Unreviewed Safety Question Evaluation # 92-0022 Revision No. 1

ORIGINATING DOCUMENT: CN-1980 REV. NO. 0

NOTE: Attach 10CFR50.59 Screening Form or License Compliance review Form to this USQE.

NOTE: Use additional sheets as necessary to provide the bases.

A.1 I Does the subject of this evaluation increase the probability of occurrence of an accident previously evaluated in the Safety Analysis Report? ☐ YES ☒ NO

Bases: SEE ATTACHMENT.

II Does the subject of this evaluation increase the consequences of an accident previously evaluated in the Safety Analysis Report? ☐ YES ☒ NO

Bases: SEE ATTACHMENT.

III Does the subject of this evaluation increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report? ☐ YES ☒ NO

Bases: SEE ATTACHMENT.

IV Does the subject of this evaluation increase the consequences of a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report? ☐ YES ☒ NO

Bases: SEE ATTACHMENT.

9707180081 970716
PDR ADOCK 05000498
P PDR

A.2 I Does the subject of the evaluation create the possibility of an accident of a different type than any previously evaluated in the Safety Analysis Report? ☐ YES ☒ NO

Bases: SEE ATTACHMENT

II Does the subject of this evaluation create the possibility of a different type of malfunction than any previously evaluated in the Safety Analysis Report? ☐ YES ☒ NO

Bases: SEE ATTACHMENT

A.3 I Does the subject of this evaluation reduce the margin of safety as defined in the basis for any Technical Specifications? ☐ YES ☒ NO

Bases: SEE ATTACHMENT

SAFETY EVALUATION SUMMARY

The following documents have been reviewed as part of the 10CFR50.59 final screening process.

UFSAR	6.4.4.1, Table II-B.2-2, 15.6.5.3, 15.B.2.1, 3.11, 12.3
Technical Specifications	Table of contents
Safety Evaluation Report	15.6.5.2, 6.4, Suppl. #4 - Sect 3.11
Amend. to Operating Lic.	Unit 1 Amendment #38, Unit 2 Amendment #29

A question by question evaluation is provided in the attachment.

B. 1. X All of the above questions were answered No; therefore, the originating document does not involve an Unreviewed Safety Question.

2. ______ One or more of the above questions was marked YES; therefore, the originating document involves an Unreviewed Safety Question. The originating document, as presented, shall NOT be implemented without prior approval by the NRC. Provide a recommendation for disposition of the Unreviewed Safety Question below. Refer to 0PGP05-ZA-0004 for processing licensing amendments. Further processing of this form to the PORC, Plant Manager and NSRB is not required. Notify Procedure Control that the evaluation involved an Unreviewed Safety Question so that Procedure Control can close the USQE number.

RECOMMENDED DISPOSITION:
APPROVE THE CHANGE.

PREPARED BY: Mark Blumberg Mark Blumberg 9/28/95
ORIGINATOR Date

REVIEWED BY: M. A. Whitley Michael A Whitley 10-4-95
QUALIFIED REVIEWER Date

APPROVED BY: D. A. Leazar D.A. Leazar 10/5/95
DEPARTMENT MANAGER Date

PORC MEETING NO. 95-050 10/20/95
Date

APPROVED BY: 11/20/95
~~PLANT~~ MANAGER Date

REMARKS:
Copy of USQE approved at meeting 95-050. The original was lost.
11/6/95

DESCRIPTION OF CHANGE:

This Change Notice updates the UFSAR for a change in design assumptions used to calculate radiological doses resulting from a Loss of Cooling Accident (LOCA) (NC6013, Rev. ~~7~~ 9, NC9004, Rev. ~~8~~ 9). This update evaluates the doses resulting from potential leakage of containment sump water into the Refueling Water Storage Tank (RWST) located in the Mechanical Auxiliary Building (MAB).

REASON FOR CHANGE:

Review of NRC Information Notice IEN 91-056 determined that potential leak paths exist which could result in radioactive sump water entering the RWST through either the Safety Injection System (SIS) pump mini-flow recirculation lines, the Containment Spray (CS) pump discharge test lines or the RWST suction isolation valves during the recirculation phase of cooling following a DBA.

The concern specifically involves the SIS pump mini-flow recirculation (SI 0011/0012 A,B,C and SI 0013/0014 A,B,C), CS pump manual test (XCS 0008 A,B,C), and the RWST suction isolation valves (SI 0001/0002 A,B,C).

Each HHSI and LHSI train contains two, in series, normally open, packless diaphragm, motor operated valves (MOVs) in the recirculation lines. These mini-flow valves function to close on the switchover from the Injection to the Recirculation Phase to prevent radioactive sump water from being lost to the RWST. The normally closed, test line isolation valves in the CS System serve only to isolate CS pump discharge from the RWST during normal and accident operations. The RWST suction isolation valve prevents radioactive sump water from contaminating the RWST during the recirculation phase.

Although these valves close, they may leak during the subsequent prolonged recirculation phases of core cooldown. Any leakage through these valves will flow into the RWST. This possible leakage was not previously considered when calculating Control Room, Technical Support Center (TSC), Offsite, Equipment Qualification (EQ), and post-accident zone doses. Therefore, the SPR on this subject (91-0475) identified a corrective action which evaluated the radiological effects of this leakage. This proposed change reflects the results of this evaluation.

This USQE revision supports the proposed changes due to difference in the calculated leakage transport time assumed in radiological calculations. USQE 92-022, Revision 0 was based upon analysis performed in Calculation MC6313, Rev. 0. This analysis calculated a transport time of 13.4 days. Recently, MC6313, Rev. 0 was superseded by Calculation MC6458, Rev. 0. This calculation determined the backleakage time to be approximately 42.3 days. The change in transport time has been incorporated in Calculations NC-6013, Rev. 9, "Control Room, TSC and Offsite Doses During a LOCA," and NC-9004, Rev. 9, "Post LOCA Zones and EQ." The increase in transport time increases the decay of radioactivity and, thus, decreases the total radioactivity transported to the RWST. This yields lower offsite and onsite doses. This change in doses is reflected in this revision of the USQE (Revision 1).

10CFR50.59 FINAL SCREENING FORM RESPONSES
Technical Justification of Change

1. Does the subject of this review involve a change to the facility as described in the SAR?

   Yes, this USQE proposes a change to UFSAR Sections 6.4, "Habitability Systems," 7.B.II.2 , "Design Review of Shielding and Environmental Qualifications of Equipment for Spaces/Systems Which May Be Used in Post Accident Operations," and 15.6.5, "Loss of Coolant Accidents". The change to these analyses introduces a previously unreviewed system interaction (ie. Previously these analyses did not assume potential leakage of containment sump water into the Refueling Water Storage Tank (RWST)).

2) Dose the subject of this review involve a change to the procedures as described in the SAR?

   No, UFSAR procedures are not impacted by this change. No UFSAR or SER procedures discuss or describe the LOCA analysis affected by this proposed change.

3) Does the subject of this review propose the conduct of tests or experiments not described in the SAR?

   No, this change is not a test or experiment, nor does it propose the conduct of a test or experiment. The change describes the effect of an additional leakage pathway on design basis dose analysis.

4) Does the proposed change affect conditions or bases assumed in the SAR or safety-related functions of equipment/systems, even though the proposed change does not entail any physical change in existing structures, systems, or procedures as described in the SAR?

