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Lew W. Myers
Senior Vice President

May 12, 2000
L-00-008

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: **Beaver Valley Power Station, Unit No. 1 and No. 2**
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
License Amendment Request Nos. 280 and 151

FirstEnergy Nuclear Operating Company (FENOC) requests NRC review and approval of proposed changes to the Beaver Valley Power Station (BVPS) Unit 1 and Unit 2 Updated Final Safety Analysis Reports (UFSARs). The proposed revisions to the UFSARs modify information on design basis accident radiological doses as a result of the recent complete reevaluation of all BVPS dose calculations. An evaluation of all BVPS Unit 1 and Unit 2 dose calculations was completed which reviewed the input parameter values, the input assumptions, and the methodology used. This license amendment is being requested in accordance with 10 CFR 50.59(c) because it has been identified that this UFSAR change is an unreviewed safety question. NRC approval is needed to revise the UFSAR analyzed dose values for several design basis accidents and to utilize different calculation methodologies. The results for the BVPS Unit 1 and Unit 2 DBA dose calculations are provided in Attachment B, except for the BVPS Unit 2 Fuel Handling Accident, which was provided in License Amendment Request 2A-155 submitted via FENOC Letter L-00-048, dated May 1, 2000.

The proposed UFSAR changes are presented in Attachment A. The safety analysis (including the no significant hazards evaluation) for the change is presented in Attachment B.

These changes have been reviewed by the Beaver Valley review committees. The changes were determined to be safe and do not involve a significant hazard consideration as defined in 10 CFR 50.92 based on the attached safety analyses.

In accordance with Generic Letter 91-18, Rev. 1, prior NRC approval is not required for Unit 1 to continue power operation. This is justified by BVPS approved Bases for Continued Operation.

A053

If there are any questions concerning this matter, please contact Mr. Thomas S. Cosgrove, Manager, Licensing at 724-682-5203.

Sincerely,

Lew W. Myers

c: Mr. D. S. Collins, Project Manager
Mr. D. M. Kern, Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)
Ms. Mary E. O'Reilly (FirstEnergy Legal Department)

I, Lew W. Myers, being duly sworn, state that I am Senior Vice President of FirstEnergy Nuclear Operating Company (FENOC), that I am authorized to sign and file this submittal with the Nuclear Regulatory Commission on behalf of FENOC, and that the statements made and the matters set forth herein pertaining to FENOC are true and correct to the best of my knowledge and belief.

FirstEnergy Nuclear Operating Company

Lew W Myers

Lew W. Myers
Senior Vice President - FENOC

STATE OF PENNSYLVANIA

COUNTY OF BEAVER

Subscribed and sworn to me, a Notary Public, in and for the County and State above named, this 12 th day of May, 2000.

Tracey A Baczek

My Commission Expires:

Notarial Seal
Tracey A. Baczek, Notary Public
Shippingport Boro, Beaver County
My Commission Expires Aug. 16, 2001
Member, Pennsylvania Association of Notaries

ATTACHMENT A-1

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 280
UFSAR Update for Revised Radiation Dose Calculations

The following is a list of the affected pages:

Affected UFSAR Pages:	Revise Pages 2.2-13, 6.4-16, 6.4-17, 11.2-2, 11.2-6, 11.3-24, 14.1-45, 14.2-4, 14.2-5, 14.2-6, 14.2-7, 14.2-8, 14.2-9, 14.2-11, 14.2-12, 14.2-14, 14.2-17, 14.2-24, 14.2-25, 14.2-45, 14.2-53, 14.3-48 through 14.3-52, 14.3-58, 14.3-59, 14B-1 through 14B-4, 14B-8 through 14B-17
Affected UFSAR Tables:	Revise Tables 2.2-12, 11.3-7, 14.1-3, 14.2-4b, 14.2-6, 14.2-6a, 14.2-8, 14.2-9, 14.2-10, 14.2-12, 14.3-10, 14.3-13, 14.3-14a, 14B-1, 14B-5, 14B-6, 14B-11, 14B-15, 14B-16,
	Delete Tables 14.2-6b, 14.2-7, 14.3-14b, 14B-3, 14B-4, 14B-9, 14B-10, 14B-13, 14B-14
Affected UFSAR Figure:	Revise Figure 14B-1

TABLE 2.2-12

MAIN CONTROL ROOM X/Q VALUES

Release Point	0-8 Hours	8-24 Hours	1-4 Days	4-30 Days
Containment Building				
Top	2.73	1.28	0.917	0.557
Edge	4.33	2.04	1.46	0.884
Auxiliary Building	43.0	20.1	14.9	9.25
Main Steam Valve House	7.60	3.51	2.59	1.58
Service Building	6.25	3.04	2.36	1.57
Turbine Building	24.3	12.2	8.90	6.26
Gaseous Waste Storage Vault	5.11	2.15	1.65	1.14
Refueling Water Storage Tank	3.77	1.81	1.33	0.850

Notes:

1. These values were effective in January 1992 and are used for analyses documented after that date. Occupancy factors are not included in values.

2. All values are in X/Q (x 10^{-4} sec/m^3).

* These values were effective August 1999 and are used to determine control room operator dose from radioactivity in spaces below the control room pressure boundary floor. This source was not previously considered in radiological consequence analysis, and is applicable to the design basis LOCA. Only these values include a reduction factor for occupancy (after 24 hours).

Containment Edge to Service Building*	60.9	49.9	19.7	4.45
Containment Top to Service Building*	45.2	36.6	14.0	2.62

15. USNRC NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations", Pacific Northwest Laboratory (November 1982).

16. DLC Calculation ERS-SFL-83-015 r0, Accident Analysis X/Q Values (1983).

17. Halliburton NUS Environmental Corporation, Control Room X/Q Values for the Beaver Valley Power Station (1991).

18. J. V. Ramsdell, Atmospheric Diffusion for Control Room Habitability Assessments, NUREG/CR-5055 (1988).

19. DLC Calculation ERS-SFL-96-021 r0, RG 1.145 Short Term Accident X/Q Values for EAB and LPZ, Unit 1 and Unit 2, based on 1986-1995 Observations (1996).

20. SWEC Calculation 07859.1003 ENVR-MET 211-EN-ME-103, Normalized Concentrations (X/Qs) at the Unit 2 Aux. Bldg. NW Corner and Unit 1 Service Bldg. Roof for Releases from the Unit 1 and Unit 2 Containment Buildings.

Electrical interlocks which prevent the operator from tripping the spray pumps inadvertently or prematurely from the main control room are accomplished by use of a control switch trip action blocking contact. Upon the receipt of a containment isolation phase B signal, a contact from the pump motor starting timer produces this signal instantaneously.

The containment spray pumps suction and discharge valves are protected from inadvertent closure by a control room operator during pump operation by either a normally closed pump switchgear auxiliary contact placed in the closing circuit of the motor-operated valve, or by access covers placed over the benchboard control switch.

To deactivate any containment spray pump would require two operator actions: (1) manually reset the containment isolation phase B train signal associated with the equipment and (2) place the control switch for the pump in the stop position.

Above the operating floor (El. 767 ft 10 inches), the quench sprays cover a volume of approximately 431,102 ~~501,630~~ cu ft and the recirculation sprays cover a volume of approximately 529,625 ~~709,050~~ cu ft. The total free volume above the operating floor is 1,028,000 cu ft. Because of the forced circulation set up by the sprays, the entire volume above the operating floor is considered to be uniformly mixed and scrubbed by the sprays.

Below the operating floor, the quench sprays cover approximately 55,443 ~~407,500~~ cu ft and the recirculation sprays cover approximately 219,711 ~~656,000~~ cu ft. The total free volume below the operating floor is 768,300 cu ft.

All of the subcompartments, with the exception of the volume below the reactor vessel, the refueling cavity and the incore instrumentation passage, are well vented and scrubbed by the sprays. The following regions are not directly covered by sprays:

Region	Area (cubic ft)
Incore instrumentation passage	6,500
Volume below reactor vessel	2,000
Volume below refueling cavity	21,750

Note that this volume is a portion of the base mat floor. As such, it has no side walls, is not enclosed and is subject to good mixing with the main spray volume.

The spray patterns for the quench and recirculation sprays are depicted in Figure 6.4-8. These spray patterns are based on minimum engineered safeguards.

The degree of mixing between the regions covered by the quench and recirculation sprays can be estimated by the amount of overlap of coverage by the quench and recirculation sprays and from the forced circulation set up by the containment sprays. The volume above the operating floor which is covered by overlapping quench and recirculation sprays is ~~349,765~~ 253,878 cubic ft. The mixing from forced circulation induced by the sprays may be visualized by inspection of Figure 6.4-9, which shows a simplified diagram of the air flow in the containment set up by the sprays.

Consider that the containment volume is split into two concentric cylindrical regions. In the outer region, air is entrained by sprays and forced downward. In the inner portion, the air moves upward. This circulation pattern is augmented by rising air currents from hot components, such as the reactor pressure vessel, steam generators and pressurizer. The effectiveness of the sprays in mixing the containment atmosphere has been verified by BNWL[2] in which measurements, made at different points in the containment systems experiment vessel (similar to the BVPS-1 containment) for iodine and noble gases, showed that iodine was completely distributed throughout the containment, even at the high points in the vessel.

The design of the containments and containment spray systems for BVPS-1 and BVPS-2, are similar to those of the Surry Power Station, Units 1 and 2, for which it has been concluded by the Atomic Energy Commission[3] that the containment spray systems provide adequate mixing of the containment atmosphere in the post LOCA environment.

A measure of the mixing between subcompartments below the operating floor (such as the steam generator cubicles and the pressurizer cubicle), that are covered by the sprayed volume, may be determined by the natural circulation set up by the difference in air/steam densities between the subcompartment volumes and the containment volume covered by the sprays. The flow due to "stack effect" can be computed from the following equation:[4]

$$Q = 7.2\,A\sqrt{h(t_i - t_o)} \qquad (6.4\text{-}1)$$

where: Q = air flow, cubic feet per minute

A = free area of inlets or outlets (assumed equal; if different, the smaller of the two is used), sq ft

h = height from inlets to outlets, ft

t_i = average temperature of inside air, °F

t_o = temperature of outside air, °F

7.2 = constant of proportionality, including a value of 50 percent for effectiveness of openings release. It also provides ~~a 30 day~~ holdup of these gases when refueling cold shutdown degassing is required.

System design provides that all the gaseous effluent from the degasifiers is directed to the gaseous waste charcoal delay subsystem for decay of most radioactive isotopes prior to compressing and discharged through the process vent. Gaseous effluent may be recycled to the volume control tank but this is not normally performed. Provision is made to direct compressed waste gas to decay tanks for control of the equilibrium activity level of the coolant fission product gas inventory and subsequent release to the atmosphere. The discharge to the atmosphere is handled by diluting the flow controlled release of waste gas with a large volume of air, discharging the air through charcoal and HEPA filters to the top of the cooling tower, approximately 500 ft above the ground. This same discharge system is also designed to handle gaseous effluent from the main condenser air ejector vents, purge and vent from the oxygen analyzers, decay tank radiation monitor aerated vents of the vent and drain system, and the gaseous discharge from the containment vacuum system. The system also handles special conditions when gases from the containment purge are vented to the top of the cooling tower.

11.2.3.2 Description

Radioactive gases enter the gaseous waste disposal system from the degasifier vent chiller of the boron recovery system and are directed by the system pressure gradient to the gaseous waste charcoal delay subsystem upstream of the overhead gas compressor. The gas is chilled to approximately 55°F to condense most of the water vapor. The compressors operate automatically in response to the suction pressure thus maintaining the degasifier's overhead components at a pressure between established limits. Radioactive gases from the degasifier vent chillers contain primarily hydrogen, water vapor and a small amount of nitrogen in the gaseous effluent. The gas is then processed through the gaseous waste charcoal delay subsystem which holds up the xenon for about 3~~0~~9 days, and the krypton about two days. Essentially all of the iodine is absorbed by the charcoal. This holdup assumes continuous stripping of 60 gpm of primary coolant letdown with a hydrogen concentration of 35 cc/kg.

One of the two overhead gas compressors directs the radioactive gas stream to a gas surge tank at a system pressure of about 65 psig. The gas flow is reduced in pressure and, as long as it meets Westinghouse specifications, can be returned to the volume control tank in the chemical and volume control system (Section 9.1). However, this method of gas reclamation is not normally used. A quantity of gas can be discharged from the surge tank to one of the three decay tanks at Unit 1 or ~~any of~~ the seven storage tanks at Unit 2 for eventual release ~~to the~~ atmosphere via the process vent on top of the cooling tower.

The gas waste decay and surge tanks are designed in compliance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Class C, with 100 percent radiography. The gas waste charcoal beds are designed in compliance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Class 3, with 100% radiography.

Process piping is designed to meet American National Standards Institute (ANSI) B31.1, Power Piping, Section 1 requirements.

The gaseous waste disposal system from the boron recovery system to an isolation valve downstream of the decay tanks is considered Seismic for design purposes. Component design data for the gaseous waste disposal system is listed in Table 11.2-1.

11.2.3.3 Evaluation

Fission product gas inventory in the reactor coolant is a function of reactor coolant system fission gas input and output. Fission gas input is determined by the reactor power level and the amount of fuel failure. Fission gas output from the reactor coolant system is determined by the amount of gas sent to the BVPS-1 decay tanks and not recycled.

Reactor coolant is letdown to adjust its chemistry and its radiation level and to provide water for seal injection.

The annual average waste gas bleed rate needed to maintain the krypton-85 inventory at acceptable levels within the reactor coolant during steady state full power operation ~~is a small~~ fraction of a cfm.

Operation of the gaseous waste disposal system, using charcoal delay beds selectively delaying xenon-133 for approximately 39 ~~30~~ days, results in an annual average atmospheric emission rate which is a very small fraction of the 10 CFR 20 limit. A tabulation ~~of nuclides~~ and their maximum calculated annual release rate is given in Appendix 11A. The doses from the gaseous nuclides released are presented in Appendix 11B.

Thirty nine days of holdup allows time for the short lived fission gases to decay to the point where krypton-85 is the controlling isotope.

In the event of modes of fuel failure which result in abnormal concentrations of fission products in the reactor coolant, adequate storage space in the decay tanks is supplied. The tanks will be allowed to go to a higher holding pressure and will thus be able to accommodate a larger volume of gas. The higher pressure will not exceed the design pressure of the system.

When charcoal delay beds are installed for the air ejectors the activity discharged from the air ejectors is assumed to be the result of one percent failed fuel and a continuous 50 lb per hr in-leakage from all steam generators. Periodic increases in the leakage rate up to 500 lb per hr will also be considered in the design. It is

D. Waste Gas System Failure (Table 14.2-8)

E. Steam Generator Tube Rupture Accident (Table 14.2-9)

F. Main Steam Line Break Accident (Table 14.2-10)

G. Rod Ejection Accident (Table 14.2-12)

H. Small Line Break Accident (Table 14.3-10)

I. Loss of Coolant Accident (Table 14.3-14b) a

The analyses indicate that the common BV1-BV2 Control Room is habitable for all design basis accidents at Beaver Valley Power Station Unit 1. Postulated doses are tabulated in Table 11.3-7.

The integrated whole body dose from the worst case radiological accident was calculated to be below the criterion dose of 5 rem. Thus, the main control room walls which must be a minimum of 24 inches thick for tornado missile protection, provide more than adequate shielding from radiation.

Special consideration has been given to the design of penetrations and structural details of the main control room so as to establish an acceptable condition of leak tightness.

The air-conditioning systems are installed within the spaces served and designed to provide uninterrupted service under accident conditions. Upon a containment isolation phase B, high chlorine or high radiation signal, the normal replenishment air and exhaust systems are isolated automatically from the main control room by tight closures in the ductwork. Breathing-quality compressed air is supplied from high-high pressure storage bottles to maintain a small positive outflow from the main control room for a period exceeding the containment leakage period. This outflow can be verified by means of a pressure gage which reads the inside and outside pressure difference in inches of water. The main control and relay rooms are also provided with an emergency ventilation system fitted with particulate and impregnated charcoal filters to introduce cleaned outside air into the protected spaces upon depletion of the high-high pressure air. This system can be used indefinitely to maintain the area pressure above atmospheric to ensure exfiltration.

The radiation levels in the main control and relay rooms are measured by gamma monitors to verify safe operating conditions.

TABLE 11.3-7

POSTULATED CONTROL ROOM ACCIDENT DOSE, REM[5][6]

(Design Basis Accidents at Unit 1)

Accident	Thyroid	Gamma	Beta	Notes
Main Steam Line Break				
Co-incident Spike	26.0	2.9E-3	3.7E-2	2
Pre-incident Spike	29.0	1.5E-3	2.0E-2	2
Small Line Break	27.0	3.0E-3	2.6E-2	4
Steam Generator Tube Rupture				
Co-incident Spike	3.13	8.12E-4	2.22E-2	4
Pre-incident Spike	8.65	9.30E-4	2.34E-2	4
Rod Ejection Accident	12.0	1.1E-3	8.9E-3	4
Fuel Handling Accident	4.3	1.4E-2	8.3E-1	4
Locked Rotor Accident	9.69	7.35E-2	1.16	1
Loss of Auxiliary AC Power	8.0E-1	7.1E-5	2.7E-3	4
Waste Gas System Rupture				
Line Break	--	< 1.0E-2	< 1.0	4
Tank Rupture	--	7.1E-4	1.3	4
DBA LOCA	14.3	1.7E-1	4.0E-1	3

See attached

NOTES:

1. Control Room isolation by area radiation monitor signal based on a setpoint with a safety limit dose rate of 1 mrem/hr gamma in the Control Room.
2. Isolation by manual operator action at T=30 minutes post-accident. In support of Alternate Repair Criteria for steam generators (ref. USNRC GL 95-05) the MSLB thyroid doses were maximized within applicable limits in order to establish the maximum allowable accident-induced leakage against which tube leakage projections, based on voltage indication, are compared. Current values are based on 8.0 gpm primary-to-secondary leakage (0-2 hour Exclusion Area Boundary thyroid dose limiting). See Section 14.2.5.1.3.
3. Control Isolation actuated by CIB signal.
4. No action required.
5. References: ERS-SFL-93-005 r0, ERS-SFL-92-033 r1, 12241/14110.39-UR(B)-456, 14110.39-UR(B)-457 r0, ERS-SFL-89-021 r1, ERS-SFL-95-008 r2.
6. Listed dose values represent the limiting bounding value.

INSERT

TABLE 11.3-7

POSTULATED CONTROL ROOM ACCIDENT DOSE, REM(5)(6)

(Design Basis Accidents at Unit 1)

Accident	CDE, Thyroid	ED~~F~~E	Skin DE	Notes
Main Steam Line Break				
Co-incident Spike	2.9E+01	<2E-01	<1E+00	2
Pre-incident Spike	1.4E+01	<2E-01	<1E+00	2
Small Line Break	2.0E+01	<2E-01	<1E+00	4
Steam Generator Tube Rupture				
Co-incident Spike	3.1E+00	<2E-01	<1E+00	4
Pre-incident Spike	1.9E+00	<2E-01	<1E+00	4
Rod Ejection Accident	7.7E+00	<2E-01	<1E+00	4
Fuel Handling Accident	~~3.2~~ 6.3E+00	<2E-01	<1E+00	4
Locked Rotor Accident	3.1E+00	<2E-01	<1E+00	1
Loss of Auxiliary AC Power	<1E+00	<2E-01	<1E+00	4
Waste Gas System Rupture				
Line Break	--	<2E-01	3.9E+00	4
Tank Rupture	--	<2E-01	<1E+00	4
DBA LOCA	5.5E+00	7.1E-01	<1E+00	3

NOTES:

1. Control Room isolation by area radiation monitor signal based on a setpoint with a safety limit dose rate of 1 mrem/hr gamma in the Control Room.
2. Isolation by manual operator action at T=30 minutes post-accident. In support of Alternate Repair Criteria for steam generators (ref. USNRC GL 95-05) the MSLB thyroid doses were maximized within applicable limits in order to establish the maximum allowable accident-included leakage against which tube leakage projections, based on voltage indication, are compared. Current values are based on ~~8~~3.0 gpm accident induced ~~primary-to-~~secondary leakage ~~(0-2 hour Exclusion Area Boundary thyroid dose limiting)~~. See Section 14.2.5.1.3.
3. Control Isolation actuated by CIB signal.
4. No action required.
5. References: ERS-~~SFL-93-005 r0~~ JTL-99-015, ERS-JTL-99-010 ERS-SFL-92-033 ~~r1~~, ERS-JTL-99-014 ERS-JTL-99-009 ~~12241/14110.39-UR(B)-456~~, ~~14110.39-UR(B)-457 r0~~ 12241/11700-UR(B)-480, ERS-SFR-89-021 ~~r1~~, ERS-SFL-95-008 ~~r2~~ ERS-AJL-? 012
6. Listed dose values represent the limiting bounding value.

References to Section 14.1

1. D. B. Fairbrother, H. G. Hargrove, "WIT-6 Reactor Transient Analysis Computer Program Description," WCAP-7980, Westinghouse Electric Corporation (November 1972).

2. H. G. Hargrove, "FACTRAN - A Fortran IV Code for Thermal Transients in UO_2 Fuel Rod," WCAP-7908-A, December 1989.

3. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984.

4. S. Altomare, R. F. Barry, "The TURTLE 24.0 Diffusion Depletion Code," WCAP-7758, Westinghouse Electric Corporation (September 1971).

5. F. M. Bordelon, "Calculation of Flow Coastdown After Loss of Reactor Coolant Pump (PHOENIX Code)," WCAP-7969, Westinghouse Electric Corporation (September 1972).

6. M. A. Mangan, "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Westinghouse Electric Corporation (October 1971).

7. J. M. Geets, R. Salvatori, "Long Term Transient Analysis Program for PWR's (BLKOUT Code)," WCAP-7898, Westinghouse Electric Corporation (June 1972).

8. J. M. Geets, "MARVEL - A Digital Computer Code for Transient Analysis of a Multiloop PWR System," WCAP-7909, Westinghouse Electric Corporation (June 1972).

9. J. Shefcheck, "Application of the THINC Program to PWR Design," WCAP-7359-L, (August 1969), Westinghouse Electric Corporation (Proprietary), and WCAP-7838, Westinghouse Electric Corporation (January 1972).

10. D. H. Risher, Jr., R. F. Barry, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979, Westinghouse Electric Corporation (November 1972).

11. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

12. ~~Combined BV1-BV2 Control Room Habitability Due to Design Basis Accidents (except LOCA) at BV1, Calculation 12241/14110.39 UR(B)-456, 1987.~~
DLC Calculation ERS-JTL-99-010, Safety Analysis of the Radiological Consequences of a Control Rod Ejection DBA at BVPS Unit 1, Control Room, EAB and LPZ Doses.

13. DLC Calculation ERS-AJL-99-012, Safety Analysis of the Radiological Consequences of a Loss of AC Power Design Basis Accident at Unit 1, Common Control Room, EAB and LPZ Dose.

Table 14.2-6. Gap inventories of fission products were determined using the release fractions identified in Regulatory Guide 1.25 except for Iodine-131, for which the release fraction is increased 20 percent in accordance with NUREG/CR-5009.

14.2.1.3 Fission Product Inventories

Core specific inventories (Curies per metric ton of uranium) of fission products were estimated with the ORIGEN~~-2~~ code.

The results of the ORIGEN calculations for isotopes that contribute to the thyroid and whole-body doses are given in Table 14.2-6A~~,~~. ~~while Table 14.2-6B lists pertinent data for the isotopes of interest.~~ Data and assumptions used in the dose calculations are given in Table 14.2-6.

14.2.1.4 Offsite Exposure from Accident During Refueling

During refueling, the fuel building will be maintained at a minimum negative pressure of 0.125 inches W.G., with a maximum exhaust flow rate of 3,000 cfm. ~~The maximum fuel building exhaust flow rate of 3,000 cfm was~~ A puff release was conservatively assumed in the fuel handling accident of the Control Room Dose Analysis.

During refueling, the containment is maintained by the supplementary leak collection and release system at a negative pressure of 0.125 inches W.G.

Exhaust flow from the fuel building will discharge through the main supplementary leak collection and release system filter bank to the SLCRS Vent.

The doses at the Beaver Valley EAB from the specified fuel handling accident are tabulated below. The doses are based on the release of all gaseous fission product activity in the gaps of 298 fuel rods in highest-power assemblies.

Thyroid ~~dose, rad~~ CDE, rem	=	~~14.60~~ 2.5E+01
~~Whole-body dose, rem~~		
~~Beta dose, D_β~~	~~=~~	~~2.96~~
~~Gamma dose, D_γ~~	~~=~~	~~0.95~~
Whole-body EDE ~~total~~, rem	=	~~3.91~~ 5.8E-01

These potential doses are well within the exposure guideline values of 10 CFR 100, paragraph 11. As defined in Standard Review Plan 15.7.4, <u>Radiological Consequences of Fuel Handling Accidents</u>, "well within" means 25% or less of the 10 CFR 100 guidelines, or values of 75 ~~rad~~ rem for thyroid doses and 6.25 rem for whole-body doses.

The doses for the Beaver Valley Common Control Room were also analyzed for the specified fuel handling accident. The doses for the duration of the accident are provided in Table 11.3-7.

~~Thyroid dose, rem = 4.33~~
~~Whole body dose, rem~~
~~Beta dose, D_β = 0.83~~
~~Gamma dose, D_γ = 0.014~~

The doses are within the criteria of 10 CFR 50, Appendix A, General Design Criterion 19 and are acceptable. Table 14.2-6 tabulates significant analysis parameters. ~~Table 11.3-7 tabulates results.~~

In the event of a fuel handling accident within the containment structure, a high-high radiation level in the purge duct discharge will trip the purge line isolation valves and isolate the containment from the supplementary leak collection and release system. During refueling in the containment, the containment isolation valves of the containment purge supply and exhaust systems are not required to close since the containment purge exhaust duct is lined up to the seismically supported leak collection system and filter train. ~~No credit is taken for containment isolation in the safety analysis for a fuel handling accident in the containment.~~ Since the containment has been maintained at subatmospheric pressure, there will be no driving force for discharge of gas from the containment, which will result in no release of activity to the environment. Once the nature of the activity release within the containment has been ascertained, a containment ventilation operation may be manually initiated, the containment iodine removal filter may be placed into operation, or other corrective action taken. ~~For analysis purposes no credit is taken for containment isolation.~~ For these conditions, the site dose would be the same as that obtained for a fuel handling accident in the fuel building.

14.2.2 Accidental Release of Waste Liquid

14.2.2.1 Identification of Causes and Accident Description

Accidents in the auxiliary system which could result in the release of waste liquid may involve the rupture or leaking of various components.

14.2.2.2 Analysis of Effects and Consequences

Liquid processing components are located within the auxiliary building, and any liquid leakage or release from the components is locally collected and transferred to sumps for subsequent pumping into the liquid waste disposal system.

Curbs and floor drains to the sump are employed to minimize the effect of leakage and spills. Outboard seal leakage from the charging pumps (Section 9.1) is contained in this manner. The ventilation system collects any gaseous radioactivity and discharges it to the monitored ventilation vent as discussed in Section 9.13.

The primary coolant recovery tanks are located in the yard area within building cubicles of sufficient capacity to retain the total liquid volume resulting from rupture of either primary coolant recovery tank without overflowing to areas outside the cubicles.

Piping running between the auxiliary building and the reactor containment, the auxiliary and fuel buildings, and the fuel building and the tanks in the yard area, are run in concrete trenches. Liquid released from such piping is collected and transferred to sumps and pumped into the liquid waste disposal system.

The liquid waste inventories in the various tanks are based on the mode of operation during any particular time duration. In order to determine the liquid waste inventory for the various process tanks the following programs are used:

1. Program ACTIVITY[15] calculates the primary coolant equilibrium activity for a variety of input parameters. The inputs include data such as thermal power level, volume of primary coolant, fraction of failed fuel, purification flow rate, and primary coolant letdown rate. The library of the program contains factors for each fission product nuclide such as decay constant, escape rate coefficient, purification factor, fission yield, and absorption cross section.

2. Program IONEXCHANGER[16] calculates the total accumulation of radioactive nuclides in a tank. The input data may include the feed rate, nuclide activity concentration, bleed rate, container volume, and duration of feed.

14.2.2.3 Conclusions

Administrative controls and batch handling of all waste liquids ensures positive control of all processing. System liquid level indicators, radiation monitors, flow control instrumentation, and piping siphon break prevent an inadvertent radioactivity release to the environment.

When accidental spillage of waste liquids does occur, it is contained within the station and does not result in any significant release of activity.

14.2.3 Accidental Release of Waste Gases

The concentration of radioactive waste gases in the reactor coolant system and auxiliary systems is a function of the rate of fission gas release to the coolant from defective fuel and the rate of removal via the auxiliary systems. ~~The areas which retain significant concentrations of radioactive gases are the volume control tank (Section 9.1) and the gas surge tank (Section 11.2).~~ Inser

The waste gas incidents consider the rupture of the volume control tank, or the gas surge tank with the subsequent release of its radioactive gas inventories to the environment.

14.2.3.1 Method of Analysis

Basis of Fission Product Inventory

Reactor coolant fission product concentrations are based on the assumption that 1.0 percent of the fuel rods in the core develop pinhole defects, resulting in the diffusion of fission product isotopes into the coolant. The rod fission product inventories are those produced at 100 percent power at a maximum calculated core thermal rating of 2,766 MWt.

The fission product gases which are removed from the reactor coolant are those which are derived from a reactor coolant letdown rate of 60 gpm. An average of 0.30 scfm of fission gases removed in the degasifier are directed to the gas surge tank prior to being sent to either the volume control tank or the waste gas decay tank. The greatest expected buildup of noble gas fission product isotopes in the volume control tank vapor phase (Table 14B-9) is approximately 6,350 Ci of Xe-133 equivalent and 0.18 Ci of I-131 equivalent. The greatest expected buildup of radioactive fission isotopes in the gas surge tank is approximately 1,780 Ci of Kr-85 equivalent. The charcoal delay beds remove essentially all iodines. Therefore a thyroid dose is not considered in the analysis of a surge tank rupture.

The gaseous waste inventories in the holdup tanks are derived using the computer programs described in Section 14.2.2.2 for the waste liquid accident.

The evaluation of the whole body dose from the release of noble gases is based upon an instantaneous formation, at ground level, of a semi-infinite cloud. The source strength within this cloud is assumed to be uniform and is based on the cloud centerline concentration at the site boundary (2,000 ft).

The following equation is used to calculate the external dose:

$$\text{Dose} = (X/Q) * Ci * CF \qquad (14.2\text{-}1)$$

where:

X/Q = ground level centerline dispersion factor at site boundary, sec per m^3

Ci = curies of a particular noble gas isotope

CF = conversion factor for each noble gas isotope, rem/(Ci-sec/m^3)

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The conversion factor, CF, is derived from the published maximum permissible concentration values in 10CFR20, Table II. For example:

(14.2-2)

CF Kr-85 = (0.5 rem/yr)/[(3*10**-7 μC/cc)*(3.154*10**+7 sec/yr)]

CF Kr-85 = 5.28 * 10**-2 rem/(μC-sec/cc)

CF Kr-85 = 5.28 * 10**-2 rem/(C-sec/m^3)

The internal (thyroid) dose from inhalation of the fission product iodine contained in the released cloud is computed using TID 14844 methods.

Meteorology

Dose calculations at the site boundary (2,000 ft) are based upon Pasquill Type "F" meteorology with a wind speed of 0.84 meter per second. A dispersion coefficient, which includes a shape factor of 0.5 to account for building wake effects is used to calculate doses at the site boundary assuming a ground level release.

The discharge of activity from the SLCRS Vent of 150 ft could result in a point source release contacting the ground at the nearest point, offsite, at the same elevation as the SLCRS Vent. This nearest point is 2,500 feet from the release point.

14.2.3.2 Results

Volume Control Tank Rupture Analysis

In the unlikely event of a sudden rupture of the volume control tank, it is assumed that the noble gases dissolved in the coolant are released in addition to the fission product inventory contained in the vapor phase (Table 14B-9). The radioactive concentration accumulated in the vapor phase is assumed to consist of the equilibrium concentration of fission product gases and to have total activity values as described in Section 14.2.3.1. The site boundary (2,000 feet) dose for a ground level release of the gaseous inventory is 0.28 rem to the whole body and negligible dose to the thyroid. The dose at the 2,500 ft offsite location based on an elevated release is approximately 0.39 rem to the whole body and negligible dose to the thyroid.

Gas Surge Tank Rupture Analysis

The rupture of the gas surge tank would suddenly release to the environment the stored concentration of Kr-85 equivalent fission products inventory. The expected fission product concentrations are described in Section 14.2.3.1. It is assumed that the hypothetical tank rupture takes place when the gas surge tank has the greatest inventory of gases (Table 14B-10). The site boundary (2,000 ft) dose for a ground level release of the fission product

Delete

~~inventory is approximately 0.07 rem whole body. The dose at the 2,500 feet offsite location based on an elevated release is approximately 0.10 rem to the whole body.~~

~~Control Room Habitability~~

~~For the purpose of demonstrating habitability of the common control room, analyses were performed for two release scenarios: (1) rupture of Waste Gas Storage Tank, (2) rupture of gaseous waste line upstream of charcoal delay beds. Table 14.2-8 tabulates significant analysis parameters. Table 11.3-7 tabulates analysis results.~~

~~14.2.3.3 Conclusions~~

~~A rupture of either the volume control tank or the gas surge tank will produce site boundary dose values much less than the dose limits of 300 rem to the thyroid and 25 rem whole body as suggested in 10 CFR 100.~~ Delete

14.2.4 Steam Generator Tube Rupture

14.2.4.1 Accident Description

The accident examined is the complete severance of a single steam generator tube. 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 reactor coolant system. 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.

Because the steam generator tube material is Inconel 600 and is a highly ductile material, it is considered that the assumption of a complete severance is conservative. The more probable mode of tube failure would be one or more minor leaks. Activity in the steam and power conversion system is subject to continual surveillance, the maximum value of this activity is given in the Technical Specifications.

The operator is expected to determine that a steam generator tube rupture has occurred, and to identify and isolate the faulty steam generator on a restricted time scale in order to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the faulty unit. The recovery procedure can be carried out on a time scale which ensures that break flow to the secondary system is terminated before water level in the affected steam generator rises into the main steam pipe. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily.

Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the

Section 14.2.3 Insert

The radiological consequence (dose) analysis for the accidental release of waste gases considers two accident scenarios. The first is a rupture of a gas decay tank with the release of it's contents directly to the environment. The second is a gaseous waste system pipe rupture. The analyses were performed using conservative assumptions based on NUREG-0800, Branch Technical Position ETSB 11-5. These accidents, as described in detail below, provide the bounding conditions and resultant radiological consequences for a waste gas release.

14.2.3.1 Method of Analysis

Reactor coolant noble gas concentrations (taken from Table 14B-6) are based on the assumption that 1.0 percent of the fuel rods in the core develop pinhole defects, resulting in the diffusion of fission product isotopes into the coolant. The rod fission product inventories are those produced at 102 percent power at a maximum core thermal power of 2705 MWt.

For the decay tank rupture accident, Xe will not be present because of the relatively long holdup in the charcoal delay bed. The Kr activity that is accumulated in a tank is calculated by assuming activity transfer from the RCS at the maximum rate of 120 gpm, holdup in the charcoal delay bed then transfer to the tank for the minimum time period required to fill the tank. This methodology minimizes activity reduction by the radioactive decay process. Activity release to the environment following a tank rupture is assumed to be a puff release from the decay tank vault directly to the environment.

For the line rupture accident, the release consists of two sources 1) 100 percent of the Xe and Kr contained in reactor coolant letdown liquid released at the maximum letdown flow rate of 120 gpm plus, 2) a fraction of the Xe and Kr that would be retained on the waste gas system charcoal delay bed during 24 hours of power operation plus 26.5 hours after reactor shutdown (the time to degas the reactor coolant system) while operating with letdown at the maximum 120 gpm. Activity release following a line rupture is assumed to be a puff release from the charcoal delay bed, and a 1 hour release of the reactor coolant letdown gases, both into the Auxiliary Building. The activity is assumed to be instantaneously released to the environment as it enters the Auxiliary Building.

Between the two accident scenarios described above, the bounding waste gas system accident is considered for the purpose of radiological consequence analysis.

Table 14.2-8 provides the significant analysis parameters for each of the accident scenarios, including relevant assumptions for calculating dose to the control room operators and at the exclusion area boundary (EAB) and the low population zone (LPZ).

14.2.3.2 Results

Thyroid committed dose equivalent (CDE), whole body dose (EDE) and skin beta dose equivalent (DE) for the control room operators are provided in Table 11.3-7.

Section 14.2.3 Insert (continued)

Doses calculated for offsite are provided below:

	Decay Tank Rupture		Line Rupture	
	0-2 h EAB (rem)	0-30 d LPZ (rem)	0-2 h EAB (rem)	0-30 d LPZ (rem)
Thyroid CDE	N/A	N/A	N/A	N/A
Whole Body EDE	<2E-01	<2E-01	2.2E-01	<2E-01

14.2.3.3 Conclusions

The maximum control room operator doses that may result due to a failure of the gaseous waste system are less than the 10 CFR Part 50 Appendix A, GDC 19 limit of 5 rem whole body, or its equivalent to any part of the body.

The maximum EAB and LPZ doses that may result due to a failure of the gaseous waste system are a small fraction of the 10 CFR Part 100.11 limits of 25 rem whole body and 300 rem thyroid, and are within the 500 mrem whole body dose specified in NUREG-0800, ETSB 11-5.

14.2.4.2 Analysis of Effects and Consequences

14.2.4.2.1 Method of Analysis

In estimating the mass transfer from the Reactor Coolant System through the broken tube, the following assumptions are made:

1. Reactor trip occurs automatically as a result of low pressurizer pressure.

2. Following the initiation of the SIS, two centrifugal charging pumps are actuated and continue to deliver flow for 30 minutes. The emergency instructions for a tube rupture accident indicate that the operator should switch off all but one pump when he has identified the accident and when a minimum on scale water level returns to the pressurizer.

3. After reactor trip, the break flow reaches equilibrium at the point where incoming safety injection flow is balanced by outgoing break flow as shown in Figure 14.2-3. The resultant break flow persists from plant trip until 30 minutes after the accident.

4. The steam generators are controlled at the safety valve setting rather than the power operated relief valve setting.

5. The operator identifies the accident type and terminates break flow to the faulty steam generator within 30 minutes of accident initiation.

The above assumptions lead to a conservative estimate of ~~132,000~~ 137,400 lb for the total amount of reactor coolant transferred to the faulty steam generator as a result of a tube rupture accident. This mass release is consistent with a 2 hour release duration the time conservatively used in the radiological analysis

The dose values calculated for the steam generator tube rupture is based on ~~132,000~~ 137,400 lb of primary coolant activity released to the environment. Equilibrium primary coolant activity values are derived by computer program "ACTIVITY", adjusted to maximum concentrations limited by facility Technical Specifications, and are tabulated in Table 14B-~~6~~ 15. Additional significant analysis parameters are provided in Table 14.2-9.

~~The Steam Generator Tube Rupture Accident has been analyzed for the 15 x 15 fuel assembly resulting in offsite doses well under the limits of 10CFR100. This analysis is conservative for the 17 x 17 fuel design since the diffusion of radioactive isotopes in the fuel pellet is temperature dependent. Since the 17 x 17 fuel operates at a lower average temperature, the release of fission products from the fuel pellet into the clad gap and subsequently into the reactor coolant is decreased. Thus less radiation is transferred from the primary to secondary system and eventually to the environment.~~

DELETE

14.2.4.2.2 Environmental Consequences of Tube Rupture

In the unlikely event of a steam generator tube rupture, the non-volatile fission and corrosion products would largely concentrate in the secondary side water of the faulty steam generator. An insignificant fraction of this activity is assumed to be carried over by the moisture droplets entrained in the steam flow exiting the faulty steam generator (the moisture is estimated at the design value of 0.25 percent of the steam mass flow). The volatile fission products are continuously released without buildup via the condenser air ejector.

With Offsite Power

If offsite power is available, all volatile activity is diverted into the containment upon a high-high radiation signal from the condenser air ejector vent monitor.

