September 29, 1982

Docket No. 50-409
LS05-82-09-082

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

Dear Mr. Linder:

SUBJECT: SEP TOPIC XV-1, DECREASE IN FEEDWATER TEMPERATURE, INCREASE IN FEEDWATER FLOW, INCREASE IN STEAM FLOW - LACROSSE BOILING WATER REACTOR (LACBWR)

By letter dated March 5, 1982, (LAC-8138), you submitted a topic assessment on the above topic. Your letter of August 26, 1982 (LAC-8534) provided additional information. The staff has reviewed your assessment and our conclusions are presented in the enclosed safety evaluation report which completes this topic for LACBWR.

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,

SE04

DSu USE(38)

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

ADD:
T. Michaels

Enclosure:
As stated

cc w/enclosure:
See next page

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LaCrosse Boiling Water Reactor

SEP Topic XV-1

Decrease in Feedwater Temperature

## I. Introduction

Loss of feedwater heating can result from a loss of steam flow to either low pressure feedwater heaters, or to the high pressure heater. Consequently, the reactor vessel receives cooler feedwater with an associated increase in core inlet subcooling and a decrease in coolant void fraction. The negative void reactivity coefficient would result in a gradual initial increase in reactor power. The reactor power would not reach the reactor scram set point and would eventually reach a steady state value slightly above 100% full power.

## 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, 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.

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 15.1.1, 15.1.2, 15.1.2, and 15.1.4.

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

This transient was analyzed in an earlier submittal (Reference 1) and the results indicate that the reactor power peaks at approximately 118% and then gradually decreases to a final steady state of 105% full power. The primary coolant pressure never exceeds 110% design pressure and remains constant throughout the transient. The MCPR during this transient remains above 1.32 at all times. Although the initial power level assumed for this analysis was 100% instead of the recommended 102% full power, the licensee has indicated that this transient is bounded by the increase of feedwater flow event (Reference 2).

VI. Conclusion

As part of the SEP review of LaCrosse the Reactor Systems Branch has evaluated the licensee's analysis of the loss of feedwater heating event. This transient is bounded by the increase of feedwater flow event and is acceptable.

VII. References

1. Anticipated Transients without Scram at the LaCrosse Boiling Water Reactor, Gulf Nuclear Fuels Company, February 28, 1974.

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

LaCrosse Boiling Water Reactor

SEP Topic XV-1

Increase in Feedwater Flow

## I. Introduction

A malfunction of the feedwater control system could cause the feedwater regulating valve to open to its maximum position and would permit excessive feedwater flow to the reactor. There would be a gradual rise in the vessel level and an increase in power because of the increased core inlet subcooling and the negative void coefficient of reactivity. The reactor would trip on the high power trip or high vessel level trip. Reactor water level would then drop due to void collapse.

## 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, 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.

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 Systems 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 Sections 15.1.1, 15.1.2, 15.1.3, and 15.1.4. 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 increase of feedwater flow event. The worst transient was verified using a modified COBRA IIIC code (Ref. 1). The initial power was assumed to be 102% and the feedwater flow rate was assumed to be the maximum available from both feedwater pumps. The licensee indicated that no credit had been taken for the turbine bypass system. The results indicate that the MCPR during this transient remains above 1.32 at all times (Ref. 2) and the reactor coolant pressure remains below 110% of the design pressure. The licensee indicates that operation of the turbine governing system initial pressure regulator (IPR) during the transient has been assumed. The assumption would result in a more limiting condition with respect to the MCPR because the IPR would close down to maintain a constant reactor pressure which in turn would maintain the reactor power at the pretransient level. This was confirmed by a recent communication with the licensee. The licensee in Ref. 1 has assumed the first scram to fail, thus satisfying the single failure criterion.

There are no automatic features that terminate the excess feedwater addition. Assuming no increase in steam flow and no operator action the water level in the vessel would reach the steam lines in about 2 minutes. The licensee addressed the consequences of continued feedwater addition in References 3 and 4. The steam line has been hydrostatically tested out to the turbine building isolation valve. Therefore, water in the steam line due to the feedwater increase would not lead to steam line rupture. With the plant configuration, water would tend to follow the path to the main condenser rather than rising toward the safety valves. The shutdown condenser is capable of controlling any pressure increase associated with this transient so the safety valves will not lift. Power-operated relief valves are not used at LaCrosse. Therefore, the consequences of delayed operator action to terminate the feedwater increase are considered to be acceptable.

## VI. Conclusion

As part of the SEP review of LaCrosse, we have evaluated the licensee's analysis of the increase of feedwater flow event. The MCPR during the transient remains above 1.32 at all times and the reactor coolant pressure does not exceed the 110% design pressure. Therefore, we conclude that the results are in conformance with SRP Section 15.1.1, 15.1.2, 15.1.3, and 15.1.4.

## VII. References


1. Letter from J. P. Madgett (DPC) to R. W. Reid (NRC), dated February 25, 1977.

2. Letter from F. Linder (DPC) to D. G. Eisenhut (NRC) (LAC-8138), dated March 5, 1982.

3. Letter from F. Linder (DPC) to D. Eisenhut (NRC) (LAC-8534), dated August 26, 1982.

4. Letter from F. Linder (DPC) to D. Crutchfield (NRC) (LAC-7633), dated June 29, 1981.

LaCrosse Boiling Water Reactor

SEP Topic XV-1

Increase in Steam Flow

## I. Introduction

An increase in steam flow can result from either inadvertent opening of the turbine bypass valve or a failure of the initial pressure regulator causing the turbine inlet valve to open. At the beginning of the transient, reactor power decreases rapidly due to the increase in coolant void content and the void coefficient of reactivity. At approximately 9 seconds into the transient, the subcooling of the water entering the reactor increases causing a power increase. Without taking credit for a reactor scram at high power, the reactor power peaks at 160% full power, then decreases rapidly and finally stabilized at 104% power (Reference 1).

## 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 the 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.

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" 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 15.1.1, 15.1.2, 15.1.3 and 15.1.4.

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 increase in steam flow event. The results indicated that failure of the turbine admission valve or the turbine bypass valve would result in an initial increase of steam flow and a decrease in core power. Initial sharp increase in steam flow will lead to a lower enthalpy of recirculation flow. Upon reaching the core inlet the cooler water will cause the previously decaying core power to increase. The transient power peaks at 160% of full power. The resulting increase in core void fraction limits the power transient and power will stabilize at 104% of full power. The CPR for this transient stays above 1.32 at all times. The event is not limiting with respect to peak system pressure and minimum critical power ratio.

## VI. Conclusions

As part of the SEP review for LaCrosse Plant, we have evaluated the licensee's analysis of the increase of steam flow event. The results indicate that the MCPR stays above 1.32 at all times (Reference 2) and the maximum reactor coolant pressure never exceeds the 110% design pressure (Reference 3). We, therefore, conclude that the results are in conformance with SRP section 15.1.3 and are acceptable.

## VII. References


1. Letter from F. Linder to D. G. Eisenhut, dated March 5, 1982.
2. Response to Question 4 - Transient Analysis of LAGBWR Reload Fuel, Nuclear Energy Service, Inc., February 18, 1977.
3. Letter from J. P. Madgett to R. W. Reed, dated April 27, 1977.