   Yes, the proposed change affects the bases assumed in the UFSAR. Previously, the UFSAR assumptions used in the LOCA dose analysis did not include the doses due to the potential leakage of containment sump water into the Refueling Water Storage Tank (RWST). The proposed change incorporates this leakage. Therefore, this is a change to the bases.

USQE RESPONSES

A.1 I) Does the subject of this evaluation increase the probability of occurrence of an accident previously evaluated in the SAR?

No. As in the previous analysis, the proposed change assumes that the Loss of Coolant Accident will occur. Therefore, there is no increase in the probability of an accident previously analyzed in the SAR.

A.1 II) Does the subject of this evaluation increase the consequences of an accident previously evaluated in the SAR?

No. The proposed change does not increase the consequences of an accident previously evaluated in the SAR.

The proposed change does not degrade or prevent action assumed in the SAR. It does add an assumption to those previously made in evaluating the radiological consequences of a LOCA described in the SAR, but the radiological consequences described in this change are bounded by those set by 10CFR100 and SER Sections 15.6.5.2.5 and 6.4 (dated April 1986). Per these documents, the acceptance criteria doses for a LOCA are as follows:

LOCATION	ACCEPTANCE CRITERIA DOSES (REM)		
	BETA SKIN	THYROID	WHOLE BODY
EZB	N/A	300	25
LPZ	N/A	300	25
CR	30	30	5
TSC	30	30	5

Where: EZB - Exclusion Zone Boundary, LPZ - Low Population Zone, CR - Control Room, and TSC - Technical Support Center

Tables 6.4-2, II.B.2-2, and 15.6-11 of the proposed change give the EZB, LPZ, CR and TSC doses as summarized below:

LOCATION	PROPOSED DOSES (REM)		
	BETA SKIN	THYROID	WHOLE BODY
EZB	1.22	137.0	2.27
LPZ	0.47	66.3 ~~66.4~~	0.73
CR	21.52	22.7 ~~23.3~~	2.43
TSC	24.55	28.6 ~~29.4~~	4.85

The values of the proposed change are less than the acceptance criteria doses given above.

Based upon these results, there is no increase in the consequences of an accident previously analyzed in the SAR.

A.1 III) Does the subject of this evaluation increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

No. The subject of this evaluation does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

The effects of RWST backleakage upon equipment qualification doses is evaluated in this proposed change. The changes are given in the table below.

ROOM	ACCIDENT EQ DOSE (RAD)	
	CURRENT	PROPOSED
62	1.6E+5 ~~1.0E+2~~	1.2E+4 ~~1.6E+5~~
63	2.0E+5 ~~1.3E+3~~	1.6E+4 ~~2.0E+5~~

~~Per memos ST-HS-HS-21529, and -21538, the proposed values have no impact on existing qualification.~~ These values are smaller than the current values and are enveloped with sufficient margin as required by 10CFR50.49, and other qualification standards (IEEE 323-1974 and NUREG-0588, Rev. 1).

A.1 IV) Does the subject of this evaluation increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

No. The subject of this evaluation does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

This change proposes making an additional assumption in the LOCA doses analysis. It evaluates the doses resulting from potential leakage of containment sump water into the Refueling Water Storage Tank (RWST) located in the Mechanical Auxiliary Building (MAB). The LOCA analysis already assumes malfunction of equipment important to safety. Per the justification to question A.1 II, this malfunction does not increase the consequences of a malfunction of this equipment.

A.2 I) Does the subject of the evaluation create the possibility of an accident of a different type than any previously evaluated in the SAR?

No. The proposed change does not create the possibility of an accident of a different type previously evaluated in the SAR. This change analyzes the effect of potential leakage of containment sump water into the RWST on LOCA doses. This additional assumption does not create the possibility of a different accident.

A.2 II) Does the subject of this evaluation create the possibility of a different type of malfunction than any previously evaluated in the SAR?

No. This proposed change does not create the possibility of a different type of malfunction than any previously evaluated in the SAR. This change analyzes the effect of potential leakage of containment sump water into the RWST on LOCA doses. This change, in itself, does not lead to a failure mode of a different type than previously evaluated.

A.3 I) Does the subject of this evaluation reduce the margin of safety as defined in the basis for any Technical Specifications?

No. The subject of this evaluation does not reduce the margin of safety as defined in the basis for any Technical Specifications.

The acceptance criteria for the TSC, Control Room, LPZ and EZP doses are given in the response to question A.1 II as defined in 10CFR100 and SER Sections 15.6.5.2.5 and 6.4. The margin of safety is reduced when the onsite (TSC and Control Room) and offsite (LPZ and EZP) doses exceed these acceptance criteria. As previously discussed in the response to question A.1.II, the TSC, Control Room, LPZ and EZP doses are below the acceptance limits.

The acceptance criteria for equipment qualification exposed to post-LOCA recirculating fluid environments is given in the SER, Supplement 4 (pg 3-24), and Amendments 38 and 29 to the facility operating license. The SER states:

> "The maximum value specified by the applicant for use in equipment qualification inside containment and in areas outside containment exposed to post-LOCA recirculating fluid environments is 1.4 E+8 Rads (gamma plus beta). This value is acceptable for use in the qualification of equipment."

Amendments 38 and 29 for Units 1 and 2, respectively, increase this value to 1.5 E+8 Rads.

The maximum dose due to RWST backleakage is ~~2.0 E+5~~ 1.6E+4 Rads, which is well below the acceptance criteria value of 1.5 E+8 Rads.

Therefore, this proposed change does not reduce the margin of safety as defined in the basis for any "Technical Specification" (where "Technical Specification" is defined as the SER document and license amendments that define the licensing basis).

UFSAR MARKUPS

CN-1980

A review has been done to show the materials for equipment which have been qualified for the original design condition of NaOH spray of 7.5 to 10.5 pH are either not affected by the change to the new pH environment or replaced with a suitable material if affected.

The Containment atmosphere is maintained below 4-volume-percent hydrogen consistent with the recommendations of RG 1.7, as discussed in Section 6.2.5. CN 1944

3.11.5.2 Radiation Environment. Safety-related systems and components are designed to perform their safety-related functions after normal operation radiation exposure plus a DBA exposure. The normal operational exposure is based on the design basis source terms presented in Sections 11.1, 11.2, 11.3, and 12.2.1 and the equipment and shielding configurations given in Section 12.3.

The effect of the Vantage 5H (V5H) fuel upgrade on radioactivity concentrations in the fluid systems was reviewed and it was determined that the original reactor coolant activity listed in Table 11.1-2 is bounding. Therefore, the FSAR analyses based on this activity are not adversely impacted by the fuel upgrade. For comparison, the activity concentrations calculated for the V5H fuel are listed in Table 11.1-2A. The corresponding reactor core activity for the V5H upgrade is shown in Table 15.A-1A. CN 1944

Safety-related system and component radiation exposures are dependent on equipment location and the particular DBA involved. In the Containment and control room area, equipment exposures are based on the DBA LOCA. For in-Containment equipment, the DBA LOCA source term is based on a release of 100 percent of the core noble gases, 50 percent of the halogens and 1 percent of the solids. This is consistent with the guidance given in RG 1.89. Control room exposures following a postulated LOCA are controlled to 5 rads or less consistent with the requirements of GDC 19 of 10CFR50, Appendix A. ~~The release pathways considered for the DBA LOCA are described in Section 15.6.5.3.~~ The source terms used correspond to a cycle length of approximately 20,000 MWD/MTU, a core average burnup of 40,000 MWD/MTU, and a discharge burnup of 60,000 MWD/MTU. These burnups are conservative relative to the planned cycle lengths for V5H fuel described in Section 4.3. CN 1980 CN 1944

Radiation source terms for safety-related components which are exposed to post-accident recirculation fluid are consistent with the recommendations of RG 1.89 (i.e., 50 percent of the core halogen inventory and 1 percent of the remaining core solid fission product inventory are mixed in the recirculation water).

