August 24, 1982

Docket No. 50-409
LS05-82-08-049

Mr. Frank Linder
General Manager
Dairyland Power Cooperative
2615 East Avenue South
LaCroese, Wisconsin 54601

Dear Mr. Linder:

SUBJECT: SEP TOPIC XV-5, LOSS OF NORMAL FEEDWATER FLOW
LACROSSE BOILING WATER REACTOR (LACBWR)

By letter dated March 5, 1982 (LAC-8138), you submitted a safety assessment report for the above topic. The staff has reviewed this assessment and our conclusions are presented in the enclosed safety evaluation report, which completes this topic for the LaCrosse Boiling Water Reactor.

This evaluation will be a basic input to the Integrated Assessment for your facility. The evaluation may be revised in the future if your facility design is changed or if NRC criteria relating to this topic is modified before the Integrated Assessment is completed.

Sincerely,

Original signed by:

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

SEO4
DSU USE (38)
ADD:
G. Staley

Enclosure:
As stated

cc w/enclosure:
See next page

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DATE	8/19/82	8/19/82	8/20/82	8/20/82	8/23/82	8/ /82	8/24/82

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

Mr. Frank Linder

cc
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SYSTEMATIC EVALUATION PROGRAM

TOPIC XV-5

LACROSSE BOILING WATER REACTOR

Topic: XV-5, Loss of Feedwater Flow

I. INTRODUCTION

Loss of Feedwater flow could occur as a result of the simultaneous tripping of all feedwater pumps or a feedwater controller failure that closes the feedwater control valve. The resulting decrease in reactor water level closes the main steam isolation valves, starts the shutdown condenser and scrams the reactor at the low water level setpoint (-12 inches). Automatic initiation of the high pressure core spray system will also occur at this low water level setpoint in order to maintain the water level above the top of the active fuel.

II. REVIEW CRITERIA

Secton 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, including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility.

Section 50.36 of 10 CFR Part 50 requires the Technical Specifications to include safety limits which protect the integrity of the physical barriers which guard against the uncontrolled release of radioactivity.

The General Design Criteria (Appendix A to 10 CFR Part 50) establish minimum requirements for the principal design criteria for water-cooled reactors.

GDC 10 "Reactor Design" requires that the core and associated coolant, control and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of anticipated operational occurrences.

GDC 15 "Reactor Coolant System Design" requires that the reactor coolant and associated protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operation, including the effects of anticipated operational occurrences.

GDC 26 "Reactivity Control System Redundancy and Capability" requires that the reactivity control systems be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded.

III. RELATED SAFETY TOPICS

Various other SEP topics evaluate such items as the reactor protection system. The effects of single failures on safe shutdown capability are considered under Topic VII-3.

IV. REVIEW GUIDELINES

The review is conducted in accordance with SRP 15.2.6. The evaluation includes review of the analysis for the event and identification of the features in the plant that mitigate the consequences of the event as well as the ability of these systems to function as required. The extent to which operator action is required is also evaluated.

V. EVALUATION

By letter dated March 5, 1982, the licensee provided the results of the analysis for the loss of feedwater flow event. The worst transient was verified using the modified COBRA IIIC Code (Ref. 1). The loss of feedwater flow accident at LaCrosse would result in a reactor vessel coolant inventory decrease. The licensee stated that with respect to fuel performance, this transient would be bounded by the accidental opening of the turbine bypass valve event (Ref. 2). The MCPR for this transient is always greater than 1.32 and the maximum reactor coolant pressure will not exceed the 110% design pressure.

VI. CONCLUSION

As part of the SEP review for LaCrosse plant, we have reviewed the licensee's analysis of loss of normal feedwater flow event according to the criteria of SRP Section 15.2.6. The calculated MCPR is always greater than 1.32 and the predicted peak pressure is well below the 110% design pressure. We therefore, conclude that the results are in conformance with SRP Section 15.2.6 and are acceptable.

VII. REFERENCE

1. Response to Question 4 - Transient Analysis for LACBWR Reload Fuel, Nuclear Energy Service Inc., February 18, 1977.

2. Letter from F. Linder to D. G. Eisenhut, dated March 5, 1982.