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U.S. NUCLEAR REGULATORY COMMISSION
DOCKETS, LICENSING & ...

Docket No. 50-237
LS05-82- 02-063

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-8, CONTROL ROD MISOPERATION; TOPIC XV-11, INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN AN IMPROPER POSITION (BWR) AND TOPIC XV-13, SPECTRUM OF ROD DROP ACCIDENTS (SYSTEMS)

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 the review of 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,

Paul W. O'Connor, Project Manager
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DRESDEN 2
Docket No. 50-237

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TOPIC: XV-8 Control Rod Misoperation (System Malfunction or Operator Error), for Dresden Unit 2

1. INTRODUCTION

The staff has evaluated the Dresden Unit 2 reactor with respect to SEP Topic XV-8, Control Rod Misoperation, using the information submitted by Commonwealth Edison in their letter of October 15, 1981. Such events can result in reactivity addition and a subsequent increase in reactor power.

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 protections systems be designed with approporate 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 25 "Protection System Requirements for Reactivity Control Malfunction" requires that specified acceptable fuel design limits not be exceeded for any single malfunction of the reactivity control systems such as accidental withdrawal of control rods.

III. RELATED SAFETY TOPICS

Topic IV-2 describes the reactivity control system and any failure modes that could lead to control rod misoperation. Other SEP topics address such items as the reactor protection system.

IV. REVIEW GUIDELINES

The review is conducted in accordance with SRP 15.4.1, 15.4.2 and 15.4.3.

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 control rod misoperation event occurs in a boiling water reactor when a rod is withdrawn out of sequence. A rod may be out of sequence by reason of being the wrong rod or by being the correct rod but moved too great a distance (i.e., too many notches). The event may occur in the startup range or in the power range.

During startup and up to a preset power level (15 to 25 percent of full power) a rod withdrawal sequence is specified and enforced by procedures, by computer software, or by hard-wired circuitry. Above the preset power level rod movements are performed in such a way as to keep the power distribution within the requirements of the limiting conditions for operation and to achieve a desired power shape (e.g., the Haling distribution).

Rod Misoperation During Startup

During startup and low power operation the rod withdrawal sequence at Dresden Unit 2 is enforced by the Rod Worth Minimizer (RWM) a computer based system that provides motion blocks to the control rod drive system when out-of-sequence rod motions are attempted. In the event that the RWM is not operable a second operator is required to approve the rod selection and motion before the rod is moved. Correctly following the withdrawal sequence constitutes normal operation and reactivity additions are designed to permit untroubled increases in power through the startup range (e.g., no periods so short that a reactor trip occurs before the operator can take action to prevent it).

The probability that a misoperation event will occur in Dresden Unit 2 during startup is very small. Nevertheless a generic analysis of the consequences of such an event has been performed and the results presented in the LaSalle County Station Final Safety Analysis Report (Docket 50-341, Section 15.4.1). The calculation is performed in two steps; first a detailed analysis, including multidimensional effects, is performed for a rod having a worth higher than would be anticipated (1.6 percent reactivity change), after which point kinetics calculations are used to extrapolate the results to rod worths to be expected for out of sequence rods. Calculations were done with an initial reactor power of one percent of rated power because a sensitivity study had shown that the consequences were maximum at this level. Transient termination was assumed to occur by means of the APRM scram at low (15 percent) power or by the degraded (worst bypass condition) IRM scram. The withdrawal speed was assumed to be the maximum value attainable and rod worths up to 2.5 percent reactivity change were analyzed. In no case did a peak enthalpy greater than 60 calories per grams result. Our acceptance criterion for fuel damage is 170 calories for this event.

On the basis that the fuel loading and control rod designs of the Dresden Unit 2 reactor are essentially the same as that of the boiling water reactors for which the generic analysis of this event was performed we conclude that the analysis is applicable to the Dresden Unit 2 reactor. Thus the analysis of this event meets the current requirements and is acceptable.

Rod Misoperation at Power

The rod misoperation event at power occurs when the operator selects and withdraws an improper rod or withdraws a proper rod beyond proper limits. In order to analyze a bounding event several conservative assumptions are made. The core is assumed to be operating at full power with an assembly or assemblies in the vicinity of the rod to be withdrawn operating at the limiting condition for operation for linear heat generation rate or critical power ratio. The rod to be withdrawn is assumed to be fully inserted. The existence of a rod pattern which would produce these initial conditions implies earlier mistakes by the operator. The core is assumed to be free of Xenon which tends to maximize rod worths. The LPRM detectors which would yield the greatest response to the event are assumed to be inoperable or bypassed so that protective action is delayed.

The calculation is performed with a three-dimensional reactor simulator code and the assumption is made that the neutron and thermal fluxes have the same time response. The results of the calculation are used to establish the rod block setting for the rod block monitor. The currently used analysis is described in NEDO-20411, "Generic Reload Fuel Application," and is the same analysis that is used for all operating boiling water reactors. We thus conclude that the analysis of this event meets the requirements that are currently used and that it is acceptable.

Operator Action

No operator action is required for this event.

Deviation from SRP 15.4.1

The criterion used for fuel damage for the startup event is 170 calories per gram which is that used for the BWR rod drop accident. The transient response for the startup event is similar to that for the rod drop accident in that the power excursion takes place in a very short time (a few hundred milliseconds). We conclude that the enthalpy rise is an acceptable measure of fuel duty rather than the SRP criterion. This is consistent with current practice for this event.

