 0.19 This dose is substantially below the 25 rem whole body value set forth in 10CFR100; it may be concluded that such an incident would not interrupt or restrict public use of areas beyond the EAB.

~~A parametric study shows that a single waste gas decay tank inventory of 200,000 Ci of noble gases (considered as Xe-equivalent) would result in a conservatively calculated whole body gamma dose of 0.3 rem at the EAB, using the whole body dose conversion factor from reference [3].~~

15.7.2 RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE

15.7.2.1 Identification of Causes and Accident Description

The accident is defined as the uncontrolled atmospheric release from the 30,000 gallon floor drain tank due to the postulated rupture of the tank. This tank is the highest potential atmospheric release source term because of its large volume and the fact that it is assumed to be 80 percent full of reactor coolant.

15.7.2.2 Analysis of Effects and Consequences

The activity in the 30,000 gallon floor drain tank, the assumptions on which the activity is based, and the radiological consequences resulting from the release of the activity are discussed in Section 15.7.2.3.

15.7.2.3 Environmental Consequences

This analysis assumes that the plant has been operating with 1 percent failed fuel for an extended period sufficient to achieve equilibrium radioactive concentrations. Floor drain tank III is assumed to contain the inventory as indicated in Table 15.7-3. The entire contents of the tank are assumed to be released to the atmosphere at ground level over a 2 hr period. Other conservative assumptions are detailed in Table 15.7-4.

Based on the foregoing model, the thyroid and whole body doses at the EAB are conservatively calculated to be ~~5.14~~ 2.3 rem and ~~1.86×10^{-4}~~ 3.8E-03 rem, respectively. The doses from this accident are well within the values set forth in 10CFR100.

15.7.3 POSTULATED RADIOACTIVE RELEASES DUE TO LIQUID TANK FAILURES

The analysis is presented in Section 2.4.12 and 2.4.13.3.

15.7.4 DESIGN BASIS FUEL HANDLING ACCIDENTS

15.7.4.1 Identification of Causes and Accident Description

The accident is defined as dropping of a spent fuel assembly in the Containment Building or spent fuel pool fuel storage area floor resulting in the rupture of the cladding of all the fuel rods in the assembly despite many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under the direct surveillance of a supervisor.

15.7.4.2 Analysis of Effects and Consequences

Method of Analysis

The method of analysis used for evaluating the potential radiological consequences of a fuel handling accident is in compliance with Regulatory Guide ~~1.25 except for those~~ 1.195. ~~provisions listed in Appendix 1A(B).~~ A two hour, ground level release is assumed for the analysis.

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

1. The accident occurs at 100 hr following the reactor shutdown, the minimum time at which spent fuel could be first moved into the fuel storage area.

2. The accident results in the rupture of the cladding of all fuel rods in a single assembly.

3. The damaged assembly is, coincidentally, the one operating at the highest power level in the core region to be discharged.

4. The power in this assembly, and the corresponding fuel temperatures establish the total fission product inventory and the fraction of this inventory which is present in the fuel pellet-cladding gap at the time of reactor shutdown.

5. The fuel pellet-cladding gap inventory of fission products is released to the refueling cavity or spent fuel pool at the time of the accident.

6. The refueling cavity or spent fuel pool retains a large fraction of the gap activity of halogens by virtue of their solubility and hydrolysis. Noble gases are not retained by the water as they are not subject to hydrolysis reactions.

Fission Product Inventories

1.195.

The actual fission product gap inventory in the fuel assembly is dependent on the linear heat generation rate of the assembly and the temperature of the fuel. However, the gap inventories assumed in fuel handling accident analyses were based on the conservative guidance contained in Regulatory Guide ~~1.25.~~ Table 15.7-6 lists the fuel assembly fission product activities at the time of the fuel handling accident. These activities are consistent with the assumptions used in analyzing the environmental consequences of postulated fuel handling accidents detailed in Section 15.7.4.3.

Iodine Decontamination Factors

An experimental test program [1] was conducted to evaluate the extent of removal of iodine released from a damaged irradiated fuel assembly. Iodine removal from the released gas takes place as the gas rises through the body of solution in the fuel storage area to the pool surface. The extent of iodine removal is determined by mass transfer from the gas phase to the surrounding liquid and is controlled by the bubble diameter and the contact time of the bubble in the solution.

In order to obtain all the necessary information regarding this mass transfer process, a number of small scale tests were conducted, using trace iodine and carbon dioxide in an inert carrier gas. Iodine testing was performed at the design basis solution conditions (temperature and chemistry) and data were collected for various bubble diameters and solution depths. This work resulted in the formulation of a mathematical expression for the iodine decontamination factor in terms of bubble size and bubble rise time.

Similar tests were conducted with carbon dioxide in an inert carrier, except that the solution temperature and chemistry were patterned after that of a deep pool where large scale tests were also performed with carbon dioxide. The small scale carbon dioxide tests also resulted in a mathematical expression for the decontamination factor in terms of bubble size and bubble rise time through the solution.

