October 20, 1982

Docket No. 50-155
LS05-82-10-071

Mr. David J. VandeWalle
Nuclear Licensing Administrator
Consumers Power Company
1945 West Parnall Road
Jackson, Michigan 49201

Dear Mr. VandeWalle:

SUBJECT: SEP TOPIC XV-19, LOSS-OF-COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY (RADIOLOGICAL) - BIG ROCK POINT

Enclosed is the staff's evaluation of SEP Topic XV-19 for the Big Rock Point Plant. This evaluation is based on safety analysis reports submitted by you on July 1, 1981 and June 2, 1982, and on an independent analysis performed by the staff. The evaluation concludes that your facility meets the acceptance criteria for this topic.

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.

Sincerely,

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

Add: R. Schad

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Big Rock Point Plant

XV-19 LOSS OF COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED 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-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 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.

Topic III-5.A, "Effects of Pipe Breaks on Structures, Systems and Components Inside Containment" ensures that the ability to achieve safe shutdown or 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 the Appendices A, B, and C to Standard Review Plan 15.6.5 and Regulatory Guide 1.3, except as noted in the evaluation below. The plant is adequately designed against a LOCA and the dose mitigating features are acceptable if the resulting doses at the exclusion area and low population zone boundaries are within the guideline values of 10 CFR Part 100.

V. LICENSEE SUBMITTAL

The licensee originally submitted an evaluation of this topic by letter dated July 1, 1981. Subsequently, a revised evaluation was submitted by letter dated June 2, 1982. Clarification of certain referenced material in the latter letter was received by telephone conversation on September 3, 1982. Because not all pathways were considered by the licensee and because no evaluation was provided for the Low Population Zone boundary, the staff performed an independent evaluation.

VI. EVALUATION

Key parameters used in this evaluation are listed in Table XV-19-1. Contributions to doses at the Exclusion Area Boundary for 0-2 hours (the Exclusion Area Boundary dose, EAB) and for 0-30 days (the Low Population Zone dose, LPZ) were calculated for (1) containment leakage, (2) Engineered Safety Feature (ESF) leakage, (3) ESF component failure, (4) emission through the purge valve prior to closure and (5) shine through the containment sphere. Each of these contributing pathways is discussed below. All of the calculations assumed a power level of 252 MW Thermal, providing a 5% uncertainty factor over the licensed power level of 240 MW Thermal.

The Exclusion Area Boundary was taken to be 0.5 mile based on a licensee submittal of April 6, 1982. The Low Population Zone was taken to be 2.5 miles, based on the staff's evaluation of SEP topic II-1.B, which was sent to the licensee on June 6, 1980.

1. All of the noble gases and 50% of the iodines were assumed to be mixed in the containment volume at the instant of the LOCA. In addition to decay and leakage to the environment, radioiodines were assumed to be removed by plateout on cool surfaces within the containment and, with limited efficiency, by spray inside containment with water from the nearby lake. The combined removal rate constant for plateout and spray was assumed to be 2.4 hr^{-1} for elemental iodine. The removal was allowed for these processes for 2 hours, during which time the containment content of radioiodine was reduced by about a factor of 200. This reduction, which is comparable to that for modern plants, was allowed because the very large final volume of water in the containment

and low power level of the plant provide for a very low concentration of iodine in the water. Therefore, although pH adjustment to alkaline conditions is not provided, the reaction reevolving iodine is not favored. In addition to this beneficial effect, the very large final volume of water reduces the free volume of the containment and increases the concentration of gaseous radioiodine. This latter effect was accounted for by increasing the leakage in units of percent by volume per day to approximate a constant leakage in cubic feet per day at higher concentration. Removal of particulate iodine by such processes as gravitational settling was allowed throughout the course of the accident.

Since the Big Rock Point containment is a large, dry (at the start of the accident) containment, much like a PWR containment with respect to its pressure response to the transient, the leakage rate was reduced at the end of one day.

2. Leakage from the ESF core spray system, which will contain radioactive materials when recirculation is started, was modeled at twice the technical specification limit, in accordance with the SRP. The plant's technical specification controls leakage in the heat exchanger tubes of the core spray system, where leakage would be into the secondary system and would provide dilution and holdup prior to potential release in the screenhouse. By practice, the other leakage paths are inspected and assured to be negligible. However, the assumption is made that small unidentified additional leakage takes place and that 10% of the iodine in all the leaking water is postulated to be evolved, beginning at the time of the start of recirculation and continuing for 30 days.