Without Offsite Power (Atmospheric Relief Only)

~~The activity release will only be through the faulted steam generator and limited by the concentration in the reactor coolant. The activity in the coolant is assumed to result from 1 percent defective fuel. For purposes of analysis, if it is conservatively assumed that it takes 30 minutes to isolate the leak and further that the charging pumps maintain the reactor coolant system pressure during this time period, then approximately 132,000 lb of coolant will be discharged to the faulted steam generator secondary side. This would result in an activity carryover of less than 25 percent of the total coolant activity in the reactor coolant system.~~

~~Because the iodine is soluble in water, considerable separation will occur. Assuming equilibrium is reached between the liquid and vapor iodine concentrations, the effective reduction factor is 0.01. Activities released to the secondary side of the steam generator, on the basis of equilibrium core conditions and 132,000 lb of reactor coolant water, are shown in Table 14.2-7.~~

~~Using the effective reduction factor given above, 0.01 for Iodines, and a X/Q value of 7.8 x 10^{-4} sec/m^3, the activity released through the safety valves results in a whole body dose of approximately 0.3 rem and a thyroid dose of approximately 0.9 rem at the site boundary. Thus, these doses are well under the limits of 10 CFR 100, even if it is assumed that the operator delays in taking action when warned by alarms and instruments.~~

~~Control Room Habitability~~

~~For the purpose of demonstrating habitability of the common control room, an analysis of the radiation doses in the common control room from a steam generator tube rupture was performed. Table 14.2-9 tabulates significant analysis parameters. Table 11.3-7 tabulates analysis results.~~

Insert

Section 14.2.4.2.2 Insert

The activity release will be through the faulted steam generator and, to a lesser extent, the two intact steam generators. Initial radioactivity concentrations in the primary coolant, secondary coolant and secondary steam are assumed to be at the upper limit specified in the facility Technical Specifications. Mass releases from these three activity compartments, and other significant parameter values used in the analysis are provided in Table 14.2-9. Although it is expected that the ruptured steam generator will be isolated within 30 minutes, the rupture release is assumed to occur over a duration of 0-2 hours. For the pre-accident iodine spike case, this assumption does not affect the rupture mass release and does not cause a significant change in the radioactivity released. However, for the co-incident iodine spike case, influence of the iodine spike is maximized, and the total accident dose will be conservatively bounded.

The accident is analyzed twice, first assuming that a pre-accident iodine spike has occurred and caused primary coolant activity to increase to the instantaneous Technical Specification limit of 21 μCi/g dose equivalent iodine 131. A second analysis was performed assuming that an iodine spike occurs co-incident with the accident, releasing iodine to the coolant at a rate 500 times the equilibrium rate that will maintain the coolant at the Technical Specification 48 hour limit of 0.35 μCi/g dose equivalent iodine 131. The co-incident spike duration is assumed to be 4 hours.

Control room operator dose is provided in Table 11.3-7. The 0-2 hour site boundary doses are 1.3E+00 rem thyroid CDE and <2E-01 rem whole body (EDE) for the pre-accident iodine spike case, and 1.4E+00 rem thyroid CDE and <2E-01 rem whole body (EDE) for the co-incident spike case.

The maximum control room operator doses that may result due to a steam generator tube rupture are less than the 10 CFR Part 50 Appendix A, GDC 19 limit of 5 rem whole body, or its equivalent to any part of the body.

The maximum site boundary doses that may result due to a steam generator tube rupture are a small fraction of the 10 CFR Part 100.11 limits of 25 rem whole body and 300 rem thyroid.

power unavailable. Although the recovery method is the same with or without offsite power available, the equipment used may be different.

Since neither the steam dump valves or the condenser would be available with offsite power unavailable, the RCS is cooled using the atmospheric steam dump valves on the intact steam generators. RCPs trip on a loss of offsite power and a gradual transition to natural circulation flow ensues. With RCPs stopped, normal pressurizer spray would not be available. Consequently, RCS pressure must be controlled using pressurizer PORVs or auxiliary spray.

The objectives of the above recovery procedure are to limit the release of radioactive effluents from the ruptured steam generators, stop primary-to-secondary leakage to prevent steam generator overfill, and restore reactor coolant inventory to ensure adequate core cooling and plant pressure control.

There is ample time available to carry out the above recovery procedure such that isolation of the affected steam generator is established before water level rises into the main steam pipes. The available time scale is improved by the termination of auxiliary feedwater flow to the faulty steam generator and the regulation of pressurizer water level with only one charging pump operating. Normal operator vigilance, therefore, assures that excessive water level will not be attained. In addition, the main steam piping and supports are capable of withstanding a solid water condition due to overfilling the steam generator as described in Section 10.3.1.1.

14.2.4.2.4 Results

Figure 14.2-3 illustrates the flow rate that ~~would~~ result through the ruptured steam generator tube. The ~~previous~~ assumptions lead to a conservative upper limit estimate of ~~132,000~~ 137,400 lb for the total amount of reactor coolant transferred to the secondary side of the faulty steam generator as a result of a tube rupture accident.

14.2.4.3 Conclusions

A steam generator tube rupture will cause no subsequent damage to the Reactor Coolant System or the reactor core. An orderly recovery from the accident can be completed even assuming simultaneous loss of offsite power. Offsite dose consequences may be calculated based on a conservative estimate of ~~132,000~~ 137,400 lb of reactor coolant transferred to the secondary side of the faulty steam generator following the accident.

"ION EXCHANGER"

The steam line break releases activity by two modes. The activity released is the steam generator equilibrium activity and that which leaks from the primary system during depressurization and cooldown. The steam generator equilibrium activity is derived from computer program ~~"IONXCHGR"~~ described in Section 14.2.2.2. ~~The inventory of iodine in the steam generator at equilibrium conditions is reduced by a factor of ten due to plateout losses on the internal surfaces of the secondary side.~~

The Westinghouse Computer Codes utilized in the transient analyses of Sections 14.1 and 14.2 are as follows:

Code Name	Reference Section
1. FACTRAN	14D.10.1
2. BLKOUT	14D.10.2
3. MARVEL	14D.10.3
4. LEOPARD	14D.10.5
5. TURTLE	14D.10.6
6. TWINKLE	14D.10.7
7. WIT	14D.10.8
8. PHOENIX	14D.10.9
9. LOFTRAN	14D.10.4

The following conditions were assumed to exist at the time of a main steam line break accident:

1. End of life shutdown margin at no load, equilibrium xenon conditions, and the most reactive assembly stuck in its fully withdrawn position: Operation of the control rod banks during core burnup is restricted in such a way that addition of positive reactivity in a steam line break accident will not lead to a more adverse condition than the case analyzed. The value of shutdown margin used is 1.77 percent.

2. The negative moderator coefficient corresponding to the end of life rodded core with the most reactive rod in the fully withdrawn position. The variation of the coefficient with temperature and pressure has been included. The k_{eff} versus temperature at 1,000 psi corresponding to the negative moderator temperature coefficient used plus the Doppler temperature effect, is

The sequence of events is shown in Table 14.2-2.

Radiological Consequences

The radiological consequences of a main steam line break (MSLB) were re-analyzed in support of the Alternate Repair Criteria (ARC) for steam generators (ref. USNRC GL 95-05)[26]. The MSLB is of interest due to the rapid depressurization of the secondary side and the high differential pressure across the steam generator tubes that can occur. Such conditions can result in accident-induced primary-to-secondary leakage. The ARC allows steam generator tubes having defects to remain in service with higher non-destructive examination (NDE) indications than would have been allowed under prior repair criteria, subject to conditions established in technical specifications. One such requirement is to project, on the basis of the NDE indication (voltage), the potential MSLB-induced leakage (95% prediction with 95% confidence), and the offsite and control room operator doses that could result.

accident induced

In lieu of calculating the radiological consequence of this event with each operating cycle, an analysis was performed to establish a maximum allowable accident-induced leakage, against which the cycle leakage projections could be compared. This leakage rate is the maximum primary-to-secondary leakage that could occur with offsite and control room operator doses remaining within applicable limits. 0-30day This re-analysis showed that the ~~30 rem 0-2 hour~~ thyroid dose ~~at the exclusion area boundary~~ was limiting with a projected leakage of ~~8~~3.0 gpm. Since steam generator tubes with NDE indications corresponding to potential leak rates greater than ~~8~~3.0 gpm will be repaired, this is expected to be the bounding case for future operating cycles. to the control room operator

The MSLB is assumed to occur between the containment wall and the main steam isolation valve, resulting in an unisolable release path to the environment. This re-analysis was performed using the guidance of the Standard Review Plan (SRP)[27] with two exceptions: (1) the dose calculation methodology (see Section 14B.8.5) is based on ICRP-26 and ICRP-30 principles rather than that described in the SRP, and (2) the primary-to-secondary leak rate is the 95% prediction 95% confidence leak rate projected on the basis of NDE indications rather than the value established by technical specification.

The analysis assumes that the unit is operating with technical specification primary and secondary coolant specific activities. In conjunction with this analysis, the reactor coolant system specific activity technical specification was reduced from 1.0 µCi/gm to 0.35 µCi/gm. A primary-to-secondary technical specification leakrate of 150 gpd is assumed in all steam generators prior to the event and in the remaining steam generators post-event. The thermodynamic analysis indicates that DNB is not exceeded and, therefore, no fuel damage is projected.

The analysis was performed assuming two iodine spike cases, pre-incident iodine spike, and accident-initiated spike (co-incident). Offsite power is assumed to be lost making the condenser unavailable for steam dump.

The release sources considered include: pre-event liquid and steam activity in all steam generators, primary-to-secondary leakage during the event. The release of primary-to-secondary leakage is assumed to continue for eight hours. Significant analysis inputs are listed in Table 14.2-10.

The 0-2 hour doses at and <2E-01 the exclusion 2.5E+01 area boundary for the co-incident iodine spike case were ~~30~~ rem thyroid committed dose equivalent (CDE), ~~0.11~~ rem effective dose equivalent (EDE)~~, 0.04 rem skin dose equivalent (DE)~~. The low population zone (LPZ) doses for the co-incident spike case were ~~18~~ rem thyroid CDE, and <2E-01 ~~0.04~~ rem EDE~~, 0.02 skin DE~~. These co-incident iodine spike doses are a small fraction of the 10 CFR Part 100 guidelines and are, therefore, acceptable. 1.5E+01

The 0-2 hour doses at the 1.5E+01 exclusion area boundary and <2E-01 for the pre-incident spike case were ~~46~~ rem thyroid CDE, ~~0.05~~ rem EDE~~, 0.03 rem skin DE~~. The LPZ doses for the pre-incident spike case were 2.8E+00 ~~9.5~~ rem thyroid CDE, ~~<0.01~~ rem EDE~~, <0.01 skin DE~~. These pre-incident iodine spike and <2E-01 doses are ~~less than~~ a small fraction of the 10 CFR Part 100 guidelines and are, therefore, acceptable. ~~[Note: EAB and LPZ doses listed above represent the limiting bounding values.]~~

The dose to control room operators was also assessed and is documented in Section 11.3.

14.2.5.1.4 Conclusions

The analysis has shown that the criteria stated earlier in this section are satisfied.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable and not precluded in the criterion, the above analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive assembly stuck in its fully withdrawn position.

14.2.5.2 Major Rupture of a Main Feedwater Pipe

14.2.5.2.1 Identification of Causes and Accident Description

A major feedwater line rupture is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve and the steam generator, fluid the instantaneous release of 18 percent of gap activity to the RCS. A coincident loss of AC power to station auxiliaries is assumed, resulting in an 8 hour plant cooldown via steam release from the secondary system to the atmosphere. Table 14.2-4b tabulates significant analysis parameters.

4.7E-01 5.8E+00

The analysis projected 0-2 hour doses at the exclusion area boundary of ~~2.3~~ rem whole body~~, 1.4 rem beta skin,~~ and ~~21.6~~ rem thyroid. The dose results of the analyses are a small fraction of the 10 CFR 100 exposure guidelines and are, therefore, acceptable. [Note: the above listed doses represent the limiting bounding values.]

Insert

14.2.8 Inadvertent Loading of a Fuel Assembly into an Improper Position

14.2.8.1 Identification of Causes and Accident Description

Fuel and core loading errors, such as can arise from the inadvertent loading of one or more fuel assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment, or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment, will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods.

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes which are more peaked than those calculated with the correct enrichments. The incore system of moveable flux detectors which is used to verify power shapes at the start of life is capable of revealing any assembly enrichment error or loading error which causes power shapes to be peaked in excess of the design value.

To reduce the probability of core loading errors, each fuel assembly is marked with an identification number and loaded in accordance with a core loading diagram. During core loading, the identification number will be checked before each assembly is moved into the core. Serial numbers read during fuel movement are subsequently recorded on the loading diagram as a further check on proper placing after the loading is completed.

The power distortion due to any combination of misplaced fuel assemblies would significantly raise peaking factors and would be readily observable with in-core flux monitors. In addition to the flux monitors, thermocouples are located at the outlet of about one-third of the fuel assemblies in the core. There is a high probability that these thermocouples would also indicate any abnormally high coolant enthalpy rise. In-core flux measurements are taken during the startup subsequent to every refueling operation.

Section 14.2.7.2.1 Insert

Calculated doses to the control room operators are provided in Table 11.3-7. These doses are within the 10 CFR Part 50 Appendix A GDC 19 limit of 5 rem to the whole body or its equivalent to any part of the body.

References to Section 14.2 (CONT'D)

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13. R. F. Barry, "LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094," WCAP-3269-26, Westinghouse Electric Corporation (September 1963).

14. "RADIOISOTOPE, A Computer Program for Calculating Residual Activities in a Closed System After One or More Decay Periods," RP-1, Stone & Webster Engineering Corporation (November 1972).

15. "ACTIVITY, A Computer Program for Calculating Fission Product Activity in Fuel, Coolant, and Selected Tanks for a Nuclear Power Plant," RP-3, Stone & Webster Engineering Corporation (January 1973).

16. "IONEXCHANGER, A Computer Program for Determining Gamma Activities in Ion Exchangers or Tanks as a Function of Time for Constant Feed Activity," RP-2, Stone & Webster Engineering Corporation (December 1972).

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

18. DLC Calculation ERS-SFL-89-021, Safety Analysis of the Dose Consequences of a Locked Rotor Accident at BVPS-1 with 18% Fuel Failure -- EAB, LPZ, Control Room.

19. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

20. ~~Combined BV1-BV2 Control Room Habitability Due to Design Basis Accidents (except LOCA) at BV1, Calculation 12241/14110.39-UR(B)-456-0, 1987.~~

21. ~~Doses to the BV1 Control Room Due to LOCA at BV1, Calculation 14110.39-UR(B)-457-0, 1987.~~ Insert

22. ~~DLC Calculation ERS-SFL-92-025 r0, Safety Analysis of the Dose Consequences of a Fuel Handling Accident at BVPS 1 to the Common Control Room, 1992.~~

23. DLC Calculation ERS-SFL-92-033 ~~r1,~~ Combined Control Room Doses Due to SGTR at Unit 1~~, 1996.~~

24. ~~DLC Calculation ERS-SFL-93-005, Safety Analysis of Consequences of Control Room Damper Response Delay (Limitorque 10 CFR 21) Unit 1 Accidents.~~ Deleted

<Page 14.2-53 Insert>

20. SWEC Calculation 12241/11700 UR(B)-480, Radiological Dose Consequences at the EAB, LPZ, and in the Control Room due to a Postulated LOCA at Beaver Valley Power Station, Unit 1, 1999.

21. DLC Calculation ERS-JTL-99-014, Safety Analysis of the Radiological Consequences of a Waste Gas System Rupture DBA at Unit 1, Control Room, EAB and LPZ doses, 1999.

22. DLC Calculation ERS-JTL-99-009, Safety Analysis of the Radiological Consequences of a Fuel Handling DBA at BVPS Unit 1, Control Room, EAB and LPZ Doses, 2000.

24. DLC Calculation ERS-SFL-95-008, Safety Analysis of the Common Control Room, EAB and LPZ Doses from a Main Steam Line Break Outside of CNMT at U1 with Increased Primary-to-Secondary Leakage, 1999.

The elastic design capability of the primary shield wall greatly exceeds the differential pressure and loading described above. Initially the shield wall was designed to contain a 100 psi differential pressure across its full height. Later design development and final loading criteria resulted in a transfer of loading to the primary shield wall as shown by Figure 14.3-85. This loading is less severe than the initial criteria given to design the area.

Results

Figures 14.3-84 through 14.3-96 graphically show the pressure response (absolute pressure and differential pressure) of the components as a function of time.

It should be noted that all of the above plots represent differential pressures between the cubicle and the containment free volume.

Absolute pressure for each of the four nodes shown in Figure 14.3-84, Figures 14.3-95 and 14.3-96 may be obtained by adding 9.51 to the value shown on the curve since a constant containment back pressure of 9.51 was assumed for the analysis. Absolute pressure for each of the steam generator cubicles can be obtained by adding the back pressure from Figure 14.3-93 to the differential pressure in Figure 14.3-87 or 14.3-88. Absolute pressure for each of the pressurizer cubicles may be obtained by adding the back pressure from Figure 14.3-94.

14.3.5 Radiological Consequences

This section addresses the radiological consequences of the postulated design basis accident. Although the preceding thermal and hydraulic analyses concluded that there would be little if any core damage, the analyses described in this section are based on assumptions that assume~~, in one case,~~ major core damage resulting in release of significant radionuclides to the containment~~, and in the other, significant, but more realistic quantities of released radionuclides. The former is titled the design base analysis or DBA-case, while the latter is titled the realistic case.[68,69]~~

14.3.5.1 Effectiveness of Spray Systems for Iodine Cleanup

The containment depressurization system (section 6.4) is designed to reduce post accident containment pressure by condensing released steam and is designed to absorb iodine present in the containment atmosphere in inorganic vapor form and in particulate form with chemical spray.

The removal of iodine vapor by chemical sprays is modeled as a mass transfer process[67]. The mass transfer coefficients for iodine in the gas phase and within the liquid droplet are used in calculating a spray removal rate coefficient, LAMBDA(V), in terms of the volume sprayed, the flow rate of the sprayed liquid and the terminal velocity of the spray droplets. Table 14.3-12 provides the formulae used, while Table 14.3-13 tabulates the input, intermediate, and resulting values of the calculation.

Particulates suspended in a containment vessel may deposit out or otherwise be removed from the vessel by a wide variety of mechanisms including: plateout, sedimentation, thermal effects, leakage from the vessel, filtration, and/or chemical sprays. Reference (a) noted that spraying predominates any other natural aerosol removal mechanism. For this reason, no analysis was made of the other natural removal processes. The particulate removal rate coefficient, LAMBDA(P), was determined in terms of collision efficiency, settling velocity, droplet size, and the flow rate of the sprayed liquid. Table 14.3-12 provides the formula used, while Table 14.3-13 tabulates the input, intermediate, and resulting values of the ~~calculation.~~

From the data provided in Section ~~6.4~~, the coverage of the quench spray system is ~~909,000~~ 486,545 cu.ft. of a possible containment free volume of 1,796,300 cu.ft. When the recirculation sprays initiate at 305 seconds, the coverage is such that only ~~89~~ 55 percent of the containment free volume is sprayed. The minimum pH of the quench spray during operation is 8.5 and the minimum pH of the containment sump and therefore the recirculation sprays is 8.0. The quench sprays are assumed to initiate instantaneously, as the actual spray start is postulated at 64 seconds, less than the regulatory guidance of 90 seconds. The preceding data is based on minimum safeguards. Credit for spray removal of iodine is taken after 305 seconds, the start of recirculation spray.

A two-region model is used for the analysis of radioiodine in the containment atmosphere, for the duration of the release period 0-3600 seconds. This model, described in Appendix 14B, calculates the iodine leakage from the sprayed region and the unsprayed region with mixing between regions.

The spray removal constants, LAMBDA(V) and LAMBDA(P), were calculated for the four time periods identified in the original FSAR analysis. The constants for vapor removal varied from 11.3 per hour to 17.5 per hour for the quench spray and recirculation spray (minimum safeguards). A value of 10 per hour will be used in the calculations. Note that this analysis did not take additional credit for the improved effectiveness in the volume where the sprays overlap. In a similar manner, a conservative particulate removal rate was determined to be 0.4~~6~~4 per hour.

For the vapor removal rate, the continued effectiveness of the recirculation sprays is limited by the partitioning factor of the sump water, ie: when iodine is evolved by the sump at a rate equal to or higher than the removal of iodine by the sprays. In this analysis, this limitation was not reached prior to termination of the containment leakage. It is noted, however, that the quench spray continues beyond the point that the containment becomes subatmospheric. Thus, fresh spray water is continuously injected, offsetting the partitioning limitation of the recirculation sprays. For this and other reasons, the iodine removal postulated herein is conservative.

14.3.5.2 Release Pathways for DBA Case

A LOCA would increase the pressure in the containment, initiating containment isolation, auxiliary feedwater, emergency core cooling, and containment spray. Normal ventilation in the auxiliary and contiguous buildings is realigned and the engineered safety features (ESF) areas are aligned and exhausted by the supplementary leak collection and release system (SLCRS). However, no credit is taken for containment leak collection and filtration prior to release to the environment. The main control room environment is ensured by immediate isolation of intake air, by a containment isolation Phase B (CIB) signal, and a supply of clean air from an emergency bottled air system. One hour after the LOCA, when the containment has returned to a subatmospheric condition, the control room ventilation system provides filtered intake air to the control room.

The doses to personnel in the control room following the DBA are provided in Section 11.3.5, while doses determined for the Population at the exclusion area boundary (EAB) and at the low population zone (LPZ) outer boundary are provided in Table 14.3-14a. These doses are due to leakage from the containment building and ESF equipment, and due to direct shine from the containment building. Figure 14B-1 of Appendix 14.B illustrates the release pathways to the general public. Table 14.3-14a tabulates all significant assumptions used in this analysis.

<u>Containment Leakage Source</u>

For a LOCA, it is postulated that 100 percent of the noble gas inventory and ~~50~~ 25 percent of the iodine inventory in the core after full power operation for ~~650~~ 1500 days are available for release from the containment atmosphere ~~at~~ time = 0. The containment structure is assumed to leak at the design basis leak rate of 0.1 volume percent per day for the first hour after the accident. Within the hour the containment is brought to subatmospheric pressure, precluding any further leakage.

<u>Engineered Safety Features Leakage</u>

In the event of a LOCA, safety injection will be initiated. At 305 seconds, recirculation spray starts. Also, at about thirty minutes, safety injection shifts to a recirculation by the low head safety injection (LHSI) pumps. Each of these systems has components located outside of the containment in the safeguards area. Table 6.3-9 and Table 6.4-4 tabulate the maximum expected leakage from these two systems. For the site boundary dose analysis, it was conservatively assumed that the ESF leakage would be ~~3.78×10^4~~ 11.356E+03 cc/hr. (70)(71) 73 74 The analysis assumes that the leakage would remain constant for 30 days. Fifty percent of the core iodine inventory is assumed mixed in the sump water that is circulated through the piping external to the containment. Ten percent of the iodine in the water leaking from the ESF piping is postulated to become airborne. A fraction of this iodine is exhausted to the environment following filtration by the SLCRS.

A surveillance test is performed every 18 months to determine the actual ESF leakage. The acceptance criteria for this test is set at 5.7 x 10^3 cc/hr. This is conservative since the acceptance criteria is based on the Control Room Dose Analysis, which assumes a leak rate of 11.356 x 10^3 cc/hr.[73][74]

Direct Dose

The direct dose is due to activity contained in the atmosphere and the sump of the containment building. This direct dose includes both direct and skyshine components. No credit was taken for shielding other than that of the containment walls.

Control Room Habitability

For the purpose of demonstrating habitability of the common control room, analyses of the radiation doses in the common control room from two loss-of-coolant accidents were performed. Table 14.3-10 tabulates significant analysis parameters associated with a small line break outside containment. Table 14.3-14~~b~~a tabulates significant analysis parameters associated with a large break LOCA. Table 11.3-7 tabulates the results from both analyses.

In addition to the ESF leakage modeled above, analyses were performed in 1992 in response to NRC Information Notice 91-56[85][86]. These analyses addressed the possibility of leakage from ECCS components to the RWST and then to the environment. This leakage was postulated to occur due to leakage through check valves and isolation valves that isolate the normal flow paths from the RWST when the ESF systems re-align for recirculation injection. As with the other ESF leakage, the analysis assumes that 50 percent of the core iodine inventory is mixed in the sump water that will be recirculated through the affected piping. The analysis assumes that the leakage through valves in credible paths is limited to a total of 1.0 gpm. In accordance with the Standard Review Plan, ~~this leak rate is~~ doubled prior to use in the analyses. ~~The~~ leakage into the RWST is postulated to start at about ~~30 minutes post-accident.~~ 0.7 hour However, since water is being drawn ~~from the~~ RWST, a negative pressure is maintained in the RWST, precluding environmental release. The environmental releases are postulated ~~to commence~~ when the pump down of the RWST ceases at about ~~1 hour~~ 1.5 hours post-accident. Doses at the LPZ and Control Room for ~~the~~ duration of the accident were assessed. The results of these calculations are provided in Table 14.3-14a and 11.3-7.

Dose Model

The releases resulting from the LOCA, including ESF leakage, presented in Table 14.3-14a are used in conjunction with the atmospheric dispersion values given in Table 2.2-11 to calculate the offsite doses using the methodology discussed in Appendix 14.B.

The total doses at the exclusion area boundary and the LPZ, presented in Table 14.3-14a are within the guidelines of 10 CFR 100. The dose to the BVPS-1 control room operators due to a LOCA at BVPS-1, as discussed in Section 11, is below the limit set in General Design Criterion 19 of 5 rem whole body, 30 rem thyroid.

14.3.5.3 ~~Realistic Case LOCA~~ Deleted

The design basis accident analysis is based on a set of assumptions, some of which are known to be overconservative. The realistic case analysis substitutes more realistic, but still conservative assumptions, where appropriate. Table 14.3-14a documents the assumptions used in the realistic case analysis opposite the DBA assumptions and also provides the results of both analyses. The more significant assumptions are:

a. The SLCRS is assumed to collect 50 percent of the containment leakage, as discussed in Appendix 14.B.10.

b. The activity released to the containment at t=0 is based on the same fractions used for the design basis case, but applies these fractions to equilibrium gap activity instead of the core inventory. Section 14.3.2.1 addresses how the peak clad temperature does not exceed that temperature necessary for fuel clad failure. On the basis of that discussion, assuming a core inventory release is unrealistic.

c. Section 6 of the FSAR addresses the thermally induced mixing of the containment atmosphere. On the basis of that discussion, an assumption of complete spray mixing is realistic. Therefore, a single region model is used as described in Appendix 14.B.

d. ESF leakage is not considered due to the insignificance of this pathway under the above assumptions. This is particularly the case when a more realistic partitioning factor is assumed for the leak fraction which goes airborne. It is also unlikely that an uncontrolled leak would be allowed to continue, without some mitigating action, for 30 days.

Reference to Section 14.3 (Cont'd)

66. Safety Evaluation in the Matter of Virginia Electric Power Company, Surry Power Stations Units 1 and 2, Docket Numbers 50-280 and 50-281, pp. 57-58, Atomic Energy Commission (February 23, 1972).

67. NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," Appendix E, June 1981.

68. ~~DLC Calculation ERS-SFL-83-016~~ Deleted

69. ~~DLC Calculation ERS-SFL-83-017~~ Deleted

70. ~~DLC Calculation ERS-SFL-84-008~~ Deleted

71. ~~DLC Letter ND1SLC:1037~~ Deleted

72. Deleted by Revision 12

73. Combined BV-1 - BV-2 Control Room Habitability Due to Design Bases Accidents (except LOCA) at BV-1, Calculation Identification Number 12241/14410.39, UR(B)-456 dated April 24, 1987. Insert

74. Doses to the BV-1 Control Room Due to a LOCA at BV-1, Calculation Identification Number 14110.39, UR(B)-457 dated May 11, 1987.

75. "American National Standard for Decay Heat Power in Light Water Reactors," ANSI/ANS - 5.1 - 1979, American Nuclear Society, August 1979.

76. D.C. Slaughterback, "A Review of Heat Transfer Coefficients for Condensing Steam in a Containment Building Following a Loss of Cooland Accident," Interim Task Report, Subtask 4.2.2.1, Idaho Nuclear Corporation, January 1970.

77. "PWR FLECHT Final Report," WCAP - 7665, Westinghouse Electric Corporation, April 1971.

78. Letter from W. J. Johnson (Westinghouse) to R. C. Jones, Jr. (Nuclear Regulatory Commission RSB NRR), Subject: Additional Information on Use of BASH Code, NS-NRC-90-3524, July 20, 1990.

79. Letter from J. N. Steinmetz (Westinghouse) to T. P. Noonan, (Duquesne Light Company), Subject: DLCo BVPS Units 1&2 ECCS Evaluation Model Changes, DLW-91-159, June 20, 1991.

80. Letter from K. E. Halliday (Duquesne Light Company) to G. D. Simmers (Westinghouse), Subject: DLCo BV Unit 1 Increased SGTP and Reduced TDF Program, ND1MNE:6312, October 12, 1992. (LOCA Parameters.)

73. DLC Calculation ERS-SFL-97-006, Unit 1 Post-LOCA CNMT Spray Removal Coefficients

74. SWEC Calculation ERS-SNW-92-009, Iodine Release from the Beaver Valley Unit 1 Refueling Water Storage Tank, 1999.

85. SWEC Calculation 12241/11700-UR(B)-480, Radiological Dose Consequences at the EAB, LPZ and in the Control Room due to a Postulated LOCA at Beaver Valley Power Station, Unit 1, 1999.

86. DLC Calculation ERS-JTL-99-015, Safety Analysis of the Radiological Consequences of a Small Line Break (Primary System Outside of Containment) DBA at BVPS Unit 1, Control Room, EAB and LPZ Doses.

References to Section 14.3 (Cont'd)

81. Letter from K. E. Halliday (Duquesne Light Company) to G. D. Simmers (Westinghouse), Subject: DLCo BV Unit 1 Increased SGTP and Reduced TDF Program, ND1MNE:6340, November 12, 1992. (ECCS Performance.)

82. D. J. Shimeck, "Spacer Grid Heat Transfer Effects During Reflood," WCAP-10484-P Addendum 1, December 1992. (Transmittal to Nuclear Regulatory Commission DCD: ET-NRC-92-3787, December 22, 1992, "Notification of Changes to the Westinghouse Large Break LOCA ECCS Evaluation Model.")

83. Letter from J. M. Hall (Westinghouse) to N. R. Tonet (Duquesne Light Company), Subject: DLCo BV Units 1&2 10 CFR 50.46 Notification and Reporting Information, DLW-93-202, January 29, 1993.

84. Letter from N. J. Liparulo (Westinghouse) to Nuclear Regulatory Commission Document Control Desk, Subject: Extension of NUREG-0630 Fuel Rod Burst Strain and Assembly Blockage Models to High Fuel Rod Burst Temperatures, ET-NRC-92-3746, September 16, 1992.

85. ~~Doses to the Combined BV1/BV2 Control Room and the EAB and LPZ due to Release from the RWST via ECCS Leakage Following a LOCA at Beaver Valley Unit 1, SWEC, calculation ERS-S&W-92-011, April 1992.~~

86. ~~Nuclear Regulatory Commission Information Notice 91-56, Potential Radioactive Leakage to Tank Vented to Atmosphere, dated September 19, 1991.~~ Insert

87. ~~DLC Calculation ERS-SFL-93-005 r0, Safety Analysis of Consequences of Control Room Damper Response Delay (Limitorque 10 CFR 21) Unit 1 Accidents, 1993.~~
Deleted

TABLE 14.1-3

PARAMETERS USED IN CONTROL ROOM HABITABILITY ANALYSIS OF THE LOSS OF AC POWERED AUXILIARIES ACCIDENT*

	Design Case
Power, MWt	2705 ~~2766~~
~~Fraction of fuel with defects~~	~~0.0026~~
Offsite AC power	Lost
Initial primary system activity (d.e. I-131), μCi/gm	0.35 ~~1.0~~
Initial secondary system activity (d.e. I-131), μCi/gm	0.1
Pre-incident primary-to-secondary leak rate, gpm	
any one S/G, gpd	150 ~~500~~
all three S/Gs, ~~gpm~~ gpd	450 ~~1.0~~
Steam generator fluid content/sg, lbm	
Liquid **	103,868 -10% ~~97,900~~
Steam **	5,807 +10% ~~6460~~
RCS fluid content, lbm **	345,800 ~~390,000~~
Steam release from steam generators, lbm	
0-2 hrs	443,878
2-8 hrs	793,664
Duration of plant cooldown by secondary system after accident, hours	8
Iodine partition factor in all steam generators prior to and during the accident	0.01
Control Room volume, ft^3	1.73E+5
Control Room normal intake, cfm	500
Control Room isolation	None
Control Room purging	None
Control Room χ/Q value 0-8 hours, sec/m^3	Table 2.2-12 ~~2.97E-3~~
Analysis references	~~12241-UR(B)-456 r0~~ ERS-AJL-99-012

*This analysis was originally performed in 1987 in support of plant modifications converting the Unit 1 Control Room to a common Unit 1 - Unit 2 facility. The analysis inputs, assumptions, and methodologies are based on regulatory requirements that existed at the time the analysis was performed. The radiological consequences of this event were not required to be evaluated as part of the licensing basis for Unit 1.

** Parameters are for 100% power, 30% steam generator tube plugging. The conditions and the stated uncertainty are bounding for this accident.

TABLE 14.2-4b

PARAMETERS USED IN RADIOLOGICAL ANALYSIS OF THE LOCKED ROTOR ACCIDENT

	Design Case
Power (MWt)	2705 ~~2766~~
Fraction of fuel with defects	
~~Pre-incident (approximate)~~	~~0.01~~
Incident	0.18
Primary-to-secondary leak rate (gpd) (per steam generator)	150
Iodine partition factor in all steam generators prior to and during the accident	0.01
Duration of plant cooldown by secondary system after accident (hr)	8
Steam release from steam generators (lb)	
0-2 hrs	443,878
2-8 hrs	793,664 ~~793,644~~
Steam generator fluid content/sg (lb)	
Liquid**	103,868 − 10% ~~88,100~~
Steam**	5,807 + 10% ~~6980~~
Iodine spike duration (hr)	4
Primary coolant fluid content (lb)**	345,800 ~~351,000~~
EAB and LPZ χ/Q values (sec/m^3)	Table 2.2-11
Initial primary system (d.e. I-131) activity, µCi/gm	0.35
Initial secondary system (d.e. I-131) activity, µCi/gm	0.1
Offsite AC power	Lost
Fraction of core inventory in gap	
Kr-85	0.3
I-131	0.12
Others	0.1
~~Control Room Habitability Analysis Parameters~~	
Control Room volume, ft^3	1.73E+5
Control Room normal intake, cfm	500
CREBAPS actuation by radiation monitor, sec	1645 ~~1085~~
Time to reach alarm concentration, sec = ~~190~~ 1450 Damper, EDG loading, instrument response delays, sec = 195	
Delay between CREBAPS depletion and manual start of emergency pressurization fans, min	20

TABLE 14.2-4b (Cont'd)

PARAMETERS USED IN RADIOLOGICAL ANALYSIS
OF THE LOCKED ROTOR ACCIDENT

Control Room pressurization rate, cfm	
Air bottles (flow rate used for activity removal)	0
Emergency filtered intake (after one hour)	600 ~~1030~~
Control Room intake filter efficiency, %	95
Control Room unfiltered infiltration, cfm	10
During fan start 20 min delay	310
Control Room purging	None
Control Room χ/Q values, sec/m^3	Table 2.2-12
Offsite χ/Q values, sec/m^3	Table 2.2-11a
Analysis references	ERS-SFL-89-021

** Parameters are for 100% power, 30% steam generator tube plugging. These conditions and the stated uncertainty are bounding for this accident.

TABLE 14.2-6

ASSUMPTIONS USED FOR THE FUEL HANDLING ACCIDENT RADIOLOGICAL ANALYSIS

Core power level, MW_t	~~2766~~ 2705
Fuel enrichment, weight % U*	3.6-5.0
~~Fuel burnup, MWD/MTU~~	~~60,000~~
~~Specific power, kW/kgU~~	~~41.16~~
Fuel assemblies in core	157
Power peaking factor	1.65
Number of failed fuel rods	298
Core inventory released to gap, %	
Iodine-131	12
Other iodines	10
Krypton-85	30
Xenon-133	10
Other xenons	10
Iodine composition released from pool, %	
Elemental	~~99.75~~ 75
Organic	~~0.25~~ 25
Pool decontamination factors	
~~Elemental~~ Iodine	~~133~~ 100
~~Organic iodine~~	~~1~~
Noble gases	1
Filter decontamination factors	
~~Elemental~~ Iodine **	~~20~~ 10
~~Organic iodine~~	~~20~~
Noble gases	1
Atmospheric diffusion factor (χ/Q), sec/m^3	~~8.91×10^{-4}~~ Tables 2.2-11a, 2.2-11b
Breathing rate, m^3/sec	3.47×10^{-4}

Basis for core inventory

Core fraction, days at 100% power	1/3, 500
	1/3, 500
	1/3, 500
Between cycle cooldown time (days)	40

Table 14.2-6 (Cont'd.)

ASSUMPTIONS USED FOR THE FUEL HANDLING ACCIDENT ANALYSIS

~~Control Room Habitability Analysis Parameters~~

Release Duration~~, days~~	Puff ~~30~~
Release rate, 1/sec* * *	13.82 ~~0.001599~~
Control Room volume, ft^3	1.73E+5
Control Room normal intake, cfm	500
Control Room isolation	None
Control Room purging	None
Control Room χ/Q values, sec/m^3	Table 2.2-12
Analysis reference ~~(Control Room only)~~	~~ERS-SFL-92-025 r0~~ ERS-JTL-99-009

* * *represents 99.9999% of the activity in the fuel building atmosphere being released in 1 second ~~2 hours~~. This release rate is greater than that suggested by the building ventilation flow rate and volume. Analysis assumed instantaneous release from fuel to the fuel building atmosphere. ~~Release rate assumption differs from that used for offsite analysis in that the offsite analysis assumed an instantaneous release. The Control Room modeling requires a continuous intake rate.~~

* Core activity content was determined for 3.6% and 5% enrichments and the maximum activity for each radionuclide was selected for the analysis.

** Filter iodine total DF combines removal efficiencies of 95% for elemental iodine, 80% for organic iodine and 1% filter bypass for both.

TABLE 14.2-6a

RESULTS OF ORIGEN-~~2~~ CALCULATIONS FOR RADIONUCLIDES OF IODINE, KRYPTON, AND XENON AT 150-HOURS COOLING TIME

Insert

Radionuclide	Curies per MTU
I-131	6.519×10^5
I-132	4.149×10^5
I-133	1.473×10^4
Kr-85	1.612×10^4
Xe-131m	1.129×10^4
Xe-133	1.138×10^6
Xe-133m	1.408×10^4

Table 14.2-6a Insert

Isotope	Activity in Core
Kr-85m	1.59E-03
Kr-85	7.58E+05
Xe-131m	8.73E+05
Xe-133m	1.03E+06
Xe-133	7.91E+07
Xe-135m	3.09E+00
Xe-135	4.67E+03
I-131	4.39E+07
I-132	2.84E+07
I-133	1.03E+06
I-135	1.90E+01

TABLE 14.2-6b

Delete

RADIONUCLIDE PROPERTIES USED IN THE FUEL HANDLING ACCIDENT ANALYSIS

Radionuclide	Dose Conversion, Rads/Curie	E_β (Mev)	E_γ (Mev)
Iodine-131	1.48×10^6	-------	-------
Iodine-132	5.35×10^4	-------	-------
Iodine-133	4.0×10^5	-------	-------
Krypton-85	-------	0.223	0.002
Xenon-131m	-------	0.140	0.164
Xenon-133	-------	0.146	0.045
Xenon-133m	-------	0.155	0.042

TABLE 14.2-7

ACTIVITIES RELEASED TO THE SECONDARY SIDE OF THE STEAM GENERATOR FOLLOWING TUBE RUPTURE ACCIDENT - (WITHOUT OFFSITE POWER)

Isotope	Activity Curies
I-131	155
I-132	53
I-133	239
I-134	33
I-135	127
Kr 85	655
Kr 85m	115
kr 87	72
Kr 88	192
Xe 133	1,609
Xe 133m	187
Xe 135	195
Xe 135m	65

delete incorporated by reference

TABLE 14.2-8

PARAMETERS USED IN ~~CONTROL ROOM HABITABILITY~~ RADIOLOGICAL ANALYSIS OF THE WASTE GAS SYSTEM FAILURE ACCIDENT*

	Design Case
Power, MWt	2705 ~~2766~~
Fraction of fuel with defects	0.01
Offsite AC power	Lost
Letdown flow rate, gpm	120
Analysis cases	
Rupture of Waste Gas Storage Tank (WGST)	
Rupture of Gaseous Waste System line upstream of charcoal delay beds	
Volume of WGST, ft^3	132
WGST feed rate, scfm	2
Gaseous Waste System operating pressure, psig	65
Gaseous Waste System operating temperature, °F	100
Charcoal bed holdup time, days	
Krypton	2.1
Xenon	38.7
Control Room volume, ft^3	1.73E+5
Control Room normal intake, cfm	500
Control Room isolation	None**
Control Room purging	None
Control Room χ/Q values, sec/m^3	
Auxiliary Building 0-8 hours (line rupture case)	Table 2.2-12
Gaseous Waste Vault 0-8 hours (WGST rupture case)	Table 2.2-12 ~~2.03E-2~~
Offsite χ/Q values, sec/m^3	Tables 2.2-11a, 2.2-11b
Analysis references	~~12241/14110.39-UR(B)-456~~ ERS-JTL-99-014

* This analysis was originally performed in 1987 in support of plant modifications converting the Unit 1 Control Room to a common Unit 1 - Unit 2 facility. The analysis inputs, assumptions, and methodologies are based on regulatory requirements that existed at the time the analysis was performed~~, and may differ from those used in the offsite dose calculations~~.

