March 04, 1982

Docket No. 50-155
LS05-82-03-019

Mr. David J. VandeWalle
Nuclear Licensing Administrator
Consumers Power Company
1945 W Parnall Road
Jackson, Michigan 49201

Dear Mr. VandeWalle:

SUBJECT: BIG ROCK POINT - SEP TOPIC XV-1, DECREASE IN FEEDWATER TEMPERATURE, INCREASE IN FEEDWATER FLOW AND INCREASE IN STEAM FLOW

In your letter dated July 15, 1981, you submitted a safety assessment report on the above topic. The staff has reviewed your assessment and our conclusions are presented in the enclosed safety evaluation report. Our report completes this topic evaluation for the Big Rock Point plant.

As noted in the evaluation of the increase in feedwater flow transient, the staff will require Technical Specifications changes if credit is to be given for operation of the turbine bypass system in the analyses.

The enclosed safety evaluation will be a basic input to the integrated safety assessment for your facility. The assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this topic are modified before the integrated assessment is completed.

Sincerely,

SE04
1/1

Dennis M. Crutchfield, Chief
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Enclosure:
As stated

cc w/enclosure:
See next page

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BIG ROCK POINT PLANT
SEP TOPIC XV-1 EVALUATION
DECREASE IN FEEDWATER TEMPERATURE

I. INTRODUCTION

Loss of feedwater heating can result from the closure of steam extraction line bleeder trip valves to either the high pressure (HP) feedwater heater or the intermediate pressure (IP) feedwater heater. These valves may close as a result of high water level on the shell side of either feedwater heater. Feedwater heating can be lost also if extraction steam is bypassed around the heaters. The first case produces a gradual cooling of the feedwater. In the second case, the steam bypasses the heater and no heating of feedwater occurs. In either case, the reactor vessel receives cooler feedwater and causes an increase in core inlet subcooling. The decrease in coolant void fraction and the negative void reactivity coefficient result in a gradual initial increase in reactor power. The rate of power increase depends on which feedwater heater is no longer functioning. The operator will respond to any power increase resulting from cold feedwater by checking the control rod pattern and if necessary, inserting control rods to maintain an acceptable power level, or shutdown the reactor if required. Failure of the operator to control power or water level in the feedwater heaters will cause the reactor power to increase above rated. If only the HP heater is lost, power will reach steady state above 100%, but below the high power trip setpoint of 125%. If the IP heater or more than one heater is lost, the reactor will trip on high power thus terminating the transient.

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.

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 occurence.

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 reliaby 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 feactures 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

The analysis for this transient was performed using the RETRAN-01 version of RETRAN (Reference 1), which is a one-dimensional transient thermal hydraulic analysis computer program. The plant initial operating conditions were assumed to be 102% of licensed power level and 1350 psia of reactor pressure. Failure of the turbine bypass valve was assumed to be the single active failure coincident with the incident. An early reactor scram on high pressure of high flux would result from failure of the bypass valve to open during this event. The resulting consequences would be similar to but less severe than a full turbine trip without bypass. Following reactor scram, the emergency condenser would actuate and control reactor pressure to less than the primary relief valve set point. Failure of the bypass valve to close, following initial opening from turbine trip, would cause a relatively rapid plant depressurization. Since the reactor had already scrammed, fuel integrity would not be challenged by such a failure. The operator would be called on to manually close the bypass valve or its isolation valve, or the isolation valve may automatically close on high condenser pressure. In any event, the CHFR limit within the core would not be exceeded.

VI. CONCLUSIONS

As part of the SEP review for the Big Rock Point plant, the staff has evaluated the licensee's analysis of the loss of feedwater heating event. The results indicate (Ref. 1) that the system pressure rises rapidly and peaks at 1380 psia, and the maximum core heat flux also peaks at 118% of the initial value. The MCPR for this event is 1.43 and the maximum reactor coolant system pressure of 1870 psia would not be violated. We therefore, find the results of the analysis for the loss of feedwater heater transient acceptable.

