Florida Power
CORPORATION

September 13, 1989
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Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Crystal River Unit 3
Docket No 50-302
Operating License No. DPR-72
Revision to FSAR Radiological Consequences

Dear Sir:

As part of the Configuration Management Program and the Technical Specification Improvement Program, Florida Power Corporation (FPC) has been re-examining the assumptions used in the FSAR Chapter 14 analyses. There have been inconsistencies identified between the Technical Specifications and certain FSAR accident assumptions. FPC is evaluating the inconsistencies to determine if they impact the FSAR results. In some cases, additional analyses have been performed to quantify the impact.

To ensure that CR-3 offsite doses remain within 10 CFR 100 limits, FPC has re-evaluated the off-site radiological consequences of the Loss-of-Coolant Accident (LOCA) and the Makeup System Letdown Line Failure Accident (LLFA) to eliminate the credit for the Auxiliary Building Ventilation (ABV) System. The ABV System contains charcoal filters which reduce the iodine dose. This system is non-safety related and is not provided with emergency power. Without emergency power, the ABV System should not be assumed to be available to provide iodine filtration.

The analyses for both accidents have assumed the ABV System is not available. The LOCA analysis used the same methodology for fission product release as that used to evaluate Crystal River Unit 3 (CR-3) control room habitability submitted in FPC's June 30, 1987 letter. The control room habitability Safety

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Evaluation Report (SER) was issued by the NRC letter dated May 25, 1989. This fission product model uses Regulatory Guide 1.4 for the upper bound assumptions. The analyses project small dose consequence increases above the values previously reported in the FSAR. A comparison of the doses is presented in the attached tables.

LOCA

FPC's re-evaluation for a hypothetical or design basis LOCA produces thyroid doses and whole body doses at the exclusion area boundary (EAB) and the low population zone (LPZ) which are in close agreement with the values described in Supplement No. 3 to the Safety Evaluation Report for CR-3 dated December 30, 1976. SER Supplement No. 3 states "The potential doses tabulated below are, therefore, conservatively derived and are well below the guideline values specified in 10 CFR Part 100." FPC has interpreted this SER statement to mean that no "unreviewed safety question" within the meaning of 10 CFR 50.59 is present due to FPC's re-evaluation. 10 CFR 100 is considered to be the acceptance limit for protection of the public health and safety.

The 1976 SER statements are not detailed enough for FPC to judge exactly how the Regulatory Guide 1.4 methodology and the 1975 CR-3 meteorological program data were used by the NRC. However, the results obtained by FPC in its re-evaluation are so close in agreement with the NRC results that a similar methodology must have been used by the NRC in 1976 for its evaluation of the hypothetical design basis LOCA. Furthermore, to ensure conservatism, FPC has used more recent NRC guidance for its re-evaluation.

As FPC noted in the CR-3 control room habitability submittal, CR-3 is not a Standard Review Plan (SRP) plant. However, to further ensure that conservative radiological consequences were produced, FPC used SRP 15.6.5, "Loss-of-Coolant Accidents Resulting From Spectrum Of Postulated Breaks Within The Reactor Coolant Pressure Boundary" as a guideline for the parameters and assumptions in the re-analysis. SRP 6.5.2, "Containment Spray As A Fission Product Cleanup System" was also used as a guideline.

LLFA

The revised LLFA results show an increase in the offsite thyroid doses which is in proportion with the decrease in the assumed ABV System charcoal filter efficiency (90% vs 0%). The whole body doses increased by 10 mrem at both the EAB and the LPZ.

The revised accident doses are much less than the limits specified by 10 CFR 100. The 1976 SER for CR-3 does not address this accident in the list of events reviewed by the staff, therefore, 10 CFR 100 is considered to be the acceptance limit for protection of the public health and safety and no "unreviewed safety question" within the meaning of 10CFR50.59 is present due to FPC's re-evaluation.

The regulatory process required by 10 CFR 50.59 is under review by the NRC and the industry. NSAC/125, "Guidelines for 10 CFR 50.59 Safety Evaluations" is being considered by the staff for endorsement. The increased dose consequences are within the licensing basis for CR-3, i.e., the 1976 SER. As long as the calculated doses remain less than the 10 CFR 100 limits, NRC review is not necessary before FPC revises the FSAR. This position is consistent with the NRC comments on NSAC/125. Until the guidance for conducting 10 CFR 50.59 reviews is formally endorsed, FPC is providing the NRC with the proposed FSAR changes for information. These changes establish a revised licensing basis for the CR-3 LOCA and the LLFA radiological consequences while recognizing that the margin between the projected releases and the design basis limits established by 10 CFR 100 have not been significantly reduced. Included are the revised FSAR Sections 14.2.2.6, 14.2.2.5.10, 14.2.2.7, 6.2.2.1., and notations of deleted pages. FPC will revise the FSAR with these changes no later than July 1, 1990.

Sincerely,

Rolf C. Widell

Rolf C. Widell, Director
Nuclear Operations Site Support

RCW/JWT/sdr

Attachments

xc: Regional Administrator, Region II
Senior Resident Inspector

COMPARISON OF LOCA RADIOLOGICAL CONSEQUENCES (Rem)

	FSAR Table 14-57	1976 SER Doses	Revised Doses
EAB (2-hr)			
Thyroid	63.1	133	134.2
Whole Body	1.55	3	2.31
LPZ (30-day)			
Thyroid	9.11	25	27.1
Whole Body	0.29	<1	0.42

COMPARISON OF LETDOWN LINE FAILURE RADIOLOGICAL CONSEQUENCES (Rem)

	FSAR Table 14-43	Revised Doses
EAB (2-hr)		
Thyroid	0.115	1.15
Whole Body	0.066	0.067
LPZ (30-day)		
Thyroid	0.0101	0.101
Whole Body	0.0058	0.0059

An analysis of the minimum containment back pressure, including the effect of the Purge System, is provided in a report from Florida Power Corporation (G. C. Moore) to the NRC (R. W. Reid), transmitted by letter dated July 11, 1980. The analysis utilizes a CONTEMPT model which employs the basis approach listed in BAW-10103A, Revision 3, yet is specific to Crystal River Unit 3 Reactor Building. It concludes that the generic 177-FA lowered loop RB pressure evaluation is conservative with respect to Crystal River Unit 3 and the ECCS conformance to 10CFR50.56 regardless of Purge System Operation.

INSERT NEW 14.2.2.5.10

14.2.2.5.10 ~~Environmental Analysis of Loss-of-Coolant Accidents~~

The analyses in the preceding Sections have demonstrated that ECCS injection will meet the Final Acceptance Criteria for LOCAs resulting from RCS ruptures ranging in size from small leaks to the complete severance of the hot leg piping. The environmental consequences from a LOCA are conservatively analyzed by assuming the activity associated with the gap of all fuel rods is released.

The activity released is shown in Table 14-52. 50% of the iodine released is assumed to plate out, and the other half is assumed to remain in the reactor building atmosphere where it is available for leakage.

The alkaline solution in the reactor building spray reduces the airborne iodine as described in Appendix A to this Chapter. 2% of the iodine available for leakage has been conservatively assumed to be present as organic iodine; the remaining iodine is present as elemental iodine. Specific parameters used and the calculated spray effectiveness are given in Table 14-53.

The reactor building pressure history for the design basis accident is shown in Figure 14-43. Although the reactor building leakage rate will decrease as the pressure decays, the leakage is assumed to remain constant at the design leak rate for the first 24 hours. Thereafter, since the reactor building will have returned to nearly atmospheric pressure, the rate is assumed to be reduced to one-half the design leak rate and remain at this value for the duration of the accident.

The atmospheric dispersion characteristics of the site are described in Section 2.3.3. The site dispersion factors for the duration of the accident are listed in Table 2-11. A breathing rate of 3.47E-4 m^3/s is assumed for the two hour exposure. For the 24 hour exposure, a breathing rate of 3.47E-4 m^3/s is assumed for the first 8 hours, and a rate of 1.74E-4 m^3/s is assumed for the remaining 16 hours. For the 30-day exposure, a breathing rate of 2.32E-4 m^3/s is assumed. The total integrated thyroid and whole body doses at the exclusion distance and the low population distance are summarized in Table 14-54.

14.2.2.5.11 Reactor Building Subcompartments Pressure Response

The results of an analysis of the reactor cavity and the steam generator compartments for the pressure response considering a homogeneous steam-water-air mixture with appropriate correlations for sonic flow through the gaps

14.2.2.5.10 Radiological Consequences Of A Loss Of Coolant Accident

Loss of Coolant Accidents (LOCA) are postulated accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the Reactor Coolant Makeup System, from piping breaks in the reactor coolant pressure boundary. The LOCA is one of the postulated accidents used to evaluate the adequacy of the plant's structures, systems, and components with respect to public health and safety.

