APPENDIX D

RADIOACTIVE SOURCE BASES

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APPENDIX D – RADIOACTIVE SOURCE BASES

This appendix presents the quantities of radioactive isotopes present in the core, the fuel rod gap, the coolant, the volume control tank, and a gas decay tank. Dose calculation models used in the safeguards analysis are presented. A general discussion of the derivations is also provided.

D.1 TOTAL ACTIVITY IN THE CORE

The total core activity used to perform dose consequence analyses for the design basis Rupture of a Steam Line (Section 14.5.5) and Loss of Coolant Accident (Section 14.9) is given in Table D.1-2 (Reference 23). The values provided reflect the higher reactor core isotopic inventories between an OFA core and a 422V+ (Heavy Bundle) core. The parameters and key assumptions used to calculate the total core inventory are listed below:

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Fuel Parameters

Parameter	Assumed Value
Total Core Thermal Power	1683 MWt
Enrichment	5.0 w/o
Uranium Mass(MTU/assembly)	0.393 (HB) 0.351 (OFA)
Cycle Burn-up	25,000 MWD/MTU
Discharge Assembly Burn-up	75,000 MWD/MTU

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D.2 ACTIVITY IN THE FUEL GAP

RG 1.183, Table 3, specifies Fission Product gap inventories for non-LOCA events. Footnote (11) to Table 3 reads:

> "The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. As an alternative, fission product gas release calculations performed using NRC approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases."

The Prairie Island fuel management program can result in some fuel assemblies being exposed to a maximum linear heat generation rate (LHGR) that exceeds 6.3 kw/ft at fuel burn-ups between 54 and 62 GWD/MTU. Thus, to account for the higher LHGR a site specific analysis was conducted (Reference 12). A computer code was developed, referred to as GAP (Reference 13), to perform the site specific gap fraction analysis.

The GAP code was developed and qualified as a safety related computer code. The GAP code implements the gap fractional release methodology presented in ANSI/ANS-5.4-1982, "American National Standard Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel" (Reference 17). The nodal input data to GAP for the temperature and specific power distribution as a function of burnup were developed with the Westinghouse PAD 4.0 code based on bounding power histories.

The GAP code is used to determine the gap release fractions for the short-lived and long-lived radionuclides. Both the ANS-5.4-1982 low-temperature and high-temperature release models are used. The bounding value as obtained for either release model is selected for the final result. These specific gap release fractions were then compared to the gap release fractions provided in Table 3 of RG 1.183. The bounding values between the regulatory value and the specific value are then used in the FHA dose analysis. The comparison of the results is shown in Table D.2-1. Due to the margin between the calculated gap fractions and those specified in Table 3 of RG 1.183, along with the minimal differences between the fuel designs (i.e., pin diameter and mass of fuel), it is assumed that the RG 1.183 gap fractions would also be bounding for the Westinghouse 422V+ fuel.

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D.3 FUEL HANDLING SOURCES

Core source terms for both a Westinghouse OFA and 422V+ were calculated for use in the fuel handling accident analysis (References 15). The inventory of the fission products in the reactor core is based on maximum full-power operation of the core at a power level equal to 1683 MWt and current licensed values of fuel enrichment and burnup. 1683 MWt includes power measurement uncertainty above the current licensed core power level.

The ORIGEN2 computer code (Reference 14) was used to determine the equilibrium core inventory. ORIGEN2 is a versatile point depletion and radioactive decay computer code for use in simulating nuclear fuel cycles and calculating the nuclide compositions and characteristics of materials contained therein.

The parameters used for the calculations are summarized in Table D.3-1. The undecayed activity in one fuel assembly is given in Table D.3-2.

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D.4 REACTOR COOLANT FISSION PRODUCT ACTIVITIES

The calculated reactor coolant specific activities for a coolant temperature of 578°F is included in Table D.4-1. The listed activities are maximum calculated values and were determined in Reference 8.

