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March 9, 1982

In reply, please
refer to LAC-8145

DOCKET NO. 50-409

U. S. Nuclear Regulatory Commission
ATTN: Mr. Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
Division of Operating Reactors
Washington, D. C. 20555

SUBJECT: DAIRYLAND POWER COOPERATIVE
LA CROSSE BOILING WATER REACTOR (LACBWR)
PROVISIONAL OPERATING LICENSE NO. DPR-45
SEP TOPIC XV-16 - RADIOLOGICAL CONSEQUENCES OF
FAILURE OF SMALL LINES CARRYING PRIMARY COOLANT
OUTSIDE CONTAINMENT

References: (1) DPC Letter, LAC-7387, Linder to Eisenhut, dated February 27, 1981.
(2) NRC Letter, Ziemann to Linder, Technical Specifications Amendment No. 18, dated February 4, 1980.

Gentlemen:

Enclosed find the Safety Evaluation Report (SER) for Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment (SEP XV-16) which has been prepared by our Radiological Protection Engineer for the La Crosse Boiling Water Reactor.

Our letter, Reference 1, identified topics for DPC to submit to the NRC for evaluation. The subject topics were listed in the schedule submitted with Reference 1.

If there are any questions regarding this report, please contact us.

Very truly yours,

DAIRYLAND POWER COOPERATIVE

Frank Linder

Frank Linder, General Manager

FL:PWS:af

cc: J. G. Keppler, Director, NRC-DRO III
NRC Resident Inspectors

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LA CROSSE BOILING WATER REACTOR

SYSTEMATIC EVALUATION PROGRAM
SAFETY EVALUATION REPORT
TOPIC XV 16

RADIOLOGICAL CONSEQUENCES OF FAILURE OF SMALL LINES CARRYING PRIMARY COOLANT OUTSIDE CONTAINMENT

The La Crosse Boiling Water Reactor (LABWR) piping has been reviewed to determine if any lines carrying primary coolant outside containment, if ruptured, could release significant quantities of radioactive materials.[1] Four lines carying primary coolant outside of containment have been identified: main steam line, feedwater line, decay heat removal start-up water, and auxiliary core spray HPSW.

The radiological effects of a main steam line break is considered in Topic XV-16.

The feedwater line and the auxiliary core spray HPSW lines meet the requirements of General Design Criterion 55[2] because of the self-actuated flow check valves on the inside of containment and the isolation valves outside containment.

The decay heat removal start-up water line has isolation valves inside and outside the containment. This line also meets the requirements of General Design Criterion 55.

Thus, there are at present no lines other than the main steam line carrying coolant outside the containment which if ruptured could release significant quantities of radioactive materials.

In the near future, as a result of requirements for a post-TMI Primary Coolant Sampling System, a small 3/8-inch I.D. water sample line will be run between the primary sample sink inside containment to a sample system outside containment near the electrical penetration panel to a quick-disconnect sample cylinder inside the feedwater heater area. This 3/8-inch I.D. sample line will have a dual isolation valve system outside containment. This sample line will be isolated during normal operations and will be opened for a sample only after an accident has been thoroughly assessed [e.g., reactor instrumentation, containment and stack airborne radioactivity monitors (SPING System) and containment grab air sample have been analyzed]. This samll sample line will have a pressure rating of 3000 psig maximum and a rating of 1400 psig normal. Coolant flow rates inside this small sample line will not exceed 1.05 gallon per minute, with a normal flow rate of 0.75 gallon per minute.

If this line should break upstream of the isolation valves assume that the maximum flow of 1.05 gpm occurs. Further assume that the reactor coolant activation is operating at the maximum technical specification limit of 0.2 μCi/gm dose equivalent I-131. Also, assume that an iodine spike occurs due to reactor shutdown or depressurization. Examination of the coolant activation measurements made continuously over the last 2-1/2 years shows only one I-131 spike greater than a factor of four over the nearly steady state level. It was an increase of nearly 20. Combining these three values the break could be releasing 0.016 Ci/min of dose equivalent I-131.

Taking no account for hold-up or plate-out, the total release rate of 0.016 Ci/min will be considered as a ground-level release. Using the reactor building dimension of 60-foot diameter and 118-foot height and the Regulatory Guide 1.3[3] approach with moderately stable atmospheric conditions

X/Q is calculated to be 1.7×10^{-3} sec/m^3. This will make the dose equivalent iodine 131 concentration at the fence equal to 4.0×10^{-7} Ci/m^3.

Assuming a breathing rate of 3.47×10^{-4} m^3/sec (Reg. Guide 1.5)[4] and an inhalation dose conversion factor of 1.49×10^{-3} mrem/pci (Reg. Guide 1.109)[5] the thyroid dose is calculated to be 1.7 rem over a two-hour period. This is considerably less than the desired limit of 10 per cent of the 10 CFR Part 100 exposure guidelines and, therefore, complying with the SRP criterion. The plant is adequately designed against failures of small lines carrying primary coolant outside the containment.

REFERENCES:

1. NRC Standard Review Plan Section 15.6.2 "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment."

2. 10 CFR 50 Appendix A Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment.

3. NRC Regulatory Guide 1.3 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors."

4. NRC Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors."

5. NRC Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluent for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I.