Multiple barriers, engineered safeguards, and administrative procedures are provided to prevent and minimize the consequences of a LOCA. Regardless of these safety provisions, it is postulated that a Design Basis LOCA of the magnitude assumed in Regulatory Guide (RG) 1.4, Rev 2 occurs. In order for a RG 1.4 fission product release to occur, fuel melting is required. Since the Emergency Core Cooling System (via high and low pressure safety injection and core flooding systems) is provided to prevent this occurrence, a more realistic analyses of a LOCA is also presented based on a reduced source term, i.e. fission product release associated with all the activity in the fuel rod gap of the core.

14.2.2.5.10.1 Acceptance Criteria

The acceptance criteria for the radiological consequences of the LOCA are that the offsite radiation exposures are within 10 CFR 100 limits, Paragraph 11. Specifically, the 2-hour dose at the exclusion area boundary (EAB) and 30 day dose at the low population zone (LPZ) are limited to 300 rem (thyroid) and 25 rem (whole body). In addition, 10 CFR 50, Appendix A, Criterion 19 requires that adequate radiation protection provision be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body (30 rem, thyroid and 30 rem beta, skin), for the duration of the accident.

14.2.2.5.10.2 Identification Of Causes And Accident Description

The Loss of Coolant Accident is postulated as the principal design bases event for assessing the potential risk to public health and safety. As a result of the LOCA, a fraction of the plant's fission product inventory is assumed to be released from the fuel assemblies into the Reactor Coolant System and later into the Reactor Building via the break in the RC System pressure boundary. High Reactor Building pressure signals from the Engineered Safeguards Actuation System (ESAS) isolates (4 psig) the Control Complex putting it into a recirculation mode of operation and initiates (30 psig) the operation of the RB Spray System. The Control Complex Emergency Fans and Charcoal Filters are manually placed in service by the operator.

The fission product inventory in the Reactor Building is reduced by radioactive decay and the action of RB Spray System as discussed in Section 6.2.2.1.1. This radioactivity is assumed to leak from the containment to the environment at a constant rate of 0.25% per day for the first 24 hours after the accident and at one-half this rate (0.125%/day) thereafter.

In the Design Basis accident, it is also postulated that fission products are released to the environment via recirculation loop leakage of engineered safety features components located outside the Reactor Building. Both operational leakage (a value of 4510 cc/hr was assumed which is twice the expected leakage given in Table 6-11) and that associated with the post-LOCA failure of a passive component (50 gpm leak occurs at 24 hrs into the event and lasts for 30 min.) is assumed. The activity released from this source is collected by the Auxiliary Building Ventilation (ABV) system and is discharged to the environment via the plant vent. Since this system would not be operative during a loss of offsite power occurrence and is not powered by the emergency diesel supply, credit for the operation of the system's charcoal filters is not assumed.

The released fission products (iodines and noble gases) are dispersed in the atmosphere with no correction made for depletion of the effluent plume of radioactive iodine due to deposition on the ground or for the radioactive decay of fission products in transit.

The offsite radiological exposure to individuals located at the exclusion and low population zones results from inhalation of radioactive iodines (thyroid dose) and immersion in the released radioactive cloud (whole body dose). The radiological exposure to operators in the control room result from (1) direct radiation from the released radioactive cloud (2) direct radiation exposure from the Reactor Building and (3) exposure to radioactive materials which leak into the control room from the radioactive cloud in the atmosphere. Direct radiation exposure to the control room operation is minimized by concrete shielding of the Reactor Building and Control Complex. Infiltration of radioactive materials into the Control Complex is minimized by the low leakage construction features of the Control Complex. The Control Complex Ventilation System is designed for zone isolation with filtered recirculated air emergency mode of operation.

14.2.2.5.10.3 Methods of Analysis

Two methods of analysis are provided in evaluating the radiological consequences of a Loss of Coolant Accident: (1) Design Basis and (2) Realistic Basis. The Design Basis method utilizes upper bound assumptions contained in Regulatory Guide 1.4 while the Realistic Basis method assumptions were made to ensure the results are conservative, but more realistic. A summary of the parameters and assumptions used in assessing the radiological consequences of the LOCA for both methods are presented in Table 14-52. The differences in the methods of analysis are in the assumed post-LOCA radiation source terms, atmospheric dispersion, and control room inleakage parameters as noted in Table 14-52.

The Design Basis method is based on RG 1.4 core inventory releases plus an additional post-LOCA activity release due to recirculation system leakage outside of containment. In the Realistic Basis, the radiation source term is limited to the core activity inventory associated with the fuel rod gap as presented in Table 14-53. The gap activity was evaluated with a digital computer code, BURPE (Ref. 21), based upon the fission product escape rate coefficients determined by ANL 5800 (Ref. 22).

The offsite atmospheric dispersion factors used in the Design Basis analysis are based upon the short term accident diffusion models presented in Section 2.3.4. In the Realistic Basis, the offsite dispersion factors are based on the 22-1/2 degree sector with the highest annual average value. The Control Complex dispersion factors for both methods are based on a 5th percentile X/Q value associated with a 1.2 meter/sec wind speed including credit for turbulent mixing within the building wake cavity. For periods greater than 8 hours, credit has been taken for the reduction in this value due to post accident control room occupancy, wind speed and wind direction persistence factors as recommended in Murphy-Campe (Ref. 26).

The total post accident leakage into the Control Complex was calculated to be 236 cfm via (1) penetrations [approximately 0 cfm], (2) door seals [5 cfm], (3) ingress/egress [10 cfm], and (4) dampers [191 cfm filtered path and 30 cfm unfiltered paths]. However, for conservatism, the total inleakage is assumed in both methods of analysis to be equal to 0.06 volume changes per hour (355 cfm) based upon a Type B Control Room as defined in RG 1.78. For the Realistic Basis, it is assumed that 191 cfm of inleakage occurs via filtered pathways. The Design Basis assumes only 70 cfm of the total inleakage to be filtered.

The activity flow path models utilized in the analyses for Reactor Building Leakage, Recirculation Loop Leakage, and Control Complex Inleakage is given in Figures 14-65, 14-66 and 14-67, respectively. Both the Design Basis and Realistic Basis analyses are based upon the assumption that the RB Spray System is functioning in a degraded (worst case) mode of operation, i.e. spray pump failure.

The design basis parameters listed in Table 14-52 were utilized as input to the TACT-III (Ref. 27) computer program to compute the offsite and unprotected control room radiation exposures. The protected control room operator dose due to inleakage of radioactive materials were calculated based upon the use of an iodine dose protection factor and a whole body dose geometry factor described by Murphy-Campe. The analytical model used to calculate the direct dose contribution from the radioactive cloud in the atmosphere is given in RG 1.4. The model was adjusted for the reduction in dose due to the control room shielding.

The direct whole body dose from the Reactor Building was calculated based on a cylindrical radiation source model and corrected for the reduction in dose due to Reactor Building concrete shielding (3.5 ft), Control Complex (2 ft) and a minimum source-receptor distance of 48 feet.

The INHEC (Ref. 28) computer code was used to compute the Realistic Basis offsite and control room doses directly. The associated assumptions and parameters utilized as input to this code are also listed in Table 14-52.

The FPC letter dated June 30, 1987 submitted the CR-3 Control Room Habitability Evaluation Report. The NRC letter dated May 25, 1989 transmitted the SER which concluded that the design of the CR-3 control room habitability system was adequate.

14.2.2.5.10.4 Radiological Consequences

The offsite and control room radiation doses, resulting from both the Design Basis and Realistic Basis analyses of the Loss of Coolant Accident, are presented in Table 14-54. In both cases, the post accident offsite and control room dose consequences satisfy the requirements of 10 CFR 100 and 10 CFR 50, Appendix A, GDC 19, respectively.

In addition, the effect of a 10 minute delay in the control room operator manually placing the control room emergency fans and filters into service was evaluated and resulted in approximately a 6% increase in the calculated control room operator thyroid dose (from 26.5 to 28 rem), with no change in whole body doses.