** The dose calculation conservatively assumed that the control room radiation monitors will not initiate control room isolation.

TABLE 14.2-9

PARAMETERS USED IN ~~CONTROL ROOM HABITABILITY~~ RADIOLOGICAL ANALYSIS OF THE STEAM GENERATOR TUBE RUPTURE ACCIDENT*

	Design Case
Power, MWt	2705 ~~2766~~
~~Fraction of fuel with defects~~	~~0.0026~~
Offsite AC power	Lost
Initial primary system activity (d.e. I-131), μCi/gm	~~1.0~~ 0.35
Initial secondary system activity (d.e. I-131), μCi/gm	0.1
Concurrent iodine spike release rates into primary coolant (Ci/sec) **	
I-131	1.16 ~~1.36~~
I-132	1.16 ~~2.52~~
I-133	2.00 ~~3.08~~
I-134	1.29 ~~3.68~~
I-135	1.52 ~~2.81~~
Iodine spike duration, hour	4
Pre-incident primary-to-secondary leak rate, gpm	
any one S/G, gpd	150 ~~500~~
all three S/Gs, ~~gpm~~ gpd	450 ~~1.0~~
Steam generator fluid content/sg, lbm	
Liquid **	103,868 -10% ~~97,900~~
Steam **	5,807 +10% ~~6460~~
RCS fluid content, lbm ** Total / Less Pressurizer	~~390,000~~ 345,800 / 314,500
RCS inventory released to ruptured steam generator, lbm	137,400
Steam release from ruptured steam generator, lbm	
0-2 hrs	48,100
2-8 hrs	28,760
Steam release from intact steam generators, lbm	
0-2 hrs	364,600
2-8 hrs	800,600
Duration of plant cooldown by secondary system after accident, hours	8
Iodine partition factor in all steam generators prior to and during the accident	0.01

*This analysis was originally performed in 1987 in support of plant modifications converting the Unit 1 Control Room to a common Unit 1 - Unit 2 facility. The analysis inputs, assumptions, and methodologies are based on regulatory requirements that existed at the time the analysis was performed, ~~and may differ from those used in the offsite dose calculations.~~

** Parameters are for 100% power, 30% steam generator tube plugging. These conditions and the stated uncertainty1 of 2 are bounding for this accident.

TABLE 14.2-9 (Cont'd.)

PARAMETERS USED IN RADIOLOGICAL ~~CONTROL ROOM HABITABILITY~~ ANALYSIS OF THE STEAM GENERATOR TUBE RUPTURE ACCIDENT*

Control Room volume, ft^3	1.73E+5
Control Room normal intake, cfm	500
Control Room isolation	None
Control Room purging	None
Control Room χ/Q values, sec/m^3	Table 2.2-12
Analysis references	ERS-SFL-92-033 ~~1~~
Offsite X/Q values, sec/m^3	Tables 2.2-11a, 2.2-11b

TABLE 14.2-10

PARAMETERS USED IN MAIN STEAM LINE BREAK RADIOLOGICAL ANALYSIS

	Design Case
Power, MWt	2705 ~~2766~~
Offsite AC Power	Lost
Initial Primary Coolant Activity, Dose Equivalent I-131, µCi/gm	0.35
Initial Secondary System Activity, Dose Equivalent I-131, µCi/gm	0.1
Initial Primary and Secondary Isotopic Concentrations	Table 14B-15
Concurrent Iodine Spike Appearance Rates, Ci/s	Table 14B-16
Pre-incident Iodine Spike Concentrations	Table 14B-16
Iodine Spike Duration, hour	4
Primary-to-Secondary leak rate	
Pre-event, any SG, gpd	150
Affected SG, gpm	8.0
Post-event, each unaffected SG, gpd	150
Steam Generator Fluid Content	
Liquid, lbm @	164,000
Steam, lbm @	6100
RCS Fluid Content, lbm	351,000
Steam Release from Affected SG, lbm	
0-30 minutes	150,000
30 min-8 hrs	1300
Steam Release from Intact SG, lbm	
0-2 hours	366,776
2-8 hours	705,393
Duration of Release, hours	8
Iodine Partition Factor	
Affected SG	1.0
Intact SGs (initially)	1.0
Intact SGs (after 1 hour)	0.01
Offsite χ/Q Values, sec/m^3	Tables 2.2-11a, 2.2-11b
Control Room Volume, ft^3	173,000
Control Room Normal Intake, cfm	500
Control Room Isolation	Manual Actuation at T=30 minutes
Control Room Purge > T=8 hours	28,000 cfm for 30 minutes
Control Room χ/Q Values, sec/m^3	Table 2.2-12
Dose Calculation Method	Section 14B.8.5
Analysis Reference	ERS-SFL-95-008

Insert

<Insert>

TABLE 14.2-10

PARAMETERS USED IN MAIN STEAM LINE BREAK RADIOLOGICAL ANALYSIS

Power, MWt	2705
Offsite AC Power	Lost
Initial Primary Coolant Activity, Dose Equivalent I-131, uCi/g	0.35
Initial Secondary Coolant Activity, Dose Equivalent I-131, uCi/g	0.1
Initial Primary and Secondary Isotopic Concentrations	Table 14B-15
Concurrent Iodine Spike Release Rates, uCi/s (Specific for the Main Steam Line Break)	I-131 1.16E+06 I-132 1.12E+06 I-133 1.99E+06 I-134 1.24E+06 I-135 1.49E+06
Pre-accident Iodine Spike Concentrations	Table 14B-16
Iodine Spike Duration, (hours)	4
Primary-to-Secondary Leak Rate	
Pre-event, each SG, gpd	150
Affected SG Accident Induced gpm	3.0
Post event, each SG, gpd	150
Steam Generator Fluid Content	
Liquid, lbm	148,104 +10%
Steam, lbm	5,781 +10%
RCS Liquid Content, lbm	
Total	329,500
Less Pressurizer	314,500
Steam Release From Affected SG, lbm	
0-30 min (Initial content and 3.104 gpm)	170,050
30 min - 8 hours (3.104 gpm)	1397
Steam Release from Intact SG, lbm	
0-2 hours	366,776
2-8 hours	705,393
Duration of Release, hours	8
Iodine Partition Factor	
Affected SG	1.0
Intact SGs (Initially)	1.0
Intact SGs (After 1 hour)	0.01

<Insert>

TABLE 14.2-10 (Continued)

PARAMETERS USED IN MAIN STEAM LINE BREAK RADIOLOGICAL ANALYSIS

Offsite X/Q Values, sec/m^3	Tables 2.2-11a, 2.2-11b
Control Room Volume, ft^3	1.73E+05
Control Room Normal Intake, cfm	500
Control Room Manual Isolation, min	30
Post-isolation Bottled Air (CREBAPS) Flow Rate, cfm	600
Unfiltered Infiltration Rate During CREBAPS Pressurization, cfm	10
Post CREBAPS Depletion Pressurization Fan Start Delay, min	20
Unfiltered Infiltration Rate During Fan Start Delay, cfm	310
Time Post-accident for Pressurization Fan Start, min	80
Pressurization Fan Flow Rate, cfm	1030
Unfiltered Infiltration Rate During Fan Run, cfm	10
Control Room Intake Filter Iodine Removal Efficiency, Total %	95
Control Room Purge Duration/Flow Rate, @T=8 hours, cfm	30 min / 28,800
Control Room X/Q Values	Table 2.2-12
Dose Calculation Method	Section 14B.8.5
Analysis Reference	ERS-SFL-95-008

*Parameters are for 0% power, 30% steam generator tube plugging. These conditions and the stated uncertainty are bounding for this accident.

0.12 I-131
0.30 Kr-85
0.10 Others

TABLE 14.2-12

PARAMETERS USED IN ~~CONTROL ROOM HABITABILITY~~ RADIOLOGICAL ANALYSIS OF THE ROD EJECTION ACCIDENT*

	Design Case
Power, MWt	2705 ~~2766~~
Offsite AC power	Lost
Release paths	
Containment design leakage	
Secondary steam release	
Fraction of fuel with defects, post event	0.10
~~Pre-event~~	~~0.0026~~
~~Post event~~	~~0.1~~
Fraction of core inventory in gap	~~0.1~~
Initial primary system activity (d.e. I-131), μCi/gm	~~1.0~~ 0.35
Initial secondary system activity (d.e. I-131), μCi/gm	0.1
Fraction of fuel melted	0.0025
Fraction of activity from melted fuel available for release to environment	
Via containment leakage	
Noble Gases	1.0
Iodines	0.25
Via primary-to-secondary leakage	
Noble Gases	1.0
Iodines	0.5
Steam generator fluid content/sg, lbm	
Liquid **	103,868 +10% ~~97,900~~
Steam **	5,807 +10% ~~6460~~
RCS fluid content, lbm **	345,800 ~~390,000~~
Containment leak rate, % volume per day	0.1
Containment sprays	None
Duration of Containment leakage, hours	1.0
Steam dump, lbm (500 seconds)	58,600
Primary-to-secondary leak rate, ~~gpm~~ gpd	~~1.0~~ 450

*This analysis was originally performed in 1987 in support of plant modifications converting the Unit 1 Control Room to a common Unit 1 - Unit 2 facility. The analysis inputs, assumptions, and methodologies are based on regulatory requirements that existed at the time the analysis was performed. ~~and may differ from those used in the offsite dose calculations.~~

** Parameters are for 100% power, 30% steam generator tube plugging. These conditions and the stated uncertainty are bounding for this accident.

TABLE 14.2-12 (Cont'd.)

PARAMETERS USED IN ~~CONTROL ROOM HABITABILITY~~ RADIOLOGICAL ANALYSIS OF THE ROD EJECTION ACCIDENT*

Duration of primary-to-secondary leakage, seconds	500
Iodine partition factor	0.01
Control Room volume, ft^3	1.73E+5
Control Room normal intake, cfm	500
Control Room isolation	None
Control Room purging	None
Control Room χ/Q values, sec/m^3	Table 2.2-12
Analysis reference	~~ERS-SFL-93-005 r0~~ EAS-JTL-99-010

TABLE 14.3-10

PARAMETERS USED IN CONTROL ROOM HABITABILITY ANALYSIS OF THE SMALL LINE BREAK ACCIDENT*

Specific for the small line break

	Design Case
Power, MWt	2705 ~~2766~~
~~Fraction of fuel with defects (approximate)~~	~~0.01~~
Offsite AC power	Lost
Break location	Letdown line upstream of Non-Regenerative Heat Exchanger in Primary Auxiliary Building
Initial primary system activity (d.e. I-131), μCi/gm	0.35
~~Initial secondary system activity (d.e. I-131), μCi/gm~~	~~0.1~~
Concurrent iodine spike release rates into primary coolant (Ci/sec)	
I-131	1.15E+00 ~~4.88E-1~~
I-132	1.16E+00 ~~7.94E-1~~
I-133	1.99E+00 ~~1.04E+0~~
I-134	1.29E+00 ~~1.14E+0~~
I-135	1.52E+00 ~~9.19E-1~~
Iodine spike duration, hour	4
RCS fluid content, lbm** Total / Less pressurizer	3.46E+05 ~~351,000~~ / 3.15 E+05
RCS release rate, lbm/sec	16.2
Duration of release, minutes	15
Steam flash fraction	0.38
Auxiliary Building holdup or filtration	None
Control Room volume, ft^3	1.73E+5
Control Room normal intake, cfm	500
Control Room χ/Q values, sec/m^3	Table 2.2-12
Analysis reference	~~ERS-SFL-93-005~~ ERS-JTL-99-015

*This analysis was originally performed in 1987 in support of plant modifications converting the Unit 1 Control Room to a common Unit 1 - Unit 2 facility. The analysis inputs, assumptions, and methodologies are based on regulatory requirements that existed at the time the analysis was performed. The radiological consequences of this event were not required to be evaluated as part of the licensing basis for Unit 1.

Replace with attached Table

TABLE 14.3-13

IODINE REMOVAL COEFFICIENTS

QUANTITY	UNITS	SYMBOL	~~FIRST PERIOD~~		~~SECOND PERIOD~~	
Time	sec	t	64-304		305-600	
Temperature	°C	T	~~125~~ 1.1952E+02		115	
Viscosity	gm/cm/sec	μ	~~0.000160~~ 1.612E-04		0.000161	
Diffusivity, liq.	cm^2/sec	D_l	~~0.0000699~~ 6.531E-05		0.0000620	
Diffusivity, gas	cm^2/sec	D_g	~~0.04636~~ 5.092E-02		0.005595	
Density, liq.	gm/cc	ρ_l	~~0.9394~~ 9.437E-01		0.9472	
Density, gas	gm/cc	ρ_g	~~0.001989~~ 1.821E-03		0.001655	
Schmidt Number	unitless	Sc	~~1.7353~~ 1.7411 E+00		1.7412	
Reynolds Number	unitless	Re	~~589.6~~ 5.4391 E-02		524.8	
Droplet Size	cm	d	~~0.125~~ 1.25E-01		~~0.125~~ 1.25E-01	
Terminal Velocity	cm/sec	v	~~379.4~~ 3.857E+02		408.4	
Mass transfer, liq.	cm/sec	K_L	~~0.003679~~ 3.4374 E-03		0.00326	
Mass transfer, gas	cm/sec	K_g	~~7.235~~ 7.6712 E+00		8.30	
			Quench	Recirc.	Quench	Recirc.
Partitioning	unitless	H	5000	1500	5000	1500
Flow Rate	cm^3/sec	F	1.20×10^5	0	1.20×10^5	3.79×10^5
Spray Ring Height	cm	h	~~2865~~ 2899	2377	2865	2377
Containment Volume	cc	V	5.09×10^{10}	5.09×10^{10}	5.09×10^{10}	5.09×10^{10}
Lambda	sec^{-1}	λ	0.003698	0	0.003649	0.004873
Lambda	hr^{-1}	λ	13.31	0	13.13	17.54
Vapor Lambda	hr^{-1}	λ	10		10	
Part. Collection	unitless	ϵ	0.0015		0.0015	
			Quench	Recirc.	Quench	Recirc.
Lambda	sec^{-1}	λ	0.000129	0	0.000128	0.000336
Lambda	hr^{-1}	λ	0.46	0	0.461	1.21
Particulate Lambda	hr^{-1}	λ	0.46		0.46	
~~MAX Iodine DF~~	~~unitless~~		~~(54 for period 64-3600 seconds)~~			

Replace with attached Table

TABLE 14.3-13

IODINE REMOVAL COEFFICIENTS

QUANTITY	UNITS	SYMBOL	~~THIRD PERIOD~~		~~FOURTH PERIOD~~	
Time	sec	t	601-1000		1001-3600	
Temperature	°c	T	105		80	
Viscosity	gm/cm/sec	μ	0.000163		0.000172	
Diffusivity, liq.	cm^2/sec	D_l	0.0000546		0.0000386	
Diffusivity, gas	cm^2/sec	D_g	0.06715		0.10013	
Density, liq.	gm/cc	ρ_1	0.9547		0.9718	
Density, gas	gm/cc	ρ_g	0.001395		0.000983	
Schmidt Number	unitless	Sc	1.7450		1.7453	
Reynolds Number	unitless	Re	467.2		355.5	
Droplet Size	cm	d	0.125		0.125	
Terminal Velocity	cm/sec	v	437		497.6	
Mass transfer, liq.	cm/sec	K_L	0.00287		0.00203	
Mass transfer, gas	cm/sec	K_g	9.462		12.51	
			Quench	Recirc.	Quench	Recirc.
Partitioning	unitless	H	5000	1500	5000	1500
Flow Rate	cm^3/sec	F	$1.20x10^5$	$3.79x10^5$	$1.20x10^5$	$3.79x10^5$
Spray Ring Height	cm	h	2865	2377	2865	2377
Containment Volume	cc	V	$5.09x10^{10}$	$5.09x10^{10}$	$5.09x10^{10}$	$5.09x10^{10}$
Lambda	sec^{-1}	λ	0.003555	0.004496	0.003140	0.003488
Lambda	hr^{-1}	λ	12.8	16.19	11.3	12.56
Vapor Lambda	hr^{-1}	λ	10		10	
Part. Collection	unitless	ε	0.0015		0.0015	
			Quench	Recirc.	Quench	Recirc.
Lambda	sec^{-1}	λ	0.000127	0.000334	0.000125	0.000328
Lambda	hr^{-1}	λ	0.458	1.2	0.45	1.18
Particulate Lambda	hr^{-1}	λ	0.46		0.46	
~~MAX Iodine DF~~	~~unitless~~		~~(54 for period 64-3600 seconds)~~			

Insert for Table 14-3-13

QUANTITY	UNITS					
Time	sec	t	64	305	900	1500
Temperature	°C	T	1.1952E+02	1.2145E+02	1.0658E+02	8.8533E+01
Viscosity	gm/cm-sec	μ	1.6123E-04	1.6102E-04	1.6400E-04	1.6990E-04
Diffusivity, liq	cm^2/sec	D_l	6.5303E-05	6.6761E-05	5.5899E-05	4.3907E-05
Diffusivity, gas	cm^2/sec	D_g	5.0913E-02	4.9092E-02	6.4235E-02	8.6183E-02
Density, liq	gm/cc	ρ_l	9.4358E-01	9.4198E-01	9.5237E-01	9.6671E-01
Density, gas	gm/cc	ρ_g	1.8189E-03	1.8851E-03	1.4618E-03	1.1278E-03
Schmidt Number	n/a	Sc	1.7411E+00	1.7399E+00	1.7465E+00	1.7480E+00
Reynolds Number	n/a	Re	5.4391E+02	5.5534E+02	4.7397E+02	3.9491E+02
Droplet Size	cm	d	1.25E-01	1.25E-01	1.25E-01	1.25E-01
Terminal Velocity	cm/sec	v	3.8572E+02	3.7948E+02	4.2540E+02	4.7594E+02
Mass Transfer, liq	cm/sec	K_l	3.4374E-03	3.5141E-03	2.9424E-03	2.3111E-03
Mass Transfer, gas	cm/sec	K_g	7.6712E+00	7.4643E+00	9.1116E+00	1.1282E+01

			QS	RS	QS	RS	QS	RS	QS	RS
Partitioning	n/a	H	5000	1500	5000	1500	5000	1500	5000	1500
Flow Rate	cm^3/sec	F	1.1987E+05	0	1.1987E+05	3.9430E+05	1.1987E+05	3.9430E+05	1.1987E+05	3.9430E+05
Drop Height	cm	h	2.8986E+03	2.4079E+03	2.8986E+03	2.4079E+03	2.8986E+03	2.4079E+03	2.8986E+03	2.4079E+03
Containment Volume	cc	V	5.0970E+10	5.0970E+10	5.0970E+10	5.0970E+10	5.0970E+10	5.0970E+10	5.0970E+10	5.0970E+10
Lambda	sec^{-1}	λ	6.5076E-03	0	6.4360E-03	5.4071E-03	7.0084E-03	4.8320E-03	7.7564E-03	4.0504E-03
Lambda	hr^{-1}	λ	2.343E+01	0	2.317E+01	1.947E+01	2.523E+01	1.740E+01	2.792E+01	1.458E+01
Vapor Lambda	hr^{-1}	λ	10		10		10		10	
Part. Collection	n/a	ε	1.50E-03		1.50E-03		1.50E-03		1.50E-03	

			QS	RS	QS	RS	QS	RS	QS	RS
Lambda	sec^{-1}	λ	1.2270E-04	0	1.2270E-04	3.3530E-04	1.2270E-04	3.3530E-04	1.2270E-04	3.3530E-04
Lambda	hr^{-1}	λ	4.417E-01	0	4.417E-01	1.207E+00	4.417E-01	1.207E+00	4.417E-01	1.207E+00
Particulate Lambda	hr^{-1}	λ	0.44		0.44		0.44		0.44	

Insert for Table 14.3-13 (continued)

Time	sec		1700		2255		3600	
Temperature	°C	T	7.9550E+01		2.5878E+01		5.7994E+01	
Viscosity	gm/cm-sec	μ	1.7321E-04		1.7999E-04		1.7971E-04	
Diffusivity, liq	cm^2/sec	D_l	3.8444E-05		2.5740E-05		2.6774E-05	
Diffusivity, gas	cm^2/sec	D_g	9.7628E-02		1.2289E-01		1.2057E-01	
Density, liq	gm/cc	ρ_l	9.7243E-01		9.8521E-01		9.8421E-01	
Density, gas	gm/cc	ρ_g	1.0158E-03		8.4102E-04		8.5572E-04	
Schmidt Number	n/a	Sc	1.7466E+00		1.7416E+00		1.7418E+00	
Reynolds Number	n/a	Re	3.6459E+02		3.1431E+02		3.1796E+02	
Droplet Size	cm	d	1.25E-01		1.25E-01		1.25E-01	
Terminal Velocity	cm/sec	v	4.9736E+02		5.3817E+02		5.3420E+02	
Mass Transfer, liq	cm/sec	K_l	2.0236E-03		1.3549E-03		1.4093E-03	
Mass Transfer, gas	cm/sec	K_g	1.2338E+01		1.4548E+01		1.4346E+01	
			QS	RS	QS	RS	QS	RS
Partitioning	n/a	H	5000	3000	5000	3000	5000	3000
Flow Rate	cm^3/sec	F	1.1987E+05	3.9430E+05	7.1794E+04	3.9430E+05	7.1794E+04	3.9430E+05
Drop Height	cm	h	2.8986E+03	2.4079E+03	2.8986E+03	2.4079E+03	2.8986E+03	2.4079E+03
Containment Volume	cc	V	5.0970E+10	5.0970E+10	5.0970E+10	5.0970E+10	5.0970E+10	5.0970E+10
Lambda	sec^{-1}	λ	8.1171E-03	6.2741E-03	5.2978E-03	4.7212E-03	5.2631E-03	4.8698E-03
Lambda	hr^{-1}	λ	2.922E+01	2.259E+01	1.907E+01	1.700E+01	1.895E+01	1.753E+01
Vapor Lambda	hr^{-1}	λ	10		10		10	
Part. Collection	n/a	ε	1.50E-03		1.50E-03		1.50E-03	
			QS	RS	QS	RS	QS	RS
Lambda	sec^{-1}	λ	1.2270E-04	3.3530E-04	7.3492E-05	3.3530E-04	7.3492E-05	3.3530E-04
Lambda	hr^{-1}	λ	4.417E-01	1.207E+00	2.646E-01	1.207E+00	2.646E-01	1.207E+00
Particulate Lambda	hr^{-1}	λ	0.44		0.26		0.26	

Based on the above results, an elemental spray lambda of 10 hr^{-1} and a particulate spray removal lambda of 0.44 hr^{-1} are appropriate. QS flow is assumed to be reduced by about 50% at T=2255 seconds, based on MAXIMUM ESF flow rates. However, the spray coefficients are based on minimum flow rates. If maximum flow rate would occur, the flow rates used above would be significantly higher. The regions covered by QS and RS are not treated seperately; therefore, the most restrictive spray removal rate is selected.

① See attached, retyped table

TABLE 14.3-14a

PARAMETERS USED IN EVALUATING THE
RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

		~~DBA CASE~~	Delete REALISTIC CASE
A. Core Parameters			
1. Core Power (MWt)	2705	~~2766~~	2766
2. Days of Operation	1500	~~650~~	650
B. Radiation Source Terms			
1. Release to containment percentage			
a. Noble gas		100	100(1)
b. Iodine		50	50(1)
2. Activity released to containment atmosphere (Ci)			
I^{130}	9.25E+05		
I^{131}	3.62E+07	~~3.42×10^7~~	2.91×10^5
I^{132}	5.30E+07	~~5.20×10^7~~	4.84×10^4
I^{133}	7.45E+07	~~7.65×10^7~~	2.15×10^4
I^{134}	8.25E+07	~~8.95×10^7~~	5.15×10^4
I^{135}	7.05E+07	~~6.95×10^7~~	1.11×10^5
I^{136}	3.27E+07		
Xe^{131m}	9.87E+05	~~5.18×10^5~~	5.38×10^3
Xe^{133m}	4.69E+06	~~4.00×10^6~~	1.81×10^4
Xe^{133}	1.49E+08	~~1.58×10^8~~	1.09×10^6
Xe^{135m}	3.12E+07	~~4.24×10^7~~	1.33×10^4
Xe^{135}	4.85E+07	~~4.31×10^7~~	8.04×10^4
Xe^{137}	1.35E+08	~~1.39×10^8~~	4.55×10^4
Xe^{138}	1.28E+08	~~1.39×10^8~~	4.55×10^4

①

TABLE 14.3-14a (Continued)

Parameters Used in Evaluating the
Radiological Consequences of a Loss-of-Coolant Accident

B. 2. Continued

	~~DBA CASE~~	REALISTIC CASE
Kr^{83m}	9.04E+06 ~~1.27×10^7~~	1.09×10^4
Kr^{85m}	1.88E+07 ~~7.75×10^5~~	3.93×10^4
Kr^{85}	7.59E+05 ~~3.06×10^7~~	1.33×10^5
Kr^{87}	3.78E+07 ~~5.89×10^7~~	4.08×10^4
Kr^{88}	5.25E+07 ~~8.38×10^7~~	8.58×10^4
Kr^{89}	6.57E+07 ~~1.08×10^8~~	1.54×10^4
Kr^{90}	7.05E+07	

3. Activity in Containment Sump at T=0 (Ci)

	DBA CASE	REALISTIC CASE
		N/A
I^{130}	9.25E+05	
I^{131}	3.62E+07 ~~3.42×10^7~~	
I^{132}	5.30E+07 ~~5.20×10^7~~	
I^{133}	7.45E+07 ~~7.65×10^7~~	
I^{134}	8.25E+07 ~~8.95×10^7~~	
I^{135}	7.05E+07 ~~6.95×10^7~~	
I^{136}	3.27E+07	

4. Containment Plateout Percentage of elemental Iodine released to containment — DBA: 50 ~~N/A(2)~~ — Realistic: N/A(2)

5. Iodine Species Percentage

	DBA CASE	REALISTIC CASE
Elemental	91	91
Organic	4	4
Particulate	5	5

C. Data and Assumptions Used to Estimate Activity Released

	DBA CASE	REALISTIC CASE
1. Containment Leakrate (%/day)	0.1	0.1
2. Containment release duration (hrs.)	1	1

TABLE 14.3-14a (Continued)

Parameters Used in Evaluating the
Radiological Consequences of a Loss-of-Coolant Accident

C. Continued

		~~DBA CASE~~	REALISTIC CASE
3. Containment leakage pathway fractions			
a. SLCRS		0	0.5
b. Bypass to environment		1	0.5
4. Containment free volume ~~(cm^3)~~ (ft^3)	1.69E6	~~5.087 x 10^{10}~~	5.087 x 10^{10}
a. Sprayed fraction (t < 305 sec)	0.27	~~0.51~~	1
b. Sprayed fraction (t > 305 sec)	0.55	~~0.89~~	1
5. Containment spray removal coefficients (hr^{-1})			
a. Elemental		10	10
b. Particulate		0.4~~6~~4	0.46
6. Mixing rate between sprayed and unsprayed regions ~~(cm^3/sec)~~ (unsprayed volumes, hr^{-1})		2 ~~1.40 x 10^7/ 3.10 x 10^6*~~	N/A
7. Elemental Iodine Decontamination Factor for Sprays	100	~~54.0~~	N/A
8. Spray delay time (sec)			
a. Quench Spray		64	64
b. Recirculation Spray		305	305

~~* T < 305/T > 305~~

①

TABLE 14.3-14a (Continued)

Parameters Used in Evaluating the Radiological Consequences of a Loss-of-Coolant Accident

		DBA CASE	REALISTIC CASE
C. Continued			
9. SLCRS Filter Efficiencies (%)			
a. Elemental iodine		99	99
b. Particulate iodine		99	99
c. Organic iodine		90	90
d. Noble gas		0	0
D. Offsite Dose Evaluation Results			
1. Dose calculation methodology		ERS-SFL-83-016	ERS-SFL-83-017
2. Activity released to environment by containment leakage (Ci)			
	I^{131}	283	0.672
	I^{132}	405	0.108
	I^{133}	629	0.495
	I^{134}	635	0.110
	I^{135}	563	0.253
	Xe^{131m}	22	0.223
	Xe^{133m}	166	0.749
	Xe^{133}	6563	4.528
	Xe^{135m}	618	0.194
	Xe^{135}	1729	3.224
	Xe^{137}	532	0.174
	Xe^{138}	1869	0.612
	Kr^{83m}	440	0.378
	Kr^{85m}	1181	1.517
	Kr^{85}	32	5.540

TABLE 14.3-14a (Continued)

Parameters Used in Evaluating the Radiological Consequences of a Loss-of-Coolant Accident

D. 2. Continued

		DBA CASE	REALISTIC CASE
	Kr^{87}	1891	1.310
	Kr^{88}	3100	3.174
	Kr^{89}	344	0.049

3. Activity released to environment by ESF leakage (Ci)

	DBA 0-2 hrs	DBA Duration	Realistic 0-2 hrs	Realistic Duration
I^{131}	7.11	954.1	N/A	
I^{132}	8.02	18.2		
I^{133}	15.42	250.4		
I^{134}	8.99	11.5		
I^{135}	13.02	71.6		

4. Atmospheric dispersion factors (sec/m^3)

		DBA CASE	REALISTIC CASE
a. Percentile		0.5%	50%
b. Highest sector (NW)	EAB 0-2 hrs.	8.9×10^{-4}	6.3×10^{-4}
c. Highest sector (NW)	LPZ 0-2 hrs.	9.5×10^{-5}	7.9×10^{-5}
	LPZ 0-8 hrs.	4.2×10^{-5}	3.6×10^{-5}
	LPZ 0-24 hrs.	2.7×10^{-5}	2.4×10^{-5}
	LPZ 30 days	6.8×10^{-6}	6.6×10^{-6}

5. Radiological exposures (rem)

	DBA CASE	REALISTIC CASE
a. Contributors(3)	Containment leakage Containment direct ESF leakage	Containment leakage Containment direct
b. Whole body dose based on	beta & gamma body surface dose rate	gamma with skin attenuation

(1)

TABLE 14.3-14a (Continued)

Parameters Used in Evaluating the Radiological Consequences of a Loss-of-Coolant Accident

D. 5. Continued	DBA CASE	REALISTIC CASE
c. 0-2 hour Dose at EAB		
i. Thyroid	252.2	0.270
ii. Whole Body	4.9	0.0013
d. 30-day Dose at LPZ		
i. Thyroid	42.3	0.034
ii. Whole Body	0.54	0.00016

NOTES:
1. Based on gap activity
2. Treated mechanistically by sprays
3. Hydrogen purge dose treated elsewhere in FSAR
   Control Room dose treated elsewhere in FSAR

(1)

TABLE 14.3-14b Delete

PARAMETERS USED IN CONTROL ROOM HABITABILITY ANALYSIS OF THE LOSS OF COOLANT ACCIDENT*

	Design Case
Power, MWt	2766
Offsite power	Lost
Containment Release Assumptions	
Fraction of core inventory available for release (after plateout)	
Noble gases	1.0
Iodines	0.25
Iodine chemical form	
Elemental	0.91
Particulate	0.05
Methyl	0.04
Containment free volume, ft^3	1.8E+6
Containment volume spray coverage fraction	
$T \leq 305$ sec	0.51
$T > 305$ sec	0.89
Containment volume mixing rate, hr^{-1}	2.0
Spray effective time, sec	
Quench spray	64
Recirculation spray	305
Iodine removal coefficients, hr^{-1}	
Elemental	10.0
Particulate	0.46
Organic	0.0
Elemental Iodine Decontamination Factor	100
Containment leak rate, % volume per day	0.1
Containment leakage duration, hours	1.0
ECCS Leakage Assumptions	
ECCS leak initiation time, sec	305
ECCS leak rate, gpm	0.05
Fraction of core iodine inventory in sump water	0.5

*This analysis was originally performed in 1987 in support of plant modifications converting the Unit 1 Control Room to a common Unit 1 - Unit 2 facility. The analysis inputs, assumptions, and methodologies are based on regulatory requirements that existed at the time the analysis was performed, and may differ from those used in the offsite dose calculations.

(1)

TABLE 14.3-14b (Cont'd.)

PARAMETERS USED IN CONTROL ROOM HABITABILITY ANALYSIS OF THE LOSS OF COOLANT ACCIDENT*

Volume of containment sump water, gal	4.59E+5
Fraction of iodine released to contiguous areas due to leak flashing	0.1
SLCRS filter efficiency, %	95
Control Room volume, ft^3	1.73E+5
Control Room normal intake, cfm	500
CREBAPS actuation by Containment Isolation phase B (CIB), sec	0
Control Room pressurization rate, cfm	
Air bottles	600
Emergency filtered intake (CIB + 1 hour)	776
Control Room intake filter efficiency, %	95
Control Room unfiltered infiltration, cfm	10
Control Room purging	None
Control Room χ/Q values, sec/m^3	

period	CNMT Leakage	ECCS Leakage
0-8 hr	4.33E-4	2.73E-4
8-24 hr	2.04E-4	1.28E-4
1-4 days	1.46E-4	9.17E-5
4-30 days	8.84E-5	5.57E-5

Analysis reference 14110.39-UR(B)-457

TABLE 14.3-14a

PARAMETERS USED IN EVALUATING THE
RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

A. Core Parameters

1. Core Power (MWt)	2705
2. Days of Operation	1500

B. Radiation Source Terms

1. Release to containment (% of core)

a. Noble gas	100
b. Iodine	50

2. Activity released to containment atmosphere (Ci)

I-130	9.25E+05
I-131	3.62E+07
I-132	5.30E+07
I-133	7.45E+07
I-134	8.25E+07
I-135	7.05E+07
I-136	3.27E+07
Xe-131m	9.87E+05
Xe-133m	4.69E+06
Xe-133	1.49E+08
Xe-135m	3.12E+07
Xe-135	4.85E+07
Xe-137	1.35E+08
Xe-138	1.28E+08
Kr-83m	9.04E+06
Kr-85m	1.88E+07
Kr-85	7.59E+05
Kr-87	3.78E+07
Kr-88	5.25E+07
Kr-89	6.57E+07
Kr-90	7.05E+07

3. Activity in Containment Sump at T=0 (Ci)

I-130	9.25E+05
I-131	3.62E+07
I-132	5.30E+07
I-133	7.45E+07
I-134	8.25E+07
I-135	7.05E+07
I-136	3.27E+07

TABLE 14.3-14a (Continued)

PARAMETERS USED IN EVALUATING THE
RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

B.	Continued	
4.	Containment Plateout (% of release to containment) of Elemental Iodine Release to Containment	50
5.	Iodine Species (%)	
	Elemental	91
	Organic	4
	Particulate	5
C.	Data and Assumptions Used to Estimate Activity Released	
1.	Containment Leak Rate (%/day)	0.1
2.	Containment Release Duration (hours)	1.0
3.	Containment Leakage Unfiltered Release Fraction	1.0
4.	Containment Free Volume (ft^3) (Minimum value conservatively used)	1.69E+06
	a. Sprayed Fraction ($T \leq 305$ sec)	0.27
	b. Sprayed Fraction ($T > 305$ sec)	0.55
5.	Containment Spray Removal Coeffecients (hr^{-1})	
	a. Elemental	10
	b. Particulate	0.44
6.	Mixing Between Sprayed and Unsprayed Regions (unsprayed volumes, hr^{-1})	2
7.	Elemental Iodine Spray Decontamination Factor	100
8.	Spray Delay Time (sec)	
	a. Quench Spray	64
	b. Recirculation spray	305
9.	SLCRS Iodine Removal Efficiency* (%)	
	a. Elemental	99
	b. Organic	80
	c. Particulate	99
	d. Noble gas	0

TABLE 14.3-14a (Continued)

PARAMETERS USED IN EVALUATING THE
RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

D. Offsite Dose Evaluation Results

1. Radioactivity Release to the Environment

INSERT

2. Atmospheric Dispersion Factors (s/m^3)

Control Room	Table 2.2-12
Offsite	Table 2.2-11a
	Table 2.2-11b

Table 14.3-14a
Insert

ENVIRONMENTAL RELEASES* (Ci) DUE TO A LOSS-OF-COOLANT ACCIDENT

Nuclide	Containment	ESF 2 Hour	ESF 30 Day	RWST 2 Hour	RWST 30 Day	Total 2 Hr Release	Total 30 day Release
KR-83M	3.42E+02	2.48E+01	2.29E+02	6.69E-03	7.29E-02	3.67E+02	5.71E+02
KR-85	3.16E+01	2.13E-07	7.55E-07	6.32E-13	1.23E-11	3.16E+01	3.16E+01
KR-85M	7.29E+02	3.16E-02	3.16E-02	2.24E-07	5.84E-07	7.29E+02	7.29E+02
KR-87	1.22E+03	1.13E-02	1.13E-02	8.05E-08	1.47E-07	1.22E+03	1.22E+03
KR-88	1.94E+03	1.12E-03	1.12E-03	1.39E-08	3.20E-08	1.94E+03	1.94E+03
KR-89	2.09E+02	1.37E-03	1.37E-03	7.16E-17	7.17E-17	2.09E+02	2.09E+02
KR-90	3.74E+01	1.70E-06	1.70E-06	1.10E-59	1.10E-59	3.74E+01	3.74E+01
XE-131M	4.11E+01	2.06E-02	6.32E+02	6.58E-06	5.83E-02	4.11E+01	6.73E+02
XE-133	6.20E+03	8.18E+00	4.19E+03	2.61E-03	3.79E+00	6.21E+03	1.04E+04
XE-133M	1.95E+02	5.81E-01	2.98E+02	1.84E-04	2.12E-01	1.95E+02	4.93E+02
XE-135	2.06E+03	8.46E+01	4.75E+03	2.61E-02	2.85E+00	2.14E+03	6.81E+03
XE-135M	7.39E+02	5.38E+02	3.03E+04	7.38E-02	9.41E-01	1.28E+03	3.10E+04
XE-137	5.29E+02	4.06E-02	4.06E-02	5.21E-14	5.22E-14	5.29E+02	5.29E+02
XE-138	1.72E+03	3.72E-03	3.72E-03	7.05E-10	8.03E-10	1.72E+03	1.72E+03
BR-82	1.23E+00	1.42E-02	3.84E-01	1.35E-03	7.00E-02	1.24E+00	1.68E+00
BR-83	3.87E+01	3.74E-01	8.79E-01	2.89E-02	1.22E-01	3.91E+01	3.97E+01
BR-85	1.26E+01	1.07E-02	1.07E-02	3.60E-12	3.61E-12	1.26E+01	1.26E+01
I-129	1.24E-05	1.45E-07	5.42E-05	1.40E-08	2.51E-06	1.25E-05	6.91E-05
I-130	8.66E+00	9.75E-02	9.58E-01	9.03E-03	1.77E-01	8.77E+00	9.80E+00
I-131	3.45E+02	4.03E+00	5.42E+02	3.88E-01	4.81E+01	3.50E+02	9.36E+02
I-132	4.55E+02	4.37E+00	9.93E+00	3.34E-01	1.37E+00	4.60E+02	4.67E+02
I-133	7.03E+02	8.04E+00	1.31E+02	7.58E-01	2.52E+01	7.12E+02	8.59E+02
I-134	6.10E+02	4.46E+00	5.71E+00	2.21E-01	4.92E-01	6.14E+02	6.16E+02
I-135	6.49E+02	7.07E+00	3.89E+01	6.32E-01	6.36E+00	6.57E+02	6.94E+02
I-136	2.24E+01	4.79E-03	4.79E-03	4.30E-22	4.30E-22	2.24E+01	2.24E+01

* Ref. ~~12241/11700-UR(B)-480, Rev.0~~

TABLE 14.3-14a (Continued)

PARAMETERS USED IN EVALUATING THE
RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

3.	ECCS Leakage Assumptions	
	a. Leak Initiation time (s)	305
	b. Leak Rate, (gpm) (Doubled in the analysis)	0.05
	c. Fraction of Core Iodine in Sump Water	0.5
	d. Sump Water Volume (gal)	514,138
	e. Leakage Flash Fraction	0.1
4.	RWST Back-leakage Assumptions	
	a. Core iodine in sump water, (% of core)	50
	b. Sump water iodine release	100% elemental
	c. Beginning of back-leakage post accident, (s)	2531
	d. Beginning of RWST release post accident, (s)	5444
	e. End of RWST release post accident, (d)	30
	f. Rate of back-leakage to RWST, (gpm) (doubled in the analysis)	1
	g. Iodine release fraction from RWST	Time dependent
5.	Control Room Parameters	
	a. Control Room Volume (ft^3)	1.73E+05
	b. Control Room Normal Intake (cfm)	500
	c. CREBAPS Actuation by Containment Isolation Phase B (CIB) (sec)	0
	d. Control Room Pressurization Flow Rate (cfm)	
	Normal Intake (T=0-92 sec)	500
	Bottled Air (T=92 sec - 60 min)	600
	Pressurization Fan Manual Start Delay (T=60-81 min)	0
	Pressurization Fans (T=81 min - 30days)	600
	e. Control Room Unfiltered Infiltration (T=92 sec - 30 days)	10
	f. Intake Filter Efficiency (total %)	95
	g. Control Room Purge	None
6.	Radiological Exposures	
	a. Contributors	Containment Leakage ESF Leakage RWST Leakage
	(Control Room Only)	Containment Direct Radiation
	(Control Room Only)	RWST Direct Radiation
	b. Control Room dose	Table 11.3-7

TABLE 14.3-14a (Continued)

PARAMETERS USED IN EVALUATING THE
RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

c. 0-2 Hour Dose at EAB (rem)

i. Thyroid (CDE)	2.0E+02
ii. Whole Body (EDE)	4.3E+00

d. 30-Day Dose at LPZ (rem)

i. Thyroid (CDE)	1.4E+01
ii. Whole Body (EDE)	4.0E-01

*Considering data B.5, C.9 and an assumed charcoal filter bypass of 1% for organic and elemental iodine, an iodine removal total efficiency of 95% is conservatively used in the analysis.