To complete the experimental program, a full size fuel assembly simulator was fabricated and placed in a deep pool for testing, where the gas released would be typical of that from the postulated damaged assembly. Tests were conducted with trace carbon dioxide in an inert carrier gas and overall decontamination factors were measured as a function of the total gas volume released. These measurements, combined with the analytical expression derived from small scale tests with carbon dioxide, permitted an in-situ measurement of both the effective bubble diameter and rise time, both as a function of the volume of gas released. Having measured the characteristics of large scale gas releases, the decontamination factor for iodine was obtained, using the analytical expression from small scale iodine testing.

Decontamination Factor = $7.3\ e^{0.313\ t/d}$

where

t = rise time (sec)

for a fuel rod internal pressure of less than 1500psia

d = effective bubble diameter (cm)

The overall test results clearly indicate that iodine will be readily removed from the gas rising through the fuel storage area solution and that the efficiency of removal will depend on the volume of gas released instantaneously from the fuel void space.

With consideration given to the total quantity of gas released from a fuel assembly, a typical pool decontamination factor for iodine is at least 760 for the 26 foot depth. 160 However, a much lower decontamination factor of 100 is conservatively selected to provide for deviations in the factors which control iodine absorption by the pool water.

15.7.4.3 Environmental Consequences

15.7.4.3.1 Postulated Fuel Handling Accident Outside Containment

The analysis of a postulated fuel handling accident is performed as follows:

1. The accident is assumed to occur 100 hr after plant shutdown.

2. All of the rods in one fuel assembly are ruptured.

3. The damaged assembly 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 and a 100 hour shutdown. A radial peaking factor of 1.65 is used.

4. The minimum water depth between the top of the damaged fuel rods and the spent fuel pool surface is 23 ft. 21

5. All of the gap activity in the damaged rods is released to the spent fuel pool. This activity consists of 10 percent of the total noble gases other than krypton-85, 30 percent of the krypton-85, and 10 percent of the total radioactive iodine in the rods at the time of the accident. The gap activities released as a result of this accident are presented in Table 15.7-6. 5 10 5 other than iodine-131, 8% for iodine-131,

6. All activity released from the spent fuel pool is released at ground level to the environment over a 2 hour period.

7. The iodine gap inventory is composed of inorganic species (99.75 percent) and organic species (0.25 percent).

8. The overall decontamination factor for the spent fuel pool is 100. 160

9. No credit is taken for iodine filtration by the primary plant ventilation system.

10. Atmospheric diffusion conditions are assumed to be the 0 to 2 hr ground level case.

The parameters used for this analysis are listed in Table 15.7-7.

Based on the foregoing assumptions, the thyroid and whole body doses at the EAB are conservatively calculated to be ~~53.9~~ 22.0 rem and ~~0.44~~ 0.087 rem, respectively. The corresponding doses at the LPZ are conservatively calculated to be ~~7.76~~ 3.3 rem and ~~6.29 x 10^-2~~ 0.013 rem. The calculated doses are within the values set forth in 10CFR100.

15.7.4.3.2 Postulated Fuel Handling Accident Inside Containment

An analysis of the radiological consequences of a fuel handling accident inside the Containment Building would use the same assumptions and yield the same results as those of a fuel handling accident outside the Containment Building. The accident described in the preceding section is considered to represent the limiting case; therefore, no specific analysis of such an accident inside the Containment is provided.

15.7.4.4 Conclusions

The possibility of a fuel handling accident is relatively small due to the many physical, administrative, and safety restrictions imposed on fuel handling operations. However, based on conservative design basis parameters, the calculated doses from a postulated fuel handling accident are within the values set forth in 10CFR100.

15.7.5 SPENT FUEL CASK DROP ACCIDENTS

The Comanche Peak Steam Electric Station Fuel Handling Building overhead crane satisfies NUREG-0554 single-failure-proof requirements and is designed to the requirements of seismic Category I (See Section 3.2). As such it can retain the maximum design load during a Safe Shutdown Earthquake and remain in place under all postulated seismic loadings. The crane is also provided with interlocks which prevent a fuel cask from being lifted more than 29.25 ft above floor elevation or from passing over the new fuel storage area during the spent fuel cask mode of operation (see Section 9.1.2.2). The crane does not pass over the spent fuel pool in any mode of operation. Based on this design approach, the radiological consequences of a spent fuel cask drop accident need not be evaluated.

REFERENCES

1. Bell, M. J., Duhn, E. R., Locante, J. and Malinowski, D. D., "Radiological Consequences of a Fuel Handling Accident," WCAP- 7518-L (Proprietary) and WCAP-7828 (Non-Proprietary), June 1970.

2. "Radiation Analysis Manual, Standard Plant Model 412, "Rev. 3, Westinghouse Electric Corporation, November 1978.

3. Killough, G. G., Begovich, C. L., Sjoreen, A. L. and Bell, L. W., "A Guide for the TACT III Computer Code," NUREG/CR-3287, Oak Ridge National Laboratory, May 1983.