TABLE 14-52

RADIOLOGICAL CONSEQUENCES OF A
LOSS OF COOLANT ACCIDENT

PARAMETER	ASSUMPTIONS Realistic Basis Analysis	Design Basis Analysis
Source Terms		
Core Thermal Power Rating, MWt	2595	2595
Activity Released To RB:		
Core Inventory:		
Iodine	N/A	50%
Noble Gases	N/A	100%
Gap Inventory	100%	N/A
Iodine Reduction Factor Due To Plateout In RB	22	N/A
Iodine Species Breakdown:		
Elemental	91%	91%
Organic	4%	4%
Particulate	5%	5%
Iodine Core Inventory Released To RB Sump	N/A	50%
Reactor Building		
Free Volume, ft^3	2,000,000	2,000,000
- Sprayed Volume, ft^3	1,304,000	1,304,000
- Unsprayed Volume, ft^3	696,000	696,000
- Air Turnover Between Sprayed And Unsprayed Volumes	4800% Of Unsprayed Volume Per Day	4800% Of Unsprayed Volume Per Day
Leakage Rate, %/Day		
0-1 Day	0.25	0.25
1-30 Days	0.125	0.125
Sump Liquid Volume, gal Post LOCA	N/A	490,182
Shield Wall Concrete Thickness, ft	3.5	3.5

TABLE 14-52 (CONTINUED)

RADIOLOGICAL CONSEQUENCES OF A
LOSS OF COOLANT ACCIDENT

PARAMETER	ASSUMPTIONS Realistic Basis Analysis	Design Basis Analysis
Recirculation Loop Leakage		
Operational, cc/hr	N/A	4510
Passive Component Failure, gpm (For 30 Min. Starting 24 Hours After The Accident)	N/A	50
Fraction Flashing To Steam, %	N/A	10
RB Spray System		
Spray System Actuation Time, sec	71	71
Spray Additive Concentration (Wt. % Of NaOH)	6	6
Flow Rate, gpm	1500	1500
Time To Reach pH = 8.5, min	9	9
Spray Removal Constants		
Elemental Iodine (Lambda$_e$), hr-1		
0-71 Sec.	0	0
71 Sec.-9 Min.	2.91	2.91
9 Min.-30 Days	16.58	16.58
Particulate Iodine (Lambda$_p$), hr-1		
0-71 Sec.	0	0
71 Sec.-30 Days	0.30	0.30
Maximum DF For Elemental Iodine By Sprays	170.4	170.4

TABLE 14-52 (CONTINUED)

RADIOLOGICAL CONSEQUENCES OF A
LOSS OF COOLANT ACCIDENT

PARAMETER	ASSUMPTIONS	
	Realistic Basis Analysis	Design Basis Analysis
Control Complex		
Free Volume, ft^3	355,311	355,311
Infiltration Rate (Total), cfm	355	355
Filtered In-Leakage	191	70
Unfiltered In-Leakage	164	285
Filtered Recirculation Flow Rate, cfm	43,500	43,500
Recirculation Charcoal Filter Efficiency, %	95	95
Environmental		
Atmospheric Dispersion:		
Offsite X/Q Values (sec/m^3)		
Exclusion (0-2 Hours)	2.56E-6	1.6E-4
Low Population Zone	3.10E-7	
0-8 Hours		1.4E-5
8-24 Hours		1.5E-6
1-4 Days		7.7E-7
4-30 Days		4.5E-7
Control Complex X/Q Values (sec/m^3)		
0-8 Hours	9.00E-4	9.00E-4
8-24 Hours	5.31E-4	5.31E-4
1-4 Days	2.07E-4	2.07E-4
4-30 Days	5.94E-4	5.94E-4
Offsite Breathing Rate (m^3/sec)		
0-8 Hours	3.47E-4	3.47E-4
8-24 Hours	1.75E-4	1.75E-4
1-30 Days	2.32E-4	2.32E-4
Control Complex Operator Breathing Rate (m^3/sec) (0-30 Days)	3.47E-4	3.47E-4

TABLE 14-53

POST LOCA GAP ACTIVITY
RELEASE INTO THE REACTOR BUILDING

Isotope	Activity, Ci
Noble Gases:	
Kr-83m	8.96E+3
Kr-85m	4.97E+4
Kr-85	4.37E+5
Kr-87	2.70E+4
Kr-88	8.86E+4
Xe-131m	8.13E+4
Xe-133m	9.50E+4
Xe-133	8.52E+6
Xe-135m	2.76E+4
Xe-135	3.44E+4
Iodines:	
I-131	6.55E+5
I-132	9.37E+4
I-133	1.41E+5
I-134	8.81E+3
I-135	4.47E+4

TABLE 14-54

OFFSITE AND CONTROL ROOM DOSES FOR
A LOSS OF COOLANT ACCIDENT

DOSE TYPE	REALISTIC BASIS ANALYSIS (Rem)	DESIGN BASIS ANALYSIS (Rem)
THYROID:		
Exclusion Boundary	1.6	134.2
Low Population Zone	0.2	27.1
Control Room	0.8	26.5
WHOLE BODY GAMMA:		
Exclusion Boundary	0.0022	2.31
Low Population Zone	0.00027	0.42
Control Room	0.04	1.88
WHOLE BODY BETA:		
Control Room	0.6	17.7

Add these references to page 14-82

26. "Nuclear Power Plant Control Room Ventilation System Design For Meeting General Criterion 19," K. G. Murphy And K. M. Campe, USAEC, 13th AEC Air Cleaning Conference, August 1974.

27. NUREG/CR-3287, "A Guide For The TACT III Computer Code", USNRC, May 1983.

28. GAI-TR-101P-A, Topical Report, "Computation Of Radiological Consequences Using the INHEC Computer Program, March 1976.

