SEP 1 8 1975

R. C. DeYoung, Assistant Director for Light Water Reactors,
Group 1, RL

WPPSS 3, 5 ADDITIONAL SER INPUT FROM AAB

PLANT NAME: WPPSS 3, 5
LICENSING STAGE: ~~CP~~
DOCKET NUMBER: 50-508/509
MILESTONE NUMBER: 24-31
RESPONSIBLE BRANCH: LWR 1-3; P. O'Keilly, LPM
REQUESTED COMPLETION DATE: August 29, 1975
DESCRIPTION OF RESPONSE: AAB SER Input Complete

Enclosed is additional input material for the following sections of the WPPSS 3, 5 Safety Evaluation Report:

6.2.3 Engineered Safety Feature Air Filtration Systems

6.4 Habitability Systems

15.0 Radiological Consequences of Accidents.

The design basis loss-of-coolant accident whole body dose of 29 rem which we calculate at the exclusion area boundary does not meet the exposure guideline for a plant at the construction permit review stage. The primary reason for the high whole body dose is that the applicant has not yet completed an analysis of the post-LOCA shield building ventilation system (SBVS) flow rates and consequently we have assumed full exhaust flow, with no credit for mixing and holdup in the shield building annulus, for the majority of the accident transient.

To meet the guideline value, we will require the applicant to reduce the containment design leak rate from 0.2% per day to 0.13% per day. When the applicant completes his analysis of the operation of the SBVS, we will re-evaluate the LOCA doses and, if warranted, consider an adjustment to the containment design leak rate.

Original Signed by
H. R. Denton

Harold R. Denton, Assistant Director
for Site Safety
Division of Technical Review
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: See page 2

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cc: w/o enclosure
R. Boyd
W. McDonald
J. Panzarella

w/enclosure
S. Hanauer
R. Heineman
TR A/D's
TR T/C's
SS B/C's
O. Parr
P. O'Reilly
K. Murphy
E. Adensam
F. Kantor

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SURNAME	F.Kantor/vg	B.Grimes	H.Denton			
DATE	9/18/75	9/18/75	9/18/75			

6.2.3 Engineered Safety Feature Air Filtration Systems

The engineered safety feature (ESF) air filtration systems proposed for the WPPSS 3, 5 plant are the shield building ventilation system, the ECCS area exhaust system, and the control room emergency filtration system. We have evaluated these systems with respect to the positions stated in Regulatory Guide 1.52 and have found that they are in agreement with these positions. We have therefore credited the ESF air filtration systems with adsorption efficiencies of 99% for removal of elemental, organic, and particulate iodines in our calculations of the radiological consequences of design basis accidents, as discussed in Section 15.0 of this report.

6.4 Habitability Systems

The emergency protective provisions of the control room related to the accidental release of radioactivity or toxic gases are evaluated in this section. Relevant portions of the control room ventilation system are described here.

6.4.1 Radiation Protective Provisions

The applicant proposes to meet General Design Criterion 19, Control Room, of Appendix A to 10 CFR Part 50, by use of concrete shielding and by installing a redundant 6000 cfm charcoal filter having a bed depth of 6 inches of charcoal.

6.4.1 Radiation Protective Provisions (Cont.)

Upon receipt of a high radiation signal at the intake louvers or a containment isolation signal, all outside air intake and exhaust valves close, isolating the control room. These signals also start both charcoal filters. Each filter processes 6000 cfm of control room air, adsorbing any radioactive iodine that may enter the control room. One of the two filters may be manually shut down and placed on standby.

The control room is of low leakage design to limit infiltration of contamination. It will be tested periodically to assure that in-leakage will not exceed 400 cfm at -0.25 inch water gauge.

Our independent calculations of the potential radiation doses to control room personnel following a LOCA show the resultant doses to be within the guidelines of Criterion 19.

