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Group 2, L

AAB INPUT TO VOGTLE SER

PLANT NAME: Alvin W. Vogtle Nuclear Plant
LICENSING STAGE: CP
DOCKET NUMBER: 50-424
RESPONSIBLE BRANCH: LWR 2-2
REQUESTED COMPLETION DATE: 2/8/74
REVIEW STATUS: AAB Review Complete

Enclosed is the Accident Analysis Branch input to the Vogtle SER covering Section 6.2.3, Air Cleanup Systems, and Section 15, Accident Analyses. Also enclosed is a revised Section 6.4.1 which is to be substituted for the same numbered section enclosed with my memorandum to you dated January 22, 1974.

/s/

Harold R. Denton, Assistant Director
for Site Safety
Directorate of Licensing

Enclosure:
As stated

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DATE➤	2/13/74					2

Section 6.2.3 - AIR CLEANUP SYSTEM

The containment spray system is also used for iodine removal from the containment atmosphere following a postulated LOCA. Sodium Hydroxide is added to the containment spray solution to enhance the iodine scrubbing function of the system. A sufficient quantity of NaOH will be injected to raise the equilibrium pH in the containment sump to a minimum value of 8.5. We have evaluated the containment spray and spray additive systems and found them effective for removal of elemental iodine, and iodine adsorbed on airborne particulate matter. The first order removal coefficients for elemental and particulate iodine are 10 hr^{-1} and 0.4 hr^{-1} respectively, in an estimated effective volume of 2.15×10^6 ft^3. The minimum sump pH of 8.5 is considered adequate to achieve and maintain a decontamination factor (DF) of 100 for elemental iodine.

6.4 HABITABILITY SYSTEM

6.4.1 CRITERION 19

The applicant proposes to meet General Design Criterion 19, Control Room, of Appendix A to 10 CFR Part 50, by use of adequate concrete shielding and by installing redundant 17,500 cfm recirculating charcoal filters in the control room ventilation system. An additional 1500 cfm charcoal filter train (also redundant) is provided for the purpose of prefiltering make-up air. These filters will be automatically activated upon an accident signal or high radiation signal. Air supply to the control room during normal operation will be supplied through pre-filters and high efficiency particulate air (HEPA) filters. We have concluded that the potential radiation dose to control room personnel following a LOCA would be within the guidelines of Criterion 19.

We have also evaluated the potential dose to control room personnel due to accidental release of radioactive materials from the AEC's Savannah River Plant. The closest reactor on the Savannah River Plant is 13 kilometers from Vogtle Plant, and the closest fuel processing facility is about 17 kilometers. In our evaluation of doses within the Vogtle control room as a result of an accident at the nearest SRP reactor we used the conservative release fractions of Regulatory Guide 1.4

based on a reactor power level normalized to 1000 MWt. We assumed that release occured through HEPA and charcoal filters which are 95% efficient for removal of iodines and 99% efficient for removal of particulates. We also assumed total release of the source term within a short time period and used instantaneous release (puff) meteorological diffusion conditions based on an equivalent Pasquill F and wind speed of 1.1 meters per second. Due to the distance and low wind speed assumptions, we allowed a 3.2 hour decay period for cloud transit time. The calculated dose to an occupant of the Vogtle control room would be well within the guidelines of Criterion 19 for control room habitability provided protective clothing and eye protection are provided for use by reactor operators during an emergency.

Based on information supplied by SRP in reference 3, we have calculated the concentration of plutonium within the control room due to an accidental release of 1.4 curies of a mixture of Pu-238 and Pu-239 at the nearest SRP fuel processing facility 17,000 meters distant. We assumed instantaneous release, no depletion of the cloud during transit, and that the plutonium was all in a soluble form. Based on the applicant's design which provides HEPA filtering of the control room's normal air intake, we assumed 99% removal of plutonium from the air entering the control room. We further assumed that the peak concentration existed for 2 hours. Based on these assumptions, the intake of plutonium by an occupant of the control room during the assumed 2-hour exposure time is calculated to be well within the intake which would accrue due to exposure for one year at the Part 20 limits of concentration for occupational exposure.