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D.5 REACTOR COOLANT TRITIUM ACTIVITIES

During the fissioning of uranium, tritium atoms are generated in the fuel at a rate of approximately 8×10^{-5} atoms per fission (1.05×10^{-2} curies/MWt-day). Other sources of tritium include neutron reactions with boron (in the coolant for shim control), neutron reactions with lithium (utilized in the coolant for pH control, and produced in the coolant neutron reactions with boron), and by neutron reactions with naturally-occurring deuterium in light water.

D.5.1 Release of Ternary-Produced Tritium

The tritium formed by ternary fission in uranium-fueled reactors, can be retained in the fuel, accumulate in the void between the fuel and cladding, react with cladding material (zirconium tritide), or diffuse through the cladding into the coolant. Operating experience at the Shippingport reactor (zirconium clad) indicated that less than 1% of the ternary-produced tritium is released to the reactor coolant. In order to insure adequate sizing of liquid waste treatment facilities, Westinghouse conservatively assumes that 30% of the ternary-produced tritium is released to coolant. This assumption then requires that the waste treatment system be sized to process approximately 4 reactor coolant-system volumes in addition to normal reactor plant liquid wastes. Anticipated ternary tritium loss to the reactor coolant is 1%.

D.5.2 Tritium Produced from Boron Reactions

The neutron reactions with boron resulting in the production of tritium are:

B^{10} (n, 2α) T

B^{10} (n, α), Li^{7} (n, nα) T

B^{11} (n, T) Be^{9}

B^{10} (n, d) Be^{9} (n, 2α) T

Of the above reactions, only the first two contribute significantly to the tritium production. the B^{11} (n, T) Be^{9} reaction has a threshold of 14 Mev and a cross section of ~5 mb. Since the number of neutrons produced at this energy are less than 10^{9} n/cm^{2}-sec the tritium produced from this reaction is negligible. The B^{10} reaction may be neglected, since Be^{9} has been found to be unstable.

D.5.3 Tritium Produced from Lithium Reactions

The neutron reactions with lithium resulting in the production of tritium are:

Li^7 (n, nα) T

Li^6 (n, α) T

In the Westinghouse-designed reactors, lithium is used to maintain the reactor coolant pH at 6.9-7.4. The reactor coolant is maintained at a target lithium nominal concentration of 3.5 ppm, until a pH of 7.3. Lithium concentration is then decreased as boron decreases, to maintain a pH of 7.3 to EOL (Reference 28).

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Short term, high startup boron levels may preclude achieving the required pH at 3.5 ppm lithium. EPRI Guidance allows for short term deviation from prescribed lithium limits, with fuel vendor concurrence (Reference 28).

Exceptions to operational prescribed lithium control:

1. A maximum lithium concentration exemption of 6.0 ppm for 24 hours during initial cycle startup. (Reference 25)
2. A one time use, maximum lithium concentration exemption of 6.0 ppm for 24 hours during a mid-cycle startup, within the first 4,000 MWD/MTU. (Reference 25)
3. A maximum lithium concentration variance of 5.0 ppm for the first 150 MWD/MTU cycle burnup. (Reference 27).

The current lithium regime is per 50.59 Evaluation 1091. (Reference 26)

A cation demineralizer is included in the Chemical and Volume Control System to remove the excess lithium produced in the B^{10} (n , α) Li^7 reactions.

The Li^6 (n , α) T reaction is controlled by limiting Li^6 inpurity in the lithium used in the reactor coolant and by lithiating the demineralizers to less than 0.001 parts of Li^6. This limitation has been in effect in Westinghouse-designed reactors since 1962.

D.5.4 Tritium Production from Deuterium Reactions

Since the amount of naturally occurring deuterium is less than 0.00015, the tritium produced from this reaction is less than 1 curie per year, and is therefore neglected.

