GPU Nuclear

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Writer's Direct Dial Number:

June 21, 1984

Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Crutchfield:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
NUREG-0737, Item II.B.2
Design Review of Plant Shielding

Your letter of December 7, 1983 requested additional information relative to NUREG-0737, item II.B.2 "Design Review of Plant Shielding." The questions were generated as a result of Region I Inspection Report No. 50-219/83-13 which reviewed our previous submittals dated January 4, 1980 and April 10, 1980.

Responses to questions 1 and 5 are provided as an enclosure to this letter. The remaining questions require that additional shielding studies be performed. A contractor has been retained to perform the necessary studies and completion is expected in September 1984. The results and responses to the remaining questions will be forwarded to you shortly thereafter.

Should you require any further information on this subject, please contact Mr. Michael Laggart, BWR Licensing Manager at (201)299-2341.

Very truly yours,

Peter B. Fiedler
Vice President and Director
Oyster Creek

PBF/dam
Enclosures

cc: Dr. Thomas E. Murley, Administrator
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631 Park Avenue
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NRC Resident Inspector
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ENCLOSURE

Question 1:

Source Terms were identified as being derived from GE data and documented in Computer Run SNUMB-7007S dated November 9, 1979. Who performed this calculation, and what type of code was used (for example, ORIGEN)?

Response:

Radiation Source Term Information for NUREG-0578 Implementation, General Electric Company, November 1979, Computer Code RIB-D was used to derive the source terms.

Question 5:

Explain the reasoning contained in the April 15, 1983 GPU letter to NRC (P.B. Fiedler to D. Eisenhut) entitled "Cycle 10 Refueling Outage Workload", for cancellation of the commitment for a SGTS Filter Tie-in. Specifically, what analyses have been performed to conclude that: (1) a single SGTS filter train is capable of handling (without changeout) effluent loading associated with an excessive MSIV leakage accident, and (2) whether radiological source contribution need be considered for any vital areas (such as the Security Building) from such a source?

Response:

By letter dated April 15, 1983, GPU Nuclear provided justification for cancelling a proposed modification to the Standby Gas Treatment System (SGTS). The justification, as stated in the letter, was based upon the NRC's evaluation of SEP Topic XV-19, "Radiological Consequences of a loss of Coolant Accident." GPU Nuclear was provided with the results of that evaluation by letter from Mr. Dennis Crutchfield to P. B. Fiedler dated September 2, 1983. A copy of this letter is attached for your convenience. The evaluation demonstrates that a single SGTS is capable of handling effluent loading. Please note in the evaluation that Main Steam Isolation Valve through valve leakage is not routed to the SGTS.

In reponse to the second part of Question No. 5, the filters for the SGTS are located below grade in a concrete pipe tunnel. Because of the obvious shielding between this location and other vital areas we did not consider it necessary to consider these components as radiation sources for other vital areas.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 2, 1982

Docket No. 50-219
LS05-82-09-011

Mr. P. B. Fiedler, Vice President and Director
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Fiedler:

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION, SAFETY EVALUATION OF SEP TOPIC XV-19, RADIOLOGICAL CONSEQUENCES OF A LOSS OF COOLANT ACCIDENT

Enclosed is the staff's final evaluation of SEP Topic XV-19 for the Oyster Creek Plant. The evaluation has been revised from the draft evaluation sent to you on June 29, 1982, based on information supplied by you. The revised portions have been marked with a line in the right hand column to designate the changes. The staff now estimates that the 30 day low population zone (LPZ) doses could exceed the allowable specified in 10 CFR 100 by approximately 14% (341 vs. 300 rem) instead of 20% as in the draft evaluation. Since the activity leakage pathway that contributes over 95% (334 rem) of the estimated dose is still from the main steam isolation valve (MSIV) leakage the recommendations outlined in the draft evaluation are still valid. These are as follow:

1. Perform a more realistic analysis for MSIV doses factoring in the effects of drywell pressure vs. MSIV leakage rate as a function of time. The total MSIV leakage then should be lower than assumed by the staff.

2. Evaluate the merits of directing the turbine building ventilation exhaust through a charcoal filter system.

3. Evaluate the merits of installing MSIV leakage prevention systems.

4. Any other procedure or system modifications that will limit the total LOCA doses from all pathways to less than 300 rem.