TABLE 15.7-1

(Sheet 1 of 2)

PARAMETERS FOR POSTULATED WASTE GAS DECAY TANK RUPTURE ACCIDENT

1. Data and assumptions used to estimate radioactive source from postulated accidents

a.	Power level (MWt)	3565
b.	Percent of fuel defected (%)	1
c.	Release of activity by nuclide	Table 15.7-2

Values increased to equal 200,000 Ci of dose equivalent Xe-133

2. All pertinent data and assumptions used to estimate activity released

   a. The maximum content of the decay tank is considered for computing noble gas inventory

   b. Radiological decay is considered for minimum time required to transfer the gases from the primary system to the decay tank

   c. Entire content of one tank is assumed to be released to the building due to failure

   d. All of the noble gas is assumed to be released to the atmosphere over a 2 hr period

3. Dispersion data

a.	Exclusion area boundary (EAB) and low population zone (LPZ) distances	2080 m and 4 miles
b.	x/Q	1.6×10^{-4} sec/m3 (0 - 2 hr)

TABLE 15.7-1

(Sheet 2)

PARAMETERS FOR POSTULATED WASTE GAS DECAY TANK RUPTURE ACCIDENT

4.	Dose data		
	a.	Method of dose calculations	Appendix 15B ~~Regulatory Guide 1.24~~ ~~(a)~~
	b.	Doses	dose = ~~0.15~~ 0.19 @EAB, whole body rem (gamma dose)

~~(a)~~ ~~Also reflected in Appendix 15B, except that average gamma energies are taken from Westinghouse Radiation Analysis Manual [2].~~

based on assumptions below

TABLE 15.7-2

GAS DECAY TANK INVENTORY FOR ACCIDENT ANALYSIS (ONE GAS DECAY TANK)

Isotope	Tank Activity (Ci)	Tank activity corresponding to 200,000 Ci DEQ Xe-133
Kr-85	2.72×10^3	4.49×10^3
Kr-85m	3.23×10^2	5.33×10^2
Kr-87	4.42×10^1	7.30×10^1
Kr-88	4.08×10^2	6.74×10^2
Xe-131m	6.80×10^2	1.12×10^3
Xe-133	7.82×10^4	1.29×10^5
Xe-133m	5.10×10^2	8.42×10^2
Xe-135	1.67×10^3	2.76×10^3

Assumptions

1. 3565 MWT
2. 40 Year continuous operation at 1 percent fuel defects
3. 100 percent stripping efficiency
4. Volume control tank purge rate of 0.7 standard cubic feet per minute (scfm)
5. Tanks switched at regular intervals
6. 8 gas decay tanks shared between two units

TABLE 15.7-3

(Sheet 1 of 3)

MAXIMUM RADIOACTIVITY IN A FLOOR DRAIN TANK FOR ACCIDENT ANALYSIS

Isotope	Floor Drain Tank Activity (Ci)	Floor Drain Tank Activity Used to Determine the offsite And Control Room Dose Consequences (Ci)
Br-83	8.63×10^{0}	N/A
Br-84	4.27×10^{0}	N/A
Br-85	5.45×10^{-1}	N/A
I-130	1.91×10^{0}	N/A
I-131	2.54×10^{2}	2.79×10^{2}
I-132	2.54×10^{2}	2.79×10^{2}
I-133	3.82×10^{2}	4.20×10^{2}
I-134	5.18×10^{1}	5.70×10^{1}
I-135	2.09×10^{2}	2.30×10^{2}
Rb-86	2.00×10^{0}	N/A
Rb-88	4.36×10^{2}	N/A
Rb-89	1.91×10^{1}	N/A
Cs-134	2.09×10^{2}	N/A
Cs-136	2.63×10^{2}	N/A
Cs-137	1.36×10^{2}	N/A
Cs-138	8.72×10^{1}	N/A
Ba-137m	1.27×10^{2}	N/A
H-3	3.18×10^{2}	N/A
Cr-51	5.00×10^{-1}	N/A
Mn-54	4.00×10^{-2}	N/A
Mn-56	1.82×10^{0}	N/A
Fe-55	1.82×10^{-1}	N/A
Fe-59	4.72×10^{-2}	N/A
Co-58	1.36×10^{0}	N/A
Co-60	1.73×10^{-1}	N/A

CPSES/FSAR

TABLE 15.7-3

(Sheet 2)

