

December 29, 1981

Docket No. 50-237  
LS05-81- 12-095



Mr. L. Del George  
Director of Nuclear Licensing  
Commonwealth Edison Company  
Post Office Box 767  
Chicago, Illinois 60690

Dear Mr. Del George:

SUBJECT: DRESDEN 2 - SEP TOPIC XV-1, DECREASE IN FEEDWATER TEMPERATURE,  
INCREASE IN FEEDWATER FLOW AND INCREASE IN STEAM FLOW

In your letter dated October 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 Dresden 2.

As noted in the evaluation of the feedwater controller malfunction event, the staff will require Technical Specifications changes to conform with current licensing practice 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,

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

Enclosure:  
As stated

cc w/enclosure:  
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UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

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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,

*for* *Walter A. Paulson*  
Dennis M. Crutchfield, Chief  
Operating Reactors Branch No. 5  
Division of Licensing

Enclosure:  
As stated

cc w/enclosure:  
See next page

8112310316

Mr. L. DelGeorge

cc

Isham, Lincoln & Beale  
Counselors at Law  
One First National Plaza, 42nd Floor  
Chicago, Illinois 60603

Mr. B. B. Stephenson  
Plant Superintendent  
Dresden Nuclear Power Station  
Rural Route #1  
Morris, Illinois 60450

Natural Resources Defense Council  
917 15th Street, N. W.  
Washington, D. C. 20005

U. S. Nuclear Regulatory Commission  
Resident Inspectors Office  
Dresden Station  
RR #1  
Morris, Illinois 60450

Mary Jo Murray  
Assistant Attorney General  
Environmental Control Division  
188 W. Randolph Street  
Suite 2315  
Chicago, Illinois 60601

Morris Public Library  
604 Liberty Street  
Morris, Illinois 60451

Chairman  
Board of Supervisors of  
Grundy County  
Grundy County Courthouse  
Morris, Illinois 60450

John F. Wolf, Esquire  
3409 Shepherd Street  
Chevy Chase, Maryland 20015

Dr. Linda W. Little  
500 Hermitage Drive  
Raleigh, North Carolina 27612

Illinois Department of Nuclear Safety  
1035 Outer Park Drive, 5th Floor  
Springfield, Illinois 62704

U. S. Environmental Protection Agency  
Federal Activities Branch  
Region V Office  
ATTN: EIS COORDINATOR  
230 South Dearborn Street  
Chicago, Illinois 60604

Dr. Forrest J. Remick  
305 East Hamilton Avenue  
State College, Pennsylvania 16801

The Honorable Tom Corcoran  
United States House of Representatives  
Washington, D. C. 20515

## DRESDEN 2

### SEP. TOPIC XV-1: DECREASE IN FEEDWATER, TEMPERATURE, INCREASE IN FEEDWATER FLOW, INCREASE IN STEAM FLOW

#### I. INTRODUCTION

Feedwater heating can be lost by closure of the steam extraction lines to the heaters or the bypassing of feedwater around the heaters. In either case the reactor vessel receives cooler feedwater and there is 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 event could occur with the reactor in either manual or automatic control mode. In the automatic control mode, there is some compensation for reactor power increase by modulation of core flow and the event generally is less severe. In the manual control mode, and assuming no corrective operator actions, the reactor power could either reach a higher equilibrium value below the scram setpoint or increase sufficiently to cause automatic scram on high neutron flux. The power history depends on both the assumed maximum decrease in feedwater temperature and the feedwater temperature time constant.

The loss of feedwater heating event results in a mild transient in which the fuel surface heat flux increases to a maximum value below that corresponding to steady state operation at the scram setpoint. This increase in power to the coolant is partially offset by the beneficial effect of the increased core inlet subcooling on the critical power ratio. However, this event can be one of the limiting events with respect to minimum critical power ratio, and is considered in reload analyses.

#### 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 ob-

jective 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 15.1.1, 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. Deviations from the criteria specified in the Standard Review Plan are identified.

### V. EVALUATION

In reference 1, the licensee reported on calculations of reactor response to a loss of feedwater heating event in which feedwater temperature decreased by 145°F. Reactor power increased to the scram setpoint in about 90 seconds with a corresponding decrease in critical power ratio of 0.17. This event was analyzed for an initial power of 100 percent instead of 102 percent as required by SRP Section 15.1.1. Use of the higher initial power could result in a slightly larger decrease in critical power ratio. However, for Dresden 2 the generator load rejection without bypass and rod withdrawal at power events result in a decrease in critical power ratio of about 0.18. Hence reevaluation of the loss of feedwater heating event with 102 percent initial power should not change the operating minimum critical power ratio limit.