Normal and accident radiation doses for the various plant areas are presented in Table 3.11-1. Safety-related equipment design doses are the sum of normal plus accident exposures. The design radiation exposures delineated in Table 3.11-1 are based on gamma and beta radiation. Radiation source terms for safety-related components outside Containment are based on gamma radiation.

Organic materials in the Containment are identified in Section 6.1.2. For the organic coating materials used inside Containment (see Section 6.1.2.1), irradiation tests performed by Oak Ridge National Laboratory have been performed for an integrated gamma dose of 1×10^9 rads (which exceeds the design calculated value in Table 3.11-1). These doses conservatively account for the surface exposure due to beta radiation in the design basis LOCA environment.

CHANGE NOTICE 1944

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITIONS

Location (Environmental Designator)	Temperature: Normal Range (max/min, °F)	Temperature: Abnormal	Temperature: Accident (°F)	Pressure: Normal	Pressure: Accident	Relative Humidity: Normal Range (max/min, %)	Relative Humidity: Accident (%)	Cumulative Radiation[2] Dosage: Normal (rads)	Cumulative Radiation[2] Dosage: Accident (rads)	Radiation Type
High Activity Spent Resin Storage Tank (Rm. 054)	104/50	120/44	125	slightly negative	0.3 psig	80/20	100	$2x10^4$	100	gamma
Reactor Make-up Water Pump Cubicles (Rm. 062)	104/50	104/44	125	slightly negative	1.2 psig	80/20	100	$2x10^3$	$1.2 x 10^4$ ~~$1.0x10^3$~~	gamma + beta
Refueling Water Storage Tank Room (Rm. 063)	104/50	111/44	130	slightly negative	2.4 psig	80/20	100	$6x10^3$	$1.6 x 10^4$ ~~$2.0x10^3$~~	gamma + beta
Non Radioactive Pipe Chase (Rm. 064)	104/50	133/44	125	slightly negative	0.3 psig	80/20	100	10^2	$3.5x10^5$	gamma
Electrical Equipment Room (Rm. 065)	104/50	131/44	135	slightly negative	1.1 psig	80/20	100	100	$1.3x10^3$	gamma
Essential Chiller & CCW Pump Room (Rms. 067, 067E, 067F)	115/50	115/44	120	slightly negative	1.1 psig	80/20	100	10^3	130	gamma
Corridor (Rms. 067A, 067B)	104/50	126/44	170	slightly negative	1.1 psig	80/20	100	10^3	100	gamma
Corridor (Rm. 067C)	104/50	125/44	195	slightly negative	1.6 psig	80/20	100	10^3	100	gamma
Corridor (Rm. 067D)	104/50	126/44	170	slightly negative	1.1 psig	80/20	100	10^3	$1.3x10^3$	gamma

CN 1944

CN 1944

CN 1980

CHANGE NOTICE 1944

room envelope. In the worst case, 235 cfm of the makeup air is filtered by the makeup units, but not by the recirculation units, before it is introduced into the control room envelope. The air-handling unit supplies the conditioned air to the control room envelope. A summary of these parameters is presented in Table 6.4-2. An unfiltered inleakage of 10 scfm to the control room envelope has also been assumed (Ref. 6.4-4). 2

~~The atmospheric releases from the Containment~~ purge valves prior to closure ~~the Refueling Water Storage Tank (ESF leakage))~~ ~~and~~ from the Fuel Handling Building (FHB) (ESF leakage) are assumed to be transported to the control room envelope air intake by the atmospheric (meteorological) conditions existing at the time. These conditions are estimated using the methods of Reference 6.4-4. The atmosphere dispersion factors for each case can be found in Table 6.4-2. 3 CN 1980

The inhalation thyroid dose and the semi-infinite cloud gamma and beta doses are calculated using the time-integrated concentrations in each area and the occupancy factors noted in Table 6.4-2. The semi-infinite cloud model remains appropriate only for the beta dose, due to the short range of beta particles. The semi-infinite cloud gamma dose calculated is divided by a geometric factor which converts the semi-infinite gamma dose to a finite dose (Ref. 6.4-4). This factor is given as:

$$GF = \frac{1173}{V_{0.338}}$$

where:

V = volume of region of interest, ft3

The resulting doses to control room personnel are given in Table 6.4-2.

The calculated thyroid dose total is less than the design limit of 30 roentgen equivalent man (rem), as is the skin beta dose total. The total whole-body gamma dose is less than the design limit of 5 rem. Thus the control room envelope HVAC System design meets the dose requirements of GDC 19 of 10CFR50, Appendix A.

6.4.4.2 Toxic Gas Protection. The general guidance contained in RG 1.78, has been considered in the design of the control room envelope HVAC system, as described in Section 9.4.1. # added CN-1979

Toxic gases which are handled onsite are kept to a minimum. During normal operation small amounts of chlorine are handled within the site boundary at the Training facility. The amount of chlorine (<300 lbs) will not impact the control room envelope. A detailed evaluation of potential hazardous chemical accidents and their impact on control room habitability is provided in Section 2.2.3.

6.4.5 Testing and Inspection

Systems and their components, listed in Section 6.4.4 above, which maintain control room envelope habitability are subjected to documented preoperational testing and inservice surveillance to ensure continued integrity. The tests conducted verify the following for both normal and emergency conditions.

TABLE 6.4-2

CONTROL ROOM DOSE ANALYSIS

Assumptions

Containment leakage assumptions (Based on a containment free volume of 3.41 x 10^6 ft^3)	0.3% (0-24 hrs) 0.15% (1-30 days)
ESF system leakage into the FHB assumptions	8,280 cm^3/hr
~~ESF system leakage into the RWST assumptions~~	~~1,940 cm^3/hr~~ | CN 1980
Pressurization makeup air inflow parameters:	
flow rate	2,200 ft^3/min | CN 1944
filter efficiency *	98.5% inorganic, 98.5% organic, 99% particulate
Control room envelope clean-up air (recirculation) parameters:	
filtered flow rate (recirculation air)	9,500 ft^3/min | CN 1944
filter efficiency	95% inorganic, 95% organic, 99% particulate
envelope free volume	274,080 ft^3
envelope unfiltered inleakage	10 ft^3/min

Meteorological dispersion factors (including wind speed and direction allowances):

	Containment Leakage Case	ESF Leakage and Purge Case
0-8 hours	1.06x10^{-3} sec/m^3	1.29x10^{-2} sec/m^3
8-24 hours	7.03x10^{-4} sec/m^3	8.55x10^{-3} sec/m^3
1-4 days	4.45x10^{-4} sec/m^3	5.42x10^{-3} sec/m^3
4-30 days	1.91x10^{-4} sec/m^3	2.32x10^{-3} sec/m^3

* 1765 cfm is filtered through makeup and recirculation filters; 235 cfm is filtered through makeup filters only. Effective filter efficiency for 2000 cfm is given above.