MAXIMUM RADIOACTIVITY IN A FLOOR DRAIN TANK FOR ACCIDENT ANALYSIS

Isotope	Floor Drain Tank Activity (Ci)	Floor Drain Tank Activity Used to Determine the Offsite and Control Room Dose Consequences (Ci)
Sr-89	3.91×10^{-1}	N/A
Sr-90	1.09×10^{-2}	N/A
Sr-91	5.63×10^{-1}	N/A
Sr-92	1.18×10^{-1}	N/A
Y-90	3.09×10^{-3}	N/A
Y-91m	3.00×10^{-1}	N/A
Y-91	5.18×10^{-2}	N/A
Y-92	1.09×10^{-1}	N/A
Y-93	3.45×10^{-2}	N/A
Zr-95	5.90×10^{-2}	N/A
Nb-95	5.90×10^{-2}	N/A
Mo-99	6.81×10^{1}	N/A
Tc-99m	6.27×10^{1}	N/A
Ru-103	5.18×10^{-2}	N/A
Ru-106	1.27×10^{-2}	N/A
Rh-103m	5.18×10^{-2}	N/A
Rh-106	1.27×10^{-2}	N/A
Ag-110m	1.27×10^{-1}	N/A
Te-125m	2.54×10^{-2}	N/A
Te-127m	2.63×10^{-1}	N/A
Te-127	1.09×10^{0}	N/A
Te-129m	1.73×10^{0}	N/A
Te-129	1.64×10^{0}	N/A
Te-131m	2.36×10^{0}	N/A
Te-131	1.09×10^{0}	N/A
Te-132	2.63×10^{1}	N/A

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TABLE 15.7-3

(Sheet 3)

MAXIMUM RADIOACTIVITY IN A FLOOR DRAIN TANK FOR ACCIDENT ANALYSIS

Isotope	Floor Drain Tank Activity (Ci)	Floor Drain Tank Activity Used to Determine the Offsite and Control Room Dose Consequences (Ci)
Te-134	2.73×10^{0}	N/A
Ba-140	3.81×10^{-1}	N/A
La-140	1.27×10^{-1}	N/A
Ce-141	5.72×10^{-2}	N/A
Ce-143	4.54×10^{-2}	N/A
Ce-144	3.54×10^{-2}	N/A
Pr-143	5.72×10^{-2}	N/A
Pr-144	3.54×10^{-2}	N/A

Assumptions

a. 3565 MWt

b. Based on 1 percent failed fuel

c. Floor drain tank of 30,000 gal capacity assumed 80 percent full of reactor coolant

TABLE 15.7-4

(Sheet 1 of 2)

PARAMETERS FOR POSTULATED RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE

1. Data and assumptions used to estimate radioactive source from postulated accident

   a. Power level (MWt) — 3565

   b. Percent of fuel defected (%) — 1

   c. Release of activity by nuclide — Table 15.7-3

2. Data and assumptions used to estimate activity release

   a. Entire content of tank is assumed to be released to the building due to failure

   b. ~~All radioactive material that escapes from the tank is assumed to be released to the atmosphere over a 2 hr period at ground level~~

   Activity release to the atmosphere takes place over a two hour period at ground level.

   c. Iodine Partition factor is 0.1

3. Dispersion data

   a. EAB distance — 2080 m

   b. x/Q — 1.6 x 10-4 sec/m^3 (0-2 hr)

4. Dose data

   a. Method of dose calculations — See Appendix 15B~~(a)~~

   b. Doses — @EAB

   thyroid dose = $\boxed{2.3}$ ~~3.16~~ rem

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TABLE 15.7-4

(Sheet 2)

PARAMETERS FOR POSTULATED RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE

whole body dose = $\boxed{0.0038}$ ~~1.14 x 10-1~~ rem (gamma dose)

~~(a)~~ ~~Except that average gamma energies are taken from Westinghouse Radiation Analysis Manual [2].~~

TABLE 15.7-6

Noble Gas and Iodine Activities Released ~~to the Environment~~ from Damaged Fuel Rods As a Result of a Fuel Handling Accident[1]

Nuclide	Curies Released
Kr-85	$\cancel{2.35 \times 10^{+3}}$ 8.62e+02
Xe-131m	$\cancel{5.47 \times 10^{+2}}$ 3.01e+02
Xe-133m	$\cancel{1.01 \times 10^{+4}}$ 5.56e+03
Xe-133	$\cancel{1.26 \times 10^{+5}}$ 6.93e+04
Xe-135m	$\cancel{6.56 \times 10^{+1}}$ 3.61e-01
Xe-135	$\cancel{2.19 \times 10^{+2}}$ 1.20e+02
I-131	$\cancel{6.22 \times 10^{+4}}$ 5.47e+04
I-132	$\cancel{5.21 \times 10^{+4}}$ 2.87e+04
I-133	$\cancel{6.31 \times 10^{+3}}$ 3.47e+03
I-135	$\cancel{4.29 \times 10^{+0}}$ 2.36e+00

1. Based on the following assumptions:

    a. The total gap inventory of one fuel assembly in the discharge region is released to the spent fuel pool.

    b. A radial peaking factor of 1.65 is applied.

    c. The accident occurs 100 hours after shutdown.

    d. ~~Gap activities correspond to 10% of the core inventory except for Kr-85 which is 30% of the core inventory.~~

    Gap activities are 8% for I-131 and 5% for all other iodines and are 10% for Kr-85 and 5% for all other noble gases.