Fifty percent of the radioiodines are assumed to be mixed in the water. The volume of the water was taken to be one-third of the containment volume, based on information contained in the Final Hazards Summary Report.

3. The SRP requires the assumption of a failure in an ESF component that allows 50 gallons per minute leakage for 30 minutes, starting at 24 hours after the accident. The thyroid dose contribution was calculated for I-131 and I-133, other isotopes having decayed prior to the failure. The specified 1500 gallon leakage represents a very small fraction of the large water inventory in the containment. Since the iodine is assumed to be slowly evolved from the spilled water, the meteorological conditions used were for the period 1 to 4 days following the accident.

4. Because the plant does not have a commitment to restrict purging during operation, the radioiodine that is contained in the primary water blown into containment and thence to the environment prior to purge valve closure contributes to the calculated thyroid doses. It was assumed that the time for a containment isolation signal to be generated is small with respect to the valve closure time. The amount of primary water released to the atmosphere was roughly estimated from release calculated for two other plants, normalized to Big Rock Point's valve closure time and valve size. The normalized values were about 25% different and the average of the two was used. Two dose contributions were calculated. The first assumed that the primary water was at the plant's technical specification limit of 35 μCi/gram radioiodine and, in the absence of any other information, that it was all I-131 (purge case a). The second assumed that the plant had adopted the BWR Standard Technical

Specification on coolant iodine content, as recommended in evaluations of other SEP topics.

5. The shine from the containment was approximately determined. Based on graphical interpolation (for the 0.5 mile EAB) or extrapolation (for the 2.5 mile LPZ) of calculated data in TID-14844, scaled to the appropriate power levels, both 0-2 hour and 0-30 day doses, respectively, to the whole body were determined. TID-14844 assumes that one percent of the solids are released from the fuel to the containment, in addition to the radioiodines and noble gases. No correction was made for that fraction of the volume of the containment which is below grade.

VII. CONCLUSIONS

The doses at the EAB and LPZ, by pathway, are summarized in Table XV-19-2. The total is given only for purge case a, since that represents the present technical specification on coolant activity for the plant. The whole body and thyroid doses for the EAB and LPZ are within the guideline values of 10 CFR 100 (25 Rem and 300 Rem). Therefore, it is concluded that the design of the dose mitigating features of the plant is acceptable for mitigating the consequences of this accident.

Table XV-19-1 Key Parameters for LOCA Evaluation Big Rock Point Plant

<u>General</u>		
Power Level (MW Thermal)		252
Containment volume (cu ft)		9.4×10^{5}
Dispersion parameters (sec/cu meter)		
0.5 mi EAB	0-2 hours	6.7×10^{-4}
2.5 mi LPZ	0-8 hours	8.0×10^{-5}
	8-24 hours	1.7×10^{-5}
	1-4 days	5.5×10^{-6}
	4-30 days	1.2×10^{-6}
Radioiodine split fractions		
	elemental	.955
	organic	.02
	particulate	.025
Breathing rate (cu meters/sec)		
	0-2 hours	3.47×15^{-4}
	2 hours-30 days	2.32×10^{-4}
<u>Containment leakage pathway</u>		
Iodine removal rate constant (hr^{-1})		
	elemental (0-2 hrs)	2.4
	organic	0
	particulate (0-30 days)	.2
Containment leakage rate (%/day)		
	0-8 hours	.5
	8-24 hours	.73
	1-30 days	.37

Table XV-19-1 (Continued)

ESF leakage pathway	
Sump volume (gal)	2.2×10^6
Leakage (gal/min)	0.4
Decontamination factor	10
Start of recirculation (hr)	5
ESF failure pathway	
Fraction of sump volume released	6.8×10^{-4}
Decontamination factor	10
Purge pathway	
Valve closure time (sec)	6
Valve size (inches)	24
Coolant released (grams)	1.6×10^6
I-131 content (μCi/gm)	
case a	35
case b	0.2

Table XV-19-2 Doses for LOCA by Pathway

	EAB		LPZ	
	Thyroid	Whole Body	Thyroid	Whole Body
Containment leakage	198	2.2	36	4.7
ESF leakage	0	0	2	0
ESF failure	0	0	1	0
Purge case a	19	0	2	0
case b	<1	0	<1	0
Shine	0	1.4	0	$<.1$
Total (purge case a)	217	3.6	41	4.7