Form 01077 (R7-89)

FORM 101.1	REVISION 13

CERTIFICATION OF ENGINEERING CALCULATION

STATION AND UNIT NUMBER Catawba Units 1 & 2

TITLE OF CALCULATION Offsite Dose from a Postulated Main Steam Line Break

CALCULATION NUMBER CNC-1227.00-00-0051

ORIGINALLY CONSISTING OF:

PAGES 1 THROUGH 29

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THESE ENGINEERING CALCULATIONS COVER QA CONDITION 1 ITEMS. IN ACCORDANCE WITH ESTABLISHED PROCEDURES, THE QUALITY HAS BEEN ASSURED AND I CERTIFY THAT THE ABOVE CALCULATION HAS BEEN ORIGINATED, CHECKED OR APPROVED AS NOTED BELOW:

ORIGINATED BY Harry E Vanpelt DATE 8/22/92

CHECKED BY James J Glenn DATE 8/22/92

APPROVED BY C J Hunzell DATE 8/22/92

ISSUED TO TECHNICAL SERVICES DIVISION ________ DATE ________

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Offsite Dose From a Postulated Main Steam Line Break

1.0 Problem

The purpose of this calculation is to determine offsite dose analysis results due to a postulated main steam line break at Catawba Nuclear Station.

2.0 QA Condition

This calculation ensures that offsite doses are within acceptance limits for the assumed primary to secondary leak rate conditions and Reactor Coolant System activity. Therefore, this is a QA1 calculation.

3.0 Method

The equations used to model iodine transport and decay, iodine and noble gas releases, and thyroid and whole body doses are developed in section 9 of this calculation. These equations are programmed with an "Excel" spreadsheet. The equations are solved in closed analytical form over numerous time steps such that if any of the input parameters change during the time frame of interest in the calculation, the values are extracted from a table of input values and used for the given mathematical function over the applicable time step. For example, Reactor Coolant System iodine concentration is solved over iterative time steps and a table of data is created to use as source term for the S/G iodine concentration and iodine release calculation. For S/G iodine concentration, we are solving for SGI, where SGI = f (SR, PF, SGM, PB, FF, SF) (these terms are defined in Section 9 of this calculation). These parameters are held constant over short time steps ($\Delta t = t_{n+1} - t_n, \ldots$), and the equation is solved over this time step. SGI solved for the time step as described now becomes the initial SGI_0 for the next time step. For this next time step, new values of the input parameters (of which SGI is a function) are extracted from tables in the "Excel" spreadsheet, and the process is repeated for the next time step. This process is repeated until we have solved for SGI over the entire time frame of interest (i.e., accident duration). Iodine release is calculated over equivalent time intervals, as is thyroid dose. The result for each interval is then summed for a total result. For the noble gases, no holdup in the S/G occurs, and hence only nuclide decay is calculated over the time steps. Based on the primary to secondary leak rate, a total whole body dose is calculated. Cases will be evaluated for pre-existent iodine spike and coincident iodine spike, for both the EAB (Exclusion Area Boundary) and the LPZ (Low Population Zone).

4.0 Codes and Standards

4.1 10 CFR Part 100

4.2 Standard Review Plan, Section 15.1.5, Appendix A

5.0 Design Criteria

The limits as specified in Codes and Standards 4.1 and 4.2 will not be exceeded.

6.0 FSAR Reference

Catawba Final Safety Analysis Report, Section 10.3 and 15.1.5 (1992 Update).

7.0 Assumptions

7.1 In the calculation of nuclide production (or appearance) rate, it is assumed that letdown demineralizer removal efficiency of iodine = 1.0. This assumption conservatively maximizes the nuclide production rate and hence yields greater Reactor Coolant System concentration versus time for the coincident (accident initiated) iodine spike case.

7.2 In the calculation of whole body doses, beta contribution to whole body dose in insignificant compared to gamma dose from the noble gases, and will be ignored.

7.3 A loss of offsite power has occurred, and all steam releases are through the S/G PORVs, rather than through the condenser (which would provide additional holdup of iodine).

7.4 All noble gases are released with a flashing fraction of 1.0. Also, all iodines are assumed to be released from the faulted S/G with a partition fraction of 1.0.

7.5 Iodine is released through the faulted S/G (i.e., the S/G affected by the main steam line break) with a flashing fraction of 1.0 (i.e., all of the iodine escapes). Iodine is released from the S/G water inventory in the intact S/Gs with a partition fraction of 0.01 (partition coefficient of 100).

7.6 In accordance with Regulatory Guide 1.4, depletion of the effluent plume due to deposition of radioiodine on the ground or radioactive decay is not

credited. Radioactive iodine in transit is conservatively assumed to be at the same source term strength as when it is released from the S/G PORVs or code safety valves.