ACTIVITY FLOW PATH MODEL
REACTOR BUILDING LEAKAGE
HYPOTHETICAL LOCA

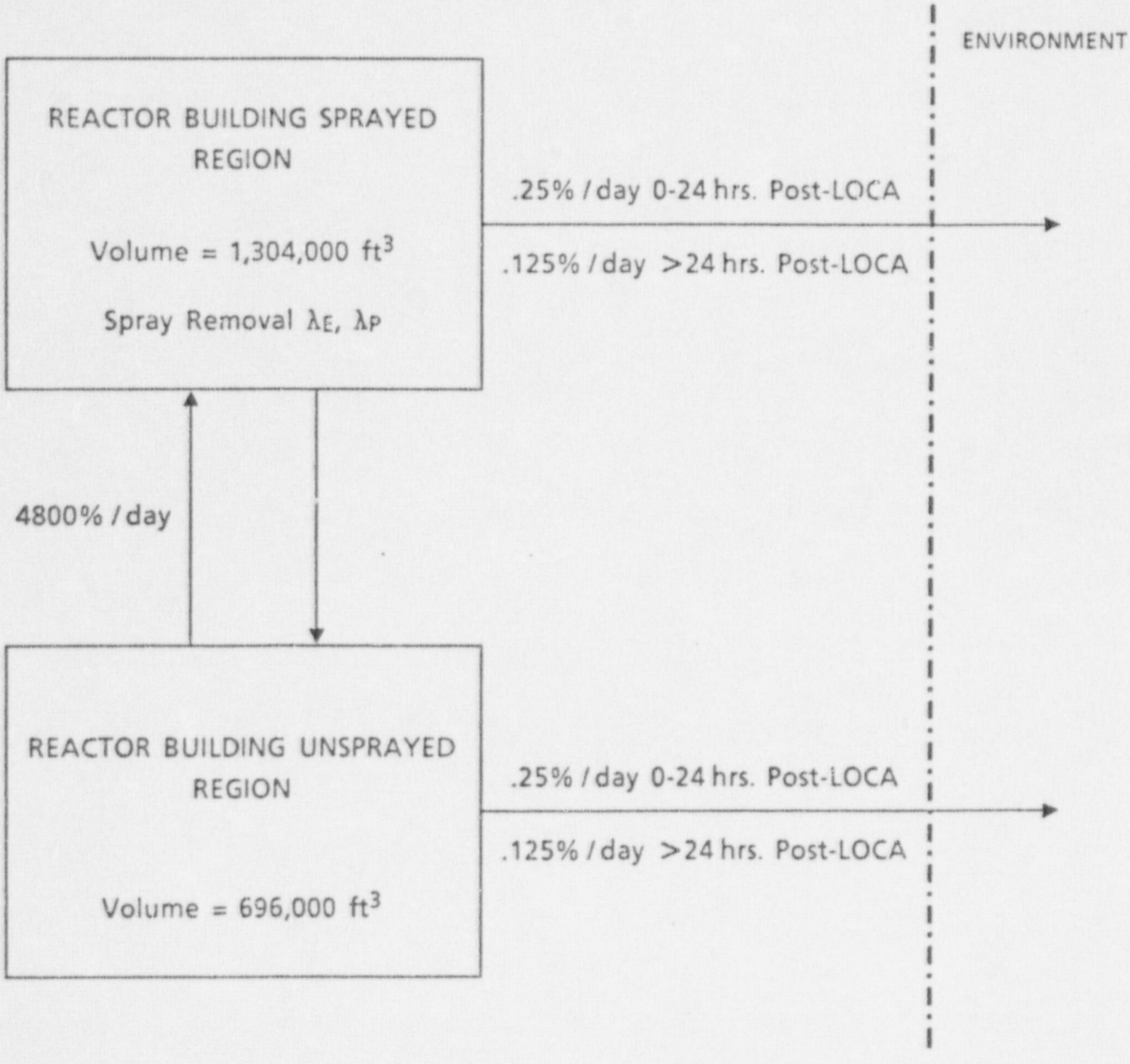


FIGURE 14-65

ACTIVITY FLOW PATH MODEL
RECIRCULATION LOOP LEAKAGE
HYPOTHETICAL LOCA

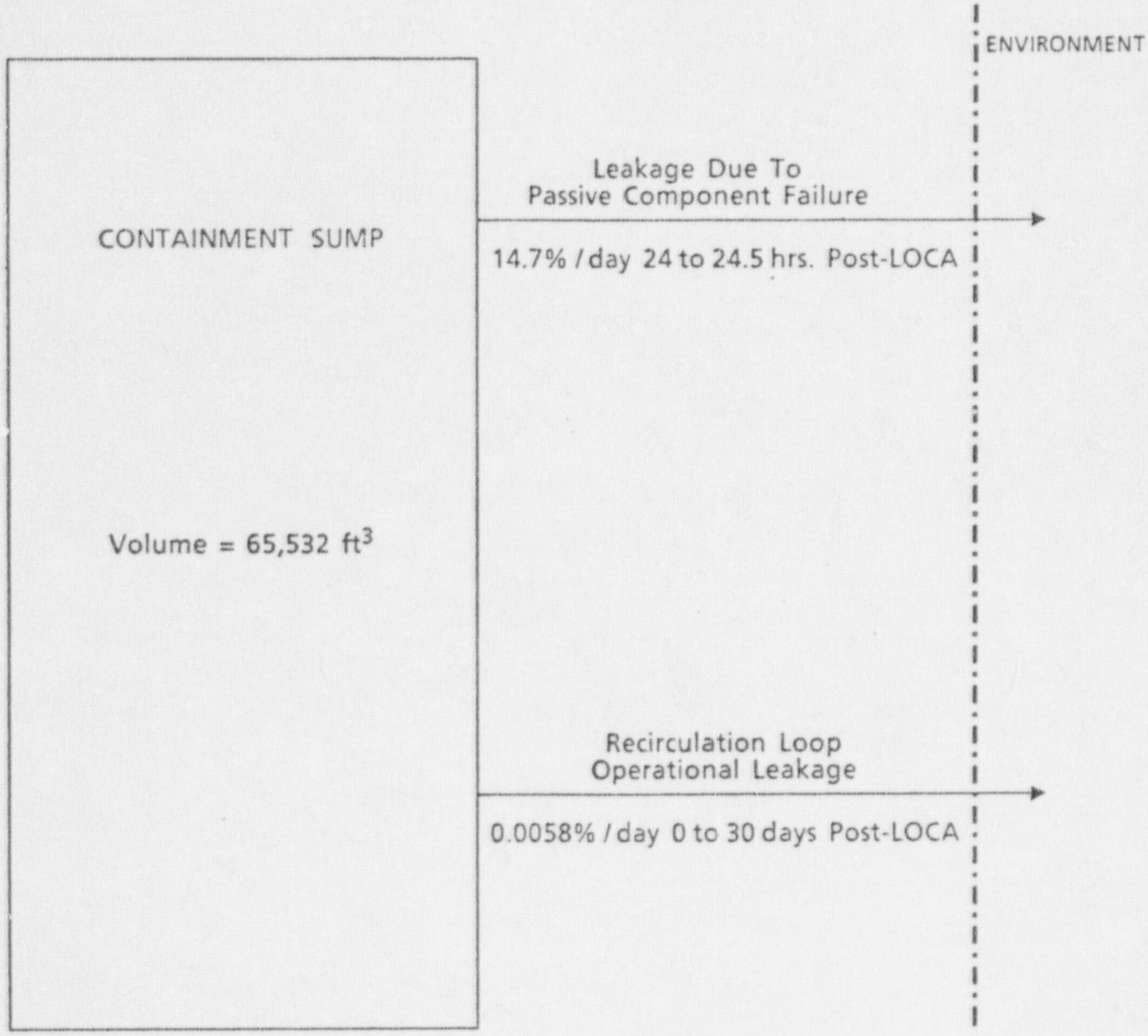
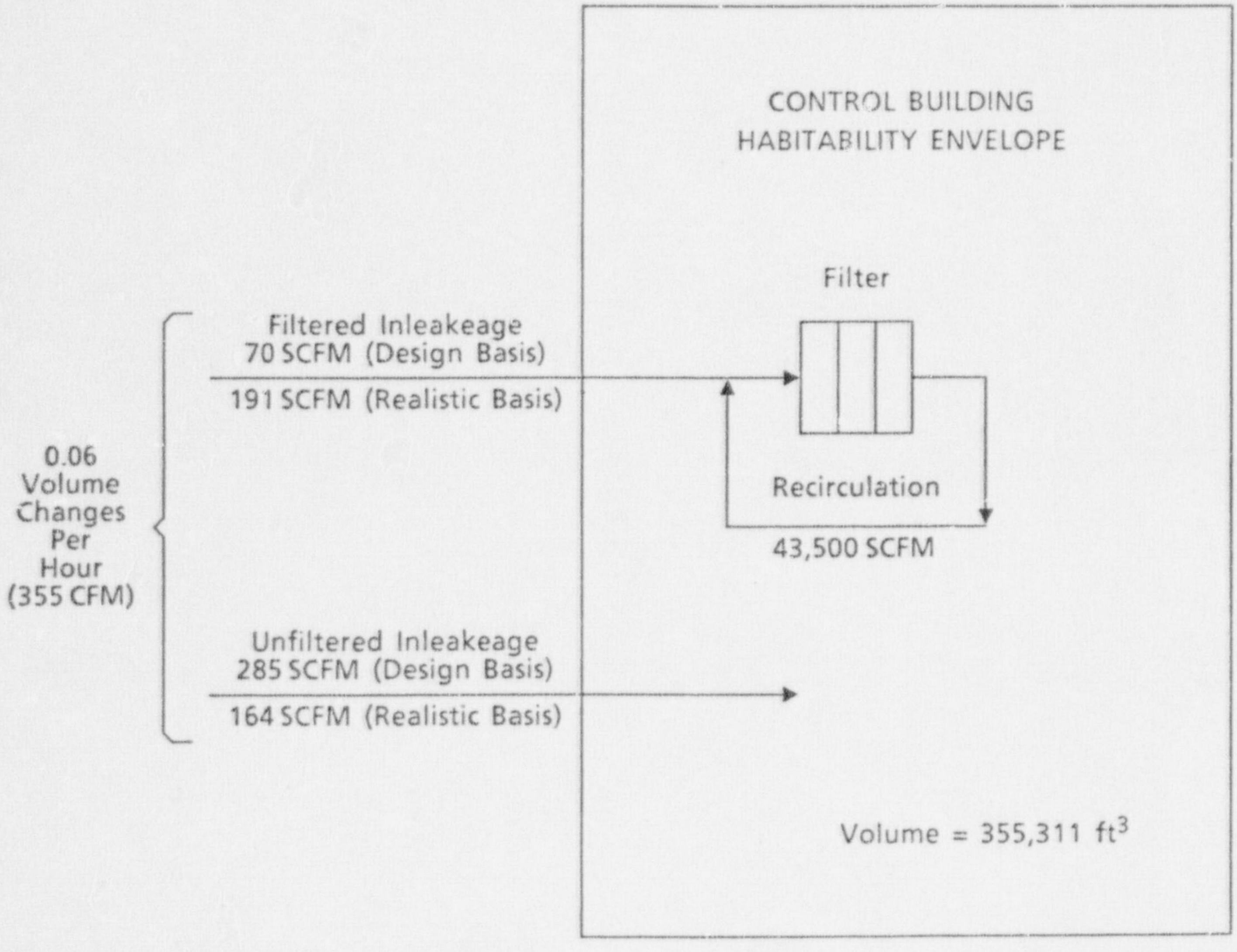


FIGURE 14-66

ACTIVITY FLOW PATH MODEL
CONTROL COMPLEX INLEAKAGE
HYPOTHETICAL LOCA



Ventilation System Mode Of Operation: Zone Isolation With Filtered Recirculating Air.

