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Regulatory File Cy.

April 28, 1977

RECEIVED MAY 5 1977 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. Olan D. Parr, Chief
Light Water Reactors Branch III
Division of Project Management

Gentlemen:

REFUELING ACCIDENT INSIDE CONTAINMENT
NO. 2 UNIT
SALEM NUCLEAR GENERATING STATION
DOCKET NO. 50-311

PSE&G was requested by your letter of March 16, 1977 to provide an evaluation of the potential consequences of a refueling accident inside the No. 2 Unit Containment Building of the Salem Nuclear Generating Station. Your request specified that our evaluation should utilize assumptions comparable to those given in Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," assuming the worst single failure. This evaluation is provided in Enclosure 1 and clearly indicates that potential site boundary radiation exposures resulting from the postulated refueling accident are well within 10CFR Part 100 Guidelines.

Enclosure 2 provides responses to your specific questions concerning containment isolation capability during a refueling accident. It should be noted that these responses address the Salem No. 1 Unit (Docket No. 50-272) and were submitted on March 30, 1977. This information has been provided since the No. 2 Unit equipment involved is similar to that installed in

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No. 1 Unit and it is therefore expected that the Technical Specification requirements will also be similar. The enclosed drawings, however, are specifically for the No. 2 Unit.

Should you have any additional concerns regarding this evaluation, please do not hesitate to contact us.

Very truly yours,

R. L. Mittl
General Manager -
Licensing and Environment
Engineering and Construction

Encl.

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ENCLOSURE 1

REFUELING ACCIDENT INSIDE CONTAINMENT
NO. 2 UNIT
SALEM NUCLEAR GENERATING STATION

(All Doses in rem)

	Minimum Exclusion Distance	Low Population Zone Distance
	1270 meters	8045 meters
Whole Body Dose	.98	.089
Thyroid Dose	59.9	5.49

Assumptions

100 hr. Hold-up time Before Fuel Transfer
(Proposed Technical Specification Requirement)
Meteorology as Calculated by NRC Staff in Salem
Safety Evaluation Report, Section 2.3, October 11, 1974
Semi-infinite Cloud Dose Model as Defined in Regulator
Guide 1.25.

No isolation

Breathing Rate = 3.47E-4 m^3/sec
No credit for charcoal filtration
Iodine DF = 100
Fuel rod gap activities calculated using assumptions
provided in Regulatory Guide 1.25:

a) Axial peaking factor of 1.72
b) Fuel Rod $\geq$ 23 feet below pool surface
c) Ratio of gap activity to total
   KR85 .3:1
   All other noble gaseous .1:1
   Iodines .1:1
d) 17 x 17 array
e) End of fuel cycle (650 days @ 3558 MWt)
f) Highest rated discharged assembly
g) Release of all gap activity in damaged assembly (other assumptions and a table of the calculated activities provided in Appendix I of the Salem FSAR)

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ENCLOSURE 2

INFORMATION NEEDED TO EVALUATE CONTAINMENT ISOLATION CAPABILITY DURING REFUELING ACCIDENT

The following are the responses to USNRC questions pertaining to the evaluation of a fuel handling accident (FHA) inside the Containment Building.

Q.1) Describe all instrumentation which would detect a fuel handling accident (FHA) inside containment. Your responses should include the following information:

a) Instrumentation function, e.g., close containment isolation valves;

b) Type of instruments and setpoints, e.g., mr/hr, and normal background reading;

c) Safety class, redundancy, power sources, and technical specification requirements;

d) A description of instrument response following a FHA taking into account instrument location;

e) Response time for the instrument to signal containment isolation after the FHA.

A.1) A fuel handling accident (FHA) inside the Containment would be detected by the containment and plant vent radiation monitors. A high radiation signal from any of these monitors will initiate automatic closure of the containment isolation valves, which are part of the Containment Purge and Pressure-Vacuum Relief System. These valves are designated as 1VC1, 1VC2, 1VC3, 1VC4, 1VC5 and 1VC6 on Figure 5.3-1 of the Salem FSAR. The Pressure-Vacuum Relief System serves to limit differentials between the Containment Building pressure and atmospheric pressure, whereas the Containment Purge System serves to supply fresh air to and vent the Containment atmosphere. The radiation monitors can monitor either the Containment atmosphere or the plant vent; automatic closure of the Containment isolation valves will occur when a high radiation alarm is received from the selected source, although the plant vent is monitored whenever the Containment is being purged.

