DISTRIBUTION:
Central Files
NRR/Reading File
AAB/Reading File

FEB 3 1976

Richard C. DeYoung, Assistant Director for Light Water Reactors, DPM

WPPSS 3, 5 SER INPUT FROM AAB

PLANT NAME: WPPSS 3, 5
LICENSING STAGE: CP
DOCKET NUMBER: STN 50-508-509
RESPONSIBLE BRANCH: LWR #3; A. Bournia, LPM
MILESTONE NUMBER: 24.31
REQUESTED COMPLETION DATE: January 5, 1976
REVIEW STATUS: AAB SER Input Complete

Enclosed are Sections 6.2.3, Containment Air Purification and Cleanup Systems (Spray), and 15.0, Radiological Consequences of Accidents, for the WPPSS 3, 5 SER from the Accident Analysis Branch. The information contained in Amendment 28, dated January 20, 1976, was included in our review.

Richard H. Vollmer, Assistant Director
for Site Analysis
Division of Site Safety and
Environmental Analysis
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: w/o enclosure
R. Boyd
R. Heineman
V. Stello
W. McDonald
J. Panzarella
P. Shuttleworth

w/enclosure
S. Hanauer
H. Denton
D. Crutchfield
SE A/Ds
SA B/Cs
R. Klecker
S. Varga
D. Eisenhut
M. Williams
J. Miller
O. Parr
A. Bournia
W. Pasedag
F. Kantor

OFFICE	SE:AAB	SE:AAB	SE:AAB	SE:AD:SA		
SURNAME	F.Kantor:dr	W.Pasedag	D.Bunch	R.Vollmer		
DATE	2- 2 -76	2- 2 -76	2- 2 -76	2- 3 -76		

6.2.3 Containment Air Purification and Cleanup Systems

The containment spray system is used for fission product scrubbing following a postulated LOCA. Sodium hydroxide is added to the spray solution to enhance the system's effectiveness for elemental iodine removal. The method of NaOH injection chosen by the applicant differs from conventional designs in that a pre-pressurized additive tank is used as the motive force for additive injection, as opposed to additive pumps or eductors. It is essential to demonstrate adequate performance of this design by a pre-operational test program. The pre-operational testing described in the PSAR is inadequate for this purpose, as it is designed to demonstrate the operability of individual components of the system, as opposed to an integrated test of the spray and spray additive systems, to verify that the system, as installed, is capable of producing the pH values assumed in the system evaluation. The applicant has made a verbal commitment to devise a system test to verify the adequacy of the design. We find this acceptable, subject to documentation in a PSAR amendment.

Subject to verification of the system design by test, we find that the pH values quoted in the PSAR meet our requirements, i.e., a spray solution pH between 9.0 and 11.0 during the additive injection phase, and a long term containment sump solution pH above 8.5. Spray solutions with these pH values have been shown effective for elemental iodine removal and retention. We calculate removal coefficients of 10.0 and 0.7 hrs^{-1} for the elemental and particulate forms of iodine, respectively, in an estimated sprayed region comprising 81% of the total free volume of the containment.

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 two-hour doses at the exclusion 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

15.0 Radiological Consequences of Accidents (Cont.)

lifting or 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 36 hours following the LOCA and 0.1% per day for the duration of the accident. For dose evaluation purposes, radioactive 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.

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

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 leakage pathway percentages to be 40% of the primary containment leakage to the shield building annulus, 50% to the ECCS area of the reactor auxiliary building, and 10% direct bypass leakage. We have used these leakage pathway percentages in our calculation of the design basis LOCA doses. The results of our calculations are shown in Table 15.1 and the assumptions used in the analysis are listed in Table 15.2. The doses we calculate for the LOCA are well within the guideline dose values of Regulatory Guide 1.4 for a plant at the construction permit review stage (150 rem thyroid and 20 rem whole body).

In modeling the releases through the shield building annulus pathway, we conservatively assumed that the SBVS operates at full exhaust throughout the course of the accident following the initial pressure transient in the annulus. In actuality, the SBVS will switch from full exhaust to full recirculation in numerous short time steps in order to control the pressure buildup in the shield building annulus following the accident. This mode of operation will provide additional holdup time for the fraction of the primary containment leakage which enters the shield building annulus.

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 at the low population zone distance from purging, 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 gaps 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 are in agreement with those given in Regulatory Guide 1.25.

TABLE 15.1

RADIOLOGICAL CONSEQUENCES OF

DESIGN BASIS ACCIDENTS

	Exclusion Area[1] 2-Hour Dose, Rem		Low Population Zone[2] 30-Day Dose, Rem	
Accident	Thyroid	Whole Body[3]	Thyroid	Whole Body[3]
Loss-of-Coolant	74	15	25	5
Hydrogen Purge	--	--	30	1
Fuel Handling	3	3	--	---

1. Exclusion area boundary distance = 1,310 meters

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

3. Dose from low penetrating beta radiation considered as a skin dose and not included in whole body dose.

TABLE 15.2

ASSUMPTIONS USED IN THE CALCULATION OF

LOSS-OF-COLLANT 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%
Shield Building Annulus Volume Between Upper and Lower Elevation of SBVS Headers	4.3×10^5 ft^3
Mixing Fraction in Annulus	50%
Primary Containment Leak Rate	
0-36 hours	0.2% per day
> 36 hours	0.1% per day
Primary Containment Leak Paths	
To Shield Building Filters	40%
To ECCS Area Exhaust Filters	50%
Direct to Atmosphere (Bypass)	10%

Shield Building Ventilation System Flow Distribution

Time Step	Recirculation Flow, cfm	Exhaust Flow, cfm
0-30 sec	0	0
30-161 sec	[illegible],000	0
161 sec - 30 days	0	10,000

Shield Building and ECCS Area Iodine Filter Efficiencies	
Elemental Iodine	99%
Organic Iodine	99%
Particulate Iodine	99%
Primary Containment Volumes	
Sprayed Volume	2.75×10^6 ft^3
Unsprayed Volume	6.5×10^5 ft^3

TABLE 15.2 (Cont'd.)

Containment Spray System		
Removal Coefficients		
Elemental Iodine		$10\ hr^{-1}$
Organic Iodine		0
Particulate Iodine		$0.7\ hr^{-1}$
Mixing Rate Between Sprayed and Unsprayed Volumes		21,500 cfm
Elemental Iodine Spray Decontamination Factor		100
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	3.4×10^6 ft^3
Holdup Time in Containment Prior to Purge Initiation	16 Days
Purge Duration	30 Days
Purge Rate	50 scfm
SBVS Filter Efficiency for Iodine	99%
4-30 Day X/Q at 4,830 meters	7.7×10^{-6} sec/m^3

TABLE 13.4

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