FIGURE 14-67

The radiological consequences of this accident are discussed in Section 14.2.2.5.10.

14.2.2.7 Maximum Hypothetical Accident

~~14.2.2.7.1 Identification of Accident~~

The Maximum Hypothetical Accident (MHA) analysis postulates a failure in the reactor coolant boundary in which fission product activity is assumed to accumulate in the reactor building atmosphere where it is available for leakage to the environment. Due to fuel cladding failure and primary system rupture, the accumulated containment inventory consists of the maximum activity from the fuel and the maximum equilibrium activity of the reactor coolant resulting from reactor operation at the design power for a sufficiently long period of time. Assumptions for fission product releases to the reactor building are assumed at a level that could result only from melting of the core; however, even in the event of a LOCA, no significant core melting would occur, since core meltdown would require a multitude of mechanical failures in safety-related systems and components, which are designed to prevent such an occurrence. Nevertheless, to assure that the operation of CR-3 does not present any undue hazard to the general public, based on fuel cladding failure and primary system rupture, an accident involving a gross release of fission products is evaluated -- 100% of the noble gases, 50% of the halogens (including iodine), and 1% of all other fission products (solids), as stipulated by TID-14844. Gases are assumed to be released through the reactor containment building immediately into the atmosphere surrounding the plant. No retention of noble gases is assumed. Only 50% of the iodine releases to containment are assumed to plate out, allowing as much as 25% of the core iodine to be released into the atmosphere. Iodine and noble gas releases available for leakage are listed in Tables 14-55 and 14-45, respectively.

Even without engineered safety features, the concentration of radionuclides in a containment atmosphere would be depleted by the natural processes of iodine plate out and radioactive decay. Engineered iodine removal mechanisms affecting fission product activity releases to the environment include washout with containment sprays and removal by charcoal filters.

14.2.2.7.2 Environmental Analysis and Results

Thyroid and whole body dose calculational methods model the minimum safety operation of engineered safeguard systems for removing airborne iodine, i.e. only one out of two building spray pumps and only one out of three reactor building air cooling units are assumed in operation. Other than activity releases, parameters for the MHA analysis are the same as those assumed for the LOCA analysis in Section 14.2.2.5.5. Thyroid doses are computed using the average iodine inventory (see Table 11-2), the atmospheric diffusion factor (see Section 2.3), the breathing rate and the containment leakage rate (see Section 5.2.1.1). Within 1 minute after the accident, isolation of the reactor building has been completed and leakage has been terminated, except for the design containment leak rate, by the reactor building isolation and cooling functions of the ESAS. Spray removal coefficients, decontamination factors and iodine source fractions are based on the sodium hydroxide spray solution. Whole body doses are based on iodine and noble gas inventories, the atmospheric diffusion factor, the containment leakage rate and beta-gamma energies of isotopes. The resulting doses are summarized in Table 14-57, and

~~are less than the 10CFR100 guideline values of 300 rem for thyroid doses and 25 rem for whole body doses.~~

14.2.2.7.3 Inhalation Dose to Reactor Operators in the Control Room

In the event of a LOCA, the ES Reactor Building 4 psig isolation signal would automatically close the control complex outside air intake (AHD-1) and atmospheric relief to outside discharge dampers (AHD-2) and open return dampers (AHD-3), thus placing the system in a recirculation mode through the normal path. In this mode of operation, the controlled access area is isolated from the control room and the remaining areas of the control complex above the 95 ft. elevation. Upon receipt of a toxic gas signal (chlorine or sulfur dioxide gases), the dampers are positioned as described for the ES signal. Upon receipt of a high radiation signal, the dampers are positioned as described for the ES signal. In addition, both the control complex normal supply fans (AHF-17A and AHF-17B) and the control complex return fans (AHF-19A and AHF-19B) are automatically stopped. The operator is required to manually change the selector switch from normal to emergency, which will open the absolute and charcoal filter damper, close the filter bypass valve and start one of the two control complex emergency supply fans (AHF-18A and AHF-18B). This fully places the system in emergency mode. All air is recirculated through the emergency filter bank as described in Section 9.7. The return air and minimum outside air, required for room pressurization, is directed through the absolute and charcoal filter before entering the coils and fan for return to the conditioned space.

The MHA assumptions presented in Section 14.2.2.7.1 apply in the calculation of the thyroid dose to the reactor operator in the control room. When the I-131 dose equivalent concentration reaches 1E-8 microCi/cc, the normal fans are tripped. The ventilation system is regulated so that a positive pressure would be maintained in the control room to ensure that all fresh air would enter through the filters. The in-leakage of outside air past the control complex isolation dampers is conservatively assumed to be 400 scfm, or approximately 1% of the total recirculation flow of 43,500 scfm. The 90% efficient emergency filters remove iodine and other particles during recirculation and fresh air changes. Fresh air change rates are dependent upon the outside air flowrate, the recirculation rates, the total air volume being recirculated of 243,000 ft^3, the number of men in the control room, and their breathing rate. For four shifts during the 30-day period of the accident, an individual would spend approximately 7.5 days in the control room. The halogens in the control room were assumed to be at equilibrium concentrations throughout the duration of the accident. The atmospheric dispersion factor, which is dependent upon the location of fresh air intake for the control room and the worst location for an operator to stand, is conservatively assumed to be 9E-4 sec/m^3. Based on the total I-131 dose equivalent activities released from containment in 30 days, and based on the ratio of control room air concentration to outside air concentration, the thyroid dose is calculated to be 0.45 rem.

14.2.2.7.4 Effects of Engineered Safeguards Systems Leakage

The engineered safeguards include HPI and LPI of the ECCS. These systems can provide an additional source of fission product leakage external to the reactor building during the recirculation phase for long-term core cooling. It is postulated that during the MHA, one of the core cooling systems undergo a pipe rupture. If the pipe breaks when the core cooling pumps are drawing on the reactor building sump, radioactive liquid would be released within the auxiliary building. The pipe break is assumed to occur at the location resulting in the greatest loss of reactor building sump fluid. Radioactivity is released by exfiltration through the charcoal filters of the auxiliary building ventilation system to the unit vent. Reactor building leakage is assumed to occur throughout the accident at the Design Basis Accident leakage rate of 0.25% by weight of contained atmosphere per 24 hrs. A detailed analysis of the potential leakage from these systems is presented in Section 6.

It is assumed that the water being recirculated from the reactor building sump through the external system piping contains 50% of the core saturation iodine inventory, which is the entire amount of iodine released from the reactor core cooling system. The 50% escaping from the RCS is consistent with TID-14844 specifications. The assumption that all iodine escaping from the reactor coolant system be absorbed by the water in the reactor building is conservative since much of the iodine released from the fuel would be plated out on the building walls. It is assumed that all of the iodine contained in water which flashes is released to the auxiliary building atmosphere. Iodine release from the remaining water is calculated using a gas/liquid partition coefficient of 9E-3. 50% of the iodine released to the auxiliary building is assumed to plate out on the walls. The remainder is assumed to be released through 90% efficient charcoal filters. The atmospheric dilution is based on the 2-hour dispersion factors shown in Table 14-23. The leakage and the resulting thyroid dose are shown in Table 14-57.

