August 6, 1982

Docket No. 50-409
LS05-82-08-006

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

Dear Mr. Linder:

SUBJECT: SEP TOPIC XV-19, LOSS OF COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY - LACROSSE BOILING WATER REACTOR

By letter (LAC-8119) dated February 26, 1982 (received May 28, 1982), you submitted a safety assessment on this topic. The staff has reviewed your assessment and our conclusions are presented in the enclosed final topic evaluation. With respect to the systems aspects of this topic, the staff concludes that the LaCrosse Boiling Water Reactor meets current licensing criteria. Potential radiological consequences will be addressed in a separate evaluation.

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 are modified before the integrated assessment is completed.

Sincerely,

SE04
DSU USE (38)

Original signed by:

Dennis M. Crutchfield, Chief
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ADD:
G. Staley
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Enclosure:
As stated

cc w/enclosure:
See next page

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Mr. Frank Linder

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IV. REVIEW GUIDELINES

The review of ECCS performance during a LOCA is conducted in accordance with Standard Review Plan Sections 15.6.5 and 6.3. A plant using stainless steel clad fuel is considered to be adequately designed against a LOCA if the Interim Acceptance Criteria are met. The radiological consequences are addressed in a separate evaluation.

V. EVALUATION

Assuming a conservative combination of circumstances which could lead to core uncovery and excessive heatup following a loss-of-coolant accident, fuel cladding integrity is maintained by successful operation of the Emergency Core Coolant System (ECCS). The following systems in the LaCrosse Boiling Water Reactor provides the necessary protection to mitigate the consequences of a loss-of-coolant accident:

(1) The High Pressure Core Spray (HPCS) system which is put into operation manually or automatically on reactor low water level or high containment building pressure.

(2) The Alternate Core Spray (ACS) system which is also put into operation manually or automatically on coincident low reactor water level and high containment building pressure.

The combined operation of the HPCS and ACS provides long-term cooling of the core. A manual depressurization system is provided to equalize reactor vessel and containment pressure following a LOCA. The LOCA analysis for LACBWR has been performed with no credit taken for blowdown to the shutdown condenser. Manual operation is permissible since at least 20 minutes is available to the operator to make a decision.

The adequacy of the LACBWR ECCS evaluation model was discussed by the licensee in Reference 2. This evaluation was made with respect to the requirements for analysis of blowdown phenomena, as prescribed in the Interim Acceptance Criteria. Small break results in the LOCA analysis have been extrapolated from large and

TOPIC XV-19 (SYSTEMS)
LOSS OF COOLANT ACCIDENTS
RESULTING FROM SPECTRUM OF POSTULATED
PIPING BREAKS
WITHIN THE REACTOR COOLANT
PRESSURE BOUNDARY

LACROSSE BOILING WATER REACTOR

I. INTRODUCTION

The objective of this review is to assure that the consequences of a Loss of Coolant Accident (LOCA) are acceptable, i.e., that the requirements of the AEC Interim Policy Statement and Appendix K to 10 CFR 50 are met. Loss-of-coolant accidents are postulated accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant make-up system, from piping breaks in the reactor coolant pressure boundary. The review consists of evaluating the licensee's analysis of the spectrum of loss-of-coolant accidents including break location, break size, initial conditions assumed, the evaluation model used, failure modes and the acceptability of auxiliary systems used.

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 systems provided for the prevention of accidents and the mitigation of the consequences of accidents.

The AEC Interim Policy Statement requires that all light water reactors shall be provided with an emergency core cooling system designed so that its performance following a LOCA satisfies the criteria set forth in the Interim Acceptance Criteria. Performance is calculated with an evaluation model satisfying the applicable requirements of Appendix K to 10 CFR 50.

The General Design Criteria (Appendix A to 10 CFR Part 50) set forth the criteria for the design of water-cooled reactors. GDC 35 "Emergency Core Cooling" requires that a system be provided to provide abundant emergency core cooling whose function is to transfer heat from the core following a loss of coolant such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (2) clad metal water reaction is limited to negligible amounts. The system should have suitable redundancy and interconnections such that function can be maintained assuming a single failure and assuming availability of only onsite or only offsite power supplies.

III. RELATED SAFETY TOPICS

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.

The adequacy of the features provided for Switchover from Injection to Recirculation modes is addressed in Topic VI-7.B.

Other SEP topics consider the emergency power supplies (VIII-2), effects of flooding of safety-related equipment (VI-7.D), prevention of boron precipitation (IX-4) as well as failure modes of the ECCS (VI-7.C). In addition, such areas as containment integrity and isolation, postaccident chemistry and Engineered Safety Feature systems are considered as part of SEP topics. Topics VI-2.D and VI-3 address the capability of the containment heat removal systems to alleviate the pressure/temperature transient so that the containment is not overpressurized.

intermediate size breaks as discussed in Reference 1. The LOCA results have shown that the HPCS is adequate, even with a single failure, to maintain core parameters within Interim Acceptance Criteria limits. The limiting single failure is the failure of one HPCS pump. The break spectrum analysis performed with the LACBWR ECCS evaluation model identified the most limiting break as an intermediate size break (0.072 sq. ft.) in the 20-inch recirculation line. The highest peak clad temperature (2296°F) is calculated for this break, with no more than 0.15% of the cladding reacting chemically. These values are within the IAC limits of 2300°F peak clad temperature and 1 percent cladding steam reaction. The ECCS performance has been found acceptable by the staff (Reference 4) based on information provided by the licensee in Reference 3.

VI. CONCLUSIONS

As part of the SEP review of LaCrosse Boiling Water Reactor, the loss-of-coolant analysis was reviewed against the Interim Acceptance Criteria, and the acceptance criteria of SRP Sections 15.6.5 and 6.3. The initial conditions relative to single failure, break size and location, power level and operating conditions have been reviewed and found to conform to the requirements of the SRP. The analysis was performed with an approved evaluation model and the results were found to be acceptable.

VII. REFERENCES

1. DPC Letter LAC-6705, Linder to D. Ziemann, SUBJECT: Information on Small Break Analysis, dated December 20, 1979.

2. NES-81A0244, "Comparison of LACBWR ECCS Results to AEC Final Acceptance Criteria," December 9, 1974.

3. Gulf United Report, "Technical Evaluation of the Adequacy of the LACBWR Emergency Core Cooling System," SS-942, May 31, 1972.

4. Safety Evaluation Supporting Amendment No. 11 to Provisional Operating License DPR-45, LaCrosse Boiling Water Reactor, March 3, 1978.