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TABLE 15.7-7

(Sheet 1 of 2)

PARAMETERS FOR POSTULATED FUEL HANDLING ACCIDENT ANALYSIS

1.	Data and assumptions used to estimate radioactive source from postulated accidents		
	a.	Power level (MWt)	3565
	b.	Release of activity	Regulatory Guide 1.25 1.195
	c.	Damage to fuel assembly	All rods ruptured
	d.	Iodine fractions (inorganic and organic)	Regulatory Guide 1.25 1.195
2.	Data and assumptions used to estimate activity released		
	a.	Spent fuel pool decontamination factor	100 160
	b.	All other pertinent data and assumptions	Regulatory Guide 1.25 1.195
3.	Dispersion data		
	a.	EAB and LPZ distances	2080 m and 4 miles
	b.	x/Q	@EAB 1.6 x 10-4 sec/m^3 (0 - 2 hr) @LPZ 2.4 2.3 x 10-5 sec/m^3 (0 - 8 hr)
4.	Dose data		
	a.	Method of dose calculation	See Appendix 15B
	b.	Dose conversion assumptions	See Appendix 15B
	c.	Doses	@EAB,

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TABLE 15.7-7

(Sheet 2)

PARAMETERS FOR POSTULATED FUEL HANDLING ACCIDENT ANALYSIS

thyroid dose = ~~53.9~~ rem, $\boxed{22.0}$
whole body dose = ~~0.44~~ rem $\boxed{0.087}$
(gamma dose)
@ LPZ
thyroid dose = ~~7.75~~ rem, $\boxed{3.3}$
whole body dose = ~~6.29 x 10-2~~ rem $\boxed{0.013}$
(gamma dose)

9. Locked Reactor Coolant Pump Rotor.

APPENDIX 15B

DOSE MODELS USED TO EVALUATE THE ENVIRONMENTAL CONSEQUENCES OF ACCIDENTS

15B.1 INTRODUCTION

This section identifies the models used to calculate offsite radiological doses that would result from releases of radioactivity due to various postulated accidents. The postulated accidents are:

1. Steam Line Break
2. Rod Ejection Accident
3. Steam Generator Tube Failure
4. Loss-of-Coolant Accident
5. Radioactive Gas Waste System Leak or Failure
6. Radioactive Liquid Waste System Leak or Failure
7. Design Basis Fuel Handling Accident
8. Primary Coolant Small Line Break Outside Containment

15B.2 ASSUMPTIONS

The following assumptions are basic to the model for the gamma and beta dose due to immersion in a cloud of radioactivity and the model for the thyroid dose due to inhalation of radioactivity.

1. The dose contribution of direct radiation from sources other than the leakage cloud is negligible compared to the dose due to immersion in the leakage cloud.
2. All radioactivity releases are treated as ground level releases regardless of the point of discharge.
3. The dose receptor is a standard man as defined by the International Commission on Radiological Protection (ICRP)[1].
4. Radioactive decay from the point of release to the dose receptor is neglected.
5. Isotopic ~~beta and gamma decay energies are taken from Table of Isotopes [2] and from reference [7].~~ whole body and beta skin dose conversion factors are taken from references [13] and [14] respectivily.
6. Atmospheric dispersion factors used in the analyses are presented in Section 2.3.

15B.3 GAMMA DOSE

The gamma dose is obtained by considering the dose receptor to be immersed in a radioactive cloud which is infinite in all directions above the ground plane, i.e., a semi-infinite cloud. The concentration of radioactive material within this cloud is taken to be uniform and equal to the maximum centerline ground level concentration that would exist in the cloud at the appropriate distance from the point of release.

The whole body dose is a result of exposure to external gamma radiation. The whole body dose due to immersion in a semi-infinite cloud is given by ~~[3]:~~

$$\cancel{D_\gamma = 0.25 \cdot X/Q \sum_i A_{R_i} \cdot \bar{E}_{\gamma i}} \quad (15B-1)$$

INSERT EQUATION 15B-1 AND DEFINITIONS FROM THE FOLLOWING PAGE

where:

D_γ is the whole body gamma dose from immersion in a semi-infinite cloud for a given time period, rem.

A_{Ri} is the activity of isotope i released during a given time period, curies

X/Q is the atmospheric dilution factor for a given time period, sec/m3

~~$\bar{E}_{\gamma i}$ is the average gamma radiation energy emitted by isotope i per disintegration, MeV/dis~~

~~The gamma energies presented in Table 15B-1 include the X-rays and annihilation gamma rays if they are prominent in the electromagnetic spectrum.~~

15B.4 THYROID INHALATION DOSE

The thyroid inhalation dose is obtained from the following expression ~~[4]:~~

$$\cancel{D_{THY} = X/Q \cdot B \cdot \sum_i Q_i \cdot (DCF)_i} \quad (15B-2)$$

INSERT EQUATION 15B-2 AND DEFINITIONS FROM FOLLOWING PAGE

where:

