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V. A. Moore, Assistant Director for Light Water Reactors Group 2, RL

SER INPUT FOR THREE MILE ISLAND UNIT 2

Plant Name:	Three Mile Island Unit 2
Docket Number:	50-320
Milestone Number:	24-24
Licensing Stage:	OL
Responsible Branch and Project Leader:	LWR 2-2 H. Silver
Technical Review Branch Involved:	Fuels Section and Physics Section of Core Performance Branch
Description of Review:	Safety Evaluation Report Input
Requested Completion Date:	August 22, 1975
Review Status:	Complete

The Safety Evaluation Report input from the Reactor Fuels Section and the Physics Section of the Core Performance Branch are enclosed. The input relates to chapters 4 and 15 of the Three Mile Island Unit 2 FSAR.

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The fuel for Three Mile Island Unit 2 consists of cylindrical UO_2 fuel pellets enriched with U235 and contained in Zircaloy-4 cladding. The fuel rod is pressurized to a high pressure with pure helium gas. A space is provided above the fuel column to allow for the accumulation of gaseous fission products which will be released from the fuel during operation. These fuel rods are arranged in a 15x15 lattice array with spacer grids along the length to provide mechanical support and shape the coolant flow distribution.

Some of the dimensions of this fuel assembly are given in Table I which is taken from Table 1.3-1 of the Three Mile Island Unit 2 FSAR.

As the fuel for Three Mile Island Unit 2 is irradiated it will undergo a phenomenon known as irradiation densification in which pores in the fuel disappear due to irradiation and the fuel becomes more dense. This can cause several effects which must be considered in the design of the fuel and operation of the plant. The densification of the fuel will cause a decrease in the length of the pellet and the radius of the pellet. These changes in dimensions will cause the following to occur.

(a) A decrease in the pellet length will cause the linear heat generation rate to increase by an amount in direct proportion to the percentage decrease in pellet length.

(b) A decrease in the pellet length can lead to generation of axial gaps within the fuel column, resulting in increased local neutron flux and the generation of a local power spike.

(c) A decrease in the pellet radius increases the radial clearance gap between the fuel pellet and fuel rod cladding causing a decrease

Table I

15x15 Fuel Assembly Dimensions

Clad Material	Zircaloy-4
Fuel Material	UO_2 sintered
Fuel Diameter	.370
Cladding Thickness, in	.0265
Cladding Outer Diameter, in	.430
Fuel Density (% Theoretical)	92.5
Fuel Enrichment (weight % U235) (core average)	2.57
Fuel Column Active Length, in	144
Spacer Grids/Assembly	8
Fuel Rod Pitch, in	.568
Maximum Design Fuel Central Temperature, °F	4400
Cladding Surface Temperature at Design Power, °F	654
Fuel Assemblies	177
DNB ratio at Design Power	1.75
Fuel Rods/Assembly	208

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in the gap thermal conductance, and consequently in the capability to transfer heat across the radial gap. This decrease in heat transfer capab .ity will cause the stored energy in the fuel pellet to increase. A decrease in radial gap conductance also will degrade the heat transfer capability of the fuel rod during various transient and accident conditions.

It is also possible that the decrease in pellet length could lead to the formation of axial gaps in the fuel column which co ild allow the cladding to creep inward (due to the effects of neutron flux and coolant pressure) so that it may collapse into a gap.

Methods have been developed by Babcock and Wilcox to deal with these effects. They are described in references (1) and (2). These methods have been reviewed by the staff and verified by independent calculations so that the staff considers these models conservative for treating the effects of fuel densification.

During the construction permit stage of Three Mile Island Unit 2 the issue of clad failure mechanisms associated with a loss of coolant accident was discussed with Babcock and Wilcox. Babcock and Wilcox was committed to a program to develop a model which would adequately consider the failure of the fuel rod cladding during a LOCA.