D.5.5 Tritium Sources from the Reactor

Reactor tritium sources are given in Table D.5-1. The basic assumptions and plant parameters used in calculating these sources are:

1.	Core thermal power		1721.4 MWt
2.	Plant load factor		0.8
3.	Core volume		613.8 ft^3
4.	Core volume fractions		
	a. UO_2		0.3118
	b. Zr + SS		0.1065
	c. H_2O		.5817
5.	Initial reactor coolant boron level		
	a. Initial cycle		890 ppm
	b. Equilibrium cycle		825 ppm
6.	Reactor coolant volume		6236 ft^3
7.	Reactor coolant peak lithium level (99% pure Li^7)		6.0 ppm
8.	Core averaged neutron fluxes (n/cm^2-sec)		
	a. E > 6 Mev		1.48×10^{12}
	b. E > 5 Mev		4.01×10^{12}
	c. 3 Mev $\leq$ E $\leq$ 6 Mev		1.15×10^{13}
	d. 1 Mev $\leq$ E $\leq$ 5 Mev		2.70×10^{13}
	e. E < 0.625 ev		1.18×10^{13}
9.	Neutron reaction cross-sections		
	a. B^{10} (n, 2α) T :	δ (1 Mev $\leq$ E $\leq$ 5 Mev) =	31.6 mb (spectrum weighted)
		δ (E > 5 Mev) =	75 mb
	b. Li^7 (n, nα) T:	δ (3 Mev $\leq$ E $\leq$ 6 Mev) =	39.1 mb (spectrum weighted)
		δ (E > 6 Mev) =	400 mb
10.	Fraction of ternary tritium diffusing through zirconium cladding		
	a. Design value		0.30
	b. Expected value		0.01

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D.5.6 Tritium Source Uncertainties

Uncertainties in the estimates of tritium generation can be attributed to uncertainties in cross-sectional data, fission yields, and neutron fluxes. A program has been implemented to follow the buildup of tritium at the R. E. Ginna Station of the Rochester Gas and Electric Company. It was the first large Westinghouse-designed PWR with zircaloy clad fuel to be placed into operation in the United States. Values for those parameters specifically requested are included in the following table. Particulars of the program are in the published literature.[9] During this program, measured values were compared to those estimated and the measured values were in agreement with calculated values to within 10%. Plant parameters for R. E. Ginna are given in Table D.5-2.

The tritium sources for some earlier plants were conservatively calculated on the following bases:

1. To establish an upper limit for tritium produced in burnable poisons, and the burnup of B^{10} by the B^{10} (n, α) Li^{7} reaction was neglected.
2. To establish an upper limit for tritium produced from the Li7 (n, nα) T reaction, it was assumed that the maximum amount of Li7 generated was present in the rods for the entire exposure period.

For the Prairie Island plant, these conservatisms were not applied. As a result, the Prairie Island numbers are a more realistic estimate of the tritium produced in the burnable poison rods used only during the initial cycle.

D.6 VOLUME CONTROL TANK ACTIVITY

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The 220-ft^3 volume control tank is assumed to contain 110 ft^3 of liquid and 110 ft^3 of vapor. Table D.6-1 lists the specific activities in the volume control tank with clad defects in 1% of the fuel rods. The assumptions used in calculating these activities are the same as those given in D.4.

Holdup Tanks

Reactor Coolant is intermittently directed to and processed from the holdup tanks. Consequently, the amount of radioactivity in the tanks is constantly changing. The activity concentration in the reactor coolant is listed, by isotope, in Table D.4-1; these values represent the maximum activity concentration in the water directed to the holdup tanks.

Reactor Makeup Water Storage Tank

Reactor makeup water is supplied as demineralized water (non-radioactive) from the secondary plant demineralizers, or as distillate from the Boric Acid Evaporator which processes reactor coolant. The maximum activity concentrations in the storage tank are tabulated below:

REACTOR MAKEUP WATER

	Source of Activity	Dominant Isotope	Maximum Concentration µCi/cc
A.	Fission Product Gases	Xe-133	0.75
B.	Fission Product Particulates	Cs-137	.005
C.	Corrosion Products	Co-58	.005

Refueling Water Storage Tank

The maximum activity concentrations in the Refueling Water Storage Tank are the same as those listed above for the Reactor Makeup Water Storage Tank.

