December 15, 1981

Docket No. 50-237
LS05-81- 12-045

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US NUCLEAR REGULATORY COMMISSION
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Mr. L. Del George
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
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Dear Mr. Del George:

SUBJECT: DRESDEN 2-SEP TOPIC XV-5, LOSS OF NORMAL FEEDWATER FLOW AND XV-9, STARTUP OF AN INACTIVE LOOP OR RECIRCULATION LOOP AT AN INCORRECT TEMPERATURE AND FLOW CONTROLLER MALFUNCTION CAUSING AN INCREASE IN BWR CORE FLOW RATE

By letter dated October 15, 1981, 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 Dresden 2.

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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Enclosures:
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DRESDEN 2

SEP TOPIC XV-5: LOSS OF NORMAL FEEDWATER FLOW

I. INTRODUCTION

Loss of feedwater flow could occur as a result of the simultaneous tripping of all feedwater pumps, or a feedwater controller failure that closes the feedwater control valves. When the feedwater flow drops to 20% of full flow, the recirculation loop control system is designed to reduce the speeds of the motor-generators (M-G's) to a minimum to protect the recirculation and jet pumps from cavitation. Loss of feedwater causes the water level in the reactor vessel to drop. The reactor protection system is designed to trip the reactor when the water level drops to the low-level set point. Due to the decay heat and the release of steam, the water level will continue to drop until the low-low water level set point is reached at which time the protection system is designed to close the main steam isolation valves, (MSIV's), trip the recirculation pumps, and start High Pressure Coolant Injection, (HPCI). An isolation condenser is designed to take away the decay heat after the MSIV's close. Should the HPCI system fail, the isolation condenser is designed to independently maintain the water level above the top of the active fuel.

The Commonwealth Edison Company (CECO) submitted two analyses of the loss of feedwater event at Dresden 2, on November, 1968 (Reference 1) and another in August, 1979 (Reference 2). This event was also assessed in a generic study for a topical report (Reference 3) which was published in May, 1977.

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 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 Redundance 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 15.2.6.

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

Since it is assumed that the reactor trips on a low-water-level signal and that the power then rapidly decreases, the critical power ratio does not decrease below its initial, steady-state value. The main concern in this event is the loss of too much water so that the fuel elements are uncovered and inadequately cooled.

From the analyses presented in Reference 2, CECO concluded that:

1. Dresden 2 is adequately equipped to mitigate the consequence of the loss of feedwater event, as it relates to core cooling, without operator assistance and with or without a stuck-open relief valve.
2. Operator actions are required only when there is a complete loss of high pressure injection (HPCI) and other water inventory systems. In such an event, timely manual depressurization followed by injection from low-pressure systems will mitigate the consequences.

   In its evaluation (Reference 4) of this report (Reference 2) the NRC stated that the basic requirements for this event, which are given in General Design Criteria 10 and 15, appear to have been met, even in the light of the TMI-2 experience.

In a generic review of incidents of moderate frequency, which includes the loss of feedwater flow event, Reference 3 concludes that a turbine trip is more limiting than a loss of feedwater in terms of minimum critical power ratio. The NRC in Reference 5 agreed with this conclusion.

VI. CONCLUSION

As part of the SEP review of Dresden 2, the analysis for loss of feedwater has been evaluated and we have concluded that the basic requirements of GDC 10 and 15 are met. The MCPR consequences of this event are bounded by those of a turbine trip. This aspect is reviewed under SEP Topic XV-3.

REFERENCES

1. Dresden Nuclear Power Station, Units 2 & 3, Safety Analysis Report; Commonwealth Edison Company, Chicago, Illinois; Chapter II; November, 1968.

2. General Electric Company; Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors; Revision 1; NEDO-24708A; August, 1979.

3. General Electric Company; Licensing Topical Report General Electric Boiling Water Reactor, Generic Reload Fuel Application; NEDE-24011-P-A; May, 1977; pps 5-11, 5-68, 5-69.

4. U. S. Nuclear Regulatory Commission; Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Generating Plants and Near-Term Operating License Applications; NUREG-0626; January, 1980; page G-3.

5. U. S. Nuclear Regulatory Commission; Safety Evaluation for the General Electric Topical Report, Generic Reload Fuel Application, (NEDE-24011-P); April, 1978; page C-62.

TOPIC XV-9

Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate

I. INTRODUCTION

The objective of this review is to assure that the consequences of core flow increase transients are acceptable, i.e., that the increase in core flow or the introduction of cooler water into the core does not lead to an unacceptable loss of fuel clad integrity or overpressurization of the primary system. The improper startup of an idle recirculation loop would cause a power increase due to a combination of a negative moderator coefficient reactivity addition and increasing recirculation flow. A flow controller malfunction can cause the scoop tube position to move at its maximum speed in the direction of increasing pump flow. This increasing pump flow will sweep voids at a faster rate and therefore cause an increase in neutron flux. The review consists of evaluating the licensee's analysis of the sequence of events, the analytical model, the values of parameters used in the analytical model and the predicted consequences of the transien

II. REVIEW OF 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 that Technical Specifications 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) set forth criteria for the design of water-cooled reactors.

GDC 10 "Reactor Design" requires that the core and associated cooling, control and protection system 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 20 "Protection System Functions" requires that the protection system be designed to initiate automatically the operation of reactivity control systems to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences.

GDC 26 "Reactivity Control System Redundance and Capability" requires that the reactivity control system 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.

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.

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.

VI. REVIEW GUIDELINES

The review is conducted in accordance with SRP 15.4.4 and 15.4.5.

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

Both of the subject transients were assessed in the General Electric generic reload ropical (Reference 1) and it was determined that the inadvertent flow increase transient was the most limiting of the transients at reduced flow.

For the FSAR analysis of the inactive loop startup, the inactive loop is assumed to be filled with 110°F water. The initial power level was 30% of full power. The analyses were performed at the end of equilibrium fuel cycle exposure condition in which the scram and void characteristics would be worst. The critical power ratio (MCPR) remains above the safety limit of 1.06 for both events. The general analytical methods have been approved by the staff (Reference 2).

For the flow controller malfunction transient, the assumed initial conditions are 65% power and 50% flow. The failed motor generator set speed controller causes the scoop tube position to move at a rate of 10%/second for nine seconds. The use of generic K_f factors to adjust the MCPR operating limits will ensure that the operating limit MCPR will not be violated for this event. Appropriate K_f factors have been reviewed and approved in Reference 2. Additional discussion of these events was provided in Reference 3 in which appropriate operator actions were identified.

VI. CONCLUSIONS

As part of the SEP review of Dresden 2, the startup of an inactive loop and flow controller malfunction transient were reviewed against the acceptance criteria of SRP Section 15.4.4 and 15.4.5. The initial conditions and analytical methods have been reviewed and found to conform to the requirements of the SRP.

REFERENCES

1. NEDE-24011-P "Generic Reload Fuel Application, with Revisions," May, 1977-March, 1978.

2. Staff Evaluation GE Generic Reload Application, May 12, 1978.

3. NEDO-24078-A "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," Volume K, August, 1979 Revision 1-December, 1980.