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July 22, 2014
L-14-245

10 CFR 50.90

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

SUBJECT:
Perry Nuclear Power Plant
Docket No. 50-440, License No. NPF-58
Response to Request for Additional Information Regarding Request for Licensing Action on Alternative Accident Source Term Radiological Dose Calculations (TAC No. MF3197)

A request for licensing action regarding alternative accident source term radiological dose calculations was submitted to the Nuclear Regulatory Commission (NRC) by letter dated December 6, 2013 (Accession No. ML13343A013). The NRC staff requested additional information in a letter dated June 24, 2014 (Accession No. ML14162A409). The requested information is provided in Attachment 1. In addition, the effects of several necessary changes to the supporting dose calculations are presented in Attachment 2. No changes were identified to the previously provided Significant Hazards Consideration.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at 330-315-6810.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 22, 2014.

Sincerely,

Ernest J. Harkness

Attachments:
1. Response to Request for Additional Information on Radiological Dose Calculations
2. Effects of Necessary Changes to Supporting Radiological Dose Calculations

cc: NRC Region III Administrator
NRC Resident Inspector
NRC Project Manager
State of Ohio (NRC Liaison)
Utility Radiological Safety Board

FirstEnergy Nuclear Operating Company (FENOC) provided a request for licensing action (RLA) regarding alternative accident source term radiological dose calculations to the Nuclear Regulatory Commission (NRC) in a letter dated December 6, 2013. The NRC staff requested additional information in a letter dated June 24, 2014. The requested information is identified using bold text, followed by the FENOC response.

1. **For those changes to the current licensing basis (CLB) parameters used in the affected dose consequence analyses, provide additional information describing for each affected design basis accident, all the basic parameters used in the dose consequence analyses. For each parameter, please indicate the CLB value, the revised value where applicable, and the basis for any changes made to the CLB values. The U.S. Nuclear Regulatory Commission (NRC) staff requests that this information be presented in separate tables for each accident evaluated.**

Response: A matrix is provided for the loss of coolant accident (LOCA), the control rod drop accident (CRDA) scenarios, and the main steam line break outside containment (MSLBOC) analyses.

LOCA

Parameter	**Current Licensing Basis** **[Existing location]**	**Proposed Licensing Basis** **[Proposed location]**	**Basis for changes**
Core Source Term Basis	General Electric (GE)12/ GE14 [GE12= Amendment 112 (power uprate), NRC Safety Evaluation (SE) pg. 4; GE14= 10 CFR 50.59 review when converted from GE12]	Global Nuclear Fuel (GNF)2 [Updated Safety Analysis Report (USAR) pg. 15.6-65]	GNF2 fuel design to be introduced at PNPP during next fuel cycle
Power Level	102% of rated thermal power = 3758 megawatts thermal (MW_t) x 1.02 ≈ 3833 MW_t [Amendment 112; NRC SE pg. 45]	102% of rated thermal power ≈ 3833 MW_t [USAR pg. 15.6-24 & 15.6-58 thru 60]	No change (existing LOCA dose calculation in support of Amendment 112 did assume 3833 MW_t although several USAR pages simply reflect the licensed 100% RTP value of 3758 MW_t)

Isotopes in source term	76 isotopes [Calculations supporting Amendment 103 (the PNPP pilot plant AST amendment)]	60 isotopes [Request for licensing action (RLA) submittal dated 12/6/2013, and Amendment TBD dated TBD]	Based on isotopes used in the RADTRAD computer code
Dose Conversion Factors	Federal Guidance Report (FGR) 11 [USAR pg. 15.0-37]	FGR 11 and 12 [USAR pg. 15.0-37]	Use of Regulatory Guide (RG) 1.183, Section 4.1.2 and 4.1.4.
Gap Release Timing	30 seconds [This assumption was not specified in the USAR but was assumed in the calculations supporting Amendment 103]	2 minutes (BWR) [USAR pg. 15.6-22 is revised to reference RG 1.183 for timing of the releases, and Table 4 of RG 1.183 specifies 2 minutes for onset of BWR gap release]	Per RG 1.183, Table 4; BWR-specific
Credit for decay during 2 minutes before start of the gap release.	Not considered	Decay considered [USAR pg. 15.6-23]	Utility decision since RG 1.183 does not specify this
Suppression Pool Scrubbing	No Credit [USAR 15.6-27]	No Credit [USAR 15.6-27]	No change
Drywell Volume	2.765×10^5 feet $(ft)^3$ [USAR pg. 15.6-58]	2.765×10^5 ft^3 [USAR pg. 15.6-58]	No change
Containment Volume (excluding drywell)	1.1654×10^5 ft^3 [USAR pg. 6.5-56 & 15.6-58]	1.1654×10^6 ft^3 [USAR pg. 6.5-56 & 15.6-58]	No change (Note: Table 6.5-9 & Table 15.6-12a markups are correcting typos)
Volume of Sprayed Region	4.81×10^5 ft^3 [USAR pg. 15.6-59]	4.812×10^5 ft^3 [USAR pg. 15.6-59]	Essentially no change (third decimal point)
Volume of Containment Unsprayed Region	6.84×10^5 ft^3 [USAR pg. 15.6-59]	6.842×10^5 ft^3 [USAR pg. 15.6-59]	Essentially no change (third decimal point)
Volume of Total Unsprayed Regions (including drywell)	9.607×10^5 ft^3 [not currently specified separately in the USAR]	9.607×10^5 ft^3 [USAR pg. 15.6-59]	No change

Flow Rate from Drywell to Unsprayed Region of the Containment	0–2 hours (hrs.) 3000 cubic feet per minute (cfm) 2–720 hrs. 2.77×10^5 cfm [USAR Table 15.6-12b on pg. 15.6-59]	0 – 0.5 hour (hr.) 0 cfm 0.5 – 2 hrs. 3000 cfm 2 – 720 hrs. 2.77×10^5 cfm [USAR Table 15.6-12b on pg. 15.6-59]	No actual change (Note: USAR Table 15.6-12b did not reflect the zero flow rate out of the drywell during the first 0.5 hour period that was in the existing calculation)
Flow Rate from Unsprayed Region of the containment back to Drywell	0 – 2 hrs. 0 cfm 2 – 720 hrs. 2.77×10^5 cfm [USAR pg. 15.6-59]	0 – 2 hrs. 0 cfm 2 – 720 hrs. 2.77×10^5 cfm [USAR pg. 15.6-59]	No actual change (USAR markup of page 15.6-59 simply rounds 2.765 to 2.77×10^5 cfm, which is the value in the existing calculation)
Primary Containment Leakage Rate (sprayed and unsprayed regions leakage)	0.2 percent of containment atmosphere per day (L_a) [USAR pg. 6.2-72 & 73; Technical Specification (TS) Section 5.5.12]	0.2 percent of containment atmosphere per day (L_a) [USAR pg. 6.2-72 & 73; TS Section 5.5.12]	No change
Percent of Primary Containment Leakage that goes into the annulus	89.92 percent (0.8992 La) [USAR 15.6-31]	89.92 percent (0.8992 La) [USAR 15.6-31]	No change (Rev. 0 of new LOCA calculation used 100 percent, but Rev. 1 corrects that to 89.92 percent)
Percent of Primary Containment Leakage that bypasses the Secondary Containment	10.08 percent (0.1008 L_a) [USAR pg. 15.6-31]	10.08 percent (0.1008 L_a) [USAR pg. 15.6-23 & 31]	No change
Credit for Reduction in Primary Containment Leakage based on containment pressure after 24 hours	Not taken	Based on containment pressure reduction after 24 hours, leakage is reduced to 69 percent of the value used during the first 24 hours [USAR pg. 15.6-30]	Credit taken for reduction in containment pressure, as permitted in RG 1.183. See pages 29 & 30 of the RLA Degree of Conformance matrix

Credit for Reduction In Secondary Containment Bypass Leakage based on containment pressure after 24 hours	Not taken	After 24 hours, leakage is reduced to 69 percent of the value used during the first 24 hours [USAR pg. 15.6-30]	Credit was taken for reduction in containment pressure, as permitted in RG 1.183
Containment Spray Initiation	10 minute auto initiation (based on high pressure and LOCA signal) [USAR 15.6-32]	30 minute manual initiation [USAR 15.6-32]	Initiation time is conservatively tripled, because auto-initiation on containment high pressure might not be achieved
Containment Spray Duration	24 hours [USAR pg. 6.5-12 & 15.6-32]	24 hours [USAR pg. 6.5-12 & 15.6-32]	No change
Spray Fall Height	53.2 feet [USAR pg. 15.6-59]	54.05 feet [USAR pg. 15.6-59]	New calculation
Containment Spray- Aerosol (particulate) Removal	Sprayed region of the containment modelled using STARNAUA methodology [USAR pg. 6.5-13, 14, 56, & 58] (Note: In Am. 103, NRC used RADTRAD 90th percentile uncertainty distributions, per pg. 10 of the NRC SE for Am. 103)	Sprayed region of the containment modelled using RADTRAD Powers model with 10th percentile uncertainty distribution. Particulate removal coefficient due to sprays is reduced by a factor of 10 after the aerosol mass is depleted by a factor of 50 [USAR pg. 6.5-13, 14, & 56]	Conservative RADTRAD model assumption, RG 1.183, Appendix A, Section 3.3
Other Inputs to Spray Modeling	Various (See USAR Table 6.5-9 "Input Parameters For The Spray Removal Analysis" [USAR pg. 6.5-56]	Q = 0.0621 (Spray Flux) Alpha = 1.422 (Ratio of unsprayed volume to sprayed volume) [USAR pg. 6.5-56]	Powers spray model inputs

Containment Spray- Elemental Iodine Removal	Sprayed region of the containment modelled using STARNAUA methodology [USAR pg. 6.5-13, 14, 56, & 58]	Sprayed region of the containment modelled using Standard Review Plan (SRP) 6.5.2 guidance. Elemental Iodine removal by sprays is terminated when a DF of up to 200 is reached [USAR pg. 6.5-13, 14, & 56]	RG 1.183 and SRP 6.5.2
Natural Deposition-Aerosol (particulate) Removal	Drywell region modelled using STARNAUA methodology [USAR pg. 15.6-28 & 29] (Note: In Am. 103, NRC used RADTRAD 90th percentile uncertainty distributions, per pg. 7 of the NRC SE for Am. 103)	Drywell and unsprayed volume of containment modelled using RADTRAD Powers Model for aerosol removal with 10th percentile uncertainty distribution [USAR 15.6-28 & 29]	Conservative RADTRAD model assumption
Natural Deposition-Elemental Iodine Removal	Drywell region modelled using STARNAUA methodology (USAR pg. 15.6-30]	Drywell, sprayed, and unsprayed volume of containment modelled using SRP 6.5.2 guidance. Similar to the spray assumptions, elemental Iodine removal is terminated when a DF of up to 200 is reached [USAR pg. 6.5-13, & Table 6.5-11]	RG 1.183 and SRP 6.5.2
Engineered Safety Feature (ESF) Leakage Pathway Direct to Environment	Time (hours) / Leak Rate 0-24: 15 gallons per hour (gph) 24–24.5: 15 gph + 50 gpm 24.5-720: 15 gph [USAR pg. 15.6-33 & 60]	Time (hours) / Leak Rate 0-24: 15 gph 24–24.5: 15 gph + 50 gpm 24.5-720: 15 gph [USAR pg. 15.6-23, 33 & 60]	No change
Offsite Breathing Rates (meters (m)^3/ second (sec))	Time (hours) 0 – 8: 3.47×10^{-4} 8 – 24: 1.75×10^{-4} 24-720: 2.32×10^{-4} [USAR pg. 15.0-37]	Time (hours) 0 – 8: 3.5×10^{-4} 8 – 24: 1.8×10^{-4} 24-720: 2.3×10^{-4} [USAR pg. 15.0-37]	RG 1.183, Section 4.1.3

Main Steam Line (MSL) leak rate (standard conditions)	250 standard cubic feet per hour (scfh) total, 100 scfh maximum per line (under standard conditions). Modelled as 100 scfh to the broken MSL, and 150 scfh to the intact MSL's [TS Surveillance Requirement (SR) 3.6.1.3.10]	250 scfh total, 100 scfh maximum per line (under standard conditions). Modelled as 100 scfh to the broken MSL, and 150 scfh to the intact MSL's [TS SR 3.6.1.3.10]	No change
Main Steam Line leak rate converted from standard to non-standard conditions (assuming elevated post-accident temperatures) during first 24 hours	Flow rate from drywell to all steam lines (both broken and intact) = 298 cfh for first 7484 seconds, and 247 cfh until 24 hours Flow rate between MSIVs in one steam line = 191 cfh (Note: USAR Table 15.6-12a says cfm vs. cfh, which was a typographical error) [USAR pg. 15.6-58]	Flow rate from drywell to the broken steam line = 1.987 cfm for first 7484 seconds, and 1.647 cfm until 24 hours Flow rate from drywell to the intact steam lines = 2.98 cfm for first 7484 seconds, and 2.47 cfm until 24 hours Flow rate in one steam line, between the MSIVs = 3.183 cfm Flow rate in the other (the intact) steam lines, between the MSIVs = 4.775 cfm [USAR pg. 15.6-58]	No change (values in USAR Table 15.6-12a appear to be changed, but that is due to mathematical conversion of existing assumed flow rates from cfh to cfm (engineering unit change only), and inclusion of more detail into this table about how flow rates to (and through) the main steam lines are attributed to the broken versus the intact main steam lines)
Credit for Main Steam Line leakage reduction associated with decreased pressure after 24 hours.	Not taken [USAR 15.6-25 & 58]	After 24 hours, leakage from the drywell to the Main Steam Lines is reduced to 69 percent of the value used during the first 24 hrs. However, the leak rate assumed between the MSIVs is conservatively not reduced after 24 hours [USAR pg. 15.6-25 & 58]	Some credit was taken for reduction in containment pressure, as permitted in RG 1.183

Aerosol Removal Efficiencies in Steam Lines	Broken steam line – between MSIVs (100 scfh) - For Particulate Iodine, solubles, and insoluble 68.1% from 0 to 0.5 hr. 83.5% from 0.5 to 1.5 hrs. 87.13% from 1.5 to 3 hrs. 89% from 3 to 5 hrs. 86.14% from 5 to 7 hrs. 81.85% from 7 to 9 hrs. 76.9% from 9 to 11 hrs. 36.53% from 11 to 720 hrs. -For Elemental Iodine – 45% Intact Steam lines RPV to IB MSIV (150 scfh) - For Particulate Iodine, solubles and insoluble 72.06% from 0 – end - For Elemental Iodine – 45% Intact Steam Lines between MSIVs - For Particulate Iodine, solubles, and insoluble 71.4% from 0 to 0.5 hr. 81.3% from 0.5 to 1.5 hrs. 83.61% from 1.5 to 3 hrs. 84.49% from 3 to 5 hrs. 83.39% from 5 to 7 hrs. 79.98% from 7 to 9 hrs. 75.58% from 9 to 11 hrs. 38.07% from 11 to 720 hrs. -For Elemental Iodine – 45% [USAR pg. 15.6-26 & 27 provide general discussion of steam line treatment]	Broken steam line – between MSIVs (100 scfh) - For Particulate Iodine, solubles, and insoluble 68.1% from 0 to 0.5 hr. 83.5% from 0.5 to 1.5 hrs. 87.13% from 1.5 to 3 hrs. 89% from 3 to 5 hrs. 86.14% from 5 to 7 hrs. 81.85% from 7 to 9 hrs. 76.9% from 9 to 11 hrs. 36.53% from 11 to 720 hrs. -For Elemental Iodine – 45% Intact Steam lines RPV to IB MSIV (150 scfh) - For Particulate Iodine, solubles and insoluble 72.06% from 0 – end - For Elemental Iodine – 45% Intact Steam Lines between MSIVs - For Particulate Iodine, solubles, and insoluble 71.4% from 0 to 0.5 hr. 81.3% from 0.5 to 1.5 hrs. 83.61% from 1.5 to 3 hrs. 84.49% from 3 to 5 hrs. 83.39% from 5 to 7 hrs. 79.98% from 7 to 9 hrs. 75.58% from 9 to 11 hrs. 38.07% from 11 to 720 hrs. -For Elemental Iodine – 45% [USAR pg. 15.6-26 & 27 provide general discussion]	No change
Annulus Exhaust Gas Treatment (AEGT) system recirculation to the annulus	Not considered, such that flow rate to the environment = 2000 cfm [Not currently specified in the USAR]	Not considered, such that flow rate to the environment = 2000 cfm [USAR pg. 15.6-31]	No change
AEGT Filtration (High Efficiency Particulate Air (HEPA) filter)	99 percent efficiency [USAR pg. 15.6-31 & 59]	99 percent efficiency [USAR pg. 6.5-4; 15.6-31 & 59]	No change

AEGT Filtration (Charcoal)	Zero percent efficiency [USAR pg. 6.5-3]	Zero percent efficiency [USAR pg. 6.5-3 & 42]	No change
Exclusion Area Boundary (EAB) X/Q (sec/m^3)	$4.3x10^{-4}$ [NRC Safety Evaluation for Amendment 103, pg. 15]	$4.3x10^{-4}$ [USAR pg. 15.6-63]	No change
Low Population Zone (LPZ) X/Q (sec/m^3)	Time (hrs) 0-8 $4.8x10^{-5}$ 8-24 $3.3x10^{-5}$ 24-96 $1.4x10^{-5}$ 96-720 $4.1x10^{-6}$ [NRC Safety Evaluation for Amendment 103, pg. 15]	Time (hrs) 0-8 $4.8x10^{-5}$ 8-24 $3.3x10^{-5}$ 24-96 $1.4x10^{-5}$ 96-720 $4.1x10^{-6}$ [USAR pg. 15.6-63]	No change
Control Room X/Q (sec/m^3)	Time (hrs) 0-8 $3.5x10^{-4}$ 8-24 $2.1x10^{-4}$ 24-96 $1.1x10^{-4}$ 96-720 $5.75x10^{-5}$ [NRC Safety Evaluation for Amendment 103, pg. 14]	Time (hrs) 0-8 $3.5x10^{-4}$ 8-24 $2.1x10^{-4}$ 24-96 $1.1x10^{-4}$ 96-720 $5.75x10^{-5}$ [USAR pg. 15.6-63 with corrections to reflect correct CLB]	No change (markups on USAR page 15.6-63 are correcting incorrect information)
Control Room Volume	$3.44x10^5$ ft^3 [USAR pg. 15.6-64]	$3.90x10^5$ ft^3 [USAR pg. 15.6-64]	Calculation for control room volume revised since pilot plant submittal
Control Room Filtration System (HEPA)	95 percent credit for HEPA removal of Particulates. [USAR pg. 15.6-35]	99 percent credit for HEPA removal of Particulates [USAR pg. 6.5-4; 15.6-35]	Acceptable value based on current and future TS test acceptance criterion
Control Room Filtration System (Charcoal)	50 percent charcoal filter efficiency for elemental and organic iodine. [USAR pg. 6.5-4, 15.6-23 & 35]	80 percent charcoal filter efficiency for elemental and organic iodine. [USAR pg. 6.5-4 & 30, 15.6-23 & 35]	Eliminated an unnecessary conservatism; 80 percent value will support a future request for licensing action to revise TS charcoal adsorber testing acceptance criterion to a value of 10% penetration (=90% efficiency) versus the current 2.5% penetration requirement

Control Room Emergency Recirculation System Actuation/Timing of the Start of Filtration	30 minutes [USAR pg. 6.4-8, 15.6-23]	30 minute manual operator actuation [USAR pg. 6.4-8, 14, & 15, 15.6-23 & 35	No credit taken for automatic actuation of filtration or for any reduction in unfiltered in-leakage until 30 minutes, to support a future request for licensing action to remove TS controls over auto-initiation instrumentation
Timing of the Reduction of Unfiltered Inleakage Into the Control Room	1375 cfm unfiltered inleakage starting at time (t)=0, for duration [USAR pg. 6.4-14 & 15, 15.6-35 & 64]	6600 cfm unfiltered inleakage from t=0 to 30 minutes 1375 cfm from t=30 min to 30 days [USAR page 6.4-14 & 15, 15.6-35 & 64]	Conservatively assumes outside air inlet is not isolated until t=30 minutes when Control Room Emergency Recirculation System is manually actuated, to support a future request for licensing action to remove TS controls over auto-initiation instrumentation
Control Room Breathing Rate	3.47×10^{-4} m^3/sec [USAR pg. 15.0-37]	3.5×10^{-4} m^3/sec [USAR pg. 15.0-37]	RG 1.183, Section 4.2.6
Control Room Occupancy Factors	Time (hours) 0 – 24 1.0 24 – 96 0.6 96 – 720 0.4 [not currently specified in the USAR]	Time (hours) 0 – 24 1.0 24 – 96 0.6 96 – 720 0.4 [USAR pg. 15.6-64]	No change

CRDA (Scenario 1 – Standard Review Plan Section 15.4.9)

Parameter	Current Licensing Basis [Current Location]	Proposed Licensing Basis [Proposed Location]	Basis For Changes
Core Source Term basis	GE14 [USAR pg. 15.4-27, 45, & 46]	GNF2 [USAR pg. 15.4-27, 45, & 46]	Change in fuel design