APPENDIX 14B

RADIATION SOURCES AND DOSE CALCULATION METHODOLOGY

This appendix presents the quantities of radioactive isotopes present in the core, fuel rod gap and coolant~~, volume control tank, and gas surge tanks~~. A brief discussion of the derivation of these quantities and the dose calculation methodology used in the assessment of the radiological consequences of the postulated accidents is also included.

14B.1 ACTIVITIES IN THE CORE

Insert

~~The calculation of the core iodine fission product inventory is consistent with the inventories given in TID 14844[1]. The fission product inventories for other isotopes which are important from a health hazards point of view are calculated using the data from APED 5398[2]~~. These inventories are given in Table 14B-1. The isotopes included in Table 14B-1 are the isotopes controlling from considerations of inhalation dose, (iodines) and from external dose due to immersion (noble gases).

14B.2 ACTIVITIES IN THE FUEL ROD GAP

Insert

The computed gap activities (Table 14B-1) are based on buildup in the fuel from the fission process and diffusion to the fuel rod gap at rates dependent on the operating temperature. The temperature dependence is accounted for by determining the core fuel fraction operating within each of ten temperature regions (Figure 14D-7), each with a release rate to the gap dependent of the mean fuel temperature within that region. Since the temperature distribution changes during core life, the highest expected values are used. The temperature dependence of the diffusion coefficient, D', for Xe and Kr in UO_2, follows the Booth expression[5] which is based upon the Arrhenius equation:

$$D'(T) = D'(1673) * EXP((-E/R) * (1/T - 1/1673)) \quad (14B.2\text{-}1)$$

where:

D'(T) = diffusion coefficient at temperature T, per second

E = activation energy, 82 Kilocalories/mole

D' (1673) = diffusion coefficient at 1673 K = 1×10^{-11} per second

T = temperature in degrees Kelvin

R = gas constant, 1.99×10^{-3} Kilocalories/mole-K

The above expression is valid for temperatures above 1,100 C. Below 1,100 C fission gas release occurs mainly by two temperature independent phenomena, recoil and knock-out, and is predicted by using D' at 1,100 C. The value used for D' (1673 K) based on data at burnups greater than 10^{19} fissions per cc, accounts for possible fission gas release by other mechanisms as well as pellet cracking during irradiation.

The diffusion coefficient for iodine isotopes was conservatively assumed to be the same as for Xe and Kr. Toner and Scott[3] observed that iodine diffuses in uranium dioxide at about the same rate as Xe and Kr and has about the same activation energy. Data reported by Belle[4] indicates that iodine diffuses at slightly slower rates than Xe and Kr.

With the diffusion coefficient determined for the fuel temperature region of interest, the fraction of radioactive fission gas which crosses the fuel boundary into the fuel rod gap is found from:

$$f = 3 * \mathrm{SQRT}\ (D'/\lambda\ ) * ((\mathrm{COTH}\ (\lambda/D')) - (D'/\lambda\ )) \qquad (14B.2\text{-}2)$$

where:

f = fraction of a given radioactive fission gas in fuel rod gap

λ = fission gas decay constant, per second

D' = diffusion coefficient, per second

The above expression is the steady-state solution of the diffusion equation in spherical geometry as given by Booth[5].

Table 14B-1 lists the total core activities as well as activities present in the gap for each pertinent isotope obtained using the above equations and the fuel temperature distribution given in Table 14B-2.

The activities in the reactor coolant as well as in the volume control tank, pressurizer and waste gas decay tanks are given in Chapter 11, including the data on which the computation of these activities is based.

14B.3 FUEL HANDLING SOURCES

Insert

The inventory of fission products in a fuel assembly is dependent on the rating of the assembly. The parameters used for the calculations of the highest rated assembly to be discharged are summarized in Table 14B-3, while the associated activities at the time of shutdown are given in Table 14B-4.

The expected end-of-life temperature and power distributions were calculated by using the radial and axial power peaking factors of

Section 14B.1 Insert

Activities in the core were calculated using the computer code ORIGEN as described in NUREG/CR-0200, and using parameter values specific to the physical and chemical makeup of the fuel and to the reactor operation. Because uranium enrichments may change from cycle to cycle and these changes may cause an increase in certain nuclides, core radionuclide inventory is calculated for a minimum expected enrichment and again for a maximum expected enrichment. The assumed core inventory used in radiological analysis is composed of a selection of the maximum value for each nuclide for the range of expected enrichments.

Section 14B.2 Insert

Where fuel rod gap activities are used in radiological analyses, the gap fractions provided in Regulatory Guide 1.25 (and NUREG/CR-5009 for I-131) are assumed. These are 0.30 Kr-85, 0.12 I-131 and 0.10 for other iodines and noble gases total activity in the core.

Section 14B.3 Insert

The inventory of fission products available for release from a damaged fuel assembly is based on the total core inventory described in 14B.1, the fraction of activity in the fuel gap described in 14B.2 and the number of fuel assemblies in the core. When used in the fuel handling accident radiological analysis, this activity is also reduced to account for delay time specified in the facility Technical Specifications which limits the post criticality time duration for performing core alterations. Additionally, the PWR radial peaking factor of 1.65, as specified in Regulatory Guide 1.25, is applied to increase the activity content to ensure that the maximum power assembly is considered in the analysis.

1.27 and 1.37, respectively. The conservative end-of-life temperature and power distributions were calculated by using the same radial power peaking factor as in the expected case, but with a higher axial power peaking factor of 1.69. Thus, the temperature/volume distribution in the fuel is changed and the maximum temperature is increased (Table 14B-3), resulting in an increased fraction of fission products in the fuel-cladding gap (Table 14B-4).

14B.4 REACTOR COOLANT FISSION PRODUCT ACTIVITIES

The parameters used in the calculation of the reactor coolant fission product inventories, together with the pertinent information concerning the expected coolant cleanup flow rate and demineralizer effectiveness, are summarized in Table 14B-5, while the results of the calculations are presented in Table 14B-6. In these calculations, the defective fuel rods were assumed to be present at the initial core loading and were uniformly distributed throughout the core. Thus, the fission product escape rate coefficients were based upon the average fuel temperature. The calculations were performed ~~for the prevailing temperature upstream of the regenerative heat exchanger. The coolant density correction of 1.35 is made in order to obtain the correct activities at the downstream temperature.~~ with proper consideration of the various coolant densities in the purification stream.

The fission product activities in the reactor coolant during operation with small cladding defects (fuel rods containing pinholes or fine cracks) in 1 percent of the fuel rods were computed using the following differential equations:

For parent nuclides in the coolant:

INSERT

$$\frac{dN_{wi}}{dt} = D{*}V_i{*}N_{ci} - \left[l_i + R{*}E_i + \frac{B'}{(B_o - t{*}B')} \right] {*}N_{wi} \quad (14B.3\text{-}1)$$

for daughter nuclides in the coolant:

$$\frac{dN_{wj}}{dt} = D{*}V_j{*}N_{cj} - \left[l_j + R{*}E_j + \frac{B'}{(B_o - t{*}B')} \right] {*}N_{wj} + l_i{*}N_{wi} \quad (14B.3\text{-}2)$$

where:

N = population of nuclide

D = fraction of fuel rods having defective cladding

R = purification flow, coolant system volumes per second

B_o = initial boron concentration, ppm

Section 14B.4 INSERT

1. First order nuclides:

$$\frac{dN_{w_i}}{dt} = \frac{hn\gamma_i}{V_w} N_{c_i}(t) - \left(\lambda_i + \frac{PF_{EQ_i} Q_1}{V_w} + \beta_i \frac{T_1}{T_2}\right) N_{w_i}(t)$$

~~(11.1-4)~~
(14B.4.1)

2. Second order nuclides:

$$\frac{dN_{w_j}}{dt} = \frac{hn\gamma_j}{V_w} N_{c_j}(t) + \lambda_i f_{ij} N_{w_i}(t)$$

$$- \left(\lambda_j + \frac{PF_{EQ_j} Q_1}{V_w} + \beta_j \frac{T_1}{T_2}\right) N_{w_j}(t)$$

~~(11.1-5)~~
(14B.4.2)

3. Third order nuclides:

$$\frac{dN_{w_k}}{dt} = \frac{hn\gamma_k}{V_w} N_{c_k}(t) + \lambda_i f_{ik} N_{w_i}(t) + \lambda_j f_{jk} N_{w_j}(t)$$

$$- \left(\lambda_k + \frac{PF_{EQ_k} Q_1}{V_w} + \beta_k \frac{T_1}{T_2}\right) N_{w_k}(t)$$

~~(11.1-6)~~
14B.4.3

where: i, j, k = First, second and third order nuclide parameters
$N_{c_i}(t)$ = Population of nuclide i per fuel region at time t (atoms per region)
$N_{w_i}(t)$ = Concentration of nuclide i in the main coolant at time t (atoms/cm³)
h = Fraction of failed fuel
n = Total number of fuel regions
γ_i = Escape rate coefficient (second⁻¹)
$\beta_i = \sigma_{a_i} \phi_{th}$ = Burnup rate (seconds⁻¹)
V_w = Volume of main coolant (cm³)
λ_i = Decay constant for isotope i (second⁻¹)
PF_{EQ_i} = Equivalent purification factor (fraction) for i
T_1 = Coolant residence time in core (seconds)
T_2 = Coolant circulation time (seconds)
f_{ij} = Branching fraction from i to j

Q_1 = Equivalent flow into purification stream (cm^3/sec)

$$= Q_p \frac{\rho_p}{\rho_w}$$

Q_p = Actual flow entering purification stream at coolant loop density (cm^3/sec)

ρ_w = Density of the main coolant (g/cm^3)

ρ_p = Density of the purification flow (g/cm^3)

The equivalent purification factor includes the effect of mixed bed demmeralizers, periodically used cation demineralizer and noble gas stripping in the volume control tank.

B' = boron concentration reduction rate by feed and bleed, ppm per second

E = removal efficiency of purification cycle for nuclide

1 = radioactive decay constant

V = escape rate coefficient for diffusion into coolant

Subscript c refers to core

Subscript w refers to coolant

Subscript i refers to parent nuclide

Subscript j refers to daughter nuclide

14B.4.1 Reactor Coolant and Secondary System Equilibrium Activities

The reactor coolant activities tabulated in Table 14B-6 are based on 1.0% failed fuel. While these activities were the basis of most design basis radiological analyses performed during original licensing, current analysis practice is to base many of these analyses on the primary and secondary equilibrium activities that correspond to the specific activity limits for reactor coolant and secondary coolant provided in technical specifications. Table 14B-15 tabulates these equilibrium activities.

14B.4.2 Reactor Coolant System Iodine Spiking

Two cases of iodine spiking are considered in current design basis radiological analyses. The first is the pre-incident spike which occurs such that the technical specification maximum 21 μCi/gm dose equivalent I-131 concentrations are reached just prior to accident initiation. The second case is the iodine spike that is initiated by the accident transient (i.e., co-incident spike). For this case, regulatory practice requires analyses to include an iodine spike appearance rate that is 500 times the iodine appearance rate that would result in RCS equilibrium concentrations equal to the 0.35 μCi/gm technical specification. Table 14B-16 tabulates the pre-incident spike concentrations and the ~~co-incident iodine spike appearance rates.~~ methodology for calculating co-incident iodine spike release rates.

14B.6 ~~VOLUME CONTROL TANK ACTIVITY~~ INSERT

The 300 cu ft volume control tank is assumed to contain 120 cu ft of liquid and 180 cu ft of vapor. Table 14B-9 lists the activities in the volume control tank with clad defects in 1 percent of the fuel rods.

14B.7 ~~GAS SURGE TANK ACTIVITY~~ INSERT

Activities in the gas surge tank are calculated assuming that it is pressurized to 65 psig with gases stripped from the reactor coolant and passed through the gaseous waste charcoal delay subsystem which provides for 30-day holdup of xenon isotopes and 2-day holdup of krypton isotopes. The assumed gas stripping rate is 60 gpm. When the operating pressure of the surge tank is reached, the feed and bleed rates for the tank are equal. Table 14-10 lists the activity in the gas surge tank.

14B.8 DOSE MODELS FOR DESIGN BASIS ACCIDENT

This section identifies the models used to calculate the offsite radiological consequences from the release of radioactivity as a result of a loss of coolant accident.

14B.8.1 Assumptions

The following assumptions are basic to both the whole body dose due to immersion in a cloud of radioactivity and the thyroid dose due to inhalation of radioactivity:

a. All radioactive releases are treated as ground level releases regardless of the point of discharge.

b. No credit is taken for cloud depletion by ground deposition and/or radioactive decay during transport to the exclusion area boundary (EAB) or the outer boundary of the low population zone (LPZ).

c. No credit is taken for collection and filtration of the containment leakage by the Supplementary Leak Collection and Release System (SLCRS) in the design basis case (DBA). ~~Credit for 50% collection of containment leakage is taken in the realistic case.~~

14B.8.2 ~~Whole Body Gamma and Beta Skin Dose (Design Basis Case)~~

DELETED

The whole body dose delivered to a receptor is based on the assumption that the receptor is immersed in a radioactive cloud that is infinite in all directions above the ground, a semi - infinite cloud. The radioactive material concentration within this cloud is assumed to be uniformly dispersed throughout the volume and assumed to be equal to the centerline ground level concentration that would exist at the receptor location.

Section 14B-6 Insert

14B-6 WASTE GAS SYSTEM DECAY TANK RUPTURE ACTIVITY RELEASE

The radioactivity available for release following rupture of a waste gas system decay tank was determined using the guidance of Regulatory Guide 1.24 and NUREG-0800 Branch Technical Position ETSB 11-5, that is, the noble gas is removed from the RCS as quickly as possible and collected in decay tanks. Additionally, the RCS activity concentration is conservatively based on equilibrium achieved while operating with 1% fuel failures, a condition prohibited by facility Technical Specifications. When a tank is full, it immediately ruptures and releases it's entire contents directly to the environment.

Hold-up in the charcoal delay bed is considered in the radiological analysis. Because of the long hold-up time for Xe, there are no Xe isotopes assumed to be present when the tank rupture occurs. Decay of Kr isotopes during activity collection and hold-up is considered in the analysis. The collection time duration is based on the time required to fill a decay tank. During this period RCS activity is conservatively assumed to remain constant, with no credit taken for activity depletion.

Section 14B-7 Insert

14B-7 WASTE GAS SYSTEM LINE RUPTURE ACTIVITY RELEASE

The radioactivity available for release following rupture of a waste gas process system component was determined using the guidance of Regulatory Guide 1.24 and NUREG-0800 Branch Technical Position ETSB 11-5. The radioactivity release consists of two source components: 1) instantaneous release of a portion of the activity collected on the charcoal delay bed following 24 hours of degassing at power plus the time period required to remove 99% of the gas following reactor shutdown and, 2) one hour continuous flow release via the rupture assuming a constant activity removal rate (no source depletion). Additionally, the RCS activity concentration is conservatively based on equilibrium achieved while operating with 1% fuel failures, a condition prohibited by facility Technical Specifications. The dose resulting from each of these two source components is summed to provide a conservative, bounding radiological consequence determination for this accident.

The whole body dose conservatively incorporates the beta skin dose. The model assumes that all of the energy of the emissions from the nuclides present in the plume are deposited in the receptor. The doses calculated from this energy absorption are corrected for the differences in absorption by air and by tissue.[6]

The whole body dose commitment due to gamma and beta is as follows:

D(WB) = XOQ x SUM[Q(i) x DCFN(i)]

Where:

D(WB) = Total whole body dose (rem)

XOQ = X/Q, Atmospheric dispersion during period of interest and at receptor location

Q(i) = Total activity of isotope, i, released during time period (Ci)

DCFN(i) = Dose conversion factor for noble gas isotope, i, (rem-cu.M/Ci-sec)

14B.8.3 ~~Thyroid Dose Commitments~~ DELETED

The thyroid dose commitment due to the uptake of radioiodine was determined using the data and methodology of ICRP-II[7] as implemented in TID-14844[1]. The dose conversion factors for the adult thyroid dose commitment due to the uptake of radioiodine are tabulated in Table 14B-11.

The total thyroid dose commitment due to the radioiodine uptake is as follows:

D(T) = XOQ x BR x SUM[Q(i) x DCFT(i)]

Where:

D(T) = Total thyroid dose commitment (rem)

BR = Breathing Rate (cu.M/sec) Table 14B-11

DCFT(i) = Dose conversion factor for radioiodine isotope, i, (rem/Ci)

14B.8.4 ~~Dose Models for Realistic Case~~ DELETED

Under ~~the~~ realistic evaluation of the DBA LOCA, ~~the whole~~ body dose model is ~~based~~ on deep-deposition ~~rather~~ than surface energy deposition, including ~~consideration~~ of the attenuation by body tissues[6]. The ~~whole~~ body dose ~~model~~ also includes the contributions from the radioiodines in the release.

14B.8.5 Updated Dose Calculation Models

Commencing with analyses performed in 1995, the whole body-gamma dose, beta skin dose, and thyroid dose commitments ~~described in Section 14B.8.1 - 14B.8.4~~ have been calculated ~~using the dose~~ quantities described in this section.

Effective Dose Equivalent (EDE) as described in ICRP-26[11]. Replaces the traditional whole body gamma dose quantity. Like the whole body dose it replaces, the EDE model assumes that the receptor is immersed in a semi-infinite cloud. The EDE model estimates the dose to each organ in the body due to radiation from this cloud, applies a weighting factor to each organ dose, and sums the weighted doses to obtain the EDE.

$$D_{EDE} = \chi / Q \times \sum_i (Q_i \times C_{EDE_i})$$

where:

D_{EDE} = Effective Dose Equivalent (EDE)

Q_i = Activity of nuclide i released

χ/Q = Atmospheric dispersion factor

C_{EDE_i} = Dose conversion factor for nuclide i (DOE/EH-0070, 1988)[15]

For the control room dose analyses, the EDE is corrected to account for the finite volume of the control room using the method of Murphy-Campe[13].

$$D_{EDE_{CR}} = \chi / Q \times \frac{V^{0.338}}{1173} \times \sum_i (Q_i \times C_{EDE_i})$$

where:

$D_{EDE_{CR}}$ = Effective Dose Equivalent (EDE) for control room

V = volume of control room, ft^3

Skin Dose Equivalent (skin DE) as described in ICRP-26. Replaces the traditional beta skin dose quantity. Assumes that the receptor is immersed in a semi-infinite cloud.

$$D_{SKIN} = \chi / Q \times \sum_i (Q_i \times C_{SKIN_i})$$

where:

D_{SKIN} = Skin Dose Equivalent (skin DE)

Q_i = Activity of nuclide i released

χ/Q = Atmospheric dispersion factor

C_{SKIN_i} = Dose conversion factor for nuclide i (DOE/EH-0070, 1988)

Thyroid Committed Dose Equivalent (thyroid CDE) as described in ICRP-26 and ICRP-30[12]. Replaces the traditional thyroid dose quantity based on the critical organ model of ICRP-2[7] used in TID14844[1].

$$D_{CDE_{thy}} = \chi / Q \times \sum_i (Q_i \times C_{CDE_i} \times BR)$$

where:

$D_{CDE_{thy}}$ = Thyroid Committed Dose Equivalent (CDE)

Q_i = Activity of nuclide i released

χ/Q = Atmospheric dispersion factor

BR = Breathing rate
- 3.47E-4 m^3/sec, 0-8 hours
- 1.75E-4 m^3/sec, 8-24 hours
- 2.32E-4 m^3/sec, >24 hours
- 3.47E-4 m^3/sec, 0-30 day control room analysis

C_{CDE_i} = Dose conversion factor for nuclide i (USEPA FGR11, 1988)[16]

14B.9 CONTAINMENT LEAKAGE MODEL - DBA CASE[8]

This section describes the model used to estimate the quantity of radionuclides released to the environment by leakage from the containment building, using design basis assumptions. ~~The realistic case leakage model is described in Section 14B.10.~~

14B.9.1 Radioiodine

Figure 14B-1 illustrates, schematically, the leakage model. The containment free volume is assumed to consist of two regions: a sprayed region and an unsprayed region. The processes acting simultaneously on the activity in the unsprayed region are:

a. Radioactive decay
b. Leakage from containment
c. Thermally induced exchange with sprayed region

For the sprayed region, scavenging of iodine by chemical sprays is added to the list above. This scavenging is effective on elemental and particulate species of iodine. The chemical removal continues until the ~~maximum spray decontamination factor (DF) is reached. This DF is based on the iodine concentration in the recirculation spray and the iodine partitioning factor.~~ containment becomes subatmospheric (1 hour after LOCA). At that time the spray decontamination factors (DF) are within the maximum allowed values.

The transport of radioiodines in and between the two regions is modeled as a first order linear process. The activity in the sprayed region (subscript "s") and in the unsprayed region (subscript "u") is determined as a function of the removal processes identified above, Z, and the exchange between regions, E, by the following differential expression:

$$dA(s)/dt = E(u)xA(u) - Z(s)xA(s)$$

$$dA(u)/dt = E(s)xA(s) - Z(u)xA(u)$$

The activity released to the environment, Q, as a function of leakage, L, is given by:

$$Q = L\int_{t=0}^{T} A(S) + L\int_{t=0}^{T} A(U)$$

The solutions to these expressions are tabulated in Table 14B-12. The model was applied to the time period ~~0-304 seconds, defined as the period of only quench spray, and with slight modification, to the period~~ 305-3600 seconds starting with the initiation of recirculation spray, and terminating when the containment becomes subatmospheric.

14B.9.2 Noble Gases

Noble gases are not affected by the containment sprays, and therefore, the two region model used for the radioiodines is replaced with a model which encompasses the entire containment free volume in a single region. The noble gas activity released to the containment, A, as a function of time, leakage, L, and radioactive decay constant, LAMBDA is given by:

$$dA/dt = -LxA - LAMBDAxA$$

The activity released to the environment, Q, is given as a function of leakage, L, and time:

$$Q = L\int_{t=0}^{T} A$$

These expressions are solved by the following:

$$Q = \{[LxA(0)] / [L+LAMBDA]\}x[1 - EXP(-(L+LAMBDA)xT)]$$

Where:

A(0) = Initial noble gas activity released to the containment at t=0, Ci

L = Leakage constant, ~~sec x 10^{-1}~~ sec^{-1}

LAMBDA = Radioactive decay, ~~sec x 10^{-1}~~ sec^{-1}

In addition, the release of noble gases generated by the decay of parent halogens/noble gases in containment is included in the calculation model.

14B.9.3 ESF Leakage

The ESF leakage model assumes that there is leakage of containment sump water to areas outside of the containment via leaks in the recirculation piping. Noble gases are not assumed to be present in the sump water and therefore are not considered in the ESF leakage model. Ten percent (10%) of the radioiodine postulated to leak from the recirculation piping is assumed to go airborne and be available for release to the environment.

All of the potential ESF leakage occurs within the areas maintained at a slight vacuum by the Supplementary Leak Collection and Release System (SLCRS), and is therefore collected. This activity is released via the release point located on the containment dome after filtration by the main filter banks. ~~The expressions of Section 14.B.9.2 were used with the initial activity reduced to account for the efficiency of the main filter banks and the 10% release fraction.~~

14B.10 ~~CONTAINMENT LEAKAGE MODEL - REALISTIC CASE[9]~~ DELETED

This section describes the model used to estimate the quantity of radionuclides released to the environment by leakage only from the containment building, using realistic case assumptions. The primary difference between the design basis and the realistic case is that credit is taken for 50% collection of the containment leakage by the Supplementary Leak Collection and Release System, in the realistic case.

14B.10.1 SLCRS Efficiency

The fraction of the Technical Specification limit for total containment leakage assumed to terminate in the areas served by the Supplemental Leak Collection and Release System (SLCRS) is conservatively set at 50 percent. This percentage was selected at the construction permit stage for BVPS-1 and was based on conservative engineering judgement since no containment had yet been built with a SLCRS. Since that time, several containment leak rate tests have been conducted on other PWR reactor containments. Calculations show that the assumption is conservative and only a maximum of 0.01 percent per day of leakage of the containment could be uncollected. The SLCRS, therefore, is at least 90 percent effective in collecting potential leakage following a DBA.

An analysis of containment leak test data from Type A, B and C tests for Main Yankee, Connecticut Yankee, Surry Unit 1[10] and Surry Unit 2[10] power stations was made. Type A tests measure the primary containment overall integrated leakage rate; Type B tests measure leakage rates across primary containment penetrations; and Type C tests measure containment isolation valve leakage rates. A summary of these tests is presented in Table 14B-13.

The boundary of the Supplementary Leak Collection and Release System includes all penetrations and valves that undergo Type B and C testing with the single exception of the equipment hatch which is sealed after the plant construction phase.

Therefore, the difference between the two leak rates, or the overall leak rate, as determined in a Type A test, and the leak rate across penetrations and valves, as determined by the combination of Type B and C test results, represents the rate of leakage that is uncollected and unfiltered, if the leakage around the equipment hatch is, as the data indicate, negligible.

The unfiltered leak rate assumed in offsite dose calculations is 50 percent of the Technical Specification limit. The unfiltered leak rates for all four nuclear power plants cited are well within 50 percent of the Technical Specification limits (38 percent or less).

Furthermore, since the containment structure is completely lined by a 3/8 inch shell with joints that are welded and then overlapped by test channels welded to the shell, the potential for leakage through the containment shell is negligible and invariant with time.

The leakage of iodine by diffusion through the liner is negligible. Solid iodine is shipped commercially in polyethylene bags in a fiberpack container with no noticeable loss by diffusion. It is expected that the diffusion of iodine through steel will be zero. Methyl iodine is not expected to diffuse either. A BNWL report notes: "Losses to the reaction vessels surfaces in the absence of hydrazine are negligible."

To evaluate the SLCRS an analysis was made of each potential leak path. The sensitivity of tests actually used on Surry Power Station were used. For welds, the sensitivity of the H-10 leak detector probe was used (minimum threshold sensitivity of 1.8×10^{-5} cc per second measuring 1/4 inch of weld). Each 1/4 inch of containment liner was assumed to leak at 1.8×10^{-5} cc per second. Testing of the containment liner weld leakage is conducted in accordance with S & W nondestructive test procedure No. 261, entitled "Quality Control Halogen Leak Testing." This procedure contains calibration procedures for all apparatus used for the test. Gross leakage testing is performed prior to the actual halogen tests. The gross leak test requires the application of a specifically prepared bubble solution which will form bubbles when air is passed through it. Following the application of the bubble solution, the welded test channel is pressurized with air to 50 psig and held for 15 minutes soak time prior to examination. Any leaks detected by this method are repaired and retested prior to the performance of the halogen test. The halogen leak test procedure requires the attachment of a special test apparatus to the welded containment. This apparatus includes a vacuum pump Freon (R-22) tank, and pressure gages, piping and valves necessary to conduct the test. The apparatus is connected to the test containment by flexible metal hose and fittings. The vacuum pump

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is started and the test containment is evacuated to 1.0 to 0.5 psia. The vacuum pump is then isolated and the Freon gas isolation valve is fully open to measure a reading of 50 psig. The isolation valve is then closed. With the leak detector calibrated in accordance with the procedure, 100 percent of the welds are scanned with the probe held as close as practicable traveling at a maximum rate of 1 inch per second.

The probe is checked using a calibrated source before and after each section of containment is tested or 20 ft, whichever is smaller. The total maximum potential uncollected containment leakage through liner welds using this ultra conservative analysis is 80.12 ft^3/day (for BVPS-1). The results of the evaluation are presented in Table 14B-10. This analysis provides justification for the 50 percent conservatively effective SLCRS on BVPS-1.

14B.10.2 <u>Release Model</u>

A single region model was used in the realistic case. This model used expressions similar to those in Section 14B.9.2, expanded to include removal by sprays for radioiodines. Figure 14B-1 illustrates the release pathways.

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18. INSERT

19. INSERT

20. INSERT

7. SWEC Calculation Package 12241/117700 UR(B)-478, Design Reactor Core Inventory (3.96% Initial Enrichment) and Associated Equilibrium Primary and Secondary Coolant Activities for BVPS.

8. SWEC Calculation Package 12241/117700 UR(B)-479, Radiological Source Terms for Accident Analyses – Composite Equilibrium Reactor Core Inventory (3.6 – 5% Initial Enrichment) and the Associated Design Primary and Secondary Coolant Activities for BVPS.

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18. SWEC Calculation Package ERS-SNW-92-009, Iodine Release from the Beaver Valley Unit 1 Refueling Water Storage Tank

19. DLC Calculation Package ERS-JTL-99-014, Safety Analysis of the Radiological Consequences of a Waste Gas System Rupture DBA at Unit 1, Control Room, EAB and LPZ Doses.

20. SWEC Calculation Package 12241/117700 UR(B)-480, Radiological Dose Consequences at the EAB, LPZ, and in the Control Room due to a Postulated LOCA at Beaver Valley Power Station, Unit 1

TABLE 14B-1

CORE AND GAP ~~ACTIVITIES~~ FRACTIONS
BASED ON FULL POWER OPERATION FOR ~~650~~ 1500 DAYS

FULL POWER: ~~2766~~ 2705 MWt

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Isotope	Curies in Core ($\times 10^7$)	Percent of Core Activity In Gap	Curies in Gap ($\times 10^5$)
I-131	6.83 =8.196	0.851	5.81
I-132	10.4	0.0933	0.967
I-133	15.3	0.280	4.29
I-134	17.9	0.0577	1.03
I-13~~4~~5	13.9	0.159	2.21
Xe-131m	0.0518	1.04	0.0538
Xe-133	15.8	0.690	10.9
Xe-133m	0.400	0.452	0.181
Xe-135	4.31	0.187	0.804
Xe-135m	4.24	0.0314	0.133
Xe-138	13.9	0.0327	0.455
Kr-83m	1.27	0.0853	0.109
Kr-85	0.0775 .2325	17.1	1.33
Kr-85m	3.06	0.128	0.393
Kr-87	5.89	0.0692	0.408
Kr-88	8.38	0.102	0.858
Kr-89	10.8	0.0142	0.154

Table 14B-1 Insert

NUCLIDE	CORE ACTIVITY (Ci)	CORE FRACTION In Gap
Kr-83m	9.04E+06	0.10
Kr-85m	1.88E+07	0.10
Kr-85	7.59E+05	0.30
Kr-87	3.78E+07	0.10
Kr-88	5.25E+07	0.10
Kr-89	6.57E+07	0.10
Kr-90*	7.05E+07	0.10
Xe-131m	9.87E+05	0.10
Xe-133m	4.69E+06	0.10
Xe-133	1.49E+08	0.10
Xe-135m	3.12E+07	0.10
Xe-135	4.85E+07	0.10
Xe-137	1.35E+08	0.10
Xe-138	1.28E+08	0.10
Br-82*	2.85E+05	0.10
Br-83*	8.96E+06	0.10
Br-85*	1.87E+07	0.10
I-130*	1.86E+06	0.10
I-131	7.24E+07	0.10
I-132	1.06E+08	0.12
I-133	1.49E+08	0.10
I-134	1.65E+08	0.10
I-135	1.41E+08	0.10
I-136*	6.57E+07	0.10

*Because of their short half-lives and/or insignificant impact on accident dose, these isotopes are considered only in the LOCA radiological analysis.

TABLE 14B-3

NUCLEAR CHARACTERISTICS OF HIGHEST RATED DISCHARGED ASSEMBLY

Reactor Power	Expected Case	Conservative Case
Rating MWt	2766	2766
102% Rating	2821	2821
Number of Assemblies	157	157
Array	15 x 15	15 x 15
Core Average Assembly Power		
At 102% Rating, MWt	17.97	17.97
Discharged Assembly (Highest Power)		
Axial Peak to Average Ratio	1.37	1.69
Peak Power, Kw/ft	12.8	15.8
Maximum Temperature, F	3500	3900
Radial Peak to Average, Ratio	1.27	1.27

Temperature/Power Distribution Local Temperature, F	% Fuel Volume	Power, MWt in Volume	% Fuel Volume	Power, MWt in Volume
>3900	0	0	0	0
3700 - 3900	0	0	1.33	0.3
3500 - 3700	0	0	2.67	0.61
3300 - 3500	1.33	0.30	4.00	0.91
3100 - 3300	2.67	0.61	5.33	1.22
2900 - 3100	4.00	0.91	6.67	1.52
2700 - 2900	5.33	1.22	8.00	1.83
2500 - 2700	6.67	1.52	9.33	2.13
<2500	80.00	18.26	62.67	14.30
	100.00	22.82	100.00	22.82

TABLE 14B-4

ACTIVITIES IN HIGHEST RATED DISCHARGED ASSEMBLY CURIES AT TIME OF REACTOR SHUTDOWN

Isotope	Total Curies	Fraction in Fuel-Cladding Gap	Fuel-Cladding Gap Curies	Fraction in Fuel-Cladding Gap	Fuel-Cladding Gap Curies
I-131	5.75×10^{5}	0.0166	9.55×10^{3}	0.0376	2.16×10^{4}
I-132	8.68×10^{5}	0.0018	1.56×10^{3}	0.0042	3.65×10^{3}
I-133	1.28×10^{6}	0.0064	8.19×10^{3}	0.0137	1.75×10^{4}
I-134	1.50×10^{6}	0.0012	1.80×10^{3}	0.0026	3.90×10^{3}
I-135	1.16×10^{6}	0.0034	3.95×10^{3}	0.0078	9.05×10^{3}
Kr-85m	2.53×10^{5}	0.0050	1.26×10^{3}	0.0108	2.73×10^{3}
Kr-85	8.25×10^{3}	0.243	2.00×10^{3}	0.353	2.91×10^{3}
Kr-87	4.86×10^{5}	0.0014	6.80×10^{2}	0.0031	1.50×10^{3}
Kr-88	6.91×10^{5}	0.0048	3.32×10^{3}	0.0108	7.46×10^{3}
Xe-133m	2.63×10^{4}	0.0091	2.39×10^{2}	0.206	5.41×10^{2}
Xe-133	1.30×10^{6}	0.0137	1.78×10^{4}	0.0310	4.03×10^{4}
Xe-135	3.53×10^{5}	0.0037	1.26×10^{3}	1.0084	2.96×10^{3}

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TABLE 14B-5

PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT ACTIVITIES

1.	Core thermal power, maximum calculated, MWt	~~2,766~~ 2705
2.	Fraction of fuel containing clad defects	0.01
3.	Reactor coolant liquid volume, including pressurizer, ft^3	~~9,387~~ 7,835
4.	Reactor coolant ~~average~~ density (lb/ft^3) ~~temperature, F~~	~~577~~ 44.13
5.	~~Purification~~ flow rate (~~normal~~) minimum, gpm	60
6.	~~Effective~~ cation demineralizer flow, gpm	6.0
7.	Volume control tank volumes	
	a. Vapor, ft^3	~~180~~ 183
	b. Liquid, ft^3	~~120~~ 136
8.	Fission product escape rate coefficients:	
	a. Noble gas isotopes, sec^{-1}	6.5×10^{-8}
	b. Br, Rb, I and Cs isotopes, sec^{-1}	1.3×10^{-8}
	c. Te isotopes, sec^{-1}	1.0×10^{-9}
	d. Mo isotopes, sec^{-1}	2.0×10^{-9}
	e. Sr and Ba isotopes, sec^{-1}	1.0×10^{-11}
	f. Y, La, Ce, Pr, Se, Zr, Nb, Tc, Ru, Rh, Sn, Sb, Nd, Pm, Sm isotopes, sec^{-1}	1.6×10^{-12}
9.	Mixed bed demineralizer decontamination factors:	
	a. Noble gases and Cs, Rb ~~134, 136, 137, Y-90, 91 and Mo-99~~	1.0
	b. All other isotopes	10.0
10.	Cation bed demineralizer decontamination factor for Cs-134, Rb ~~136, 137, Y-90, 91 and Mo-99~~	10.0
	Noble gases, halogens, others	1.0

TABLE 14B-5 (CONT'D)

PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT ACTIVITIES

11. Volume control tank noble gas stripping fraction ~~(closed system):~~

Isotope	Stripping Fraction
Kr-85	2.3×10^{-5}
Kr-85m	2.7×10^{-1}
Kr-87	6.0×10^{-1}
Kr-88	4.3×10^{-1}
Xe-133	1.6×10^{-2}
Xe-133m	3.7×10^{-2}
Xe-135	1.8×10^{-1}
Xe-135m	8.0×10^{-1}
Xe-138	1.0

INSERT

Table 14B-5 (continued) INSERT

Kr-83m	7.7E-01
Kr-85m	5.8E-01
Kr-85	6.5E-05
Kr-87	8.3E-01
Kr-88	6.8E-01
Kr-89	9.9E-01
Xe-131m	1.3E-02
Xe-133m	6.9E-02
Xe-133	3.0E-02
Xe-135m	9.4E-01
Xe-135	3.0E-01
Xe-137	9.8E-01
Xe-138	9.4E-01

12. Thermal Neutron Flux in Fuel Region (n/s-cm^2)	4.64E+13
13. Thermal Neutron Flux in Coolant Region in Core (n/s-cm^2)	5.34E+13
14. Circulation time of Primary Coolant (sec)	12.2
15. Residence Time of Coolant In Core (sec)	0.808
16. Density of Purification Flow (lb/ft^3)	61.29
17. Volume Control Tank	
Water Volume (ft^3)	136
Vapor Volume (ft^3)	183
Temperature (°F)	115
Pressure (psig)	20
Purge Rate (ft^3/min)	0.0
18. Reactor Operation Time (days)	1,150

TABLE 14B-6

REACTOR COOLANT EQUILIBRIUM FISSION AND CORROSION PRODUCT ACTIVITIES
(Based on Parameters Given in Table 14B-5)

Isotope	Activity (μCi/cc)	Isotope	Activity (μCi/cc)
Br-84	2.9×10^{-2}	Cs-137	0.91
Rb-88	2.3	Cs-138	0.66
Rb-89	6.9×10^{-2}	Ba-140	3.0×10^{-3}
Sr-89	2.9×10^{-3}	La-140	1.1×10^{-3}
Sr-90	6.9×10^{-5}	Ce-144	2.5×10^{-4}
Sr-91	1.4×10^{-3}	Pr-144	2.5×10^{-4}
Sr-92	5.2×10^{-4}	Kr-85	7.8
Y-90	8.4×10^{-5}	Kr-85m	1.4
Y-91	4.7×10^{-4}	Kr-87	0.86
Y-92	5.1×10^{-4}	Kr-88	2.3
Zr-95	4.9×10^{-4}	Xe-133	19.2
Nb-95	4.9×10^{-4}	Xe-133m	2.2
Mo-99	2.3	Xe-135	2.3
I-131	1.8	Xe-135m	0.78
I-132	0.63	Xe-138	0.48
I-133	2.9	Mn-54	5.6×10^{-4}
I-134	0.39	Mn-56	2.1×10^{-2}
I-135	1.5	Co-58	1.8×10^{-2}
Te-132	0.19	Co-60	5.4×10^{-4}
Te-134	2.14×10^{-2}	Fe-59	7.5×10^{-4}
		Cr-51	6.8×10^{-4}

INSERT

Table 14B-6 Insert

DESIGN REACTOR COOLANT NOBLE GAS AND IODINE ACTIVITIES
(Based on Parameters Given in Table 14B-5)

NUCLIDE	ACTIVITY (μCi/gram)	NUCLIDE	ACTIVITY (μCi/gram)
Kr-83m	3.95E-01	Br-84	3.62E-02
Kr-85m	1.38E+00	Rb-88	2.69E+00
Kr-85	1.45E+02	Rb-89	1.54E-01
Kr-87	9.26E-01	Sr-89	3.38E-03
Kr-88	2.95E+00	Sr-90	1.98E-04
Kr-89	7.51E-02	Sr-91	1.40E-03
		Sr-92	9.95E-04
Xe-131m	4.70E+00	Y-90	5.41E-05
Xe-133m	3.93E+00	Y-91	4.60E-04
Xe-133	2.91E+02	Y-92	8.49E-04
Xe-135m	8.99E-01	Zr-95	5.94E-04
Xe-135	9.67E+00	Nb-95	6.01E-04
Xe-137	1.85E-01	Mo-99	7.11E-01
Xe-138	6.35E-01	Te-132	2.78E-01
		Te-134	2.85E-02
I-131	2.69E+00	Cs-137	3.27E+00
I-132	1.06E+00	Cs-138	9.67E-01
I-133	4.03E+00	Ba-140	3.85E-03
I-134	5.94E-01	La-140	1.31E-03
I-135	2.32E+00	Ce-144	4.34E-04
		Pr-144	4.37E-04
Mn-54	4.80E-03		
Co-58	1.38E-02		
Co-60	1.59E-03		
Fe-59	9.00E-04		
Cr-51	9.30E-03		

TABLE 14B-9

INSERT ~~VOLUME CONTROL TANK ACTIVITIES~~

Isotope	Activities (Curies)
Kr-83m	1.41
Kr-85m	6.35
Kr-85	6.05×10^{3}
Kr-87	3.96
Kr-88	1.06×10^{1}
Xe-131m	1.39
Xe-133m	1.05×10^{1}
Xe-133	3.85×10^{2}
Xe-135m	3.59
Xe-135	1.08×10^{1}
I-131	Negligible
I-132	Negligible
I-133	Negligible
I-134	Negligible
I-135	Negligible

delete incorporate by reference

TABLE 14B-10

~~INSERT~~ ~~GAS SURGE TANK ACTIVITY~~

Assumptions: Tank at operating pressure - 80 psia

Clad defects in 1 percent of fuel rods

Operation at ~~2,766~~ MWt

Tank contains worst gaseous activity inventory associated with gas stripping of reactor coolant letdown to compensate for fuel burnup.