TABLE 14-52

REPLACE WITH REVISED TABLE 14-52

~~ACTIVITY RELEASE FROM MAXIMUM BREAK SIZE LOCA~~

Isotope	Activity, Ci
Noble Gases:	
Kr-83m	8.96 E+3
Kr-85m	4.97 E+4
Kr-85	4.37 E+5
Kr-87	2.70 E+4
Kr-88	8.86 E+4
Xe-131m	8.13 E+4
Xe-133m	9.50 E+4
Xe-133	8.52 E+6
Xe-135m	2.76 E+4
Xe-135	3.44 E+4
Iodine-131	6.55 E+5
Iodine-132	9.37 E+4
Iodine-133	1.41 E+5
Iodine-134	8.81 E+3
Iodine-135	4.47 E+4

REPLACE WITH REVISED TABLE 14-53

TABLE 14-53

~~REACTOR BUILDING SPRAY SYSTEM EFFECTIVENESS~~

Parameter	1 Spray Operates	2 Sprays Operate
Spray Flow, gpm	1500	3000
Effective Fall Height, ft	96	96
RB Free Volume, ft^3	2,000,000	2,000,000
Spray Drop Diameter, microns	1080	1080
Average Removal Time Constant, hr^{-2}		
Elemental	12.78	25.56
Aerosol Particles	0.37	0.73
Organic	0.0	0.0

REPLACE WITH REVISED TABLE 14-54

TABLE 14-54

~~ENVIRONMENTAL DOSES RESULTING FROM MAXIMUM BREAK SIZE LOCA~~

2-Hour Dose at Exclusion Distance, Rem

	Original FSAR Value	Cycle 3 Value	Cycle 7 Value
Thyroid	0.549	2.19	3.01
Whole Body	0.0174	0.016	0.008

30-Day Dose at Low Population Distance, Rem

	Original FSAR Value	Cycle 3 Value	Cycle 7 Value
Thyroid	0.073	0.517	0.25
Whole Body	0.011	0.0081	0.004

LOCA During Reactor Building Purge

Purge Valve Closing Time, s	5.0
Iodine Released, equiv Ci	4.2 I-131 Dose
Increase in 2-Hour Thyroid Dose at Exclusion Distance Due to Purge Valve Closing Time, Rem	0.213

TABLE 14-55

~~MHA IODINE ACTIVITY AVAILABLE FOR LEAKAGE~~

~~WITH 50% PLATE OUT~~

ISOTOPE	Reactor Building Activity, Ci
~~I-131~~	~~1.61 E+7~~
~~I-132~~	~~2.45 E+7~~
~~I-133~~	~~3.62 E+7~~
~~I-134~~	~~4.24 E+7~~
~~I-135~~	~~3.29 E+7~~

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TABLE 14-56

~~MHA RADIOACTIVITIES AVAILABLE FOR LEAKAGE~~

ISOTOPE	Reactor Building Activity, Ci
Kr-83m	7.3 E+6
Kr-85m	2.1 E+7
Kr-85	5.4 E+5
Kr-87	3.9 E+7
Kr-88	6.0 E+7
Xe-131m	5.5 E+5
Xe-133m	3.1 E+6
Xe-133	1.3 E+8
Xe-135m	3.4 E+7
Xe-135	2.6 E+7

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TABLE 14-57

~~MHA ENVIRONMENTAL DOSES~~

2-Hour Dose at Exclusion Distance, rem

	Original FSAR Dose			Cycle 7 Dose
Thyroid	23.3	23.4*	26.1**	63.1**
Whole Body	2.01	2.01*	2.02**	1.55**

30-Day Dose at Low Population Distance, rem

	Original FSAR Dose			Cycle 7 Dose
Thyroid	2.65	2.66*	2.89**	9.11**
Whole Body	0.29	0.29*	0.29**	0.29**

Engineered Safeguards Leakage

Iodine concentration in liquid, I-131 dose equivalent, Ci/ml	0.034
Liquid leakage, ml/hr	2165
Leakage that flashes, ml/hr	90
Thyroid dose at exclusion distance, rem	0.0191

* Considers throttling of reactor building spray pumps at time of recirculation (34.4 minutes after accident with two LPI, two HPI, and two RPS pumps operating) to 1200 gpm.

** Considers reactor building spray pump flow of 1200 gpm.

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assembly prior to being recirculated. The sole function of the sump screen assembly is to prevent small debris in the recirculating water from entering the associated systems.

A 1-1/2 inch grating cover above the sump inlet is designed to prevent large debris from entering the sump area. Dislodged debris and paint chips present in the recirculation water, smaller than 1-1/2 inch size, will flow into the reactor building sump preceding the sump screen assembly. High density particles will have a tendency to settle out and be retained by the 3 foot weir preceeding the sump screen assembly. The velocity of flow through the sump screen is relatively low and in a downward direction, therefore permitting suspended debris to settle out and collect in the debris hoppers. Particles smaller than 1/4 inch in size which are not retained by the weir or sump screen assembly will flow through the associated Decay Heat Removal System and Reactor Building Spray System with no additional restrictions, thus returning to their originating source (reactor building proper).

GAI report 2009, "Borated Water and Sodium Hydroxide Storage Tank Drawdown Transient Analysis," (Appendix 14A) addresses the ES operation of the Reactor Building Spray System and Decay Heat Removal System. This report demonstrates that utilizing the sodium hydroxide tank with a 10.5 to 12.0 weight percent sodium hydroxide solution, the maximum pH of the spray solution is not greater than 11.0 during all modes of operation and that the resulting doses from the Maximum Hypothetical Accident (MHA) are within 10CFR100 limits. In addition, it demonstrates that a relatively high pH (at least 8.0) is maintained in the sump after mixing and dilution with primary coolant, borated water from ECCS injection, and Core Flooding Tank (CFT) inventories. The operation of the system during the following modes of operation was analyzed and found to be acceptable:

a. Full flow mode in which all components function as designed.

b. Half flow mode in which one train does not operate.

c. Valve failure mode in which BSV-36F or 37F in the NaOH addition line fails closed.

d. Spray pump failure mode in which BSP-3A or 3B fails.

e. Decay heat pump failure mode in which DHP-3A or 3B fails.

6.2.2.1.1 Building Spray Nozzles

SPRACO-1713A spray nozzles are used in the spray headers. They are ramp bottom swirl chamber type nozzles of one piece construction; they have a 3/8 inch orifice and deliver a hollow cone spray pattern. Each nozzle will deliver 15.7 gal/minute at 40 psi with a spray angle of 63°. The drop size distribution used in the design distribution produces a conservative evaluation of the system's iodine removal rate. The measured spray drop size distribution is based on the results of spray tests performed by Spray Engineering Company of Burlington, Mass. during 1970 and 1971. The following paragraphs describe how the tests were performed.

A SPRACO-1713A spray nozzle was positioned ten feet above the plane of drop size measurement. The nozzle sprayed water straight down at a rate of 15.3 gpm at a 40 psi differential pressure. The plane of measurement was divided into eight concentric regions each six inches wide and then into four quadrants, which gave a total of 32 zones. The fraction of the total spray flow was measured for each zone. High speed photographs were also taken in each zone to measure the spatial drop size distribution at that location. The photographs were taken with a three micro-second exposure and with the field of spray limited to a 2 inch thick radial section across the zone. The photographic negatives from each zone were analyzed for the number of drops in every 25 micron interval, using a Mann Model 880 Comparator. The total drop count was about 33,000 drops. The end result of the experimental measurements made by Spray Engineering Company was 32 histograms (one of each zone) showing the number frequency of spatially distributed drops versus the drop size, and tables summarizing the amount of spray flow in each zone. This spatial drop size distribution data was then analyzed, as follows, to obtain the temporal mass drop size distribution shown in Figure 6-13. The percentage of the spray's mass flow rate (P_d) which contained only drops of a specified size (d) or smaller was calculated as follows:

$$P_d = \sum_{D=1}^{d} \sum_{Z=1}^{32} (N_D \, v_D \, V_D \, \rho \, F_D)_Z$$

where P_d = percentage of spray's mass flow rate which contains only drops of size (d) or smaller;

N_D = the number of spatially distributed drops in a given zone which are in drop size group (D_O) (group widths are 25 microns);

Motors

The reactor building spray pump motors are designed to the same requirements as the ECCS motors. Refer to Section 6.1.2.4.

6.2.2.5 Reliability Considerations

A failure analysis has been made on all active components of the system to show that the failure of any single active component will not prevent fulfilling the design function. This analysis is shown in Table 6-6.

6.2.2.6 Missile Protection

Protection against missile damage is provided by direct shielding or by physical separation of duplicate equipment. The spray headers are located outside and above the primary and secondary concrete shield.

6.2.3 DESIGN EVALUATION

The Reactor Building Spray System, acting independently of the Reactor Building Emergency Cooling System, is capable of limiting the containment pressure after a LOCA to a level which is below the design pressure and reduces building pressure to near atmospheric level. The Reactor Building Spray System is at least equivalent in heat removal capacity to the Reactor Building Emergency Cooling System and is designed for long term post-accident operation. In combination with emergency cooling units, it affords redundant alternative methods to maintain containment pressure at a level below design pressure. Any of the following combinations of equipment will provide sufficient heat removal capability to accomplish this:

a. The Reactor Building Spray System.

b. Three reactor building emergency cooling units.

c. One reactor building emergency cooling unit and the Reactor Building Spray System operating at one-half capacity.