Power Level	Plant startup, at low power level, after operation at 3,833 MW_t [USAR pg. 15.4-26 thru 29, & 44]	Plant startup, at low power level, after operation at 3,833 MW_t [USAR pg. 15.4-26 thru 29, & 44]	No change
Coincident loss of offsite power (LOOP)?	Yes [USAR pg. 15.4-27 & 28]	Yes [USAR pg. 15.4-27 & 28]	No change (per SRP 15.4.9 Section III, Assumption 1)
Isotopes Considered	Considered Iodine, Krypton, and Xenon isotopes [USAR pg. 15.4-29]	Considered Iodine, Krypton, Xenon, Bromine, Cesium, and Rubidium isotopes [USAR pg. 15.4-29]	Added isotopes per RG 1.183, Section 3, Tables 3 & 5
Assumed number of failed fuel rods	1107 [USAR pg. 15.4-27, 28, & 44]	1376 (two full rows of bundles around the dropped control rod) [USAR pg. 15.4-27, 28, & 44]	Conservative increase in number of failed rods
Iodine Fractions, %	Organic 0 Elemental 100 Particulate 0 [USAR pg. 15.4-44]	Organic 0.15 Elemental 4.85 Particulate 95.0 [USAR pg. 15.4-44]	RG 1.183, Appendix C, Item 3.6
Radial Peaking Factor	1.7 [Reload Analysis Parameter]	2.0 [Reload Analysis Parameter]	Current radial peaking factor in analyses
Gap and Melt fraction	Group Gap Melt Noble Gas 10% 100% Halogen 10% 50% [USAR pg. 15.4-29]	Group Gap Melt Noble Gas 10% 100% Halogen 10% 50% Alkali Metals 12% 25% [USAR pg. 15.4-29]	Added alkali metals per RG 1.183, Section 3.2, Tables 1 & 3
Activity released to condenser	Group Noble Gas 100% Halogen 10% [USAR pg. 15.4-29]	Group Noble Gas 100% Halogen 10% Remaining 1% [USAR pg. 15.4-29]	RG 1.183, Appendix C, Item 3.3
Activity available for release from condenser	Group Noble Gas 100% Halogen 10% [USAR pg. 15.4-29]	Group Noble Gas 100% Halogen 10% Particulate 1% [USAR pg. 15.4-29]	RG 1.183, Appendix C, Item 3.4

Condenser Leakage	The leakage rate from the condenser is assumed to be 1% per day for the duration of the event (24 hours). [USAR pg. 15.4-29 (24 hour termination time is from SRP 15.4.9 Assumption III.12)]	The leakage rate from the condenser is assumed to be 1% per day for the duration of the event (24 hours) [USAR pg. 15.4-29]	No change
Release to Environment	All activity leaking from the condenser is assumed to leak directly to the environment without mixing in the turbine building. [USAR pg. 15.4-28 & 44]	All activity leaking from the condenser is assumed to leak directly to the environment without mixing in the turbine building. [USAR pg. 15.4-28 & 44]	No change
Atmospheric Dispersion X/Q	Time EAB LPZ 0-2 hrs. 4.3E-4 4.8E-5 2-8 hrs. -- 4.8E-5 8-24 hrs. -- 3.3E-5 1-4 Days -- 1.4E-5 4-30 Days -- 4.1E-6 [USAR pg. 15.4-45]	Time EAB LPZ CR 0-2 hrs. 4.3E-4 4.8E-5 3.5E-4 2-8 hrs. -- 4.8E-5 3.5E-4 8-24 hrs. -- 3.3E-5 2.1E-4 1-4 Days -- 1.4E-5 1.1E-4 4-30 Days -- 4.1E-6 5.75E-5 [USAR pg. 15.4-45]	No change except for the addition of control room (CR) X/Q's
Control Room Dose	Not calculated [Control room doses were not previously required to be determined for CRDA]	Calculated [USAR pg. 15.4-47]	RG 1.183, Section 4.4
Control Room Emergency Recirculation System Filtration (Charcoal and HEPA)	N/A [Control room doses were not previously required to be determined for CRDA]	Not credited [USAR pg. 6.5-4 & 30, 15.4-45]	No credit taken
AEGT System Filtration (Charcoal and HEPA)	Not credited [Releases are from outside containment, so AEGT system is N/A, therefore it is not credited]	Not credited [USAR pg. 6.5-3, 4, & 42]	No change

CRDA (Scenario 2 – NEDO-31400 "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor")

Parameter {See Scenario 1, with the following changes based on NEDO-31400}	Current Licensing Basis [Current Location]	Proposed Licensing Basis [Proposed Location]	Basis For Changes
Power Level	Plant startup, at low power level, after operation at 3,833 MW_t, aligned to the Offgas system [USAR pg. 15.4-26, 27, 28, & 30]	Plant startup, at low power level, after operation at 3,833 MW_t, aligned to the Offgas system [USAR pg. 15.4-26, 27, 28, & 30]	No change
Coincident loss of offsite power (LOOP)?	No, so Offgas System remains running to hold up all but the noble gases [USAR 15.4-30]	No, so Offgas System remains running to hold up all but the noble gases [USAR 15.4-30]	No change

MSLBOC

Parameter	Current Licensing Basis [Current Location]	Proposed Licensing Basis [Proposed Location]	Basis for Changes
Reactor Coolant Inventory	Original GE-supplied reactor coolant inventory for: a) Design basis analysis = TS pre-accident spike of 4.0 microcuries (μCi)/ gram (gm), and b) Realistic analysis = TS maximum equilibrium value of 0.2 μCi/gm [USAR pg. 15.6-13 & 15]	Revised GE-supplied reactor coolant inventory for: a) Design basis analysis = TS pre-accident spike of 4.0 μCi/gm, and b) Realistic analysis = TS maximum equilibrium value of 0.2 μCi/gm [USAR pg. 15.6-13 & 15]	RG 1.183, Appendix D, Assumption 2
Source Term	Considered Iodine, Krypton, and Xenon isotopes as coolant activity [USAR pg. 15.6-52 & 56]	Considered Iodine, Krypton, Xenon, and Bromine isotopes as coolant activity [USAR pg. 15.6-52 & 56]	Revised GE-supplied reactor coolant isotopes
Iodine Fractions (%)	Organic 0 Elemental 100 Particulate 0 [USAR pg. 15.6-54]	Organic 0.15 Elemental 4.85 Particulate 95.0 [USAR pg. 15.6-54]	RG 1.183, Appendix D, Assumption 4.4

Isolation Valve Closure Time, and mass of coolant released to the environment	Various closure times are listed on current USAR pages, but the resultant mass release (consisting of 14,311 pounds of steam and 127,376 pounds of liquid) listed in the USAR is consistent with the supporting calculation [USAR 15.6-12 & 13]	The supporting calculation's conservative valve closure time of 6.05 seconds will be reflected in the USAR, with its resultant mass release of 14,311 pounds of steam and 127,376 pounds of liquid [USAR 15.6-12 & 13]	Clarification of actual (conservative) valve closure timing in the existing calculation; no change in the amount of reactor coolant released
Atmospheric Dispersion X/Q	EAB LPZ 6.7E-4 8.2E-5 [USAR pg. 15.6-55]	Time EAB LPZ CR 0-2 hrs. 4.3E-4 4.8E-5 3.5E-4 2-8 hrs. -- 4.8E-5 3.5E-4 8-24 hrs. -- 3.3E-5 2.1E-4 1-4 Days -- 1.4E-5 1.1E-4 4-30 Days -- 4.1E-6 5.75E-5 [USAR pg. 2.3-89 & 15.6-55]	Utilized updated X/Q values and added control room
Control Room Dose	Not calculated [Control room doses were not previously required to be determined for MSLBOC]	Calculated [USAR pg. 15.6-53, 55 & 57]	RG 1.183, Section 4.4
Control Room Emergency Recirculation System Filtration (Charcoal & HEPA)	N/A [Control room doses were not previously required to be determined for MSLBOC]	Not credited [USAR pg. 6.5-4 & 30]	No credit taken
AEGT System Filtration (Charcoal and HEPA)	Not credited [Releases are from outside containment, so AEGT system is N/A, therefore it is not credited]	Not credited [USAR pg. 6.5-3, 4, & 42]	No change

2. **On page 7 of the submittal evaluation, FirstEnergy Nuclear Operating Company addressed the impact of the proposed changes on the radiological habitability of the Technical Support Center. However, the application appears to be silent on the impact of those changes on the calculations supporting the establishment of numeric thresholds for emergency action levels (EALs) related to readings on: (1) radiological effluent radiation monitors, and (2) the in-containment high range radiation monitor readings used in the fission product barrier matrix EALs. In addition to the source term change and core inventory changes, the NRC staff notes that you have proposed changes in other analysis assumptions (e.g., filter removal efficiencies, containment leakage rates, drywell flows, containment spray credit, etc.) that could potentially affect the validity of the previous EAL thresholds. Explain the impact of the changes on these EAL thresholds.**

Response: The current PNPP Emergency Plan utilizes the Nuclear Utility Management and Resources Council (NUMARC) EAL methodology, NUMARC/NESP-007, *Methodology for Development of Emergency Action Levels*. When the plant was originally licensed in the mid-1980's, the dose assessment program utilized a source term based on the licensing-basis loss of coolant accident (LOCA) in the Final Safety Analysis Report (FSAR). Subsequently, in October of 1988, the NRC published NUREG-1228, *Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents*. This guidance reflected a more current understanding of accident source terms and their offsite consequences. The NUREG noted on page 1-6 that licensing basis evaluations "normally should not be used to estimate actual accident source terms or offsite doses." The NUREG (beginning on page 2-5) provided core inventory assumptions to be used for source term estimation, expressed in curies per electrical megawatt (Ci/MWe), such that the source term could be adjusted for various plants. The NUREG explained on page 1-2 that "studies of the uncertainties associated with source term estimation indicate that source term projections based on accident conditions are only accurate within a factor of 100 or more..." Therefore, the generic, scalable source term provided by NUREG-1228 is considered to provide an acceptable standard for use at various plants, including PNPP. In the early 1990s at PNPP, when dose assessment software was updated, the source term provided in NUREG-1228, scaled for PNPP's licensed power level at that time, was utilized.

The dose assessment program was updated again in 2012, utilizing the Meteorological Information and Dose Assessment Software (MIDAS) program. As noted in NUREG-1228 on page 1-3, "if a change in assumptions does not result in a change to the source term by at least 1 order of magnitude, it is not worth considering..." Therefore, it was determined that the NUREG-1228 source term, scaled for PNPP's five percent uprate, would be used as input for the dose assessment program (for events postulated to occur with the plant at power). The dose assessment program was utilized to back-calculate plant effluent monitor values that would result in 1 Rem Total Effective Dose Equivalent (TEDE) and/or 5 Rem Child Thyroid doses offsite (the current EAL thresholds).

Similarly, for item (2) in the NRC RAI (the in-containment high-range radiation monitors), the PNPP EALs are based on the NUMARC/NESP-007 guidance. That guidance discusses the determination of in-containment radiation monitor readings for fuel failure events, specifically, for the fuel clad barrier on page 5-20 and the primary containment barrier on page 5-23. Page 5-23 includes a reference to NUREG-1228. Similar to the discussions above for the effluent radiation monitors, the NUREG-1228 conclusion that unless a change in assumptions results "in a change to the source term by at least 1 order of magnitude, it is not worth considering" can also be applied to the in-containment radiation monitor reading thresholds. Since the licensing basis source term is not changing significantly with the adoption of the GNF2 fuel, and a NUREG-1228-based source term based on curies per electrical megawatt would not change, it is concluded that the PNPP in-containment radiation monitor reading values specified in the EALs do not need to be changed as a result of the proposed change to the licensing basis source term for the LOCA event.

In summary, it is concluded that implementation of a new licensing basis source term, such as proposed by this request for licensing action (RLA), does not invalidate the current numeric thresholds in the Emergency Plan EALs, including the (1) radiological effluent radiation monitors, and (2) the in-containment high range radiation monitors. Other proposed changes to licensing basis analysis assumptions, such as those listed in the NRC RAI (filter efficiencies, containment leakage rates, drywell flows, containment spray credit, etc.), also would not invalidate use of the current numeric thresholds in the EALs.

3. **Address whether any non-safety-related systems and components are credited in the accident source term (AST) analyses. If so:**
   a. **Describe the independence (electrical and physical separation) of these systems from the safety-related systems. Provide a detailed discussion on why a fault on the non-Class 1E electrical circuit will not propagate to the Class 1E electrical circuit.**
   b. **Describe the redundancy of these systems and how these systems meet the single failure criterion.**

Response: No non-safety-related systems or components are credited to reduce doses in the alternative AST analyses. Conservative assumptions in several event dose analyses assume the normal control room ventilation system continues to operate for various lengths of time post-accident, which increases the amount of unfiltered inleakage into the control room versus use of an assumption that the systems shut down or otherwise isolate quickly, so this is not considered to be a 'credit.' The Scenario 1 CRDA analysis notes that if the mechanical vacuum pumps are running at a very low plant power level, the pumps will shut down as a result of the Standard Review Plan 15.4.9-required Assumption 1 of a loss of offsite power (LOOP); this has been previously reviewed and approved by the NRC for the CRDA at PNPP.

4. **Address whether any loads are being added to the emergency diesel generators (EDGs). If so, describe their impact on the capability and capacity of the EDGs. Also, describe changes, if any, being made to the EDG loading sequence to support this license amendment request.**

Response: No new loads are being added to the EDGs, and there are no changes being made to the EDG loading sequence in support of this amendment request.

5. **Discuss why there are no changes to the Environmental Qualification profiles as a result of the full implementation of the AST.**

Response: A calculation was performed to determine the impact on equipment environmental qualification due to the post-accident fuel failure radiation dose for GNF2 fuel. The maximum radiation dose rate and the integrated dose (over the course of an accident) following failure of GNF2 fuel was compared to the maximum dose rate and the integrated dose for the current design basis source term. It was determined that the post-accident doses from the GNF2 source term trended closely with the original source term, with the GNF2 fuel post-accident doses being slightly higher. The increases for the maximum dose rates were on the order of 10.5 percent, and the integrated dose increases ranged from zero percent to approximately eight percent. The increases represent the expected change in the post-accident dose profiles as a result of the transition to the GNF2 fuel. Equipment Qualification packages were reviewed to determine if these increases would exceed the existing radiation qualification of equipment. In all cases, the existing qualification doses bound the expected increased doses resulting from the transition to GNF2 fuel. Formal updates to the environmental qualification zone profiles, equipment calculations, and affected auditable file packages, are ongoing to reflect the results of the above review of the GNF2 post-accident doses.

Several necessary changes were identified in two of the calculation summaries included as addendums to the request for licensing action (RLA) submitted on December 6, 2013; specifically Addendum 4, "Summary of Loss of Coolant Accident (LOCA) Dose Calculation," and Addendum 5, "Summary of Control Rod Drop Accident (CRDA) Dose Calculation." Updated summaries of the calculations are included in this attachment, following this discussion (42 pages in the LOCA summary, and 2 pages in the CRDA summary). The issue of these calculations needing to be changed after their submittal has been entered into the FENOC Corrective Action Program.

As a result of the calculation changes, and a re-examination of the proposed Updated Safety Analysis Report (USAR) markups that were included as Addendum 1 of the RLA, updates of the proposed USAR markups are also provided at the end of this attachment, immediately following the CRDA dose calculation update (31 revised or additional USAR pages (the new pages do not include revised USAR text – they are provided only for the purpose of providing information/context)).

The changes to the LOCA calculation include:

1) Correction of the total elemental iodine removal coefficient in the sprayed region of containment to 20.975 $hour^{-1}$,
2) Correction of the equilibrium source term values for Y90, Y91, Y92, Zr95, Zr97, Nb95, and Mo99,
3) Correction of engineered safety feature (ESF) leakage timing so that ESF leakage is assumed throughout the 720 hours of the analysis,
4) Removal of the unnecessary hydrogen mixing system model (sensitivity),
5) Increase of the unfiltered flow rate into the control room during the first 30 minutes to be consistent with the value assumed in the CRDA and main steam line break analyses (10 percent above nominal system intake flow rate),
6) Corrected containment leakage flow rates into the annulus (secondary containment), and
7) Additional changes to improve readability and flow of information.

The change to the CRDA calculation clarifies the description of the iodine releases assumed in the calculation.

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CALCULATION COMPUTATION (BODY OF CALCULATION):	

ATTACHMENTS	
Attachment 1: PNPP ESF.out	25 Pages
Attachment 2: PNPP LOCA.out	338 Pages
Attachment 3: PNPP LOCA TSC.out	42 Pages
Attachment 4: PNPP ESF TSC.out	26 Pages
SUPPORTING DOCUMENTS (*For Records Copy Only*)	

OBJECTIVE OR PURPOSE:

This calculation replaces the current loss-of-coolant accident (LOCA) dose calculation (PSAT 08401T.03, DIN 25) and supports the transition to GNF2 fuel. In addition, certain excess conservatisms contained in the current LOCA dose calculation will be removed to increase the margin of safety. This calculation will be performed in accordance with the guidance provided in Regulatory Guide 1.183 (DIN 7) for application of an alternative radiological source term and will demonstrate that the offsite and onsite post-accident doses comply with the requirements and acceptance criteria of 10 CFR Part 50.67.

SCOPE OF CALCULATION/REVISION

Revision 0

PNPP will be transitioning to GNF2 fuel in future outages. The purpose of this calculation is to prepare a dose analysis supporting this transition and to establish the new design basis LOCA dose analysis using the RADTRAD 3.03 computer program, which was developed for the Nuclear Regulatory Commission (NRC) and is in common use for this type of application in the nuclear industry. This calculation may also be used to support a license amendment request (LAR) associated with the GNF2 fuel transition.

Revision 1

The changes incorporated in Revision 1 are: 1) Correction of the total elemental iodine removal coefficient for the sprayed region to 20.975 hr^{-1}, 2) Correction of the equilibrium core source term for Y90, Y91, Y92, Zr95, Zr97, Nb95, and Mo99, 3) Correction of the ESF leakage timing, and 4) removal of the unnecessary hydrogen mixing system model (sensitivity). Revision 1 also incorporates Addendum A-01 and Post-It-Note P-01.

SUMMARY OF RESULTS/CONCLUSIONS

The post-accident offsite, Control Room, and Technical Support Center doses for a postulated design basis LOCA meet the requirements of 10 CFR Part 50.67. The LOCA dose results, including all leakage pathways, from Table 11-1, are given below:

Location	LOCA Dose (rem TEDE)	Regulatory Limit (DIN 7, 14) (rem TEDE)
Exclusion Area Boundary (EAB)	21.2	25 (0.25 Sv)
Low Population Zone (LPZ)	6.9	25 (0.25 Sv)
Control Room	3.0	5 (0.05 Sv)
Technical Support Center (TSC)	0.5	5 (0.05 Sv)

LIMITATIONS OR RESTRICTIONS ON CALCULATION APPLICABILITY:

This calculation determines the radiological dose consequences resulting from the reactor coolant release that accompanies a postulated design basis LOCA, which are reported in USAR Section 15.6.5. This calculation will become the licensing basis LOCA dose analysis after the transition to GNF2 fuel.

IMPACT ON OUTPUT DOCUMENTS

The results of this calculation will be incorporated into the USAR following LAR approval.

