

January 4, 1982

Docket No. 50-237
LS05-82-01-004

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 TOPICS XV-7 AND XV-15

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

Enclosures:
As stated

cc w/enclosures:
See next page

SE04
5/11
Add: McKenna
DSA USE EX (16)

8201070001 820104
PDR ADOCK 05000237
P PDR

OFFICE	SEP:MM	SEP:SL	SEP:BC	ORB#5 PM	ORB#5 C	AD:SA:DL
SURNAME	EMcKenna:dp	Gova Lina	WRussell	P O'Connor	DCrutchfield	GLinas
DATE	12/28/81	12/28/81	12/28/81	1/4/82	1-4-82	1-4-82

Mr. L. DelGeorge

DRESDEN 2
Docket No. 50-237

cc

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

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

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

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

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

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

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. Wolfe, Esquire
3409 Shepherd Street
Chevy Chase, Maryland 20015

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

SEP TOPIC XV-7: LOSS OF FORCED COOLANT FLOW, REACTOR COOLANT PUMP ROTOR SEIZURE AND SHAFT BREAK

DRESDEN 2 NUCLEAR POWER PLANT

SUBJECT: LOSS OF FORCED REACTOR COOLANT FLOW INCLUDING TRIP OF PUMP MOTOR AND FLOW CONTROLLER MALFUNCTIONS

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 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.

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 (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 breached 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 a letter dated October 15, 1981, and Section 4.3.3 of the FSAR, 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 the motor generator (M-G) set speed controller, or failure of the pump. 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 two reactor coolant recirculation pumps in operation. The Dresden 2 is also permitted to operate with only one reactor coolant recirculation pump for four weeks at 50% power. The results of the licensee's analysis show that 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 Dresden 2 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.

DRESDEN 2

SUBJECT: 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.

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 Sections 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 27, 1981 and October 15, 1981, and in Section 4.3.3 of the FSAR, provides the results of an analysis for the subject topic. The licensee does not address the pump shaft break accident in his analysis. However, the results of analyses for other BWR plants indicated 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 two reactor coolant recirculation pumps in operation.

Dresden 2 is also permitted to operate with only one reactor coolant recirculation pump for four weeks at 50% power. The results of the licensee analysis show that the single reactor coolant recirculation pump rotor seizure during either one or two pump operation will have an MCHFR higher than allowable

MCHFR for Dresden 2. The results of the licensee's analysis also show that the reactor coolant pressure will increase slightly during the first two seconds after the accident and decrease during the remaining of the transient.

The licensee has not provided the results of an analysis for this event in combination with a single failure. Since this event does not cause a reactor trip or any engineered safety feature initiation during this transient, we could not identify any single failure which would 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 Heat Flux Ratio (MCHFR) remains above the allowable MCHFR limit.
2. The requirements of GDC 31 are met with respect to integrity of the primary system boundary to withstand the postulated accident.

SEP TOPIC XV-15

DRESDEN 2 NUCLEAR POWER PLANT

SUBJECT: 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.

A failed open relief valve blows down to the suppression pool. The safety valves discharge directly to drywell atmosphere.

On relief valve opening, 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.

Should the feedwater system become unavailable due to a single failure or loss of offsite power, the high pressure coolant injection (HPCI) system could provide water. HPCI would be automatically actuated on low-low water level.

If the pressure regulator fails to respond, the increased steam flow would cause a decrease in steam pressure and close the main steam isolation valves (MSIV's). This event has been discussed in Section 1.3 of the report submitted by Commonwealth Edison on October 15, 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 margin of safety during normal operational 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 specific 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 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 the Engineered Safety Features (ESF) and containment cooling systems. 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 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

Inadvertent opening of a safety/relief valve causes a negligible pressure reduction due to partial closure of the turbine control valve by the pressure regulator. The net change in power level and coolant conditions within the fuel assemblies is negligible and operating thermal margins are relatively unaffected. Therefore, minimum critical power ratio (MCPR) will not change significantly.

The licensee has submitted an evaluation of this SEP topic in Reference 1 and has referred to the accident analysis of Reference 2. However, this analysis (Ref.2) generically evaluates the effects of stuck open relief valves as an inventory threatening event. The purpose of this analysis was to show no fuel uncover for this event. The intent of Reference 2 was not to look at MCPR as required by Standard Review Plan (SRP) Chapter 15. References 1 and 2 did not address the potential for fuel failure as a result of this transient. On the basis of our evaluation of other similar plants (References 3 and 4) the staff feels, however, that there will not be any fuel failure due to stuck open relief valve event since MCPR does not change much as discussed above in the preceding paragraph. Therefore, no additional information is required from the licensee.

A possible single failure that would affect the course of this event would be a failure in the feedwater system. The failure of feedwater system will result in actuation of the HPCI. There should not be any fuel failure as a consequence of feedwater system failure since (i) the MCPR will occur before actuation of HPCI, (ii) there will not be any fuel uncover, and (iii) either natural circulation or forced flow is sufficient to prevent boiling transition at decay heat power level.

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

VI. CONCLUSION

As part of the SEP review of Dresden 2 Nuclear Power Plant, 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. REFERENCES

1. Transmitting Accident and Transient Topic Assessments Letter, T. Rausch (Commonwealth Edison) to D. Crutchfield (NRC), for the Dresden 2 Nuclear Power Plant, Docket No. 50-237, October 15, 1981.
2. NEDO-24708A - August 1979 Rev. 1 - December 1980, Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors Volumes 1 and 2 (Section 3.5.2.1.7).
3. Letter, D. Crutchfield (NRC) to W. Council (Northeast Nuclear Energy Co.) October 28, 1981. Subject: SEP Topic XV-15 for Millstone 1 (Docket 50-245).
4. Letter, D. Crutchfield (NRC) to I. Finfrock (Jersey Central Power & Light Co.) December 4, 1981. Subject: SEP Topic XV-15 for Oyster Creek (Docket 50-219).