7.7 For the intact S/Gs, the flashing fraction is assumed to be 0.15.

7.8 For the intact S/Gs, the primary bypass fraction is 0.05.

7.9 For the intact S/Gs, the scrubbing fraction (for flashed flow) is conservatively assumed to be 0. This will maximize the releases of radioactive iodine.

7.10 For the intact S/Gs, tube bundle uncovery lasts for 30 minutes. When the tube bundle is covered, it may be assumed that the primary leakage mixes homogeneously with the S/G water inventory. For steam generator tube rupture accidents, this is not expected to occur. However, for small leaks where the primary water is cooled in the leak path, it is not necessary to consider flashing. As stated in reference 8.3, "in the following analysis it has been assumed that leak geometry allows jet-like discharge of primary water into a shell side containing saturated water at low enthalpy. If flashing did not occur, then neither volatilization nor atomization of leaked water would occur and the leaked water would merely mix with the boiler water. While flashing is thought to be a realistic expectation for the tube rupture accident, flashing of coolant would not occur for very small leaks where the water is cooled in the leak path. Also, if subcooled water were present in the generator, flashed water would be condensed immediately and iodine would remain in water solution." However, it is conservatively assumed that the flashing fraction and primary bypass exist for the duration of the accident (8 hour LPZ dose calculation) for the intact S/G. This is to account for any S/G tube bundle uncovery that may occur as the operator steams the intact S/G to remove core decay heat.

7.11 Within 8 hours, the primary system has been cooled and depressurized such that primary to secondary leakage will cease.

7.12 No decrease in primary to secondary leak rate occurs over time. This is a very conservative assumption because the decreasing primary side temperature and pressure can be numerically credited by decreasing the leakage by the following ratio for a time step: $(\Delta P_{ending} / \Delta P_{beginning})^{0.5}$, where the pressure is assumed to be that value associated with 50 °F subcooling in the Reactor Coolant System, and where a cooldown rate is assumed.

7.13 For conversion from volume to mass in the calculation of primary to secondary leakage used as input (see section 10 of this calculation), it is assumed that the Reactor Coolant System is at 400 °F. The density

conversion for this temperature is ~ 0.14 gal/lb (see reference 8.11, page 4-5). This assumption conservatively maximizes the calculated primary to secondary leak rate as opposed to, for example, an RCS temperature of 550 °F (0.1628 gal/lbm).

7.14 The primary to secondary leakage in the intact S/G is the maximum amount allowed per S/G by the Catawba Technical Specifications.

7.15 In the calculation of core decay heat (which is used as an input to calculate steam mass released versus time for the intact S/Gs), it is assumed that the core has been irradiated to 30,000 MWD/MTU, that the initial core loading of heavy metal includes 88 MTU, that it has been enriched to 3.6% ^{235}U, and that it is burned at 1.02 times the RTP of 3411 MWt.

7.16 Iodine decay to xenon is an insignificant contribution to whole body doses and is ignored.

7.17 In the calculation of steam released from the intact S/Gs, it is assumed that the Reactor Coolant System sustains a cooldown rate of 60 °F/hour (8 hrs * 60 °F/hour = 480 °F total cooldown in 8 hours). The Catawba Technical Specifications (reference 8.12) limit the allowable cooldown rate to 100 °F/hour (3/4.9.1). However, it is implausible to obtain this cooldown rate (i.e., 800 °F in 8 hours with a beginning Reactor Coolant System temperature of 600 °F) with the accident duration lasting the assumed 8 hour period. Hence, this assumption is a corollary to assumption 7.11.

7.18 In the calculation of steam released from the intact S/Gs, it is assumed that the Reactor Coolant System temperature is equal to 550 °F. Per reference 8.11, this yields a density conversion of 0.1628 gal/lbm. The volume of the Reactor Coolant System is 73,300 gallons (see reference 8.13). This density conversion conservatively ignores the Reactor Coolant System cooldown which will occur over the duration of the accident, where the intact S/Gs are being fed and steamed to remove core decay heat, and cooldown and depressurize to terminate the primary to secondary leakage. An assumption of a lower Reactor Coolant System temperature would yield a lower density conversion factor, and hence yield a higher value for total calculated Reactor Coolant System mass, and a higher calculated value for steam mass released. For example, the conversion for 556 °F RCS temperature is ~ 0.1628 gal/lbm, and the conversion for 400 °F RCS temperature is ~ 0.14 gal/lbm. Accounting for RCS cooldown in the calculation of steam releases would increase the steam released by (1-(0.1628/0.14)) = 16.3%. However, a reduced Reactor Coolant System temperature would also yield a reduced S/G temperature and pressure, which would correspond to a higher value for h_{fg}. For example, h_{fg} for 556 °F RCS temperature is 632 BTU/lbm, while h_{fg} for 400 °F RCS

temperature is 825.9 BTU/lbm. Assuming that the S/G $P_{saturation}$ will closely track with RCS temperature, this would decrease the steam mass released by (1-(825.9/632)) = 30.7%. Therefore, these considerations cancel out, and in fact, not accounting for RCS cooldown in the calculation of steam releases is slightly conservative.