REFERENCES:

1. Letter from R. A. Vincent to D. M. Crutchfield dated July 15, 1981, Enclosure entitled "Plant Transient Analysis of the Big Rock Point Nuclear Reactor"

BIG ROCK POINT PLANT

SEP TOPIC XV-1 EVALUATION

INCREASE IN FEEDWATER FLOW

I. INTRODUCTION

Failure of the feedwater control system which causes the feedwater regulating valve to open to its maximum position will permit excessive feedwater flow to the reactor. There is a gradual rise in the steam drum level and an increase in power because of the increase core inlet subcooling and the negative void coefficient of reactivity. As a result, the reactor will trip on high power approximately 15 seconds into the event. A high steam drum level alarm may not occur before the high power scram. Following reactor scram, the steam drum level will drop because of void collapse. The operator will then have approximately two minutes before a high drum level alarm occurs, at which time the operator will place the main feedwater valve control on remote manual mode and proceed to trip the feedwater pumps and the turbine. If the operator fails to respond to the high level alarm and terminate feedwater flow, water will reach the safety valves and discharge through these valves. The reactor depressurization system (RDS) and the core spray system (CSS) may be needed for long-term core cooling.

II. REVIEW CRITERIA

Section 50.34 of 10CFR 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 operation 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 assigned 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.

V. EVALUATION

The licensee utilized the RETRAN computer code to evaluate the consequences of a feedwater controller failure. The results are an increase in reactor power to a maximum value of 125.6%, a MCPR of 1.57 and maximum coolant pressure remains below the allowed (1870 psia). The plant initial operating condition was assumed to be 102% of licensed power level and the turbine bypass system is also assumed to be functional. In response to the power increase, the core heat flux increases to a maximum of 108% of the initial value. The increases are not as large as those resulting from load rejection without bypass, e.g., 350.5% of initial core power and 121.4% of initial heat flux.

VI. CONCLUSION

As part of the SEP review of Big Rock Point, we have evaluated the licensee's analysis of a feedwater controller failure event. Reference 1 indicates that the MCPR is 1.57 for this event and the maximum allowable reactor coolant pressure (110% of the design pressure) would not be violated. However, because the turbine bypass system is not a safety related system, and failure of the system in this transient would result in a more limiting condition than events such as load rejection or turbine trip, we require the licensee to:

1) Reanalyze the event of increase in feedwater flow assuming the

turbine bypass system fails to operate and demonstrate that the MCPR and maximum reactor coolant pressure satisfy the criteria stated in SRP sections 15.1.1, 15.1.2, 15.1.3, and 15.1.4. or

2) Institute a surveillance program for the turbine bypass system and specify the limitations to either reactor power or minimum critical power ratio in the Technical Specifications to cover the case where the turbine bypass system is found inoperable.

VII. REFERENCES

1. Letter from R. A. Vincent to D. M. Crutchfield dated July 15, 1981, Enclosure entitled "Plant Transient Analysis of the Big Rock Point Nuclear Reactor."

BIG ROCK POINT

SEP TOPIC XV-1 EVALUATION

INCREASE IN STEAM FLOW

I. INTRODUCTION

Failure of the initial pressure regulator (IPR) control system can cause the turbine admission valves to open, allowing maximum steam flow to the turbine. Primary system pressure will decrease and the resulting void fraction will cause a slight decrease in reactor power. The operator must take manual action to close the turbine admission valves until steam flow returns to the same level prior to failure. If control of turbine steam flow and primary system pressure has not been reached by the time the primary system pressure decreases to approximately 900 psig, the operator would, by procedure, trip the turbine (stop valve closed). Failure of the operator to perform these actions will not adversely affect the reactor operation as the reactor would still be operating at a steady power level of approximately 100% power.

The turbine bypass valves can also fail in such a way that the bypass valves open, partially or fully, causing an increase in steam flow and produce the same effect on the reactor as discussed previously. However, a rapid decrease in the main generator output will result. This is opposite to the effect produced by an IPR failure because steam is diverted from the turbine in this case. Manual action would be required to close the bypass valves or close the main steam isolation valves which will result in a reactor scram. The resulting effect is found to be less severe than IPR failure.

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.

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 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

Failure of the turbine admission valves would result in an initial increase of steam flow and a slight decrease in core power. Initial sharp increase in steam flow will lead to a lowe enthalpy of recirculation flow leaving the steam drum. Upon reaching the core inlet the cooler water will cause the previously decaying core power to increase. The transient power peaks at approximately 99% of the original power level. The resulting increase in core void fraction limits the small power transient and power will level off at approximately 98% of its initial value.

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 Big Rock Point, we have evaluated the licensee's treatment of the failure of an IPR to the open position. The results provided in Reference 1 indicate that the MCPR is 1.67 for this event and the maximum reactor coolant pressure (1870 psia) would not be violated. Therefore, we concluded that the results are in conformance with SRP Section 15.1.3 and are acceptable.

VII. REFERENCES

1. Letter from R. A. Vincent to D. M. Crutchfield dated July 15, 1981, Enclosure entitled "Plant Transient Analysis of the Big Rock Point Nuclear Reactor."