DOCUMENT INDEX

Din No.	Document Number/Title	Revision, Edition, Date	Reference	Input	Output
1	Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 103 to Facility Operating License No. NPF-58	N/A	☒	☐	☐
2	NUREG/CR-6604, RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation	December 1997	☒	☐	☐
3	NUREG-1465, Accident Source Terms for Light-Water Nuclear Power Plants	February 1995	☒	☐	☐
4	Federal Guidance Report 11, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion	Second Printing, 1989	☒	☐	☐
5	Federal Guidance Report 12, External Exposure to Radionuclides in Air, Water, and Soil	September 1993	☒	☐	☐
6	NUREG/CR-5966, A Simplified Model of Aerosol Removal by Containment Sprays	June 1993	☒	☐	☐
7	Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"	July 2000	☒	☐	☐
8	Not Used		☐	☐	☐
9	GEH-KL1WX23P-017, from E. G. Thacker II (GE) to E. S. Tomlinson III (FENOC), PNP GNF2 Fuel Transition: F0802 Source Term Output Files	May 7, 2012	☐	☒	☐
10	DES/98-0845, Telephone and Conference Memorandum by Paul J. Roney/DES, Perry Control Room Atmospheric Dispersion Factors (Chi/Q)	12/2/98	☐	☒	☐
11	PSAT 150.01C.03, Dose Calculation Data Base for Application of the Revised DBA Source Term to the Perry Power Uprate	Revision 2	☐	☒	☐
12	PSAT 04202H.04, Aerosol Decay Rates (Lambda) in Drywell	Revision 0	☐	☒	☐
13	PSAT 04202U.03, Dose Calculation Data Base for Application of the Revised DBA Source Term to the CEI Perry Nuclear Power Plant	Revision 2	☐	☒	☐
14	10 Code of Federal Regulations 50.67, Accident Source Terms	December 23, 1999	☒	☐	☐
15	PNPP Technical Specifications	Amendment 150	☐	☒	☐
16	M26-001, M26, Volume Calculation, Control Room Envelope	Revision 2	☐	☒	☐
17	PSAT 04202H.13, Offsite and Control Room Dose Calculation	Revision 1	☐	☒	☐
18	PERRY USAR	Revision 17	☒	☐	☐

Din No.	Document Number/Title	Revision, Edition, Date	Reference	Input	Output
19	NEI 99-03, Control Room Habitability Assessment Guidance	March 2003	☒	☐	☐
20	Regulatory Guide 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants"	Revision 2, March 1978	☒	☐	☐
21	PNPP Technical Specification Section 5.5.7, PNPP Ventilation Filter Testing Program (VFTP)	Amendment 143	☒	☐	☐
22	NUREG/CR-6604, Supplement 1, RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation	June 8, 1999	☒	☐	☐
23	NUREG/CR-6604, Supplement 2, RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation	October 2002	☒	☐	☐
24	10 Code of Federal Regulations 20 , Standards For Protection Against Radiation	Oct. 1, 2007	☒	☐	☐
25	PSAT 08401T.03, Perry Plant Total Effective Dose Equivalent (TEDE) Calculation	Revision 5	☐	☒	☐
26	PNPP Calculation 3.2.15.17, Containment Water Pool pH Post-Accident	Revision 0	☐	☒	☐
27	GE letter from D. Braden (GE) to E. Root (CEI), GE-PAIP-651, DRF A22-00084-00, dated 3/12/01, "Additional Containment Response Curves"	N/A	☐	☒	☐
28	10 Code of Federal Regulations Part 50 - Domestic Licensing of Production and Utilization Facilities	77 FR 39906, Jul. 6, 2012	☒	☐	☐
29	10 Code of Federal Regulations Part 100 - Reactor Site Criteria	77 FR 39910, Jul. 6, 2012	☒	☐	☐
30	RADTRAD Computer Program Certification, FNOCPP184, CEI-120		☒	☐	☐
31	CEI Calculation 3.2.6.4, Post-LOCA Doses with Spray at 10 min for 6 Hours and Control Room Inleakage of 90 CFM	Revision 0	☒	☐	☐
32	Wolfram Mathworld http://mathworld.wolfram.com/OblateSpheroid.html	accessed 7/20/2012	☒	☐	☐
33	NUREG/CR-0009, Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels	0	☒	☐	☐
34	NUREG-800, Standard Review Plan (SRP) 6.5.2, Containment Spray As A Fission Product Cleanup System	Revision 4, March 2007	☒	☐	☐
35	PNPP Drawing 320-0661-00000, Containment Spray System	Revision T	☐	☒	☐
36	PNPP Drawing D-314-661, Sheet 3, Containment Vessel Spray Ring "A" Piping	Revision B	☐	☒	☐
37	PNPP Drawing D-314-661, Sheet 8, Containment Vessel Spray Ring "B" -	Revision B	☐	☒	☐

Din No.	Document Number/Title	Revision, Edition, Date	Reference	Input	Output
	Piping				
38	PNPP drawing SS-304-661, Sheet 105.2, Piping Isometric – Containment Spray System	Revision C	☐	☒	☐
39	PNPP Drawing D-314-661, Sheet 7, Containment Vessel Spray Ring "D"	Revision B	☐	☒	☐
40	PNPP Drawing SS-304-661, Sheet 102.2, Piping Isometric – Containment Spray System Reactor Building	Revision B	☐	☒	☐
41	PNPP Drawing D-314-661, Sheet 6, Containment Vessel Spray Ring "F"	Revision B	☐	☒	☐
42	PSAT 04202H.08, Steamline: Particulate Decontamination Calculation.	Revision 1	☐	☒	☐
43	PSAT 04212H.02, Drywell Sweep-Out Rate and Related Thermal-Hydraulic Conditions Inside Containment	Revision 1	☒	☐	☐
44	NUREG-0800, Standard Review Plan, 15.6.5, Appendix B, Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage from Engineered Safety Feature Components Outside Containment	Revision 1, July 1981	☒	☐	☐
45	PNPP Drawing 511-0016-00000, Reactor Building – Steel Framing Sections and Details	Revision 0	☐	☒	☐
46	CEI Calculation 3.2.6.3, LOCA Doses as a Function of Spray Initiation Time	Revision 0	☒	☐	☐
47	003.008-001-00, FSAR Figure 3.8-1, "Typical Section of Reactor Building Complex."	Revision 12	☒	☐	☐
48	NUS letter from A. E. Mitchell (NUS) to R. F. Zucker (CEI), CSA-B106187-12, PY-NUS/CEI-1474, dated 3/2/98, "Habitability Chi/Qs for TSC", Attached to PNPP Calc 5.7.1.2, page 4.	N/A	☐	☐	☒
49	PNPP Calculation 5.7.1.2, Technical Support Center – Final Dose	Revision 0	☐	☐	☒
50	PNPP Drawing E-013-011, Final Plant Layout, Section A-A	Revision B	☐	☒	☐
51	PNPP Drawing E-002-002, Final Plant Layout, Section A	Revision 15	☐	☒	☐
52	PNPP Drawing D-912-610, Control Room HVAC and Emergency Recirculation System	Revision FF	☐	☒	☐
53	WASH-1400, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," NUREG 75/014, Nuclear Regulatory Commission, Washington, DC.	1975	☒	☐	☐
54	D.I. Chanin, J.L. Sprung, L.T. Ritchie, and H-N Jow, "Melcor Accident Consequence Code System (MACCS) User's Guide," NUREG/CR-4691, Vol. 1, Sandia National Laboratories, Albuquerque, NM.	1990	☒	☐	☐
55	NUREG/CR-6189, D.A. Powers et al, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments."	July 1996.	☒	☐	☐
56	PERRY SPECIFIC TECHNICAL GUIDELINE	REVISION 20	☒	☐	☐

AW
6.27.14

1.0 PURPOSE

The purpose of this calculation is to prepare a dose analysis supporting the transition to GNF2 fuel and to establish the new design basis loss-of-coolant accident (LOCA) dose analysis. This calculation makes the current LOCA dose calculation (PSAT 08401T.03, DIN 25) and Technical Support Center calculation (Calculation 5.7.1.2, DIN 49) historical and removes certain conservatisms contained in the current LOCA dose calculation to increase the margin of safety. This calculation will be performed in accordance with the guidance provided in Regulatory Guide 1.183 (DIN 7) for application of an alternative accident source term and will demonstrate that the offsite and onsite post-accident doses comply with the requirements and dose limits of 10 Code of Federal Regulations (CFR) Part 50.67 (DIN 14). Onsite doses calculated include the Technical Support Center (TSC) dose.

The evaluation of the limiting design basis loss-of-coolant accident will use the RADTRAD 3.03 Code instead of the proprietary STARDOSE code used in PSAT 08401T.03. RADTRAD 3.03 was developed for the NRC and is commonly used in the nuclear power industry for applications of this type. Excess conservatisms removed from the current calculation (PSAT 08401T.03) are given in detail in the following sections and are summarized below:

- Increased Control Room Emergency Recirculation System (CRERS) charcoal filter efficiency for elemental and organic iodine from 50% to 80%.
- Credit for decay during the two (2) minute onset of the gap release.
- Credit for elemental and aerosol removal in the unsprayed containment region.
- Credit for reduced containment and annulus bypass leakage after 24 hours based on post-accident containment pressure.
- Increased CRERS HEPA filter efficiency from 95% to 99%.

An additional conservatism added to this calculation is the removal of credit for auto-initiation of the CRERS. Isolation of the normal ventilation system and actuation of CRERS is assumed to be performed manually from the control room at 30 minutes post-accident.

2.0 BACKGROUND

The Perry Nuclear Power Plant (PNPP) pilot Alternative Source Term (AST) submittal to the NRC was based on the LOCA analysis presented in PSAT 08401T.03, Revision 5 (DIN 25). This analysis utilized the POLESTAR proprietary computer code STARDOSE to determine the offsite and onsite consequences of a LOCA.

The NRC, in approving the PNPP pilot Alternative Source Term (AST) submittal, performed a confirmatory radiological consequence calculation that evaluated potential fission product release pathways following a postulated LOCA. The NRC calculation was documented in the Perry Safety Evaluation by the Office of Nuclear Reactor Regulation related to Amendment No. 103 (DIN 1). The NRC staff used the RADTRAD Code.

The guidance of Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (DIN 7), will be used to identify the conservatisms currently being applied in the Perry design basis LOCA model. Regulatory Guide (R.G.) 1.183 establishes an acceptable Alternative Source Term (AST) and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST. This calculation will remove some conservatism per the guidance of R.G. 1.183.

3.0 ACCEPTANCE CRITERIA

The post-accident offsite and control room doses must meet the requirements of 10 CFR Part 50.67, "Accident Source Term."

10 CFR 50.67 gives the limits applicable to plants revising their accident source terms. The dose limits specified are given in § 50.67, (b)(2)(i), (ii), and (iii) as follows:

> (b)(2)(i) - An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).
>
> (b)(2)(ii) - An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).
>
> (b)(2)(iii) - Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

For plants implementing the alternative radiological source term methodology, the dose limits of 10 CFR 50.67, given above, replace the limits given in 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance." which are expressed in terms of whole body and thyroid dose as follows:

> (a)(1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.
>
> (a)(2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

As noted above, the dose limit for control room personnel is specified in 10 CFR Part 50.67 (DIN14).

4.0 METHOD OF ANALYSIS

This calculation will evaluate the total effective dose equivalent (TEDE) for the PNPP design basis radiological accident (LOCA) using the revised accident source term based on Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (DIN 7). The TEDE dose is defined as the sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures) (DIN 24). The RADTRAD Code, Version 3.03, will be used to calculate radiological consequences in terms of TEDE. RADTRAD (Radionuclide Transport and Removal and Dose Estimation) calculates fission product transport and removal along with the resulting radiation doses at selected receptors. The code is described in NUREG/CR-6604, "A Simplified Model for Radionuclide Transport and Removal and Dose Estimation" (DIN 2, DIN 22, and DIN 23). RADTRAD 3.03 was certified for this application (DIN 30) in accordance with the ENERCON computer code certification procedure [ENERCON CSP 3.02].

5.0 ASSUMPTIONS

5.1 Control Room Emergency Recirculation System (CRERS)

Upon receipt of an Engineered Safety Feature (ESF) actuation system signal or high radiation signal, the PNPP control room heating, ventilation, and air conditioning (HVAC) system is designed to automatically isolate and activate the CRERS; however, this analysis conservatively assumes that the normal HVAC system continues to operate with an outside air intake (6000 cfm+10% margin) and exhaust to the environment (4800 cfm) until the CRERS is manually actuated at 30 minutes.

Each redundant CRERS subsystem has a high efficiency particulate air (HEPA) filter, charcoal adsorbers and a post HEPA filter. Operation of the CRERS fans, charcoal adsorbers, and HEPA filters are credited in this analysis. The CRERS is an ESF system that is tested in accordance with R.G. 1.52 (DIN 20). The current test acceptance criterion for the CRERS charcoal adsorbers requires a penetration of less than 2.5% (DIN 21). Based on the testing requirements, a charcoal adsorber removal efficiency of 95% could be justified; however, for additional operational margin, elemental and organic iodine removal efficiency is assumed to be 80%. Technical Specification 5.5.7 (DIN 21) states that each HEPA filter is tested to show a penetration and system bypass of less than 0.05% when tested in accordance with Regulatory Guide 1.52 (DIN 20). A penetration and bypass of less than 0.05% allows credit for a particulate removal efficiency of 99% per Regulatory Position C.5.c of Regulatory Guide 1.52. This analysis uses a HEPA filter efficiency of 99 percent for aerosol particulates.

5.2 Hydrogen Mixing System

The hydrogen mixing system is manually initiated. For this analysis, operation of the hydrogen mixing system is not assumed. Due to the minimal (500 cfm) flow rate between the drywell and containment there is little effect on drywell or containment radionuclide concentrations due to operation of this system. Requiring operators to manually initiate the hydrogen mixing system early in the accident is not a good utilization of operator effort because of the minimal impact on accident doses.

5.3 Control Room Inleakage

As described in Section 6.4 of the PNPP USAR (DIN 18), the control room is normally maintained at a slightly positive pressure to the surrounding areas from the 6600 cfm (includes a 10% tolerance on flow rate) fresh air makeup and out leakage of 4800 cfm. In the isolated mode, there is no intake from outside air sources and the control room pressure would eventually reach that of the surrounding areas. After the CRERS is initiated, the maximum control room unfiltered inleakage of 1375 cfm, will be used (DIN 11). The leakage out of the control room envelope is also modeled as 1375 cfm to avoid pressurization of the control room envelope.

5.4 Drywell Flows

Leakage from the drywell into the primary containment is due to steaming from the heated reactor core in accordance with R.G. 1.183, Appendix A, Assumption 3.7. This leakage is assumed during the two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). The termination of the release from the core is due to core recovery and reflood. Instead of evaluating all of the potential steaming rates due to various reflooding scenarios, this analysis will assume that there is a homogenous mixture in the drywell and containment starting at two hours. The assumption of a well-mixed drywell and containment atmosphere at two hours is appropriate because the EAB radiological doses consider the worst two hours as opposed to the first two hours as was done for the previous TID 14844 source term methodology. The assumption of a well-mixed drywell and containment atmosphere is implemented by assuming a high mixing flow (2.77E+05 cfm, approximately one drywell volume per minute) between the two volumes. The mixing flow is conservatively assumed to continue for the duration of the accident instead of isolating the drywell after the core is quenched.

The flow rate from the Drywell to the Wetwell, which bypasses the suppression pool, is given in PSAT 150.01C.03 (DIN 11, page 6) as follows:

Table 5-1

Drywell and Wetwell Mixing Flows

Time After Gap Release (hours)	Flow from DW to WW (cfm)	Flow from WW to DW (cfm)
0 - 0.5	0	0
0.5 – 2.0	3000	0

2.0 – 720	2.77E+05	2.77E+05

5.5 Containment Leakage Rate

The primary containment consists of a drywell, a wetwell and supporting systems to limit fission product leakage during and following the postulated LOCA with rapid isolation of the containment boundary penetrations. The secondary containment will collect and retain fission product leakage from the primary containment and will release fission products to the environment through the Annulus Exhaust Gas Treatment System (AEGTS). During normal operation, the shield building is maintained at a slight negative pressure. Following a DBA, it is expected to remain negative, however for a short period it may not be maintained below the design negative pressure value of 0.25-inch water gauge (USAR 6.5.3.2.1). Therefore, it will be assumed that the primary containment leakage is released directly to the environment for the first 40 seconds following the LOCA (DINs 11 and 25). However, because the gap release does not begin until two minutes post-accident, this 40-second period when there may be direct release to the environment is not considered.

5.6 AEGTS Filtration

The AEGTS includes HEPA filters which are periodically tested to demonstrate compliance with Regulatory Guide 1.52. Particulate removal by the HEPA filters is assumed to be 99% in accordance with Regulatory Guide 1.52 (DIN 20). The system also contains 4-inch deep activated charcoal adsorbers to remove elemental and organic iodine; however, this analysis conservatively assumes a removal efficiency of 0% for the charcoal adsorbers to allow operational flexibility.

The AEGTS extracts and filters a maximum of 2000 cfm from the annulus. During an accident, the maximum expected discharge to the atmosphere is 1000 cfm (DIN 18). The balance of the filtered AEGTS flow is routed back to the annulus. This analysis conservatively assumes that 2000 cfm is discharged directly to the environment with no recirculation (holdup) of iodine in the annulus.

5.7 Containment Spray

Manual initiation of containment spray is assumed at thirty minutes instead of automatic initiation on high pressures and low water level per previous (DIN 25) analysis. Containment spray is assumed to end at 24 hours at which time the radionuclide removal by containment spray is terminated. USAR Subsection 6.5.2.3 gives a discussion of the non-mechanistic assumption that sprays will operate up to 24 hours. In an actual event, spray use would not necessarily be suspended at 24 hours if appropriate conditions for their use still existed. Therefore, the assumption that sprays stop at 24 hours is not intended to be interpreted as a commitment to stop using sprays after 24 hours. In addition, the statement that sprays will operate up to 24 hours implies that the sprays will not necessarily operate continually for 24 hours.

The containment sprays will be run when it is appropriate, and not necessarily the entire time during the first 24 hours of a LOCA. However, this does not invalidate the assumption used in this calculation. The accident guidance to operators is symptom based, rather than event based. Most postulated LOCAs will not result in large radiation releases and would not require containment sprays to run for 24 hours for removal of radioactivity from the containment. Sprays are also used to reduce containment pressure, as needed, by steam condensation and containment heat removal. If a high radiation signal is present from the containment radiation monitor and pressures are elevated in containment, the sprays would be operated. However, if containment gauge pressure is reduced to near zero and use of the sprays is terminated by the operators, this does not have an adverse impact on off-site doses (or the dose calculations) since the driving pressure for containment and main steam line leakage has been eliminated. The dose calculations assume that the maximum allowable leakage (L_a) corresponding to the peak post-accident pressure (P_a) remains during this first 24 hour period, so if containment pressure is reduced to substantially less than P_a, a reduction in leakage and the resultant offsite doses will follow. If containment pressure increases again, and the high radiation signal is present, sprays would be actuated again.

5.8 ECCS Leakage

Consistent with the previous analysis (DIN 25), this analysis assumes that the ECCS leakage is 15 gallons per hour (gph) for the entire duration of the accident. Additionally, leakage from a gross failure of a passive component is assumed to occur at a rate of 50 gallons per minute (gpm) starting 24 hours into the accident and lasting 30 minutes in accordance with NUREG-0800 SRP 15.6.5, Appendix B (DIN 44). Regulatory Guide 1.183, Appendix A, Section 5.2 states that engineered safeguards feature leakage should be assumed to start at the earliest time that recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. For PNPP, ECCS leakage may begin up to 30 minutes post-accident but is assumed to begin at the onset of gap release at two minutes and continue for the duration of the event. This is a conservative assumption which maximizes the dose contribution for this release pathway.

5.9 MSIV Leakage Rate

There are four main steam lines; each line has an inboard MSIV, an outboard MSIV, and a third isolation (shutoff) valve. This analysis assumes a double guillotine pipe rupture in one of the four main steam lines upstream of the inboard MSIV and failure of all four third main steam shutoff valves (1N11-F0020A, B, C, and D) to close as a result of a common power failure (single-failure criterion). A total of a 250 scfh maximum allowable leakage limit (TS SR 3.6.1.3.10) is assumed to occur: (1) 100 scfh through the broken steam line, (2) 100 scfh through a second intact steam line, and (3) the remaining 50 scfh through a third intact steam line. This is modeled as 100 scfh through the broken steam line and 150 scfh through the unbroken steam lines. The calculation converts this to non-standard conditions, as explained in more detail in Section 6.12, "MSIV Flows."

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5.10 Bypass Leakage

Secondary containment bypass leakage is ~~in addition to~~ A PORTION OF the containment allowable leakage, L_a. The leakage paths include all pathways which could potentially allow leakage to bypass secondary containment. Therefore, any bypass leakage releases would not be treated by an ESF filtration system prior to being released to the environment. The secondary containment bypass leakage is currently limited to 5.04% of L_a, when pressurized to $\geq P_a$, by Technical Specification SR 3.6.1.3.9 (DIN 15) even though the previous LOCA dose calculation (DIN 25) assumed a leakage of 10.08% of L_a. The containment bypass leakage will be maintained at 0.1008 L_a in this analysis to allow for an increase in the Technical Specification allowable leakage limit.

5.11 Source Term Release

In accordance with R.G. 1.183 (DIN 7), only the gap and in-vessel release phases are considered in this design basis LOCA dose calculation. The core source terms are assumed to be released at a constant rate such that the release is completed by the end of the specified release period. Assumptions regarding release fractions and timing are consistent with Tables 1 and 4 of R.G. 1.183 (DIN 7).

Table 1 of R.G. 1.183 is given below:

BWR Core Inventory Fraction Released Into Containment

Group	Gap Release Phase	Early In-vessel Phase	Total
Noble Gases	0.05	0.95	1.0
Halogens	0.05	0.25	0.3
Alkali Metals	0.05	0.20	0.25
Tellurium Metals	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Noble Metals	0.00	0.0025	0.0025
Cerium Group	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

Table 4 of R.G. 1.183 is reproduced below:

LOCA Release Phases

	PWRs		BWRs	
Phase	**Onset**	**Duration**	**Onset**	**Duration**
Gap Release	30 sec	0.5 hr	2 min	0.5 hr
Early In-Vessel	0.5 hr	1.3 hr	0.5 hr	1.5 hr

6.0 Design Input

6.1 Plant Grade

The PNPP plant grade elevation is 620 feet (DIN 50).

6.2 Core Source Terms and Releases

The PNPP core source term release magnitude, timing and chemical form are based on Regulatory Guide 1.183 (DIN 7). The core source terms were developed by GE Hitachi (DIN 9). The calculated inventories are based on 2-year GNF2 refueling cycles and serve as input for design basis accident analyses based on Regulatory Guide 1.183 source term assumptions. The fission product inventory calculations were performed using the ORIGEN2 code. The Ci/MW multipliers developed in DIN 9 are applied here to generate the core source terms at the onset of the event. A reactor power level of 3833 MWt will be used based on 102% of the rated thermal power, 3758 MWt, as defined in Technical Specification 1.1, Definitions, page 1.0-5, Amendment 112 (DIN 15).

The GNF2 fuel source terms are based on the GNF2 equilibrium source activity given below. The source terms include fission products, actinides, and activation products. The listing of isotopes given in Table 6-1, below, is based on the isotopes used in the RADTRAD computer code. As stated in the RADTRAD User's Manual, NUREG/CR-6604 (DIN 2), the 60 isotope nuclide file is based on isotopes selected in WASH-1400 [DIN 53] with the addition of 6 isotopes used in the MACCS code [DIN 54].