The Reactor Building Spray System will deliver 3,000 gpm through the spray nozzles within 68.2 seconds after the reactor building pressure reaches the actuation set point.

INSERT NEW 6.2.3.1

6.2.3.1 RB Spray System Iodine Removal Evaluation

The iodine removal function of the RB Spray System has been evaluated for fully effective and minimum safeguards operation in the following cases. Analysis of each case includes the condition of a single active failure of any active component.

1. Full Flow Case - Normal mode in which all components function as designed. Spray flow is 3000 gpm.

2. Half Flow Case - A half flow mode in which one string of pumps and valves do not operate, i.e. one diesel fails to operate and all other components function as designed. The B string was selected as the failure for this analysis. Spray flow is 1500 gpm.

3. BST-1 Valve Failure Case - Valve BSV-11 (B-side) fails closed and all other components function as designed. In this case, the total spray flow is 3000 gpm, but only Train A with a flow of 1500 gpm receives sodium hydroxide (NaOH).

4. Spray Pump Failure Case - Failure of the spray pump (B-side) and all other components function as designed. Spray flow is 1500 gpm.

5. Decay Heat Pump Failure - Failure of the decay heat pump (B-side) and all other components function as designed. The total spray flow is 3000 gpm. However, Train B receives a reduced amount of sodium hydroxide due to failure of the decay heat pump.

The iodine in the post accident Reactor Building atmosphere is assumed to exist in three chemical forms, i.e. elemental, organic (methyl), and iodine sorbed on airborne particulate matter. The RB Spray System with iodine absorbing additive (i.e. NaOH) remove these three forms with varying degrees of effectiveness. The removal of each form of iodine is described mathematically by a first order exponential removal process with a removal rate coefficient.

The SPIRT computer code was used to evaluate the spray removal constants for the elemental form of iodine. Hand calculational methods (Ref. 2) were used to determine the removal constants for particulate iodines. Since the spray additive sodium hydroxide (NaOH) is not very effective in enhancing the removal rate of organic forms of iodine, the removal of methyl iodide was conservatively assumed to be zero. A summary of the assumptions and parameters used in evaluating the effectiveness of the spray system is presented in Tables 6-15 and 6-16.

The capacity of the spray solution to absorb elemental iodine from the post accident RB atmosphere is strongly dependent upon the pH of the spray solution via the equilibrium iodine partition coefficient. The recommended values (Ref. 3) of partition coefficients for sodium hydroxide buffered spray solution varies from 50 to 5000 over a pH range of 6.5 to 8.5. The spray solution pH is a function of (a) mode of spray system operation, (b) rate of drawdown from the ECCS storage tanks, and (c) rate of sodium hydroxide injection. The spray solution pH values for each operating mode were determined as part of the RB Spray System and ECCS Storage Tank Drawdown

Analysis (Ref. 4). The pH values are presented in Table 6-17. These values are based upon assuming the minimum NaOH concentration (6 wt.%), the minimum level in the NaOH storage tank (BST-1), the maximum borated water concentration, and maximum level in the borated water storage tank (BWST). The elemental iodine spray removal constants shown in Table 6-18 are based upon the Table 6-17 pH values.

The effectiveness of the spray system is assumed to cease once the concentration of elemental iodine in the atmosphere reaches the equilibrium limit, i.e. the maximum allowable decontamination factor (DF) is reached. The DF is defined as the ratio of the initial iodine concentration in the RB atmosphere when 50 percent of the core iodine is instantaneously released to the concentration of iodine in the RB atmosphere at some time later. This value was determined to be 170.4.

The spray removal constants determined for the particulate iodines are as follows:

Spray Flow (gpm)	p (/hr)
One Header - 1500	0.30
Two Headers - 3000	0.60

Add these references to page 6-33

2. NUREG/CR-0009, "Technicalogical Bases For Models Of Spray Washout Of Airborne Contaminants In Containment Vessels," USNRC, October 1978.

3. ANSI/ANS-56.5-1979, "PWR And BWR Containment Spray System Design Criteria", November 1979.

4. Florida Power Corporation, Crystal River Unit 3, "Reactor Building Spray And ECCS Storage Tanks Drawdown Analysis", B&W Document 86-1146656-01, November 1983, FPC Document M-83-0001

TABLE 6-15

IODINE REMOVAL EVALUATION
REACTOR BUILDING SPRAY SYSTEM

PARAMETER/COMPONENT	ASSUMPTION
Spray System:	
Spray Nozzle Type SPRACO MODEL 1713A	
Number Of Spray Drop Sizes	56
Spray Drop Size Distribution	Table 6-16
Spray Flow Rate (One/Two Header), gpm	1500/3000
Collection Drop Efficiency	1.0
Spray Solution Chemistry:	
Spray Additive	NaOH (6 wt.%)
Spray Storage Temperature, F	40
Spray pH Range	7.2 to 11
Partition Coefficient (H) Elemental Iodine	310 to 5000
Reactor Building Design:	
RB Free Volume, ft^3	2,000,000
RB Free Diameter, ft	130
Fraction Of RB Volume Sprayed, %	65.2
Fall Height (One/Two Header), ft	109/110
Maximum Post-Accident Atmospheric Temp, F	281
Liquid Volume RB Sump, gal	490,182
Interior Surfaces:	
RB Surface Area Impacted By Sprays, ft^2 (One/Two Header)	37,900/38,760
Laminar Boundary Layer Surface Area, ft^2	4084
Turbulant Boundary Layer Surface Area, ft^2 (One/Two Header)	58,320/59,180
Spray Water Wall/Flow Fraction	0.1
Delta T Across Wall/Gas Boundary, F	1.0

TABLE 6-16

SPRAY DISTRIBUTION FOR SPRACO MODEL 1713A NOZZLE

Data Point Number	Drop Size (cm)	Relative Frequency (Fraction)
1	3.75-3	0.011
2	6.25-3	0.027
3	8.75-3	0.056
4	1.125-2	0.105
5	1.375-2	0.095
6	1.625-2	0.080
7	1.875-2	0.070
8	2.125-2	0.051
9	2.375-2	0.066
10	2.625-2	0.044
11	2.875-2	0.026
12	3.125-2	0.022
13	3.375-2	0.017
14	3.625-2	0.020
15	3.875-2	0.023
16	4.125-2	0.011
17	4.375-2	0.011
18	4.625-2	0.015
19	4.875-2	0.012
20	5.125-2	0.013
21	5.375-2	0.011
22	5.625-2	0.016
23	5.875-2	0.012
24	6.125-2	0.008
25	6.375-2	0.008
26	6.625-2	0.007
27	6.875-2	0.011
28	7.125-2	0.009

TABLE 6-16 (CONTINUED)

SPRAY DISTRIBUTION FOR SPRACO MODEL 1713A NOZZLE

Data Point Number	Drop Size (cm)	Relative Frequency (Fraction)
29	7.375-2	0.011
30	7.625-2	0.009
31	7.875-2	0.008
32	8.125-2	0.007
33	8.375-2	0.006
34	8.625-2	0.006
35	8.875-2	0.008
36	9.125-2	0.006
37	9.375-2	0.005
38	9.625-2	0.005
39	9.875-2	0.005
40	1.013-1	0.004
41	1.038-1	0.005
42	1.063-1	0.004
43	1.088-1	0.005
44	1.113-1	0.005
45	1.138-1	0.005
46	1.163-1	0.004
47	1.188-1	0.005
48	1.213-1	0.005
49	1.238-1	0.007
50	1.288-1	0.005
51	1.313-1	0.002
52	1.338-1	0.002
53	1.413-1	0.001
54	1.438-1	0.001
55	1.613-1	0.001
56	1.738-1	0.002

TABLE 6-17

RESULTS OF DRAWDOWN ANALYSIS FOR A MINIMUM OF 6.0 WT% SODIUM HYDROXIDE IN THE STORAGE TANK

Case	INITIAL SPRAY PH			TIME POST-LOCA TO REACH SPRAY PH OF 8.5			Post-LOCA Time To Sump Recirculation (min)
	Train A	Train B	Average A & B	Train A	Train B	Average A & B	
Full Flow	7.3	8.1	-	6.0 min.	3.7 min.	-	41.82
Half Flow	7.3	-	-	8.0 min.	-	-	81.82
BST-1 Valve Failure	7.5	-	7.5	3.75 min.	-	6.7 min.	41.48
Spray Pump Failure	7.4	-	-	9.0 min.	-	-	48.80
Decay Heat Failure	7.5	4.1	-	5.0 min.	26.0 min.	-	58.32

TABLE 6-18

ELEMENTAL IODINE
SPRAY REMOVAL CONSTANTS

Case	pH	Initial e(hr^{-1})	Time To Achieve pH $\geq$ 8.5 (Min)	e(hr^{-1}) pH $\geq$ 8.5
Full Flow	7.3	4.61	6.0	31.09
Half Flow	7.3	2.30	8.0	16.58
BST-1 Valve Failure:				
With One Header	7.5	3.55	3.75	16.58
With Two Headers	7.5	7.07	6.70	31.09
Spray Pump Failure	7.4	2.91	9.0	16.58

Decay Heat Pump Failure: This Situation Is Bounded By The BST-1 Valve Failure (With Two Headers) And Spray Pump Failure Cases.