7.19 In the calculation of steam released from the intact S/Gs, a value of 632 BTU/lbm is assumed for the S/G h_{fg}. This value corresponds to a steam pressure of ~ 1100 psia. This is conservative, since a lower value for steam pressure would yield a higher value for h_{fg} and hence lower calculated steam releases from the intact S/Gs.

7.20 The calculation of steam released from the intact S/Gs, core decay heat and a Reactor Coolant System cooldown rate is assumed as heat load. Stored energy in the primary system components is ignored.

7.21 In the calculation of S/G iodine concentration, it is assumed that the intact S/G water inventory is equal to 60,000 lbm (see reference 8.19).

8.0 References

8.1 Standard Review Plan (NUREG-0800), Section 15.6.3, "Radiological Consequences of a Steam Generator Tube Failure," and Section 15.1.5, Appendix A, "Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR."

8.2 Regulatory Guide 1.4, Rev. 2, June 1974, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors."

8.3 NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident."

8.4 NUREG/CR-2683, "Iodine Behavior in Steam Generator Tube Rupture Accidents."

8.5 Westinghouse WCAP-11002, "Evaluation of Steam Generator Overfill Due to Steam Generator Tube Rupture Accident."

8.6 Westinghouse Supplement 1 to WCAP-10698, "Evaluation of Offsite Doses For a Steam Generator Tube Rupture Accident."

8.7 Westinghouse WCAP-13132, "The Effect of Steam Generator Tube Bundle Uncovery on Radioiodine Release."

8.8 Peter S. Tam (USNRC), "A Mathematical Model to Describe Decontamination Factors (DFs) in Steam Generators," Proceedings of the Topical Meeting on Thermal Reactor Safety (Americal Nuclear Society).

8.9 Annals of the ICRP, ICRP Publication 30, Supplement to Part 1, Vol. 3, 1979, "Limits for Intakes of Radionuclides by Workers."

8.10 Regulatory Guide 1.109, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the purpose of Evaluating Compliance With 10 CFR Part 50, Appendix I."

8.11 Cameron Hydraulic Data, Ingersoll-Rand, 16th Edition, 1979.

8.12 Catawba Technical Specifications, Section 3.4.6.2.

8.13 Catawba Drawing CN-1680-122, Sheet 3, Rev. 1.

8.14 Catawba Calculation CNC-1108.01-00-0001, Rev. 1, "Diffusion Calculation for Calculations for Catawba Nuclear Station."

8.15 Catawba Calculation CNC-1227.00-00-0052, "Reactor Coolant System Radiation Monitor Setpoint Calculation."

8.16 Catawba Calculation CNC-1201.30-00-0012, "Nuclear Fuel Decay Heat Methodology, Inputs and Results."

8.17 Catawba Final Safety Analysis Report, 1992 Update.

8.18 ORNL/NUREG/TM-102, "Nuclear Decay Data for Radionuclides Occurring in Routine Releases from Nuclear Fuel Cycle Facilities."

8.19 G. B. Swindlehurst letter (file: CN-1552.12), 1-11-89.

8.20 ASME Steam Tables

8.21 PIR 2-C92-0412

8.22 Catawba Calculation CNC-1223.43-01-0004, "SMFE5760, 5770, 5780 & 5790 Bore Verification."

8.23 John R. LaMarsh, "Introduction to Nuclear Engineering," 2nd Edition, 1983, Addison-Wesley Publishing Company.

8.24 Nuclides and Isotopes (Chart of the Nuclides), 14th Edition, G.E. Nuclear Energy.

8.25 Westinghouse WCAP-8637, "Iodine Behavior Under Transient Conditions in the Pressurized Water Reactor."

8.26 EPRI NP-4595, "Iodine Spiking," May, 1986.

8.27 W. F. Pasedag (USNRC), "Iodine Spiking in BWR and PWR Coolant Systems," Proceedings of the Topical Meeting on Thermal Reactor Safety, American Nuclear Society.

8.28 R. J. Lutz & W. Chubb, "Iodine Spiking - Cause and Effect," Transactions of the American Nuclear Society, 28, pg 649-650 (1978).

8.29 K. H. Neeb & E. Schuster, "Iodine Spiking in PWRs: Origin and General Behavior," Transactions of the American Nuclear Society, 28, pg 650-651 (1978).

8.30 John C. Ho, "Pressurized Water Reactor Iodine Spiking Behavior Under Power Transient Conditions," Transactions of the American Nuclear Society, pg 378-379.