Table 6-1
Source Term

Isotope	GNF2 Equilibrium (Ci/MWth)
Co58	2.647E+02
Co60	4.827E+02
Kr85	3.789E+02
Kr85m	6.737E+03
Kr87	1.283E+04
Kr88	1.804E+04
Rb86	6.882E+01
Sr89	2.425E+04
Sr90	3.016E+03
Sr91	3.064E+04
Sr92	3.346E+04
Y90	3.118E+03
Y91	3.152E+04
Y92	3.362E+04
Y93	3.928E+04
Zr95	4.440E+04

Isotope	GNF2 Equilibrium (Ci/MWth)
Zr97	4.490E+04
Nb95	4.463E+04
Mo99	5.105E+04
Tc99m	4.470E+04
Ru103	4.309E+04
Ru105	3.046E+04
Ru106	1.750E+04
Rh105	2.871E+04
Sb127	3.016E+03
Sb129	8.906E+03
Te127	2.997E+03
Te127m	4.049E+02
Te129	8.762E+03
Te129m	1.304E+03
Te131m	3.965E+03
Te132	3.850E+04
I131	2.714E+04
I132	3.914E+04
I133	5.495E+04
I134	6.025E+04
I135	5.150E+04
Xe133	5.302E+04
Xe135	1.934E+04
Cs134	6.926E+03
Cs136	2.162E+03
Cs137	4.190E+03
Ba139	4.877E+04
Ba140	4.709E+04
La140	5.002E+04
La141	4.440E+04
La142	4.278E+04
Ce141	4.460E+04
Ce143	4.090E+04
Ce144	3.670E+04
Pr143	3.957E+04
Nd147	1.795E+04
Np239	5.619E+05
Pu238	1.338E+02
Pu239	1.291E+01
Pu240	1.749E+01

Isotope	GNF2 Equilibrium (Ci/MWth)
Pu241	5.748E+03
Am241	7.237E+00
Cm242	1.799E+03
Cm244	1.124E+02

6.2.1 Onset of Gap Release

Table 4 of Regulatory Guide 1.183 tabulates values acceptable to the NRC for the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset of the gap release is the time following the initiation of the accident (i.e., time = 0) prior to the start of the gap release. For a BWR the onset is 2 minutes. Credit will be taken for decay prior to the onset of the gap release at 2 minutes.

6.2.2 Release fractions

The release fractions used in this analysis are consistent with Table 1 of R.G. 1.183 (DIN 7) which is reproduced in Section 5.11 of this calculation.

6.3 Suppression Pool Iodine Re-evolution

The impact of any postulated iodine re-evolution from the suppression pool has been evaluated and shown to be negligible based on the pool pH level. If the pH is maintained above 7, very little (less than 1%) of the dissolved iodine will be converted to elemental iodine (DIN 1, DIN 7). The Standby Liquid Control System (SLCS) is used for controlling and maintaining long-term suppression pool water pH levels to 7 or above. The pH of post-accident water in the containment will remain above 7 for the entire duration of the postulated LOCA (DIN 26). As such, this analysis will not consider any impact to the offsite or control room doses due to iodine re-evolution from the suppression pool. Also, in accordance with Appendix A of Regulatory Guide 1.183 (DIN 7), because the suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment can be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide.

6.4 Dose Conversion Factors

The effective dose conversion factors for the TEDE calculations are based on FGR 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (DIN 4) and FGR 12, "External Exposure to Radionuclides in Air, Water, and Soil" (DIN 5). These reports tabulate dose coefficients for external exposure to photons and electrons emitted by radionuclides distributed in air, water, and soil, as well as, dose coefficients for the committed dose equivalent to tissues of the body per unit activity of inhaled or ingested radionuclides. These dose coefficients for exposure to radiation are intended for the use in calculating the dose equivalent to organs and tissues of the body and are endorsed by the NRC in Regulatory Guide 1.183, Sections 4.1.2 (FGR

11) and 4.1.4 (FGR 12). Dose conversion factors for the 60-isotope, 9 element NUREG 1465 (DIN 3) accident source term composition are included in the RADTRAD input.

6.5 Atmospheric Dispersion Factors

The atmospheric dispersion factors (χ/Q values) for the LPZ and EAB are obtained from PSAT 04202U.03 (DIN 13, page 10) and Calculation 3.2.6.3 (DIN 46). The Control Room atmospheric dispersion factors are documented in DES/98-845 (DIN 10). The Technical Support Center atmospheric dispersion factors are documented in PY-NUS/CEI-1474 (DIN 48) and Revision 0 of the TSC Dose Evaluation (DIN 49). The χ/Q values, based on a ground level release, are given below:

Table 6-3
χ/Q (sec/m^3)

	Location	
Time Interval	EAB	LPZ
0 to 2 hrs	4.3E-4	4.8E-5
2 to 8 hrs		4.8E-5
8 to 24 hrs		3.3E-5
24 to 96 hrs		1.4E-5
96 to 720 hrs		4.1E-6

Table 6-4
Control Room and TSC χ/Q (sec/m^3)

Time Interval	CONTROL ROOM	TSC
0 to 8 hrs	3.5E-4	5.1E-5
8 to 24 hrs	2.1E-4	4.1E-5
24 to 96 hrs	1.1E-4	3.1E-5
96 to 720 hrs	5.75E-5	1.1E-5

6.6 Breathing Rate and Occupancy Factors

The breathing rates applied in the calculation of the inhalation dose are consistent with those reported in Sections 4.1.3 and 4.2.6 of R.G. 1.183 (DIN 7).

Table 6-5
Breathing Rates (m^3/s)

Time Period	EAB	LPZ	Control Room/TSC
0 to 8 hours	3.5E-4	3.5E-4	3.5E-4
8 to 24 hours	1.8E-4	1.8E-4	3.5E-4
1 to 30 days	2.3E-4	2.3E-4	3.5E-4

The control room and TSC occupancy factors are consistent with those reported in Section 4.2.6 of R.G. 1.183 and are tabulated below.

Table 6-6
Control Room and TSC Occupancy Factors

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

6.7 Containment Volumes

The volumes of the containment regions are from CEI Calculation 3.2.6.4, Revision 0, Page 3A of 33 (DIN 31).

Table 6-7
Containment Volumes

Region	Volume (ft^3)
Unsprayed Containment	684,226
Sprayed Containment	481,174
Drywell	276,500

Note: the above volumes are shown as rounded values in the RADTRAD screen views but the actual values are used in the RADTRAD input file.

6.8 Technical Support Center Doses

Doses to personnel in the Technical Support Center (TSC) are calculated in the same manner as the doses to the Control Room operators except for the TSC specific atmospheric dispersion, χ/Q, values and TSC data. The additional data needed to determine the TSC doses is as follows (DIN 49, page 9 and 10, and DI 5.7.1 page 18):

Table 6-8
TSC Data

Parameter	Value
TSC Volume (ft^3)	113,412
HVAC Flow Rate (cfm)	37,000
Recirculation Filter Flow Rate (cfm)	6,000
Charcoal Filter Bed Depth (in)	2
Filtered Damper Inleakage (cfm)*	12
Unfiltered Inleakage (cfm)**	27.2

*Added to unfiltered inleakage

** After recirculation is initiated at 60 minutes (includes 10 cfm for ingress and egress)

For the TSC charcoal removal efficiency, a removal efficiency of 80% will be used to provide margin as was done for the Control Room charcoal adsorber removal efficiency. Note that in the RADTRAD files, the TSC is labeled as the Control Room. The normal HVAC flow is assumed to operate for the first 60 minutes after which it is isolated and the recirculation filter is initiated.

6.9 Mixing Between the Unsprayed Containment and Sprayed Containment

The mixing rate between the unsprayed containment and the sprayed containment was determined to be 71,400 cfm in Calculation PSAT 04202U.03, Rev. 0 (DIN 13, page 6).

6.10 Containment Leakrate

The maximum allowable primary containment leakrate, L_a, is 0.2 volume percent per day at the peak containment pressure (P_a) of 7.80 psig per Technical Specification 5.5.12 (DIN 15). Per Assumption 5.5, primary containment leakage is released directly to the environment for the first 40 seconds following the LOCA, when the shield building may not be at a negative pressure. This potential 40 second release directly to the environment is not modeled because it has no dose significance. Following this 40-second period, the annulus will collect and retain any fission product leakage from the primary containment and will release fission products to the environment through the AEGTS. Secondary Containment Bypass leakage is a portion of the total containment leakage, L_a. Technical Specification SR 3.6.1.3.9 (DIN 15) limits the secondary containment bypass leakage to equal to or less than 5.04 percent of the primary containment leak rate. The containment bypass leakage for this calculation is assumed to be 0.1008 L_a. (Assumption 5.10)

Therefore, the leakrate for the sprayed and unsprayed (including drywell) regions of the containment is calculated below:

$$Leakrate\ from\ Sprayed\ Region = \frac{[(4.812e+05\ ft^3) * 0.2\%]}{24\ hr * 60\frac{min}{hr}}$$

$$Leakrate\ from\ Sprayed\ Region = 0.668\ cfm$$

Subtracting the bypass leakage of 0.1008*L_a (i.e., 0.067 cfm) gives:

$$Leakrate\ from\ Sprayed\ to\ Annulus = 0.668 - 0.067\ cfm = 0.601\ cfm$$

$$Leakrate\ from\ Unsprayed\ Regions = \frac{[(6.842e+05\ ft^3 + 2.765e+05\ ft^3) * 0.2\%]}{24\ hr * 60\frac{min}{hr}}$$

$$Leakrate\ from\ Unsprayed\ Regions = 1.334\ cfm$$

Subtracting the bypass leakage of 0.1008*L_a (i.e., 0.134 cfm) gives:

$$Leakrate\ from\ Unsprayed\ \text{Regions to}\ Annulus = 1.334 - 0.134\ cfm = 1.2\ cfm$$

<u>Bypass Leakage:</u>

$$Sprayed\ Region\ Bypass\ Leakage = 0.668\ cfm * 0.1008 = 0.067\ cfm$$

$$Usprayed\ Regions\ Bypass\ Leakage = 1.334\ cfm * 0.1008 = 0.134\ cfm$$

$$Total\ Bypass\ Leakage = 0.202\ cfm$$

Table 6-9
Containment Leakrate Summary

Leakage	Leakrate (ft^3/min) 40 sec. to 24 hrs
From Sprayed Region to Annulus	0.601
From Unsprayed Regions to Annulus	1.2
Bypass Sprayed Region to Environment	0.067
Bypass Unsprayed Regions to Environment	0.135 (rounded up)

6.11 Leakage after 24 Hours

Containment leakage depends upon containment pressure and will be reduced at 24 hours as allowed by Regulatory Guide 1.183, Appendix A, Section 3.7. Based on the post-accident containment pressure curve for a MSLB (DIN 27), the containment pressure at 24 hours post-accident is 18.1 psia. This value was obtained by digitizing the containment pressure curve and finding the pressure at 24 hours. Atmospheric pressure at the elevation of the PNPP site (620 ft AMSL) is 14.37 psia. Because the flow rate is proportional to the square root of the differential pressure, the reduction in flow rate may be estimated as follows (assuming all other parameters remain constant):

$$\frac{L_{24hr}}{L_a} \propto \frac{\sqrt{P_{24hr}}}{\sqrt{P_a}} = \frac{\sqrt{18.1-14.37}}{\sqrt{7.8}} = 0.691 \Rightarrow 0.69$$

Because the secondary containment bypass leakage also depends upon containment pressure, this leakage rate will also be reduced to 69% of the pre- 24 hour values beginning 24 hours post-accident. Therefore, this analysis will reduce all leakage flows to 69% of the ~~pef~~ PRE -24 hour value after 24 hours. AW 6-26-14

Table 6-10

Containment Leakrate after 24 hours

Leakage	Leakrate (ft^3/min)
From Sprayed Region to Annulus	0.415
From Unsprayed Regions to Annulus	0.828
Bypass Sprayed Region to Environment	0.0462
Bypass Unsprayed Regions to Environment	0.0932

6.12 MSIV Flows

The flows given in Section 5.9 are based on MSIV leakage rate testing requirements at standard conditions. The drywell atmosphere will not be at standard conditions after the reactor blowdown. Calculation PSAT 04202H-04 (DIN 12, page A3) determined that the total MSIV flow rate from 0 to 7484 seconds was 298 cfh based on the minimum post-accident drywell pressure of 15.7 psia and minimum temperature of 215°F. From 7484 to 86400 seconds, the MSIV flow rate is 247 cfh based on the minimum pressure of 15.7 psia and temperature of 100°F.

Based on the assumed flow split given in Section 5.9, the flow through the broken steam line is:

$$Q_{broken\ line}(t \leq 7484\ seconds) = \frac{298\ cfh}{60\ {min}/{hr}} * \frac{100\ scfh}{250\ scfh} = 1.987\ cfm$$

$$Q_{broken\ line}(7484 < t \leq 86400\ seconds) = \frac{247\ cfh}{60\ {min}/{hr}} * \frac{100\ scfh}{250\ scfh} = 1.647\ cfm$$

The flow through the intact steam lines (100 scfh through a second intact steam line and the remaining 50 scfh through a third intact steam line) is given below.

$$Q_{intact\ lines}(t \leq 7484\ seconds) = \frac{298\ cfh}{60\ {min}/{hr}} * \frac{150\ scfh}{250\ scfh} = 2.98\ cfm$$

$$Q_{intact\ lines}(7484 < t \leq 86400\ seconds) = \frac{247\ cfh}{60\ {min}/{hr}} * \frac{150\ scfh}{250\ scfh} = 2.47\ cfm$$

Consistent with Section 6.2 of Appendix A to R.G. 1.183, this leak rate may be reduced by as much as a factor of 2 after 24 hours, if supported by plant specific analysis. As calculated above, this analysis will reduce all initial leakage flows to 69% of the pre-24 hour leak rate after 24 hrs. This leakrate is conservatively based on the initial flows (i.e., t<7484 seconds). The MSIV leakage flows at this time become:

Table 6-11
Steam Line Leakrate after 24 Hours

Leakage Path	Leakrate (ft^3/min)
Broken Steam Line	1.371
Intact Steam Lines	2.056

The main steam line leakage rate is required to be less than or equal to 100 scfh (Technical Specification SR 3.6.1.3.10) when tested at P_a (7.80 psig). The test temperature is assumed to be 70°F and the RCS operating temperature is 552°F (saturated temperature at steam dome pressure of 1045 psig, LCO 3.4.12). Converting the 100 scfh leakage rate to operating temperature gives a leakage rate of 191 cfh. The MSIV flow after the first piping segment from the drywell will be based on a constant maximum steam line flow of 191 cfh (3.183 cfm) (DIN 11, page 8) for the broken steam line and 191 cfh*150 scfh/100 scfh = 4.775 cfm for the unbroken lines. No reduction in flow at 24 hours for these line segments will be taken. This conservative model is unchanged from the previous analysis (DIN 25).

6.13 Radionuclide Removal Mechanisms

Removal mechanisms for elemental iodine and aerosols will be applied in this calculation using NRC correlations incorporated into the RADTRAD 3.03 code.

6.13.1 Removal by Deposition

<u>Elemental Iodine Removal by Deposition</u>

Elemental iodine removal is credited in the drywell and containment volumes. Airborne elemental iodine is removed by deposition to the walls in the drywell and containment. As reported in Section 5.1.2 of NUREG/CR-0009, DIN 33), this process is driven by the temperature differences between the surfaces and the atmosphere. The removal factor reported in NUREG/CR-0009 is given by the following equation.

$$\lambda = \frac{K_g A}{V}$$

where:

λ = removal rate constant due to surface deposition,

k_g = average mass transfer coefficient 0.137 cm/s (16.18 ft/hr) from page 17 of NUREG/CR-0009,

A = surface area for wall deposition, and
V = volume of contained gas.

This formula is also reported in Standard Review Plan 6.5.2 (DIN 34) as a method of calculating the total elemental iodine removal capability. These removal constants are applied until a decontamination factor (DF) of 200 has been obtained.

Volume and Area Calculations

Drywell

Volume

For all volume calculations, surfaces other than the inner and outer building wall will be conservatively neglected. The PNPP drywell volume of 276,500 ft^3 from CEI Calculation 3.2.6.4, Revision 0, Page 3A of 33 (DIN 31) is used in this calculation.

Wall Surface Area

Considering the 36'6" inside radius (DIN 45) of the drywell cylinder and the approximately 66 foot height above the suppression pool high water level (DIN 51), the area of the inner drywell wall is calculated to be **15,000 ft^2**. The use of the suppression pool high water level is conservative because it minimizes the wall surface area available for deposition.

$$Area = \pi Dh = \pi * 73' * 66' = 15136.2\, ft^2\ or, 15{,}000\, ft^2$$

Sprayed Containment Region

Volume

Although in some parts of the containment, the containment spray would fall directly to the suppression pool, the refueling floor (grating) at El. 689'-6" would affect a large fraction of the containment spray. As such, the only containment volume credited with spray removal is that area above the refueling floor. The upper containment (sprayed) region volume of 481,174 ft^3 from CEI Calculation 3.2.6.4, Revision 0, Page 3A of 33 (DIN 31) is used in this calculation.

Wall Surface Area

The surface area is taken as the containment wall area above the refueling floor at 689'-6" (DIN 47) and below the containment spring line at 727' (DIN 47). Using the containment radius of 60' (DIN 45) the surface area is calculated below as 14,137 ft^2.

$$Area = \pi Dh = \pi * 2 * 60' * (727' - 689.5') = 14{,}137\, ft^2$$

The surface area of the oblate elliptical spheroid above the spring line is given by: (DIN 32)

$$S = 2\pi a^2 + \pi\frac{c^2}{e}\, ln\left(\frac{1+e}{1-e}\right)$$

Where "a" is the equatorial radius and "c" is the polar radius and the ellipticity, "e" is given by:

$$e = \sqrt{1-\frac{c^2}{a^2}}$$

Substituting 60' for "a" and (757' – 727' = 30') for "c" (DIN 47) gives:

$$e = \sqrt{1-\frac{(30')^2}{(60')^2}} = 0.866$$

$$S = 2\pi * (60')^2 + \pi\frac{(30')^2}{0.866}\, ln\left(\frac{1+0.866}{1-0.866}\right) = 31{,}218\, ft^2$$

The surface area of the dome is half of this total area: S = 31,218 ft^2/2 = 15,609 ft^2

The total area available for plateout is therefore 29,746 ft^2 or, 29,000 ft^2.

Unsprayed Containment Region

Volume

The volume of the unsprayed containment region is 684,226 ft^3 per from CEI Calculation 3.2.6.4, Revision 0, Page 3A of 33 (DIN 31).

Wall Surface Area

Considering the 41'6" outside radius of the drywell (DIN 45) and the approximately 96 foot height (689'-6" – 593'-4" = 96.17') above the suppression pool high water level (DIN 51), the area of the outside drywell wall is calculated to be 25,000 ft^2. The use of the suppression pool high water level is conservative because it minimizes the wall surface area available for deposition.

$$Area = \pi Dh = \pi * 2 * 41.5' * 96' = 25{,}032\, ft^2\ or, 25{,}000\, ft^2$$

The radius of the unsprayed containment wall is 60' giving a surface area of

$$Area = \pi Dh = \pi * 2 * 60' * 96' = 36{,}191\, ft^2\ or, \quad 36{,}000\, ft^2$$

This gives a total surface area of 61,000 ft^3

Using the above wall areas and volumes, the removal rate constants are given below:

Table 6-12

Elemental Iodine Deposition Removal Factors

Node	Volume (ft^3)	Wall Area (ft^2)	Removal Factor (hr^{-1})
Drywell	276,500	15,000	0.878
Sprayed Containment	481,174	29,000	0.975
Unsprayed Containment	684,226	61,000	1.443

Airborne elemental iodine removal by deposition to the walls in the drywell and containment is assumed to end when a DF of 200 is reached.

Aerosol (particulate) Removal by Deposition

Regulatory Guide 1.183, Appendix A, Section 3.2 (DIN 7), discusses the reduction in airborne radioactivity in the containment by natural deposition. This section references the model in NUREG/CR-6189 (DIN 55) as an acceptable model. This model (the "Powers" model) is incorporated into the RADTRAD code. Aerosol removal in the drywell and unsprayed containment region is based on the 10% Powers Aerosol model in RADTRAD. Note that, for unsprayed regions, the reactor and accident type used in the Powers aerosol model must be reset to "BWR-Design Basis Accident" prior to each execution of the RADTRAD code.

6.13.2 Removal by Sprays

Aerosol Removal by Sprays

A simplified model for estimating the fission product aerosol removal by containment sprays following a postulated LOCA is used. The model for aerosol removal by sprays built into the RADTRAD 3.03 code is the Powers model. The model was developed using values of 10, 100, and 2500 cm^3 H_2O/ cm^2-s for the spray water flux. The model should not be used for spray water fluxes and fall heights outside of these ranges (DIN 2, 22, and 23). The Powers model was derived by correlating the results of Monte Carlo uncertainty sampling analyses assessing the uncertainties in aerosol properties, aerosol behavior, spray droplet behavior, and the initial and boundary conditions expected to be associated with a postulated LOCA in the containment. The Powers mechanistic model requires that the user specify the following:

1. *Q*, the spray water flux, in cfm/sq ft;
2. *H*, the fall height, in meters;
3. *ALPHA*, the ratio of unsprayed volume to sprayed volume,
4. *PCT*, the uncertainty percentile selected for the model (10th, 50th, 90th percentiles).

The spray alpha is 6.8423E+05/4.8117E+05 = 1.422. The other two parameters used in this evaluation that are not treated as uncertainty distributions for Perry are (1) spray water flux, and (2) mean spray fall height. These parameters are specified based on plant specific design information. The "best estimate" value is associated with the 50th percentile, or median values; the lower bound is associated with the 10th percentile; and the reasonable upper bound, or largest decontamination factor (DF), with the 90th percentile. For aerosol removal by containment spray, the RADTRAD Powers Model 10th percentile uncertainty distribution for fission product in aerosol form is used in this analysis. Note that the reactor and accident type used in the Powers aerosol model must be reset to "BWR-Design Basis Accident" prior to each execution of the RADTRAD code.

The PNPP LOCA dose analysis credits spray removal of aerosols in the sprayed region of the containment. The Powers spray removal model implemented in RADTRAD requires the spray flux and spray height as inputs. The spray flow is 5250 gpm (D-302-0661-00000, Rev. G, DIN 35) per train. Technical Specification 3.6.1.7 requires that the spray flow from the RHR system be ≥5250 gpm. Because the Powers model spray removal is a direct function of spray flow rate, a lower bound of the spray flow (i.e., 5250 gpm) is conservative.

$$Spray\ Flux = \frac{Spray\ Flow}{Sprayed\ Area} = \frac{5250\ gpm * 0.1337\ cfm/gpm}{\pi * (60')^2} = 0.06206\ {}^{cfm}/_{sq\ ft}$$

The average droplet fall height is dependent on the available train of containment spray. As shown below, the headers for the "A" Train are located above the headers for the "B" Train per drawing D-320-0661-00000, Rev. G. If the flow rate through all nozzles is assumed to be equal, the average drop height can be calculated by the nozzle-weighted average of the drop heights. The average drop height is used because the train operating post-accident is unknown. The drop height is based on the distance above the operating floor at El. 689'-6" (DIN 51).