Reactor coolant system volume is 9,387 ft^3.

Isotope	Total Activity Curies
Kr-83m	4.01×10^{-7}
Kr-85m	6.55×10^{-2}
Kr-85	1.69×10^{3}
Kr-87	2.33×10^{-10}
Kr-88	1.11×10^{-3}
Xe-131m	3.84×10^{-1}
Xe-133m	5.30×10^{-2}
Xe-133	8.70×10^{1}
Xe-135m	5.60×10^{-3}
Xe-135	7.40×10^{-3}
I-131	Negligible
I-132	Negligible
I-133	Negligible
I-134	Negligible
I-135	Negligible

delete incorporate by reference

TABLE 14B-11

THYROID DOSE CONVERSION FACTORS[1]

Nuclide	Sv/Bq	~~rem/Ci~~
I^{131}	2.92E-07	~~1.48×10^6~~
I^{132}	1.79E-09	~~5.35×10^4~~
I^{133}	4.86E-08	~~4.00×10^5~~
I^{134}	2.88E-10	~~2.50×10^4~~
I^{135}	8.46 E-09	~~1.24×10^5~~

BREATHING RATES[2]

Time Period	m^3/sec
0-8 hours	3.47 10^{-4}
8-24 hours	1.75 10^{-4}
24-duration	2.32 10^{-4}

(1) EPA 520, Federal Guidance Report No. 11
~~TID 14844, Calculation of Distance Factors for Power and Test Reactor Sites~~ (conversion factor, 3.7 E+09 mrem-Bq/Sv-μCi)

(2) U. S. Nuclear Regulatory Commission, Regulatory Guide 1.4

TABLE 14B-13

SUMMARY OF TYPE A, B AND C CONTAINMENT LEAKAGE TESTS

Plant	Maximum Tech Spec Leakage (% Day)	Type A Test (%/Day)	Type B + C Tests (%/Day)		B + C / A (%)		A- (B + C) (%/Day)	A- (B + C) (Tech Spec %)
Maine Yankee	0.25 (.1875)	0.0541 ± 0.0187	0.0112 ± 0.00003		21		0.0429 ± 0.0187	17
		(1)	(2)	(3)	(2)	(3)	(2)	
Connecticut Yankee	0.25 (.1875)	0.0426 ± 0.0038	0.0423	0.0719	99	169	0.0003 ± 0.0038	0.1
Surry 1	0.1 (.075)	0.052 ± 0.004	0.014		27		0.038 + 0.004	38
Surry 2	0.1 (.075)	0.022 ± 0.007	0.002		10		0.020 ± 0.007	20
Ginna	0.1 (.0731)	0.0219 ± 0.0168	None					

NOTES: (1) Test conducted during 1967

(2) Test conducted during 1967

(3) Test conducted during 1970

DELETE

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TABLE 14B-14

PARAMETERS USED IN CALCULATING FRACTION OF LEAKAGE COLLECTED BY THE SLCRS

Structure	Area
Containment	
Containment Dome	2.49×10^4 ft^2
Containment Wall	4.82×10^4 ft^2
Containment Floor	1.24×10^4 ft^2
Containment Wall below Waterproof Membrane	1.54×10^4 ft^2
Total Containment Area above Waterproof Membrane	5.78×10^4 ft^2
Contiguous Areas Serviced by SLCRS	
Safeguards Area	4.41×10^3 ft^2
Cable Vault Area	4.82×10^3 ft^2
Total Contiguous Area	9.23×10^3 ft^2

$$\text{Ratio} = \frac{\text{Total Contiguous Area}}{\text{Total potential containment weld leakage area}} = \frac{9.23 \times 10^3 \text{ ft}^2}{5.78 \times 10^4 \text{ ft}^2}$$

Ratio = 15.97% of potential weld leakage is into areas serviced by the SLCRS.

Potential containment leakage from liner weld = 105.72 ft^3/day

0.1597 x 105.72 = 16.88 ft^3/day leakage into contiguous areas serviced by SLCRS.

Leakage Sources (Uncollected)	Leakage
Uncollected type C piping leakage	less than 90 ft^3/day by test
Equipment Hatch	99.86 ft^3/day
Dome Hatch (welded closed)	4.72×10^{-3} ft^3/day (negligible)

TABLE 14B-14, (Cont'd)

PARAMETERS USED IN CALCULATING FRACTION OF LEAKAGE COLLECTED BY THE SLCRS

Leakage Sources (Uncollected)	Leakage
Maximum potential uncollected containment liner weld leakage to atmosphere	88.84 ft^3/day
Total maximum potential uncollected leakage	278.70 ft^3/day

RESULTS

Maximum potential uncollected leakage	0.016 percent of containment volume per day
Effectiveness of SLCRS (percent of Technical specification leakage collected)	greater than 84%

DELETE

TABLE 14B-15

PRIMARY AND SECONDARY EQUILIBRIUM ACTIVITIES
Corresponding to 0.35 μCi/gm Dose Equivalent I-131 in RCS(1)

Nuclide	RCS μCi/gm	SG Liquid μCi/gm	SG Steam μCi/gm
Kr-83m	4.02E-02 ~~4.65E-02~~		~~4.53E-07~~ 6.45E-07
Kr-85m	1.41E-01 ~~2.27E-01~~		~~2.21E-06~~ 1.89E-06
Kr-85	1.48E+01 ~~1.20E+00~~		~~1.17E-05~~ 2.00E-04
Kr-87	9.44E-02 ~~1.30E-01~~		~~1.26E-06~~ 1.27E-06
Kr-88	2.64E-01 ~~3.46E-01~~		~~3.37E-06~~ 3.55E-06
Kr-89	7.65E-03 ~~1.09E-02~~		~~1.07E-07~~ 1.03E-07
Xe-131m	4.79E-01 ~~1.17E-02~~		~~1.14E-07~~ 6.45E-0[illegible]
Xe-133m	4.00E-01 ~~3.33E-01~~		~~3.25E-06~~ 5.42E-06
Xe-133	2.97E+01 ~~2.84E+00~~		~~2.77E-05~~ 4.00E-04
Xe-135m	9.16E-02 ~~1.18E-01~~		~~1.15E-06~~ 8.61E-06
Xe-135	9.85E-01 ~~3.48E-01~~		~~3.40E-06~~ 1.44E-05
Xe-137	1.89E-02 ~~1.77E-02~~		~~1.72E-07~~ 2.55E-07
Xe-138	6.47E-02 ~~7.28E-02~~		~~7.10E-07~~ 8.72E-07
I-131	2.74E-01 ~~2.72E-01~~	8.38E-02 ~~8.11E-02~~	~~8.11E-04~~ 8.38E-04
I-132	1.08E-01 ~~9.48E-02~~	1.35E-02 ~~1.44E-02~~	~~1.44E-04~~ 1.35E-04
I-133	4.11E-01 ~~4.24E-01~~	9.13E-02 ~~1.06E-01~~	~~1.06E-03~~ 9.13E-04
I-134	6.05E-02 ~~5.93E-02~~	1.76E-03 ~~2.54E-03~~	~~2.54E-05~~ 1.76E-05
I-135	2.36E-01 ~~2.28E-01~~	3.19E-02 ~~3.88E-02~~	~~3.88E-04~~ 3.19E-04

Notes:

1. Steam generator liquid based on 0.1 μCi/gm D.E. I-131 in steam generator.

2. Ref: ~~ERS-SFL-96-012 r1, 1996~~
   ERS-AJL-99-007

TABLE 14B-16

RCS IODINE SPIKE ACTIVITIES

Pre-incident Concentration, µCi/gm
(Corresponding to 21 µCi/gm d.e. I-131)

I-131	1.64E+01 ~~16.3~~
I-132	6.48E+00 ~~5.69~~
I-133	2.46E+01 ~~25.4~~
I-134	3.63E+00 ~~3.56~~
I-135	1.42E+01 ~~13.7~~

~~Co-incident Iodine Spike Appearance Rates, Ci/sec~~
~~(500x Equilibrium Rate for 0.35 µCi/gm d.e. I-131)~~

~~I-131~~	~~0.49~~
~~I-132~~	~~0.92~~
~~I-133~~	~~1.10~~
~~I-134~~	~~1.34~~
~~I-135~~	~~1.02~~

Notes:

1. Ref: ~~ERS-SFL-96-012 r1, 1996~~
ERS-AJL-99-007

INSERT

Table 14B-16 INSERT

PARAMETERS AND ASSUMPTIONS AND MODEL USED
FOR CALCULATING IODINE RELEASE RATES INTO REACTOR COOLANT
DUE TO A CONCURRENT IODINE SPIKE

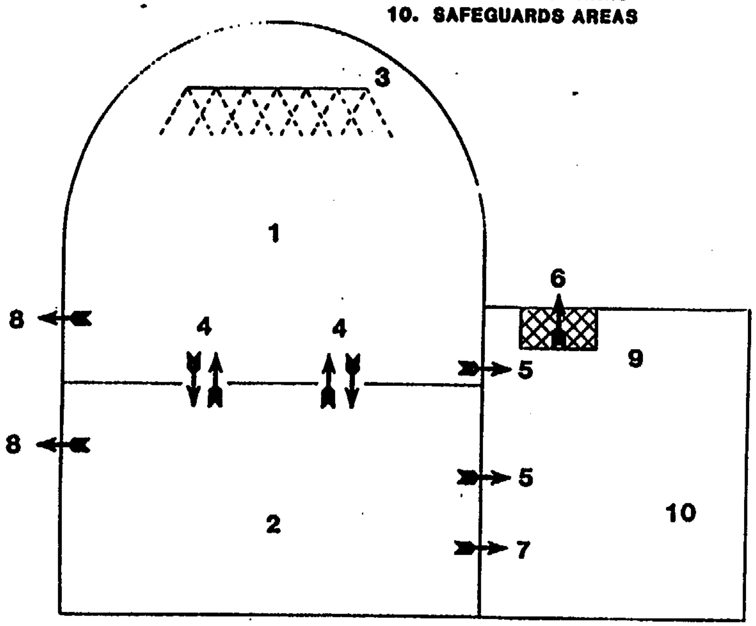
Thyroid dose conversion factors	<table><tr><th>Nuclide</th><th>mrem/µCi</th></tr><tr><td>I-131</td><td>1.08E+03</td></tr><tr><td>I-132</td><td>6.44E+00</td></tr><tr><td>I-133</td><td>1.80E+02</td></tr><tr><td>I-134</td><td>1.07E+00</td></tr><tr><td>I-135</td><td>3.13E+01</td></tr></table>
Nuclide decay constants (λ_r)	<table><tr><th>Nuclide</th><th>second^{-1}</th></tr><tr><td>I-131</td><td>9.9783E-07</td></tr><tr><td>I-132</td><td>8.3713E-05</td></tr><tr><td>I-133</td><td>9.2568E-06</td></tr><tr><td>I-134</td><td>2.1963E-04</td></tr><tr><td>I-135</td><td>2.9129E-05</td></tr></table>
Reactor coolant system leakage (L)	Technical Specification maximum allowable values
Reactor coolant system mass (M)	Limiting value specific to the accident
Letdown purification removal (E)	1.0
Letdown purification flow rate (F)	120 gpm
Technical Specification equilibrium concentrations (EQ)	Table 14B-15
Formula for iodine loss constant	$\lambda_{total} = (F*E/M) + (L/M) + \lambda_r$
Concurrent iodine spike release rate (RR)	$RR = EQ * M * \lambda_{total}$

NOTES:

Formulas for iodine release rates from EPRI Report, "Review of Iodine Spike Data from PWR Power Plants in Relation to SGTR with MSLB, TR-103680)"
This Table is applicable to design basis accident analyses performed subsequent to December 1998.

REV. 2 (1/84)

1. SPRAYED REGION (Vs)
2. UNSPRAYED REGION (Vu)
3. CONTAINMENT SPRAYS
4. EXCHANGE BETWEEN REGIONS (E)
5. COLLECTED LEAKAGE
6. SLCRS RELEASE TO ENVIRONMENT
7. ESF LEAKAGE DURING RECIRCULATION
8. UNCOLLECTED LEAKAGE
9. MAIN FILTER BANKS
10. SAFEGUARDS AREAS



NOTES:

(1) FOR DBA MODEL, ALL CONTAINMENT LEAKAGE IS UNCOLLECTED
(2) WHEN RECIRCULATION SPRAYS INITIATE, CONTAINMENT IS ONLY 55 ~~89%~~ SPRAYED
(3) PATHWAY 7 COMMENCES AT t=305 SEC. LEAKAGE INCREASES AT t=1500 SEC WITH INITIATION OF LOW HEAD SAFETY INJECTION IN RECIRC MODE.

FIGURE 14B-1
LOSS-OF-COOLANT ACCIDENT RELEASE PATHWAYS
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

ATTACHMENT A-2

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 151
UFSAR Update for Revised Radiation Dose Calculations

The following is a list of the affected pages:

Affected UFSAR Pages:	Revise Pages 6.4-5, 6.4-8, 11.1-2, 12.2-1, 12.3-15, 15.0-11a, 15.0-12, 15.0-14, 15.0-15, 15.1-22, 15.1-23, 15.2-13, 15.2-22, 15.3-10, 15.3-11, 15.3-12, 15.4-29, 15.4-44, 15.4-45, 15.4-46, 15.6-4, 15.6-5, 15.6-9, 15.6-10f, 15.6-10g, 15.6-10h, 15.6-20, 15.6-22, 15.6-25, 15.6-26, 15.7-2, 15.7-6, 15A-1, 15A-5, 15A-7
Affected UFSAR Tables:	Revise Tables 1.8-1, 6.5-2, 11.1-1, 11.1-2, 11.1-3, 11.1-6, 11.1-7, 12.2-1, 12.2-3, 15.0-7a, 15.0-7b, 15.0-8b, 15.0-9a, 15.0-10a, 15.0-12, 15.0-13, 15.0-14, 15.1-3, 15.2-2, 15.3-3, 15.4-3, 15.6-2, 15.6-5b, 15.6-11, 15.6-12, 15.7-1, 15.7-2
	Delete Tables 6.4-2, 15.1-4, 15.1-5, 15.2-3, 15.4-4, 15.6-3, 15.6-6, 15.6-7, 15.6-13, 15.7-3
	Add Table 15A-1a

TABLE 1.8-1 (Cont)

RG No. 1.23, Rev. 0
UFSAR Reference Section 2.3.3

ONSITE METEOROLOGICAL PROGRAMS (FEBRUARY 17, 1972)

Onsite meteorological programs for Beaver Valley Power Station - Unit 2 will follow the guidance of this regulatory guide.

RG No. 1.24, Rev. 0
UFSAR Reference Section 15.7.1.3

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED WATER REACTOR RADIOACTIVE GAS STORAGE TANK FAILURE (MARCH 23, 1972)

Beaver Valley Power Station - Unit 2 evaluation of the potential radiological consequences of a pressurized water reactor radioactive gas storage tank rupture meets the intent of this regulatory guide. The following alternatives were considered prudent:

Paragraph C.1.a

In recognition of specific plant equipment arrangements for gaseous waste handling, the system component producing the worst environmental impact was identified and additional conservatism was appropriately applied.

Paragraph C.2

Atmospheric diffusion (X/Q) values were calculated using the latest approved techniques which are provided in Regulatory Guide 1.145.

BVPS takes some alternatives to Section C.3 dose calculation methodology.

RG No. 1.25, Rev. 0
UFSAR Reference Section 15.4.7

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 (MARCH 23, 1972)

The assumptions used for evaluating the potential radiological consequences of a fuel handling accident in the fuel handling and storage facility at Beaver Valley Power Station - Unit 2 meet the intent of this regulatory guide with the following alternative:

Paragraph C.1.j specifies iodine removal efficiencies of 90 percent for inorganic species and 70 percent for organic species. However, the efficiencies used are those given in Table 2 of Regulatory Guide 1.52 (that is, 95 percent for both organic and inorganic species if 2-inch charcoal bed depth is provided; 99 percent if 4 or more inches of charcoal bed depth is provided) since these represent more realistic values.

Paragraph C.1.d specifies that the analysis should be performed assuming 10% of the total radioactive iodine in the rods of the time of the accident. However, the iodine percentages used are 12% I-131 and 10% of the other iodine nuclides. This is in keeping with NUREG-5009, as referenced by the USNRC in the safety evaluation report for license amendment 12.

BVPS takes some alternatives to Section C.3 dose calculation methodology.

<u>RG No. 1.26, Rev. 3</u>
UFSAR Reference Section 3.2.2

<u>QUALITY GROUP CLASSIFICATIONS AND STANDARDS FOR WATER-, STEAM-, AND RADIOACTIVE-WASTE-CONTAINING COMPONENTS OF NUCLEAR POWER PLANTS (FEBRUARY 1976)</u>

Quality group classifications and standards for water-, steam-, and radioactive-waste-containing components of Beaver Valley Power Station - Unit 2 meet the intent of Regulatory Guide 1.26 with the following alternatives:

1. The safety class terminology of ANSI N18.2 and ANSI 18.2a-1975 is used instead of the quality group terminology. Thus, the terms Safety Class 1, Safety Class 2, Safety Class 3, and Non-nuclear Safety (NNS) Class are used instead of Quality Groups A, B, C, and D, respectively, and are consistent with present nuclear industry practice.

2. Paragraph NB-7153 of the ASME Section III Code requires that there be no valves between a code safety valve and its relief point unless special interlocks prevent shutoff without other protection capacity. Therefore, as an alternative to Paragraphs C.1.e and C.2.c, a single safety valve designed, manufactured, and tested in accordance with ASME III Division 1 is considered acceptable as the boundary between the reactor coolant pressure boundary and a lower safety class or NNS class line.

3. Portions of the emergency diesel generator cooling water system, considered by the vendor to be parts of the engine (as distinguished from auxiliary support systems), were built to the manufacturer's standards rather than ASME III. These are identified in Table 3.2-1 and Section 9.5.5. The components used are of high quality, proven by experience, and were designed, fabricated, erected, and tested under the vendor's Quality Assurance Program which meets the requirements of 10CFR50, Appendix B. Similar equipment has been accepted by the NRC for other nuclear power plant applications.

TABLE 1.8-1 (Cont)

8. Paragraph C.1 (IEEE Standard 384-1974 - Section 3, ISOLATION DEVICES)

   The use of two independent Class 1E overcurrent devices (breakers or fuses) in series, provides electrical separation between Class 1E and non-Class 1E circuits under the following conditions:

   1. Coordination is provided between each of the two series devices and the main Class 1E feeder breaker.

   2. These devices are included in a surveillance program during normal plant operation.

RG No. 1.76, Rev. 0
UFSAR Reference Section 3.3.2.1

DESIGN BASIS TORNADO FOR NUCLEAR POWER PLANTS (APRIL 1974)

All applicable Beaver Valley Power Station - Unit 2 structures, systems, or components important to safety will be designed to withstand, or will be enclosed in structures which will withstand, the six descriptive parameters given in Table I of Regulatory Guide 1.76 for the Region I location.

RG No. 1.77, Rev. 0
UFSAR Reference Section 15.4.8

ASSUMPTIONS USED FOR EVALUATING A CONTROL ROD EJECTION ACCIDENT FOR PRESSURIZED WATER REACTORS (MAY 1974)

The guidance of this regulatory guide was followed for the Beaver Valley Power Station - Unit 2 analysis of a control rod ejection accident as provided in Section 15.4.8 with the following alternatives:

The rod ejection accident is considered a faulted condition as stated in ANSI N18.2 rather than as an emergency condition as implied by Paragraph C.2.

The meteorological model as described in Regulatory Guide 1.145, Revision 0, was used in the atmospheric diffusion analysis since it supersedes the meteorological portion of Regulatory Guide 1.77.

Dose conversion methodology is based on that described in ICRP Reports 26 and 30.

6.4.2.5 Shielding Design

The design of the control room envelope includes adequate radiation shielding and ventilation control to maintain acceptable radiation levels in the main control room under accident conditions, as discussed in Section 12.3.2.

In accordance with General Design Criterion 19, personnel exposure is limited to 5 Rem whole body, or its equivalent to any part of the body, for the duration of any accident postulated in Chapter 15.

The postulated accident radioactivity sources inside and outside the control room envelope are stated in Chapter 15.

The effects of the LOCA and all other design basis accidents are evaluated to determine the doses which the main control room personnel might receive at BVPS-2. The LOCA analysis is based on:

1. Major reactor coolant system (RCS) pipe rupture (LOCA) at BVPS-2 or a

2. Major RCS pipe rupture (LOCA) at BVPS-1.

For purposes of analysis, it is assumed that each accident occurs with a seismic event and loss of offsite power. Accidents are not postulated to occur simultaneously.

The main control room personnel are potentially exposed to sources from several locations following the LOCA. The sources considered for the design of control room shielding include: 1) the containment building (direct and skyshine dose), 2) the external cloud (from containment and emergency core cooling system (ECCS) leakage), 3) sources in adjacent buildings, and 4) iodine collection on the main control room intake filter.

The containment building is considered as one of the sources of radiation used for main control room shielding design due to its location and the large amount of activity contained within its bounds. A significant fraction of the containment free air volume is located above grade and its dose contribution is evaluated to determine the main control room 30 day whole body gamma dose. The containment atmosphere source is based on 100 percent of the noble gas and 50 percent of the halogen core inventory. The containment atmosphere LOCA specific activities for 12 time intervals are presented in Table 12.2-13.

, used for control room shielding design,

The external cloud is due to containment leakage during the first hour of the LOCA plus the ECCS leakage over 30 days. The containment leakage contribution to the cloud source is a function of the containment airborne inventory available for leakage and the of a HEPA filter and carbon adsorber with effective iodine removal efficiency of 95 percent. These emergency supply filtration units and associated air handling equipment are designed to Seismic Category I and Safety Class 3 requirements.

The evaluation of radiation exposure to personnel in the main control room envelope examined the contribution from the four LOCA sources defined in Section 6.4.2.5. In addition, the inhalation dose from inleakage into the main control room of radionuclides in the external cloud was also examined for each of the DBAs considered in Chapter 15. Dose calculations are based on the source terms and pertinent parameters defined in Chapter 15 for each DBA, and the flux-to-dose conversion factors given in ~~Table 6.4-2.~~ Appendix 15A.

~~The limiting design basis accident for the main control room personnel whole body gamma and beta skin dose is the LOCA. The small line break outside containment (SLB) is the limiting DBA for the thyroid dose.~~

Exposure from inhalation is principally attributable to airborne radioactivity in the main control room envelope due to:

1. Intake prior to main control room isolation,
2. Inleakage during main control room isolation, or
3. Post-isolation ventilation intake.

The CIB signal isolates the control room almost immediately after a LOCA. ~~For DBAs that do not cause a CIB signal, control room isolation is initiated by a high radiation signal from redundant Category 1 area monitors centrally located in the BVPS-2 control room except for the main steamline break (MSLB) and the locked rotor accident (LRA) which do not initiate a high radiation signal.~~ For the MSLB ~~and the LRA,~~ manual operator action by t=30 min post-accident is needed to maintain habitability.

The analyses consider a conservative selection of parameters to calculate the thyroid dose. Ventilation intake prior to control room isolation and an assumed 10 cfm unfiltered inleakage are the main contributors to the thyroid dose. The maximum normal ventilation intake rate of 500 cfm (for both BVPS-1 and BVPS-2 intakes) prior to isolation and an appropriate clean up rate post-isolation are used to maximize the dose estimate. The analysis also assumes coincident loss of offsite power.

Radiation doses to a control room operator due to the various postulated DBA's are summarized in Table 15.0-13.

TABLE 6.4-2

FLUX-TO-DOSE CONVERSION FACTORS*

DELETE

Energy (Mev/photon)	mRem/hr (Mev/cm^2 sec)
0.4	2.46×10^{-3}
0.8	2.10×10^{-3}
1.3	1.83×10^{-3}
1.7	1.69×10^{-3}
2.2	1.55×10^{-3}
2.5	1.49×10^{-3}
3.5	1.32×10^{-3}
6.15	1.08×10^{-3}

NOTE:

*Based on ANSI/ANS-6.1.1-1977 (N666) (American Nuclear Society 1977).

TABLE 6.5-2 (Cont)

Particulate Iodine Removal Coefficient

Parameter	Units		Value
E	Dimensionless		0.0015
d	microns		1,000
V	ft^3		1.356×10^6
F	gpm	Upper header Lower header	593 1,824
h	ft	Upper header Lower header	104 78.5
λ_p	hr^{-1}		~~0.83~~ 0.825

t = Time (seconds)

F = Fission rate (fissions per second in fuel region)

α_i = Fission yield for isotope i (atoms per fission)

λ_i = Decay constant for isotope i ($seconds^{-1}$)

γ_i = Escape rate coefficient ($seconds^{-1}$)

$\beta_i = \sigma_{a_i} \phi_{th}$ = Burnup rate ($seconds^{-1}$)

f_{ij} = Branching fraction from i to j

h = Fraction of failed fuel.

The program has a basic library of 167 nuclides with a capability of 200 nuclides. Library data include decay scheme information, production information, and decay gamma spectra in seven energy groups. Input data include time intervals, initial source inventory in the fuel, neutron flux, and power level. The program output describes the system analyzed, as well as the operating history, the activities, and associated gamma spectral information for the input time interval.

The calculation of the core iodine fission product inventory is consistent with the inventories given by DiNunno (et al 1962). The fission product inventories are calculated using the appropriate data from Meek and Rider (1974), Lederer (et al 1968), Nucleonics Handbook of Nuclear Research and Technology (1963), Goldberg (et al 1966), and Perkins (1963). The core iodine and noble gas fission product inventories are presented in Table 11.1-1 based on continuous operation of the unit at 2,766 MWt. ~~These inventories are used in~~ <Insert> ~~the evaluation of the postulated accidents in Chapter 15.~~

Fuel assembly source terms for shielding design are calculated using the ACTIVITY 2 computer code and are presented in Chapter 12.

Fuel element heat loadings and stresses, as well as fuel operating experience, are presented in Chapter 4.

11.1.2 Radionuclide Inventory in Fuel Element Gap

The gap activity is that fraction of the gaseous activity in the core that diffuses to the fuel gaps. In accordance with the guidance provided in Regulatory Guides 1.25 and 1.77, the noble gas and iodine inventory in the fuel gap region is conservatively assumed to be 10 percent ~~(30 percent for Kr-85 for the fuel handling accident analysis) for the accident analysis described in Chapter 15.~~ Table 11.1-1 presents the core gap activities.

<Insert>

<insert Section 11.1.1>

These inventories are not used in Chapter 15 design accident analysis. For this, a new core inventory (Table 15.0-7b) and primary and secondary system design activity concentrations (Table 15.0-8b) were developed in 1999 using updated, conservative facility design and operating parameter values as analysis inputs. The values and parameters maintained in Chapter 11 provide the historical basis for facility design and remain adequate for analyses not related to Chapter 15 accidents.

<insert Section 11.1.2>

Core gap activity values used in Chapter 15 accident analyses are provided in Table 15.0-7b. These were developed in 1999 using updated, conservative facility design and operating parameter values as analysis inputs. In addition, a Kr-85 gap fraction of 0.30 was used in accordance with Regulatory Guide 25, and an I-131 gap fraction of 0.12 was used in accordance with NUREG/CR-5009. The values and parameters maintained in Chapter 11 provide the historical basis for facility design and remain adequate for analyses not related to Chapter 15 accidents.

TABLE 11.1-1

IODINE AND NOBLE GAS INVENTORY IN REACTOR CORE AND FUEL ROD GAPS*

Isotope	Core (Ci)	Fraction of Core Activity in Gap	Fuel Rod Gap Activity (Ci)
I-131	6.9×10^7	0.1	6.9×10^6
I-132	9.9×10^7	0.1	9.9×10^6
I-133	1.6×10^8	0.1	1.6×10^7
I-134	1.8×10^8	0.1	1.8×10^7
I-135	1.4×10^8	0.1	1.4×10^7
Kr-83m	1.2×10^7	0.1	1.2×10^6
Kr-85m	3.0×10^7	0.1	3.0×10^6
Kr-85	6.8×10^5	0.1**	6.8×10^4
Kr-87	5.9×10^7	0.1	5.9×10^6
Kr-88	8.3×10^7	0.1	8.3×10^6
Kr-89	1.1×10^8	0.1	1.1×10^7
Xe-131m	4.2×10^5	0.1	4.2×10^4
Xe-133m	3.7×10^6	0.1	3.7×10^5
Xe-133	1.6×10^8	0.1	1.6×10^7
Xe-135m	4.2×10^7	0.1	4.2×10^6
Xe-135	4.1×10^7	0.1	4.1×10^6
Xe-137	1.4×10^8	0.1	1.4×10^7
Xe-138	1.4×10^8	0.1	1.4×10^7

NOTES:

*Based on 650 days of operation at 2,766 MWt, ~~for use in Chapter 15.~~

~~**In fuel handling accident analysis, the fraction used for Kr-85 is 0.3, in accordance with Regulatory Guide 1.25.~~

Refer to Table 15.0-7b for values used in Chapter 15 accident analyses.

TABLE 11.1-2 (Cont)

Nuclide	Design (μCi/g)	Expected (μCi/g)
Other		
Activation Products		
H-3	3.5	1.0
Subtotal	3.5	1.0
Total (excluding H-3)	7.5×10^{1}	4.1
Total (including H-3)	7.9×10^{1}	5.1

Refer to Table 15.0-8b for values used in Chapter 15 accident analyses.

TABLE 11.1-3 (Cont)

Parameter	Value
Thermal neutron flux (n/cm^2-sec)	4.16x10^{13}
Operating time (650 EFPD) (hr)	15,600
Coolant cycle time (sec)	11.3
Coolant in core time (sec)	0.9
Degasification factor	1.0
Secondary side equilibrium time (hr)	1.0 x 10^4
Volume control tank volumes	
Vapor (ft^3)	175
Liquid (ft^3)	125
Total secondary liquid per steam generator (lb)	9.93x10^4
Total steam generator blowdown flow (lb/hr) (15 gpm/steam generator is the minimum flow corresponding to secondary side design activities).	2.24x10^4
Fraction removed from steam generator blowdown (purification factors for design and expected cases)	
Noble gases	0.0
Halogens	0.855
Cs, Rb	0.500
Others	0.899
Tritium	0.0
Ratio of condensate Demineralizer flow rate to total steam flow rate	0.733

Refer to Table 15.0-8a for parameter values used to determine fission and activation product activities for Chapter 15 accident analyses.

TABLE 11.1-6 (Cont)

Nuclide	Design* (μCi/g)	Expected** (μCi/g)
Te-134	$4.5x10^{-6}$	
Cs-134	$5.9x10^{-4}$	$7.3x10^{-6}$
Cs-136	$3.3x10^{-4}$	$3.7x10^{-6}$
Cs-137	$3.4x10^{-3}$	$5.3x10^{-6}$
Cs-138	$1.1x10^{-4}$	
Ba-137m	$3.1x10^{-3}$	$1.6x10^{-6}$
Ba-140	$5.2x10^{-6}$	$3.2x10^{-8}$
La-140	$2.2x10^{-6}$	$2.1x10^{-8}$
Ce-141	$8.4x10^{-7}$	$1.3x10^{-8}$
Ce-143	$5.4x10^{-7}$	$3.0x10^{-9}$
Ce-144	$5.7x10^{-7}$	$6.5x10^{-9}$
Pr-143	$8.1x10^{-7}$	$6.5x10^{-9}$
Pr-144	$5.7x10^{-7}$	$3.6x10^{-9}$
Np-239	$4.5x10^{-6}$	$1.9x10^{-7}$
H-3	$4.4x10^{-3}$	$1.0x10^{-3}$
Total (excluding H-3)	$2.2x10^{-2}$	$8.7x10^{-5}$
Total (including H-3)	$2.6x10^{-2}$	$1.1x10^{-3}$

NOTES:

*Based on 1,188 lb/day primary to secondary leak rate.
**Based on 100 lb/day primary to secondary leak rate.

Refer to Table 15.0-8b for concentrations used in Chapter 15 accident analyses.

TABLE 11.1-7 (Cont)

Nuclide	Design* (μCi/g)	Expected** (μCi/g)
Tc-99m	$9.7x10^{-6}$	$7.5x10^{-9}$
Ru-103	$1.0x10^{-9}$	$6.5x10^{-12}$
Ru-106	$9.7x10^{-11}$	$1.3x10^{-12}$
Rh-103m	$1.0x10^{-9}$	$3.9x10^{-12}$
Rh-106	$9.7x10^{-11}$	$7.0x10^{-13}$
Te-125m	$2.9x10^{-10}$	$3.3x10^{-12}$
Te-127m	$5.9x10^{-9}$	$3.3x10^{-11}$
Te-127	$4.1x10^{-9}$	$8.0x10^{-11}$
Te-129m	$1.2x10^{-7}$	$1.9x10^{-10}$
Te-129	$1.1x10^{-7}$	$1.2x10^{-10}$
Te-131m	$6.0x10^{-8}$	$3.0x10^{-10}$
Te-131	$1.4x10^{-8}$	$3.7x10^{-11}$
Te-132	$7.7x10^{-7}$	$3.2x10^{-9}$
Te-134	$1.1x10^{-8}$	-
Cs-134	$1.5x10^{-6}$	$7.3x10^{-9}$
Cs-136	$8.3x10^{-7}$	$3.7x10^{-9}$
Cs-137	$8.4x10^{-6}$	$5.3x10^{-9}$
Ba-137m	$7.8x10^{-6}$	$1.6x10^{-9}$
Ba-140	$1.3x10^{-8}$	$3.2x10^{-11}$
La-140	$5.5x10^{-9}$	$2.1x10^{-11}$
Ce-141	$2.1x10^{-9}$	$1.3x10^{-11}$
Ce-143	$1.4x10^{-9}$	$3.0x10^{-12}$
Ce-144	$1.4x10^{-9}$	$6.5x10^{-12}$
Pr-143	$2.0x10^{-9}$	$6.5x10^{-12}$
Pr-144	$1.4x10^{-9}$	$3.6x10^{-12}$
Np-239	$1.1x10^{-8}$	$1.9x10^{-10}$
H-3	$4.4x10^{-3}$	$1.0x10^{-3}$
Total (excluding H-3)	$3.1x10^{-4}$	$1.4x10^{-6}$
Total (including H-3)	$4.7x10^{-3}$	$1.0x10^{-3}$

NOTES:

*Based on 1,188 lb/day primary to secondary leak rate.
**Based on 100 lb/per day primary to secondary leak rate.

Refer to Table 15.0-8b for concentrations used in Chapter 15 accident analyses.

12.2 RADIATION SOURCES

12.2.1 Contained Sources

The radiation source terms used for shield design analyses are based upon full power operation, shutdown conditions, and accident conditions. Normal operation source locations are shown on Figures 12.3-1 through 12.3-5.

The sources of radioactivity contained in the streams of the various radioactive waste management systems are the nuclides generated in the reactor core, the activation of nuclides in the reactor coolant system (RCS) and the air surrounding the reactor vessel. Tables 11.1-3 and 12.2-1 present the principal parameters which are used in establishing normal operation design radiation source inventories. Dimensions and locations for equipment containing all major sources in normal operation are given in Table 12.2-2.

The reactor core source description is similar to that given in Section 4.1.1 of Topical Report RP-8A (SWEC 1975).

The activity of a spent fuel assembly is calculated using appropriate fission yields, decay constants, and thermal neutron cross-sections. Isotopic inventories are based on full power operation for 650 days. The core inventory at shutdown and 100 hours after shutdown is given in Table 12.2-3. The corresponding source strength in MeV/sec, assuming a radial peaking factor of 1.65 for one fuel assembly, is given in Table 12.2-4. The primary and secondary side system inventories are given in Tables 11.1-2, 11.1-6, and 11.1-7. Based on selected data in Topical Report RP-8A (SWEC 1975), source strengths for various auxiliary systems are presented in Tables 12.2-5 through 12.2-9.

Sources used in the evaluation of equipment qualification and post-accident access doses are determined using NUREG-0737 (USNRC 1980) values for fractional releases of the core inventory which is given in Table 12.2-3. The specific activities for the contained accident sources are given for various times from T=0 to T=6 months after the accident in Tables 12.2-10 through 12.2-14.

A discussion of systems which contain major sources of radiation follows.

12.2.1.1 Sources for Normal Full Power Operation Shield Design

The main sources of activity during normal full power operation are N-16 from coolant activation processes, fission products from fuel clad defects, and corrosion and activation products.

Each BVPS-2 system is shielded according to the amount of activity present and adjacent zoning and access criteria. The systems which are the major contributors to radiation levels in the plant are:

Radiation source terms used in Chapter 15 dose analyses, and the bases for those source terms may differ from the shielding design values herein.

TABLE 12.2-1

SELECTED PARAMETERS USED IN CALCULATION OF DESIGN RADIATION SOURCE INVENTORIES

Characteristic	Parameters
Power level (MWt)	2,766
Failed fuel fraction *	0.01
Primary to secondary leak rate (gpd)	144
Reactor operating time (days)	650

NOTE:

* Failed fuel fraction of 0.01 indicates the failure of the fuel which produces 1 percent of the reactor power.

Refer to Chapter 15 for parameters used in accident radiological consequence analyses for control room operator and offsite doses.

TABLE 12.2-3 (Cont)

Nuclide	0-Hour Decay (μCi)	100-Hour Decay (μCi)
Pm-147	$0.17x10^{14}$	$0.17x10^{14}$
Pm-149	$0.24x10^{14}$	$0.67x10^{13}$
Pm-151	$0.97x10^{13}$	$0.85x10^{12}$
Sm-151	$0.77x10^{10}$	$0.80x10^{10}$
Sm-153	$0.36x10^{13}$	$0.81x10^{12}$

NOTE:

*Less than 1.0 microcurie.

Refer to Chapter 15 for values used in accident radiological consequence analyses for control room operator and offsite doses.

dose in excess of 5 Rem (or equivalent organ dose). This dose includes the 30 day direct radiation dose from activity inside the containment (assuming no cleanup), the external radiation contribution from the postulated radioactive plume leaking from the containment and engineered safety features (ESF) system leakage. Leakage from the containment is assumed until the ESF system returns the containment to subatmospheric pressure and terminates the leakage as discussed in Chapter 15. The radiation sources in the containment during the assumed DBA are calculated by the methods described in Regulatory Guide 1.4 and are discussed in Section 15.6.5. These sources are assumed to be evenly distributed throughout the containment. The containment is then treated as a volume source and the 30 day direct radiation dose inside the 24-inch thick concrete wall of the main control room is calculated using typical shielding computational techniques.

The released activity is assumed to leak from the containment structure at a rate of 0.1 percent of the contained volume per day for sixty minutes, and is converted into a semi-infinite volume source surrounding the main control room.

Sources such as containment and ESF system leakage which contribute to personnel doses from the intake of the outside atmosphere into the control room, are described in Section 6.4.2.5.

provided in Table 15.0-13.

The integrated whole body ~~gamma~~ dose from all sources described in Section 6.4.2.5 is ~~calculated to be 0.7 Rem in 30 days~~. This dose is well below the dose criterion of 5 Rem. Thus, the main control room walls, which must be a minimum of 24 inches thick for tornado missile protection, provide more than adequate shielding from radiation. ~~Control room dose values are summarized in Table 15.0-13.~~

Special consideration has been given to the design of penetrations and structural details of the main control room so as to establish an acceptable condition of leaktightness.

The control room air-conditioning systems are installed within the spaces served and are designed to provide uninterrupted service under accident conditions. Upon a containment isolation Phase B signal or a high radiation signal from the control room redundant area monitors, the normal replenishment air and exhaust systems are isolated automatically from the main control room by tight closures in the ductwork. Breathing-quality compressed air is supplied from high pressure storage bottles to maintain a small positive outflow from the main control room for a period exceeding the containment leakage period. The main control computer, and mechanical rooms are included in the control room envelope.