CHANGE NOTICE 1944

TABLE 6.4-2 (Continued)

CONTROL ROOM DOSE ANALYSIS

Assumptions (Cont'd)

Occupancy assumptions:

24 hours, control room	100%
1-4 days, control room envelope	60%
4-30 days, control room envelope	40%

Breathing rate of operator 3.47×10^{-4} m^3/sec

Results

Operator dose, 0-30 day period (rem):	Thyroid	Whole-Body Gamma	Skin Beta	
Containment leakage	21.03	1.69	21.52	
~~ESF leakage into the FHB~~	~~1.58~~	~~6.6×10^{-3}~~	~~3.94×10^{-4}~~	CN 1944
~~ESF leakage into the RWST~~	~~0.62~~	~~3.8×10^{-3}~~	~~6.1×10^{-4}~~	CN 1980
~~Containment purging~~	~~0.058~~	~~6.1×10^{-3}~~	~~9.86×10^{-4}~~	CN 1944
Direct dose from Containment	---	0.07	---	
Direct dose from cloud of released fission products	---	0.67	---	
Iodine filter loading	---	2.72×10^{-3}	---	
Total	~~23.29~~ 22.67	2.43	21.52	CN 1980

CHANGE NOTICE 1944

TABLE 7.A.II.B.2-2

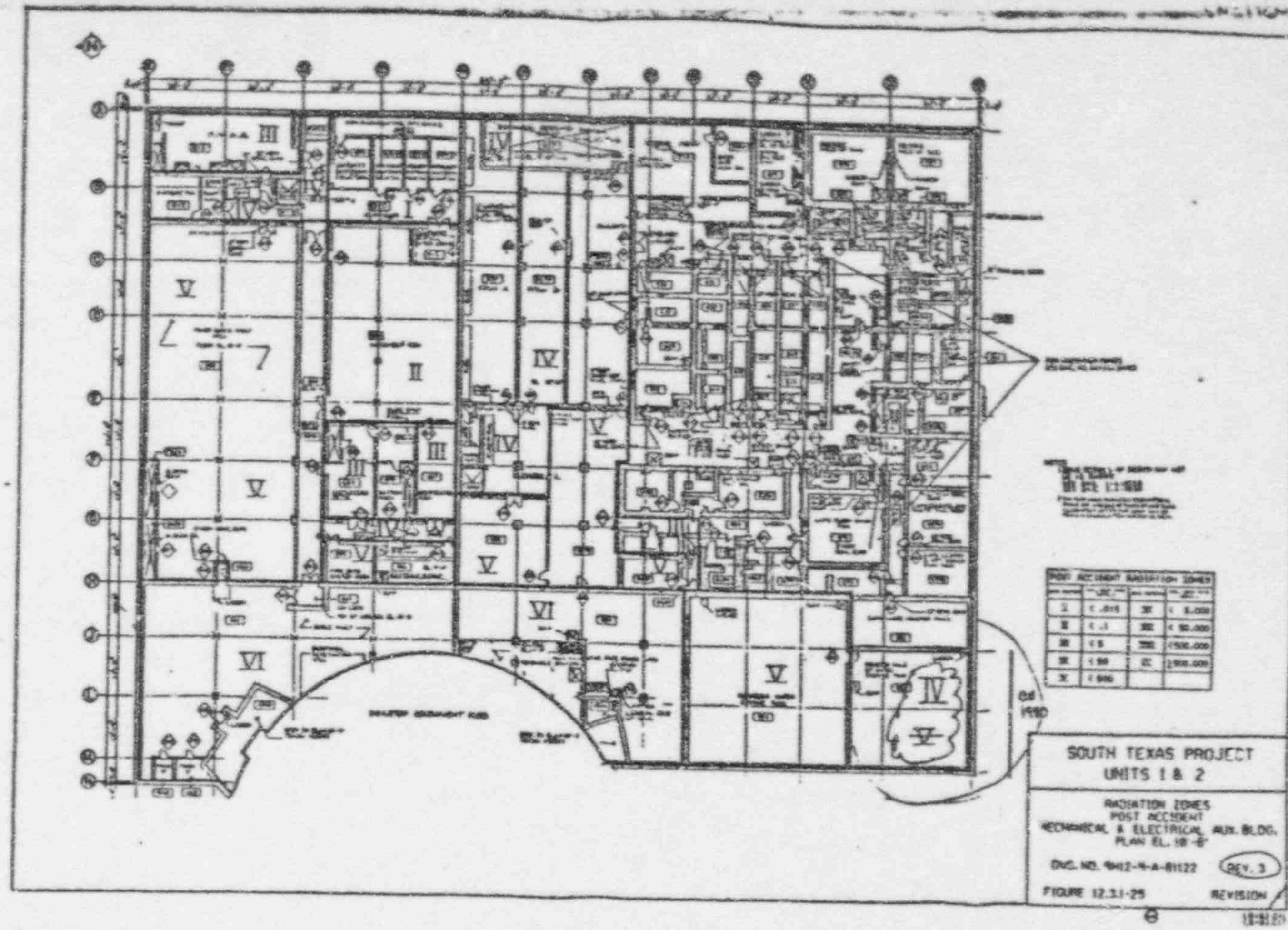
POST-ACCIDENT RADIATION LEVELS/DOSES

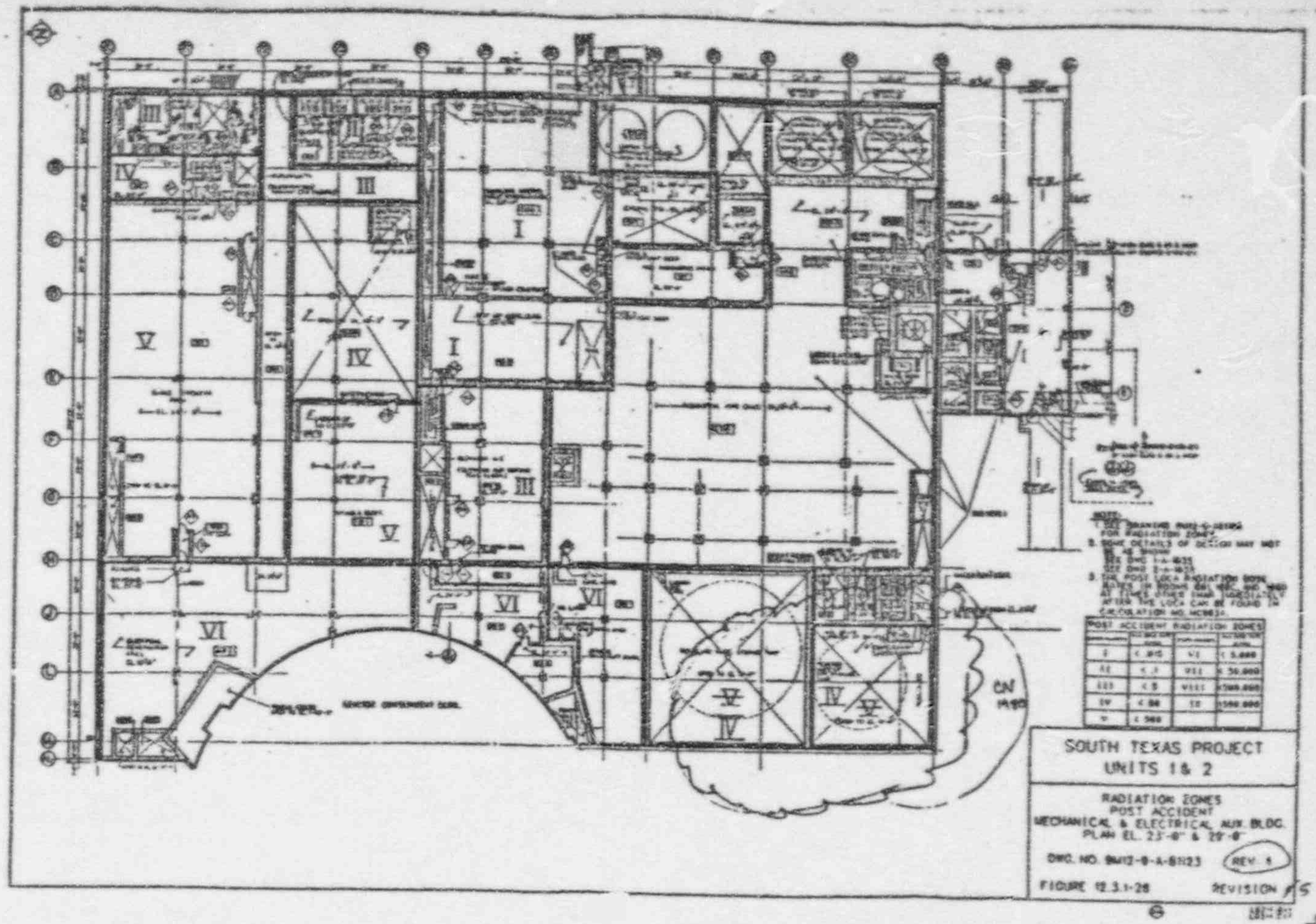