6.4.2 Toxic Gas Protective Provisions

Control room habitability following a postulated toxic gas release is essential to ensure that operators can maintain the plant in a safe condition.

Using the procedures described in Regulatory Guide 1.78, chlorine was identified as being the only substance potentially hazardous to the control room operators. The applicant has provided protective provisions against a chlorine release including quick-acting chlorine detectors, automatic isolation of the control room, and breathing

6.4.2 Toxic Gas Protective Provisions (Cont.)

apparatus capable of protecting personnel for extended periods of time. These provisions comply with Regulatory Guide 1.95 and are acceptable.

15.0 Radiological Consequences of Accidents

The postulated design basis accidents analyzed by the applicant to determine the offsite radiological consequences are the same as those analyzed for previously licensed PWR plants. These include a loss-of-coolant accident (LOCA), a steam line break accident, a steam generator tube rupture, a fuel handling accident, a rupture of a radioactive gas storage tank, and a control rod ejection accident. We have reviewed these accidents and further evaluated the LOCA and the fuel handling accident. The offsite doses we calculated for these accidents and the assumptions used in the analyses are given in Sections 15.1 and 15.2 of this report.

On the basis of our experience with the evaluation of the steam line break and the steam generator tube rupture accidents for PWR plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the permissible reactor coolant system and secondary coolant system radioactivity concentrations.

At the operating license stage of review, we will include limits in the technical specifications on the reactor coolant system and secondary coolant system activity concentrations such that the potential 2-hour

15.0 Radiological Consequences of Accidents (Cont.)

doses at the exclustion radius, as calculated by the staff for these accidents, will be small fractions of the guideline doses of 10 CFR Part 100. Similarly, we will include limits in the technical specifications on gas decay tank activity so that any single failure such as lifting and failure to close of a relief valve will not result in doses that are more than a small fraction of the 10 CFR Part 100 guideline values.

The control rod ejection accident will also be evaluated at the operating license stage of our review. A technical specification will limit the allowable operational leakage of reactor coolant into the steam generator secondary side to assure that the radiological consequences of this accident will be well within the dose guidelines of 10 CFR Part 100.

15.1 Loss-of-Coolant Accident Dose Analysis

The WPPSS 3, 5 pressurized water reactors are each surrounded by a double containment structure consisting of a low leakage steel containment vessel and an outer reinforced concrete shield building to minimize the offsite radiological consequences of the design basis loss-of-coolant accident (LOCA). The applicant has specified a design leak rate for the primary containment of 0.2% per day for the first 24 hours following the LOCA and 0.1% per day for the duration of the accident. For dose evaluation purposes, radio-

15.1 Loss-of-Coolant Accident Dose Analysis (Cont.)

active materials that leak from the primary containment following a postulated LOCA can take any of the following pathways to the environment:

1. Leakage to the annulus between the primary and secondary containment structures (the shield building annulus) which will be treated by the shield building ventilation system.

2. Leakage to the ECCS area of the reactor auxiliary building which will be treated by the ECCS area exhaust system.

3. Direct bypass leakage which will not be treated.

Both the shield building ventilation system (SBVS) and the ECCS area exhaust system are engineered safety features.

The applicant has determined the allowable percentages of the primary containment leakage for each leakage pathway based on a limiting dose criterion of 150 rem to the thyroid at the exclusion area boundary. These allowable leakage pathway percentages are 40% of the primary containment leakage to the shield building annulus, 57.2% to the ECCS area of the reactor auxiliary building, and 2.8% direct bypass leakage. We have used these leakage pathway percentages to calculate the LOCA doses shown in Table 15.1. Other assumptions used in our analysis are given in Table 15.2.

With a primary containment design leak rate of 0.2% per day, we calculate a 2-hour exclusion area boundary thyroid dose of 149 rem.