14.2.2.6 Makeup System Letdown Line Failure Accident

14.2.2.6.1 Identification of Cause

A break in fluid-bearing lines that penetrate the reactor containment may result in the release of radioactivity to the environment. There are no instrument lines connected to the RCS that penetrate the containment. However, there are other piping lines such as those associated with the Makeup and Purification (MU) System and the Decay Heat Removal (DH) System that penetrate the containment. For fluid penetrations in piping systems that do not serve to limit the consequences of accidents, leakage is minimized by a double-barrier design to ensure that no single credible failure or malfunction of an active component will result in either unacceptably high leakage or the loss of the capability to isolate a piping break. The installed double barriers consist of closed piping, both inside and outside the containment, and various types of isolation valves.

The most severe piping rupture identified for which radioactivity release may occur during normal plant operation is in the Makeup and Purification System. This involves a rupture of the letdown line just outside the containment and upstream of the letdown control valves. A rupture at this point produces a loss of reactor coolant condition until the RCS pressure drops below the pressure for actuation of the Engineered Safeguards to isolate the reactor building. When this pressure is reached, the building isolation signal initiates closure of the letdown isolation valves inside the containment. Closure of the isolation valves stops the release of reactor coolant and fission products to the auxiliary building, thus terminating the loss-of-coolant phase of the accident.

14.2.2.6.2 Safety Evaluation Criterion

The acceptance criterion for the evaluation of this accident is that the resultant doses shall not exceed 10CFR100 limits. (Dose limits are 300 rem thyroid dose and 25 rem whole body dose.)

14.4.4.6.3 Methods of Analysis

The CRAFT2 computer code was used to determine the reactor coolant mass release rates and the primary system response for the rupture of the letdown line. The multinode model includes a detailed model of the RCS as well as noding for simulation of the letdown piping, valves, and coolers.

For purposes of calculating the mass of reactor coolant released, the reactor is assumed to be operating at 2603 MWt with a letdown flow of 140 gpm prior to the rupture. The rupture is modeled as a complete severance of the 2 1/2 inch nominal diameter letdown line at a location between containment penetration number 333 and the downstream isolation valve (MUV-49). As a consequence of the failure, the makeup control valve is assumed to move to the fully opened position to provide the maximum available makeup flow. This assumed control action delays the times for the trip of the reactor and the actuation of ESAS and consequently increases the releases of reactor coolant mass and the fission products to the auxiliary building.

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Automatic actuation of ESAS is assumed to occur at a pressure setpoint of 1350 psig, which corresponds to the nominal value of 1500 psig with an adjustment for possible instrument error equal to 6% of the 2500 psig range of the measurement. The letdown isolation valve is assumed to reach the fully closed position 7.4 seconds after the ESAS pressure setpoint is reached. This time period includes both the instrumentation delay time and the valve stroke time.

Dose calculations are based on a core power level of 2544 MWt with the fission product concentrations corresponding to 1 percent defective fuel rods. Ten percent of the iodine contained in the mass of reactor coolant is assumed to volatilize and become airborne in the auxiliary building. The remaining 90% is assumed to remain in the liquid which drains into the auxiliary building sump. The airborne radioactive nuclides in the auxiliary building are filtered through High Efficiency Particulate Air (HEPA) and charcoal filters in the Auxiliary Building Ventilation System (ABV) before being exhausted to the environment. The analysis is based on a conservatively estimated efficiency of 90% for iodine removal by the charcoal filters. The assumptions used in the evaluation of the off-site doses are summarized in Table 14-41.

14.2.2.6.4 Results of Analysis

The calculated time for the RCS to depressurize and reach the actuation pressure for the ESAS is 745 seconds. At a time of 752 seconds, the isolation valve is completely closed. The total mass of reactor coolant that escapes through the break and is released to the auxiliary building is 45,760 pounds.

The fission product activities released to the environment during the accident are listed in Table 14-42. The dose consequences of the letdown line rupture accident are presented in Table 14-43. The table presents: (1) the thyroid dose due to inhalation of iodine activity; and (2) the whole body doses from gamma radiation due to immersion in the gas cloud for individuals located at the outer boundaries of either the exclusion area or the low population zone for the first two hours after the accident. The resulting doses are small fractions of the 10CFR100 limits.

An analysis was performed to determine the offsite doses with no iodine filtration by the ABV System. The ABV System is non-safety-related and is not provided with emergency power. Tables 14-42 and 14-43 provide a comparison of the dose consequences. The results show an increase in the offsite thyroid dose by a factor of 10 which is in proportion with the decrease in the assumed ABV System charcoal filter efficiency (90% vs 0%). The whole body doses increased by 10 mrem at both the EAB and LPZ. These doses are much less than the limits specified by 10CFR100.

TABLE 14-41

ANALYSIS ASSUMPTIONS FOR THE MU SYSTEM LETDOWN LINE RUPTURE ACCIDENT

Data and Assumptions Used to Estimate Radioactive Source

Power level, MWt	2544
Percent of fuel rods leaking, %	1.0
Escape rate coefficient	Table 11-1

Reactor Coolant Activity

Nuclide	Activity, Micro-Ci/cc
Kr 85m	1.48
85	4.36
87	0.779
88	2.41
Xe 131m	1.63
133m	2.58
133	238.0
135m	0.294
135	4.88
138	0.421
I 131	3.47
132	1.17
133	3.7
134	0.461
135	1.88

Data and Assumptions Used to Estimate Radioactivity Released

Total mass of reactor coolant released to auxiliary building, lb	45,760
Charcoal filter efficiency for	
Iodine, %	90/0[a]
Noble gas, %	0
Fraction of iodine airborne	0.1/1.0[a]

Dispersion Data

EAB, m	1340
LPZ boundary, m	8047
Atmospheric dispersion percentile, %	5
0-2 hour atmospheric dispersion factors, s/m^3	
at EAB	1.6 E-4
at LPZ boundary	1.4 E-5

a) Reflects analyses with and without credit taken for iodine filtration by the auxiliary building ventilation system.

TABLE 14-42

ACTIVITY RELEASED TO ENVIRONMENT DUE TO
RUPTURE OF THE MU SYSTEM LETDOWN LINE

Nuclide	Activity, Ci (a)	(b)
Kr 85m	44.6	44.6
85	131.0	131.0
87	23.5	23.5
88	72.6	72.6
Xe 131m	49.1	49.1
133m	77.7	77.7
133	7170.0	7170.0
135m	8.85	8.85
135	147.0	147.0
138	12.7	12.7
I 131	10.4	104.0
132	3.52	35.2
133	11.1	111.0
134	1.39	13.9
135	5.66	56.6

(a) With 90% iodine filtration by the auxiliary building ventilation system assumed.

(b) With no iodine filtration by the auxiliary building ventilation ~~s~~ system assumed.

TABLE 14-43

SUMMARY OF RESULTANT DOSES FOR THE
MU SYSTEM LETDOWN LINE RUPTURE ACCIDENT

~~Total Integrated Dose at Exclusion Boundary~~	
~~Thyroid, Rem~~	~~0.115~~
~~Whole Body, Rem~~	~~0.066~~
~~Total Integrated Dose at Low Population Zone~~	
~~Thyroid, Rem~~	~~0.0101~~
~~Whole Body, Rem~~	~~0.0058~~

2-hour Total Integrated Doses at the Exclusion Area Boundary

	Thyroid, REM	Whole Body, REM
90% ABVS iodine filtration	0.115	0.066
0% ABVS iodine filtration	~~0~~ 1.15	0.067

30-day Total Integrated Doses at the Low Population Zone

	Thyroid, REM	Whole Body, REM
90% ABVS iodine filtration	0.0101	0.0058
0% ABVS iodine filtration	0.101	0.0059