The Containment/Plant Vent Monitoring/Sampling System consists of three (3) separate radiation monitors--a particulate monitor, a gaseous monitor, and an iodine monitor. The pertinent information associated with each of these radiation monitors is as follows:

	Monitor Type		
	Particulate	Gas	Iodine
Background (CPM)	1000	839	180/Min
Alarm Setpoint (CPM)	7000	30,000	15,000
Sensitivity (CPM/uci/cc)	4.4×10^{11}	2.1×10^{6}	3.0×10^{9}

The particulate, gas and iodine monitors are designed and qualified for Seismic Class I service. The containment and plant vent radiation monitors and the sampling system are connected to vital power sources.

Although the system is not redundant, assurance of function is provided in that the system logic is designed such that any one of the three (3) monitors will initiate isolation. Loss of power to any monitor will also automatically initiate isolation. Additionally, two source range neutron flux monitors are

A.1) Cont'd.

required to be in service during the refueling operation, and the operator is provided with control room indication of the two area radiation monitors located inside the Containment Building on Elevation 130'. The Technical Specifications require that the Containment Purge and Pressure-Vacuum Relief Isolation System be operational during refueling operations.

Q.2) Describe the response of the containment isolation valves following the FHA. Include valve closure times including expected valve closure time as well as Technical Specification requirements.

A.2) The response time for Containment Building ventilation isolation has been determined during pre-operational testing. The results obtained do not differ significantly from Technical Specification requirements. The protection logic response time is in the order of 0.02 seconds. The response time for the initiation of ventilation isolation after a high radiation alarm is 0.10 seconds. The isolation valves close within two (2) seconds of receipt of an isolation signal.

Q.3) Provide the transit time from the point where a monitor can respond to a release from the FHA to the inboard isolation valve based on the maximum air velocity (peak centerline velocity) at maximum exhaust flow. Also include the transit time based on average velocity and normally expected air flows. Conservatively assume that the FHA is a puff release closest to an exhaust grill.

A.3) It was assumed that the fuel handling accident (FHA) was a "puff" release as close as possible to an exhaust grill, i.e., a Containment Fan Cooler Unit intake. It would take 13 seconds for the "puff" of radioactivity to travel from the Containment Fan Cooler Unit inlet through the ventilation ductwork and to Elevation 195' in the plant vent where the radiation monitoring sampling line inlet is located; this time lapse has been calculated assuming that only the Fan Cooler Unit furthest from the Containment Purge line inlet is operating, however, the results of the analysis are independent of both this assumption and of the mass transit from pool surface to the Fan Cooler Unit duct inlet. It has also been calculated that it would take approximately 10 seconds for the radioactive "puff" to reach the radiation monitor from the sampling line inlet, and approximately 2 seconds (see response to Question 2) for the containment isolation valves to close. Therefore, it has been estimated that 25 (13 + 10 + 2) seconds elapse from the time that the "puff" enters the Containment Fan Coolers until the containment isolation valves are completely closed.

The plant vent continues upward along the side of the Containment Building above Elevation 195' where the radiation monitor inlet tubing is located. It will take approximately 3 seconds for the gas to go from Elevation 195' to the plant vent discharge. Therefore, 16 (13 + 3) seconds elapse from the time where the "puff" enters the containment fan cooler inlet inside of the Containment until it is discharged from the plant vent. Since purge flow from the Containment is 35,000 CFM, the maximum volume of radioactive air released is 5075 $ft.^3$ due to purge flow and 3923 $ft.^3$ due to the amount of air still in the ventilation ductwork up to the plant vent discharge after the isolation valves are closed. Therefore the total amount of radioactive air released to the atmosphere is 5075 + 3932 = 9007 $ft.^3$

A.3) Cont'd.

This analysis is based on average velocity; the maximum peak centerline velocity is equal to the average velocity since turbulent flow is present throughout the ventilation ductwork.

Q.4) Provide drawings of the containment which clearly show the location of the radiation monitors relative to the ventilation exhaust system including all exhaust inlets and duct arrangement up to the outboard isolation valves.