8.31 EGG-NERD-8648, "Probability of the Iodine Spike Release Rate During an SGTR."

8.32 NUREG/CR-4817 (ORNL/TM-10330), "Iodine Partition Coefficient Measurements at Simulated PWR Steam Generator Conditions: Interim Data Report," May, 1987.

8.33 NUREG/CR-5365, "Iodine Speciation and Partitioning in PWR Steam Generator Accidents," October, 1989.

8.34 EP/2/A/5000/1D, "Steam Line Break Outside Containment."

8.35 EP/1/A/5000/1E3, "SGTR with Continuous NC System Leakage: Subcooled Recovery."

9.0 Calculations (Mathematical Modeling Methodology)

Reactor Coolant System Activity (Pre-existent Iodine Spike)

Pre-existent iodine activity in the Reactor Coolant System is depleted over time in this model based on radioactive decay and break flow, as per the following mathematical function:

$C_i = C_o\ [e^{-(\lambda+(BF/RCS\ mass))t}]$

Where:

C_i	= Concentration of nuclide at time t, Ci/lbm
C_o	= Initial concentration, Ci/lbm
λ	= Nuclide decay constant, min^{-1}
BF	= Break mass flow, lbm/min
RCS_{mass}	= Reactor Coolant System mass, lbm
t	= time, min

In the Excel spreadsheet, C_o is the concentration of nuclide *i* calculated for the previous time step, and is used as input to the subsequent time step as described above.

Reactor Coolant System Activity (Coincident Iodine Spike)

Reference 8.1 states the following: "The reactor trip and/or primary system depressurization associated with the MSLB creates an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model which assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value stated in the NSSS vendor standard technical specifications or from the plant specific technical specifications, as appropriate (i.e., concurrent iodine spike case)." The following mathematical model is derived for the coincident (or accident-initiated) iodine spike:

Calculation of nuclide production rate:

$$\frac{dN}{dt} = P_n - \lambda N - [f_L \eta/V]N$$

Where:

P_n	= Production rate for nuclide, atoms/min
λ	= Nuclide decay constant, min^{-1}

N	=	Number of atoms
f_L	=	Letdown flow, gal/min
V	=	Reactor Coolant System volume, gal
η	=	L/D demineralizer efficiency (assumed = 1)

$$\frac{dN}{dt} = 0 \text{ at equilibrium}$$

$$0 = P_n - \lambda N - [f_L \eta/V]N$$

$$P_n = \lambda N + [f_L \eta/V]N$$

$$P_n = N[\lambda + (f_L \eta/V)]$$

This formula is utilized to obtain nuclide production rate prior to the start of an accident. It is not necessary to include a break flow term at this point. Additionally, it is not necessary to account for parent, grandparent, or great-grandparent production of the nuclide of interest (which is part of a radioactive decay chain). We are calculating an equilibrium nuclide production rate; whether the nuclide is produced directly from fission and release from the fuel pin plenum or from decay of a precursor is of no consequence.

Calculation of concentration versus time based on production rate:

$$\frac{dN}{dt} = P_n - \lambda_d N - \lambda_{bf} N$$

Where:

λ_d	=	Nuclide decay lambda, min^{-1}
λ_{bf}	=	Nuclide loss due to break flow, (lbm/min / lbm)

$$\frac{dN}{dt} = P_n - N[\lambda_d + \lambda_{bf}]$$

Rearranging the above equation we obtain:

$$\frac{dN}{dt} + N[\lambda_d + \lambda_{bf}] = P_n$$

$$\frac{dN}{dt} + C1N = C2$$

Where:

$C1 = \lambda_d + \lambda_{bf}$
$C2 = P_n$

Integrating Factor = $e^{C1(t)}$

$$d e^{C1(t)} N = C2 e^{C1(t)} dt$$

$$e^{C1(t)} N = \int C2 e^{C1(t)} dt$$

$$e^{C1(t)} N = \frac{C2}{C1} e^{C1(t)} + C3$$

Where C3 is the constant of integration.

$$N = \frac{C2}{C1} + C3\, e^{-C1(t)}$$

At $t = 0$, $N = N_o$

$$N_o = \frac{C2}{C1} + C3$$

$$C3 = N_o - \frac{C2}{C1}$$

$$N = \frac{C2}{C1} + \left[N_o - \frac{C2}{C1}\right] \left[e^{-C1(t)}\right]$$

$$N = N_o e^{-C1(t)} + \frac{C2}{C1} \left[1 - e^{-C1(t)}\right]$$

Substituting for C1 and C2, we obtain:

$$N = N_o e^{-(\lambda_d+\lambda_{bf})t} + \frac{P_n}{\lambda_d+\lambda_{bf}} \left[1 - e^{-(\lambda_d+\lambda_{bf})t}\right]$$