Continuous Occupancy Areas:	30 day Doses (Rem)			
	Gamma	Beta	Thyroid	
Control Room	2.43	21.50	22.67 ~~22.90~~	CN 1944 CN 193
Technical Support Center	4.85	24.55	28.62	CN 1944

Infrequent Access Areas:

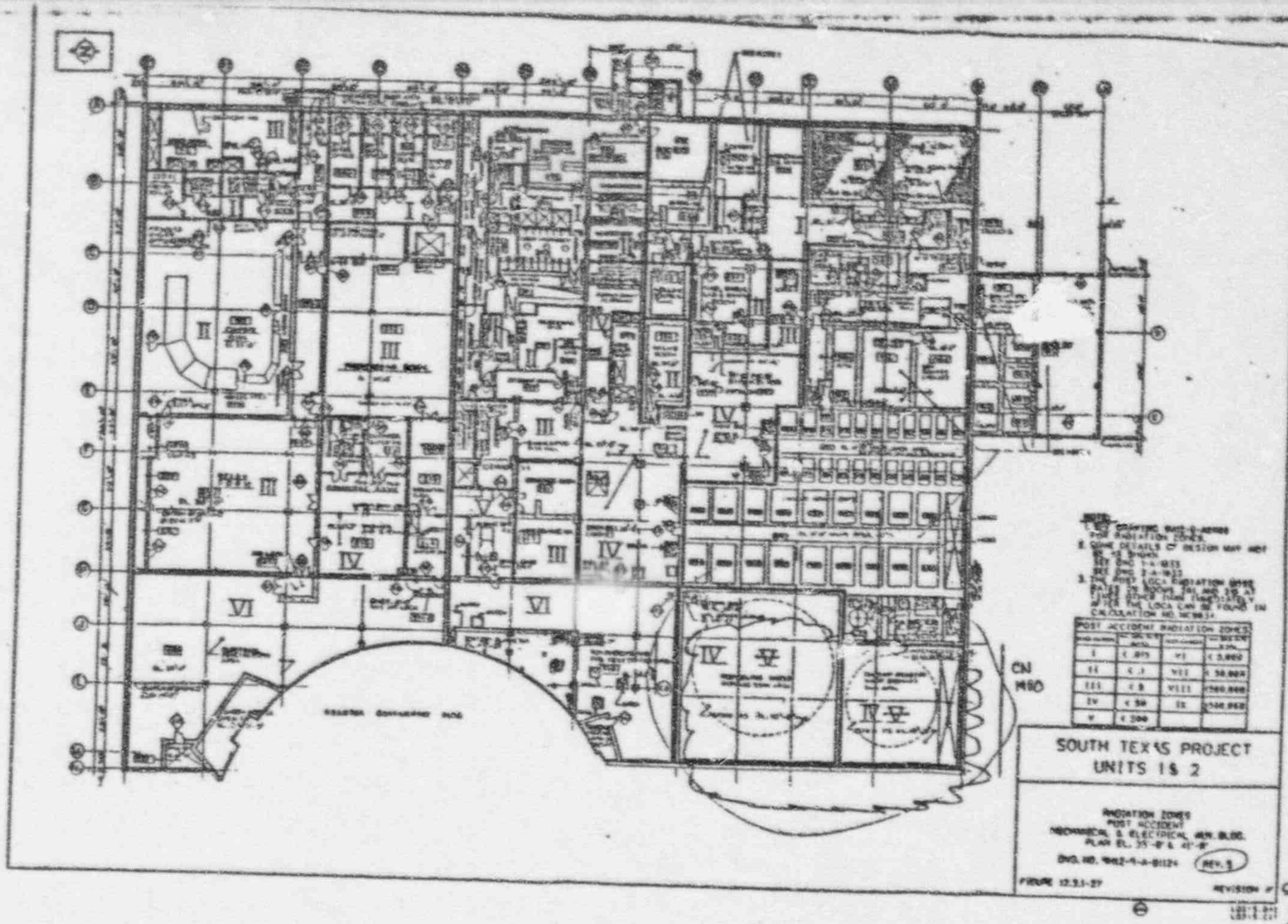
UFSAR Figure Reference	Area	Dose Rate (R/Hr) Time after accident 1 hr	1 day	1 wk	1 month
12.3.1-36	Post-accident sample station	.75	4.5×10^{-2}	1.1×10^{-2}	6×10^{-4}
12.3.1-27	Health Physics counting room	6×10^{-3}	3.6×10^{-4}	9×10^{-5}	4.8×10^{-6}
12.3.1-27	Radwaste counting room	3.1×10^{-2}	1.8×10^{-3}	4.6×10^{-4}	2.4×10^{-5}
12.3.1-28	Plant vent radiation monitor	4.74	.28	7.1×10^{-3}	3.8×10^{-3}
12.3.1-25	Auxiliary shutdown panel	8×10^{-4}	4.8×10^{-5}	1.2×10^{-5}	6.4×10^{-7}

CHANGE NOTICE 1944


POST ACCIDENT RADIATION ZONES
SOUTH TEXAS PROJECT
UNITS 1 & 2
RADIATION ZONES
POST ACCIDENT
MECHANICAL & ELECTRICAL AUX. BLDG.
PLAN EL. 10'-0"
DWG. NO. 9H12-9-A-01122
REV. 3
FIGURE 12.3.1-25
REVISION 5