### VI. CONCLUSIONS

As part of the SEP review for Dresden 2, we have evaluated the licensee's analysis

of the loss of feedwater heating event. We conclude that the loss of feedwater heating event is bounded by generator load rejection without bypass. We, therefore, find results acceptable even though an initial power of 100% was assumed instead of 102% as required by the SRP acceptance criteria.

References

1. Y1003J01A06, "Supplemental Reload Licensing Submittal for Dresden Nuclear Power Station Unit 2 Reload 5", October, 1980.

INCREASE IN FEEDWATER FLOWI. INTRODUCTION

Failure of the feedwater controller to maximum demand results in an increase in reactor power and vessel inventory. There is a gradual initial increase in power because of the increased core inlet subcooling and the negative void coefficient of reactivity. The steam/feedwater flow mismatch leads to a high vessel water level trip of the main turbine. The turbine trip results in a pressurization transient, with attendant power transient, which is mitigated by reactor scram due to turbine stop/control valve closure and initiation of the turbine bypass system. The limiting conditions occur during the pressurization portion of the overall event.

A feedwater controller failure at partial power gives a larger steam/feedwater flow mismatch. However, failure at rated power can be more severe in terms of maximum reactor pressure and minimum critical power ratio. A feedwater control failure event at rated power is similar to the turbine trip event at rated power with turbine bypass operable. However, for the feedwater controller event, the turbine trip signal occurs when the reactor is at above rated power. Hence this event can be limiting with respect to minimum critical power and is evaluated in reload analyses.

To meet current criteria, surveillance of the turbine bypass system is required. Since the bypass system was assumed to operate in the analysis of this event, limitations to either reactor power or minimum critical power ratio would be required in the Technical Specifications to cover the case where the bypass system is found inoperable.

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 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 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 15.1.1, 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. Deviations from the criteria specified in the Standard Review Plan are identified.

## V. EVALUATION

In Reference 1, the licensee evaluated the consequences of a feedwater controller failure leading to an increase in feedwater flow to 110%. The initial power was 100% instead of 102% as required in SRP Section 15.1.2. However, the reduction in critical power ratio was only 0.09 compared to a reduction of 0.18 for load rejection without bypass. Hence, reevaluation of this event for an initial power of 102% would not result in a reduction in the operating minimum critical power ratio limit.

## VI. CONCLUSIONS

As part of the SEP review of Dresden 2, we have evaluated the licensee's analysis of a feedwater controller failure. We conclude that this event is bounded by load rejection without bypass. We, therefore, find the results acceptable even though an initial power of 100% was assumed instead of 102% as required by the SRP acceptance criteria.

To meet current criteria, surveillance of the turbine bypass system is required. Since the bypass system was assumed to operate in the analysis of this event, limitations to either reactor power or minimum critical power ratio would be required in the Technical Specifications to cover the case where the bypass system is found inoperable.

## REFERENCES

1. Y1003J01A06, "Supplemental Reload Licensing Submittal for Dresden Nuclear Power Station Unit 2 Reload 5," October 1980.

DRESDEN 2, SEP TOPIC XV-1 EVALUATION

INCREASE IN STEAM FLOW

I. INTRODUCTION

Failure of the pressure regulator in an open position results in full opening of the turbine admission valves and partial opening of the turbine bypass valves. The total steam flow rate resulting from the regulator failure is limited to approximately 110 percent of rated flow by a maximum flow limiter. The increased steam flow rate results in a drop in reactor pressure and inventory. The increase in core void fraction produces an initial decrease in core power and increase in vessel level. The vessel level increase can cause trip of the main turbine. Reactor scram then results from turbine stop/control valves closure. If the turbine trip signal or high water level is not reached, an MSIV closure on low steam pressure occurs. Reactor scram then results from position switches on the MSIV's. Since the turbine trip or MSIV closure occurs at relatively low reactor pressure and power, the pressure regulator failure event is not of consequence with respect to peak system pressure or minimum critical power ratio.

## 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 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 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 15.1.1, 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. Deviations from the criteria specified in the Standard Review Plan are identified.

## V. EVALUATION

The licensee considered this event in the FSAR but did not reanalyze the event for later fuel cycles. The event is not limiting with respect to peak system pressure or minimum critical power ratio.

## VI. CONCLUSIONS

As part of the SEP review for Dresden 2, we have evaluated the licensee's treatment of the failure of a pressure regulator to the open position. We conclude that the event is bounded by load rejection without bypass and is in conformance with SRP Section 15.1.3.