Based on information supplied by SRP in Reference 6, we have estimated the concentration at the Vogtle Plant site due to accidental releases of tritium from the processing facilities at the 200H Area, 18,000 meters distant. Assuming a 60 meter release height and no bouyancy effects, the peak short term value of X/Q calculated at the Vogtle Plant site due to a puff release is 3.0×10^{-9} per cubic meter. If the released tritium gas remains in the gaseous form, bouyancy is expected to increase the effective stack height by at least 100 meters or more. If the released tritium gas burns, the heated cloud is also expected to increase the effective stack height by about 100 meters. The short term X/Q for an effective stack height of 160 meters and a puff release is calculated to be 9×10^{-11} per cubic meter, and the resulting short term peak concentration of tritium in its oxide form at the Vogtle Plant would be about 1.4×10^{-3} uc/ml. Although this is about 300 times the value of the occupational MPC for tritium listed in Appendix B of Part 20, the cloud passage time and average cloud concentration is estimated to result in exposure of control room occupants to significantly less than the 2000 MPC-hours resulting from exposure to occupational MPC for one year, and therefore to be well within the guidelines of Criterion 19.

Although we have not independently confirmed all of the Accident assumptions which lead to the release terms provided by Savannah River, we believe that the systems proposed for the Vogtle plant provide assurance that even releases substantially greater than those postulated would not lead to a situation in which the Vogtle control room was not habitable. On the basis of the above, we conclude that the Vogtle Plant control room meets the guidelines of Criterion 19 provided that protective clothing is available within the control room to minimize the beta skin dose in the event of an accidental release of radioactive material from SRP; and that procedures for its use by control room occupants is provided.

1. Memorandum dated December 18, 1973 from N. Stetson, Manager, Savannah River Operations Office, to J. M. Hendrie, Deputy Director for Technical Review, Directorate of Licensing, AEC.

2. Savannah River memorandum DPST-73-508 dated December 4, 1973. (Enclosure to Reference 1 above)

3. Letter dated July 12, 1973 from N. Stetson, Manager, Savannah River Operations Office, to John F. O'Leary, Director, Directorate of Licensing, AEC.

4. DPST-73-330 "Presentation at the Meeting with Representatives of the Directorate of Licensing," April 24, 1973." May 9, 1973.

5. DP-1323, "The Savannah River Plant Site" January, 1973.

6. Letter dated February 1, 1974, from N. Stetson, Manager, Savannah River Operations Office, to J. M. Hendrie, Deputy Director for Technical Review, Directorate of Licensing, AEC, and attachment thereto.

7. Meteorology and Atomic Energy 1968, U. S. Atomic Energy Commission, July 1968.

SECTION 15   ACCIDENT ANALYSIS

We have selected five highly unlikely accidents (design basis accidents) that are representative of the spectrum of types and physical locations of postulated accidents that would involve the various engineered safety feature systems provided. We performed conservative analyses of these design basis accidents to assess the capability of the engineered safety features to control the possible escape of fission products from the facility. The design basis accidents analyzed were (1) loss-of-coolant, (2) refueling, (3) steam-line-break, (4) steam generator tube rupture, and (5) waste gas decay tank rupture.

On the basis of our experience with the evaluation of postulated accidents such as a steam line break, a steam generator tube rupture and waste gas decay tank rupture for PWR plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the permissible primary and secondary coolant system radioactivity concentrations and gas decay tank activities so that potential offsite doses are relatively small. We will include appropriate limits in the Technical Specifications in the context of our review of the operating license application.

