December 4, 1981

Docket No. 50-219
LS05-81-12-012

Mr. I. R. Finfrock, Jr.
Vice President - Jersey Central
Power & Light Company
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Finfrock:

SUBJECT: OYSTER CREEK - SEP TOPICS XV-7 AND XV-15

By letters dated August 25, 1981 and May 7, 1981 respectively, you submitted safety assessment reports for the above topics. The staff has reviewed these assessments and our conclusions are presented in the enclosed safety evaluation reports, which complete these topics for Oyster Creek.

These evaluations will be a basic input to the integrated assessment for your facility. The evaluations may be revised in the future if your facility design is changed or if NRC criteria relating to these topics are modified before the integrated assessment is completed.

Sincerely,

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

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DSU USE (51)
ADD:
B. Fell

Enclosures:
As stated

cc w/enclosures:
See next page

OFFICE	SEPB:DL	SEPB:DL	SEPB:DL	ORB#5:PM	ORB#5:BC	AD:SA:DL	
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Mr. I. R. Finfrock

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SYSTEMATIC EVALUATION PROGRAM
TOPIC XV-7

OYSTER CREEK

TOPIC XV-7: LOSS OF FORCED RECIRCULATION FLOW

I. INTRODUCTION

A decrease in reactor coolant flow occuring 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 to 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.

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 (Ref. 1), 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 (Ref. 2), 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 breeched during normal operations including anticipated operational occurrences.

C. General Design Criterion 26 (Ref. 3) 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.

The specific 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.

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.

V. EVALUATION

The licensee, in letters dated May 7, 1981 and August 25, 1981, provides 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 drive motor connections, M-G set eakers 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. No reactor trips occur due to the decreased flow. During power operation, there are five reactor coolant recirculation pumps in operation. The plant technical specifications also permit operation with only four reactor coolant recirculation pumps at full power. The results of the licensee's analysis show that the five pump trip is the limiting transient. During this event the reactor coolant pressure is decreasing and the minimum critical heat flux ratio (MCHFR) is increasing.

- The licensee has not provided the results of an analysis for this event in combination with a single failure. Since this event however, does not cause a reactor trip or any engineered safety feature initiation during this transient, we could not identify any single failure which will lead to unacceptable results.

VI. CONCLUSIONS

The staff concludes that the Oyster Creek plant design with regard to transients that are expected to occur during plant life and result in a loss or decrease in forced reactor coolant flow is acceptable and meets the relevant requirements of General Design Criteria 10, 15 and 26. This conclusion is based on the following:

1. The applicant has met the requirements of GDC 10 and 26 with respect to demonstrating that the specified acceptable fuel design limits are not exceeded for this event. This requirement has been met since the results of the analysis showed that the thermal margin limits (MCHFR) are satisfied.

2. The applicant has met the requirements of GDC 15 with respect to demonstrating that the reactor coolant pressure boundary limits have not been exceeded for this event. This requirement has been met since the analysis showed that the maximum pressure of the reactor coolant and main steam systems did not exceed 110% of the design pressure.

3. The applicant has met the requirements of GDC 26 with respect to the capability of the reactivity control system to provide adequate control of reactivity during this event while including appropriate margin for stuck rods since the specific acceptable fuel design limits were not exceeded.

SYSTEMATIC EVALUATION PROGRAM
TOPIC XV-7

OYSTER CREEK

TOPIC XV-7: 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.

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

GDS 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, 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, in letters dated May 7, 1981 and August 25, 1981, provides the results of an analysis for the subject topic. Based on a qualitative comparison, the licensee has identified that the single reactor coolant recirculation pump rotor seizure is more limiting than the pump shaft break accident. This is because it 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 analysis indicates that, after a single reactor coolant recirculation pump seizure, the reactor power decreases in response to the reduced circulation flow. No reactor trip occurs and therefore, a loss of offsite power following turbine trip need not be assumed in this accident analysis. During power operation, there are five reactor coolant recirculation pumps in operation. The plant Technical Specifications also permit operation with only four reactor coolant recirculation pumps at full power. The results of the licensee analysis show that the single reactor coolant recirculation pump rotor seizure during four pump operation is the most limiting case with a bounding MCPR of 1.46. This MCPR is higher than the allowable MCPR of 1.34 for Oyster Creek. The results of the licensee's analysis also shows that the reactor coolant pressure decreases during the postulated accident.