D_{THY} = thyroid inhalation dose, rem

X/Q = atmospheric dilution factor for a given time period, sec/m^3

B = breathing rate for a given time period t, m^3/sec

Q_i = total activity of iodine isotope i released in time period t, curies

$(DCF)_i$ = dose conversion factor for iodine isotope i, ~~rem/curie~~ Sv/Bq inhaled

Equations for page 15B-2

Replace Equation (15B-1)

$$D_{\gamma} = K \cdot \chi/Q \sum_{i} A_{Ri} \cdot DCF_{\gamma i}$$

Insert

K is the conversion factor 3.7E12 rem-Bq/Ci-Sv

$DCF_{\gamma i}$ is the whole body gamma dose conversion factor for isotope I, Sv-m^3/Bq-sec

Replace Equation (15B-2)

$$D_{THY} = K \cdot \chi/Q \cdot B \cdot \sum_{i} Q_i \cdot (DCF)_i$$

Insert

K = the conversion factor 3.7E12 rem-Bq/Ci-Sv

The thyroid inhalation dose conversion factors and offsite breathing rates used in the above model are given in Table 15B-1.

15B.5 CONTROL ROOM DOSE

The thyroid inhalation, whole body gamma, and beta skin dose models for the major contributors to the Control Room dose are described below. The dose to the Control Room occupants due to a postulated accident is calculated on the basis of source strength, atmospheric transport, dosimetry and Control Room emergency pressurization and filtration considerations as illustrated in the following equations ~~[7]~~.

The thyroid inhalation dose is obtained from the following expression:

$$\cancel{D_{THY}} = \sum_i \sum_j \cancel{BR \cdot DCF_i} \cdot \int_{T_{j-1}}^{T_j} \cancel{A_j(t)dt} \qquad (15B\text{-}3)$$

INSERT EQUATION 15B-3 AND DEFINITIONS FROM FOLLOWING PAGE

where:

i	=	isotope index
j	=	time interval index
BR	=	breathing rate, m^3/sec
DCF_i	=	dose conversion factor for iodine isotope i, ~~rem/curie~~ Sv/Bq inhaled
$A_i(t)$	=	airborne concentration of iodine isotope i at time t (sec), in the Control Room, $curies/m^3$

The thyroid inhalation dose conversion factors and Control Room breathing rate used in the above model are presented in Table 15B-1.

The whole body gamma dose due to inleakage is calculated using the following equation:

$$\cancel{D_R} = (\cancel{R/2}) \cancel{\Sigma A_j (K_R)}_i \qquad (15B\text{-}4)$$

INSERT EQUATION 15B-4 AND DEFINITIONS FROM FOLLOWING PAGE

where:

D_R	=	Whole body gamma dose, rem
~~R~~	~~=~~	~~radius of an equivalent hemispherical control room, m~~
A_i	=	time integrated concentration of nuclide i, ~~Ci-hr/m³~~ Ci-sec/m3
~~$(K_R)_i$~~	=	~~dose conversion factor for nuclide i, Rem-m³/Ci-hr defined by equation 15B-5~~

Equations for page 15B-3

Replace Equation (15B-3)

$$D_{THY} = K \cdot \sum_i \sum_j BR \cdot DCF_i \cdot \int_{Tj-1}^{Tj} A_j(t)dt$$

Insert

K = the conversion factor 3.7E12 rem-Bq/Ci-Sv

Replace Equation (15B-4)

$$D_R = K \cdot 1/GF \cdot \sum_i A_i \cdot (DCF)_{\gamma i}$$

Insert

K = the conversion factor 3.7E12 rem-Bq/Ci-Sv

GF = Geometry Factor to adjust the dose to reflect the finite cloud in the control room

$DCF_{\gamma i}$ = the whole body gamma dose conversion factor for isotope i, Sv-m^3/Bq-sec

INSERT EQUATION 15B-5 AND DEFINITIONS FROM FOLLOWING PAGE

$$\cancel{(K_R)_i = 3.7\times10^6 \sum_j S_{ij}C_j} \qquad (15B\text{-}5)$$

~~S_{ij}~~ - ~~gamma energy emitted per disintegration for nuclide i at energy E_j, Mev/dis~~

~~C_j~~ - ~~flux to dose rate conversion factor for gamma energy E_j, rem-cm²-sec/Mev-hr [5]~~

~~3.7×10^6~~ - ~~Units conversion factor, dis-m³/Ci-sec-cm³~~

The whole body gamma dose conversion factors ~~(K_R), and gamma energies E_j,~~ are presented in Table 15.B-1

The beta skin dose due to inleakage is calculated using the following equation ~~[3]:~~

$$\cancel{D}_B = \cancel{0.23} \sum \cancel{A}_i E_{B_i} \qquad (15B\text{-}6)$$

INSERT EQUATION 15B-6 AND DEFINITIONS FROM FOLLOWING PAGE

where:

DB = beta skin dose from immersion in a semi-infinite cloud, rem

Ai = time integrated concentration of nuclide i, Ci-sec/m3

~~E_{Bi}~~ = ~~average beta energy emitted by isotope i, MeV/dis~~

The beta ~~energies include conversion electrons if they are prominent in the~~ skin dose conversion factors ~~electromagnetic spectrum and are computed as one-third the maximum beta energy for a given spectrum. The values of E_B~~ are presented in Table 15.B-1.