We may substitute the expression derived for P_n earlier into this equation. The Standard Review Plan, Section 15.1.5, Appendix A, requires that the production

rate be set equal to 500 times the normal equilibrium production rate. Hence, we have:

$$N = N_0 e^{-(\lambda_d+\lambda_{bf})t} + \frac{500N_0(\lambda+(f_L\eta/V))}{\lambda_d+\lambda_{bf}}[1 - e^{-(\lambda_d+\lambda_{bf})t}]$$

Recall that $A = N\lambda$; multiplying both sides by λ yields:

$$A = A_0 e^{-(\lambda_d+\lambda_{bf})t} + \frac{500A_0(\lambda+(f_L\eta/V))}{\lambda_d+\lambda_{bf}}[1 - e^{-(\lambda_d+\lambda_{bf})t}]$$

We may also substitute concentration for activity, and the units remain consistent.

$$C = Co_p\, e^{-(\lambda_d+\lambda_{bf})t} + \frac{500Co_i\,(\lambda+(f_L\eta/V))}{\lambda_d+\lambda_{bf}}[1 - e^{-(\lambda_d+\lambda_{bf})t}]$$

In the above expression, Co_i is the initial Reactor Coolant System concentration of nuclide *i* at equilibrium. The term Co_p is the concentration calculated for the previous time step in Excel.

S/G Iodine Concentration

The following mathematical function is derived to model iodine concentration versus time in the S/Gs. It is only applicable to the intact S/G cases, as a flashing fraction of 1.0 is assumed for the faulted S/G, which will release all of the iodine rather than holding it up in the S/G.

$$\frac{dSGI}{dt} = \frac{BF(1-FF-PB+(SF*FF))*RSCI}{SGM} - \lambda\,(SGI) - \frac{SGI*SR*PF}{SGM}$$

Where:

SGI	= S/G Iodine Concentration, Ci/lbm
BF	= Break Flow, lbm/min
FF	= Flashing Fraction
PB	= Primary Bypass (liquid entrained in the flash fraction)
RCSI	= Reactor Coolant System Iodine Concentration, Ci/lbm
SR	= S/G Steaming Rate, lbm/min
PF	= S/G Partition Factor
SGM	= S/G Mass, lbm
SF	= Scrubbing Fraction (iodine removal fraction from flash fraction)

$$\frac{dSGI}{dt} = \frac{BF(1-FF-PB+(SF*FF))*RSCI}{SGM} - (SGI) * \frac{\lambda + (SR*PF)}{SGM}$$

Solving in the same manner as above, we obtain:

$$SGI = SGI_0 e^{-(\lambda+(SR*PF)/SGM)t} + \frac{(BF(1-FF-PB+(SF*FF))*RCSI)/SGM}{\lambda + ((SR*PF)/SGM)} [1 - e^{-(\lambda+(SR*PF/SGM),t}]$$

Dose Conversion Factors

1 Sievert = 100 Rem (Dose Equivalent) (see reference 8.23, page 466)

1 Becqerel = 1 DPS = 2.703E-11 Ci = 27 pCi (see reference 8.23, page 22)

The ICRP-30 DCFs are in units of Sv/Bq, so these values must be converted to mRem/pCi with the following conversion factor: 1E5/27 = 3704.

Nuclide	ICRP-30	Conversion	mRem/pCi
I-131	2.9E-7	3704	1.07E-3
I-132	1.7E-9	3704	6.3E-6
I-133	4.9E-8	3704	1.81E-4
I-134	2.9E-10	3704	1.07E-6
I-135	8.5E-9	3704	3.15E-5

Reference 8.9 is used to obtain thyroid dose conversion factors (ICRP-30). Dose conversion factors from reference 8.10 are used to calculate whole body dose.

Calculation of Steam Released From Decay Heat and Cooldown Rate

To model steam releases versus time for the intact S/Gs, the following mathematical function is used.

Steam Mass Released = $[DH + (CR * RCS_{mass})\, C_p]\ [1 / h_{fg}]\ [1/60]$

Where:

DH	= Core Decay Heat, BTU/Hr
CR	= Cooldown Rate, °F/Hr
RCS_{mass}	= Reactor Coolant System Mass, lbm
C_p	= Specific Heat, BTU/lbm-°F = 1
h_{fg}	= Heat of Vaporization, BTU/lbm
1/60	= Conversion, 1 Hr per 60 Minutes

Core decay heat is calculated with ANSI/ANS-5.1 methodology. The model has been certified in reference 8.16. The inputs for the core decay heat calculation are as outlined in assumption 7.15.