POST ACCIDENT RADIATION ZONES
SOUTH TEXAS PROJECT
UNITS 1& 2
RADIATION ZONES
POST ACCIDENT
MECHANICAL & ELECTRICAL AUX. BLDG.
PLAN EL. 23'-0" & 29'-0"
FIGURE 12.3.1-28
REVISION 5



CN
M80
POST ACCIDENT RADIATION ZONES
SOUTH TEXAS PROJECT
UNITS 1 & 2
RADIATION ZONES
POST ACCIDENT
MECHANICAL & ELECTRICAL AUX. BLDG.
REV. 5
FIGURE 12.3.1-27
REVISION # 5

value once a DF of 100 is reached and no additional credit is taken for deposition after a DF of 200 is reached. For particulate iodine, a spray removal rate of 6.9 hr^{-1} is assumed until a DF of 50 is reached and it is then reduced to 10% of this value until a DF of 1000 is reached. 3

15.6.5.3.1.3 Containment Leakage Doses - Doses resulting from activity leakage from the Containment have been calculated using the models presented in Appendix 15.B. The thyroid, whole-body gamma and skin beta doses are presented in Table 15.6-11 for the EZB distance of 1,430 meters and the outer boundary of the LPZ at 4,800 meters.

15.6.5.3.2 ESF Leakage Contribution: A potential source of fission product leakage following a LOCA is the leakage from Engineered Safety Features (ESF) components which are located in the Fuel Handling Building (FHB). This leakage may be postulated to occur during the recirculation phase for long-term core cooling and Containment cooling by sprays. The water contained in the Containment sumps is used after the injection phase and is recirculated by the ECCS pumps and the Containment spray pumps.

~~An additional potential source of fission product leakage from ESF components is via valve seat leakage in isolation valves in the Low Head Safety Injection pump recirculation lines, the High Head Safety Injection pump recirculation lines, the Containment Spray System test lines, and the Refueling Water Storage Tank (RWST) suction line to the RWST. The RWST is vented to the Mechanical Auxiliary Building atmosphere.~~ 3

15.6.5.3.2.1 Fission Product Source Term - Since most of the radioiodine released during the LOCA would be retained by the Containment sump water, due to operation of the CSS and the ECCS, it is conservatively assumed that 50 percent of the core iodine inventory is introduced to the sump water to be recirculated through the external piping systems.

Because noble gases are assumed to be available for leakage from the Containment atmosphere and are not readily entrained in water, the noble gases are not assumed to be part of the source term for this contribution to the total LOCA dose.

15.6.5.3.2.2 Leakage Assumptions - The amount of water in the Containment sumps at the start of recirculation is the total of the RCS water and the water added due to operation of the engineered safeguards, i.e., the ECCS and CSS. This amount has been calculated to be 512,494 gallons. This value is conservatively low to maximize iodine concentration in the sump water.

ESF leakage into the FHB 3

The ECCS recirculation piping and components external to the Containment are designed in accordance with applicable codes and are described in Section 6.3. The CSS is described in Sections 6.2.2 and 6.5.2.

The maximum potential recirculation loop leakage is tabulated in Table 15.6-12. Each recirculation subsystem includes a high-head safety injection (HHSI) pump, a low-head safety injection (LHSI) pump, a residual heat exchanger, the Containment sump, and associated piping and valves. Thus three separate subsystems are provided for recirculation, as well as for injection, each of which is adequate for long-term cooling.

Since three redundant subsystems are available during recirculation, leakage for any component in any subsystem can be terminated by shutting down the LHSI and HHSI pump associated with that subsystem and by closing the appropriate pump suction and discharge isolation valves.

Maximum potential recirculation leakages are indicated in Table 15.6-12. The leakage rate assumed for dose calculation purposes is conservatively twice the leakage rate given in Table 15.6-12.

The iodine partition factor applicable for this leakage is assumed to be 0.1.

~~ESF Leakage into the RWST~~

~~The potential leakage from the valves given in Table 15.6-12 was used to calculate the maximum potential ESF leakage into the RWST. The total leakage from these valves is ten times their total design leak rate. Leakage of fission products past these valves is assumed to begin upon recirculation and continues at the same rate for the duration of the accident.~~

~~Transit times for the fission products to travel from the leaking valves to the RWST were calculated by using the valve leak rate and the volume of water in the piping to be displaced. The isolation valves on the Containment Spray System (CSS) test lines are physically closest to the RWST and leakage from these valves are conservatively calculated to reach the RWST 13.4 days following an accident. Once the leakage from the CSS test line isolation valves reaches the RWST, the combined leak rate from the LHSI and HHSI recirculation isolation valves and the CSS test line isolation valves is assumed to be flowing into the RWST.~~

~~Due to the large diameter piping involved, leakage from the RWST suction isolation valves would not reach the RWST before the motive force, i.e. the pressure in the RCB, is negligible. Therefore, leakage from these valves is not considered in the radiological analysis.~~ 3

~~Credit is taken for radioactive decay during the ECCS injection phase and the piping transit time to the RWST. No credit is taken for dilution in either the RWST or in the atmosphere of the Mechanical Auxiliary Building (MAB) or for the holdup time of fission products in the atmosphere of the MAB. Once the leakage flow reaches the RWST, it is assumed to be immediately dispersed into the environment.~~

~~The iodine partition factor applicable for this leakage is assumed to be 0.1.~~

15.6.5.3.2.3 ESF Leakage Doses - The iodine activity, once released to the atmosphere of the FHB, is assumed to be quickly transported by the ventilation system through the exhaust filters and released to the environment at ground level. The iodine filtration efficiency is assumed to be 95 percent.

CNF

~~Iodine activity which reaches the RWST is assumed to be immediately dispersed into the environment.~~ 3

For the first 30 minutes, the analyses consider the potential imbalance of exhaust flow, resulting in flow through one train of heaters being below the setpoint for heater energization (12,000 cfm through the three filter banks). During this time, the filter efficiency is assumed to be 90 percent for elemental iodine and 30 percent for organic iodine for that train of filters. Within 30 minutes, operator action to isolate that train of filters occurs. 2

The offsite doses due to the recirculation leakage are presented in Table 15.6-11 for the EZB of 1,430 meters for the initial two-hour period and the LPZ outer boundary distance of 4,800 meters for the 30-day duration of the accident.

15.6.5.3.3 Containment Purge Contribution: In the event of a LOCA coincident with the Containment supplementary purge system in operation, the purge is assumed to be isolated within 23 seconds following LOCA initiation. During normal power operation, the Containment supplementary purge system vents the containment at 5,000 ft3/min. However, for this analysis, the maximum flow rate due to the pressure spike inside the Containment was used (88,900 ft3/min for each purge line, intake and exhaust). The Containment purge system is described in Section 9.4. 2

The Containment airborne iodine inventory available for release is assumed to be the flashed portion of the total primary coolant iodine inventory based upon a preexisting iodine spike level of 60 $\mu Ci/g$ dose equivalent I-131. For noble gases, 100 percent of the primary coolant inventory based upon 1 percent failed fuel is assumed to be available for release. No failed fuel is assumed since isolation occurs prior to the core reaching a temperature which could cause a fuel failure.