Since that time the response of a power reactor to a LOCA has become a generic review item. The subject of the behavior of Babcock and Wilcox cladding during a LOCA has been considered under this generic topic. The experimental effort performed by Babcock and Wilcox to demonstrate the adequacy of the behavior of their cladding under LOCA conditions is described in reference (3). A description of the analytical procedures used to calculate the condition of the cladding during a LOCA is given in reference (4). The staff status report on the conformance of Babcock and Wilcox to the ECCS Final Acceptance Criteria is given in reference (5). Reference (5) also describes certain modifications which the staff has required in the Babcock and Wilcox cladding swelling and rupture model. With these modifications, the staff considers the Babcock and Wilcox model for the behavior of the fuel rod cladding during a LOCA to be in conformance with the AEC ECCS Final Acceptance Criteria.

During the construction permit stage Babcock and Wilcox discussed with the staff and with the ACRS proposed high burnup fuel tests. The primary purpose of the High Burnup Program was to demonstrate the capability of the fuel design for future operation at higher power levels and to determine the swelling rate of UO_2 as a function of burnup using fuel rods of the same design as those in the core in order to advance the state of the art. In addition to determining the swelling rate, the effects of several other variables, including the density, heat rate, cladding restraint and the resulting clad strain, were to be investigated.

A description of the proposed tests was given in the PSAR[6]. Babcock and Wilcox considers that these tests were not necessary to show that the fuel for Three Mile Island Unit 2 could operate safely to the proposed maximum burnup at the proposed maximum linear heat generation rate. The staff concurs. However, since the tests were meant to be prototypical of Three Mile Island Unit 2 fuel, the staff remained interested

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in the outcome of these tests, and pursued the subject during the Operating License stage of Three Mile Island Unit 2. Babcock and Wilcox has reported the results to the staff in the FSAR. These tests have demonstrated the capability of Babcock and Wilcox fuel of a design similar to that used in Three Mile Island Unit 2 to operate to a high burnup.

A possible fuel failure mechanism in fuel rods is pellet cladding mechanical interaction (PCMI). If the fuel is operated so that certain combinations of power, time at power, power increase, and rate of power increase are exceeded it is possible for the fuel pellet to interact with the cladding so as to perforate the cladding. The staff is presently pursuing this subject with Babcock and Wilcox as well as the other nuclear reactor vendors on a generic basis. Possible limits on the above named variables of power, power increase, time at power, and rate of power increase are being explored with the industry. Experience with the first cycle of operation of the Oconee-1 reactor which will use similar fuel showed no PCMI related failures.

References

1. "Fuel Densification Report", BAW-10054, Revision 1, Babcock and Wilcox May 1973.

2. "TAFY- Fuel Pin Temperature and Gas Pressure Analysis", BAW-10044, Babcock and Wilcox, April 1972.

3. "Effect of Fuel Rod Failure on Emergency Core Cooling", BAW-10009, June 1970.

4. "B&W's ECCS Evaluation Model Report with Specific Application to 177 FA Class Plants with Lowered Loop Arrangement", BAW-10091, August 1974.

5. "Status Report By the Directorate of Licensing in the Matter of Babcock and Wilcox ECCS Evaluation Model Conformance to 10 CFR 50, Appendix K". U.S. Atomic Energy Commission, Regulatory Staff.

6. Three Mile Island Unit 2 Preliminary Safety Analysis Report, Supplement No. 3, Answers to Staff Questions 11.0-AC and U.1-AC.

4.3 Nuclear Design

Our review of the nuclear design of the Three Mile Island Nuclear Station - Unit 2 (TMI-2) was based on information supplied by the applicant in the FSAR and amendments thereto and discussions with the applicant and the reactor supplier, Babcock & Wilcox. The nuclear design features of this plant are essentially identical to those of Rancho Seco Unit 1 which we have previously reviewed and approved. Extensive use was made of the startup report for Rancho Seco (Reference 1) in the performance of this review.