The radiation levels in the main control room are measured by installed area and airborne monitors to verify safe operating conditions.

The computer code ORIGENS was used to calculate core radioactivity inventory. ORIGENS is distributed by the Radiation Safety Information Computational Center, Oak Ridge, TN. This code is readily available, and is a commonly used code for this purpose. The code input parameters used for this calculation are provided in Table 15.0-7a. The revised core radioactivity inventory values for the radionuclides used in design basis radiological accident radiological consequence analyses are provided in Table 15.0-7b.

15.0.9.2 Activities in the Fuel Pellet Clad Gap

For accident analysis, the core gap activities are based on the guidance provided in Regulatory Guides 1.25 and 1.77. The noble gas and iodine inventory in the fuel gap region is assumed to be 10 percent (30 percent for Kr-85 and 12 percent for I-131) ~~for the fuel handling accident and the locked rotor accident)~~ of the core inventory. The values are presented in Table 15.0-7.

For design basis accident radiological consequence analyses performed subsequent to December 1998, the core gap radioactivities are presented in Table 15.0-7b.

15.0.9.3 Primary and Secondary Side Coolant Activities

The equilibrium concentrations in the RCS and the secondary coolant system have been calculated assuming full power operation for the following cases: 1) one percent fuel defects, 2) normal operations using the guidelines of NUREG 0017 (USNRC 1976), and 3) plant Technical Specification iodine concentrations. The Technical Specification activities are used in the analysis of the main steam line break (MSLB), the locked rotor accident, the rod ejection accident, the failure of small lines carrying primary coolant outside containment, and the steam generator tube rupture. The Technical Specifications for BVPS-2 restrict the concentration in the primary and secondary systems to 0.35 and 0.1 μCi/gm I-131 dose equivalent, respectively.

In December 1998, the primary and secondary side coolant radioactivities for use in performing design basis radiological consequence analyses were revised. This revision was made as part of a larger effort which included revision of core radioactivity inventory (all used as accident source terms). The reanalyses were performed using updated plant operating and design parameters that had been changed since the radioactivities were last calculated.

The parameters used to calculate the revised primary and secondary side coolant radioactivities are presented in Table 15.0-8a. The revised core inventory values given in Table 15.0-7b are used as the basis for calculating the revised primary coolant and secondary side coolant and steam radioactivities. The computer codes and methodology used to determine these remain unchanged from those used to determine the former primary coolant and secondary side coolant and steam radioactivities. The exception noted in the preceding paragraph is a change to the basis for primary coolant radioactivity from 1.0 μCi/gm I-131 dose equivalent to a revised, lower value of 0.35 μCi/gm. This alone causes the primary system Technical Specification radioactivity concentrations to be lower than the former values by a factor of 1/0.35. The revised primary coolant and secondary side coolant and steam radioactivity concentrations are presented in Table 15.0-8b. The original (1.0 μCi/gm) concentrations are presented in Table 15.0-8.

15.0-8b

For the waste gas system rupture analysis, primary coolant concentrations with 1 percent fuel defects are assumed. These RCS concentrations are given in Table ~~11.1-2.~~ The calculation of releases due to a liquid-containing tank failure uses expected normal operation concentrations of 0.12 percent fuel defects. These concentrations are also presented in Table 11.1-2.

15.0.9.4 Iodine Spiking Concentrations

The analysis of an MSLB, steam generator tube rupture, and the failure of small lines carrying primary coolant outside containment include equilibrium coolant iodine concentrations augmented by iodine spiking. Both pre-accident and concurrent iodine spiking models are considered.

steam dump, feedwater control, and pressurizer pressure control. The ECCS, including the accumulators, is also modeled.

LOFTRAN is a versatile program which is suited to both accident evaluation and control studies as well as parameter sizing.

LOFTRAN also has the capability of calculating the transient value of DNBR based on the input from the core limits illustrated on Figures 15.0-1 and 15.0-1a. The core limits represent the minimum value of DNBR as calculated for typical or thimble cell.

LOFTRAN is further discussed by Burnett (1972).

15.0.11.3 TWINKLE

The TWINKLE program is a multi-dimensional spatial neutron kinetics code, which was patterned after steady state codes presently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, or three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2,000 spatial points, and performs its own steady state initialization. Aside from basic cross section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, and control rod motion. Various edits are provided, for example, channelwise power, axial offset, enthalpy, volumetric surge, pointwise power, and fuel temperatures.

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients which cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further described by Risher and Barry (1975).

15.0.11.4 THINC

The THINC Code is described in Section 4.4.

15.0.12 Radiological Consequences

The radiological consequences of each of the design basis accidents (DBA) were analyzed based on assumptions discussed in the respective sections. Specific parameters used in these analyses are tabulated in the corresponding sections.

Initial core and core gap activities, coolant Technical Specification equilibrium concentrations, pre-accident iodine spike primary coolant concentrations, and concurrent iodine spiking appearance rates are discussed in Section 15.0.9. Coolant concentrations at design basis and Technical Specification limit used in design basis accident radiological consequence analyses prior to 12/98 are given in Section 11.1. Subsequent to 12/98, these values were recalculated using updated plant design and operating parameters and the ORIGENS computer code. The revised values are provided in Table 15.0-8b. ~~The releases~~

~~to the environment resulting from each accident are presented in the respective sections.~~

Accident atmospheric dispersion coefficients (X/Q) for the exclusion area boundary and low population zone were used to calculate the potential offsite doses. The 0.5 percent sector-dependent X/Q values, presented in Table 15.0-11, were determined as described in Section 2.3.4. Main control room X/Q values for the LOCA are ~~also~~ given in Table 15.0-~~11~~ 14.

The atmospheric releases discussed in each accident section are used in conjunction with the appropriate X/Q values of Table 15.0-~~11~~ 14 to calculate the potential offsite doses for the corresponding accidents and the potential control room dose due to a LOCA. The methodology for determining the doses is discussed in Appendix 15A. The resulting EAB and LPZ doses are presented in Table 15.0-12 for all postulated accidents. The potential doses to main control room personnel due to DBAs are presented in Table 15.0-13.

For all cases the potential offsite doses are within the limits of 10 CFR 100, while the potential doses for the main control room due to a LOCA are within the limits of GDC 19 of Appendix A to 10 CFR 50.

15.0.13 References for Section 15.0

Bordelon F.M. et al 1974a. SATAN-VI Program: Comprehensive Space Time Dependent Analysis of Loss-of-Coolant. WCAP-8302 (Proprietary) and WCAP-8306.

Bordelon F.M. et al 1974b. LOCTA-IV Program: Loss-of-Coolant Transient Analysis. WCAP-8305.

Burnett, T.W.T. et al 1972. LOFTRAN Code Description. WCAP-7907, June 1972. (Also supplementary information in letter from T.M. Anderson, NS-TMS-1802, May 26, 1978 and NS-TMS-1824, June 16, 1978.)

ERS-MPD-91-035. "Assessment of the Doses in the Unit 2 Control Room Due to a Locked Rotor Accident at Unit 2 Assuming 18% Failed Fuel."

Hunin C. 1972. FACTRAN, A FORTRAN IV Code Thermal Transients in a UO_2 Fuel Rod. WCAP-7908.

Risher, Jr. D.H. and Barry R.F. 1975. TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code. WCAP-7979-P-A (Proprietary) and WCAP-8028-A, (Non-Proprietary).

~~SWEC 1998. Beaver Valley Power Station Unit 2, Radiological Accident Analyses Update, Site Boundary and Control Room Calculations, Radiological Source Term Report.~~

SWEC Calculation 12241/11700 UR(B)-478 "Design Reactor Core Inventory (3.96 w/o Initial Enrichment) and Associated Primary and Secondary Coolant Activities for BVPS"

SWEC Calculation 12241/11700 UR(B)-479 "Radiological Source Terms for Accident Analyses - Composite Equilibrium Reactor Core Inventory (3.6 w/o - 5 w/o Initial Enrichment) and the Associated Design Primary and Secondary Coolant Activities for BVPS"

TABLE 15.0-7a

PARAMETERS AND ASSUMPTIONS USED FOR CALCULATING
REACTOR CORE RADIONUCLIDE INVENTORY USING ORIGENS

A. Inputs for fuel, cladding and coolant material compositions

Uranium mass per fuel assembly	463.5 Kg/assembly

Uranium isotopic compositions*:

Initial Enrichment	Number of Assemblies	U-234(%)	U-235(%)	U-236(%)	U-238(%)
3.820	1	0.0327	3.8203	0.0017	96.145
3.604	8	0.0316	3.6037	0.0091	96.356
4.008	20	0.0373	4.0078	0.0010	95.954
3.608	56	0.0319	3.6077	0.0013	96.359
4.210	8	0.0333	4.2104	0.0018	95.755
4.000	20	0.0328	4.0000	0.0013	95.966
4.400	44	0.0390	4.4000	0.0013	95.560
0.740*	157	0.0057	0.7400	0.0010	99.253

* These values are representative of the natural uranium blankets located in the top and bottom six inches of each assembly.

Cladding material and density	ZIRLO - 6.44 gm/cm^3
Average coolant density in active core region	0.719 gm/cm^3
Average boron concentration in reactor coolant during a fuel cycle	Figure 1
Average temperature in fuel	1200°F
Average temperature in cladding	632°F
Average temperature in coolant	580.2°F

B. Inputs for fuel cell geometry

Fuel cell type	Square cell (17x17)
Cell dimension	0.496 inches
Fuel pin diameter	0.3225 inches
Clad outer diameter	0.374 inches
Clad inner diameter	0.329 inches

* The values shown are representative of 3.96% average enrichment. Core inventory calculations were performed using 3.6% and 5.0% average enrichments to represent bounding conditions. The maximum individual nuclide values are selected from these calculations for use in DBA analysis. See Table 15.0-7b.

TABLE 15.0-7b

IODINE AND NOBLE GAS INVENTORY IN REACTOR CORE AND FUEL ROD GAPS*

Nuclide	Core (Ci)	Fraction of Core Activity in Gap	Activity in Fuel Rod Gap (Ci)
I-130+	1.85E6	n/a	n/a
I-131	~~7.21E7~~ 7.24E7	0.12	~~8.65E6~~ 8.69E6
I-132	1.06E8	0.1	1.06E7
I-133	~~1.49E8~~ 1.49E8	0.1	~~1.48E7~~ 1.49E7
I-134	~~1.63E8~~ 1.65E8	0.1	~~1.63E7~~ 1.65E7
I-135	1.41E8	0.1	1.41E7
I-136+	6.54E7	n/a	n/a
Kr-83m	~~8.47E6~~ 9.04E6	0.1	~~8.47E5~~ 9.05E5
Kr-85m	~~1.73E7~~ 1.88E7	0.1	~~1.73E6~~ 1.88E6
Kr-85	~~7.17E5~~ 7.59E5	0.3	~~2.15E5~~ 2.28E5
Kr-87	~~3.45E7~~ 3.78E7	0.1	~~3.45E6~~ 3.78E6
Kr-88	~~4.78E7~~ 5.25E7	0.1	~~4.78E6~~ 5.25E6
Kr-89	~~5.91E7~~ 6.57E7	0.1	~~5.91E6~~ 6.57E6
Kr-90+	7.05E7	n/a	n/a
Xe-131m	~~9.80E5~~ 9.87E5	0.1	~~9.80E4~~ 9.87E4
Xe-133m	~~4.68E6~~ 4.69E6	0.1	~~4.68E5~~ 4.69E5
Xe-133	~~1.48E8~~ 1.49E8	0.1	~~1.48E7~~ 1.49E7
Xe-135m	~~3.10E7~~ 3.12E7	0.1	~~3.10E6~~ 3.12E6
Xe-135	~~4.06E7~~ 4.85E7	0.1	~~4.06E6~~ 4.85E6
Xe-137	1.35E8	0.1	1.35E7
Xe-138	~~1.25E8~~ 1.28E8	0.1	~~1.25E7~~ 1.28E7

NOTES:

* Based on 1500 days of operation at 2,705 MWt.

Kr-85 and I-131 gap activity fractions are in accordance with Regulatory Guide 1.25 and NUREG/CR-5009.

This table is applicable to design basis accident radiological consequence analyses performed subsequent to December 1998.

The core values represent the maximum for the expected range of core uranium enrichment.

+ Isotopes included only in the DBA LOCA analysis core release source term.

Br-82+	2.85E5	n/a	n/a
Br-83+	8.96E6	n/a	n/a
Br-85+	1.87E7	n/a	n/a

TABLE 15.0-8b

1% FAILED FUEL AND TECHNICAL SPECIFICATIONS PRIMARY AND SECONDARY SIDE IODINE AND NOBLE GAS CONCENTRATIONS*

Nuclide	1% Failed Fuel Primary Coolant	Technical Specification Primary Coolant**	Secondary Liquid***	Secondary Steam
I-131	2.6~~8~~9E+00	2.74E-01	8.38E-02	8.38 ~~8.4~~E-04
I-132	1.06E+00	1.08E-01	1.35E-02	1.35 ~~1.4~~E-04
I-133	4.03 ~~4.02~~E+00	4.11 ~~4.10~~E-01	9.13E-02	9.13 ~~9.3~~E-04
I-134	5.94 ~~5.99~~E-01	6.05 ~~6.03~~E-02	1.76E-03	1.76 ~~1.8~~E-05
I-135	2.32E+00	2.36 ~~2.37~~E-01	3.19 ~~3.20~~E-02	3.19 ~~3.2~~E-04
Kr-83m	3.~~71~~95E-01	4.02 ~~3.08~~E-02	****	6.45 ~~6.07~~E-07
Kr-85m	1.38 ~~1.27~~E+00	1.41 ~~1.30~~E-01		1.89 ~~1.75~~E-06
Kr-85	1.45 ~~1.37~~E+02	1.48 ~~1.40~~E+01		2.00 ~~1.89~~E-04
Kr-87	9.26 ~~8.45~~E-01	9.44 ~~8.64~~E-02		1.27 ~~1.17~~E-06
Kr-88	2.59 ~~2.35~~E+00	2.64 ~~2.40~~E-01		3.55 ~~3.24~~E-06
Kr-89	~~6.75~~E-02 7.51	7.65 ~~6.90~~E-03		1.03E-7 ~~9.30E-08~~
Xe-131m	4.70 ~~4.66~~E+00	4.79 ~~4.77~~E-01		6.45 ~~6.42~~E-06
Xe-133m	3.93 ~~3.92~~E+00	4.00 ~~4.01~~E-01		5.42E-06
Xe-133	2.91 ~~2.90~~E+02	2.97E+01		4.00E-04
Xe-135m	8.99 ~~8.93~~E-01	9.16 ~~9.13~~E-02		8.61 ~~8.64~~E-06
Xe-135	9.67 ~~8.09~~E+00	9.85 ~~8.28~~E-01		1.44 ~~1.23~~E-05
Xe-137	1.85E-01	1.89E-02		2.55E-07
Xe-138	6.35 ~~6.22~~E-01	6.47 ~~6.36~~E-02		8.72 ~~8.57~~E-07

NOTES:

* All concentration have units of μCi/gm.

** Technical Specification primary coolant concentrations correspond to 0.35 μCi/gm I-131 dose equivalent.

*** Technical Specification secondary liquid concentrations correspond to 0.1 μCi/gm I-131 dose equivalent.

**** All noble gas activity is assumed to leak directly from the primary coolant into the steam phase.

This table is applicable to design basis accident radiological consequence analyses performed subsequent to December 1998.

TABLE 15.0-9a

PRE-ACCIDENT IODINE SPIKE
PRIMARY COOLANT IODINE CONCENTRATIONS

Nuclide	Primary Coolant Concentration (μCi/gm)
I-131	1.64E+01
I-132	6.48 ~~6.51~~E+00
I-133	2.46E+01
I-134	3.63 ~~3.62~~E+00
I-135	1.42E+01

NOTES:

This table is applicable to design basis accident radiological consequence analyses performed subsequent to December 1998.

TABLE 15.0-10a

PARAMETERS AND ASSUMPTIONS AND MODEL USED FOR CALCULATING IODINE RELEASE RATES INTO REACTOR COOLANT DUE TO A CONCURRENT IODINE SPIKE

Thyroid dose conversion factors	Nuclide	mrem/µCi
	I-131	1.08E+03
	I-132	6.44E+00
	I-133	1.80E+02
	I-134	1.07E+00
	I-135	3.13E+01
Nuclide decay constants (λ_1)	Nuclide	second^{-1}
	I-131	9.9783E-07
	I-132	8.3713E-05
	I-133	9.2568E-06
	I-134	2.1963E-04
	I-135	2.9129E-05
Reactor coolant system leakage (L)	Technical Specification maximum allowable values	
Reactor coolant system mass (M)	Limiting value specific to the accident	
Letdown purification removal (E)	1	
Letdown purification flow rate (F)	135 ~~120~~ gpm	
Technical Specification equilibrium concentrations (EQ)	Table 15.0-8b	
Formula for iodine loss constant	$\lambda_{total} = (F*E/M) + (L/M) + \lambda_1$	
Concurrent iodine spike release rate (RR)	$RR = EQ * M * \lambda_{total}$	

NOTES:

Formulas for iodine release rates from EPRI Report, "Review of Iodine Spike Data from PWR Power Plants in Relation to SGTR with MSLB, TR-103680)"

This table is applicable to design basis accident radiological consequence analyses performed subsequent to December 1998.

TABLE 15.0-12

POTENTIAL DOSES DUE TO POSTULATED ACCIDENTS
(Rem)

		Exclusion Area Boundary			Low Population Zone*		
Postulated Accident	FSAR Section	CDE Thyroid	Whole Body EDE ~~Gamma~~	~~Beta Skin~~	CDE Thyroid	Whole Body EDE ~~Gamma~~	~~Beta Skin~~
Loss of nonemergency ac power to the station auxiliaries	15.2.6	~~$1.5x10^{-1}$~~ <1E+00	~~$5.2x10^{-4}$~~ <1E-01	~~$4.1x10^{-4}$~~	~~$2.1x10^{-2}$~~ <1E+00	~~$6.5x10^{-5}$~~ <1E-01	~~$6.8x10^{-5}$~~
Locked rotor	15.3.3	6.8E+00 ~~$3.7x10^{1}$~~	5.5E-01 ~~3.6~~	~~2.2~~	3.1E+00 ~~$1.6x10^{1}$~~	<1E-01 ~~$3.6x10^{-1}$~~	~~$2.3x10^{-1}$~~
Rod ejection	15.4.8	2.4E+01	<1E-01		1.2E+00	<1E-01	
Containment leakage		~~$4.1x10^{1}$~~	~~$1.9x10^{-1}$~~	~~$6.5x10^{-2}$~~	~~2.0~~	~~$9.2x10^{-3}$~~	~~$3.2x10^{-3}$~~
Secondary side		~~$2.2x10^{-1}$~~ <1E+00	~~$5.1x10^{-1}$~~ <1E-01	~~$3.7x10^{-1}$~~	~~$1.1x10^{-2}$~~ <1E+00	~~$2.5x10^{-2}$~~ <1E-01	~~$1.8x10^{-2}$~~
Small line break - loss-of-coolant	15.6.2	~~$1.6x10^{1}$~~ 6.3E+00	~~$7.0x10^{-2}$~~ <1E-01	~~$2.4x10^{-2}$~~	~~$8.2x10^{-1}$~~ <1E+00	~~$3.4x10^{-3}$~~ <1E-01	~~$1.2x10^{-3}$~~
Steam generator tube rupture	15.6.3	1.7E+01	<1E-01		<1E+00	<1E-01	
Pre-accident iodine spike		~~71.6~~	~~$2.0x10^{-1}$~~	~~$1.0x10^{-1}$~~	~~3.6~~	~~$7.0x10^{-3}$~~	~~$5.0x10^{-3}$~~
Concurrent iodine spike		~~13.4~~ 7.0E+00	~~$2.0x10^{-1}$~~ <1E-01	~~$2.0x10^{-1}$~~	~~$8.0x10^{-1}$~~ <1E+00	~~$9.0x10^{-3}$~~ <1E-01	~~$7.0x10^{-3}$~~
Loss-of-coolant	15.6.5	2.2E+02	4.9E+00		1.1E+01	2.6E-01	
Containment leakage		~~$2.7x10^{2}$~~	~~5.3~~	~~2.5~~	~~$1.3x10^{1}$~~	~~$2.6x10^{-1}$~~	~~$1.2x10^{-1}$~~
ECCS leakage		~~$8.3x10^{-1}$~~	~~$1.3x10^{-2}$~~	~~$5.1x10^{-3}$~~	~~$6.3x10^{-1}$~~	~~$1.2x10^{-2}$~~	~~$1.1x10^{-2}$~~
ECCS backleakage to RWST		~~0.0~~	~~0.0~~	~~0.0~~	~~6.9~~	~~$7.0x10^{-3}$~~	~~$3.4x10^{-3}$~~
Waste gas system rupture	15.7.1		2.9E-01				
Line rupture			~~$3.1x10^{-1}$~~	~~$1.9x10^{-1}$~~			
Tank rupture			~~$1.6x10^{-1}$~~ <1E-01	~~1.5~~			
Fuel handling	15.7.4	$2.9x10^{1}$	2.33	~~6.58~~	1.4	$1.1x10^{-1}$	~~$3.2x10^{1}$~~
Main steam line break	15.1.5	1.6E+01			2.4E+00		
Pre-accident Iodine spike		~~1.8E+01~~	< 1E-01	~~< 1E+00~~	~~2.8E+00~~	< 1E-01	~~< 1E+00~~
Concurrent Iodine spike		~~2.9E+01~~ 2.7E+01	< 1E-01	~~< 1E+00~~	~~1.4E+01~~ 1.3E+01	< 1E-01	~~< 1E+00~~

NOTE:

* For duration of accident

TABLE 15.0-13

Control Room Doses, rem, From Design Basis Accidents ~~6~~

Accident	CDE Thyroid	EDE ~~Gamma~~	Skin DE ~~Beta~~	Notes
Small Line Break	4.9E+00 ~~8.1~~	<2E-01 ~~8.0E-4~~	<1E+00 ~~7.7E-3~~	3,~~5~~
Steam Generator Tube Rupture				
Co-incident Spike	1.1E+00 ~~1.9~~	<2E-01 ~~3.0E-4~~	<1E+00 ~~6.1E-3~~	3,~~5~~
Pre-incident Spike	~~8.7~~ 2.3E+00	~~5.0E-4~~ <2E-01	~~7.9E-3~~ <1E+00	3,~~5~~
Rod Ejection Accident	3.4E+00 ~~4.9~~	<2E-01 4.9E-4	<1E+00 ~~3.8E-3~~	3,~~5~~
Fuel Handling Accident	2.3	9.3E-3	5.3E-1	3,5
Locked Rotor Accident	7.5E+00 ~~1.7~~	<2E-01 ~~1.6E-2~~	<1E+00 ~~2.3E-1~~	~~1,4,7~~ 3
Loss of Auxiliary AC Power	<1E+00 ~~2.1~~	<2E-01 ~~1.8E-4~~	<1E+00 ~~1.2E-2~~	3,~~5~~
Waste Gas System Rupture				
Line Break	---	<2E-01 ~~5.8E-2~~	<1E+00 ~~1.3~~	3,~~5~~
Tank Rupture	---	~~3.5E-2~~ <2E-01	~~9.7~~ 6.9E+00	3,~~5~~
DBA LOCA	~~1.3~~ 2.0E+00	~~3.2E-1~~ 3.3E-01	~~1.2E-1~~ <1E+00	2,~~5~~
Main Steam Line Break				
Co-incident Spike	2.8 ~~3.0~~E+00	< 2E-01	< 1E+00	1,4,~~5~~,~~8~~
Pre-incident Spike	1.3 ~~1.4~~E+00	< 2E-01	< 1E+00	1,4,~~5~~,~~8~~

Notes

1: Isolation by manual operator action at T=30 minutes post-accident.
2: Control Isolation actuated by CIB signal.
3: No action required.
4: Purge of Control Room atmosphere for 30 minutes at 16,900 cfm at no later than T=8 hr post-accident initiation.
5: Reference: ERS-SFL-93-004
~~6: Listed dose values represent the bounding value which may be higher than current analysis results.~~
~~7: Reference: ERS-MPD-91-035~~
~~8: Reference: ERS-SFL-96-010~~

TABLE 15.0-14

ACCIDENT METEOROLOGICAL PARAMETERS

Control Room Release Point*	0-8 Hours	8-24 Hours	1-4 Days	4-30 Days
Containment Building				
-Top	1.20	0.591	0.445	0.264
-Edge	1.88	0.932	0.706	0.418
Auxiliary Building	10.4	5.15	4.04	2.46
Main Steam Valve House	1.59	0.786	0.596	0.376
Service Building	2.21	1.11	0.851	0.517
Turbine Building	2.72	1.43	1.10	0.630
Gaseous Waste Storage Vault	17.4	9.36	7.69	5.55
RWST**	0.825	0.407	0.313	0.189
EAB***	0-2 Hours			
	12.5			
LPZ***	0-8 Hours	8-24 Hours	1-4 Days	4-30 Days
	0.604	0.433	0.210	0.0744

NOTES:

* These values were effective in 1/92 and are used for analyses documented after that date. Occupancy factors are not included in values. ~~Analyses performed prior to that date use the values from Table 15.0-11.~~

All values are in X/Q (x 10^{-4} sec/m^3)

** This value was effective 4/92 and is used for analyses documented after that date.

*** These values are applicable to design basis accident radiological consequence analyses performed subsequent to December 10/96. ~~Analyses performed prior to that date use the values from Table 15.0-11.~~

15.1.5.3 Radiological Consequences

The SLB is postulated to occur in a main steam line outside the containment. Steam and feedwater isolation valves shut automatically, and the AFWS starts, supplying feedwater to each of the steam generators until the feedwater system is manually isolated from the affected steam generator (that is, the one associated with the broken main steam line).

The conservative analysis assumes that BVPS-2 is operating with Technical Specification iodine concentrations in the primary coolant and secondary coolant systems and with a primary to secondary Technical Specification leakage of 450 gpd. The equilibrium primary and secondary coolant systems noble gas and iodine concentrations are presented in Table 15.0-8b.

The radiological consequences are determined assuming each of the following occurrences:

1. Pre-accident iodine spike, and

2. Accident-initiated concurrent iodine spike.

The pre-accident iodine spike is the result of a primary plant transient which will increase the primary system iodine concentrations to the levels shown in Table 15.0-9a. The accident-initiated or concurrent iodine spike is modeled by assuming that the iodine release rates from the fuel rods into the primary coolant are 500 times the Technical Specification equilibrium release rates. The iodine release rates for the concurrent iodine spiking conditions are calculated for the limiting SLB as detailed in Table 15.0-10a. The results used in the MSLB analysis are given in Table 15.1.3.

In 1999, the radiological consequences of a MSLB outside of containment was re-analyzed in support of the Alternate Plugging Criteria (APC) for steam generators (ref. T/S amendment 115). The MSLB is of interest due to the rapid depressurization of the secondary side and the high differential pressure across the steam generator tubes that can occur. The APC allows steam generator tubes having outside diameter stress corrosion cracking (ODSCC) to remain in service with higher NDE indications than would be allowed under prior repair criteria, subject to conditions established in technical specifications. One such requirement is to project, on the basis of the NDE indication (voltage), the potential leakage (95 percentile/95% confidence) should a MSLB occur, and, on the basis of this projected leakage, the resulting offsite and control room doses.

In lieu of calculating the radiological consequence of this event for each operating cycle, an analysis was performed to establish a maximum allowable accident leakage, against which the leakage projections could be compared. For this analysis, the thyroid dose was maximized at 10% of the 10 CFR 100 guideline of 300 rem (co-incident iodine spike). Analyses were also performed for control room habitability. ~~It~~ was determined that the EAB thyroid dose, assuming a co-incident iodine spike, would be limiting at a projected accident leak rate of ~~3~~ gpm. This maximum allowable leakage is used in the dose calculations in addition to the traditionally assumed technical specification primary-to-secondary leak rate.

2.5

The radiological consequences due to a postulated main steam line break (MSLB) accident are evaluated based on the assumptions listed in Table 15.1-3. The offsite power is assumed to be lost, thereby making the condenser unavailable for steam dump. The steam released from the secondary system is assumed to be released directly to the environment at ground level.

The environmental releases due to a MSLB following a pre-accident iodine spike ~~are presented in Table 15.1-4 while Table 15.1-5 provides~~ and the releases resulting from the MSLB with concurrent iodine spike ~~These releases~~ are combined with the atmospheric dispersion values presented in Table 15.0-14 to calculate offsite doses. The methodology employed in the dose calculations is discussed in Appendix 15A with results presented in Table 15.0-12. The radiological consequences for either a pre-accident iodine spike or concurrent iodine spike with a MSLB do not exceed 10 percent of the dose guidelines of 10 CFR 100, that is, 2.5 Rem to the whole body and 30 Rem to the thyroid.

15.1.5.4 Conclusions

The analysis has shown that the criteria stated in Section 15.1.5.1 are satisfied with the exclusion of the radiological criteria. Although DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the analysis, in fact, shows that the DNB design bases is met as stated in Section 4.4. The radiological consequences are a small fraction of the dose guidelines of 10 CFR 100.

15.1.6 References for Section 15.1

Burnett, T. W. T., et al 1972. LOFTRAN Code Description. WCAP-7907, June, 1972. Also supplementary information in letter from T. M. Anderson, NS-TMA-1802, May 26, 1978 and NS-TMA-1824, June 16, 1978.

DLC, Safety Analysis of the EAB, LPZ and Control Room Doses from a Main Steam Line Break Outside of CNMT at Unit 2 with Increased Primary-to-secondary Leakage (SG APL), ERS-SFL-96-010.

DLC Technical Evaluation Report No. 12075, BVPS-2 Design Basis for Safety Limit Associated with Overpower ΔT Trip Setpoint.

Hollingsworth, S. D. and Wood, D. C. 1978. Reactor Core Response to Excessive Secondary Steam Releases, WCAP-9227.

Moody, F. W. 1965. Transactions of the ASME. Journal of Heat Transfer, Figure 3, page 134.

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

USNRC Voltage Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking, Generic Letter 95-05.

TABLE 15.1-3

ASSUMPTIONS USED FOR THE MAIN STEAM LINE BREAK ACCIDENT

	~~Expected~~	~~Technical Specification~~
Primary coolant concentrations	~~Table 11.1-2~~	Table 15.0-8b
Primary to secondary leak rate	~~0.009 gpm~~	450 gpd
Secondary coolant concentrations	~~Table 11.1-6~~	Table 15.0-8b
Primary coolant concentrations due to pre-accident iodine spike	~~0.0~~	Table 15.0-9a
Concurrent iodine spike release rates (Ci/S) (Refer to Table 15.0-10a for release rate determination methodology)	~~Table 15.0-10a~~	I-131 ~~1.16~~ 1.28 I-132 ~~1.16~~ 1.20 I-133 ~~2.00~~ 2.19 I-134 ~~1.28~~ 1.32 I-135 ~~1.52~~ 1.63
Duration (hrs)	~~4~~	4
Iodine partition factor for initial and long term steam release from affected steam generator	~~1.0~~	1.0
Iodine partition factor in non-affected steam generator prior to and during accident after one hour	~~0.01~~	0.01
Time to isolate affected steam generator (hr)	~~8~~	8
Initial steam and water release from affected steam generator (0 - 30 min)	all of the liquid and steam mass initially present	
Long term steam release from affected steam generator (0 - 8 hr)	mass available due to total primary-to-secondary leak rate	
Steam release from 2 non-affected steam generators (lb)		
0 - 2 hr	~~336,776~~	336,776
0 - 8 hr	~~705,393~~	705,393

TABLE 15.1-3 (Cont)

	~~Expected~~	~~Technical Specification~~
Steam generator fluid content/SG*		
liquid weight (lb)	~~162,946~~	148,133+10% ~~162,946~~
steam weight (lb)	~~6,360~~	5,782+10% ~~6,360~~

* Corresponds to bounding conditions for this accident

TABLE 15.1-4

ENVIRONMENTAL RELEASES FROM A
MAIN STEAM LINE BREAK ACCIDENT WITH
PRE-ACCIDENT IODINE SPIKE

DELETE

Nuclide	Releases (Ci) 0-2 Hr	0-8 Hr
Kr-83m	3.99E-02	7.11E-02
Kr-85m	1.68E-01	4.43E-01
Kr-85	2.10E+02	8.31E+01
Kr-87	7.91E-02	1.17E-01
Kr-88	2.85E-01	6.29E-01
Kr-89	3.97E-04	3.97E-04
Xe-131m	7.13E-01	2.80E+00
Xe-133m	6.06E-01	2.44E+00
Xe-133	4.44E+01	1.75E+02
Xe-135m	2.64E+00	8.86E+00
Xe-135	2.49E+00	1.88E+01
Xe-137	1.31E-03	1.31E-03
Xe-138	1.62E-02	1.62E-02
I-131	3.05E+01	9.88E+01
I-132	8.24E+00	1.52E+01
I-133	4.21E+01	1.30E+02
I-134	2.84E+00	3.50E+00
I-135	2.15E+01	5.76E+01

<incorporated by reference>

TABLE 15.1-5

ENVIRONMENTAL RELEASES FROM A MAIN STEAM LINE BREAK ACCIDENT WITH CONCURRENT IODINE SPIKE

DELETE

Nuclide	Releases (Ci) 0-2 Hr	0-8 Hr
Kr-83m	3.99E-02	7.10E-02
Kr-85m	1.68E-01	4.43E-01
Kr-85	2.10E+02	8.29E+01
Kr-87	7.90E-02	1.17E-01
Kr-88	2.85E-01	6.29E-01
Kr-89	3.96E-04	3.96E-04
Xe-131m	7.13E-01	2.83E+00
Xe-133m	6.10E-01	3.12E+00
Xe-133	4.44E+01	1.84E+02
Xe-135m	5.78E+00	7.13E+01
Xe-135	3.44E+00	1.01E+02
Xe-137	1.31E-03	1.31E-03
Xe-138	1.62E-02	1.62E-02
I-131	4.66E+01	4.72E+02
I-132	3.42E+01	2.17E+02
I-133	7.50E+01	7.32E+02
I-134	2.81E+01	1.09E+02
I-135	5.18E+01	4.56E+02

<incorporated by reference>

The assumptions used in the analysis are similar to the loss of normal feedwater flow incident (Section 15.2.7) except that power is assumed to be lost to the RCPs at the time of reactor trip, and only one motor driven auxiliary feedwater pump is conservatively assumed to deliver flow.

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

Results

The transient response of the RCS following a loss of ac power is shown on Figures 15.2-9 through 15.2-13 for three loops initially in operation, and on Figures 15.2-9a through 15.2-13a for two loops initially in operation.

The first few seconds after the loss of power to the RCPs will closely resemble the complete loss of flow incident (Section 15.3.2), that is, core damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor. After the reactor trip, stored and residual decay heat must be removed to prevent damage to either the RCS or the core. The LOFTRAN results show that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown.

The calculated sequence of events for this accident is listed in Table 15.2-1.

15.2.6.3 Radiological Consequences

A loss of nonemergency ac power to BVPS-2 auxiliaries would result in a turbine and reactor trip and loss of condenser vacuum. Heat removal from the secondary system would occur through the steam generator power relief valves or safety valves. The parameters to be used in analysis of the radiological consequences of the loss of ac power are summarized in Table 15.2-2.

No fuel damage is postulated to occur from this transient. The environmental releases due to a loss of ac power and actuation of the steam safety valves are <insert> ~~presented in Table 15.2-3.~~ These releases are combined with the atmospheric dispersion values presented in Table 15.0-11 to calculate offsite doses. The results of the analysis, as shown in Table 15.0-12, indicate the doses to the unrestricted area are within the limits prescribed by 10 CFR 20.

15.2.6.4 Conclusions

Analysis of the natural circulation capability of the RCS has demonstrated that sufficient heat removal capability exists following RCP coastdown to prevent fuel or clad damage. The radiological consequences of this event are within the limits specified by 10 CFR 20.

<insert above>

Calculated using the radiological source terms presented in Tables 15.0-7b and 15.0-8b and the parameters presented in Table 15.2-2.

removal capability of the AFWS, and makeup is provided by the high head safety injection pumps.

The major difference between the two cases analyzed can be seen in the plots of hot- and cold-leg temperatures, Figures 15.2-21 and 15.2-22 (with offsite power available) and Figures 15.2-27 and 15.2-28 (without offsite power). It is apparent that for the initial transient (300 seconds), the case without offsite power results in higher temperatures in the hot-leg. For longer times, however, the case with offsite power results in a more severe rise in temperature due to the addition of pump heat.

The pressurizer fills more rapidly for the case with power due to the increased coolant expansion resulting from the pump heat addition; hence, more water is relieved for the cases with power. As previously stated, however, the core remains covered with water for all cases.

15.2.8.3 Radiological Consequences

The feedwater line break with the most significant consequences would be one that occurred inside the containment between a steam generator and the feedwater check valve. In this case, the contents of the steam generator would be released to the containment. Since no fuel failures are postulated, the radioactivity released is less than that for the steam line break (SLB), Section 15.1.5.3. Furthermore, automatic isolation of the containment would further reduce any radiological consequences from this postulated accident.

15.2.8.4 Conclusions

Results of the analyses show that for the postulated feedwater line rupture, AFWS capacity is adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the reactor core. Radiological doses from the postulated feedwater line rupture would be less than those previously presented for the postulated SLB.

15.2.9 References for Section 15.2

Burnett, T. W. T. et al 1972. LOFTRAN Code Description. WCAP-7907. (Also, supplementary information in letter from T. M. Anderson, NS-TMA-1802 May 26, 1978 and NS-TMA-1824, June 16, 1978.)

Mangan, M. A. 1972. Overpressure Protection for Westinghouse Pressurized Water Reactors. WCAP-7769.

Westinghouse Electric Corporation 1974. Westinghouse Anticipated Transients Without Trip Analysis. WCAP-8330.

DLC, Calculation ERS-ATL-99-013, "Safety Analysis of the Radiological Consequences of a loss of AC Power Design Basis Accident at unit 2, Common Control Room, EAB and LPZ Doses".

TABLE 15.2-2

PARAMETERS USED FOR THE LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES ACCIDENT

Parameters	~~Expected~~	~~Technical Specification~~
Power (MWt)	~~2,766~~	~~2,766~~ 2705
~~Fraction of fuel with defects~~	~~0.0012~~	~~0.0026~~
Primary coolant concentrations	~~Table 11.1-2~~	Table 15.0-8b
Secondary coolant concentrations	~~Table 11.1-6~~	Table 15.0-8b
Primary to secondary leak rate (~~gpm~~ gpd)	~~0.009~~	~~1.0~~ 450
Iodine partition factor in all steam generators prior to and during the accident	~~0.01~~	0.01
Duration of plant cooldown by secondary system after accident (hr)	~~8~~	8
Steam release from 3 steam generators (lb)		
0-2 hr	~~443,878~~	443,878
2-8 hr	~~793,664~~	793,664
Feedwater flow to 3 steam generators (lb)		
0-2 hr	~~527,065~~	527,065
2-8 hr	~~874,470~~	874,470
Steam generator fluid content/SG *		
Liquid (lb)	~~99,300~~	~~99,300~~ 103,172 - 10%
Steam (lb)	~~8,700~~	~~8,700~~ 6534 + 10%

* Correspond to bounding conditions for this accident.

TABLE 15.2-3

ENVIRONMENTAL RELEASES DUE TO A LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES

DELETE

Nuclide	Total Releases (Ci) 0-2 Hr	Total Releases (Ci) 0-8 Hr
Kr-83m	$3.7x10^{-2}$	$6.7x10^{-2}$
Kr-85m	$2.3x10^{-1}$	$5.9x10^{-1}$
Kr-85	1.3	5.3
Kr-87	$8.7x10^{-2}$	$1.3x10^{-1}$
Kr-88	$3.0x10^{-1}$	$6.6x10^{-1}$
Kr-89	$4.8x10^{-4}$	$4.8x10^{-4}$
Xe-131m	$1.3x10^{-2}$	$5.1x10^{-2}$
Xe-133m	$3.5x10^{-1}$	1.4
Xe-133	3.1	$1.2x10^{1}$
Xe-135m	$4.5x10^{-2}$	$7.8x10^{-2}$
Xe-135	$3.7x10^{-1}$	1.2
Xe-137	$9.2x10^{-4}$	$9.2x10^{-4}$
Xe-138	$1.5x10^{-2}$	$1.5x10^{-2}$
I-131	$1.5x10^{-1}$	$4.1x10^{-1}$
I-132	$3.3x10^{-2}$	$5.1x10^{-2}$
I-133	$2.0x10^{-1}$	$5.1x10^{-1}$
I-134	$6.9x10^{-4}$	$8.7x10^{-4}$
I-135	$8.6x10^{-2}$	$1.9x10^{-1}$

<incorporated by reference>

2. Locked Rotor with Three Loops operating, Loss of Power to the Remaining Pumps

   The transient results for this case are shown on Figures 15.3-17 through 15.3-20. The results of these calculations are summarized in Table 15.3-2b. The peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. Also, the peak cladding surface temperature in considerably lose than 2,700°F. Both the peak RCS pressure and the peak cladding surface temperature for this cans are similar to the 3-loop transient with power available as discussed on the previous page.