This evaluation will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. This assessment may be

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revised in the future if your facility design is changed or if NRC criteria relating to this subject is modified before the integrated assessment is completed.

Sincerel

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

Mr. P. B. Fiedler

cc

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XV-19 LOSS-OF-COOLANT ACCIDENTS RESULTING FROM A SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY (RADIOLOGICAL CONSEQUENCES) - OYSTER CREEK

I. INTRODUCTION

Loss-of-coolant accidents (LOCA's) are postulated breaks in the reactor coolant pressure boundary resulting in a loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system. LOCA's result in excessive fuel damage or melt unless coolant is replenished. Excessive fuel damage can result in significant radiological consequences to the environment via leakage from the containment. SEP Topix XV-19 is intended to assure that the radiological consequences of a design basis LOCA from containment leakage, ESF leakage, containment purge and leakage through the main steam isolation valves (MSIV's) are within the exposure guideline values of 10 CFR Part 100.

II. REVIEW CRITERIA

Section 50.34 of 10 CFR Part 50 requires that each applicant for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility. The LOCA is one of the postulated accidents used to evaluate the adequacy of these structures, systems, and components with respect to the public health and safety.

In addition, 10 CFR Part 100.11 provides dose guideline values for reactor siting assessments.

III. RELATED SAFETY TOPICS

Topic II-2.C, "Atmospheric Transport and Diffusion Characteristics for Accident Analysis" provides the meteorological data used to evaluate the offsite doses. Topic III-5.A, "Effects of Pipe Breaks on Structures, Systems and Components Inside Containment" ensures that the ability to achieve safe shutdown or to mitigate the consequences of an accident are maintained. Various other topics examine such areas as containment integrity and isolation, post accident chemistry, ESF systems, combustible gas control and control room habitability.

IV. REVIEW GUIDELINES

The review of the radiological consequences of a LOCA was conducted in accordance with Appendices A, B, and D to Standard Review Plan 15.6.5 and Regulatory Guide 1.3. The plant is adequately designed against a LOCA and the dose mitigating features are acceptable only if the resulting doses at the exclusion area and low population zone boundaries are within the guideline values of 10 CFR Part 100.

V. EVALUATION

In the licensee submittal to NRC, the licensee provided a full spectrum of loss-of-coolant accidents as a result of various primary system pipe break sizes. The submittal, however, did not provide sufficient detail to permit an independent analysis and questions were sent to the licensee on April 7, 1982 by teletype. Based on the licensee's response to the questions dated April 28, 1982 (in a letter from Drew G. Holland of GPU Nuclear to Robert Fell of NRC), the staff performed an analysis of the radiological consequences according to the current NRC criteria.

The radiological consequences of this accident result from the following sources:

1. Containment Leakage: The licensee in his April 28, 1982 letter indicates that there is no containment leakage which bypasses the SGTS filters. (Because any bypass leakage paths can alter the conclusions reached in this evaluation, the licensee should confirm this statement by submitting the details on how each leakage path was considered in arriving at the conclusion that no containment leakage bypasses the area processed by the SGTS.) The calculated dose from conta' nt leakage is derived solely from the 0.5% per day Technical Specification leakage limit from the primary containment, complete mixing in the secondary containment and then processing by the SGTS prior to release o the environment.

   Based on information provided by the licensee on filter efficiencies, the staff has determined that an appropriate value for the filter efficiencies is 90%.

2. Main Steam Isolation Valve Leakage: Oyster Creek does not have a main steam isolation valve leakage control system (MSIV-LCS). In our analysis, we have assumed that the MSIV's leak at a rate of 11.5 scfh. The value of 11.5 scfh was determined from the acceptance criteria of the plant's test program for

these valves. The staff has estimated that a holdup of fission products will occur in the 100 foot section of main steam piping between the outboard isolation valve and the turbine stop valves. Leakage is assumed to occur at ground level.

The resulting 0-30 day LPZ doses based on the 11.5 scfh per MSIV is 334 rem for the thyroid and 0.2 rem whole body. The length of the main steam pipe section between the outboard main steam isolation valve and the turbine stop valves is critical to this conclusion. The estimated length of pipe (100 feet) was supplied by the licensee, and because of its importance to the calculation, should be verified by the licensee.