Table 6-13
PNPP Containment Spray Heights

RHR Train	Header Designation	Header Elevation (ft)	Reference Drawing	Height - H_i (ft)	Number of Nozzles[7] - N_i	Ni*Hi
A	A	735.250	D-314-661, Sheet 3, Rev. B, (DIN 36)	45.75	129	5901.75
	C	744.250	SS-304-661, Sheet 105.2, Rev. C, (DIN 38)	54.75	113	6186.75
	E	750.500	SS-304-661, Sheet 102.2, Rev. B, (DIN 40)	61.00	102	6222
B	B	737.000	D-314-661, Sheet 8, Rev. B, (DIN 37)	47.50	129	6127.5
	D	745.750	D-314-661, Sheet 7, Rev. B, (DIN 39)	56.25	113	6356.25
	F	752.000	D-314-661, Sheet 6, Rev. B, (DIN No. 41)	62.50	104	6500
				Total	690	37294.25
				Average (ft)	54.05	

[7]320-0661-00000, Rev. G (DIN 35)

$$H_{avg} = \frac{\sum_i H_i N_i}{\sum_i N_i}$$

where:

N_i is the number of nozzles on header i

H_i is the height of header i above the operating floor (ft)

The average fall height for both trains combined is therefore **54.05 ft**.

As discussed in SRP 6.5.2 (DIN 34), because the removal of particulate material depends markedly upon the relative sizes of the particles and the spray drops, the aerosol spray removal lambda is assumed to decrease by a factor of 10 after the aerosol mass has been depleted by a factor of 50.

<u>**Elemental Iodine Removal by Sprays**</u>

SRP 6.5.2 provides guidance on calculating the spray lambda for removal of elemental iodine. The following formula is valid for lambdas greater than 10 per hour with a maximum of 20 per hour to prevent extrapolation beyond the existing data.

$$\lambda_s = \frac{6 * k_g * T * F}{V * D}$$

where:

λ_s = first-order removal coefficient by spray,
k_g = the gas-phase mass-transfer coefficient,
T = the fall time of the drops, which may be estimated by the ratio of the average fall height to the terminal velocity of the mass-mean drop,
F = volume flow rate of the spray pump,
V = containment building net free volume, and
D = mass-mean diameter of the spray drops.

Gas Phase Mass Transfer Coefficient

The gas-phase mass-transfer coefficient, k_g, can be determined by back-calculation from a solved case with slightly different assumptions. Specifically, the example on Page 106 of NUREG/CR-0009, "Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels", 1978, (DIN 33) uses the stagnant film model to determine the spray removal coefficient for a PWR case with a 1713 spray nozzle and the following parameters.

λ_s = 14.2 hr^{-1}
F = 1500 gpm
V = 1.75E6 ft^3
Height = 90 ft
Temp = 250°F

Solving the above equation for k_g, gives:

$$k_g = \frac{\lambda_s * V * D}{6 * T * F}$$

To calculate k_g, the values of the mass-mean drop diameter, D, and the fall time of the drops, T, are needed. The PNPP spray nozzles are Spraco 1713A nozzles (DIN 35). Recent test results with the Spraco 1713A nozzles presented in Figure 4 of NUREG/CR-5966 (DIN 6) have a mean droplet size of 234 µm (NUREG/CR-5966, page 7) and an upper diameter of about 1500 µm. The mass-weighted average drop size, however, will be larger than 234 microns since the larger drops have exponentially more mass. This volume-weighted size distribution (which is directly related to the mass-weighted distribution) is reported in Figure 7 of NUREG/CR-5966 which illustrates an average of the volume

weighted distribution to be approximately 1200 microns. A value of 1200 microns will be applied in this calculation. The terminal velocity of 1200 μm drops can be found to be approximately 400 cm/s from Figure 16 of NUREG/CR-5966. Conservatively assuming the velocity is equal to the terminal velocity, a 90 foot (2743 cm) fall height gives a fall time of 6.86 seconds. Using the above data to determine the gas-phase mass-transfer coefficient, k_g, gives:

$$k_g = \frac{\lambda_s * V * D}{6 * T * F} = \frac{14.2\ \text{hr}^{-1} \frac{\text{hr}}{60\ \text{min}} * 1.75\text{E}6\ \text{ft}^3 * 1200 * 10^{-6}\ \text{m} * 100\ {}^{\text{cm}}/_{\text{m}}}{6 * \frac{90\ \text{ft}}{400\ {}^{\text{cm}}/_{\text{sec}}} * 30.48 \frac{\text{cm}}{\text{ft}} * 1500\ \text{gpm} * 0.1337 \frac{\text{ft}^3}{\text{gal}}} = 6\ \frac{\text{cm}}{\text{sec}}$$

For PNPP, the average fall height of the spray drops is calculated to be 54.05 ft (1647 cm). The terminal velocity of 1200 μm drops can be found to be approximately 400 cm/s from Figure 16 of NUREG/CR-5966. The drop fall time is calculated to be 4.1 seconds. The spray flow is 5250 gpm from D-302-0661-00000, Rev. G, (DIN 35) and the sprayed volume of the containment is 481,174 ft^3 from CEI Calculation 3.2.6.4, Revision 0, page 3A of 33 (DIN 31). From the SRP equation, below, the PNPP spray lambda for elemental iodine can be calculated to be 107.66 hr.

$$\lambda_s = \frac{6 * k_g * T * F}{V * D} = \frac{6 * 6 \frac{\text{cm}}{\text{sec}} * 4.1\ \text{sec} * 5250\ \text{gpm} * 0.1337 \frac{\text{ft}^3}{\text{gal}} * 60 \frac{\text{min}}{\text{hr}}}{481{,}174\ \text{ft}^3 * 1200 * 10^{-6}\ \text{m} * 100\ {}^{\text{cm}}/_{\text{m}}} = 107.66\ \text{hr}^{-1}$$

This result is reasonable considering the 14.2 hr^{-1} value calculated for the PWR case described in NUREG/CR-0009, the much higher spray flow rate at PNPP, and the smaller sprayed volume at PNPP. Since the SRP allows a maximum lambda of 20 hr^{-1}, this calculation will apply a spray removal lambda of 20 hr^{-1} for elemental iodine. As discussed previously, elemental iodine is removed by deposition to the walls in the containment with a removal coefficient of 0.975 hr^{-1} for the sprayed region which gives a total elemental iodine removal coefficient for the sprayed region of containment as 20.975 hr^{-1}. As discussed in SRP 6.5.2, the maximum decontamination factor is 200 for elemental iodine. The effectiveness of the spray in removing elemental iodine will be presumed to end at that time, post-LOCA, when the maximum elemental iodine DF is reached.

6.14 Annulus Model

The Annulus Exhaust Gas Treatment System (AEGTS) is an engineered safety features system designed to collect, process, and release the fission product leakage from the primary containment into the shield building. The system is operated continuously during normal operation and maintains a slight negative pressure in the shield building. The AEGTS is a redundant system consisting of pre-HEPA filters, charcoal adsorbers and post-HEPA filters. Reduction in release activity by ESF ventilation filtration systems may be credited where applicable if filter systems used in these applications are evaluated against the guidance of Regulatory Guide 1.52 (DIN 20). The AEGTS charcoal adsorbers are not credited for reducing the released activity, so testing in accordance with R.G. 1.52 is not necessary.

The AEGTS HEPA filter is tested in accordance with Regulatory Guide 1.52 to verify a penetration and system bypass of less than 0.05% (DIN 21). Aerosol removal by the HEPA filters is therefore assumed to be 99%. As discussed previously, no credit for charcoal filtration of the annulus exhaust is taken in this calculation.

6.15 Deposition in Main Steam Lines

The deposition in the main steam lines will use the aerosol removal efficiencies from PSAT 08401T.03 (DIN 25) which was based on PSAT 04202H.08 (DIN 42). These removal efficiencies include a 10% increase in aerosol penetration to add conservatism to the main steam line leakage pathway. The removal efficiencies are given below.

Table 6-14
Main Steam Line Removal Fractions

Time after release (hr)	MSL 1 (failed steamline)	MSL 2 (pipe to intact steamlines)	MSL 3 (intact steamlines)
0.0	0.681	0.7206	0.714
0.5	0.835	0.7206	0.813
1.5	0.8713	0.7206	0.8361
3.0	0.89	0.7206	0.8449
5.0	0.8614	0.7206	0.8339
7.0	0.8185	0.7206	0.7998
9.0	0.769	0.7206	0.7558
11.0	0.3653	0.7206	0.3807
720	0.0	0.0	0.0

The elemental iodine removal efficiency is 0.45 for all steam lines (DIN 25).

7.0 ACCIDENT SCENARIO AND CHRONOLOGY

0 minutes to 2 minutes

A design basis double-ended guillotine break occurs in a main steam line upstream of the inboard MSIV, releasing reactor coolant into the drywell. The drywell is pressurized driving drywell atmosphere out the MSIVs and into containment via the drywell bypass. All MSIV and containment leakage is initially directed to the environment. The AEGTS system achieves a 0.25-inch vacuum in the secondary containment at 40 seconds (Assumption 5.5) and draws 2000 cfm of secondary containment atmosphere through a HEPA filter and charcoal bed before release to the environment. Because there is no core damage during this 40 second drawdown period, it is not included in the RADTRAD model. Following this drawdown period, all primary containment leakage is directed to the secondary containment except for the containment bypass leakage, which is assumed to bypass secondary containment and is released directly to the environment. No credit for elemental or organic iodine removal by the AEGTS charcoal adsorbers is taken. Particulate removal by the HEPA filters is assumed to be 99% in accordance with Regulatory Guide 1.52. The control room and offsite dose points begin to accumulate dose from the ECCS, MSIV and containment leakage. The control room normal ventilation mode is assumed to continue until the CRERS is manually initiated at 30 minutes.

2 minutes to 32 minutes

The gap release begins by releasing the gap source terms into the drywell at a constant rate over the 30-minute release period following the onset of gap release at two minutes post-accident. ECCS leakage is assumed to begin at this time leaking contaminated suppression pool water (10% of iodine - all forms) directly to the environment even though the postulated core damage is occurring because no ECCS injection is assumed to be available during the first two hours. The control room and offsite dose points begin to accumulate dose from the ECCS, MSIV and containment leakage. At 30 minutes post-accident, the control room normal ventilation system is manually isolated, and the CRERS is manually initiated. The CRERS fans recycle 27,000 cfm of control room atmosphere through HEPA filters and charcoal adsorbers. Manual initiation of containment spray is assumed at thirty minutes. Manual initiation of containment spray at 30 minutes is reasonable based on Emergency Operating Procedure guidance (DIN56) requiring operation of containment spray based on the "Pressure Suppression Pressure" curve contained in the EOP. The containment pressure threshold is met within 30 seconds of the LOCA per DIN 27. AW 6.27.14

32 minutes to 122 minutes

At 30 minutes, the control room normal ventilation system is manually isolated, and the CRERS is manually initiated. The CRERS fans recycle 27,000 cfm of control room atmosphere through HEPA filters and charcoal adsorbers. The in-vessel release begins at 30 minutes after the onset of gap release by releasing the in-vessel source terms into the drywell at a constant rate over a 90-minute release period. Manual initiation of containment spray is assumed at thirty minutes. Manual initiation of containment spray at 30 minutes is reasonable based on Emergency Operating Procedure guidance requiring operation of containment spray based on the "Pressure Suppression Pressure" curve contained in the EOP. The containment pressure threshold is met within 30 seconds of the LOCA per DIN 27.

122 minutes to 24 hours

The source term release from the vessel is terminated at 2 hours after the onset of gap release with the actuation of ECCS, which results in large amounts of steam evolution and large flows out of the drywell into the containment. The drywell and lower containment region are assumed to become well-mixed at 2 hours. / 6-17-14

24 hours to 30 days

Releases to the environment via the containment bypass, MSIV leakage, ECCS leakage and AEGTS exhaust continue for 30 days. As discussed above, containment leakage is reduced at 24 hours based on containment pressure.

8.0 MODEL DEVELOPMENT

This analysis considers the following three pathways through which source terms can be released from the containment.

- ECCS liquid leakage outside of containment
- MSIV leakage
- Containment airborne leakage (containment bypass and containment leakage)

These three pathways are discussed below. RADTRAD modeling capabilities allow incorporation of the MSIV leakage and the containment airborne leakage into one model, therefore; the three release pathways are addressed in two RADTRAD models.

8.1 ECCS Liquid Leakage

8.1.1 Source Terms

The gap and core activity is released to the drywell atmosphere based on the release fractions and timing reported in Tables 1 and 4 of R.G. 1.183 and is assumed to be immediately dissolved in the suppression pool. Only halogens are modeled in this analysis. Noble gases are not soluble and, with the exception of iodine, all other radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase. This is consistent with the guidance of R.G. 1.183, Appendix A.

8.1.2 Volumes

The suppression pool inventory expected during the LOCA is 114,379 ft^3 (DIN 11). No credit is taken for holdup in the Auxiliary Building where the ECCS systems are located.

8.1.3 Flows

The earliest that the containment spray system could potentially be automatically initiated to spray the containment is 10 minutes post-accident if high containment pressure combined with other LOCA signals is sensed. For this calculation it is conservatively assumed that the ECCS system leakage begins immediately after the LOCA (at the beginning of the gap release at two minutes post-accident).

Consistent with the previous analysis (DIN 25), this analysis assumes that the ECCS leakage is 15 gph (0.0334 cfm) for the entire duration of the accident. This is twice the established administrative limit of 7.5 gph. Additionally, leakage from a gross failure of a passive component is assumed to occur at a rate of 50 gpm (6.68 cfm) starting 24 hours into the accident and lasting 30 minutes in accordance with NUREG-0800 (DIN 44).

8.1.4 Removal Mechanisms

Because the suppression pool temperature will not exceed 212°F (DIN 11) during the accident, ten percent of the iodine in the ECCS leakage is assumed to become airborne consistent with R.G. 1.183, Appendix A. Natural removal mechanisms and holdup in the auxiliary building are conservatively neglected. Consistent with Section 5.6 of R.G. 1.183, Appendix A, the chemical species of these airborne source terms is assumed to be 97% elemental and 3% organic.

8.1.5 Model

The ECCS liquid leakage model is illustrated in Figure 1.

8.1.6 Results

The radiological doses for the ECCS liquid leakage transport path are reported in Table 11-1. The RADTRAD output file, including the input summary, is listed in Attachment 1.

8.2 MSIV Leakage

8.2.1 Source Terms

As discussed previously, the PNPP core source terms have been developed with the ORIGEN2 methodology. These source terms are released into the drywell based on the release fractions and timing reported in Tables 1 and 4 of R.G. 1.183.

8.2.2 Volumes

This analysis assumes a double guillotine pipe rupture in one of the four main steam lines upstream of the inboard MSIV and failure of all four main steam shutoff valves (1N11-F0020A, B, C, and D) valves to close as a result of a common power failure (single-failure criterion). The maximum allowable MSIV leakage of 250 scfh is modeled to occur through two pathways: (1) through the broken steam line and, (2) through the second and third intact steam lines. The volume of the ruptured main steam line between the MSIVs is 146 ft^3 (DIN 11). Leakage past the second MSIV in this line is released directly to the environment. The volume of the two intact steam lines between the reactor vessel and the inboard MSIVs is 440 ft^3 (DIN 25). The leakage past the first MSIVs in these lines is released to the volume between the first and second MSIVs which is 292 ft^3, two times the volume between the MSIVs in one steam line (146 ft^3) (DIN 11). Leakage past the second MSIVs in these lines is also released directly to the environment. This configuration was previously identified in DIN 17 to be limiting with respect to dose consequences.

8.2.3 Flows

This calculation will apply a maximum MSIV leak rate of 250 scfh with the worst-case main steam line leaking no more than 100 scfh. The leakage limit is assumed to occur: (1) 100 scfh through the broken steam line, (2) 100 scfh through a second intact steam line, and (3) the remaining 50 scfh through a third intact steam line. As stated above, leakage is modeled to occur through two paths, one path consisting of the broken steam line and a second path consisting of the second and third intact steam lines. All leakage past the outboard MSIVs is assumed to be released to the environment.

The drywell atmosphere will not be at standard conditions after the reactor blowdown. The MSIV leakage rate must be converted to a flow at the drywell conditions. The MSIV leakage rates at the drywell conditions were determined from PSAT 04202H.04 Rev. 0 (DIN 12). The leakage rates are reduced at 2 hours when well-mixed conditions between the drywell and primary containment apply. Additionally, the flow in the main steam lines past the inboard MSIV is represented as well-mixed. The total MSIV leakage rates (DIN 1 and 11) are 298 cfh for the first two hours and 247 cfh thereafter. The maximum flow rate is 191 cfh (DIN 11) through any single main steam line to the environment. The values used in the model are given in Section 6.12.

In addition to the leakage through the MSIVs, the drywell will also continue to leak activity into the containment over this 2 hour period. This calculation will assume a leakage rate of 3000 cfm for the drywell bypass flow consistent with PSAT 08401T.03 (DIN 25).

8.2.4 Release Points

All MSIV leakage past the outboard MSIV is assumed to be released directly to the environment. No credit for holdup in the auxiliary building or turbine building is taken.

8.2.5 Model

The RADTRAD model applied for this leakage path, as well as the containment airborne leakage path, is illustrated in Figure 2.

8.3 Containment & Containment Bypass Leakage

8.3.1 Volumes

In addition to the main steam lines, the following volumes are used in the LOCA airborne leakage dose calculation (DIN 11 and DIN 31):

Table 8-2
LOCA Volumes

Volume in Model	Name in Model	Description	Volume (ft^3)
1	Drywell	Drywell	2.765E+05
5	Sprayed	Sprayed Region of the Containment above the Operating Floor at El. 208'10"	4.812E+05
6	Unsprayed	Unsprayed Region of the Containment	6.842E+05
7	Annulus	Secondary Containment	1.96E+05
9	Control Room	Control Room	390,020

The volume of the Perry control room has recently been re-evaluated. The current volume to be used in Control Room Dose calculations is 390,020 ft^3 (DIN 16).

8.3.2 Flows

From Drywell Volume Into Containment (Suppression Pool Bypass)

The flow rate from the Drywell to the Wetwell are given below, see PSAT 04212H.02 (DIN 43):

Table 8-3
Drywell Flows

Time After Gap Release (hours)	Flow from DW to WW (cfm)	Flow from WW to DW (cfm)
0 - 0.5	0	0
0.5 – 2.0	3000	0
2.0 – 720	2.77E+05	2.77E+05

At two hours, the drywell and unsprayed portion of the containment will be assumed to become instantly well-mixed without credit for suppression pool scrubbing in accordance with Regulatory Guide 1.183, Section 3.7 (DIN 7).

From Unsprayed and Sprayed Containment Volumes to the Environment

By the time the gap release begins, the containment is completely isolated and only containment leakage is assumed. The design basis containment leakage for Perry is 0.2% per day. Since the AEGTS will not completely draw down the annulus for 40 seconds, a 40 second positive pressure period is assumed in which all containment leakage is assumed to leak directly to the environment, but because there is no radionuclide release during this 40 second time period, before gap release which begins at two minutes, this leakage does not contribute to the onsite or offsite doses and is not included in the model.

From Unsprayed and Sprayed Containment Volumes to the Annulus

The majority (89.92%) of the total containment leakage (L_a) is drawn into the annulus by the AEGTS. Although the primary containment is enclosed by the secondary containment, there are systems that penetrate both the primary containment and the shield building boundaries that could create potential pathways through which fission products in the primary containment could bypass the leakage collection and filtration systems associated with the shield building. The Perry Technical Specification SR 3.6.1.3.9 (DIN 15) limit the secondary containment bypass leakage to equal to or less than 5.04 percent of the primary containment leak rate. This analysis uses a bypass leakrate of 10.08 percent of the primary containment leak rate.

From Unsprayed and Sprayed Containment Volumes to the Annulus

As stated above, the majority (89.92%) of the total containment leakage (L_a) is drawn into the annulus where it is filtered by the installed HEPA filters at a credited efficiency of 99% before being released into the environment.

Mixing Between the Unsprayed Containment and Sprayed Containment

The mixing rate between the unsprayed containment and the sprayed containment is 71,400 cfm, Calculation PSAT 04202U.03, Rev. 0 (DIN 13).

From Secondary Containment

The only flow from secondary containment is via the AEGTS system which draws 2000 cfm through a charcoal-filter unit and HEPA filter unit. The HEPA filters are tested per Regulatory Guide 1.52 and therefore are credited for a 99% removal efficiency in the analysis; however, no credit is taken for the charcoal adsorbers in this analysis.

8.3.3 Removal Mechanisms

Natural removal mechanisms for elemental iodine and aerosols will be applied in this calculation using NRC correlations. Elemental iodine removal is credited in the drywell and containment volumes. Aerosol removal is credited only in the drywell and unsprayed region of the containment since containment spray will adversely impact the particle size distribution in the containment.

Fission product removal by containment sprays is considered. The Perry containment spray system is initiated manually based on high radiation readings or is initiated automatically approximately 10 minutes following a LOCA based on pressure and low water level. In this calculation, sprays are assumed to be manually initiated at 30 minutes. The Powers model for aerosol removal by sprays which is built into the RADTRAD code is used in this analysis. Consistent with the guidance in Section 3.3 of Appendix A to R.G. 1.183, the maximum spray decontamination factors for elemental iodine is 200 based on Standard Review Plan 6.5.2, Section III, D. After the aerosol mass has been depleted by a factor of 50, the spray removal lambda is assumed to decrease by a factor of 10.