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D.7 GAS DECAY TANK ACTIVITY

The activity in one gas decay tank was calculated assuming that one tank, initially void of activity, is filled with all the gaseous activity that could be stripped off from the entire Reactor Coolant System operating with 1% fuel defects. The specific activities presented in Table D.4-1 were multiplied by the Reactor Coolant System volume of 6100 ft^3. Table D.7-1 lists the total activity in one gas decay tank, which is also the total gaseous activity in the reactor system with the exception of KR-85. The KR-85 inventory in Table D.7-1 represents the activity at the end of the 60-year plant life.

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D.8 DOSE CALCULATION MODELS FOR SAFEGUARDS ANALYSIS

D.8.1 Introduction

This section identifies the models used to calculate the offsite radiation dose that would result from radioactivity release due to various postulated accidents.

Sections D.8.2 – D.8.4 apply to:

1. Volume control tank rupture
2. Gas decay tank rupture

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Section D.8.5 applies to:

1. Rupture of a main steam line
2. Loss of coolant accident
3. Steam Generator Tube Rupture Accident
4. Locked Pump Rotor Accident
5. Control Rod Ejection Accident

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D.8.2 Assumptions

The following assumptions are basic to the models used:

1. Direct radiation from the source point is negligible compared to whole body radiation due to submersion in the radioactive leakage cloud.
2. All radioactive material releases are treated as ground sources regardless of the point of discharge. The atmospheric dispersion factors discussed in Appendix H apply to all radioactive releases.
3. The dose receptor is a standard man as defined by the International Commission on Radiological Protection (ICRP). [5]
4. Radioactive decay from the point of release to the dose receptor is neglected.

D.8.3 Whole Body Dose (Gamma and Beta)

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

The general equation for calculating whole body dose due to gamma radiation and whole body skin dose is:

$$D_{wb} = 0.246\ (X/Q)\ \Sigma_i Q_i E_i^* \qquad (1)$$

where:

D_{wb} = whole body dose due to gamma radiation, rem

X/Q = site dispersion factor, sec/m^3

Q_i = total activity of isotope i released, curies

E_i = effective decay energy from isotope i, Mev/disintegration

*Unit Conversion factor of 0.25 versus 0.246 is used for MSLB and LOCA.

D.8.4 Thyroid Inhalation Dose

The thyroid dose is obtained from the following expression:

$$D_T = \Sigma_t B_t \, (X/Q)_t \, \Sigma_i A_{i,t} \, (DCF)_i$$

where:

D_T	=	thyroid inhalation dose, rem
$(X/Q)_t$	=	site dispersion factor for time interval t, sec/m^3
B_t	=	breathing rate for time interval t, m^3/sec
$A_{i,t}$	=	total activity of iodine isotope i released in time period i, curies
$(DCF)_i$	=	dose conversion factor for iodine isotope i, rem/curie inhaled

For accident analyses over various time intervals t, site atmospheric dispersion factors are presented in Appendix H.

The dose conversion factors are given in Table D.8-2 for each significant iodine isotope. Standard man breathing rates are given in Table D.8-3.

D.8.5 Dose Models used for Main Steam Line Break, Loss of Coolant Accident, Steam Generator Tube Rupture Accident, Locked Pump Rotor Accident, and Control Rod Ejection Accident

The RADTRAD computer code is used to calculate radiological dose consequences at the EAB, the LPZ, and the Control Room for the Main Steam Line Break, Loss of Coolant Accident, Steam Generator Tube Rupture Accident, Locked Pump Rotor Accident, and Control Rod Ejection Accident.