VI Conclusions

Based on the evaluation presented above we conclude that Dresden Unit 2 meets present day requirements for the rod misoperation events.

TOPIC: XV-11, INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN AN IMPROPER POSITION (BWR) FOR DRESDEN UNIT 2

I. INTRODUCTION

The staff has evaluated the Dresden Unit 2 reactor with respect to SEP Topic XV-11, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position". We used the information submitted by Commonwealth Edison by their letter of October 15, 1981 in our evaluation which follows.

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 operating 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 13 "Instrumentation and Control" requires that instrumentation and controls be provided to monitor variables over anticipated ranges for normal operations, anticipated operational occurrences and for accident conditions as appropriate to assure adequate safety.

10 CFR Part 100.11 provides dose guidelines for reactor siting against which calculated accident dose consequences may be compared.

III. RELATED SAFETY TOPICS

None.

IV. REVIEW GUIDELINES

This review is conducted in accordance with SRP 15.4.7.

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 fuel misloading event consists of the inadvertent loading of and operation with a fuel assembly in an improper position. For boiling water reactors two different events are considered; a fuel assembly loaded into an improper core location; and an assembly properly located but improperly oriented (i.e., rotated by 90 or 180 degrees). It is possible but unlikely that a loading error would be detected by incore instrumentation. Thus no credit is taken for instrumentation in the analyses. The acceptance criterion for this event (Standard Review Plan, Section 15.4.7) is that normal operation with a misloaded assembly that cannot be detected by the instrumentation system shall not lead to offsite consequences greater than a small fraction of the 10 CFR Part 100 guidelines.

The analysis procedure employed for this event at Dresden 2 is described in NEDE-24011, "Generic Reload Fuel Application," a General Electric Topical report which has been reviewed and approved by the staff. A reload specific analysis is performed for the rotated bundle event but the mislocated bundle

event has been analyzed generically for operating plants. Based on this analysis it is concluded that a reload specific calculation is no longer required. These procedures represent current state-of-the-art for boiling water reactor analyses. No operator action is required.

VI. CONCLUSION

We conclude that the analysis of this event for the Dresden 2 reactor meets the criteria which are applied to the present generation of boiling water reactors and is acceptable. The analyses show that for neither type of misloading event are core fuel thermal limits violated. We therefore conclude that the requirements of Section 15.4.7 of the Standard Review Plan with respect to consequences are met.

TOPIC: XV-13, "SPECTRUM OF ROD DROP ACCIDENTS (BWR)"
DRESDEN UNIT 2

I. INTRODUCTION

The staff has evaluated the Dresden Unit 2 reactor with respect to SEP Topic XV-13, "Spectrum of Rod Drop Accidents". We used the information submitted by Commonwealth Edison by their letter of October 15, 1981 in our evaluation which follows.

II. REVIEW CRITERIA

Section 50.35 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 operating of the facility, including determination of the margins of safety during normal operations and transients 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 criteria for water-cooled reactors. 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

SEP Topic XV-19 considers the effects of rupture of the reactor coolant pressure boundary of the ejected rod.

IV. REVIEW GUIDELINES

The review is conducted in accordance with SRP 15.4.9.

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 sytems 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.

The potential radiological consequences are assessed in a separate evaluation.

V. EVALUATION

The rod drop accident in a boiling water reactor occurs when a rod blade becomes disconnected from its drive and sticks in the core. The drive is then withdrawn leaving the rod behind. At some later time, the rod blade becomes unstuck and falls rapidly out of the core. If the control rod has sufficient reactivity worth the potential for localized fuel damage exists.

In order to limit the worth of a potential dropped rod, a rod withdrawal sequence is defined and enforced, in Dresden Unit 2, by a Rod Worth Minimizer, a computer based device which blocks rod motion if an attempt is made to withdraw an out of sequence rod. The analysis of the rod drop event is then done for the maximum potential dropped rod under the assumption that the sequence is followed.

The analysis is performed generically and is described in General Electric Topical Report NEDO-10527, "Rod Drop Accident Analysis for Large Boiling Water Reactors," and Supplements 1 and 2 to that report. The bounding

analysis procedure is described in NEDO-24011-A, "Generic Reload Fuel Application." Both of these reports have been reviewed and approved by the staff and the methods and procedures are used on all currently operating BWR/3s, 4s, and 5s.

In the generic analysis the consequences of the accident (i.e., the maximum enthalpy deposited in the fuel) were obtained as a function of dropped rod worth for conservatively chosen values of the important input parameters. These include the Doppler reactivity coefficient, the shape of the scram curve, the rod drop speed and reactivity insertion shape, and the scram insertion time. No credit is taken in the generic analysis for feedback due to moderator heatup.

Sensitivity studies have shown that, for potential dropped rod worths of less then one percent reactivity change, it may be concluded that no further analysis is required. For worths greater than one percent a bounding analysis is done. In this analysis the values of the input parameters are compared to those used in the generic analysis. If all parameters are within bounds it is concluded that the generic analysis is applicable. If any parameters are non-conservative a cycle specific analysis is performed. This is the current practice for boiling water reactors and it is followed by Dresden.

No operator action is required and there are no deviations from the Standard Review Plan.

VI. CONCLUSION

On the basis that the same approved models and procedures are employed at Dresden Unit 2 as at current generation boiling water reactors to analyze the rod drop accident, we conclude that this analysis meets present day requirements and the results of the analysis are acceptable.