August 06, 1982

Docket No. 50-409
LS05-82-08-008

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

Dear Mr. Linder:

SUBJECT: SEP TOPIC XV-7, LOSS OF FORCED COOLANT FLOW, REACTOR COOLANT PUMP ROTOR SEIZURE, REACTOR COOLANT PUMP SHAFT BREAK - LACROSSE BOILING WATER REACTOR (LACBWR)

By letter dated July 7, 1981 (received March 11, 1982), you submitted a topic assessment on the above topic. 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 are modified before the integrated assessment is completed.

Sincerely,

SE04
DSU USE (38)

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

ADD:
T. Michaels

Enclosure:
As stated

cc w/enclosure:
See next page

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8/ /82

OFFICE	SEPB:DL	SEPB:DL	SEPB:DL	ORB#5:PM	ORB#5:BC	AD:SA:DL	
SURNAME	EMcKenna:dk	CGrimes	WRussell	RDudley	DCrutchfield	GLainas	
DATE	8/ /82	8/3/82	8/3/82	8/5/82	8/5/82	8/4/82	

Mr. Frank Linder

cc

Fritz Schubert, Esquire
Staff Attorney
Dairyland Power Cooperative
2615 East Avenue South
La Crosse, Wisconsin 54601

O. S. Heistand, Jr., Esquire
Morgan, Lewis & Bockius
1800 M Street, N. W.
Washington, D. C. 20036

Mr. R. E. Shimshak
La Crosse Boiling Water Reactor
Dairyland Power Cooperative
P. O. Box 275
Genoa, Wisconsin 54632

Mr. George R. Nygaard
Coulee Region Energy Coalition
2307 East Avenue
La Crosse, Wisconsin 54601

Dr. Lawrence R. Quarles
Kendal at Longwood, Apt. 51
Kenneth Square, Pennsylvania 19348

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
Rural Route #1, Box 276
Genoa, Wisconsin 54632

Town Chairman
Town of Genoa
Route 1
Genoa, Wisconsin 54632

Chairman, Public Service Commission
of Wisconsin
Hill Farms State Office Building
Madison, Wisconsin 53702

Alan S. Rosenthal, Esq., Chairman
Atomic Safety and Licensing Appeal Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

U. S. Environmental Protection
Agency
Federal Activities Branch
Region V Offioe
ATTN: Regional Radiation Representative
230 South Dearborn Street
Chicago, Illinois 60604

Mr. John H. Buck
Atomic Safety and Licensing Appeal Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. Ralph S. Decker
Route 4, Box 190D
Cambridge, Maryland 21613

Charles Bechhoefer, Esq., Chairman
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. George C. Anderson
Department of Oceanography
University of Washington
Seattle, Washington 98195

James G. Keppler, Regional Administrator
Nuclear Regulatory Commission, Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Thomas S. Moore
Atomic Safety and Licensing Appeal Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

LaCrosse Nuclear Plant

SEP TOPIC XV-7(a)

Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunction

I. INTRODUCTION

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. A resulting increase in fuel temperature and accompanying fuel damage could then result if specified acceptable fuel damage limits are exceeded during the transient. A number of transients that are expected to occur with moderate frequency and that result in a decrease in forced reactor coolant flow rate are addressed in SRP 15.3.1 and SRP 15.3.2. For boiling water reactors (BWRs), partial and complete recirculation pump trips and malfunctions of the recirculation flow controller that cause decreasing flow are reviewed.

II. REVIEW CRITERIA

Section 50.34 of 10 CFR Part 50 requires that each applicant for an 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 loss of forced reactor coolant flow is one of the postulated transients used to evaluate the adequacy of these structures, systems and components with respect to the public health and safety.

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. The staff acceptance criteria are based on meeting the relevant requirements of the following regulations:

A. General Design Criterion 10, as it relates to the reactor coolant system being designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences.

B. General Design Criterion 15, as it relates to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to assure that the pressure boundary will not be breached during normal operations including anticipated operational occurrences.

C. General Design Criterion 26 as it relates to the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences. This is accomplished by assuring that appropriate margin for malfunctions, such as stuck rods, are accounted for.

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 15.3.1 and 15.3.2. The evaluation includes reviews 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. Deviations from the criteria specified in the Standard Review Plan are identified.