Calculation of Iodine Released

IR = [(RCSI * BF * (FF + PB - (SF * FF))) + (SR * SGI * PF)] * ΔT

Where

RCSI = Reactor Coolant System Iodine concentration, Ci/lbm
BF = Break flow, lbm/min
FF = Flashing fraction
PB = Primary bypass fraction
SF = Scrubbing fraction
SR = Steaming rate, lbm/min
SGI = S/G iodine concentration, lbm/min
PF = Partition fraction
ΔT = Min

Calculation of Thyroid Dose

Dose = IR * χ / Q * BR * DCF * 1E9

Where

IR = Iodine Released, Ci
χ / Q = Dispersion Factor, s / m^3
BR = Breathing Rate, m^3 / s
DCF = Dose Conversion Factor, mRem / pCi
1E9 = Conversion Factor, Rem-pCi / mRem-Ci
= 1E-3 Rem/mRem * 1E12 pCi/Ci

Calculation of Whole Body Dose

WB Dose = RCSC * PSLR * χ / Q * DCF * ΔT * 31.71

Where

RCSC = Reactor Coolant System noble gas concentration, Ci/lbm
PSLR = Primary to secondary leak rate, lbm / min
χ / Q = Dispersion factor, s / m^3
DCF = Dose Conversion Factor, mRem-m^3 / pCi-yr
ΔT = Time, min
31.71 = Conversion factor, yr-pCi-Rem / sec-Ci-mRem

= (1E12 pCi/Ci / 3.1536E7 sec/Yr) * 1E-3 Rem/mRem

RCSC in the equation above is also decayed versus time based on the decay λ for the given nuclide.

Calculation of Nuclide Decay λs

Decay constants are included in the Excel spreadsheet for use in depletion models delineated earlier in section 9 of this calculation. Decay constants are calculated as follows, and are in units of min^{-1}.

$$\lambda = \frac{\ln 2}{T_{1/2}} = \frac{0.693}{T_{1/2}}$$

Values for $T_{1/2}$ are taken from reference 8.18, and also verified with reference 8.24.

Nuclide	T½	λ
Kr-83m	109.8	6.31E-3
Kr-85m	268.8	2.58E-3
Kr-85	5.634E6	1.23E-7
Kr-87	76.3	9.08E-3
Kr-88	170.4	4.07E-3
Xe-131m	17,136	4.04E-5
Xe-133m	3150.7	2.20E-4
Xe-133	7552.8	9.18E-5
Xe-135m	15.65	4.43E-2
Xe-135	544.98	1.27E-3
Xe-138	14.17	4.89E-2
I-131	11,578	5.99E-5
I-132	138	5.02E-3
I-133	1248	5.55E-4
I-134	52.6	1.32E-2
I-135	396.6	1.75E-3

The equations programmed into Excel are contained in attachment 1.

10.0 Inputs and Results

Inputs for the Intact S/G

The iodine source term (iodine spectrum to yield 1 μCi/gm) and the total Reactor Coolant System gross activity (for calculation of whole body doses) are taken from reference 8.15.

ICRP-30 DCFs are used in the calculation of thyroid dose; Regulatory Guide 1.109 DCFs are used in the calculation of whole body dose.

Maximum primary to secondary leak rate allowed by Catawba Technical Specifications = 0.4 gpm total and 150 gpd per S/G.

(150 gal/day) / (24 hours/day * 60 min/Hour) = 0.1042 gpm per S/G

0.1042 gpm * 4 S/Gs = 0.4168 gpm total; hence there is little difference in the total amount allowed for all four S/Gs and four times the maximum amount allowed in an individual S/G. The primary to secondary leak rate in the intact S/Gs will be taken to be 0.1042 gpm.

PSLR = 0.1042 gal/min / 0.14 gal/lbm = 0.744 lbm/min

Flashing fraction = 0.15 (see assumption 7.7)

Primary bypass fraction = 0.05 (see assumption 7.8)

EAB χ/Q = 4.78E-4 (see reference 8.14)

LPZ χ/Q = 6.85E-5 (see reference 8.14)

60 °F/Hr Reactor Coolant System cooldown rate in the calculation of steam releases (see assumption 7.17).

Breathing rate = 3.47E-4

Core decay heat is calculated with ANSI/ANS-5.1 methodology, which is certified in reference 8.16. The assumptions used to obtain core decay heat values versus decay time are found in assumption 7.15. Results for the first five decay times are contained in attachment 2. Results for additional decay times were calculated with the same methodology, but results were not attached due to the large quantity of computer output.