4.3.1 Design Bases

The design bases presented for the nuclear design of the fuel and reactivity control systems are satisfactory and comply with all applicable General Design Criteria of 10 CFR 50, Appendix A.

4.3.2 Description

4.3.2.1 Nuclear Design Description

Descriptions of the fuel assembly enrichments, physics of the fuel burnout process, burnable poison distribution, soluble boron concentrations, delayed neutron fractions, and neutron lifetimes have been provided. The values presented for these parameters meet the design bases and satisfy the applicable sections of the General Design Criteria.

4.3.2.2 Power Distribution

We have reviewed the methods used by B&W to calculate power distributions for both steady state and transient conditions. The major computation tool is PDQ-7, a diffusion theory code with industry-wide usage. We have reviewed the procedures employed by B&W to prepare cross-sections for use in these calculations. These procedures are similar to others used in the industry.

The startup report for Rancho Seco Unit 1 provides comparisons between calculated and measured power distributions for a reactor similar to TMI-2. These comparisons showed that total peaking factors were predicted to within ~7.0%. This value is within the 7.5% nuclear uncertainty factor which is applied to calculated peaking factors. In addition, our consultant (BNL) has performed an independent audit calculation of heat generation rates for the BOL (equilibrium xenon) power distribution for Rancho Seco 1. The results for peaking factors agreed with those calculated by B&W within ~ 3.5%.

On the basis of our review, we conclude that the applicant has made suitable predictions of core power distributions.

4.3.2.3 Reactivity Coefficients

Comparisons of calculated and measured reactivity coefficients are presented in the Rancho Seco Unit 1 startup report[1]. Isothermal temperature coefficient and moderator coefficient were measured at zero power. Power Doppler and Moderator coefficients were measured at various power levels. Calculation and measurement agreed to within $\sim 1 \times 10^{-5}$ $\Delta k/k/^\circ F$ at zero power for both coefficients. The measured moderator coefficient at near full power (91.4%) was within $\sim 0.5 \times 10^{-5}$ $\Delta k/k/^\circ F$ of the calculated value. The measured power Doppler coefficient at near full power agreed to within $\sim 4 \times 10^{-5}$ $\Delta k/k/\%FP$ with the calculated value. On the basis of this good agreement between measurement and calculation for an essentially similar reactor, we conclude that the applicant has made suitable predictions of the reactivity coefficients for TMI-2.

4.3.2.4 Control Requirements

To allow for changes in reactivity due to reactor heatup, load following, and fuel burnup with consequent fission product buildup, a significant amount of excess reactivity is built into the core. The applicant has presented information on first cycle reactivity control distribution for TMI-2, which is operated in the "feed and bleed" mode.

Soluble boron is used to control reactivity changes due to:

- moderator deficit from ambient to operating temperature
- equilibrium xenon and samarium buildup
- fuel depletion and fission product buildup throughout cycle life (that part not controlled by lumped burnable poison)
- transient xenon resulting from load following.

Regulating rods are used to control reactivity changes due to:

- moderator deficit from HZP to HFP
- power level changes (Doppler)

Lumped burnable poison rods are used for radial flux shaping and to control part of the reactivity change due to fuel burnup and fission product buildup. Part length control rods are used to maintain an axially balanced power distribution.

The applicant has presented data to show that adequate control exists to satisfy the above requirements with enough additional control to provide a shutdown $k_{eff} \leq 0.99$ during the initial and equilibrium fuel cycles with the most reactive rod stuck out of the core. Comparisons between calculated and measured rod worths have been presented for the similar Rancho Seco Unit 1 reactor in the startup report(1) for that unit. These comparisons showed a deviation of $\sim 6\%$ between measured and calculated values with the measured values being smaller. The stuck rod worth was measured to be about 11% lower than the calculated value. The rod worth available for shutdown was $\sim 3.5\%$ lower than predicted - well within the 10% uncertainty allowed for this value in safety analyses.

The soluble boron system is capable of shutting down the system and of maintaining it in the cold shutdown condition at any time during core life. This condition satisfies the requirements of General Design Criterion 26.