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D.9 REFERENCES

1. J. J. DiNunno, et al, "Calculation of Distance Factors for Power and Test Reactor Sites", TID 14844, March 1962.
2. J. O. Blomeke and Mary F. Todd, "Uranium-235 Fission Product Production as a Function of Thermal Neutron Flux, Irradiation Time, and Decay Time", ORNL 2127 (4 Volumes), August and December 1957.
3. D. F. Toner and J. L. Scott, "Fission Product Release from UO_2," Nuclear Safety Volume III No. 2, December 1961.
4. J. Belle, Uranium Dioxide: Properties and Nuclear Applications, Naval Reactors Division of Reactor Development, USAEC, 1961.
5. "Report of ICRP Committee II on Permissible Dose for Internal Radiation (1959)," Pergamon Press, 1963.
6. Merril Eisenbud, Environmental Radioactivity, McGraw Hill Book Company, p. 93, 1963.
7. Reference (5), pp. 27-28.
8. Westinghouse Calculation Note CN-REA-07-36, "Prairie Island Core Activity Inventory and Reactor Coolant Activity", Revision 0.
9. J. Locante, D. D. Malinowski, "Tritium in Pressurized Water Reactors" Transactions of the Annual Meeting of the ANS, 1971, Volume 14 No. 1.
10. Deleted
11. Deleted
12. Calculation GEN-PI-048, Non-LOCA Gap Release Factors for Use in Prairie Island Alternate Source Term Assessments.
13. Fauske and Associate Inc. (FAI), GAP Computer Code.
14. ORNL/TM-11018, Standard - and Extended - Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code, ORNL Chemical Technology Division, December.
15. Calculation GEN-PI-077, "Fuel Handling Accident Dose Analysis - AST"
16. Deleted
17. ANSI/ANS-5.4-1982, "American National Standard Method for Calculating the Fractional Release of Volatile Fission Products from oxide Fuel."
18. Calculation 12400604-UR(B)-001, Waste Gas Tank Rupture Dose Analysis, Revision 0A.
19. Calculation GEN-PI-078, "Main Steam Line Break (MSLB) Accident Analysis Using AST"
20. Calculation GEN-PI-079, "Post-LOCA, EAB, LPZ, and CR Doses - AST"

21. ANSI/ANS 6.1.1-1977, “Neutron and Gamma Ray Fluence-to-Dose Factors”.

22. USNRC Regulatory Guide 1.4, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors”, Revision 2.

23. Westinghouse Calculation CN-REA-07-36, “Prairie Island Core Activity Inventory and Reactor Coolant Activity”.

24. EPA Federal Guidance Report 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion”, 1988.

25. Letter, J.T. Fuller (Westinghouse) to D. Mienke (Xcel Energy), “Exemption from Westinghouse Supplementary Chemistry Guidelines for Startup Lithium Limits at Prairie Island Units 1 and 2”, Letter No. NF-XCEL-11-135, dated November 7, 2011

26. 50.59 Evaluation 1091, “Changes to Lithium Control Strategy in RPIP-3006, PRIMARY CHEMISTRY GUIDELINES”, January 19, 2012

27. Letter J.T. Fuller (Westinghouse) to D. Mienke (Xcel Energy), “Updated Westinghouse Supplement (Revision 4) to EPRI PWR Primary Water Chemistry Guidelines, Rev. 6 (09-IC-21)”, Letter No. NF-XCEL-09-153, dated October 21, 2009

28. Pressurized Water Reactor Primary Water Chemistry Guidelines, Volume 1, Rev. 6, final report date December 2007

29. Letter, T.J. Wengert (NRC) to M.A. Schimmel (NSPM), “Prairie Island Nuclear Generating Plant, Units 1 and 2 - Amendment RE: Measurement Uncertainty Recapture Power Uprate (TAC NOS. ME3015 and ME3016)”, dated August 18, 2010.