Results for the Intact S/G

EAB whole body dose results (krypton) = 3.07E-5 Rem

EAB whole body dose results (xenon) = 6.11E-5 Rem

EAB Total = (3.07E-5 + 6.11E-5) * 3 Intact S/Gs = **2.754E-4 Rem**

LPZ whole body dose results (krypton) = 9.52E-6 Rem

LPZ whole body dose results (xenon) = 3.33E-5 Rem

LPZ Total = (9.52E-6 + 3.33E-5) * 3 Intact S/Gs = **1.285E-4 Rem**

EAB thyroid dose results (pre-existent) = 9.22E-2 Rem

EAB Total (pre-existent) = 9.22E-2 * 3 Intact S/Gs = **2.766E-1 Rem**

LPZ thyroid dose results (pre-existent) = 5.94E-2 Rem

LPZ Total (pre-existent) = 5.94E-2 * 3 Intact S/Gs = **1.782E-1 Rem**

EAB thyroid dose results (coincident) = 5.58E-2 Rem

EAB Total (coincident) = 5.58E-2 * 3 Intact S/Gs = **1.674E-1 Rem**

LPZ thyroid dose results (coincident) = 1.36E-1 Rem

LPZ Total (coincident) = 1.36E-1 * 3 Intact S/Gs = **4.08E-1 Rem**

Output for the intact S/G case is contained in attachment 3 of this calculation.

Inputs for the Faulted S/G

CASE 1 (0.6 gpm Increased PSLR)

The iodine source term (iodine spectrum to yield 1 μCi/gm) and the total Reactor Coolant System gross activity (for calculation of whole body doses) are taken from reference 8.15.

ICRP-30 DCFs are used in the calculation of thyroid dose; Regulatory Guide 1.109 DCFs are used in the calculation of whole body dose.

PSLR = (0.1042 + 0.6 gal/min) / 0.14 gal/lbm = 5.03 lbm/min

Flashing fraction = 1.0 (see assumption 7.4)

Primary bypass fraction = 0.0

EAB χ/Q = 4.78E-4 (see reference 8.14)

LPZ χ/Q = 6.85E-5 (see reference 8.14)

Breathing rate = 3.47E-4

CASE 2 (2.73 gpm Increased PSLR)

Same as CASE 1 input above, except for the primary to secondary leak rate, calculated as follows:

PSLR = (0.1042 + 2.73 gal/min) / 0.14 gal/lbm = 20.244 lbm/min

Results for the Faulted S/G

CASE 1

EAB whole body dose results (krypton) = 2.07E-4 Rem

EAB whole body dose results (xenon) = 4.13E-4 Rem

EAB Total = 2.07E-4 + 4.13E-4 = **6.2E-4 Rem**

LPZ whole body dose results (krypton) = 6.43E-5 Rem

LPZ whole body dose results (xenon) = 2.25E-4 Rem

LPZ Total = 6.43E-5 + 2.25E-4 = **2.89E-4 Rem**

EAB thyroid dose results (pre-existent) = **2.88 Rem**

LPZ thyroid dose results (pre-existent) = **1.59 Rem**

EAB thyroid dose results (coincident) = **1.79 Rem**

LPZ thyroid dose results (coincident) = **3.92 Rem**

CASE 2

EAB whole body dose results (krypton) = 8.34E-4 Rem

EAB whole body dose results (xenon) = 1.66E-3 Rem

EAB Total = 8.34E-4 + 1.66E-3 = **2.49E-3 Rem**

LPZ whole body dose results (krypton) = 2.59E-4 Rem

LPZ whole body dose results (xenon) = 9.07E-4 Rem

LPZ Total = 2.59E-4 + 9.07E-4 = **1.17E-3 Rem**

EAB thyroid dose results (pre-existent) = **11.6 Rem**

LPZ thyroid dose results (pre-existent) = **6.36 Rem**

EAB thyroid dose results (coincident) = **7.19 Rem**

LPZ thyroid dose results (coincident) = **15.7 Rem**

Output for the faulted S/G cases is contained in attachment 4 of this calculation.

Total Dose Calculation Results

Adding the dose results of the intact S/Gs and the faulted S/Gs yields the following tables:

Case 1 (0.6 gpm Increased Primary to Secondary Leak Rate)

Calculation Type	Whole Body (Rem)	Thyroid Dose (Rem) Pre-existent Iodine Spike	Thyroid Dose (Rem) Accident-Initiated Iodine Spike
EAB (2 Hour)	8.95E-4	3.16	1.96
LPZ (8 Hour)	4.18E-4	1.77	4.33

Case 2 (2.73 gpm Increased Primary to Secondary Leak Rate)

Calculation Type	Whole Body (Rem)	Thyroid Dose (Rem) Pre-existent Iodine Spike	Thyroid Dose (Rem) Accident-Initiated Iodine Spike
EAB (2 Hour)	2.77E-3	11.88	7.36
LPZ (8 Hour)	1.30E-3	6.54	16.11

As can be seen from the above tables, the results are within the pertinent 10 CFR Part 100 limits (i.e., 300 Rem thyroid for the pre-existent iodine spike case, and "a small fraction of Part 100 limits," or 10% of the Part 100 limits for the coincident iodine spike case, as well as 2.5 Rem whole body).