The whole body gamma dose to Control Room personnel due to a cloud external to the Control Room is calculated using the following equation:

$$D_C = \sum_j \left(X/Q_j \cdot \sum_j A_{ij} \cdot CF \right) \qquad (15B\text{-}7)$$

where:

D_c = whole body gamma dose due to external cloud shine, rem

X/Q_j = atmospheric dispersion factor for the time period j, calculated at the Control Room air intake assuming a ground level release, sec/m3

A_{ij} = total activity of nuclide i released during time period j, Ci

Equations for page 15B-4

Replace Equation (15B-5)

$$GF = 1173/V^{0.338}$$

Insert

V = control room volume, ft^3

Replace Equation (15B-6)

$$D_B = K \cdot \sum_i A_i \cdot (DCF)_{Bi}$$

Insert

K = the conversion factor 3.7E12 rem-Bq/Ci-Sv

DCF_{Bi} = the beta skin dose conversion factor for isotope i, Sv-m^3/Bq-sec

ANISND solves the one-dimensional Boltzmann transport equation for neutrons or gamma rays in rectangular, spherical or cylindrical geometry. The source may be fixed, fission, or a subcritical combination of the two. Criticality search may be performed on any one of several parameters. Cross sections may be weighted using the space and energy dependent flux generated in solving the transport equation. The external cloud dose calculations use Bugle 80, a P-3 cross section library.

5. EFFLMDA [10]

EFFLMDA calculates the effective spray removal constant as a function of time. The calculation is based on a multi- region model with intermixing among the sprayed and unsprayed regions in a reactor containment.

The EFFLMDA results are merged with DRAGON input data. This enables the DRAGON program, a single spray region computer code, to simulate a multi-region spray model with step changes in the effective spray constants.

6. ~~TITAN5 [12]~~ RADTRAD [12]

Program RADTRAD ~~TITAN5 performs~~ performs similar analysis functions as described above for DRAGON.

REFERENCES

1. ICRP Publication 2, Report of Committee II on Permissible Dose for Internal Radiation (1959). The International Commission on Radiological Protection.

2. ~~Table of Isotopes, sixth edition, by C.M. Lederer, J.M. Hollander, I. Perlman. University of California Berkeley Lawrence Radiation Laboratory.~~ Not Used.

3. ~~Regulatory Guide 1.4, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors, Revision 2, June, 1974, U.S. Nuclear Regulatory Commission.~~ Not Used.

4. ~~TID 14844, Calculation of Distance Factors for Power and Test Reactor Sites, by J.J. DiNunno, F.D. Anderson, P.E. Baker and R.L. Waterfield, Division of Licensing and Regulation, U.S. Nuclear Regulatory Commission.~~ Not Used.

5. ~~TID-7004, Reactor Shielding Design Manual, March 1956.~~ Not Used.

6. RADIOISOTOPE, SWEC proprietary computer code NU-007, Version 01, Level 02, April, 1985.

7. DRAGON, SWEC proprietary computer code NU-115, Version 04, Level 02, September 1984 and Version 05 Level 00, April, 1986.

8. QADMOD, SWEC proprietary computer code NU-137U, Version 00, Level 03, November 1985.

9. ANISND, SWEC proprietary computer code NU-146, Version 01, Level 01, January 1985.

10. EFFLMDA, SWEC proprietary computer code NU-159, Version 01, Level 00, December, 1984.

11. International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, Volume 3 No. 1-4, 1979.

~~12. TITAN5, Westinghouse proprietary computer code, Version 4.10, October 7, 1993.~~

12. NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," December 1997.

NUREG/CR-6604, Supplement 1, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," June 1999.

NUREG/CR-6604, Supplement 2, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," October 2002.

13. K. F. Eckerman and J.C. Ryman, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report No. 12, EPA 402-R-93-081, Environmental Protection Agency, September 1993.

14. DOE/EH-0070, "External Dose-Rate Conversion Factors for Calculation of Dose to the Public," July 1988.

15. NRC Regulatory Issue Summary 2001-19: "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests," 10/18/01.

TABLE 15B-1

(Sheet 1 of 2)