On the basis of our review, which has included the comparison between calculated and measured rod worths, we conclude that the applicant's assessment of reactivity control requirements is suitably conservative and that adequate negative reactivity worth has been provided by the control system to assure shutdown capability.

4.3.2.5 Control Rod Patterns and Reactivity Worth

The full length control rod assemblies are divided into two groups - control rods and shutdown (or safety) rods. Load changes will be made with the control rods and/or the soluble boron system. Rod insertion will be controlled by power-dependent insertion limits given in the Technical Specifications. These limits ensure that:

1) There is sufficient negative reactivity available to permit the rapid shutdown of the reactor with ample margin.
2) The worth of control rods that might be ejected in the very unlikely event of failure of a pressure barrier in a control rod drive mechanism will be no greater than that which has been shown to have acceptable consequences in the safety analysis.
3) The overall peaking factor does not exceed the limiting value used in the accident analysis.

We have reviewed the calculated rod worths and the methods used by B&W to obtain the worths. Our consultant, BNL, has performed independent calculations of rod worths for the similar Rancho Seco Unit 1 reactor. The BNL calculation agreed with the B&W values to within 2% for the regulating groups. We have also reviewed the comparison between calculated and measured values (see Section 4.3.2.4 above).

On the basis of our review, we have concluded that the rod groupings proposed for TMI-2 satisfy the requirements for safe shutdown and power distribution control and that values for ejected rod worth and stuck rod worth are sufficiently conservative.

4.3.2.6 Stability

The stability of the reactor to xenon-induced power oscillations and the control of such transients have been discussed by the applicant. The reactor is calculated to be always stable to azimuthal oscillations, but under certain conditions (BOL, full power, equilibrium xenon) may experience non-damped axial oscillations.

The stability of 177 fuel assembly B&W plants was investigated during startup tests for the Oconee Unit 1 reactor[2]. A diagonal (combination of axial and azimuthal) oscillation was induced at 75% FP and the reactor response was monitored for ~ 72 hours. The azimuthal component of the oscillation was damped but the axial component was divergent. At ~ 70 hours into the transient, the part length rods were used to suppress the axial imbalances which was reduced to near zero where it was kept.

While the Oconee reactor is a rodded plant and TMI-2 is unrodded, the behavior of the two reactors is expected to be similar. Xenon stability is dependent on flux level, power shape, and moderator coefficient. Of these, two (flux level and power shape) are more stabilizing for TMI-2 and one (moderator coefficient) is more stabilizing for Oconee. In all three cases the differences are small.

On the basis of this demonstration of the azimuthal stability of a similar reactor and the ability of the control system to suppress axial oscillations, we conclude that the reactor will not experience uncontrolled oscillation.

4.3.3 Analytical Methods

We have reviewed the analytical methods used by B&W to perform core design. The major design tool is PDQ-7 as diffusion theory code with industry-wide usage. Cross-sections for use with this code are prepared in a manner similar to that used by others in the industry. Comparisons between calculated and measured design parameters have been made during startup tests on six reactors designed by B&W. In all cases, the comparisons have been satisfactory. On the basis of our review, we conclude that the analytical methods used for the design of TMI-2 are acceptable.

REFERENCES

1. Sacramento Municipal Utility District, Rancho Seco Nuclear Generating Station, Startup Report, March 1975 (Docket 50-312).

2. Duke Power Company, Oconee Nuclear Station Unit 1, Startup Report, November 16, 1973 (Docket 50-269).

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15.0 ACCIDENT ANALYSIS

15.1 Uncontrolled Control Rod Group Withdrawal

We have reviewed the analysis of the rod withdrawal transient at low power (startup accident) and at full power. We have reviewed the range of parameters assumed in these analyses and the results of the calculations and we conclude that the analyses are satisfactory. The transients are terminated by the negative Doppler coefficient, the high pressure trip, or the nuclear overpower trip. The design overpower condition is not reached in any transient and the peak pressure never exceeds allowable limits.