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TABLE D.1-1 DELETED

TABLE D.1-2 EQUILIBRIUM CORE INVENTORY (POWER LEVEL: 1683 MWt)

Nuclide	Activity (Curie)	Nuclide	Activity (Curie)	Nuclide	Activity (Curie)
AM-241	6.54E+03	NB-95	7.64E+07	XE-133	9.15E+07
BA-139	8.06E+07	ND-147	2.89E+07	XE-133M	2.86E+06
BA-140	7.73E+07	NP-239	9.08E+08	XE-134M	1.39E+06
BR-83	5.09E+06	PR-143	6.71E+07	XE-135	2.07E+07
BR-84	9.08E+06	PR-144	6.29E+07	XE-135M	1.95E+07
BR-85	1.08E+07	PU-238	2.48E+05	XE-137	8.23E+07
CE-141	7.39E+07	PU-239	1.45E+04	XE-138	7.72E+07
CE-143	6.76E+07	PU-240	2.33E+04	Y-90	5.99E+06
CE-144	6.25E+07	PU-241	6.32E+06	Y-91	5.34E+07
CM-242	2.79E+06	RB-86	1.32E+05	Y-92	5.49E+07
CM-244	6.13E+05	RB-88	2.88E+07	Y-93	6.28E+07
CS-134	1.49E+07	RH-105	4.94E+07	ZR-95	7.50E+07
CS-136	3.12E+06	RU-103	7.40E+07	ZR-97	7.49E+07
CS-137	7.88E+06	RU-105	5.26E+07		
CS-138	8.46E+07	RU-106	3.08E+07		
I-129	1.90E+00	SB-127	4.26E+06		
I-130	1.45E+06	SB-129	1.30E+07		
I-131	4.50E+07	SE-79	5.40E+00		
I-132	6.51E+07	SR-89	3.98E+07		
I-133	9.13E+07	SR-90	5.71E+06		
I-134	1.02E+08	SR-91	5.02E+07		
I-135	8.72E+07	SR-92	5.43E+07		
KR-85	7.15E+05	TC-99M	7.31E+07		
KR-85M	1.08E+07	TE-127	4.21E+06		
KR-87	2.12E+07	TE-127M	7.36E+05		
KR-88	2.82E+07	TE-129	1.22E+07		
LA-140	8.13E+07	TE-129M	2.35E+06		
LA-141	7.26E+07	TE-131M	8.90E+06		
LA-142	6.99E+07	TE-132	6.35E+07		
MO-99	8.31E+07	XE-131M	5.06E+05		

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TABLE D.2-1 COMPARISON OF GAP FRACTIONAL RELEASES FOR PRAIRIE ISLAND FUEL RODS TO RG 1.183, TABLE 3

Radionuclide or Radionuclide Group	Fractional Release	
	Bounding Prairie Island Result	**RG 1.183, Table 3**
I-131	0.036	0.08
Kr-85	0.056	0.10
Other Noble Gases	0.024	0.05
Other Halogens	0.013	0.05
Alkali Metals	0.072	0.12

The GAP code predicts that in all cases, the fractions in Table 3 of RG 1.183 are bounding. Note: GAP results are currently not available for the Westinghouse 422V+ Fuel. However, as stated in Section D.2, the RG 1.183 values are assumed to bound this fuel type as well due to small differences in fuel type. Therefore, the fractions in Table 3 of RG 1.183 are used in the FHA dose analysis.

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TABLE D.3-1 NUCLEAR CHARACTERISTICS OF HIGHEST POWER DISCHARGE ASSEMBLY

Parameter	**Assumed Value**
Core Power	1683 MWt (including power measurement uncertainty)
Uranium Mass	43.202 MTU (Westinghouse OFA) 47.533 MTU (Westinghouse 422V+)
U-235 Enrichment	4.92 w/o (Westinghouse OFA) 4.95 w/o (Westinghouse 422V+)
Burn- Up	65,000 MWD/MTU
Max Radial Peaking Factor	1.77 (1.90 is used in the FHA dose consequence analysis)
Post-Shutdown Decay Time	50 hours