Delete and Replace with following Table 15B-1

PHYSICAL DATA FOR DOSE CONVERSIONS

~~Isotope~~	~~E Average Beta Energy (MeV/dis)[2][7]~~	~~E Average Gamma Energy (MeV/dis)[2][7]~~	~~DCF Thyroid Dose Conversion Factor (rem/Ci)[4]~~	~~DCF Thyroid Dose Conversion Factor Used for SLB (rem/Ci)[11]~~	~~K_γ (**) Gamma Dose Converson Factor Used for Control Room (rem-m^3/Ci-hr)~~
~~I-131~~	~~0.209~~	~~0.375~~	~~1.48×10^6~~	~~1.07×10^6~~	~~2.84~~
~~I-132~~	~~0.421~~	~~2.29~~	~~5.35×10^4~~	~~6.29×10^3~~	~~17.2~~
~~I-133~~	~~0.403~~	~~0.636~~	~~4.00×10^5~~	~~1.81×10^5~~	~~3.53~~
~~I-134~~	~~0.558~~	~~2.51~~	~~2.50×10^4~~	~~1.07×10^3~~	~~13.0~~
~~I-135~~	~~0.475~~	~~1.46~~	~~1.24×10^5~~	~~3.14×10^4~~	~~11.5~~
~~Kr-83m~~	~~0.034~~	~~0.005~~	-	-	~~0.0062~~
~~Kr-85m~~	~~0.101~~	~~0.156~~	-	-	~~1.15~~
~~Kr-85~~	~~0.288~~	~~0.0021~~	-	-	~~0.016~~
~~Kr-87~~	~~1.014~~	~~0.856~~	-	-	~~7.87~~
~~Kr-88~~	~~0.307~~	~~2.00~~	-	-	~~10.4~~
~~Kr-89~~	~~1.00~~	~~2.22~~	-	-	~~6.70~~
~~Xe-131m~~	~~0.135~~	~~0.022~~	-	-	~~0.025~~
~~Xe-133m~~	~~0.191~~	~~0.024~~	-	-	~~0.25~~
~~Xe-133~~	~~0.126~~	~~0.073~~	-	-	~~0.23~~

TABLE 15B-1

(Sheet 2)

PHYSICAL DATA FOR DOSE CONVERSIONS

Delete and Replace with following Table 15B-1

~~Isotope~~	~~Ē Average Beta Energy (MeV/dis)[2][7]~~	~~Ē Average Gamma Energy (MeV/dis)[2][7]~~	~~DCF Thyroid Dose Conversion Factor (rem/Ci)[4]~~	~~DCF Thyroid Dose Conversion Factor Used for SLD (rem/Ci)[11]~~	~~K_γ (1) Gamma Dose Converson Factor Used for Control Room (rem-m^3/Ci-hr)~~
~~Xe-135m~~	~~0.090~~	~~0.440~~	-	-	~~3.12~~
~~Xe-135~~	~~0.302~~	~~0.248~~	-	-	~~1.89~~
~~Xe-137~~	~~1.37~~	~~0.192~~	-	-	~~1.11~~
~~Xe-138~~	~~0.565~~	~~0.932~~	-	-	~~5.74~~

~~BREATHING RATES [3]~~

~~Time Period (hours)~~	~~Breathing Rates (m^3/sec)~~
~~0-8~~	$\cancel{3.47 \times 10^{-4}}$ ~~(2)~~
~~8-24~~	$\cancel{1.75} \times \cancel{10^{-4}}$
~~24-720~~	$\cancel{2.32 \times 10^{-4}}$

~~(1) Calculated using equation 15B-5~~

~~(2) Used as Control Room breathing rate for the duration of the accident~~

Table 15B-1

Dose Conversion Factors

Isotope	Thyroid DCF(1) (Sv/Bq)	Whole Body DCF(2) (Sv-m^3/Bq-sec)	Beta-Skin DCF(3) (Sv-m^3/Bq-sec)
I-131	2.9E-7	1.82E-14	8.66E-15
I-132	1.7E-9	1.12E-13	3.02E-14
I-133	4.9E-8	2.94E-14	2.44E-14
I-134	2.9E-10	1.30E-13	3.87E-14
I-135	8.5E-9	7.98E-14	2.16E-14
Kr-83m	N/A	1.50E-18	0.00E+00
Kr-85m	N/A	7.48E-15	1.38E-14
Kr-85	N/A	1.19E-16	1.35E-14
Kr-87	N/A	4.12E-14	9.08E-14
Kr-88	N/A	1.02E-13	2.13E-14
Xe-131m	N/A	3.89E-16	4.07E-15
Xe-133m	N/A	1.37E-15	8.46E-15
Xe-133	N/A	1.56E-15	2.85E-15
Xe-135m	N/A	2.04E-14	5.85E-15
Xe-135	N/A	1.19E-14	1.75E-14
Xe-138	N/A	5.77E-14	3.99E-14

Breathing Rates (m^3/sec)

Time Period (hours)	Offsite	Control Room
0 – 8	3.5E-4	3.5E-4
8 – 24	1.8E-4	3.5E-4
24 - 720	2.3E-4	3.5E-4

(1) Thyroid DCFs from ICRP-30 (Reference 11)
(2) Whole-body DCFs from Federal Guidance Report 12 (Reference 13)
(3) Beta-skin DCFs from DOE/EH-0070 (Reference 14). Skin doses are modeled without the contribution from photon emissions as supported by the NRC in Reference 15.