The following section determines when these DFs were determined to occur. As discussed in Section 3.3 of Appendix A to R.G. 1.183, these DFs are based on the inventories at the end of the in-vessel release phase. Containment spray is assumed to end at 24 hours and the aerosol removal by containment spray is terminated.

<u>Decontamination Factor Reductions</u>

As discussed above, the elemental iodine removal by natural deposition is neglected after a DF of 200 is reached. Based on the elemental iodine lambda of 0.878 hr^{-1} in the drywell, a DF of 200 would be reached in approximately six hours without any leakage or decay. The output in Attachment 2 indicates that the Drywell 2-hour post-accident release (i.e., 2.0000 hr) elemental I-131 inventory of 2.4087E+22 atoms has reduced to 1.29E+20 at 3.0 hours post-accident representing a DF of 186. This calculation will therefore model the elemental iodine removal to end at 3.0 hours in the drywell.

In the containment, the total (sprayed + unsprayed) elemental I-131 inventory is 3.1526E+21 atoms after the drywell is flushed at two hours post-accident (i.e., 2.0000 hr). The total activity in both regions of the containment is considered because, if only the activity in the sprayed region of the containment was considered, a longer period of the higher lambdas would be applicable. At 3.75 hours, the total elemental I-131 inventory in the sprayed and unsprayed containment regions is 1.6644E+19 atoms, representing a DF of 189. This calculation will model spray removal to end at 3.75 hours.

The particulate removal (Powers Model) in the sprayed region of the containment is reduced by a factor of 10 when the aerosol activity is reduced by a DF of 50. In the containment, the total (sprayed + unsprayed) particulate inventory is 8.13 kg after the drywell is flushed at two hours. At 4.95 hours, the total aerosol inventory is 1.64E-01 kg, representing a DF of 49.7. This calculation will model this removal coefficient to be reduced at 4.95 hours. This is accomplished by reducing the spray flow used in the Powers Model by a factor of ten at this time.

8.3.4 Release Points

All source terms released via containment leakage are released through the plant vent.

8.3.5 Model

The containment airborne model is illustrated in Figure 2 which is based on the time at which the gap release begins. This figure also includes the MSIV leakage transport pathways.

8.4 Control Room

Although the current configuration of the control room HVAC system would automatically initiate the control room recirculation on a LOCA signal, this analysis assumes that the CRERS is manually initiated at thirty minutes. Once the CRERS is initiated, CRERS fans recycle 27,000 cfm of control room atmosphere through HEPA filters and charcoal adsorbers before being returned to the control room. The normal control room recirculation air flow is 45,000 cfm (DIN 52) including 6,600 cfm of outside air for ventilation. To represent the normal positive pressurization in the control room, the exfiltration air flow is modeled as 4,800 cfm before isolation at 30 minutes. The RADTRAD code only allows a single control room recirculation air flow. As a result, the normal recirculation air flow is not modeled. This is acceptable because the normal recirculation flow does not change the radionuclide concentration in the control room.

After isolation, unfiltered inleakage of 1375 cfm is assumed to be drawn from the control room intake duct for the duration of the postulated accident (30 days). The flow from the control room to the environment is also set at 1375 cfm to avoid pressurization.

Consistent with the requirements of R.G. 1.183, the contribution to the control room dose due to shine from the containment building (0.13 Rem) and release plume (0.002 Rem) must be considered. These dose contributions are given in PSAT 04202H.13 (DIN 17). These 30-day doses to the control room were generated with the previous TID 14844 source term that assumed an instantaneous release to the containment of 100% of the core inventory of noble gases and 50% of the radioiodines. These assumptions are conservative compared to the Alternative Source Term methodology (AST) due to removal of radioiodines from the containment atmosphere by sprays and deposition thereby reducing the radionuclide concentration in containment. In addition, the total halogen release fraction is 0.3 for the AST methodology providing additional margin. Based on these considerations, the previously calculated shine and plume doses are considered bounding for this analysis.

8.5 RADTRAD MODEL

The models developed for the analysis are illustrated in Figures 1 and 2.

9.0 Operator Actions

The operator actions assumed in this analysis include the following:

1. Manual initiation of containment spray at 30 minutes
2. Manual initiation of CRERS at 30 minutes
3. The pH calculation (DIN 26) assumes initiation of Standby Liquid Control (~~SBLC~~) to control suppression pool pH SLC

AW 6.27.14

Figure 1

Fission Product Transport Model

(ECCS Leakage Pathway)

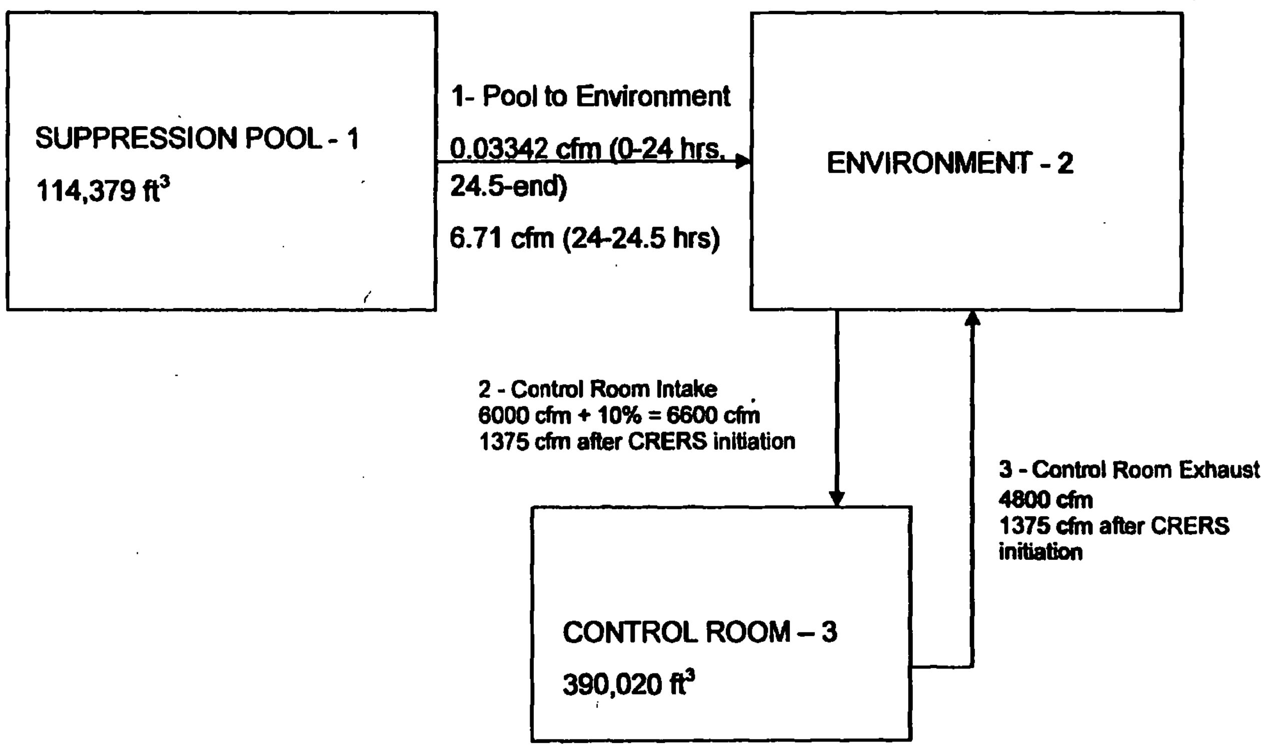
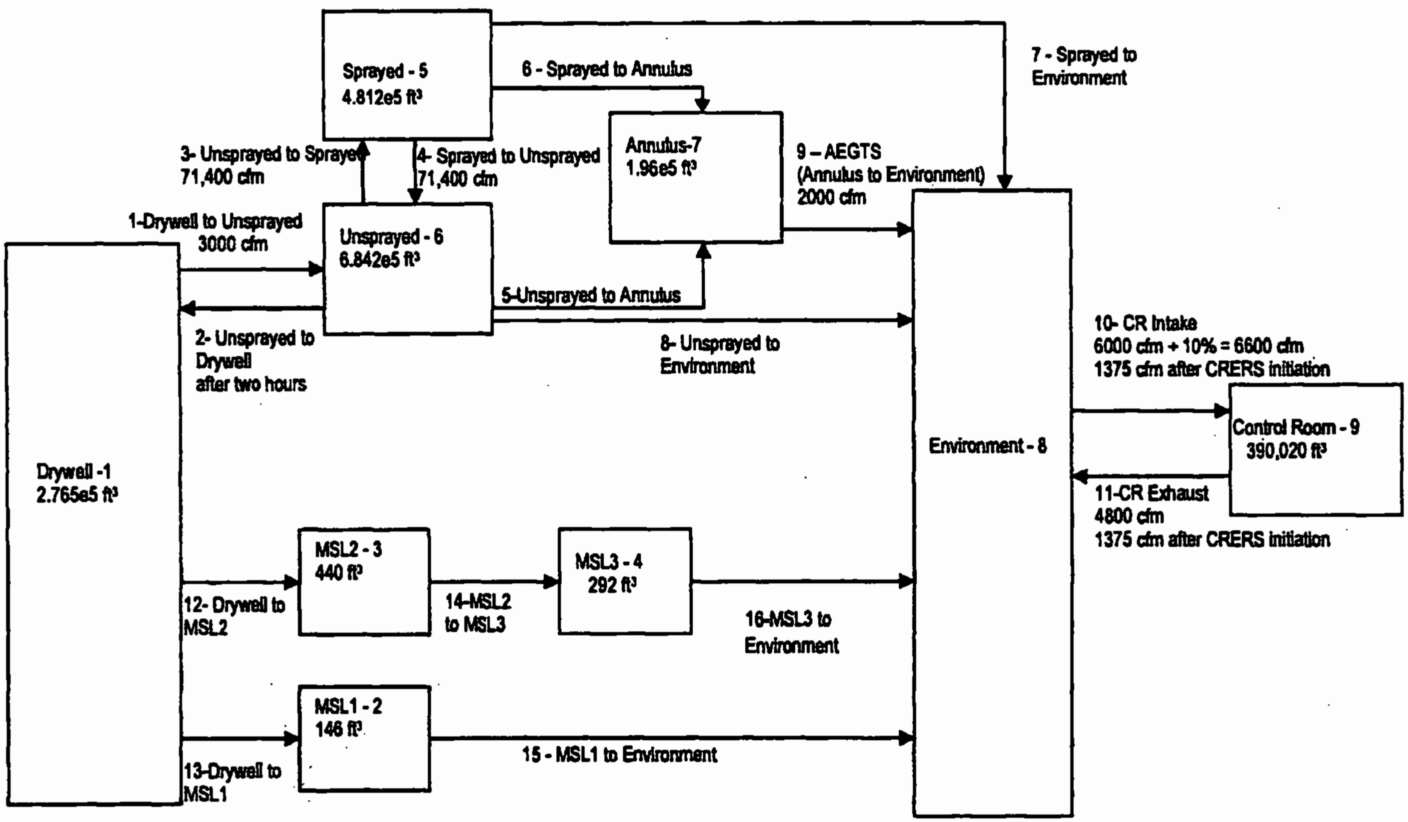


Figure 2
FissION Product Transport Model
(MSIV and Containment Leakage Pathways)
Gap Release Phase



10.0 COMPUTATION

The RADTRAD output files, which include the input summary, are given in Attachments 1 and 2. The RADTRAD input and output files used for this calculation are identified below:

Description	ECCS Leakage	Containment Leakage
Plant scenario file	PNPP ESF.psf	PNPP LOCA.psf
Auxiliary RADTRAD Input Files		
Nuclide Inventory File	PNPP ESF.nif	PNPP LOCA.nif
Release Fraction and Timing File	pnpp_esf.rft	PNPP_DBA.rft
Dose Conversion Factors	Fgr11&12.inp	Fgr11&12.inp
Output File	**PNPP ESF.out**	**PNPP LOCA.out**

Files for the TSC dose calculation are:

Plant Scenario Files	Output Files
PNPP ESF TSC.psf	PNPP ESF TSC.out
PNPP LOCA TSC.psf	PNPP LOCA TSC.out

For the TSC analyses, the nuclide inventory files, release fraction and timing files, and the dose conversion factor files for the LOCA and ESF cases are the same as above.

11.0 Overall Results

Table 11-1 presents the dose results for individual leakage pathways for MSIV leakage, containment leakage, containment bypass, ECCS leakage, and shine dose. Control room shine dose is from DIN 17 and 25.

Table 11-1
Dose Results (rem TEDE)

Pathway	EAB	LPZ	Control Room	TSC
Containment & MSIV Leakage	20.4	5.0	1.7	0.36
ECCS Leakage	0.79	1.83	1.15	0.05
Shine Dose			0.132	0.132*
Total	21.2	6.9	3.0	0.5
Regulatory Limit	25	25	5	5

*Assumed to be the same as the Control Room

CALCULATION ADDENDUM

☐ BV1	☐ BV2	☐ DB	☒ PY
TITLE/SUBJECT: (MUST MATCH ORIGINAL CALCULATION TITLE (SUBJECT)) Control Rod Drop Accident Radiological Analysis using Alternative Source Terms			

Classification	☒ Tier 1 Calculation	☒ Safety-Related/Augmented Quality	☐ Nonsafety-Related
Open Assumptions?	☐ Yes ☒ No	If Yes, Enter Tracking Number	Initiating Document
(Perry Only)	Referenced In Atlas? ☐ Yes ☒ No		
(Perry Only)	Referenced In USAR Validation Database ☐ Yes ☒ No		

Computer Program(S)				
Program Name	Version / Revision	Category	Status	Description
N/A				

Originator (Print, Sign & Date)	Reviewer/Design Verifier(Print, Sign & Date)	Approver(Print, Sign & Date) 6/25/14
A. Widmer A. Widmer 6-24-14	Marvin Morris Marvin Morris 6/25/2014	M. Bensi for D. Culler

OBJECTIVE OR PURPOSE OF ADDENDUM:

The purpose of this addendum is to clarify assumption 3.1.5.8 with respect to the iodine fractions released, and Iodine fractions input into the RADTRAD code. One objective is to prevent an error likely situation from occurring during the next revision.

SCOPE OF ADDENDUM:

There is a discrepancy between Assumption 3.1.5.8 and the input into the RADTRAD code. This addendum is initiated to ensure that the next full revision corrects the discrepancy and to formally document that there is no impact on the results or conclusions.

LIST NEW DOCUMENTS TO BE ADDED TO THE DOCUMENT INDEX (DIN):

DIN No.	Document Number/Title	Revision, Edition, Date	Reference	Input	Output
			☐	☐	☐
			☐	☐	☐

SUMMARY OF RESULTS/CONCLUSIONS OF ADDENDUM:

There are no changes to the results of this calculation. This addendum documents a discrepancy between assumption 3.1.5.8 and the input values into attachments 1, 2, and 3.

The assumption states that the iodine release from the turbine and condenser is assumed to be 97% elemental and 3% organic and is consistent with guidance in section 3.6 of Appendix C to Regulatory Guide 1.183 Rev. 0. Attachment 1 (pages 4, 9, 27, and 32) and Attachment 2 (pages 2, 9) used iodine fractions of 95% aerosol, 4.85% elemental, & 0.15% organic. Attachment 3 (pages 2, 8) used iodine fractions of 97% elemental and 3% organic. Attachment 3 utilized the correct release from the turbine and condenser as it matches the release fractions contained in assumption 3.1.5.8.

Future revisions of this calculation shall assume the iodine fractions released from the turbine and condenser to be 97% elemental and 3% organic unless updated regulations dictate otherwise.

As stated above, this inconsistency does not impact the results or conclusions of Revision 1. For the dose associated with the EAB, LPZ, and Control Room (without isolation), the form of the iodine released does not impact the dose to those areas as no filtration was credited in the calculation for the offsite locations or the Control Room (without isolation). As a result, there is no change in the consequences for those areas. For the Control Room evaluations which assume isolation and filtration, there is also no impact on Control Room doses because the efficiencies for the Control Room filtration were set to 80% for both the charcoal and HEPA filters. This ensures that the elemental iodine, organic iodine, and aerosol iodine are filtered the same. There would be no change to the isolated control room dose unless the filtration efficiencies were changed such that the removal efficiency for different iodine forms are different. As evidence, attachment 2 and attachment 3 of the base calculation utilized the different iodine fractions. It is noted that the calculated EAB and LPZ doses are identical as seen on Page 30 of Attachment 2 and Page 44 of Attachment 3.

LIMITATIONS OR RESTRICTIONS CREATED BY ADDENDUM:

None

IMPACT OF ADDENDUM ON OUTPUT DOCUMENTS:

None

DESCRIBE WHERE THE ADDENDUM WILL BE EVALUATED FOR 10CFR50.59 APPLICABILITY:

RAD/SCREEN/EVAL 13-02880 for Revision 1 of this remains applicable.

LIST SUPPORTING DOCUMENTS: (Include total number of pages)

Design Verification Record 1 page
Calculation Review Checklist 3 pages
Design Interface Summary 7 pages

LIST ATTACHMENTS: (Include total number of pages)

None

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS AND SYSTEMS

Certain structures, components and systems of the nuclear plant are considered important to safety because they perform safety actions required to avoid or mitigate the consequences of abnormal operational transients or accidents. The purpose of this section is to classify structures, components and systems according to the importance of the safety function they perform. In addition, design requirements are placed upon such equipment to assure the proper performance of safety actions, when required.

3.2.1 SEISMIC CLASSIFICATION

Plant structures, systems and components important to safety are designed to withstand the effects of a Safe Shutdown Earthquake (SSE) and remain functional if they are necessary to assure:

a. The integrity of the reactor coolant pressure boundary,

b. The capability to shut down the reactor and maintain it in a safe condition, or

c. The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of <10 CFR 100>.

or <10 CFR 50.67> (future revisions to design basis analyses that compare consequences to 10 CFR 100 will be updated to <10 CFR 50.67>)

Plant structures, systems and components (including their foundations and supports) designed to remain functional in the event of an SSE are designated as Seismic Category I, as indicated in <Table 3.2-1>.

Structures, components, equipment, and systems designated as Safety Class 1, Safety Class 2 or Safety Class 3 (see <Section 3.2.3> for a discussion of safety classes) are classified as Seismic Category I except for (1) those noted in <Table 3.2-1> and (2) those portions of

(M25-C002A or B) is also deactivated and the electric heating coil in the charcoal filter train is automatically energized upon receipt of an emergency signal.

The emergency recirculation system causes the supply air to be filtered through the charcoal filter train (M26-D001A, B) before being distributed to the control room. This system is idle during normal plant operation. During periods of loss of offsite power, emergency power will be supplied by the standby diesel generators.

The degree to which the recommendations of <Regulatory Guide 1.52> are followed is given in <Table 6.5-1>.

The main components of this system are located in the control complex at Elevation 679'-6" and consist of two 100 percent capacity filter trains. Each filter train includes the following sequential components: demisters, roughing filters, electric heating coil, HEPA prefilters, charcoal filters, HEPA after filters, centrifugal fan, isolation damper, and check damper.

The fans, filter elements and dampers are of standard industrial design, manufactured in accordance with Quality Assurance (QA) requirements of Safety Class 3, Seismic Category I items. The filter racks, frames and housing are specially designed to satisfy the system space requirements and also meet the above QA requirements.

Design information for the major components in this system is listed in <Table 6.4-3>.

6.4.2.3 Leak Tightness

The control room system is designed so that, when operating in a normal mode (admitting outside air), the system automatically maintains a positive differential pressure between the control room and the outside

The design basis radiological calculations for a

Normally, the control room boundary inleakage is maintained at a value consistent with pre-operational testing such that the actual inleakage is substantially less than 1375 cfm.

Throughout the life of the plant, various plant activities may need to be performed which temporarily degrade the control room boundary such that the unfiltered inleakage significantly exceeds 1375 cfm. ~~If a~~ postulated LOCA ~~were to occur under these conditions, parametric analyses have shown~~ that it is acceptable to delay the restoration of the control room boundary, provided that once it is restored, the actual unfiltered inleakage would be reduced below 1375 cfm for the remainder of the accident. This allows for temporary degradations of the boundary to occur without impacting overall accident dose to the control room operators. Administrative controls are utilized during planned degradations to ensure the boundary can be restored within the bounding parameters of ~~the analyses <Figure 6.4-4 (1)> and <Figure 6.4-4 (2)>.~~

assume an unfiltered inleakage of ~~6000~~ 6600 cfm for the first 30 minutes, which shows

to at or

analysis.

rapidly

to

6.4.4.2 Toxic Gas Protection

No toxic materials which could interfere with control room occupancy are stored in the plant. Sodium hypo-chlorite, rather than chlorine, is used as a biocide. No chlorine is stored on site. The potential effects of offsite and onsite hazardous materials are discussed in <Section 2.2.2> and <Section 2.2.3>. Protection against offsite toxic gases are detailed in <Section 6.4.1.g>.

6.4.4.3 Control Room Emergency Recirculation System

The general arrangement and control of the control room emergency recirculation system is as described in <Section 6.4.2.2.2>. Detailed information concerning the emergency filter is presented in <Section 6.5.1>. The equipment is shielded, housed in a Seismic Category I structure, separated, redundant, and powered from the

the design basis LOCA <Section 15.6.5.5.1.9> credits an 80 percent removal efficiency of elemental and organic iodines by the charcoal filters in the CRERS. The Steam System Piping Break Outside Containment <Section 15.6.4>, Control Rod Drop Accident <Section 15.4.9>, and the Fuel Handling Accident <Section 15.7.4> and <Section 15.7.6>, do not take credit for the charcoal filters in the CRERS.

organic species of iodine. For the CRERS, ~~the alternative source term LOCA analysis and one fuel handling accident sensitivity case assumed an elemental and organic removal efficiency of only 50% for the charcoal adsorbers~~. For the FHAES, the alternative source term FHA analysis took no credit for the charcoal adsorbers. The CRERS and FHAES charcoal adsorber beds are 2 inches deep. Exhaust air for both plenums is maintained at less than 70 percent relative humidity.