11.0 Delineation of Conservatisms in Analysis

Primary to Secondary Leak Rate

Per discussions with Westinghouse, the values of increased primary to secondary leak rate are based on a primary to secondary ΔP of 2560 psi at a Reactor Coolant System temperature of 616 °F . It should be noted that this is a highly conservative approach. This ΔP would only be applicable to a feedline break (which increases Reactor Coolant System temperature due to the loss of heat sink). A steam line break rapidly decreases the Reactor Coolant System

temperature due to the blowdown of steam from the faulted S/G. The Catawba Nuclear Station Emergency Procedures direct the operator to arrest a subsequent heatup and repressurization of the Reactor Coolant System, after initiation of safety injection, by feeding and steaming the intact S/Gs. The mechanism to cause a coincident iodine spike is present in a steam line break. If a fuel pin is leaking plenum gases at some escape rate and a depressurization of the Reactor Coolant System occurs, the rate at which the plenum gases escape may be increased. However, this mechanism is not present in a feedline break because the Reactor Coolant System pressure increases. A pre-existent iodine spike may be present during a feedline break; however, the pre-existent iodine spike is not the limiting case (as seen in the above tables, the coincident iodine spike is the limiting case). Therefore, the use of the primary to secondary leak rates based on a ΔP of 2560 psi does not coincide with the mechanism for a coincident iodine spike, which is the most limiting case.

Iodine Concentrations

The following iodine concentration (I-131 Dose Equivalent Concentration) is calculated from individual nuclide values calculated by the spreadsheet at a time of 8 hours:

Nuclide	Concentration (Ci/lbm)	DCF (ICRP-30)	Ci*DCF
I-131	8.9782E-2	1.07E-3	9.607E-5
I-132	6.8863E-3	6.30E-6	4.338E-7
I-133	1.8614E-1	1.81E-4	3.369E-5
I-134	4.5067E-2	1.07E-6	4.822E-8
I-135	1.3845E-1	3.15E-5	4.361E-6

Σ (Ci*DCF) = 1.346E-4

Σ (Ci*DCF) / I-131DCF = 1.258E-1 I-131 D.E.C. (Ci/lbm)

(1.258E-1 Ci/lbm*1E6 μCi/Ci)/453.5924 gm/lbm = 277.34 μCi/gm I-131 D.E.C.

The above values are obtained from the speadsheet which has used the previously derived model to calculate nuclide concentrations, based on 1) An appearance rate calculated for 1 μCi/gm I-131 D.E.C., and 2) An increase in this appearance rate of 500 times (per the Standard Review Plan). This is a highly conservative model. Reference 8.26 contains the following equation for spiking factor (increase in coolant activity following a transient):

$$S = \frac{K\,(\Delta P)}{P_2}$$

Where:

S	= Spiking Factor
K	= Gap-to-Coolant Rate Constant (250)
ΔP	= The Differential Pressure for the Transient
P_2	= The Power or Pressure at the Beginning of the Transient

To prevent rapid cooldown and PTS concerns for the Reactor Coolant System, the faulted S/G will not be fed with auxiliary feedwater. However, at a Reactor Coolant System pressure of 385 psig and temperature of 350 °F (reference 8.35), the Residual Heat Removal System will be aligned to remove core decay heat, and the faulted S/G will then be able to be filled with water inventory without causing cooldown concerns, essentially terminating the releases through the faulted S/G (due to the decrease in primary to secondary ΔP as well as the presence of scrubbing and partitioning of the primary to secondary leakage). The following maximum spiking factor is calculated with the EPRI model, assuming a beginning Reactor Coolant System pressure of 2235 psig:

$S = (250)\,(2235 - 385) / 2235 = 206.94$

As can be seen from the I-131 D.E.C. calculated above from Standard Review Plan methodology, the value calculated with EPRI methodology is 74.6% of the Standard Review Plan value. This also assumes that the beginning Reactor Coolant System concentration is 1 μCi/gm I-131 D.E.C. Operating history for Catawba Nuclear Station shows that the normal Reactor Coolant System concentration is approximately 0.003 - 0.005 μCi/gm I-131 D.E.C. Use of a concentration value of 0.005 μCi/gm would yield another factor of conservatism of 200.

Reference 8.31 recommends that a new value be considered for release rate from the fuel sheath to coolant due to the conservatism inherent in the Standard Review Plan methodology (this report finds the SRP method conservative by at least a factor of 10).

S/G Iodine Partition Coefficient

The Standard Review Plan partition coefficient of 100 (partition fraction = 0.01) is used in this analysis. While the intact S/G is not determined to be a significant dose contributor (based on the above analysis), a more realistic value for partition coefficient would decrease the calculated offsite dose. Reference 8.32 provides evidence which supports a value of 1E3 and higher for a S/G partition coefficient.