Q.4) The following is a list of drawings that show the portion of the ventilation system which is pertinent to our analysis:

A.4)

Drawing Number	Title
207622-A-8851	Auxiliary Building - Ventilation Ducts - El. 100'
207623-A-8851	Auxiliary Building - Ventilation Ducts - El. 122'
207624-A-8851	Auxiliary Building - Ventilation Equip. & Ducts - El. 122'
207635-A-8826	Reactor Containment - Ventilation - Plan El. 130' & Above
207636-A-8826	Reactor Containment - Ventilation - Plan Below El. 130'
207637-A-8826	Reactor Containment - Ventilation - Sections
223101-A-8989	Auxiliary Building - Plant Vent - Sheet 1
223102-A-8989	Auxiliary Building - Plant Vent - Sheet 2
223103-A-8989	Auxiliary Building - Plant Vent - Sheet 3
223104-A-8989	Auxiliary Building - Plant Vent - Sheet 4
239634-A-1520	North Penetration Area - El. 78', 100' 130' and Roof Above 100' - External Tubing - Radiation Monitoring

Q.5) If the summation of the instrument response time (question 1.e) and valve closure time (question 2) is greater than the gas transit time (question 3), provide an analysis as to the volume and amount of radioactive exhaust air which could be released. Your response should include the following:

a) Duct sizes

b) Maximum (peak) air velocity

c) Average air velocity

d) Containment isolation valve closure characteristics

e) Exhaust system flow rates

f) Methodology used to calculate gas transit times from the pool surface to the inlet to the exhaust system

g) Air velocity profiles over the pool surface. You should consider the effects of pool water temperature on air flow trajectories.

A.) The following is a tabulation of the duct sizes, velocities and flow rates, etc. for that portion of the ventilation system pertinent to the analysis of a FHA inside the Containment Building. Note that only one value for velocity is given since in turbulent flow, the maximum peak velocity equals the average velocity.

A.5) Cont'd.

From - To	Cross Sect. Area	Length	Flow Rate	Velocity	Air Transport Time
FC - Ring Duct Disch.	29.4 ft^2	195.9 ft	55,000 ft^3/min	1870.7 ft/min	6.28 sec
Ring Duct - 36"	7.07 ft^2	26 ft	35,000 ft^3/min	4950 ft/min	0.315 sec
36" - 63"x50"	21.9 ft^2	58 ft	35,000 ft^3/min	1598 ft/min	2.18 sec
63"x50"- 54"x60"	22.5 ft^2	20 ft	35,000 ft^3/min	1555 ft/min	0.771 sec
54"x60"- 84"x63"	36.8 ft^2	18 ft	95,000 ft^3/min	2580 ft/min	0.418 sec
84"x63"- 72"x90	45 ft^2	34 ft	95,000 ft^3/min	2110 ft/min	0.967 sec
72"x90"- 99"x72"	49.5 ft^2	20 ft	114,490 ft^3/min	2320 ft/min	0.517 sec
92"x72"-102"x72"	51 ft^2	63 ft	114,490 ft^3/min	2240 ft/min	1.684 sec
Total time for radioactive particle to reach monitor tubing after entering FCU					13.132 sec =13 se
El.195'-102"x72"	51 ft^2	118 ft	114,490 ft^3/min	2240 ft/min	3.15 sec = 3 se
Total time for radioactive particle to reach plant vent discharge					16.28 sec

We have not considered the gas transit time from the pool surface to the inlet to the exhaust system nor have we considered the air velocity profiles over the pool surfaces. It is our opinion that this analysis is not required since our system design is such that the time for a puff of radioactive air to go from the pool to the exhaust system is common to both the monitoring system and the purge system.

Our calculations, as seen in answer to Question No. 3 above, estimate that a maximum 9007 ft.^{3} of radioactive air is released to the atmosphere if a FHA occurs inside the Containment.

Q.6) Describe any charcoal filters which would mitigate the consequence of the FHA. If so, include the following information: type (e.g., kidney), redundancy, power sources, safety grade, technical specification requirements.

A.6) Charcoal filters are provided in both the Containment Iodine Removal System (subsystem of the Containment Ventilation System) and the Auxiliary Building Exhaust System. The Containment Iodine Removal System is contained within the Containment Building and is a recirculation system. The Auxiliary Building system when used in conjunction with the Containment Purge System is a once-through system. Both systems use pleated bed adsorber cells which consist of 1-inch thick charcoal beds.

The charcoal filters are designed to Seismic Class I criteria. Although the filters do not have any power source, they are supplied by safety related equipment. Also the Technical Specification requires periodic testing to assure operability of the filters.