The licensee has not provided the results of an analysis for this event in combination with a single failure. Since this event however, does not cause a reactor trip or any engineered safety feature initiation during this transient, we could not identify any single failure which will lead to unacceptable results.

VI. CONCLUSIONS

The staff concludes that the consequences of a postulated reactor coolant recirculation pump rotor seizure or broken shaft event meet the requirements set forth in the General Design Criteria 27, 28, and 31 regarding control rod insertability and core coolability. This conclusion is based upon the following:

1. The licensee has demonstrated that there is no fuel damage as a result of a postulated reactor coolant recirculation pump rotor seizure accident, because the Minimum Critical Power Ratio (MCPR) remains above the allowable MCPR limit.

2. The requirements of GDC 31 are met with respect to integrity of the primary system boundary to withstand the postulated accident.

SYSTEMATIC EVALUATION PROGRAM
TOPIC XV-15

OYSTER CREEK

TOPIC XV-15: INADVERTENT OPENING OF A BWR SAFETY/RELIEF VALVE

I. INTRODUCTION

The inadvertent opening of a safety or relief valve results in a reactor coolant inventory decrease and a decrease in reactor coolant system pressure. Neutron flux decreases due to additional void formation.

The pressure regulator senses the pressure decrease and partially closes the turbine control valves. No reactor trip occurs, and conditions stabilize at a power level near the initial power. The feedwater system is used to makeup the continuing loss of inventory.

If the pressure regulator fails to respond, the increased steam flow would cause a decrease in steam pressure and close the MSIV's. This event has been discussed in Section 1.3 of the report submitted by GPU Nuclear on May 7, 1981 (Reference 1).

If a Power Actuated Relief Valve (PARV) opens and fails to reclose, the torus could experience an increase in temperature since relief valves discharge into the torus. Closure of the MSIV's would not halt the blowdown since the relief valves are upstream of the isolation valves.

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.

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 occurrence.

GDC 15 "Reactor Coolant System Design" requires that the reactor coolant and associated protection systems be designed with suffient 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 discuss such items as ESF. Topic XV-19 reviews the spectrum of loss of coolant accidents.

IV. REVIEW GUIDELINES

The review is conducted in accordance with SRP 15.6.1.

The evaluation includes review of the analysis for the event and identification of the features in the p ant 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 has evaluated this event in Reference 1, which refers to an analysis performed for the inadvertent opening of a relief valve in Reference 2. No scram trip levels are reached during this transient. The opening of the valve drops the neutron power from the rated power level to about 94.2 percent at 2.1 seconds. Mid-core pressure drops from about 1034 psig to a low of about 1021 psig, within 10 seconds and then begins to rise. This pressure drop is sensed by the pressure regulator and the steam flow to the turbine is decreased by the closing of the turbine control valves by automatic action of the pressure regulator. Therefore, the reactor does not respond with a continuing pressure and temperature decrease. The net change is relatively insignificant. Therefore, minimum critical power ratio will not change significantly and no vessel or fuel stress limits are approached.

A possible single failure that would affect the course of this event would be a failure in the feedwater system so that water level is not maintained. Without feedwater, the auto depressurization sytem (ADS) would actuate to lower the pressure sufficiently to permit ~~low~~ the core spray systems to inject water into the reactor coolant system.

The effect of the inadvertent safety/relief valve opening on the suppression pool temperature is treated separately under Unresolved Safety Issue (USI) A-39.

VI. CONCLUSION

As part of the SEP review of Oyster Creek Nuclear Generating Station, the analysis presented by the licensee has been evaluated against the criteria of SRP Section 15.6.1 and found to be in conformance with the acceptance criteria.

VII. REFERENCE

1. Oyster Creek Nuclear Generating Station, Systematic Evaluation Program, Docket No. 50-219-1, May 7, 1981.

2. Amendment No. 65 to FSAR, Docket No. 50-219, December 30, 1970.