The calculated sequence of events for the two cases analyzed is shown in Table 15.3-1. Figure 15.3-17 shows that the core flow reaches a new equilibrium value by 10 seconds. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

Following reactor trip, Beaver Valley Power Station - Unit 2 (BVPS-2) will approach a stabilized condition at hot standby; normal plant operating procedures may then be followed to maintain a hot condition or to cool the plant to cold shutdown. The operating procedures would call for operator action to control RCS boron concentration and pressurizer level using the CVCS, and to maintain steam generator level through control of the main feedwater system or AFWS. Any action required of the operator to maintain BVPS-2 in a stabilized condition will be in a time frame in excess of ten minutes following reactor trip.

15.3.3.3 Radiological Consequences

The radiological consequences of a postulated locked rotor accident are analyzed assuming 18% failed fuel. The primary to secondary system leakage rate is at the Technical Specification value of ~~1 gpm.~~ 450 gpd The primary coolant and secondary side iodine and noble gas concentrations prior to the start of the accident are presented in Table 15.0-8b ground level until secondary side pressure decreases below the relief valve actuation value.

The environmental releases for a postulated locked rotor accident, ~~shown in Table 15.3-4~~ are combined with the atmospheric dispersion values presented in Table 15.0-~~11~~ 14 to determine the exclusion area boundary and low population zone doses given in Table 15.0-12. The methodology used in calculating the offsite doses is discussed in Appendix 15A. The radiological consequences for a locked rotor event do not exceed a small fraction of 10 CFR 100 guidelines.

15.3.3.4 Conclusions

Since 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.

Since the peak cladding surface temperature calculated for the hot spot during the worst transient remains considerably less than 2,700°F, the core will remain in place and intact with no loss of core cooling capability. No fuel failures are predicted for the locked rotor accident (Van Houten 1979).

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 an RCP 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 RCP rotor seizure event. Reactor trip is initiated on a low flow signal in the affected loop.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generator is reduced, first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the PORVs, and opens the pressurizer safety valves, in that sequence. The PORVs are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure reducing effect as well as the pressure reducing effect of the spray is not included in the analysis.

This event is classified as an ANS Condition IV incident (a limiting fault as defined in Section 15.0.1.

15.3.4.2 Radiological Consequences

The radiological consequences for an RCP shaft break event are less severe than those from the locked rotor incident (Section 15.3.3).

15.3.4.3 Conclusions

The consequences of a RCP shaft break are less severe than those calculated for the locked rotor accident (Section 15.3.3). With a failed shaft, the impeller could be free to spin in a reverse direction as opposed to being fixed in position as assumed in the locked rotor analysis. However, the net effect on core flow is negligible, resulting in only a slight decrease in the end point (steady state) core flow. For both the shaft break and locked rotor incidents, reactor trip occurs very early in the transient. In addition, the locked rotor analysis conservatively assumes that DNB occurs at the beginning of the transient.

15.3.5 References for Section 15.3

Baldwin, M.S., et al 1975. An Evaluation of Loss of Flow Accident caused by Power System Frequency Transients in Westinghouse PWRs, WCAP-8424, Revision 1.

Burnett, T. W. T, et al 1984. LOFTRAN Code Description, WCAP-7907-PA (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984.

Hargrove, H.G., "FACTRAN-A Fortran Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

Van Houten, R. 1979. Fuel Rod as a Consequence of Departure from Nucleate Boiling or Dryout. Office of Nuclear Regulatory Research, USNRC, Washington, D.C. NUREG-0562.

DLC, Calculation "Assessment of the Doses in the Unit 2 Control Room Due to a Locked Rotor Accident at Unit 2 Assuming 18% Failed Fuel (Includes Offsite Dose Added in Rev. 1)

TABLE 15.3-3

PARAMETERS USED FOR THE LOCKED ROTOR ACCIDENT

Parameter	
Power (MWt)	~~2,766~~ 2705
~~Fraction of fuel with defects prior to the accident~~	~~.0026~~
Primary coolant concentrations prior to the accident	Table 15.0-8b
Secondary coolant concentrations prior to the accident	Table 15.0-8b
Primary to secondary leak rate (~~gpm~~ gpd)	~~1.0~~ 450
Iodine partition factor in all steam generators prior to and during the accident	0.01
Duration of plant cooldown by secondary system after accident (hr)	8
Steam release from steam generators (lb)	
0-2 hr	443,878
2-8 hr	793,664
Feedwater flow to steam generators (lb)	
0-2 hr	527,065
2-8 hr	874,470
Steam generator fluid content/SG (lb)*	
Liquid	~~92,000~~ 102,230 -10%
Steam	~~7,190~~ 6,152 +10%
~~Control room pressurization fan flow, cfm~~	~~1,030~~
~~Control room purge flow, cfm~~	~~16,900~~
~~CREBAPS flow assumed, cfm~~	~~0~~

* Corresponds to bounding conditions for this accident.

and subsequently manually withdraw the control rods, a process that takes several hours. The BVPS-2 Technical Specifications require that the operator determine the estimated critical position of the control rods prior to approaching criticality thus assuring that the reactor does not go critical with the control rods below the insertion limits. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the source range reactor trip after receiving P-6 from the intermediate range (nominally at 10^5 cps). Too fast a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly leaving insufficient time to manually block the source range reactor trip. Failure to perform this manual action results in a reactor trip and immediate shutdown of the reactor.

After reactor trip with all loops in service there is at least 35 minutes for operator action prior to return to criticality. The required operator action is the opening of valves 2CHS*LCV-115B and D to initiate boration and the closing of valves 2CHS*LCV115C and E terminate dilution.

Dilution During Full Power Operation

With the reactor in manual control and no operator action taken to terminate the transient, the power and temperature rise will cause the reactor to reach the overtemperature ΔT trip setpoint resulting in a reactor trip. After reactor trip, with all loops in service, there is at least 16 minutes for operator action prior to return to criticality. The required operator action is the opening of valves 2CHS*LCV115B and D and the closing of valves 2CHS*LCV115C and E. The boron dilution transient in this case is essentially the equivalent of an uncontrolled rod withdrawal at power. The maximum reactivity insertion rate for a boron dilution transient is conservatively estimated to be 1.8 pcm/sec and is within the range of insertion rates analyzed for uncontrolled rod withdrawal at power. It should be noted that prior to reaching the overtemperature ΔT reactor trip the operator will have received an alarm overtemperature ΔT and an overtemperature ΔT turbine runback.

With the reactor in automatic rod control the pressurizer level controller will limit the dilution flow rate to the maximum letdown rate, approximately ~~120~~ 135 gpm. If a dilution rate in excess of the letdown rate is present, the pressurizer level controller will throttle charging flow down to match the letdown rate.

analysis, the accident for BVPS-2 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 Westinghouse cores are under-moderated, and bowing will tend to increase the under-moderation at the hot spot. Since the 17 by 17 fuel design is also under-moderated, the same effect would be observed. 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.

15.4.8.3 Radiological Consequences

The radiological consequences of a postulated rod ejection accident are analyzed in accordance with the methods described in Regulatory Guide 1.77. Initial concentrations of radionuclides in the primary and secondary coolant systems are based on Technical Specification limits. Also postulated is a Technical Specification primary-to-secondary leakage of 450gpd ~~1 gpm.~~ The primary and secondary system concentrations prior to the accident are shown in Table 15.0-8b. The rod ejection accident produces an adverse core power distribution which results in localized fuel rod damage. The assumed damage includes the breach of the fuel clad, releasing a fraction of the core gap activity, plus melting of a fraction of the core fuel pins which reach or exceed the initiation temperature of fuel melting. The quantity of nuclides released from the fuel is based on the assumption that 10 percent of fuel rods experience clad damage and 0.25 percent of fuel rods experience melting. The fuel and gap activity is presented in Table 15.0-7b.

Activity released into the primary coolant from melted fuel pins and the fuel rod gap is assumed to be immediately released into the containment through the break in the RCS located in the reactor pressure vessel head. The activity released to the containment is assumed to leak from the building to the environment at the design basis leak rate as a ground level release. No credit is taken for the removal of iodines due to the containment spray system; however, the containment will reach subatmospheric conditions and leakage will cease within 1 hour due to the effects of the spray systems.

Concurrently, the activity in the 450 gpd coolant is assumed to leak to the secondary side at the rate of ~~1 gpm~~. Offsite power is assumed to be lost making the condenser unavailable for steam dump. The primary-to-secondary leakage activity along with the initial secondary system inventory is assumed to be released from the secondary system to the environment as a ground level release.

The releases to the environment from the containment leakage and secondary side are ~~presented separately in Table 15.4-4 and are~~ calculated based on the assumptions summarized in Table 15.4-3.

The offsite doses are computed by the methods discussed in Appendix 15A using the environmental releases in combination with the atmospheric dispersion values listed in Table 15.0-~~11~~ 14.

The radiological consequences of the postulated rod ejection accident are presented in Tables 15.0-12, and 15.0-13.

15.4.8.4 Conclusions

Conservative analyses indicate that the described fuel and cladding 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 stress to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the reactor coolant system. The analyses have demonstrated that the fission product release, as a result of the number of fuel rods entering DNB, is limited to less than 10 percent of the fuel rods in the core.

The calculated dose values for the rod ejection accident are well within the 10 CFR 100 guidelines of 75 Rem thyroid and 6 Rem whole body.

15.4.9 Spectrum of Rod Drop Accidents in a Boiling Water Reactor

This section applies only to BWRs, and is not applicable to BVPS-2.

15.4.10 References for Section 15.4

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

Burnett, T.W.T. 1969. Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors. WCAP-7306.

Burnett, T.W.T. 1972. LOFTRAN: Code Description WCAP-7907. Also supplementary information letter from T.M. Anderson, NS-TMA-1802, May 26, 1978 and NS-TMA-1824, June 16, 1978.

Hunin, C. 1972. FACTRAN: A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod. WCAP-7908.

Liimataninen, R.C. and Testa, F.J. 1966. Studies in TREAT of Zircaloy 2-Clad, UO_2-Core Simulated Fuel Elements. ANL-7225, p. 177.

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

Risher, D.H., Jr. and Barry, R.F. 1975. TWINKLE: A Multi-Dimensional Neutron Kinetics Computer Code, WCAP-7979-P-A (Proprietary) and WCAP-8028-A.

Taxelius, T.G. (Ed) 1970. Annual Report - Spert Project, October 1968, September 1969. Idaho Nuclear Corporation 1N-1370.

Westinghouse 1974. Westinghouse Anticipated Transients Without Trip Analysis. WCAP-8330.

DLC, Calculation ERS-JTL-99-011 "Safety Analysis of the Radiological Consequences of a Control Rod Ejection DBA at BVPS Unit 2, Control Room, EAB and LPZ Doses"

TABLE 15.4-3

PARAMETERS USED FOR THE ROD CONTROL
CLUSTER EJECTION ACCIDENT

Parameters	~~Expected~~	~~Technical Specification~~
Power (MWt)	~~2,766~~	~~2,766~~ 2,705
~~Fraction of fuel with defects, prior to rod ejection~~	~~0.0012~~	~~0.0026~~
Primary coolant concentrations	~~Table 11.1-2~~	Table 15.0-8b
Secondary coolant concentrations	~~Table 11.1-6~~	Table 15.0-8b
Primary-to-secondary leak rate (~~gpm~~ gpd)	~~0.009~~	~~1.0~~ 450
~~Containment free volume (ft³)~~	~~$1.8x10^6$~~	~~$1.8x10^6$~~
Core and gap activity	~~Table 15.0-7~~	Table 15.0-7b
Fraction of core activity in fuel-clad gap		
Iodine *	~~0.10~~	0.10
Noble gases*	~~0.10~~	0.10
Fraction of fuel experiencing clad failure	~~0.0~~	0.10
Fraction of core activity released to reactor coolant from failed fuel		
Iodine	~~NA~~	0.10
Noble gases	~~NA~~	0.10
Melted fuel fraction	~~0.0~~	0.0025
Fraction of activity from melted fuel available for release from containment		
Iodine	~~NA~~	0.25
Noble gases	~~NA~~	1.0
Fraction of activity from melted fuel available for release from secondary system		
Iodine	~~NA~~	0.5
Noble gases	~~NA~~	1.0
Containment leak rate 0-1 hr (%/day)	~~0.1~~	0.1

* Gap fractions assumed for I-131 = 0.12 and Kr-85 = 0.30

TABLE 15.4-3 (Cont)

Parameters	~~Expected~~	~~Technical Specification~~
Iodine partition factor in steam generators prior to and during accidents	~~0.01~~	0.01
Steam dump from relief valves (lb)	~~58,600~~	58,600
Duration of dump from relief valves (sec)	~~500~~	500
Steam generator fluid content/SG (lb)**		
Liquid	~~99,300~~	~~99,300~~ 102,230−10%
Steam	~~8,700~~	~~8,700~~ 6,152 +10%

** Corresponds to bounding conditions for this accident.

TABLE 15.4-4

RADIOACTIVITY RELEASED
TO THE ENVIRONMENT
DUE TO ROD EJECTION ACCIDENT

DELETE

Nuclide	From Containment (Ci)	Releases From Secondary System (Ci)	Total (Ci)
Kr-83m	5.2	$2.6x10^1$	$3.1x10^1$
Kr-85m	$1.5x10^1$	$6.5x10^1$	$8.0x10^1$
Kr-85	$3.8x10^{-1}$	1.6	$2.0x10^1$
Kr-87	$2.4x10^1$	$1.3x10^2$	$1.5x10^2$
Kr-88	$3.7x10^1$	$1.7x10^2$	$2.1x10^2$
Kr-89	4.5	$2.4x10^2$	$2.4x10^2$
Xe-131m	$2.2x10^{-1}$	$9.0x10^{-1}$	1.1
Xe-133m	2.0	8.0	$1.0x10^1$
Xe-133	$8.3x10^1$	$3.4x10^2$	$4.2x10^2$
Xe-135m	$1.9x10^1$	$9.0x10^1$	$1.1x10^2$
Xe-135	$2.3x10^1$	$8.7x10^1$	$1.1x10^2$
Xe-137	6.9	$3.1x10^2$	$3.2x10^2$
Xe-138	$2.4x10^1$	$3.1x10^2$	$3.3x10^2$
I-131	$3.0x10^1$	$1.6x10^{-1}$	$3.0x10^1$
I-132	$4.0x10^1$	$2.1x10^{-1}$	$4.0x10^1$
I-133	$7.0x10^1$	$3.5x10^{-1}$	$7.0x10^1$
I-134	$5.5x10^1$	$3.6x10^{-1}$	$5.5x10^1$
I-135	$5.9x10^1$	$3.0x10^{-1}$	$5.9x10^1$

<incorporated by reference>

limit value throughout the transient; thus, the departure from nucleate boiling (DNB) design-basis as described in Section 4.4 is met.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

15.6.2.1 Identification of Causes and Accident Description

Lines connected to the RCS and penetrating the containment, as well as isolation provisions are identified in Table 6.2-60.

There are no instrument lines connected to the RCS that penetrate the containment. There are, however, the sample lines from the hot and cold legs of reactor coolant loops and the steam and liquid space of the pressurizer, and the CVCS letdown and excess letdown lines that penetrate the containment. The sample lines and the CVCS letdown and excess letdown lines are all provided with normally open containment isolation valves on both sides of the containment wall. In all cases, the containment isolation valves are designed in accordance with the containment isolation requirements of General Design Criterion 55 (Section 6.2.4).

The most severe small line and with a coincident loss of heat exchanger cooling rupture with regard to radioactivity release during normal BVPS-2 operation is a complete severance of the 2-inch letdown line at a location outside containment, ~~upstream~~ downstream of the letdown heat exchanger. This event would result in a loss-of-reactor coolant at the rate of approximately ~~160 gpm~~ 16.0 lbm/s based on a density of 44.13 ~~57~~ lbs/ft^3 and on the flow restriction provided by two of the three letdown line orifices in service (the 45-gpm orifice and one of the 60-gpm orifices), shown on Table 9.3-8 and Figure 9.3-24.

The time required for the operator to identify the accident and isolate the rupture is expected to be less than 15 minutes. Diverse instrumentation in the form of letdown line pressure and flow downstream of the postulated break location, volume control tank level and pressurizer level with indication at the main control board will allow detection of the failure by the operator. In addition, a control room operator can determine specific plant areas which are experiencing high radiation after receiving plant high radiation annunciation. The operator would isolate the letdown line rupture by closing the letdown orifice isolation valves, 2CHS*AOV200A, B, and C of the pressurizer low level isolation valves, 2CHS*LCV460A and 2CHS*LCV460B. All valves are provided with control switches with indicating lights at the main control board and at the emergency shutdown panel. All valves are air-operated and designed to fail close on loss of air or electrical power. There are no single failures that would prevent isolation of the letdown line rupture.

15.6.2.2 Analysis of Effects and Consequences

Method of Analysis

The amount of primary coolant released is conservatively estimated by assuming critical flows in the ruptured letdown line. The mass of fluid released from the postulated break was calculated using the Zaloudek correlation in WCAP-8312A (Reference 2) for subcooled liquids and the theoretical model developed by Moody for saturated conditions. Immediately after the rupture, the Moody model is used for a saturated liquid until the liquid in the letdown line between the orifices and rupture point is depleted. After the liquid is depleted, Zaloudek's subcooled correlation is used at the orifice and continues until isolation occurs at 15 minutes after the break. These critical flow correlations are in accordance with WCAP-8312A.

15.6.2.3 Radiological Consequences

after

The failure outside the containment of small lines carrying primary coolant is postulated to occur in the letdown line ~~to~~ the letdown heat exchanger. The rupture of this line will result in the loss of primary coolant, with isolation occurring within 15 minutes. The rupture will result in the discharge of primary coolant directly into the auxiliary building or into the contiguous areas, with the radioactivity released to the environment at ground level. All potential locations for the small line break in the auxiliary building are within ventilation zones of the supplementary leak collection and release system (SLCRS). A small line break in the contiguous areas would be serviced by the SLCRS after receipt of a high radiation signal from a QA Category II ventilation monitor. However, the conservative analysis does not take credit for SLCRS operation.

The assumptions for evaluating the radiological consequences of the postulated small line failure are summarized in Table 15.6-2. The conservative analysis assumes primary coolant Technical Specification equilibrium activities as presented in Table 15.0-8b

Additionally, a concurrent iodine spike is postulated to occur with iodine release rates into the primary coolant ~~as shown in Table 15.0-10.~~ The resulting releases to the environment based on the stated assumptions. ~~and postulated activities are presented in Table 15.6-3.~~

and 15.0-13

The radiological consequences resulting from a postulated failure of a small line carrying primary coolant outside containment are presented in Tables 15.0-12. The offsite doses are determined using the calculated environmental releases for this accident and the atmospheric dispersion values given in Table 15.0-~~11~~. 14 The methodology for calculating the offsite doses is discussed in Appendix 15A. The radiological consequences for this event are a

at a rate calculated using the methodology provided in Table 15.0-10a.

The RCS depressurization is performed using normal pressurizer spray if the reactor coolant pumps (RCPs) are running. However, if offsite power is lost or the RCPs are not running for some other reason, normal pressurizer spray is not available. In this event, RCS depressurization can be performed using the pressurizer power operated relief valves (PORVs) or auxiliary pressurizer spray.

5. Terminate SI to stop primary to secondary leakage.

   The previous actions will have established adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, SI flow must be stopped to terminate primary to secondary leakage. Primary to secondary leakage will continue after SI flow is stopped until RCS and ruptured steam generator pressures equalize. Charging flow, letdown, and pressurizer heaters will then be controlled to prevent repressurization of the RCS and reinitiation of leakage into the ruptured steam generator.

Following SI termination, the plant conditions will be stabilized, the primary to secondary break flow will be terminated, and all immediate safety concerns will have been addressed. At this time a series of operator actions are performed to prepare the plant for cooldown to cold shutdown conditions. Subsequently, actions are performed to cool down and depressurize the RCS to cold shutdown conditions and to depressurize the ruptured steam generator.

15.6.3.2 Analysis of Effects and Consequences

An SGTR results in the leakage of contaminated reactor coolant into the secondary system and subsequent release of a portion of the activity to the atmosphere. Therefore, an analysis must be performed to assure that the offsite radiological consequences resulting from an SGTR are within the allowable guidelines. One of the major concerns for an SGTR is the possibility of steam generator overfill since this could potentially result in a significant increase in the offsite radiological consequences. Therefore, an analysis was performed to demonstrate margin to steam generator overfill, assuming the limiting single failure relative to overfill. The results of this analysis demonstrated that there is margin to steam generator overfill for BVPS Unit 2. An analysis was also performed to determine the offsite radiological consequences, assuming the limiting single failure relative to offsite doses without steam generator overfill. Since steam generator overfill does not occur, the results of this analysis represent the limiting consequences for an SGTR for BVPS Unit 2. The analyses to demonstrate margin to overfill ~~and to determine the offsite radiological consequences~~ for a design basis SGTR for BVPS Unit 2 are presented in Schrader, 1990, and the results of the offsite radiological consequences analysis are discussed below.

i. Accident Initiated Spike - The initial primary coolant iodine concentration is ~~1~~ 0.35 μCi/gm of Dose Equivalent (D.E.) I-131. Following the primary system depressurization associated with the SGTR, an iodine spike is initiated in the primary system which increases the iodine release rate from the fuel to thecoolant to a value 500 times greater than the release rate that corresponds to the initial primary system iodine concentration. The duration of the iodine spike is 4.0 hours.

ii. Pre-Accident Spike - A reactor transient has occurred prior to the SGTR and has raised the primary coolant iodine concentration from 1 to ~~60~~ 21 μCi/gram of D.E. I-131.

b. The initial secondary coolant iodine concentration is 0.1 μCi/gram of D.E. I-131.

c. The chemical form of iodine in the primary and secondary coolant is assumed to be elemental.

d. The initial noble gas concentrations in the reactor coolant are based upon ~~approximately 0.26% fuel defects.~~ design concentrations reduced to correspond to the Technical Specification limit D.E. I-131.

3. Dose Calculations

The iodine transport model utilized in this analysis was proposed by Postma and Tam (NUREG 0409). The model considers break flow flashing, droplet size, bubble scrubbing, steaming, and partitioning. The model assumes that a fraction of the iodine carried by the break flow becomes airborne immediately due to flashing and atomization. Removal credit is taken for scrubbing of iodine contained in the atomized coolant droplets as a function of the height of the secondary water level above the rupture site. The fraction of primary coolant iodine which is not assumed to become airborne immediately mixes with the secondary water and is assumed to become airborne at a rate proportional to the steaming rate and the iodine partition coefficient. This analysis conservatively assumes an iodine partition coefficient of 0.01 between the steam generator liquid and steam phases. Droplet removal by the dryers is conservatively assumed to be negligible. The iodine transport model is illustrated in Figure 15.6-69.

The following assumptions and parameters were used to calculate the activity released to the atmosphere and the offsite doses following a SGTR.

a. The mass of reactor coolant discharged into the secondary system through the rupture and the mass of steam released from the ruptured and intact steam generators to the atmosphere are presented in Table 15.6-5a.

b. The time dependent fraction of rupture flow that flashes to steam and is immediately released to the environment is presented in Figure 15.6-70. The break flow flashing fraction was conservatively calculated assuming that 100 percent of the break flow comes from the hot leg side of the steam generator, whereas the break flow actually comes from both the hot leg and cold leg sides of the steam generator.

c. In the iodine transport model, the time dependent iodine removal efficiency for scrubbing of steam bubbles as they rise from the rupture site to the water surface conservatively assumes that the rupture is located at the intersection of the outer tube row and the upper anti-vibration bar (approximately 4 inches below the apex of the tube bundle). However, the tube rupture break flow was conservatively calculated assuming that the break is at the top of the tube sheet. The water level relative to the top of the tubes in the ruptured and intact steam generators is shown in Figure 15.6-71. The iodine scrubbing efficiency is determined by the method suggested by Postma and Tam (NUREG 0409). The iodine scrubbing efficiencies are shown in Figure 15.6-72.

The activity released to the environment by the flashed rupture flow can be written as follows:

$$A_r = \sum_j IA_j \, (1 - eff_j)$$

where:

A_r = total iodine released to the environment by flashed primary coolant

IA_j = (integrated activity in rupture flow during time interval j) (flashing fraction for time interval j)

eff_j = iodine scrubbing efficiency during time interval j

d. The total primary to secondary leak rate is assumed to be 450 gpd ~~1.0 gpm~~ as allowed by the BVPS Unit 2 Technical Specifications. The leak rate is assumed to be ~~0.35 gpm~~ 150 gpd for each of the intact steam generators and ~~0.30 gpm~~ 150 gpd for the ruptured steam generator. The leakage to the intact steam generators is assumed to persist for the duration of the accident.

e. The iodine partition coefficient between the liquid and steam of the ruptured and intact steam generators is assumed to be 0.01.

f. No credit was taken for radioactive decay during release and transport, or for cloud depletion by ground deposition during transport to the site boundary or outer boundary of the low population zone.

g. Short-term atmospheric dispersion factors (χ/Qs) are provided in Table 15.0-~~11~~ 14 and breathing rates are provided in Appendix 15A.

4. Offsite Dose Calculation Models

The models used to calculate the offsite thyroid CDE, whole-body EDE ~~gamma~~, and beta-skin doses are presented in Appendix 15A.

5. Results

~~The calculated nuclide releases resulting from an SGTR are presented in Table 15.6-6 for the pre-accident iodine spike case and in Table 15.6-7 for the accident initiated iodine spike case.~~ Thyroid and whole-body ~~gamma, and beta-skin~~ doses at the Exclusion Area Boundary and Low Population Zone are presented in Table 15.0-12. All doses are within the allowable guidelines as specified by Standard Review Plan 15.6.3 and 10CFR100.

15.6.5.4 Radiological Consequences

A LOCA would increase the pressure in the containment initiating containment isolation, auxiliary feedwater, emergency core cooling, and containment spray. Normal ventilation in the auxiliary and contiguous buildings is realigned and the ESF areas are aligned and exhausted by the supplementary leak collection and release system (SLCRS). However, no credit is taken for containment leak collection and filtration prior to release to the environment. The main control room environment is ensured by immediate isolation of intake air, by a containment isolation Phase B (CIB) signal, and a supply of clean air from an emergency bottled air system. One hour after the LOCA, when the containment has returned to a subatmospheric condition, the control room ventilation system provides filtered intake air to the control room. The post-accident monitoring system (Section 7.5) is available for monitoring important post-LOCA parameters.

Doses from the LOCA are calculated for the power plant operators located in the control room as well as at locations on the exclusion area boundary and low population zone (LPZ) outer boundary. The doses are due to leakages from the containment building and ECCS, in addition to the direct shine from the containment building and other emergency systems.

Containment Leakage Source

For a LOCA, it is postulated that 100 percent of the noble gas inventory and 25 percent of the iodine inventory in the core after full power operation for ~~650~~ 1500 days are available for release from the containment atmosphere. The containment structure is assumed to leak at the design basis leak rate of 0.1 volume percent per day for the first hour after the accident. Within the hour the containment is brought to subatmospheric pressure, precluding any further leakage.

The iodine concentration in the containment atmosphere is reduced by a caustic spray additive which is injected into the containment by the quench spray system (QSS). The QSS is activated by the CIB signal and provides caustic spray to the nozzles within 90 seconds of accident initiation. A two region model is used to evaluate the effect of the spray on the concentration of iodine in the containment atmosphere. (Section 6.2.2.3.3). This model accounts for mixing between the sprayed and unsprayed regions of the containment.

Emergency Core Cooling System Leakage Source

In the event of a LOCA, emergency core cooling will be initiated. It is postulated that leakage will occur from the ECCS outside containment in the safeguards area, the auxiliary building, and the rod control building. A combined leakage rate in the buildings is expected to be 9.4×10^{-3} gpm, as shown in Table 15.6-12. Fifty percent of the core iodine inventory is assumed mixed in the sump water that is circulated through the piping external to the

The environmental releases resulting from the LOCA, including ECCS leakage, ~~presented in Table 15.6-13,~~ are used in conjunction with the atmospheric dispersion values given in Table 15.0-~~11~~ 14 to calculate the offsite and control room doses using the methodology discussed in Appendix 15A.

The potential dose to the plant operators in the control room is due to inleakage into the control room of the external cloud, and direct doses from 1) immersion in the external cloud, and 2) radiation sources in the containment and emergency systems located outside containment including the intake filters in the control room. The operating procedures for the control room ventilation system are described in Section 6.4. The accident analysis considers a conservative post-accident ventilation rate in the control room to evaluate the thyroid dose from ~~inhalation.~~ Since the unfiltered in-leakage into the control room ~~during isolation~~ is the main contributor to the inhalation dose, a ~~minimum air clean~~ up rate is assumed. Specific control room shielding details, which limit the 30 day exposure to control room personnel, are described in Section 12.3.

The control room walls provide the necessary shielding to protect personnel from the external cloud due to containment building and ECCS leakage. Conversion factors were developed, as presented in Appendix 15A, to calculate the control room operator dose due to these sources. Parameters required to calculate the control room doses are provided in Tables 15.6-~~11~~ 12 through 15.6-14.

The total doses at the exclusion area boundary and the LPZ, presented in Table 15.0-12 are within the guidelines of 10 CFR 100. The dose to the BVPS-2 control room operators due to a LOCA at the BVPS-2 plant, as presented in Table 15.0-13, is below the limit set in General Design Criterion 19 of 5 Rem whole body, or its equivalent to any part of the body.

16. DLW-91-159, "DLCo BVPS Units 1&2 ECCS Evaluation Model Changes," Steinmetz (W) to Noonan, (DLCo), June 20, 1991.

17. DLW-93-202, "DLCo BV Units 1&2 10 CFR 50.46 Notification and Reporting Information," Hall (W) to Tonet (DLCo), January 29, 1993.

18. WCAP-10484 Addendum 1; "Spacer Grid Heat Transfer Effects During Reflood," D. J. Shimeck, December 1992.

19. ET-NRC-92-3746, "Extension of NUREG-0630 Fuel Rod Burst Strain and Assembly Blockage Models to High Fuel Rod Burst Temperatures," Liparulo (W) to NRC Document Control Desk, September 16, 1992.

20. WCAP-8341-P, WCAP-8342-NP, "Westinghouse ECCS Evaluation Model Sensitivity Studies," July 1984.

21. WCAP-8340-P, WCAP-8356-NP, "Westinghouse ECCS - Plant Sensitivity Studies," July 1974.

22. NS-EPR-2538, E. P. Rahe (W) to R. L Tedesco and T. P. Speis (NRC), "Concerning Maximum Safety Injection in the LBLOCA Analysis," December 22, 1981.

23. Meyer, P. E., "NOTRUMP - A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A, (Proprietary), WCAP-10080-NP, (Non-Proprietary), August 1985.

24. Lee, N. et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A, (Proprietary), August 1985.

25. NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants."

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

~~27. SWEC, calculation ERS-S&W-92-012, "Doses to the Combined BV1/BV2 Control Room and the EAB and LPZ Due to Release from the RWST Via ECCS Leakage Following a LOCA at Beaver Valley Unit 2," dated 4/92.~~

27 ~~28.~~ Schrader, K. J., Holderbaum, D. F., Lewis, R. N., Marmo, C.A., and Rubin, K., "LOFTTR2 Analysis for A Steam Generator Tube Rupture for Beaver Valley Power Station Unit 2," WCAP-12737 (Proprietary)/WCAP-12738 (Non-Proprietary), October 1990.

28 ~~29.~~ Postma, A. K., Tam, P. S., "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture", NUREG-0409.

29 ~~30~~. Eckerman, K. F., et al., "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Federal Guidance Report No. 11, USEPA, EPA-520/1 88-020.

30 ~~31~~. Schrader, K. J., Holderbaum, D. F., Lewis, R. N., Marmo, C.A., and Rubin, K., "LOFTTR2 Analysis for A Steam-Generator Tube Rupture for Beaver Valley Power Station Unit 2," WCAP-12737 (Proprietary)/WCAP-12738 (Non-Proprietary), October 1990.

31 ~~32~~. Postma, A. K., Tam, P. S., "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture", NUREG-0409.

32. SWEC Calculation 12241 US(B)-182, Maximum Mass Releases due to Letdown Line Break Assuming Manual Line Isolation in 15 Minutes for Dose Calculations.

33. DLC Calculation ERS-AJL-99-017, Safety Analysis of the Radiological Consequences of a Steam Generator Tube Rupture DBA at Unit 2, Common Control Room, EAB and LPZ Doses (Based on WCAP 12738).

34. DLC Calculation ERS-JTL-99-016, Safety Analysis of the Radiological Consequences of a Small Line Break (Primary System Outside of Containment) DBA at BVPS Unit 2, Control Room, EAB and LPZ Doses.

35. SWEC Calculation 02695.3025 US(B) ERS-SNW-92-0 Iodine Release From the Beaver Valley Unit 2 Refueling Water Storage Tank.

36. SWEC Calculation 12241/11700 UR(B)-481, Radiological Dose Consequences at the EAB, LPZ, and in the Control Room due to a Postulated LOCA at Beaver Valley Power Station, Unit 2.

TABLE 15.6-2

PARAMETERS USED FOR THE
SMALL LINE CARRYING
PRIMARY COOLANT FAILURE

Characteristics	~~Expected~~	~~Technical Specification~~
Power (MWt)	~~2,766~~	~~2,766~~ 2705
~~Fraction of failed fuel~~	~~0.0012~~	~~0.0026~~
Line failure	Letdown line ~~to~~ from ~~inlet of~~ letdown heat exchanger	
Break size (in)	~~2~~	2
Time required to detect and isolate failure (min)	~~15~~	15
Primary coolant release rate from break $\left(\frac{\text{lb m}}{\text{sec}}\right)$	~~20~~	~~20~~ 16.0
Temperature of released primary coolant (°F)	~~287~~	549
Fraction of iodine assumed airborne from pipebreak	~~0.1~~	~~0.4~~ 0.38
Supplementary leak collection and release system iodine filter efficiency (%)	~~95~~	0
Primary coolant concentrations	~~Table 11.1-2~~	Table 15.0-8b
Iodine spiking - release rate~~s~~ calculation methadology (assumed to occur for duration of accident)	~~Table 15.0-10~~	Table 15.0-10a

TABLE 15.6-3

SMALL LINE FAILURE
RELEASES TO THE ENVIRONMENT

DELETE

Nuclide	Releases (Ci)
Kr-83m	9.2×10^{-1}
Kr-85m	4.5
Kr-85	2.4×10^{1}
Kr-87	2.6
Kr-88	6.9
Kr-89	2.2×10^{-1}
Xe-131m	2.3×10^{-1}
Xe-133m	6.6
Xe-133	5.6×10^{1}
Xe-135m	3.6
Xe-135	7.0
Xe-137	3.5×10^{-1}
Xe-138	1.5
I-131	1.2×10^{1}
I-132	2.0×10^{1}
I-133	2.7×10^{1}
I-134	2.7×10^{1}
I-135	2.3×10^{1}

<included by reference>

TABLE 15.6-5b

PARAMETERS USED IN EVALUATING
RADIOLOGICAL CONSEQUENCES OF A
STEAM GENERATOR TUBE RUPTURE

I. Source Data

A. Core power level, MWt	~~2766~~ 2,705
B. Total steam generator tube leakage, prior to accident, ~~gpm~~ gpd	~~1.0~~ 450
C. Reactor coolant iodine activity:	
1. Accident Initiated Spike	The initial RC iodine activities based on 0.35 ~~1~~ µCi/gram of D.E. I-131 are presented in Table 15.0-8.b The iodine appearance ~~rates~~ rate calculation methodology ~~assumed~~ for the accident initiated spike ~~are~~ is presented in Table 15.0-10 a
2. Pre-Accident Spike	Primary coolant iodine activities based on 21 ~~60~~ µCi/gram of D.E. I-131 are presented in Table 15.0-9 a
3. Noble Gas Activity	~~The initial RC noble gas activities based on 0.26% fuel defects are presented in~~ Table 15.0-8
D. Secondary system initial activity	Dose equivalent of 0.1 µCi/gm of I-131, presented in Table 15.0-8 b
E. Reactor coolant mass, ~~grams~~ lbm	~~1.91 x 10^8^~~ 3.887E+05
F. Initial steam generator water mass (each), ~~grams~~ lbm* Liquid Steam	~~4.5 x 10^7^~~ 103,172 -10% 6,534 +10%
G. Offsite power	Lost at time of reactor trip
H. Primary-to-secondary leakage duration for intact SG, hrs.	8
I. Species of iodine	100 percent elemental

*Corresponds to bounding conditions for this accident.

TABLE 15.6-5b

II. Activity Release Data	
A. Ruptured steam generator	
1. Rupture flow	See Table 15.6-5a
2. Rupture flow flashing fraction	See Figure 15.6-70
3. Iodine scrubbing efficiency	See Figure 15.6-72
4. Total steam release, lbs.	See Table 15.6-5a
5. Iodine partition factor	0.01
6. Location of tube rupture	Intersection of outer tube row and upper anti-vibration bar
B. Intact steam generators	
1. Total primary-to-secondary leakage, ~~gpm~~ gpd (150 gpd @)	~~0.7~~ 300
2. Total steam release, lbs	See Table 15.6-5a
3. Iodine partition factor	0.01
C. Condenser	
1. Iodine partition factor	0.01
D. Atmospheric Dispersion Factors	See Table 15.0-~~11~~ 14

TABLE 15.6-6

ENVIRONMENTAL RELEASES DUE TO A
STEAM GENERATOR TUBE RUPTURE
FOR PRE-ACCIDENT IODINE SPIKE CASE

DELETE

Nuclide	Total Releases (Ci) 0-2 hr	0-8 hr
Kr-83m	5.7	5.7
Kr-85m	3.1E1	3.1E1
Kr-85	1.7E2	1.7E2
Kr-87	1.6E1	1.6E1
Kr-88	4.6E1	4.6E1
Kr-89	1.7E-1	1.7E-1
Xe-131m	1.7	1.7
Xe-133m	4.8E1	4.8E1
Xe-133	4.1E2	4.1E2
Xe-135m	8.2	8.2
Xe-135	4.9E1	4.9E1
Xe-137	3.2E-1	3.5E-1
Xe-138	5.4	5.4
I-131	6.5E1	6.6E1
I-132	2.1E1	2.1E1
I-133	1.0E2	1.0E2
I-134	1.2E1	1.2E1
I-135	5.3E1	5.4E1

<included by reference>

TABLE 15.6-7

ENVIRONMENTAL RELEASES DUE TO A STEAM GENERATOR TUBE RUPTURE FOR ACCIDENT INITIATED IODINE SPIKE CASE

DELETE

Nuclide	Total Releases (Ci) 0-2 hr	0-8 hr
Kr-83m	5.7	5.7
Kr-85m	3.1E1	3.1E1
Kr-85	1.7E2	1.7E2
Kr-87	1.6E1	1.6E1
Kr-88	4.6E1	4.6E1
Kr-89	1.7E-1	1.7E-1
Xe-131m	1.7	1.7
Xe-133m	4.8E1	4.8E1
Xe-133	4.1E2	4.1E2
Xe-135m	8.2	8.2
Xe-135	4.9E1	4.9E1
Xe-137	3.2E-1	3.5E-1
Xe-138	5.4	5.4
I-131	9.9	1.1E1
I-132	1.6E1	1.7E1
I-133	2.2E1	2.4E1
I-134	2.2E1	2.2E1
I-135	1.9E1	2.1E1

<included by reference>

TABLE 15.6-11

PARAMETERS USED FOR THE LOCA-CLOUD ANALYSIS

Parameters	
Power level (MWt)	~~2,766~~ 2705
Operating time (days)	~~650~~ 1500
Core inventory	Table 15.0-~~7~~b
Iodine reduction factor due to effects of plateout	0.5
Core inventory available for release from containment following plateout (%)	
Noble gases	100
Iodine	25
Iodine composition (%)	
Elemental	91
Particulate	5
Organic	4
Containment free volume (ft^3)	~~$1.8x10^6$~~ 1.71E+06
Sprayed region (~~78%~~) 79%	~~$1.4x10^6$~~
Unsprayed region (~~22%~~) 21%	~~$4.0x10^5$~~
Spray effective time (sec)	~~83~~ 85.5
Volume mixing rate (hr^{-1})	2
Containment leak rate (%/day)	0.1
Elemental iodine decontamination factor	100
Iodine removal coefficient (hr^{-1})	
Elemental	10
Particulate	~~0.65~~ ~~0.44~~ 0.825
Organic	0
Duration of containment leakage (hr)	1.0

TABLE 15.6-12

PARAMETERS USED FOR EMERGENCY CORE COOLING SYSTEM LEAKAGE ANALYSIS

Parameters	
Leak initiation time sec (~~min~~)	~~5~~ 716
Leak rate (gpm)	
5 min to 30 days (Doubled in the analysis)	~~9.4x10^-3^~~ 0.01
Fraction of core iodine inventory in sump water	0.50
Sump water volume (gal)	~~8.3x10^5^~~ 935,654
Peak ESF fluid temperature ~~Sump water temperature at start of recirculation spray~~ (°F)	~~280~~ 217.8
Iodine released to building atmosphere due to flashing (%)	10
Supplementary leak collection and release iodine filter efficiency (%)	95
Leak rate into RWST (gpm) 30 min to 30 days (Doubled in the analysis)	1.0
Release from RWST to environment starts at (~~hr~~) sec	~~2~~ 8800

TABLE 15.6-13

ENVIRONMENTAL RELEASES DUE TO A
LOSS-OF-COOLANT ACCIDENT

DELETE

Nuclide	Releases (Ci) 0-2 Hr	Releases (Ci) 0-30 Days
Kr-83m	4.2×10^{2}	4.2×10^{2}
Kr-85m	1.2×10^{3}	1.2×10^{3}
Kr-85	2.8×10^{1}	2.8×10^{1}
Kr-87	1.9×10^{3}	1.9×10^{3}
Kr-88	3.1×10^{3}	3.1×10^{3}
Kr-89	3.5×10^{2}	3.5×10^{2}
Xe-131m	1.8×10^{1}	2.2×10^{2}
Xe-133m	1.5×10^{2}	3.6×10^{2}
Xe-133	6.7×10^{3}	1.4×10^{4}
Xe-135m	1.4×10^{3}	2.0×10^{3}
Xe-135	2.0×10^{3}	3.9×10^{3}
Xe-137	5.4×10^{2}	5.4×10^{2}
Xe-138	1.9×10^{3}	1.9×10^{3}
I-131	2.0×10^{2}	2.8×10^{2}
I-132	2.6×10^{2}	2.6×10^{2}
I-133	4.5×10^{2}	4.7×10^{2}
I-134	4.0×10^{2}	4.0×10^{2}
I-135	3.8×10^{2}	3.9×10^{2}

<included by reference>

The inlet line rupture analysis assumes that the noble gas inventory released to the environment from the ruptured line is based on the sum of the following: a 1-hour release to the environment of degasifier effluent and the release of a fraction of the radioactivity adsorbed on the charcoal delay beds. The fraction of noble gases released from the charcoal delay beds are given in Table 15.7-2. ~~The total activity released from the ruptured line plus the charcoal delay beds is listed in Table 15.7-3.~~

For the GWST rupture, it is assumed that ~~100 percent of~~ considering holdup and decay, the noble gases produced from the complete degasification of the primary coolant is contained in the seven GWSTs. Since the tanks are not isolated from each other, the rupture of one tank is assumed to cause the release of the contents of all seven tanks. ~~The releases from a GWST rupture are shown in Table 15.7-3.~~ The accident atmospheric dispersion values are given in Table 15.0-~~11~~ 14.