15.1 Loss-of-Coolant Accident Dose Analysis (Cont.)

This dose meets the 150 rem thyroid dose guideline value and is in close agreement with the applicant's calculated dose. However, our 2-hour whole body dose of 29 rem at the exclusion area boundary exceeds the 20 rem guideline value for a plant at the construction permit review stage, as stated in Regulatory Guide 1.4. The applicant calculated a 2-hour whole body dose less than the 20 rem guideline value because, contrary to the staff's dose model for site evaluation purposes, the beta and gamma doses were not added in computing the whole body dose.

A primary reason for the high computed whole body dose is the assumption made with regard to the operation of the SBVS throughout the course of the accident. The applicant has not yet completed his analysis of SBVS exhaust and recirculation flow rates following a LOCA. Consequently, we and the applicant have assumed that, following the initial full recirculation phase which ends at 430 seconds (7.2 minutes) after the start of the accident, the SBVS operates at full exhaust for the duration of the LOCA. It is conservatively assumed that all of the primary containment leakage into the shield building annulus during this period is exhausted directly to the atmosphere through the SBVS filters and no credit is taken for mixing or holdup in the annular volume. In effect, this is equivalent to a direct bypass leakage pathway for the noble gases which are not removed by filtration and are the primary contributors to the whole body dose. Another reason for the high whole body dose is the large percentage

15.1 Loss-of-Coolant Accident Dose Analysis (Cont.)

(57.2%) of the primary containment leakage which is assumed to go to the ECCS area of the auxiliary building. As no credit is taken for mixing or holdup in the auxiliary building, this pathway is also equivalent to direct bypass leakage for the noble gases.

To meet the dose guidelines of Regulatory Guide 1.4, we will require the applicant to reduce the primary containment design leak rate from 0.2% per day to 0.13% per day. In addition, we will require the applicant to determine the allowable percentages of the primary containment leakage for each leakage pathway using as a limiting dose criterion the beta plus gamma radiation whole body dose model. When the applicant completes his analysis of the post-LOCA operation of the SBVS and the primary containment leakage pathway percentages, we will re-evaluate the LOCA doses and, if warranted, consider adjustments to the containment design leak rate.

15.1 Loss-of-Coolant Accident Dose Analysis (Cont.)

As part of the loss-of-coolant accident (LOCA), we have considered the consequences of leakage of containment sump water which is circulated by the ECCS outside the containment after a postulated LOCA. We have assumed the sump water to contain a mixture of iodine fission products in agreement with Regulatory Guide 1.7. At the time of the recirculation mode of operation, about 2200 seconds after the accident, the sump water is circulated outside of the containment to the Reactor Auxiliary Building to be cooled. If a source of leakage should develop, such as from a pump seal, a portion of the iodine would become gaseous and would exit to the atmosphere. As the ECCS area is served by an engineered safety feature air filtration system, we find that the offsite doses from possible equipment leakage would be within the guidelines of 10 CFR Part 100, even for substantial amounts of leakage.

The applicant will provide redundant hydrogen recombiners for the purpose of controlling any accumulation of hydrogen within the primary containment after a design basis LOCA. In the event both recombiners fail, the applicant has provided a backup purge system which discharges to the shield building annulus and subsequently to the atmosphere through the SBVS filters. Assuming operation of the SBVS at full exhaust and with no credit for mixing or holdup in the annulus, we have computed the additional dose an individual might receive due to purging the containment after the accident. The calculated doses are shown in Table 15-1

15.1 Loss-of-Coolant Accident Dose Analysis (Cont.)

and the assumptions used in the analysis are listed in Table 15.3. We find that the calculated doses from purging at the low population zone distance, when added to the LOCA doses, are well within the guidelines of 10 CFR Part 100.

15.2 Fuel Handling Accident

For the analysis of the fuel handling accident, we have assumed that a fuel assembly was dropped in the fuel pool during refueling operations and that all of the fuel rods in the assembly were damaged thereby releasing the volatile fission gases from the fuel rod gap into the pool. The radioactive material that escaped from the fuel pool was assumed to be released to the environment over a two-hour time period with the iodine activity reduced by filtration through the fuel building exhaust system. The dose results are shown in Table 15.1 and the assumptions and parameters used in the analysis are shown in Table 15.4. The dose model and dose conversion factors employed in the analysis were in agreement with those given in Regulatory Guide 1.25.