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TABLE D.3-2 ACTIVITY IN ONE FUEL ASSEMBLY UNDECAYED

Isotope	Activity in One Fuel Assembly at Shutdown (Ci)
Kr-85**	2.245E+04
Kr-85M	4.696E+05
Kr-87	3.329E+05
Kr-88	4.428E+05
I-131***	1.131E+06
I-132	1.022E+06
I-133	1.434E+06
I-134	1.602E+06
I-135	1.369E+06
Xe-131M	7.945E+03
Xe-133	1.437E+06
Xe-133M	4.491E+04
Xe-135	3.250E+05
Xe-135M	3.062E+05
Xe-138	1.212E+05

* The activity values have been adjusted by the radial peaking factor.

** Kr-85 activity has been multiplied by a factor of 2 to account for additional fractional release relative to other noble gases.

*** I-131 activity has been multiplied by a factor of 1.6 to account for additional fractional release relative to other iodine isotopes.

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TABLE D.4-1 REACTOR COOLANT SPECIFIC FISSION PRODUCT ACTIVITIES (1683 MWT AT 578°F)* from Reference 8

Isotope	µCi/g	Isotope	µCi/g
Noble Gases		**Non-volatile Fission Products**	
Kr-83	0.392		
Kr-85	10.9	Br-84	4.22E-02
Kr-85m	1.59	Rb-88	4.09
Kr-87	1.03	Rb-89	1.64E-01
Kr-88	2.89	Sr-89	3.46E-3
Kr-89	0.082		
Xe-131m	2.05		
Xe-133	260	Sr-90	2.51E-4
Xe-133m	3.65	Y-90	6.88E-05
Xe-135	8.24	Sr-91	4.76E-03
Xe-135m	0.502	Y-91	4.74E-04
Xe-137	0.172		
Xe-138	0.609	Sr-92	1.14E-03
		Y-92	9.68E-04
		Zr-95	5.69E-04
		Nb-95	5.73E-04
		Mo-99	6.90E-01
		I-129	5.51E-08
		I-130	0.0254
		I-131	2.55
		Te-132	2.64E-01
		I-132	2.8
		I-133	3.88
		Te-134	2.85E-02
		I-134	0.582
		Cs-134	2.96
		I-135	2.23
		Cs-136	3.21
		Cs-137	2.16
		Cs-138	9.35E-01
		CE/Pr-144	4.42E-04

*Specific activities based on heavy bundle fuel with fuel management multiplier or OFA fuel without fuel management multiplier. The higher of the two is listed.

TABLE D.5-1 REACTOR COOLANT TRITIUM ACTIVITIES (curies per year)

Tritium Source	Total Produced	Released to the Coolant: Design Value	Released to the Coolant: Expected Value
Ternary Fissions	6614	1982	66
Burnable Poison Rods (Initial Cycle)	390	120	3.9
Soluble Poison Boron (Initial Cycle)	323	323	323
(Equilibrium Cycle)	323	323	323
Li-7 Reaction (Initial Cycle)	7	7	7
(Equilibrium Cycle)	11	11	11
Li-6 Reaction (Initial Cycle)	3.5	3.5	3.5
(Equilibrium Cycle)	5.5	5.5	5.5
Deuterium Reaction	.6	.6	.6
Total-Initial Cycle	7338	2436	404
Total-Equilibrium Cycle	6954	2322	406

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TABLE D.5-2 PLANT PARAMETERS FOR R.E. GINNA