The methodology used in computing the doses is discussed in Appendix 15A. The calculated whole body doses at the exclusion area boundary are presented in Table 15.0-12, and do not exceed 0.5 Rem.

15.7.2 Radioactive Liquid Waste System Leak or Failure (Atmospheric Release)

This section of the Standard Review Plan (NUREG-0800) has been deleted.

15.7.3 Postulated Radioactive Releases Due to Liquid Containing Tank Failures

15.7.3.1 Identification of Causes and Accident Description

The postulated radioactive release due to liquid-containing tank failures is classified as an ANS Condition III event, an incident which may occur during the lifetime of a plant.

All tanks have been qualitatively considered for radioactive releases. The failure and subsequent release of the contents of the tanks with the largest inventory of radioactivity most likely to infiltrate the nearest potable water supply in an unrestricted area were evaluated. The following tanks were considered for release: coolant recovery tank - located on Beaver Valley Power Station - Unit 1 (BVPS-1) and utilized by both BVPS-1 and Beaver Valley Power Station - Unit 2 (BVPS-2), refueling water storage tank (RWST) (Section 6.2.2), waste drain tank (Section 11.2), and the steam generator blowdown hold tank (Section 11.2). The most limiting liquid-containing tank failure postulated is the RWST. Although the coolant recovery tank and the steam generator blowdown hold tank have larger radionuclide inventories than the RWST, a rupture of either of these tanks would not be as limiting since the liquid pathway to the river is through ground water.

15.7.5.2 Analysis of Effects and Consequences

The details of spent fuel cask handling are provided in Section 9.1.5.

15.7.5.3 Radiological Consequences

Since a spent fuel cask drop exceeding 30 feet cannot occur, no radiological analysis need be performed for a spent fuel cask drop accident.

15.7.6 References for Section 15.7

Underhill, D.W. 1972. Effects of Rupture in a Pressurized Noble Gas Adsorption Bed; Nuclear Safety Volume 13 Number 6.

U.S. Nuclear Regulatory Commission (USNRC 1976). Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code). NUREG-0017.

USNRC 1978. Preparation of Radiological Effluent Technical Specification for Nuclear Power Plants. NUREG-0133.

USNRC 1981. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (formerly issued as NUREG 75/087). NUREG-0800.

USNRC 1988. Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors. NUREG/CR-5009.

DLC 1989. Unit 2 Fuel Handling Accident Doses at EAB, LPZ, Common Control Room. ERS-SFL-89-019.

DLC Calculation, ERS-JTL-99-008, Safety Analysis of the Radiological Consequences of a Waste Gas System Rupture at BVPS Unit 2, Control Room, EAB and LPZ Doses.

TABLE 15.7-1

PARAMETERS USED FOR THE
WASTE GAS SYSTEM FAILURE

	~~Expected~~	~~Design~~
Power (MWt)	~~2,766~~	~~2,766~~ 2,705
Fraction of fuel with defects	~~0.0012~~	0.01
Reactor coolant concentrations	~~Table 11.1-2~~	~~Table 11.1-2~~
Letdown flow (gpm)	~~60~~	~~120~~ 135
Charcoal delay beds holdup times (~~days~~ hrs)		
Kr	~~2.6~~	~~0.4~~ 9.6
Xe	~~46~~	~~7.26~~ 174
Fraction of noble gases released from charcoal delay beds	~~Table 15.7-2~~	Table 15.7-2
Duration of release for the inlet line rupture (hrs)	~~1~~	1

TABLE 15.7-2

FRACTIONS OF NOBLE GASES RELEASED FROM CHARCOAL DELAY BEDS

Nuclide	Expected - Design
Kr-83m	0.650
Kr-85m	0.353
Kr-85	0.0444
Kr-87	0.786
Kr-88	0.502
Kr-89	1.000
Xe-131m	0.00739
Xe-133m	0.0361
Xe-133	~~0.016~~ 0.0156
Xe-135m	1.000
Xe-135	0.195
Xe-137	1.000
Xe-138	1.000

TABLE 15.7-3

WASTE GAS SYSTEM FAILURE
ENVIRONMENTAL RELEASES

DELETE

Nuclide	Line Rupture Releases (Ci)	Tank Rupture Releases (Ci)
Kr-83m	3.2×10^{1}	5.2
Kr-85m	1.6×10^{2}	3.2×10^{2}
Kr-85	3.2×10^{2}	2.0×10^{4}
Kr-87	8.1×10^{1}	2.3
Kr-88	2.5×10^{2}	1.8×10^{2}
Kr-89	3.0	
Xe-131m	4.3	
Xe-133m	2.2×10^{2}	
Xe-133	1.3×10^{3}	
Xe-135m	4.1×10^{1}	
Xe-135	3.2×10^{2}	
Xe-137	4.9	
Xe-138	2.5×10^{1}	

<included by reference>

APPENDIX 15A

DOSE METHODOLOGY

The radiological consequences of the design basis accidents (DBAs) are represented by the calculated results of thyroid doses, whole-body gamma doses, and beta skin doses at the exclusion area boundary (EAB), the low population zone (LPZ), and the main control room. The doses at the EAB are based on the release of radionuclides over a period of 2 hours following the occurrence of a postulated accident. For accidents lasting beyond 2 hours, doses are calculated for the LPZ based on releases over the duration of the accident, up to 30 days following the occurrence of an accident. The control room dose is based on releases over a 30-day period following the loss-of-coolant accident (LOCA).

15A.1 Original Licensing Basis

Thyroid doses are calculated based on Regulatory Guide 1.4, June 1974 and the following equation:

$$D_{thy} = \sum_i Q_i\ (\chi/Q)\ (B.R.)\ (C_{thy_i}) \qquad (15A\text{-}1)$$

where:

D_{thy} = thyroid dose (Rem)

Q_i = activity of iodine isotope i released (Ci)

χ/Q = atmospheric dispersion factor (sec/m^3)

B.R. = breathing rate (m^3/sec)

C_{thy_i} = thyroid dose conversion factor for iodine isotope i (Rem/Ci) (DiNunno et al 1962)

The χ/Q values presented in Table 15.0-~~11~~ 14 were calculated using the methodology described in Section 2.3. For persons offsite, the breathing rates are assumed to be:

3.47×10^{-4} m^3/sec, 0 to 8 hours
1.75×10^{-4} m^3/sec, 8 to 24 hours
2.32×10^{-4} m^3/sec, >24 hours

These values are taken from Regulatory Guide 1.4.

External whole-body gamma doses and beta skin doses are calculated using Equations 15A-2 and 15A-3 derived from equations in Regulatory Guide 1.4.

QADMOD

Program QADMOD calculates dose rates at a series of detector locations with shielding for a number of different source points representing volumetric sources. The program is a modified version of the QAD P-5 program written at the Los Alamos Scientific Laboratory by R. E. Malenfant. This program has been upgraded to include: 1) the FASTER geometry routines, 2) a point source option, 3) a translated cylindrical source volume option, and 4) internal library data for conversion factors, build-up factor coefficients, and mass attenuation factors for several materials and compositions.

Insert

15A.1 References for Section 15A

DiNunno, J. J.; Anderson, F. D.; Baker, R. E.; and Waterfield, R.L. 1962. Calculation of Distance Factors for Power and Test Reactor Sites, TID 14844.

U.S. Atomic Energy Commission (USAEC) 1974. Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors. Regulatory Guide 1.4, Revision 2.

Page 15.A-5 Insert

TRAILS_PC

Program TRAILS_PC performs calculations involving transport of radioactive species between compartments that are related by first order linear processes. It is specifically structured to evaluate the transport of radioactivity in design basis accidents, and for calculating dose rates and doses at a user defined offsite location and in the control room. This code was developed and tested at BVPS and has been benchmarked against the SWEC DRAGON code.

PERC2

Program PERC2 is identical to DRAGON in terms of the environmental transport and dose conversion, but it includes the following:

- Provision of time-dependent releases from the reactor coolant system to the containment atmosphere
- Provision for airborne radionuclides other than noble gas and iodine, including daughter ingrowth.
- Provision for calculating organ doses other than thyroid.
- Provisions for tracking time-dependent inventories of all radionuclides in all control regions of the plant model.
- Provision for calculating energies as well as activities for the inventoried radionuclides to permit direct equipment qualification and vital area access assessment.

(NOTE: These provisions are necessary to treat the kind of source term described in NUREG-1465, a methodology not employed in BVPS analyses.)

ORIGEN

Program ORIGEN, calculates fuel depletion, actinide transmutation, fission product buildup and decay and associated radiation source terms. At BVPS, ORIGEN has been used to develop reactor core inventory, and decayed inventories after various cool down times. These values are used in design basis radiological consequence analyses. This code was developed for the NRC at the Oak Ridge National Laboratory. This code is documented as part of the SCALE package in NUREG/CR-0200.

U.S. Atomic Energy Commission (USAEC) 1974. Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors. Regulatory Guide 1.4, Revision 2.

Kocher, DC, External Dose-Rate Conversion Factors for Calculation of Dose to the Public, DOE/EH-0070, 1988

Eckerman, K. F., etal, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, EPA-520/1-88-020, 1988

ICRP, Recommendations of the International Commission of Radiological Protection, ICRP Publication 26, 1977

ICRP, Limits for Intakes of Radionuclides by Workers, ICRP Publication 30, 1979

Murphy, K. G. and Campe, K. W., Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19, published in proceedings of 13th AEC Air Cleaning Conference

DLC Calculation ERS-SFL-96-004, TRAILS_PC: Transport of Radioactive Material in Linear Systems, PC Version

DLC Calculation ERS-SFL-96-017 ASCOT_PC: Assessment of Containment Transport, PC Version

DLC Calculation ERS-SFL-96-001, QAD/CGGP_PC, a Point Kernel Photon Shielding Code With Combinatorial Geometry and Geometric Progression Buildup Factors

DLC Calculation ERS-SFL-88-020, Combinatiorial Geometry Point Kernal Photon and Neutron Shielding Code, QAD-CG, DLC Version 1.0

US NRC NUREG/CR-0200, ORIGEN-S: Scale System Module to Calculate Fuel depletion, Actinide Transmutation, Fission Product Buildup and Decay, and associated Radiation Source Terms

Table 15A-1a

FLUX-TO-DOSE CONVERSION FACTORS
(Used in 1999 (SWEC PERC2 code) reanalysis of Unit 2 LOCA)

Energy Mean MeV	(rem/hr)/ (MeV/cm^2-s)
0.01	3.96E-04
0.025	3.20E-05
0.0375	1.05E-05
0.03	1.94E-05
0.0575	4.67E-06
0.085	3.11E-06
0.125	2.61E-06
0.225	2.51E-06
0.375	2.49E-06
0.4	2.46E-06
0.575	2.29E-06
0.5	2.36E-06
0.85	2.07E-06
1.25	1.86E-06
1.75	1.67E-06
2.25	1.54E-06
2.75	1.44E-06
4.25	1.23E-06
3.5	1.32E-06
5	1.16E-06
7.5	1.02E-06
7	1.04E-06
9.5	9.62E-07

ATTACHMENT B

Beaver Valley Power Station, Unit Nos. 1 and 2
License Amendment Request Nos. 280 and 151
UFSAR Update for Revised Radiation Dose Calculations

A. DESCRIPTION OF AMENDMENT REQUEST

The proposed amendment would revise the Beaver Valley Power Station (BVPS) Units 1 and 2 calculated doses and associated descriptions/information listed in the UFSAR for the Design Basis Accidents (DBAs). An evaluation of all of the BVPS Unit 1 and 2 dose calculations was completed which reviewed the input parameter values, the input assumptions, and the methodology utilized. The resultant DBA dose calculation revisions necessitate associated revisions to the UFSAR. Additionally, some changes were made in response to Generic Letter 99-02.

This change is not to plant hardware. It is intended to reflect revised analyses results only and is necessary to allow correction of the licensing basis to reflect conservative assumptions used in the revised dose analysis for the DBAs listed in the Unit 1 and Unit 2 UFSARs.

The UFSAR changes are identified in Attachments A-1 and A-2.

This amendment is being requested in accordance with 10 CFR 50.59(c). The differences between the present analyses of record and the new analyses are sufficiently significant such that NRC review is warranted. Several analyzed dose values have increased for several DBAs. The increases in calculated dose values and changes to analysis methodology for each DBA is described below in Part C.

B. DESIGN BASES

The BVPS Unit 1 Design Basis Accidents are described in the Unit 1 UFSAR. The radiological dose calculation information for the postulated DBAs are described throughout Chapters 11 and 14 of the UFSAR. Specific radiological dose information is calculated and provided for the following DBAs: Loss of Offsite AC Power (Section 14.1.11), Fuel Handling Accident (Section 14.2.1), Accidental Release of Waste Gas (Section 14.2.3), Steam Generator Tube Rupture (Section 14.2.4), Major Secondary System Pipe Rupture (Section 14.2.5), Rod Cluster Control Assembly Ejection (Section 14.2.6), Single Reactor Coolant Pump Locked Rotor (Section 14.2.7), and Loss of Reactor Coolant from Small Ruptured Pipes/Loss of Coolant Accidents (Section 14.3).

The BVPS Unit 2 Design Basis Accidents are described in the Unit 2 UFSAR. The radiological dose calculation information for the postulated DBAs are described throughout Chapters 11 and 15 of the UFSAR. Specific radiological dose information is calculated and provided for the following DBAs: Steam System Piping Failures (Section 15.1.5), Loss of AC Power (Section 15.2.6), Reactor Coolant Pump Shaft Seizure (Section 15.3.3), Rod Cluster Control Assembly Ejection (Section 15.4.8), Failure of Small Lines Carrying Primary Coolant Outside Containment (Section 15.6.2), Steam Generator Tube Rupture (Section 15.6.3), Loss of Coolant Accidents (Section 15.6.5), Waste Gas System Failure (Section 15.7.1), and Fuel Handling Accidents (Section 15.7.4).

C. JUSTIFICATION

Information Common to Both Units' Radiological Dose Analyses

The proposed revisions to the BVPS Unit 1 and 2 UFSARs as a result of revised radiological dose calculations are a result of an extensive review re-assessing the dose calculations' input parameter values, input assumptions, consistency with current design basis, calculation methodologies and conservatisms. This action was taken as a follow-up extent of condition action described in BVPS Unit 2 Licensee Event Report (LER) 97-008, Revision 1, issued on March 30, 1998. Following the changes, several DBA dose values were determined to be above the dose values currently listed in the UFSAR. These proposed changes to the BVPS Unit 1 or Unit 2 UFSAR have been determined to be an Unreviewed Safety Question pursuant to 10 CFR 50.59. Since this extensive evaluation and the resulting changes have caused some DBA dose values to increase and other dose values to decrease, this submittal will address each UFSAR DBA which calculates a dose value.

Reactor core inventory was recalculated using updated fuel and operating parameters. This, as well as other parameter updates, supported recalculation of primary coolant, secondary coolant and secondary steam design and Technical Specification limits activity concentration. This work was performed using the methodology of NUREG/CR-0200, ORIGEN and SWEC proprietary ACTIVITY computer codes. Previously, the only Unit 1 accident that used the ORIGEN methodology for source term calculation was the Fuel Handling Accident. Other Unit 1 analyses used ACTIVITY methodology exclusively. Previously, the only Unit 2 accident that used the ORIGEN methodology for source term calculation was the Main Steam Line Break. Other Unit 2 analyses used ACTIVITY methodology exclusively.

Recent changes to the Technical Specification for reactor coolant activity limits were considered in all accident analyses where this Technical Specification parameter was used as a source term. The recent changes for reactor coolant activity limits were associated with steam generator alternate repair criteria (NRC Generic Letter 95-05) and were implemented via License Amendment No. 205 for Unit No. 1 and via License Amendment No. 101 for Unit No. 2.

Revised dose conversion factors/dose quantities and offsite atmospheric dispersion factors (chi over Q – only where applicable) were used for all accidents. Previously at Unit 1, only the more recent Main Steam Line Break Accident analysis used these revised dispersion factors. These revised dose conversion factors and atmospheric dispersion factor methodologies were reviewed and accepted by NRC for use at Unit 1 in License Amendment No. 205 in the Main Steam Line Break analysis pursuant to implementing the steam generator alternate repair criteria. Previously at Unit 2, the more recent Main Steam Line Break and Locked Rotor Accident analyses used these. These methodologies were reviewed and accepted by NRC for use at Unit 2 in Unit 2 License Amendment Nos. 101 and 103.

NRC Generic Letter 99-02 provides for a safety factor of ≥ 2, relating organic iodine adsorption by charcoal filters and allowable test penetration. This provision is not currently met for accidents which take credit for organic iodine removal by BVPS Unit 1 SLCRS. This license amendment request includes appropriate changes to the BVPS Unit 1 dose calculations to reduce the SLCRS filtration efficiency as discussed in FENOC Letter L-00-046 dated May 12, 2000, which transmitted License Amendment Request 263/138 Revision 1 for BVPS Unit 1 and 2, respectively.

The offsite skin doses (LPZ & EAB*) currently listed in the Unit 1 and Unit 2 UFSAR will be deleted. There is no 10 CFR 100 limit nor a Standard Review Plan limiting criterion for calculated skin dose quantity following a design basis accident. Thus these values will be deleted from the UFSAR, though they were calculated in the recent reevaluation of BVPS DBA radiation dose calculations. In addition, all references to realistic analyses will be deleted as these analyses are not required as provided in Regulatory Guide 1.70, Section 15.

Consistent with recent similar information received in other NRC license amendment safety evaluation reports for BVPS, the BVPS Unit 1 and Unit 2 UFSAR will only list a threshold dose value when the calculated value is determined to be very small. For all design basis accidents, if the calculated dose does not exceed the threshold value, the following values will be listed in the UFSAR:

* LPZ: Low Population Zone; EAB: Exclusion Area Boundary

Control Room (CR) Dose:	Thyroid: Whole Body: Skin:	<1E+00 rem <2E-01 rem <1E+00 rem
Exclusion Area Boundary (EAB) Dose & Low Population Zone (LPZ) Dose:	Thyroid: Whole Body:	<1E+00 rem <1E-01 rem

In addition, all doses above the threshold will be listed with two significant digits.

The methodology changes described above are those that generally affect all or most of the DBA calculations. Other methodology changes unique to the specific DBA analysis are described below. Specific details are provided in the marked-up UFSAR pages in Attachment 1.

Information Specific To Unit 1's Radiological Dose Analyses

Loss of Offsite AC Power (UFSAR Section 14.1.1)

Certain parameters and assumptions were changed to reflect corrected or conservative analysis input parameter values or input assumptions on plant design and operation. No new analysis methodology was used which had not been previously reviewed and approved by the NRC for BVPS Unit 1.

The revised analysis did not show an increase in dose for this accident.

Fuel Handling Accident (UFSAR Section 14.2.1)

Dose	Location	UFSAR Current Dose	UFSAR Revised Dose	10 CFR Limit
Thyroid	EAB	14.6	2.5E+01	300
Thyroid	CR	3.2	6.3E+00	30

Certain parameters and assumptions were changed to reflect corrected or conservative analysis input parameter values or input assumptions on plant design and operation. No new analysis methodology was used which had not been previously reviewed and approved by the NRC for BVPS Unit 1. Because this accident takes credit for organic iodine removal by SLCRS, a reanalysis was performed to add the Generic Letter 99-02 safety factor of ≥ 2. As a consequence of this, the doses increased.

Control room operator dose increased from 3.2 to 6.26 rem, and EAB thyroid dose increased from 14.6 to 24.6 rem. These are reported in the UFSAR revision as "6.3E+00 rem" and 2.5E+01 rem." This increase may be attributed to use of the Generic Letter safety factor of ≥ 2.

Accidental Release of Waste Gas (UFSAR Section 14.2.3)

Dose Increases as a result of revised radiological analyses:

Dose	Location	UFSAR Current Dose	UFSAR Revised Dose	10 CFR Limit
Skin (line break)	CR[#]	<1.0 rem	3.9 E+00 rem	30.0 rem
Whole Body (line break)	CR	<1.0 E-02 rem	<2E-01 rem	5.0 rem

The revised calculated doses are well within the applicable regulatory limit.

Certain parameters and assumptions were changed to reflect corrected or conservative analysis input parameter values or input assumptions on plant design and operation. In addition, some analysis methodology was changed. The offsite dose analysis was previously based on volume control tank and waste gas surge tank ruptures. This was original licensing basis. A control room analysis was performed circa 1987 for Unit 2 (common control room) licensing action; however, the offsite analysis was not updated. Consequently, the offsite radiological analysis was changed to conform to the control room dose analysis. The new analysis follows guidance provided in NUREG-0800 (ETSB 11-5) and Regulatory Guide 1.25. To ensure that the revised analysis represents the bounding case for a Waste Gas System Rupture, the radioactivity release quantities for the volume control tank and gas surge tank were adjusted for the source term revisions discussed above and compared to the revised analysis. The release point to the environment for the volume control tank and gas surge tank ruptures is the same as that for the new, limiting line rupture case. The release quantities for the revised analysis are higher; therefore, the accident consequences are bounded. Methodology was also changed for the control room analysis. This change was limited to using a more conservative method for calculating the environmental release radiological source term. This also now conforms to the accident analysis performed for Unit 2. The new methodology is summarized in the attached UFSAR markups.

[#] Control Room

Control room operator beta (skin) dose increased from <1.0 to 3.85 rem, and whole body (gamma) dose increased from <1E-02 to 0.0295 rem. These are reported in the UFSAR revision as "3.9E+00 rem" and "<2E-01 rem". This increase may be attributed, for the most part, to update of plant operating input parameter values, for example, addressing conservative values of reactor coolant and steam generator mass.

<u>*Steam Generator Tube Rupture*</u> *(UFSAR Section 14.2.4)*

Dose Increases as a result of revised radiological analyses:

Dose	Location	UFSAR Current Dose	UFSAR Revised Dose	10 CFR Limit
Skin (coincident spike)	CR	0.0222 rem	<1E+00 rem	30.0 rem
Skin (preincident spike)	CR	0.0234 rem	<1E+00 rem	30.0 rem
Thyroid (coincident spike)	EAB	0.9 rem	1.4E+00 rem	300.0 rem

The revised calculated doses are well within the applicable regulatory limit.

Certain parameters and assumptions were changed to reflect corrected or conservative analysis input parameter values or input assumptions on plant design and operation. In addition, some analysis methodology for the offsite dose analysis was changed. The revised analysis uses the methodology of the current analysis of record for the control room operator dose.

Control room operator beta (skin) dose increased from 0.0222 to 0.0451 rem for the co-incident iodine spike case, and from 0.0234 to 0.0443 rem for the pre-incident iodine spike case. These are reported in the UFSAR revision as "<1E+00 rem", consistent with our current policy for reporting small doses therein. This increase may be attributed, for the most part, to update of plant operating input parameter values.

Exclusion area boundary thyroid dose increased from 0.9 rem to 1.37 rem. This is reported in the UFSAR revision as "1.4E+00 rem". This increase may be attributed to methodology changes employed to conform with current regulatory guidance.

Major Secondary System Pipe Rupture (UFSAR Section 14.2.5)

Dose Increases as a result of revised radiological analyses:

Dose	Location	UFSAR Current Dose	UFSAR Revised Dose	10 CFR Limit
Thyroid (coincident spike)	CR	26.0 rem	2.9E+01 rem	30.0 rem

The revised calculated doses are within the applicable regulatory limit.

Certain parameters and assumptions were changed to reflect corrected or conservative analysis input parameter values or input assumptions on plant design and operation.

Control room operator thyroid dose increased from 26.0 rem to 28.9 rem for the co-incident iodine spike case. This is reported in the UFSAR revision as "2.9E+01" rem, consistent with our current policy for reporting doses therein. This increase may be attributed, for the most part, to update of plant operating input parameter values.

Rod Cluster Control Assembly Ejection (UFSAR Section 14.2.6)

Dose Increases as a result of revised radiological analyses:

Dose	Location	UFSAR Current Dose	UFSAR Revised Dose	10 CFR Limit
Skin	CR	8.9E-3 rem	<1E+00 rem	30.0 rem

The revised calculated dose is well within the applicable regulatory limit.

Certain parameters and assumptions were changed to reflect corrected or conservative analysis input parameter values or input assumptions on plant design and operation. No new analysis methodology was used which had not been previously reviewed and approved by the NRC for BVPS Unit 1.

Control room operator beta (skin) dose increased from 0.009 to 0.014 rem. This is reported in the UFSAR revision as "<1E+00 rem". This increase may be attributed, for the most part, to update of plant operating input parameter values.

Single Reactor Coolant Pump Locked Rotor (*UFSAR Section 14.2.7*)

Certain parameters and assumptions were changed to reflect corrected or conservative analysis input parameter values or input assumptions on plant design and operation. In addition, some analysis methodology was changed. The co-incident iodine spike, previously assumed to occur, is removed from the analysis. This is justified because of the 18% failed fuel assumption. Consistent with License Amendment No. 103 received on Unit 2, these source terms are considered mutually exclusive for analysis purposes. This change does not significantly affect the analysis results.

The revised analysis did not show an increase in dose for this accident.

Loss of Reactor Coolant from Small Ruptured Pipes/Small Line Break (*UFSAR Section 14.3*)

Dose Increases as a result of revised radiological analyses:

Dose	Location	UFSAR Current Dose	UFSAR Revised Dose	10 CFR Limit
Skin	CR	0.026 rem	<1E+00 rem	30.0 rem

The revised calculated doses are well within the applicable regulatory limit.

Certain parameters and assumptions were changed to reflect corrected or conservative analysis input parameter values or input assumptions on plant design and operation. No new analysis methodology was used which had not been previously reviewed and approved by the NRC for BVPS Unit 1.

Control room operator beta (skin) dose increased from 0.026 to 0.0438 rem. This is reported in the UFSAR revision as "<1E+00 rem". This increase may be attributed, for the most part, to update of plant operating input parameter values.

Loss of Coolant Accident (*UFSAR Section 14.3*)

Dose Increases as a result of revised radiological analyses:

Dose	Location	UFSAR Current Dose	UFSAR Revised Dose	10 CFR Limit
Whole Body	CR	0.170 rem	7.1E-01 rem	5.0 rem

The revised calculated doses are well within the applicable regulatory limit.

Certain parameters and assumptions were changed to reflect corrected or conservative analysis input parameter values or input assumptions on plant design and operation. In addition, some analysis methodology was changed. The dose analysis is performed using SWEC PERC2 computer code. This is functionally equivalent to the previously used DRAGON code. An additional control room dose source is included in the revised analysis – shine from the area beneath the control room that is not within the control room ventilation envelope. For this one source, it was necessary to calculate a new atmospheric dispersion factor. The methodology of Murphy and Campe (1974) was used to calculate this one source, consistent with the Unit 2 analysis. Because thyroid dose due to iodine is limiting, this new source has little effect on the accident consequences.

Because this accident takes credit for organic iodine removal (for the ECCS portion of the release) by SLCRS, the potential change in accident consequences was evaluated. It is determined that the analysis assumption used for total iodine removal efficiency is sufficiently conservative such that reanalysis is not necessary. However, the UFSAR description of the accident is changed to recognize that a safety factor of ≥ 2 for organic iodine removal efficiency is required.

Control room operator whole body (gamma) dose increased from 0.170 to 0.71 rem. This is reported in the UFSAR revision as "7.1E-01 rem". This increase may be attributed, for the most part, to the additional source described above.

Information Specific To Unit 2's Radiological Dose Analyses

Steam System Piping Failures *(UFSAR Section 15.1.5)*

Certain parameters and assumptions were changed to reflect corrected or conservative analysis input parameter values or input assumptions on plant design and operation. No new analysis methodology was used which had not been previously reviewed and approved by the NRC for BVPS Unit 2.

The revised analysis did not show an increase in dose for this accident.

Loss of AC Power *(UFSAR Section 15.2.6)*

Certain parameters and assumptions were changed to reflect corrected or conservative analysis input parameter values or input assumptions on plant design and operation. No

new analysis methodology was used which had not been previously reviewed and approved by the NRC for BVPS Unit 2

The revised analysis did not show an increase in dose for this accident.

Reactor Coolant Pump Shaft Seizure (UFSAR Section 15.3.3)

Dose Increases as a result of revised radiological analyses:

Dose	Location	UFSAR Current Dose	UFSAR Revised Dose	10 CFR Limit
Thyroid	CR	1.7 rem	7.5E+00 rem	30.0 rem

The revised calculated dose is well within the applicable regulatory limit.

Certain parameters and assumptions were changed to reflect corrected or conservative analysis input parameter values or input assumptions on plant design and operation. In addition, some analysis methodology was changed. Isolation of the control room is assumed NOT to occur in the revised analysis. Although previously control room isolation was assumed, control room isolation is now deleted both in the analysis as a conservative measure and to simplify the calculation. Note the control room isolation function remains operationally unchanged; it is just not credited in the analysis.

Control room operator thyroid dose increased from 1.7 to 7.46 rem. This is reported in the UFSAR revision as "7.5E+00 rem". This increase may be attributed, for the most part, to removal of the control room isolation assumption.

Rod Cluster Control Assembly Ejection (UFSAR Section 15.4.8)

Dose Increases as a result of revised radiological analyses:

Dose	Location	UFSAR Current Dose	UFSAR Revised Dose	10 CFR Limit
Skin	CR	0.0038 rem	<1E+00 rem	30.0 rem

The revised calculated dose is well within the applicable regulatory limit.

Certain parameters and assumptions were changed to reflect corrected or conservative analysis input parameter values or input assumptions on plant design and operation. No

new analysis methodology was used which had not been previously reviewed and approved by the NRC for BVPS Unit 2.

Control room operator beta (skin) dose increased from 0.0038 to 0.00451 rem. This is reported in the UFSAR revision as "<1E+00 rem". This increase may be attributed, for the most part, to update of plant operating input parameter values.

Failure of Small Lines Carrying Coolant Outside Containment (USFAR Section 15.6.2)

Dose Increases as a result of revised radiological analyses:

Dose	Location	UFSAR Current Dose	UFSAR Revised Dose	10 CFR Limit
Skin	CR	0.0077 rem	<1E+00 rem	30.0 rem

The revised calculated dose is well within the applicable regulatory limit.

Certain parameters and assumptions were changed to reflect corrected or conservative analysis input parameter values or input assumptions on plant design and operation. No new analysis methodology was used which had not been previously reviewed and approved by the NRC for BVPS Unit 2.

Control room operator beta (skin) dose increased from 0.0077 to 0.0106 rem. This is reported in the UFSAR revision as "<1E+00 rem". This increase may be attributed, for the most part, to update of plant operating input parameter values.

Steam Generator Tube Rupture (UFSAR Section 15.6.3)

Dose Increases as a result of revised radiological analyses:

Dose	Location	UFSAR Current Dose	UFSAR Revised Dose	10 CFR Limit
Skin (coincident spike)	CR	0.0061 rem	<1E+00 rem	30.0 rem
Skin (preincident spike)	CR	0.0079 rem	<1E+00 rem	30.0 rem
Skin (preincident spike)	LPZ	0.0050 rem	Deleted	None

The revised calculated doses are well within the applicable regulatory limit.

Certain parameters and assumptions were changed to reflect corrected or conservative analysis input parameter values or input assumptions on plant design and operation. No

new analysis methodology was used which had not been previously reviewed and approved by the NRC for BVPS Unit 2.

Control room operator beta (skin) dose increased from 0.006 to 0.012 rem for the co-incident iodine spike case, and from 0.008 to 0.012 rem for the pre-incident iodine spike case. These are reported in the UFSAR revision as "<1E+00 rem". Low population zone beta (skin) dose increased from 0.0050 rem to 0.0051 rem. This is not reported in the UFSAR revision as there is no regulatory criterion for offsite skin dose limitation for design basis accidents.

Loss of Coolant Accidents (UFSAR Section 15.6.5)

Dose Increases as a result of revised radiological analyses:

Dose	Location	UFSAR Current Dose	UFSAR Revised Dose	10 CFR Limit
Whole Body	CR	0.32 rem	3.3E-01 rem	5.0 rem
Thyroid	CR	1.3 rem	2.0E+00 rem	30.0 rem

The revised calculated doses are well within the applicable regulatory limit.

Certain parameters and assumptions were changed to reflect corrected or conservative analysis input parameter values or input assumptions on plant design and operation. No new analysis methodology was used which had not been previously reviewed and approved by the NRC for BVPS Unit 2.

Control room operator whole body (gamma) dose increased from 0.320 to 0.330 rem, and thyroid dose increased from 1.3 to 2.0 rem. These are reported in the UFSAR revision as "3.3E-01 rem" and "2.0E+00 rem". This increase may be attributed, for the most part, to a change in an analysis assumption. Previous analyses assumed a control room isolation to occur prior to plume arrival. The new analysis assumes instantaneous transport and control room intake between T=0 and isolation.

Waste Gas System Failure (UFSAR Section 15.7.1)

Dose Increases as a result of revised radiological analyses:

Dose	Location	UFSAR Current Dose	UFSAR Revised Dose	10 CFR Limit
Skin (line break)	EAB	0.19 rem	Deleted	None
Skin (tank rupture)	EAB	1.5 rem	Deleted	None

The revised calculated doses are well within the applicable regulatory limit.

Certain parameters and assumptions were changed to reflect corrected or conservative analysis input parameter values or input assumptions on plant design and operation. No new analysis methodology was used which had not been previously reviewed and approved by the NRC for BVPS Unit 2.

Exclusion area boundary beta (skin) dose increased from 0.19 to 0.757 rem for the line break case, and from 1.5 to 4.97 rem for the tank rupture case. However, these will not be reported in the UFSAR revision as there is no regulatory criterion for offsite skin dose limitation for design basis accidents.

Fuel Handling Accidents (UFSAR Section 15.7.4)

The Fuel Handling Accident was revised to support BVPS Unit 2 License Amendment Request (LAR) 2A-155, Revision of Requirements Associated with Containment Closure. This LAR was submitted to the Commission in FENOC Letter L-00-048, dated May 1, 2000. Please reference this LAR for details regarding the dose calculation. [Note it is discussed here solely for completeness of addressing all BVPS dose calculations.]

D. SAFETY ANALYSIS

The proposed revision to the BVPS Unit 1 and 2 UFSARs as a result of revised radiological dose calculations are a result of an extensive review reassessing the dose calculations' input parameter values, input assumptions, consistency with current design basis, calculation methodologies and conservatisms. Following the changes, several DBA dose values were recalculated to be above the dose values currently listed in the UFSAR. These proposed changes to the BVPS Unit 1 or Unit 2 UFSAR have been determined to be an Unreviewed Safety Question pursuant to 10 CFR 50.59. However, each of these increases in calculated DBA dose values remain within the applicable DBA previously approved regulatory limit. In addition, the only higher dose value which approaches a regulatory limit is the BVPS Unit 1 Major Secondary System Pipe Rupture DBA. The dose for BVPS Unit 1 Major Secondary System Pipe Rupture DBA increases from 26.0 rem to 28.9 rem, which remains within the Standard Review Plan limit of 30 rem. This value remains below the Standard Review Plan limit in accordance with the steam generator alternate repair criteria as previously approved via BVPS Unit 1 License Amendment No. 205.

Thus, since each dose increase remains within the applicable DBA previously approved regulatory limit, it is recommended that the proposed UFSAR changes in Attachments A-1 and A-2 be approved for BVPS Unit 1 and Unit 2, respectively.

E. NO SIGNIFICANT HAZARDS EVALUATION

The no significant hazard considerations involved with the proposed amendment have been evaluated. The evaluation focused on the three standards set forth in 10 CFR 50.92(c), as quoted below:

> The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:
>
> (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
>
> (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
>
> (3) Involve a significant reduction in a margin of safety.

The following evaluation is provided for the no significant hazards consideration standards.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Following a reevaluation of the calculated dose values for BVPS Unit 1 and Unit 2 design basis accidents (DBAs) as described in their respective UFSAR, several calculated dose values were identified to be increased. These increases were small and remained within the applicable DBA previously approved regulatory limit.

The increases for each DBA were as a result of revised plant data being used in the dose calculation, revised calculation assumptions, or new methodology. These changes were not the result of plant hardware changes. The changes were intended to ensure that accurate, current and conservative licensing basis information and assumptions were used for DBA dose analyses. The UFSAR changes are proposed to reflect the revised analyses results for the Unit 1 and Unit 2 UFSAR.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   BVPS Unit 1 and Unit 2 calculations which are used to determine DBA calculated dose values were revised. The changes were as a result of revised plant data being used in the dose calculation, revised calculation assumptions or new methodology. The changes were intended to ensure that accurate, current and conservative licensing basis information and assumptions were used for DBA dose analyses. The DBA events themselves remain the same postulated events as previously described within the BVPS Unit 1 and Unit 2 UFSARs. These changes were not the result of plant hardware changes. The changes were only in the calculations. The UFSAR changes are proposed to reflect the revised analyses results for the Unit 1 and Unit 2 UFSAR.

3. Does the change involve a significant reduction in a margin of safety?

   This amendment request addresses only proposed changes to the Unit 1 and Unit 2 UFSAR, which was determined to involve an Unreviewed Safety Question pursuant to 10 CFR 50.59. This request does not propose modifying any Technical Specification criteria. This request proposes that several calculated dose values for BVPS Unit 1 and Unit 2 DBAs be increased following a reevaluation of their design basis calculations. These proposed increases are small and remained within the applicable DBA previously approved regulatory limit.

F. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission has provided guidance concerning the application of standards in 10 CFR 50.92 by providing certain examples (March 6, 1986 51FR7751) of amendments that are considered not likely to involve a significant hazards consideration. The proposed amendment is similar to example (vi) stated in the March 6, 1986 Federal Register Notice, in that this proposed change results in some increase in the consequences of a previously analyzed accident, but where the results of the change are clearly within acceptable criteria. The proposed increases in calculated doses remain within the applicable DBA previously approved regulation limits.

G. ENVIRONMENTAL CONSIDERATION

This license amendment request changes the calculated design basis accident dose values identified in the BVPS Unit 1 and Unit 2 UFSARs. The identified increased dose values remain below the dose requirements of 10 CFR 100 and 10 CFR 50, GDC 19. It has been determined that this license amendment request involves no significant increase in the amounts, and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. This license amendment request may change requirements with respect to installation or use of a facility component located within the restricted area; however, the category of this licensing action does not individually or cumulatively have a significant effect on the human environment. This amendment is necessary to allow correction of the licensing basis to reflect corrected and conservative input and assumptions used in the BVPS Unit 1 and Unit 2 analyses for design basis accidents. Accordingly, this license amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this license amendment request.

H. UFSAR CHANGES

UFSAR changes are required. See Attachments A-1 and A-2.