September 01, 1982

Docket No. 50-29
LS05-82-09-008

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: SEP TOPIC XV-19, LOSS-OF-COOLANT ACCIDENTS RESULTING FROM A SPECTRUM OF PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY - YANKEE

Enclosed is the staff's final evaluation of SEP Topic XV-19 for the Yankee Plant. This evaluation is based on our review of your topic safety assessment report submitted by letter dated January 4, 1982, and an independent evaluation performed by the staff. The staff's conclusion is that the doses calculated for this topic exceed 10 CFR Part 100 guidelines.

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 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.

SE04

DSU USE(11)

Sincerely,

Original signed by:

Ralph Caruso, Project Manager
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

OFFICE	SEPB:DL	SEPB:DL	SEPB:DL	SEPB:DL	ORB#5:PM	ORB#5:BC	AD:SA:DL
SURNAME	TMichaels:dk	MBoyle	CGrimes	WRussell	RCaruso	DCrutchfield	TIppolito
DATE	8/31/82	8/26/82	8/31/82	8/31/82	8/31/82	9/1/82	9/1/82

Mr. James A. Kay

Yankee
Docket No. 50-29
Revised 3/30/82

cc
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Board of Selectmen
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SYSTEMATIC EVALUATION PROGRAM
TOPIC XV-19

YANKEE

LOSS OF COOLANT ACCIDENTS RESULTING FROM A SPECTRUM OF PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

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. A LOCA will 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 Topic XV-19 is intended to assure that the radiological consequences of a design basis LOCA from containment leakage and leakage from engineered safety features outside containment 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 guidelines for reactor siting against which calculated accident dose consequences may be compared.

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, including those shown in Table 1.

Topic III-5.A, "Effects of Pipe Breaks on Structures, Systems, and Components Inside Containment," ensures that the ability to safely shut down or mitigate the consequences of an accident is maintained. Various other related topics cover 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 the Appendices A and B to Standard Review Plan 15.6.5, TID-14844, R.G. 1.4 and current staff practice. The plant is considered adequately designed against a LOCA, and the dose mitigating features are acceptable, if the resulting doses at the Exclusion Area Boundary and the outer boundary of the Low Population Zone are within the guideline values of 10 CFR Part 100.

V. EVALUATION

Staff reviewed the licensee's submittal for evaluation of loss-of-coolant accident (LOCA). The licensee determined that the total radiological consequences of such an accident meet the exposure guidelines of 10 CFR Part 100.11 with respect to the adequacy of the distances to the Exclusion Area Boundary and the Low Population Zone outer boundary. The analysis included the contributions from containment leakage post-LOCA, leakage from ESF systems outside containment with an assumed leak rate of 20 gallons per day, and shine or direct radiation through the containment.

The staff reviewed this analysis and performed an independent analysis of the radiological consequences from the three pathways mentioned above. The assumptions used in the calculations are listed in Table 1; the calculated doses are shown in Table 2. The dose from the containment leakage and ESF leakage outside containment pathways was calculated by the methods of Standard Review Plan (SRP) 15.6.5, Appendices A and B. We assumed that decay was the only removal mechanism for the radioactive material assumed to be released to containment, unlike the licensee, who used a plate-out removal coefficient of 2.5 per hour.

For the iodine postulated to be released from leaking ESF components outside containment, we gave no credit for hold-up, plate-out, or filtration. Since there are no technical specification limits on the leakage of these components, we followed the current staff practice and assumed a leak rate of 1 gpm. The licensee assumed that the leakage would be 20 gallons per day. Although the contribution from this recirculation system leakage represents a substantial contribution to the total doses, our review of the sensitivity of the calculated doses to the leakage parameter indicates that reasonable variations of the assumed 1 gpm leak rate (including the licensee's value of 20 gallons/day) would not result in doses which would change the conclusions reached below.

The shine or direct gamma dose evaluation is not considered in SRP 15.6.5, because the modern plants for which SRP 15.6.5 was written have thick concrete walls which reduce the potential shine dose to a negligible amount. However, Yankee's containment is a steel sphere, which provides far less shielding from shine than a steel-lined reinforced concrete containment. Guidance for the evaluation of this dose pathway is found in 10 CFR 100.11, which states, "The calculations described in Technical Information Document 14844 may be used as a point to departure for consideration of particular site requirements which may result from evaluation of the characteristics of a particular reactor..." We decided that, based on the characteristics of the Yankee plant, we should evaluate the shine doses. We used the calculative method outlined in TID-14844. More recently developed ways of calculating the shine dose may give a more accurate answer, but the shine doses, shown in Table 2, are so small that refining the calculation would make little difference.

<u>Conclusions</u>

The calculated doses, shown in Table 2, resulting from a loss-of-coolant accident exceed the dose guidelines of 10 CFR 100.11. A major contributor to the calculated dose is from the postulated leakage of recirculated core cooling water outside containment. It is reasonable to assume that the leakage could be reduced by appropriate surveillance and maintenance, and limited by Technical Specifications to lower values. Also, the postulated release of airborne iodine from this leakage could be reduced by orders of magnitude by filtering this release pathway. These types of changes could be included in the consideration of the LOCA dose in the Integrated Assessment for this plant.

Table 1

Assumptions Used in the Calculation of Offsite Doses
Following a Design Basis LOCA

Reactor power level	600 MWt
Fraction of noble gases available for release	100%
Fraction of iodines available for release	25%
Containment leak rate	0.2% per day, first 24 hours 0.1% per day, after 24 hours
For shine dose:	
distance to Exclusion Area Boundary (EAB)	3100 feet
distance to nearest Low Population Zone (LPZ) outer boundary	4452 feet
credit for vapor barrier (containment) shielding	none
calculative method	As in TID-14844 (100% of the noble gases, 50% of the iodines, and 1% of the solid fission products).
ESF leakage	
Long term	1 gpm for ½ hour to 30 days after accident.
Short term passive failure	50 gpm for 30 minutes starting 24 hours after accident.
Fraction of core inventory in ECCS water	50% of iodine, no noble gases
Fraction of iodine in leaked water released to environment	10%
Atmospheric Dispersion Coefficients	0-2 hr. EAB 2.8×10^{-4} sec/m^3 0-8 hr. LPZ 2.8×10^{-5} sec/m^3 8-24 hr. LPZ 1.9×10^{-5} sec/m^3 24-96 hr. LPZ 1.6×10^{-5} sec/m^3 96-720 hr. LPZ 1.0×10^{-5} sec/m^3

Table 2

CALCULATED OFFSITE DOSES RESULTING FROM A LOCA

	Exclusion Area Boundary (0-2 hours)		Nearest Outer Boundary of Low Population Zone (0-30 days)	
	Dose in rems		Dose in rems	
	Thyroid	Whole Body	Thyroid	Whole Body
Containment Leakage	162	1.0	244	0.4
Leakage from ESF components	126	0.3	435	0.3
Shine (direct gamma)	-	1.4	-	0.2
Total	288	2.7	679	0.9