3. Post-LOCA Leakage from ESF Systems Outside Primary Containment: Because the ECCS leakage will be to the reactor building and the SGTS includes an ESF grade filtration system which filters the reactor building exhaust, we have not calculated the doses from passive failures (according to Appendix B to Standard Review Plan Section 15.4.5). We have calculated the doses resulting from anticipated operational leakage. No Technical Specification limit on the leakage from ESF systems outside containment exists. We have assumed one gpm total leakage in the calculation of the ESF component leakage contribution to the LOCA doses.

4. Containment Purge: The existing purge valves will close in about one minute from an initiating signal. The licensee in his April 28, 1982 letter indicates plans to replace these valves with ones that will close within 5 seconds. The licensee should submit confirmation of these plans and a schedule for their installation. The staff has evaluated the potential contribution to the LOCA dose from operation of the purge system during the

onset of an accident and has determined that the contribution is much less than 0.1 rem and, therefore, is negligible.

VI. CONCLUSION

The calculated doses and assumptions used to arrive at these doses are presented in Table XV-1 and XV-2, respectively. The evaluation indicates that the 0-30 day LPZ thyroid dose guideline is exceeded by approximately 14%. The staff notes that a major portion of this dose is attributed to MSIV leakage. As noted earlier, the licensee needs to provide information to support the statement (in the April 28, 1982 letter) that no containment leakage bypasses the area served by the SGTS.

The staff concludes that because of the uncertainties in the calculation of the doses and because the estimated thyroid doses exceed the 0-30 day 10 CFR 100 thyroid dose LPZ guideline value by only approximately 14%, any plant backfit considerations can be appropriately pursued during the integrated assessment.

TABLE XV-1

RADIOLOGICAL CONSEQUENCES OF A LOCA AT OYSTER CREEK

	Duration		Exclusion Area Boundary		Low Population Zone	
	From Hrs.	To Hrs.	Thyroid Rem	Whole Body Rem	Thyroid Rem	Whole Body Rem
CONTAINMENT LEAKAGE	0.0	2.0	3.8	0.1	1.4	0.1
	2.0	4.0	-	-	3.8	0.1
	4.0	8.0	-	-	0.3	0.1
	8.0	24.0	-	-	0.2	0.1
	24.0	96.0	-	-	1.0	0.2
	96.0	720.0	-	-	0.4	0.2
MSIV* LEAKAGE	37.5	96.0	-	-	170	0.1
	96.0	720.0	-	-	164	0.1
ESF LEAKAGE	0.0	2.0	<0.1	<0.1	-	-
	0.0	720.0	-	-	0.01	<0.01
Total LOCA doses			3.8	0.2	341	1.0

* The leakage from this source is assumed to start 37.5 hours into the accident and, therefore, there is no contribution to the EAB dose.

TABLE XV-2

ASSUMPTIONS USED IN THE ANALYSIS OF THE RADIOLOGICAL CONSEQUENCES OF A LOCA AT OYSTER CREEK

1.	Reactor stretch power (Mwt)	1934
2.	Fission product release fractions (percent)	
	a. Iodines	25
	b. Noble gases	100
3.	Primary containment volume (cubic feet)	180,000
4.	Primary containment leak rate (%/day)	0.5
5.	SGTS filter efficiency (percent) (all forms of iodine)	90
6.	MSIV leak rate (scfh)	11.5
7.	SGTS bypass leakage	0
8.	ESF leakage into reactor building (gpm)	1.0
9.	Purge system flow rate (cfm)	1000
10.	Time required for purge system isolation (sec)	5
11	X/Q's (sec/cubic meters) Ground level release for MSIV leakage	
	0-2 hour EAB* (414 m)	7.6 E-4***
	0-8 hour LPZ** (1208 m)	6.5 E-5
	8-24 hour " "	4.3 E-5
	1-4 day " "	1.7 E-5
	4-30 day " "	4.8 E-6
	Elevated release used for containment leakage (fumigation conditions)	
	0-2 hour EAB	1.1 E-4
	0-4 hour LPZ	4.2 E-5
	(non-fumigation conditions)	
	4-8 hour LPZ	9.1 E-7
	8-24 hour LPZ	2.5 E-7
	1-4 day LPZ	1.7 E-7
	4-30 day LPZ	2.5 E-7

*Exclusion Area Boundary (10 CFR 100)
**Outer boundary of Low Population Zone (10 CFR 100)
***7.6 E-4 = 7.6×10^{-4} = .00076