Section 15 - LOSS OF COOLANT ACCIDENT

In our analysis of the consequences of a LOCA, we modelled the primary and secondary containment as a set of three control volumes, consisting of the region of the primary containment covered by containment spray, an unsprayed containment volume, and an effective secondary containment volume consisting of 50% of the free volume of the enclosure building. The primary containment was assumed to leak at the design leak rate of 0.3% per day for the first 24 hours, and 0.15% thereafter. During the first two minutes after the accident, all of this leakage is assumed to be released directly to the environment. After the first two minutes, Section 15 - LOSS-OF-COOLANT ACCIDENT

37.5% of the primary containment leakage was assumed to be released untreated to the environs, while the remainder is assumed to enter the enclosure building ventilation system, where it is filtered, a small fraction (1/100) exhausted, and the remainder circulated to the enclosure building. Technical Specifications will be written at the operating license stage for this bypass leakage fraction, the lines, valves, and penetrations which are the bypass leak paths, and the tests which will be performed to assure that this leakage is not exceeded. The applicant will be required to commit to the inclusion of such technical specifications and certify that tests are feasible to measure leakage through all bypass leakage paths prior to issuance of a construction permit.

Recirculation/filtration units similar to those of the enclosure building are provided for the electrical and piping penetration rooms, which results in fission product filtration and hold-up characteristics essentially the same for these three volumes comprising the secondary containment. For this reason, it is not necessary to specify the fraction of the leakage being treated by each system. The most significant parameters of our analysis of the consequences of a LOCA are tabulated in Table I. The results of the analysis are summarized in Table II.

Refueling Accident

In this accident, it is assumed that a fuel assembly is dropped during refueling operations and that, as a result of the fall, all of the fuel rods are damaged. Our analysis of the consequences are consistent with the conservative assumptions in Regulatory Guide 1.25. Activity released to the environs is assumed to be released through the fuel handling building ventilation filters within a two-hour period. The resultant calculated doses at the exclusion radius boundary are 12 rem thyroid and 1 rem whole body, and at the low population zone radius, 4 rem thyroid and 0.5 rem whole body.

TABLE I

LOCA Dose Calculation Input Parameters

Power	3565 MWt
Primary Containment Volumes	
Sprayed regions	2,145,000 ft^3
Unsprayed regions	605,000 ft^3
Enclosure Building Volumes	
Effective Mixing Volume	1,705,000 ft^3
Primary Containment Leak Rate	
0-24 hours	0.3% per day
24 hrs-720 hrs.	0.15% per day
Fraction of Primary Leakage Released untreated:	
0-2 min.	100%
2 min-24 hrs.	37.5%
24 hrs.-720 hrs.	37.5%
Enclosure Bldg. Filtration System:	
Filtration Flow Rate	20,000 cfm
Enclosure Bldg. Exhaust flow	2,000 cfm
Filter Efficiencies for Iodine Forms:	
Elemental	90%
Organic	70%
Particulate	90%

Table II

LOCA Dose Calculation Results

Thyroid Doses	
2 hr dose at site boundary	122 rem
30 day dose at LPZ	70 rem
Whole Body Doses	
2 hr dose at site boundary	7 rem
30 day dose at LPZ	8 rem

TABLE III

REFUELING ACCIDENT

Shutdown Time	100 hours
Total Number of Fuel Rods in the Core	50,952
Number of Fuel Rods Involved in the Refueling Accident	264
Power Peaking Factor	1.65
Iodine Fractions Released from Pool	
Elemental	75%
Organic	25%
Filter Efficiencies	
Elemental	90%
Organic	70%

X/Q Values, Sec/M^3

0 - 2 hours @ 1098 meters	2.8×10^{-4}
0 - 2 hours @ 3220 meters	1.0×10^{-4}

Spray removal coefficients:

elemental iodine	10 hr^{-1}
organic iodine	0
particulate iodine	0.4 hr^{-1}

X/Q values:

0-2 hrs. @ site boundary	2.8×10^{-6}
0-8 hrs @ LPZ	1.0×10^{-6}
9-24 hrs @ LPZ	2.1×10^{-5}
24-96 hrs @ LPZ	8.7×10^{-6}
96-720 hrs @ LPZ	2.5×10^{-6}