The HEPA filter efficiency of all the plenums is 99.97 percent on particles 0.3 microns and larger. However, ~~no credit was taken for the HEPA filters in the alternative source term analysis for the fuel handling accident~~.

the LOCA analysis only credits the HEPA filters in the AEGTS and CRERS at an efficiency of 99 percent. The other design basis radiological calculations do not take credit for the HEPA filters in the AEGTS, CRERS, or FHAES.

Additional bases for the design of the CRERS, FHAES and AEGTS are presented in <Section 6.4>, <Section 9.4.2>, and <Section 6.5.3> respectively.

6.5.1.2 System Design

The design features of the CRERS, FHAES and AEGTS are compared to the recommendations of <Regulatory Guide 1.52> in <Table 6.5-1>, <Table 6.5-2>, and <Table 6.5-3> respectively.

Design of the activated charcoal adsorber plenums used in the CRERS, FHAES and AEGTS follows the guidelines of <Regulatory Guide 1.52> and ERDA 76-21.

Each charcoal adsorber plenum contains the following:

a. Demisters to remove large particles and water droplets (about 1 micron diameter).

b. Roughing filters to remove large particles (about 1 micron).

e. Gasketless activated charcoal adsorber beds to remove gaseous elemental and organic iodines.

f. HEPA filters downstream of the charcoal beds to remove charcoal particles that may be entrained in the air stream.

g. A fan external to the plenum.

h. Instrumentation.

i. Test ports.

j. Water deluge system for fire protection.

Plenum housings and filter support frames are shop fabricated. Potential leakage and bypass paths are closed by seal welding. No caulking or sealant is used. Housings are fabricated of carbon steel sheet. Filter support frames are of unpainted stainless steel.

Roughing and HEPA filters are mounted in frames in accordance with the recommendations of ERDA 76-21.

The activated charcoal adsorber is bulk loaded into the permanently installed, gasketless adsorber section which is seal welded to the housing and support frames of the plenum. Tray type activated charcoal adsorber units are not used.

Spent charcoal adsorber material is vacuumed from the bottom or top of the plenum and is loaded into 55 gallon drums for shipment off site. New charcoal adsorber material is added at the top of the adsorber section. Personnel are not directly exposed to potentially contaminated adsorber material during the changing procedure.

6.5.1.6 Materials

Estimated quantities of materials used in the activated charcoal adsorber plenums for CRERS, FHAES and AEGTS are listed in <Table 6.5-4>, <Table 6.5-5>, and <Table 6.5-6> respectively. The governing specifications for the various materials are also listed and provide information regarding chemical composition of materials used.

There are no radiolytic or pyrolytic decomposition products from the ESF filter systems. Actuation of the activated charcoal adsorber plenum water deluge fire protection systems will extinguish a charcoal fire before pyrolytic decomposition products are formed. None of these systems are located in areas where gamma radiation sources are sufficiently strong to cause radiolytic decomposition products. Therefore, decomposition products do not affect any engineered safety features.

6.5.2 CONTAINMENT SPRAY SYSTEM

6.5.2.1 Design Bases

a. The containment spray system (CSS) is a part of the residual heat removal (RHR) system.

b. The CSS provides containment cooling following a loss-of-coolant accident, in addition to being a fission product removal mechanism. Refer to <Section 6.2.2> for the heat removal function of the CSS.

c. The CSS consists of two completely redundant and independent loops. (Loops "A" & "B")

d. The CSS is designed to remain operable in the containment accident environment, which is discussed in <Section 3.11>.

are not subject to clogging by particles less than 1/4 inch in maximum dimension. Each nozzle header is independently oriented to ensure efficient coverage of the containment volume.

The minimum water supply flow rate to the containment spray system is 5,250 gpm.

There are no spray additives for the CSS (other than the pH buffering chemical, boron solution, from the standby liquid control system, which is injected into the reactor vessel and suppression pool following a design basis LOCA). The CSS will automatically initiate after 10 minutes of a LOCA signal if containment pressure exceeds the high pressure setpoint. If containment pressure is less than high pressure setpoint, the control room operator can actuate the system manually.

The sprayed and unsprayed volumes and regions of the containment, with their associated mixing rates, are discussed in <Section 15.6.5>.

The CSS takes no credit for ventilation.

6.5.2.3 Design Evaluation

No changes to this page.
Provided for context.

The containment spray mode of the RHR system is safety-related and is designed to operate following the postulated design basis loss-of-coolant accident. A high degree of system reliability is maintained through system quality control, by general equipment arrangement to provide access for inspection and maintenance and by periodic testing. A single failure analysis of the RHR system is given in <Section 6.2.2>.

Because of the large volume of the containment atmosphere swept by the sprays, the spray mode serves as a removal mechanism for fission products postulated to be dispersed in the containment atmosphere following an accident. Radioiodine in its various forms is the fission

~~No changes to this page. Provided for context.~~ (aerosol) and elemental

product of primary concern in the evaluation of a loss-of-coolant accident. The major benefit of the containment spray is its capacity to collect and remove ~~particulate~~ iodines from the containment atmosphere and thus reduce ~~its~~ their release to the environment. Offsite and control room operator doses are a function of both the rate of removal and the final equilibrium decontamination factor. The dose calculation assumes (non-mechanistically) that the containment spray will operate for up to 24 hours. However, the dose calculations also expand on this assumption, noting the following:

1) The dose calculations assume the sprays are run for the first 24 hours, then are suspended. This is the most important time period for scrubbing of radiation down into the suppression pool. However, in an actual event, spray use would not necessarily be suspended at 24 hours, if appropriate conditions for their use still existed. Therefore, the phrase "up to" is <u>not</u> intended to be interpreted to stop using sprays after 24 hours.

2) The phrase "up to" <u>is</u> intended to mean that in an actual event, the sprays will be run when it is appropriate, and not necessarily the entire time during the first 24 hours of a LOCA. This does not invalidate the assumptions in the dose calculations. The accident guidance to operators must be written to be symptom based, rather than event based. Most postulated LOCAs will not result in large radiation releases. Therefore, it would not be appropriate to run containment sprays for 24 hours following such an event. Another critical factor in spray use is containment pressure. Use of the sprays will work to reduce containment pressures, due to steam condensation and the containment heat removal function that they provide. In the majority of cases, if a high radiation signal is present from the containment radiation monitor and pressures are elevated in containment, the sprays would be run. However, if containment pressure gets reduced to near zero and use of the sprays is terminated by the operators, this does not have an

Elemental iodine removal is credited in the drywell and containment volumes. Airborne elemental iodine is removed by deposition to the walls in the drywell and containment. As reported in Section 5.1.2 of NUREG/CR-0009 (Reference 1), this process is driven by the temperature differences between the surfaces and the atmosphere.

adverse impact on offsite doses (or the dose calculations) since the driving pressure for containment and MSIV leakage has been eliminated. The dose calcs assume that the maximum allowable leakage (La) corresponding to the peak postaccident pressure (Pa) remains during the entire 24 hours period, so if containment pressure actually gets reduced to substantially less than Pa, a reduction in leakage and the resultant offsite doses will follow.

6.5.2.3.1 Iodine Removal Performance Evaluation

elemental, organic, and particulate (aerosol)

The iodine removal analysis is based on the assumptions presented below and in <Table 6.5-9>.

sprayed and unsprayed

The analysis uses the flow associated with only one RHR pump operating in the containment spray mode. It is conservatively assumed that the containment spray system directly sprays approximately 41 percent of the total containment free volume (excluding the drywell). <Section 15.6.5> provides a description of the volumes and flow paths used in the analyses.

the

deposition factors and drywell

elemental

The calculated iodine removal ~~rates~~ for the containment ~~sprays~~ are given in <Table 6.5-11> ~~for the elemental and particulate iodines as well as other particulates. Because of the large surface area of the initially airborne particulate, the elemental iodine is assumed to be adsorbed onto the particulate and to be removed with it~~.

It has been conservatively assumed in these evaluations of spray removal effectiveness that organic iodine forms are not removed by the sprays.

The model for particulate (aerosol) iodine removal is built into the RADTRAD code. Input parameters are presented in <Table 6.5-9>. The particulate removal coefficient is reduced by a factor of 10 after the aerosol mass has been depleted by a factor of 50. Particulate removal is assumed to end at 24 hours when the sprays are assumed to be stopped.

elemental iodine

These removal ~~constants~~ factors are applied until a decontamination factor (DF) of ~~200 has been~~ up to 200 is obtained.

6.5.2.3.2 Evaluation of Analytical Assumptions

6.5.2.3.2.1 Iodine Retention by Spray Solution

The equilibrium between the concentrations of iodine in the liquid and vapor phases is given by the partition coefficient, H, which is a function of iodine concentration, pH and temperature. In accordance with (Reference 3) re-evolution of iodine does not have to be considered, (i.e., H will be very large) as long as the pH of the suppression pool is maintained greater than or equal to 7.0 postaccident.

6.5.2.3.2.2 Elemental Iodine and Particulate Removal Constant

~~See insert on next page~~ Deleted

~~The calculational model used to determine the elemental and particulate iodine removal constant is Polestar Applied Technology's "STARNAUA" computer code (Reference 4) which incorporates the spray removal modeling features given in Appendix E of (Reference 3). The input data required by the computer code is given in <Table 6.5-9>. The mean drop fall height of 53.2 feet was calculated by taking a weighted average of the height of each ring above the operating floor and the associated spray flow rate as follows:~~

$$\cancel{h = (h_1m_1 + h_2m_2 + h_3m_3) / (m_1 + m_2 + m_3)}$$

~~where:~~

$\cancel{h_1 = 45.75'}$

$\cancel{h_2 = 54.75'}$

$\cancel{h_3 = 61.0'}$

~~and m = ring flow rate = number of nozzles x flow rate per nozzle~~

~~or~~ $\cancel{m_1 = 129 \times 15.22 \text{ gpm} = 1963}$

$\cancel{m_2 = 114 \times 15.22 \text{ gpm} = 1735}$

$\cancel{m_3 = 102 \times 15.22 \text{ gpm} = 1552}$

~~The spatial and temporal distributions are derived from analysis using the USNRC's computer code "SPIRT" (Reference 1).~~

Insert for USAR Section 6.5.2.3.2.2

The two Control Room Emergency Recirculation System (CRERS) subsystems each have a high efficiency particulate air filter, charcoal adsorbers, and a post HEPA filter. The CRERS is an ESF system that is tested in accordance with R.G. 1.52 (Reference 4). The calculation model (Reference 6) assumed an elemental and organic iodine removal efficiency of 80 percent for the charcoal adsorber removal efficiency.

Each HEPA filter is tested to show a penetration and system bypass of less than 0.05 percent when tested in accordance with Regulatory Guide 1.52 (Reference 4). A penetration and bypass of less than 0.05 percent allows credit for a particulate removal efficiency of 99 percent per Regulatory Guide 1.52. The analysis therefore used a CRERS HEPA filter efficiency of 99 percent for aerosol particulates.

The AEGTS includes HEPA filters and 4-inch deep charcoal filters. Particulate removal by the HEPA filters is assumed to be 99 percent in accordance with Regulatory Guide 1.52. The analysis conservatively assumed a removal efficiency of 0 percent for the charcoal adsorbers.

A simplified model for estimating the fission product aerosol removal by containment sprays following a postulated LOCA was used in the analysis. The model for aerosol removal by sprays built into the RADTRAD code is the Powers model. The Powers model was derived by correlating the results of Monte Carlo uncertainty sampling analyses assessing the uncertainties in aerosol properties, aerosol behavior, spray droplet behavior, and the initial and boundary conditions expected to be associated with a postulated LOCA in the containment.

Input parameters for the Powers model are presented in <Table 6.5-9>.

The information in the above insert is provided elsewhere in the USAR or in the USAR changes proposed as part of this request for licensing action, and the above information was inappropriate for Section 6.5.2.3, which is the Design Evaluation description for the Containment Spray system.

6.5.4 ICE CONDENSER AS A FISSION PRODUCT CLEANUP SYSTEM

This section is not applicable to PNPP.

6.5.5 REFERENCES FOR SECTION 6.5

1. Postma, A. K.; Sherry, R. R.; Tam, P. S.; "Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels," <NUREG/CR-0009>, October 1978.

2. ANSI/ANS-56.3-1979, "American National Standard for PWR and BWR Containment Spray System Design Criteria."

3. Electrical Power Research Institute, "Generic Framework for Application of Revised Accident Source Terms to Operating Plants," TR-105909, Interim Report, November, 1995.

~~4. Polestar Applied Technology, Inc., "STARNAUA, A Code for Evaluating Severe Accident Aerosol Behavior in Nuclear Power Plant Containments: A Code Description and Validation and Verification Report," PSAT C101.02, Revision 1, February 23, 1996.~~

4. Regulatory Guide 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in LightWater-Cooled Nuclear Power Plants", Revision 2, March 1978.

5. NUREG-0800, "Standard Review Plan (SRP) 6.5.2, Containment Spray As A Fission Product Cleanup System," Revision 4, March 2007.

6. Calculation 3.2.15.16, "Design Basis LOCA Dose Evaluation Using Alternate Source Terms," Revision 0 ~~October 2013~~.

1, July 2014

TABLE 6.5-9

INPUT PARAMETERS FOR THE SPRAY REMOVAL ANALYSIS

~~Containment minimum pressure, psig~~	~~1~~
~~Containment minimum temperature, °F~~	~~100~~
~~Total~~ Containment (excluding drywell) net free volume, ft^3	~~1.1654~~ $\boxed{1.1654}$ x 10^6
Sprayed containment volume, F^3	481174
Unsprayed containment volume, F^3	684226
Mean spray fall height, ft	~~53.2~~ $\boxed{54.05}$
Number of spray pumps operating	1
Spray flow rate, gpm	5,250
Spray solution pH	7.0
~~Number of drop size groups~~	~~56~~
Q, Spray Flux, cfm/ft^2	0.0621
Alpha, unsprayed/sprayed volume	1.422
Pct, uncertainty percentile	10
~~Geometric mean drop size for spatial distribution, cm~~	~~4.95 x 10^{-3}~~
~~Geometric mean standard deviation~~	~~2.9~~
~~Geometric mean particle size for incoming aerosol, cm~~	~~4.4 x 10^{-5}~~
~~Geometric mean standard deviation~~	~~1.81~~

~~No wall condensation~~

~~No condensation on water droplets~~

~~No consideration of particle hygroscopicity~~

TABLE 15.0-4

DOSE CONVERSION FACTORS[1]

Isotope	Thyroid (rem/Ci)	Whole Body 0.25xMeV/dis
I-131	1.49E+6	8.72E-2
I-132	5.35E+4	5.13E-1
I-133	3.97E+5	1.55E-1
I-134	2.54E+4	5.32E-1
I-135	1.24E+5	4.21E-1
Kr-83m		5.02E-6
Kr-85		3.72E-2
Kr-85m		5.25E-4
Kr-87		1.87E-1
Kr-88		4.64E-1
Kr-89		5.25E-1
Xe-131m		2.92E-3
Xe-133m		8.00E-3
Xe-133		9.33E-3
Xe-135m		9.92E-2
Xe-135		5.72E-2
Xe-137		4.53E-2
Xe-138		2.81E-1

Breathing Rates

Breathing Rate (m^3/~~sec~~) (m^3/sec)

Time Period (hr)	MSLB ~~LOCA~~	FHA, LOCA, CRDA
0-8	3.5 ~~3.47~~E-4[2]	3.5E-4[2]
8-24	1.75E-4	1.8E-4
24-720	2.32E-4	2.3E-4

NOTES:

[1] The following dose conversion factors (DCF's) are used in the alternative source term analyses;

CRDA, MSLB ~~LOCA~~ - ~~DCF's for inhalation:~~ EPA Federal Guidance Report 11 - ~~1988~~ 1989 (Reference ~~9)~~ 11) and EPA Federal Guidance Report 12 - 1993 (Reference 13).

LOCA, FHA - CEDE: EPA Federal Guidance Report 11 - 1989 (Reference 11)
DDE/EDE: MACCS2 computer code (Reference 12), which used Federal Guidance Report 12 - 1993 (Reference 13).

[2] This breathing rate was used for the duration of the Control Room radiological consequence analyses.

TABLE 15.4-12

CONTROL ROD DROP ACCIDENT
EVALUATION PARAMETERS

		Scenario 1 Assumptions	Scenario 2 Assumptions
I.	Data and assumptions used to estimate radioactive source from postulated accidents.		
A.	Power level	3,833 MWt	3,833 MWt
B.	Burnup	N/A	N/A
C.	Fuel damaged	~~1,107~~ 1376 rods[1]	~~770~~ 1376 rods
D.	Release of activity by nuclide	<Table 15.4-13>	N/A
E.	Iodine fractions, %		
	(1) Organic	~~0~~ 0.15	~~0~~ 0.15
	(2) Elemental	~~100~~ 4.85	~~100~~ 4.85
	(3) Particulate	~~0~~ 95	~~0~~ 95
F.	Reactor coolant activity before the accident.	N/A	N/A
II.	Data and assumptions used to estimate activity released.		
A.	Condenser leak rate (%/day)	1.0	N/A
B.	Turbine building leak rate (%/day)	N/A	N/A
C.	Valve closure time (sec)	N/A	~~5~~ N/A
D.	Adsorption and filtration efficiencies		
	(1) Organic iodine	N/A	N/A
	(2) Elemental iodine	N/A	N/A
	(3) Particulate iodine	N/A	N/A
	(4) Particulate fission products	N/A	N/A
E.	Recirculation system parameters		
	(1) Flow rate	N/A	N/A
	(2) Mixing efficiency	N/A	N/A
	(3) Filter efficiency	N/A	N/A
F.	Containment spray parameters (flow rate, drop size, etc.)	N/A	N/A
G.	Containment volumes	N/A	N/A

~~The RAST analysis was pursued initially to support an increase in the main steam line leak rate to 250 scfh and to eliminate the MSIV Leakage Control System.~~

LOCA

The ~~RAST~~ analysis is based on the following:

Regulatory Guide 1.183

- using a reactor accident source term developed from <NUREG-1465>,
- credit for decay during the two (2) minute onset of the gap release,
- ~~relying on natural deposition of fission product aerosol in the~~ drywell and unsprayed regions of the containment,
- relying on natural deposition of fission product aerosol in the main steam lines,
- controlling the pH of the water in the containment to prevent iodine re-evolution,
- operating the containment spray system for up to 24 hours <Section 6.5.2.3>,
- not crediting iodine removal by charcoal adsorbers in the Annulus Exhaust Gas Treatment System (AEGTS),
- delaying actuation of the control room emergency recirculation system for up to 30 minutes,
- utilizing ~~decreasing~~ elemental and organic iodine removal efficiencies of 80 percent for the control room emergency recirculation system charcoal adsorbers ~~from 95 percent to 50 percent~~,
- utilizing an ~~increasing the~~ engineered safety feature system leakage outside primary containment value of 15 gallons per hour (gph), and
- utilizing a ~~increasing the~~ maximum allowable secondary containment bypass leakage of $0.1008L_a$ ~~by 50 percent~~.

The ~~RAST~~ analysis considers the following four potential fission product release pathways following the design basis LOCA:

- main steam isolation valve leakage,
- containment leakage,

The analysis conservatively assumes that the fission product leakage from the main steam lines is released directly into the environment. The leakage past the MSIVs is conservatively assumed to begin immediately after the accident. In actuality, the three intact steam lines would contain trapped steam which would be relatively cooler and more dense as compared to the atmosphere in the reactor vessel upper head during the overheating of the core. This condition would greatly inhibit mixing between the activity released from the core and the steam leaking through the three intact steam lines and the three associated sets of MSIVs. However, for conservatism, all of the lines are assumed to be leaking contaminated drywell atmosphere.

Other significant conservatisms in the analysis of steam line transport include:

STET the typed words, and add the handwritten insert...

(1) ~~No consideration of reduced steam line mass leak rate with decreasing drywell pressure~~, in the region between the Inboard and outboard MSIVs (no reduction in flow for this portion of the main steam lines),

(2) No consideration of steam line mass leak rate for two closed MSIVs in series, and

STET ~~(3)~~ No consideration of particulate removal and even plugging of the extremely small MSIV leak paths due to particulate deposition at the entrance to or within the leak path as the gas flow accelerates to sonic or near-sonic conditions.

Two configurations were analyzed to cover all single-failure possibilities. In the first configuration (Configuration 1), the inboard MSIV on the affected line was assumed to fail open, and this line was assumed to leak at 100 scfh. The three intact lines were then assumed to leak at 100 scfh, 50 scfh, and 0 scfh to maximize flow rates through the lines, which in turn maximizes the activity release. At 20 minutes after the start of release the third safety-related and seismically-qualified isolation valves (just outboard of the outboard in the main steam lines

Elemental iodine retention efficiency is based on a comparison of deposition and resuspension rates from (Reference 17) and ~~is~~ was initially ~~conservatively~~ set at 50%. ~~All of the "unfiltered" iodine is conservatively assumed to be released in organic form.~~ ~~Additionally,~~ For conservatism, the main steam line aerosol removal efficiency (the ability of the steam lines to retain aerosol fission products) was slightly reduced in the analysis. This aerosol removal efficiency is equivalent to an increase in aerosol penetration of 10 percent. This was done to further increase the dose from the main steam line pathway.