15.6.5.3.3.1 Containment Purge Doses - The offsite doses calculated due to Containment purging are presented in Table 15.6-11 for the EZB of 1,430 meters and LPZ outer boundary distance of 4,800 meters.

15.6.5.4 Core and System Performance.

15.6.5.4.1 Mathematical Model: The requirements of an acceptable ECCS Evaluation Model are presented in Appendix K of 10CFR50 (Ref. 15.6-2).

Large Break LOCA Evaluation Model

The analysis of a large break LOCA transient is divided into three phases: blowdown, refill, and reflood. There are three distinct transients analyzed in each phase, including the thermal hydraulic transient in the RCS, the pressure and temperature transient within the Containment, and the fuel and clad temperature transient of the hottest fuel rod in the core. Based upon these considerations, a system of inter-related computer codes has been developed for the analysis of the LOCA.

The description of the various aspects of the LOCA analysis methodology is given in WCAP-8339 (Ref. 15.6-4). This document describes the major phenomena codes which ensure compliance with the acceptance criteria. The SATAN-VI, WREFLOOD, COCO, and LOCTA-IV codes, which are used in the LOCA analysis, are

TABLE 15.6-10 (Continued)

PARAMETERS USED IN ANALYSIS OF
LOSS-OF-COOLANT ACCIDENT OFFSITE DOSES

Parameter	
Activity assumed mixed in Containment sump water available for ESF leakage	
Noble gases	None
Iodines	50% core activity Table 15.A-1
ESF system leakage rate assumed into the FHB, cm^3/hr	Twice that of Table 15.6-12
ESF system leakage into the RWST assumed, cm^3/hr	Table 15.6-12
~~Transit time from leaking LHSI, HHSI, CSS isolation valves to the RWST~~	~~13.4 days~~
~~Transit time from leaking RWST suction isolation valves to the RWST~~	~~>30 days~~
Amount of water in which mixing of iodine occurs, gal	512,494
Iodine partition factor for leakage	0.1
FHB filtration efficiency, %	95
Supplementary purge rate, scfm (for each of two lines, intake and exhaust)	88,900
Time before isolation of purge, sec	23
Meteorology	5 percentile Table 15.B-1
Dose model	Appendix 15.B

CN 198

TABLE 15.6-11

DOSE RESULTING FROM LARGE BREAK LOSS-OF-COOLANT ACCIDENT

Parameter		
Containment Leakage Doses		
Exclusion Zone Boundary 0-2 hr		CN 1944
thyroid, rems	1.173×10^{3}	
whole body gamma, rems	2.26	
skin beta, rems	1.21	
Low Population Zone 0-30 days		
thyroid, rems	63.54	
whole body gamma, rems	7.3×10^{-3}	
skin beta, rems	4.7×10^{-3}	
ESF Leakage Doses (from leakage into the FHB)		
Exclusion Zone Boundary 0-2 hr		
thyroid, rems	2.54×10^{-3}	
whole body gamma, rems	8.03×10^{-4}	
skin beta, rems	2.27×10^{-4}	
Low Population Zone 0-30 days		
thyroid, rems	3.75×10^{-3}	
whole body gamma, rems	3.93×10^{-4}	
skin beta, rems	1.40×10^{-4}	
~~**ESF Leakage Doses (from leakage into the RWST)**~~		CN 198
~~Exclusion Zone Boundary, 0-2 hr~~		
~~thyroid, rems~~	~~0.0~~	
~~whole body gamma, rems~~	~~0.0~~	
~~skin beta, rems~~	~~0.0~~	
~~Low Population Zone, 0-30 days~~		
~~thyroid, rems~~	~~0.14~~	
~~whole body gamma, rems~~	~~3.60×10^{-5}~~	
~~skin beta, rems~~	~~1.30 x 10 [illegible]~~	
Containment Purging Doses		
Exclusion Zone Boundary 0-2 hr		CN 194
thyroid, rems	19.42	
whole body gamma, rems	1.0×10^{-2}	
skin beta, rems	7.6×10^{-3}	
Low Population Zone 0-30 days		
thyroid, rems	2.39 ~~2.40~~	CN 1980
whole body gamma, rems	1.23×10^{-3}	
skin beta, rems	9.30×10^{-4}	

 Revision 5

CHANGE NOTICE 1944

TABLE 15.6-11 (Continued)

DOSE RESULTING FROM LARGE BREAK LOSS-OF-COOLANT ACCIDENT

Total Doses

Exclusion Zone Boundary 0-2 hr	
thyroid, rems	1.37×10^2
whole body gamma, rems	2.27
skin beta, rems	1.22
Low Population Zone 0-30 days	
thyroid, rems	6.63 ~~6.65~~ $\times 10^1$
whole body gamma, rems	0.73
skin beta, rems	0.47

CN 1980

CN 1944

* Exclusion Zone Boundary is at 1,430 m. Outer boundary of Low Population Zone is at 4,800 m.

CHANGE NOTICE 1944

TABLE 15.6-12

MAXIMUM POTENTIAL RECIRCULATION LOOP LEAKAGE
EXTERNAL TO CONTAINMENT

3

Leakage into the FHB

Item	Leakage (cm3/hr)
Low-Head Safety Injection pumps	30
High-Head Safety Injection pumps	60
Valves	4,050
Total	4,140

Leakage into the RWST

Item	
Low-Head Safety Injection pump recirculation line isolation valves	
High-Head Safety Injection pump recirculation line isolation valves	
Containment spray system test line isolation valves	
Total leakage	1,740 cm^3/hr

3

CN 1980

APPENDIX 15.B

DOSE MODELS

This appendix describes the mathematical models and parameters used for the fission product transport from the postulated accident site to the environment and for the radiological dose calculations.

15.B.1 General Accident Parameters

This section describes the parameters used in analyzing the radiological consequences of postulated accidents. The site-specific, 5-percentile, short-term dispersion factors for the worst sector (assuming ground level releases) are given in Table 15.B-1. (See Section 2.3.4 for additional details on meteorology.) The breathing rates used are presented in Table 15.B-2. The thyroid (via inhalation pathway), beta skin, and gamma body (via submersion pathway) dose factors based upon Reference 15.B-3 are given in Table 15.B-3.

15.B.2 Offsite Radiological Consequences Calculational Models

This section presents the models and equations used for calculating the integrated activity released to the environment, the accident flow paths, and the equations for dose calculations. Two major release models are considered:

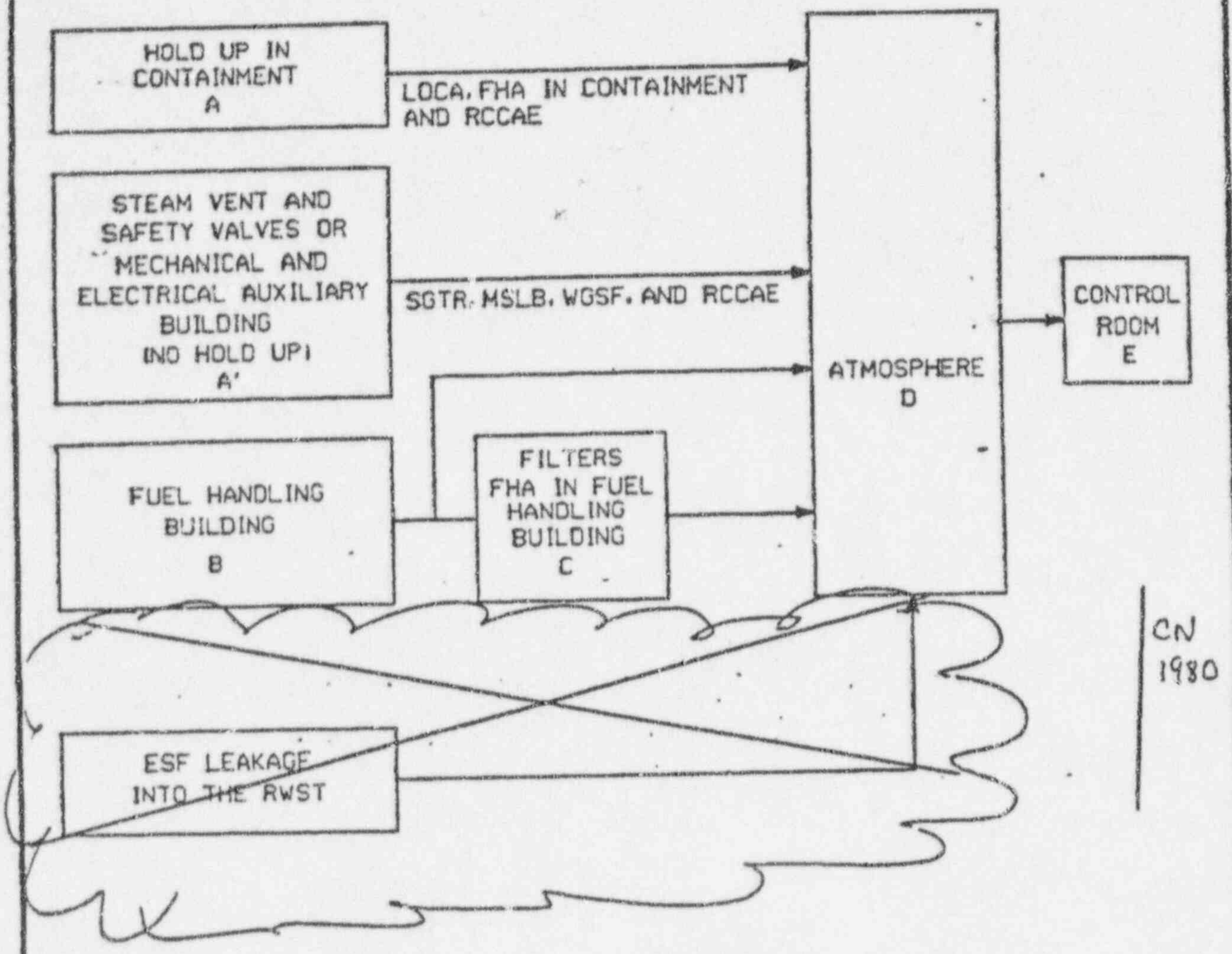
- A single holdup system with no internal cleanup
- A holdup system wherein a two-region spray model is used for internal cleanup

15.B.2.1 Accident Release Pathways. The release pathways for the major accidents are given in Figure 15.B-2. The accidents and their pathways are as follows:

1. Loss-of-Coolant Accident (LOCA)

   Immediately following a postulated LOCA, the release of radioactivity from the containment is to the environment with the containment spray and Engineered Safety Features (ESF) systems in full operation. The release in this case is calculated using Equations 15.B.2-6 and 15.B.2-7, which take into account a two-region spray model within the Containment. The release of radioactivity to the environment due to assumed ESF system leakages in the Fuel Handling Building (FHB) will be via ESF filters and is calculated using Equation 15.B.2-5.

   ~~The release of radioactivity to the environment due to assumed ESF leakages into the Refueling Water Storage Tank (RWST) is immediately dispersed into the environment and is calculated using Equation 15.B.2-5. However, no credit is taken for the holdup time of fission products in the atmosphere of the Mechanical Auxiliary Building (MAB).~~ 3



SOUTH TEXAS PROJECT
UNITS 1 & 2

RELEASE PATHWAYS

FIGURE 15.B-2

REVISION 5

L8411.831

SOUTH TEXAS PROJECT UNITS 1 & 2
HOUSTON LIGHTING & POWER
CALCULATION COVER SHEET

CALCULATION NUMBER: **NC-9004**

CALCULATION TITLE: **Post LOCA Radiation Zones and EQ Radiation**

SUBJECT: **Determination of Post LOCA Radiation Zones and EQ Radiation**

BUILDING/AREA/SYSTEMS: **Various**

DISCIPLINE: **NUCLEAR**

QUALITY CLASS: **4 (Safety Related)**

UNIT: **9 (Units 1 & 2)**

CALCULATION STATUS: **Final**

OBJECTIVE: **To determine the post accident (LOCA) radiation zones and doses for the plant. These doses will be used for equipment qualification (EQ) purposes (For EQ Design Criteria see TPNS # 4E019NQ1009, and UFSAR Chapter 3, Section 3.11).**

SCOPE: **The calculation is applicable to Units 1 & 2.**

RESULTS: **The results are given on sheets 9-22.**

TOTAL NUMBER OF SHEETS: **979**

REV NO.	9				
PREPARER	W. M. Blumberg WMBlumberg 10/4/95				
REVIEWER	S. F. Huang SF Huang 10/4/95				
SUPERVISING ENGINEER	10-4-95 D. E. Orr MA Whaley for				
DEPARTMENT DIRECTOR	10/4/95 D. A. Leazar D Leazar				
D.C. ISSUE DATE	10.9.95				

1

SOUTH TEXAS PROJECT UNITS 1 & 2
HOUSTON LIGHTING & POWER
CALCULATION COVER SHEET

CALCULATION NUMBER:	NC-6013
CALCULATION TITLE:	Control Room, TSC and Offsite LOCA Radiation Doses
SUBJECT:	Calculation of Control Room, TSC and Offsite Doses during a LOCA
BUILDING/AREA/SYSTEMS:	N/A
DISCIPLINE:	NUCLEAR
QUALITY CLASS:	4 (Safety Related)
UNIT:	9 (Units 1 & 2)
CALCULATION STATUS:	Final
OBJECTIVE:	To determine the Control Room, TSC and Offsite Doses during a LOCA.

Revision 9 removes the RWST backleakage dose contribution. Calculation MC-6458 (Ref. 68) shows that the RCS backleakage to the RWST will not contribute to the LOCA doses at 30 days. The minimum time for the backleakage to reach the RWST is 42 days.

SCOPE: The calculation is applicable to Units 1 & 2.

RESULTS: The results for shutoff of the CSS at approximately 6.3 hours are given on pages 9 and 10.

The doses for shutoff of the CSS at 30 an 45 minutes after the start of the accident are given on page M-22 and M-23. These are not the current design basis dose.

TOTAL NUMBER OF SHEETS: 1155

REV NO.	9				
PREPARER	W. M. Blumberg W M Blumberg 10/4/95				
REVIEWER	S. F. Huang SF Huang 10/2/95				
SUPERVISING ENGINEER	10-4-95 D. E. Coe M. C. W... for				
DEPARTMENT DIRECTOR	10/6/95 D. A. Leazar D. C. Leazar				
D.C. ISSUE DATE	10-9-95				