TABLE 15.1

RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS

	Exclusion Area* 2-Hour Dose, Rem		Low Population Zone** 30-Day Dose, Rem	
Accident	Thyroid	Whole Body	Thyroid	Whole Body
Loss-of-Coolant				
Case 1 (0.2%/day leak rate)	149	29	130	10
Case 2 (0.1%/day leak rate)	97	19	85	3
Hydrogen Purge	---	---	30	2
Fuel Handling	3	5	---	---

* Exclusion area boundary distance = 1,310 meters

** Low population zone distance = 4,830 meters (3 miles)

TABLE 15.2

ASSUMPTIONS USED IN THE CALCULATION OF LOSS-OF-COOLANT ACCIDENT DOSES

Power Level	4100 MWt	
Operating Time	3 Years	
Fraction of Core Inventory Available for Leakage		
Iodines	25%	
Noble Gases	100%	
Initial Iodine Composition in Containment		
Elemental	91%	
Organic	4%	
Particulate	5%	
Primary Containment Volume	3.7×10^6 ft^3	
Shield Building Annulus Volume	7.5×10^5 ft^3	
Mixing Fraction in Annulus	50%	
Primary Containment Leak Rate	Case 1	Case 2
0-24 Hours	0.2%/day	0.1%/day
>24 Hours	0.1%/day	0.05%/day
Primary Containment Leak Paths		
To Shield Building Filters	40%	
To ECCS Area Exhaust Filters	57.2%	
Direct to Atmosphere (Bypass)	2.8%	

Shield Building Ventilation Flow Distribution

Time Step	Recirculation Flow, cfm	Exhaust Flow, cfm
0-30 sec	0	0
30-250 sec	0	10,000
250-430 sec	10,000	0
430 sec-30 days	0	10,000

Shield Building and ECCS Area Iodine Filter Efficiencies 99%

TABLE 15.2 (Cont'd.)

Minimum Exclusion Area Boundary Distance	1,310 m
Low Population Zone Distance	4,830 m
X/Q Values	
0 - 2 hours @ EAB	1.0×10^{-3} sec/m^3
0 - 8 hours @ LPZ	1.0×10^{-4} sec/m^3
8 - 24 hours @ LPZ	6.7×10^{-5} sec/m^3
1 - 4 days @ LPZ	2.8×10^{-5} sec/m^3
4 - 30 days @ LPZ	7.7×10^{-6} sec/m^3

TABLE 15.3

HYDROGEN PURGE DOSE INPUT PARAMETERS

Power Level	4100 MWt
Containment Volume	3.7×10^6 ft^3
Holdup Time in Containment Prior to Purge Initiation	16 Days
Purge Duration	30 Days
Purge Rate	50 scfm
SBVS Filter Efficiency for Iodines	99%
4-30 Day χ/Q at 4,830 meters	7.7×10^{-6} sec/m^3

TABLE 15.4

ASSUMPTIONS USED IN THE FUEL HANDLING

ACCIDENT ANALYSIS

Power Level	4100 MWt
Number of Fuel Rods Damaged	236
Total Number of Fuel Rods in Core	56,876
Radial Peaking Factor of Damaged Rods	1.65
Shutdown Time	72 Hours
Inventory Released From Damaged Rods (Iodines and Noble Gases)	10%
Pool Decontamination Factors	
Iodines	100
Noble Gases	1
Iodine Fractions Released From Pool	
Elemental	75%
Organic	25%
Filter Efficiency for Iodine Removal	99%
0-2 Hour χ/Q Value at 1,310m	1.0×10^{-3} sec/m^3