¶. Similar to the aerosol removal efficiency reduction, the elemental iodine retention efficiency was also reduced by 10% for conservatism, to 45%.

15.6.5.5.1.2 Fission Product Transport in Drywell

The most limiting DBA, with respect to the offsite and control room radiological consequences, is considered a large-break LOCA as a result of a double guillotine pipe rupture in one of the four main steam lines upstream of the inboard MSIV. It is further conservatively assumed that all fission products are released directly to the drywell and leaked into the primary containment and into the main steam lines, bypassing the suppression pool. The analysis also assumes that at a point two hours after accident initiation (when the ECCS is assumed to be able to reach the core and reflood it) the fission products are homogeneously distributed between the drywell and the primary containment. The objective of this well mixed approach is to achieve an appropriate balance for the design of drywell leakage mitigative devices such as the MSIVs as well as containment leakage mitigative features such as the HEPA filters in the annulus exhaust gas treatment system.

As characterized in ~~<NUREG-1465>~~ Reference 19, the gap releases and the early in-vessel fission product releases terminate 2 hours after accident initiation. For the fission product releases to terminate, the reactor vessel would need to be reflooded. In lieu of evaluating all of the potential steaming rates due to various reflooding scenarios, the analysis assumes that a substantial amount of fission products will end up in the primary containment as well as in the drywell, and as such, exhaust

mitigative features such as the HEPA filters in the annulus ~~effluent~~ gas treatment system are designed to accommodate a significant portion of the source term. The 2-hour assumption for the homogeneous mixture of the source term between the drywell and the containment is used since it provides an appropriate balance, because the "worst 2 hours" are considered for the EAB radiological dose results, as opposed to simply the first 2 hours as was done when the TID source term was used.

The radiological consequences are dependent upon the drywell bypass leakage prior to the termination of fission product release at 2 hours. Because of this sensitivity, the analysis uses a steaming rate of an intact core without relocation to the lower head region, on the order of 3,000 cfm. For the period prior to 2 hours, the analysis conservatively does not credit steaming due to relocation, cooling from alternative water sources, or the release of hydrogen gas, all of which would provide a higher steaming rate and remove more of the fission products from the drywell region.

Elemental Iodine and

15.6.5.5.1.3 Aerosol Deposition ~~Within the Drywell~~

Activity released to the drywell as a result of the design basis loss-of-coolant-accident is initially airborne and can be removed from the atmosphere in one of four ways:

(1) Convection from the drywell to the containment

(2) Natural removal within the drywell (e.g., particulate sedimentation)

(3) Leakage into the broken steam line and through the MSIVs

(4) Leakage back into the reactor vessel and through the MSIVs

and containment

Elemental iodine removal is credited in the drywell volumes. Airborne elemental iodine is removed by deposition to the walls ~~in the drywell~~. This process is driven by the temperature differences between the surfaces and the atmosphere. The calculated removal constants are applied until a decontamination factor (OF) of 200 has been obtained. Aerosol removal in the drywell is modeled using the Power's removal model as given in NUREG/CR-6189 (Reference 20). The lower bound decontamination coefficient associated with the 10th percentile uncertainty was used for conservatism.

unsprayed regions of Containment (including

(DF) of up to

The leakage ~~contribution~~ is small by design; and therefore, the two principal mechanisms for depletion of activity in the drywell atmosphere (other than by radioactive decay) is convection from the drywell to the containment and natural removal within the drywell.

depletion due to MSIV

~~Convection is conservatively calculated (i.e., minimized) by taking into account only a very low estimate of the post-LOCA boiloff of residual water inside the reactor vessel following the start of core damage. This assumes that the ECCS is initially ineffective in preventing core damage. The boiloff is assumed to begin at the start of the fuel release phase of the accident, as defined in Section 5.2.1 of (Reference 18). The value calculated is 4.4 lbm/sec.~~

Following the fuel release phase of the accident, the restoration of ECCS (thus arresting further core damage) would quench the core debris, and results in a rapid sweep-out of the drywell into the containment as discussed in Section 5.2.3 of (Reference 18).

For the design basis analysis, a negotiated licensing basis was established for the transport of activity between the containment and the drywell. The negotiated basis in effect mixes activity between the regions and does not consider a sweep-out of the activity after two hours. The negotiated parameters are in <Table 15.6-12b>.

~~Natural removal of activity due to physical processes (i.e., other than by radioactive decay) can be associated with many effects, including sedimentation, diffusiophoresis, and thermophoresis. Only sedimentation processes (described in Section 5.2.3 and Appendix E of (Reference 18)) are credited in this analysis. The Polestar Applied Technology "STARNAUA" computer code (Reference 15) is used for the calculation of natural particulate removal in the drywell. The key input assumptions are given on <Table 15.6-12d> and the removal rates ("lambdas", as calculated by STARNAUA) are shown on <Table 15.6-12e>. These removal~~

~~rates are also assumed to apply to elemental iodine (see Section 5.2.3 of (Reference 18)). Note that the STARNAUA analysis considers flow out of the drywell and sedimentation simultaneously. In this way the removal rates (which improve with increasing particulate concentration) are not overestimated. The particulate release from the drywell (associated with the drywell-to-containment convection discussed above) becomes the input for the STARNAUA calculation for the sprayed region of the containment <Section 6.5.2.3> after being reduced by a factor of 2.44 to account for the fact that the sprayed region is only 41% of the containment free volume. Here again, the intent is to ensure that the particulate concentration (and therefore, the rate of particulate removal) in the sprayed region of the containment is not overestimated.~~

15.6.5.5.1.4 Containment Leakage Pathway

The primary containment consists of a drywell, a wetwell, and supporting systems to limit fission product leakage during and following the postulated LOCA with isolation of the containment boundary penetrations. The design basis leak rate of the primary containment is 0.2 volume percent per day. The analysis ~~conservatively assumes~~ the design basis leak rates ~~stays constant~~ for the ~~entire~~ duration of the accident (30 days).

reduces

at 24 hours as permitted by <Regulatory Guide 1.183>

remaining

to ~~0.69La~~

69 percent of the leak rates assumed during the first 24 hours

The secondary containment (shield building) which surrounds the primary containment will collect and retain fission product leakage from the primary containment and will release fission products to the environment in a controlled manner through the AEGTS. AEGTS will maintain the secondary containment pressure negative following a DBA by the time the gap release could migrate outside the containment structure. Therefore, if a short period of time exists post-LOCA when the annulus pressure is not negative, the dose calculations would not be affected.

Although the primary containment is enclosed by the secondary containment, there are systems that penetrate both the primary

~~No changes to this page. Provided for context.~~

containment and the shield building boundaries that could create potential pathways through which fission products in the primary containment could bypass the leakage collection and filtration systems associated with the shield building. The analysis conservatively assumes 10.08% of the primary containment leakage bypasses the secondary containment (the Technical Specifications limit bypass leakage to a lower limit).

The analysis assumes 89.92 percent of the primary containment leak rate goes into the secondary containment for its radiological consequence analysis. This leakage is collected in the shield building and processed through the AEGTS HEPA filters before being released into the environment. The remaining 10.08 percent of the primary containment leak rate is assumed to bypass the shield building and to be released directly to the environment for the entire duration of the postulated LOCA.

15.6.5.5.1.5 Annulus Exhaust Gas Treatment System

The AEGTS is an engineered safety features system and is designed to collect, process, and release the fission product leakage from the primary containment into the shield building. The AEGTS is a redundant system consisting of two 100 percent capacity subsystems. Each subsystem has a design capacity of 2000 cfm and consists of, among other things, a HEPA pre-filter, one 4-inch deep charcoal adsorber, and a HEPA post-filter. The system is designed to Seismic Category I standards and is located in a Seismic Category I structure.

The system is operated continuously during normal plant operation, and it maintains a slight negative pressure in the shield building. The analysis assumes a 99 percent removal efficiency for fission products in aerosol form for HEPA filters. The analysis however does not consider any fission product removal by the charcoal adsorbers in the AEGTS. The analysis also conservatively assumes that the entire 2000 cfm flow is discharged directly to the environment with no recirculation (holdup) of iodine in the annulus.

15.6.5.5.1.9 Control Room Habitability

Upon receipt of an ESF actuation system signal or high radiation, the control room Heating, Ventilation, and Air Conditioning (HVAC) system is designed to automatically switch to the emergency recirculation mode of operation (CRERS). The analysis conservatively assumes a 30-minute delay in actuation of the CRERS.

The CRERS is a redundant system and each subsystem has a design flow capacity of 30,000 cfm. The analysis uses a conservative recirculation flow rate of 27,000 cfm. Each subsystem consists of, among other things, a High-Efficiency Particulate Air (HEPA) filter, charcoal adsorbers, and a HEPA post-filter. The analysis also uses a conservative HEPA filter efficiency of ~~95~~ percent for aerosol particulate and ~~a 50~~ percent charcoal filter removal efficiency for iodine in elemental and organic forms.

an 80 99 ~~6000~~ 6600

During normal operation, the HVAC system is designed to pressurize the control room envelope with 45,000 cfm recirculation airflow and with 6,000 cfm outside makeup air. During an emergency, when the system operates in the emergency recirculation mode, the outside makeup air is isolated and the control room envelope is not pressurized relative to adjacent areas. To be conservative, the analysis uses ~~1,375~~ cfm unfiltered inleakage to the control room during the emergency recirculation mode ~~for the entire duration of the accident~~. The major parameters and assumptions used in the analysis are listed in <Table 15.6-14>.

first 30 minutes, followed by 1,375 cfm unfiltered inleakage in the

~~An exemption was granted by the NRC from the control room dose acceptance criterion of <10 CFR 50, Appendix A>, General Design Criteria 19, "Control Room." The exemption permits use of a 5 rem TEDE acceptance criteria in lieu of "5 rem whole body, or its equivalent to any part of the body," as currently stated in GDC 19 for the control room.~~

after 30 minutes

15.6.7 REFERENCES FOR SECTION 15.6

1. Moody, F. J., "Maximum Two-Phase Vessel Blowdown From Pipes," ASME Paper Number 65-WA/HT-1, March 15, 1965.

TABLE 15.6-12a

LOSS-OF-COOLANT ACCIDENT PARAMETERS AND ASSUMPTIONS
USED IN RADIOLOGICAL CONSEQUENCE CALCULATIONS
MAIN STEAM ISOLATION VALVE LEAKAGE PATHWAY

Parameter	Value
Reactor power (3758 MWt x 1.02)	$\boxed{3833}$ ~~3758~~ MWt
Drywell volume	2.765×10^{5} ft^3
~~Wetwell~~ Containment (excluding drywell) volume	1.165×10^{5} ft^3 $\boxed{6}$
Volume of one main steam line between MSIV's	146 ft^3
~~Volumetric flow rate, drywell to all main steam lines (total leakage)~~	~~298 cfh from t = 0 to t = 7484 seconds~~ ~~247 cfh from t = 7484 seconds to 30 days~~
~~Volumetric flow rate (maximum), one main steam line to environment~~	~~191 cfm~~

Volumetric flow rate, drywell to broken steam line	
0 to 7484 seconds	1.987 ft^3/min
7484 seconds to 24 hours	1.647 ft^3/min
24 hours to 30 days	1.371 ft^3/min
Volumetric flow rate, drywell to intact steam lines	
0 to 7484 seconds	2.98 ft^3/min
7484 seconds to 24 hours	2.47 ft^3/min
24 hours to 30 days	2.056 ft^3/min
Volumetric flow rate (maximum) in one main steam line, between the MSIV's, then to environment, t = 0 to 30 days	3.183 ft^3/min
Volumetric flow rate in intact main steam lines, between the MSIV's, then to environment, t = 0 to 30 days	4.775 ft^3/min

TABLE 15.6-12b

LOSS-OF-COOLANT ACCIDENT PARAMETERS AND ASSUMPTIONS
USED IN RADIOLOGICAL CONSEQUENCE CALCULATIONS
CONTAINMENT LEAKAGE PATHWAY

Parameter	Value
Reactor power (3758 x 1.02)	~~3758~~ 3833 MWt
Volume of sprayed region Containment	4.812x 10^5 ft^3
Volume of unsprayed region	6.842x 10^5 ft^3
~~Flow rate from drywell to unsprayed region~~	
~~0~~ 0.5 - 2 hours	3000 ft^3/min
2 hours - 30 days	~~2.765~~ 2.77 x 10^5 ft^3/min
Flow rate from unsprayed region to drywell	
0 - 2 hours	0 ft^3/min
2 hours - 30 days	~~2.765~~ 2.77 x 10^5 ft^3/min
Flow rate between drywell and sprayed region	0 ft^3/min
Flow rate from sprayed region to unsprayed region	71,400 ft^3/min
Flow rate from unsprayed region to sprayed region	71,400 ft^3/min
Containment leak rate to environment from sprayed region	
~~0 - 40 seconds~~	~~0.67 ft^3/min~~
40 seconds - ~~30 days~~ 24 hours	0.067~~0~~ ft^3/min
24 hours - 30 days	0.0462 ft^3/min
Spray removal rate for particulate	~~90~~ 10 percent uncertainty distribution
Spray fall height	~~53.2~~ 54.05 ft
Spray removal rate for elemental iodine (sprayed region only)	<Table 6.5-11>
Containment leak rate to environment from total unsprayed region	
~~0 - 40 seconds~~	~~1.34 ft^3/min~~
40 seconds - ~~30 days~~ 24 hours	0.135 ft^3/min
24 hours - 30 days	0.0932 ft^3/min
Containment leak rate to annulus from sprayed region	
~~0 - 40 seconds~~	~~0 ft^3/min~~
40 seconds - ~~30 days~~ 24 hours	~~0.603~~ 0.601 ft^3/min
24 hours - 30 days	0.415 ft^3/min
Containment leak rate to annulus from total unsprayed region	
~~0 - 40 seconds~~	~~0 ft^3/min~~
40 seconds - ~~30 days~~ 24 hours	1.20~~0~~ ft^3/min
24 hours - 30 days	0.828 ft^3/min
Annulus volume	1.96 x 10^5 ft^3
Flow rate from annulus to environment	2000 ft^3/min
Annulus exhaust gas treatment system filter efficiency	
particulate	99 percent
elemental and organic iodine	0
Volume of total unsprayed region (including drywell)	9.607 x 10^5 ft^3
0 - 0.5 hours	0 ft^3/min

Insert for Table 15.6-12b

Containment leak rate to environment from sprayed region	
~~0 to 40 sec~~	~~0.735 ft^3/min~~ e
40 sec to 24 hours	0.067 ft^3/min
24 hrs to 30 days	0.04621 ft^3/min e
Containment leak rate to environment from unsprayed region	
~~0 - 40 seconds~~	~~1.046 ft^3/min~~ e
40 seconds - 24 hours	0.135 ~~0.096~~ ft^3/min
24 hours - 30 days	~~0.0662~~ ft^3/min 0.0932
Containment leak rate to annulus from unsprayed region	
~~0 - 40 seconds~~	~~0 ft^3/min~~ e
40 seconds - 24 hours	1.20 ~~0.95~~ ft^3/min
24 hours - 30 days	~~0.656~~ ft^3/min 0.828
Containment leak rate to annulus from sprayed region	
~~0 - 40 seconds~~	~~0 ft^3/min~~ e
40 seconds - 24 hours	0.601 ~~0.668~~ ft^3/min
24 hours - 30 days	~~0.461~~ ft^3/min 0.415

total

This revised information was marked directly into Table 15.6-12b on page 15.6-59, thereby obviating the need for this Insert page

TABLE 15.6-12c

LOSS-OF-COOLANT ACCIDENT PARAMETERS AND ASSUMPTIONS
USED IN RADIOLOGICAL CONSEQUENCE CALCULATIONS
ENGINEERED SAFETY FEATURE (ESF) LEAKAGE PATHWAY

ECCS Leakage Model

Parameter	Value
Plant power (3758 MWt x 1.02)	~~3758~~ 3833 MWt
Release fractions and timing	As specified for BWR in <~~NUREG-1465~~ Regulatory Guide 1.183> (gap and early in-vessel iodine releases only)
Release location	Directly from core to suppression pool
Suppression pool water volume	114,379 ft^3
ECCS leak rate	
0 - 24 hours	15 gph
24 - 24.5 hours	15 gph and 50 gpm for 30 minutes
24.5 hours - 30 days	15 gph
Partition factor	10

TABLE 15.6-14

CONTROL ROOM MODEL

Parameter	Value
Volume	3.90 ~~3.44~~ x 10^5 ft^3
~~Flow rate - unfiltered inleakage~~	~~1375 ft^3/min~~
~~Flow rate - exhaust~~	~~1375 ft^3/min~~
Recirculation flow rate	
0 - 0.5 hour	0
0.5 hour - 30 days	2.7 x 10^4 ft^3/min
Recirculation filter efficiencies	
particulate	99 ~~95%~~
elemental and organic iodine	80 ~~50%~~
Flow rate - unfiltered inleakage	
0 - 0.5 hour	6600 ~~6000~~ ft^3/min
0.5 hour - 30 days	1375 ft^3/min
Flow rate - exhaust	
0 - 0.5 hour	4800 ft^3/min
0.5 hour - 30 days	1375 ft^3/min
Occupancy Factors	
0 - 24 hours	1.0
24 - 96 hours	0.6
96 - 720 hours	0.4

TABLE 15.6-15

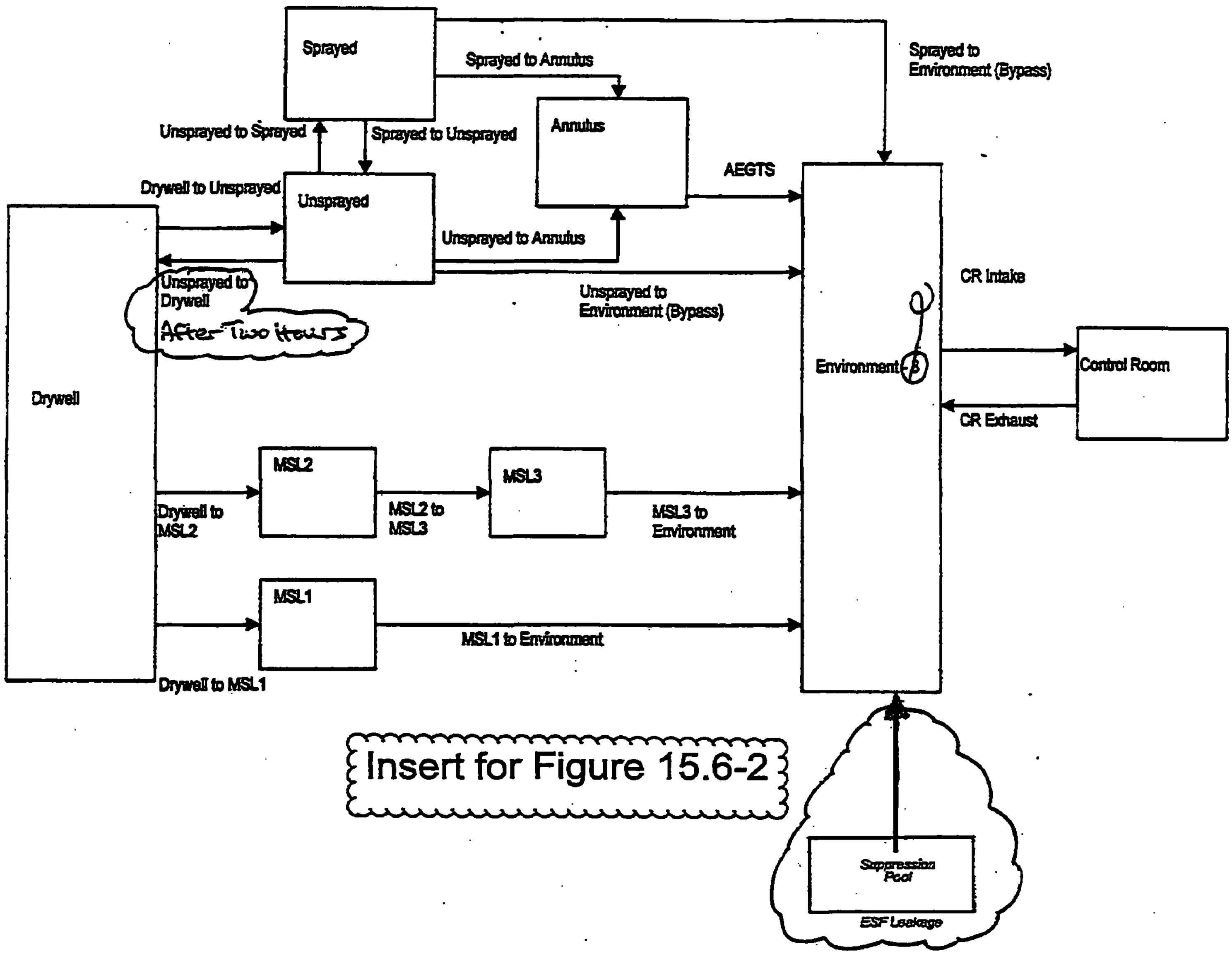
LOSS-OF-COOLANT ACCIDENT
(DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS(1) ~~MA~~ STET Note (1)

		Dose (Expressed as TEDE, Rem)	Licensing Basis Limit (TEDE, Rem)
1.	Offsite Doses		
	Exclusion area (863 Meters)	21.2 ~~6.5~~ 6.9	25
	Low population zone (4,002 Meters)	~~10.1~~ ~~2.8~~ 3.0	25
2.	Control Room Doses (0-30 days)	~~4.3~~	5

NOTE:

(1) GNF2 Fuel, 1350 Effective Full Power Days (EFPD)
~~The above radiological effects have been updated to reflect the scaled increases associated with Power Uprate to 3,758 MWt.~~

Fission Product Transport Model
(MSIV and Containment Leakage Pathways)



Insert for Figure 15.6-2