The specified criteria necessary to meet the relevant requirements of GDC 10, 15 and 26 for incidents of moderate frequency are:

a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values.

b. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs and the CPR remains above the MCPR safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

d. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, the number of fuel failures must be assumed for all rods for which the DNBR or CPR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding.

V. EVALUATION

The licensee, by letters dated July 7, 1981, and February 25, 1977, has provided the results of an analysis for the subject topic. The analysis indicates that a loss of reactor coolant flow can result from loss of power to the pump, failure of flow control, or pump failure. The decreasing core flow causes a core heat-up due to the flow-power mismatch. The increased void formation inserts negative reactivity to drop power back to a level compatible with the lower flow. The reactor would scram if recirculation flow decreases to 30% of full flow. The licensee's analysis shows that throughout the event, reactor coolant pressure decreases as the void formation adds negative reactivity and the critical power ratio (CPR) remains above 1.32 (Ref. 1, 2).

VI. CONCLUSION

As part of the SEP review of LaCrosse Plant, we have evaluated the licensee's analysis of the loss or decrease in forced reactor coolant flow and have concluded that this event is a mild transient and would not cause unacceptable consequences.

VII. REFERENCE

1. Letter from F. Linder to D. G. Eisenhut, dated July 7, 1981.

2. Letter for J. P. Madgett to R. W. Reid, dated Februarv 25, 1977.

LaCrosse Nuclear Plant

SEP TOPIC XV-7(b)

Reactor Coolant Pump Rotor Seizure

and Reactor Coolant Pump Shaft Break

I. INTRODUCTION

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a reactor coolant pump in a pressurized water reactor (PWR) or recirculation pump in a boiling water reactor (BWR). Flow through the affected loop is rapidly reduced. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate later in time. This topic is intended to cover both of these accidents.

II. REVIEW CRITERIA

Section 50.34 of 10 CFR Part 50 requires that each applicant for an 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 reactor coolant pump rotor seizure and reactor coolant pump shaft break are two of the postulated accidents used to evaluate the adequacy of these structures, systems, and components with respect to the public health and safety.

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 27 "Combined Reactivity Control System Capability", requires that the reactivity control systems, in conjunction with poison addition by the emergency core cooling system, has the capability to reliably control reactivity changes to assure that under postulated accident conditions, and with appropriate margin for stuck rods, the capability to cool the core is maintained.

GDC 28 "Reactivity Limits" requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core.

GDC 31 "Fracture Prevention of Reactor Coolant Pressure Boundary" requires that the boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fractures is minimized.

10 CFR Part 100.11 provides dose guidelines for reactor siting against which calculated accident dose consequences may be compared.

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 15.3.3 and 15.3.4. 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. Deviations from the criteria specified in the Standard Review Plan are identified.

V. EVALUATION

The licensee, by letters dated July 7, 1981, and February 25, 1977, has provided the results of an analysis for the subject topic. The licensee did not address in detail the pump shaft break accident in his analysis. However, the results of analyses for other BWR plants indicate that the single reactor coolant recirculation pump rotor seizure is more limiting than the pump shaft break accident. The result occurs because the pump seizure produces a greater initial power to flow mismatch and more of a decrease in the minimum critical power ratio (MCPR). The single reactor coolant recirculation pump shaft break has a less severe effect with respect to MCPR. The licensee's analysis of loss of flow events shows that throughout the event, reactor coolant pressure decreases as the void formation adds negative reactivity. The reactor coolant pressure would not exceed the 100% design pressure, and the CPR remains above 1.32 (Ref. 1, 2).

The reactor coolant recirculation pump seizure analysis indicates that the reactor power decreases rapidly due to the increase in voiding as the recirculation flow decreases. The reactor would scram if the recirculation flow decreases to 30% of full flow. Although the effects of a single failure coincident with the transient was not addressed in the analysis, a single failure is not identified by us that would cause unacceptable consequences (e.g., violation of the SEP acceptance criteria).

VI. CONCLUSION

As part of our SEP review of the LaCrosse Plant, we have evaluated the licensee's analysis of the recirculation pump seizure and pump shaft break and have concluded that this event is a mild transient and reactor trip is not expected to occur. Therefore, the submitted evaluation is acceptable to the staff.

VII. REFERENCES

1. Letter from F. Linder to D. G. Eisenhut, dated July 7, 1981.

2. Letter from J. P. Madgett to R. W. Reid, dated February 25, 1977.