15.2 Control Rod Misoperation

Control rods may be misaligned from their group average by as much as nine inches without appreciable effect on power peaking factors. If a rod is misaligned by more than nine inches, it is defined to be a dropped rod. This definition covers both the action of dropping a rod and of sticking a rod while moving a group. The maximum worth of a rod which may be dropped while operating at full power is 0.65% $\Delta k/k$.

We have reviewed the analysis of the dropped rod accident. The most severe accident occurs at EOL when the moderator and Doppler temperature coefficients have their most negative values. The initial power decrease is followed by a return to full power as the reactivity decrease due to the dropped rod is offset by the temperature decrease. Startup tests at Rancho-Seco Unit 1 have shown that heat generation rate and DNBR limits are not exceeded at full power when the most reactive rod is dropped into the core. The test was conducted at 40% FP and the measured peak heat generation rate was extrapolated to 102% FP with all uncertainties included.

On the basis of our review, we conclude that the discussion of the control rod misoperation transient is adequate.

15.3 Rod Ejection Accident

This design basis accident is assumed to be caused by the physical failure of a pressure barrier component in a control rod drive mechanism which results in the rapid ejection of a control rod assembly. The maximum worth rod which may be ejected is limited by Technical Specifications to 1.0% $\Delta k/k$ at zero power and 0.65% $\Delta k/k$ at full power.

We have reviewed the rod-ejection analysis presented in the FSAR. The requirements of Regulatory Guide 1.77 are met. Analyses were performed for zero and full power conditions at BOL and EOL. The limiting case is that a full power at BOL. The environmental consequences of the postulated accident are shown to be acceptable.

Three-dimensional effects are treated in the analysis by assuming a larger than normal radial peaking factor (to account for the effect of the ejected rod) with the design axial peaking factor.

The peaking factor is assumed to be unchanged during the transient. Actually the power would be depressed prior to initiation of the transient and would peak during the transient. To compensate for the neglect of the change in peaking factor the reactivity feedback during the transient is based on the average rod in the core (i.e., a peaking factor of 1.0). Calculations in two dimensions (using the TWIGL code) have shown that this procedure is conservative in the expected range of ejected rod worths.

Birkhofer, et al[1] have reported the need for three-dimensional time-dependent calculations to predict correctly peak flux and temperature distributions for super-prompt-critical reactivity excursions. B&W has submitted comments[2] on this article, observing that the Birkhofer article addressed a BWR which has several core features that tend to make three-dimensional effects more important than in a PWR. Among these are the magnitudes of the feedback coefficients, the geometrical design of the control rod, and the reactivity control scheme. B&W further observes that the rod worth for the case analyzed was 1.67% $\Delta k/k$, which is more than a factor of two larger than the Technical Specification limit for B&W reactors. It is to be expected that three-dimensional effects would increase in importance rapidly as a function of rod worth. Also, the two-dimensional problem analyzed by Birkhofer had an axial peaking factor of unity as opposed to a value 1.7 used in B&W analyses. Application of this factor to the two-dimensional results will bring them in line with those of the three-dimensional calculations.

We agree with the comments of B&W regarding the weaknesses of the Birkhofer article. Until full three-dimensional time-dependent calculations are performed, however, we are unable to ascertain the magnitude of the uncertainty invovled in the synthetic three-dimensional treatment performed by B&W. Meanwhile, we accept the analysis of the rod ejection accident on the basis that the computed maximum enthalpy of the hottest fuel rod is of the order of 200 cal/gm, which is far below our acceptance criterion of 280 cal/gm. It is very unlikely that residual three-dimensional effects could cause the 280 cal/gm value to be exceeded.

REFERENCES

1. A. Birkhofer, A. Schmidt, and W. Werner, Nuclear Technology, 24, pp. 7-12, October 1974.

2. Letter, J. Mallay to V. Stello, dated February 5, 1975.