1.	Core thermal power		1300 MWt
2.	Plant load factor		0.8
3.	Core volume		613.8 ft^3
4.	Core volume fractions		
	a. UO_2		0.3118
	b. Zr + SS		0.1065
	c. H_2O		.5817
5.	Initial reactor coolant boron level		
	Initial cycle		1300 ppm
6.	Reactor coolant volume		6000 ft^3
7.	Reactor coolant peak lithium level		
	(99.9% Li^7)		2.2 ppm (max.)
8.	Core averaged neutron fluxes (n/cm^2-sec)		
	a. E > 6 Mev		2.59×10^{12}
	b. E > 5 Mev		5.17×10^{12}
	c. 3 Mev $\leq$ E $\leq$ 6 Mev		1.62×10^{13}
	d. 1 Mev $\leq$ E $\leq$ 5 Mev		4.82×10^{13}
	e. E < 0.625 ev		2.30×10^{13}
9.	Neutron reaction cross-sections		
	a. B^{10} (n, 2α) T:	δ (1 Mev $\leq$ E $\leq$ 5 Mev) =	31.6 mb
		δ (E > 5 Mev) =	75 mb
	b. Li^7 (n, nα) T:	δ (3 Mev $\leq$ E $\leq$ 6 Mev) =	85.6 mb
		δ (E > 6 Mev) =	400 mb
10.	Expected Fraction of ternary tritium diffusing through zirconium cladding		0.01

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TABLE D.6-1 VOLUME CONTROL TANK SPECIFIC ACTIVITIES (from Reference 8)

Isotope	µCi/cc
Vapor	
Kr-85	2.29
Kr-85m	6.37
Kr-87	2.73
Kr-88	11.69
Xe-133	1770
Xe-133m	25.08
Xe-135	45.31
Xe-135m	0.41
Xe-138	0.58
Liquid	
I-131	0.27
I-132	0.29
I-133	0.42
I-134	0.062
I-135	0.24

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TABLE D.7-1 GAS DECAY TANK ACTIVITY (Reference 8, 18, 29)

Assumptions: Volume of the tank immaterial to this calculation

Operation at 1683 MWt (includes 0.36% measurement uncertainty)

Clad defects in 1% of fuel rods.

Isotope	Total Activity Curies
Kr-85*	8.15×10^4
Kr-85m	1.98×10^2
Kr-87	1.28×10^2
Kr-88	3.60×10^2
Xe-133m	4.54×10^2
Xe-133	3.21×10^4
Xe-135	1.02×10^3
Xe-135m	5.89×10^1
Xe-138	7.60×10^1

* Kr-85 inventory represents the activity at the end of the 60-year plant life.

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TABLE D.8-1 EFFECTIVE DECAY ENERGIES FOR NOBLE GASES (MEV)

Isotope	Gamma	Beta
Kr-83m	0.0016	0.0
Kr-85	0.0022	0.25
Kr-85m	0.158	0.229
Kr-87	0.614	1.24
Kr-88	1.95	0.36
Kr-89	1.83	0.61
Xe-131m	0.02	0.0
Xe-133	0.045	0.1
Xe-133m	0.041	0.0
Xe-135	0.248	0.303
Xe-135m	0.431	0.0
Xe-137	0.188	1.77
Xe-138	1.13	0.615
Br-82	2.24	0.439
Br-83	0.0074	0.911
Br-84	0.636	2.93

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TABLE D.8-2 THYROID DOSE CONVERSION FACTORS FOR IODINE INHALATION

(Reference 24)
[Used for post MSLB and LOCA dose analyses]

Isotope	DCF, rem/curies inhaled
I-131	1.08×10^{6}
I-132	6.44×10^{3}
I-133	1.80×10^{5}
I-134	1.07×10^{3}
I-135	3.13×10^{4}

(Reference 11)
[Used for post WG dose analysis]

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Isotope	DCF, rem/curies inhaled
I-129	5.50×10^{6}
I-130	2.18×10^{6}
I-131	1.48×10^{6}
I-132	5.34×10^{4}
I-133	4.00×10^{5}
I-134	2.5×10^{4}
I-135	1.25×10^{5}

Different reference sources may publish slightly different values for the dose conversion factors (DCF). In practice these difference do not affect the analysis results.

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TABLE D.8-3 STANDARD MAN BREATHING RATES (OFFSITE ONLY)

Time Period	Breathing Rate, m^3/sec
0 - 8 hours	3.47×10^{-4}
8 - 24 hours	1.75×10^{-4}
> 24 hours	2.32×10^{-4}

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