Safety Evaluation

License No. DPR-50

Docket Number 50-289

Three Mile Island Nuclear Station, Unit 1

Cycle 3 Reload

Introduction

By letter dated January 26, 1977[1], Metropolitan Edison Company (the licensee) requested changes in the technical specifications appended to Operating License DPR-50 for the Three Mile Island Nuclear Station, Unit 1 (TMI-1). The proposed changes relate to the discharge of the batch 2 fuel assemblies and replacement with fresh batch 5 assemblies plus assemblies saved from cycle 1a, thus constituting refueling of the reactor for operation in cycle 3. In addition, the proposed changes include operating limits based on an evaluation of ECCS performance calculated in accordance with an acceptable evaluation model that conforms to the requirements of the Commission's regulations in 10 CFR Section 50.46.

Reload Description

The TMI-1 reactor core consists of 177 fuel assemblies, each with a 15x15 array of fuel rods. The reload in preparation for cycle 3 operation[2,3] consists of the removal of all batch 2 assemblies, the relocation of batch 3 and 4 assemblies, and the introduction of 13 batch 1a and 48 new batch 5 assemblies. The batch 5 assemblies will be located at the core periphery and the batch 1a assemblies will occupy 13 positions within the mixed central zone.

Fuel Mechanical Design

The outside dimensions and configuration of the new Mark B-4 (Batch 4 & 5) fuel assemblies and older Mark B-3 (Batch 3) fuel assemblies are identical except that the Mark B-4 have spring-type flexible spacers and the Mark B-3 have corrugated-type flexible spacers. This new fuel rod spacer has been previously reviewed and found acceptable by the NRC staff on the basis of no significant mechanical or material change to the reactor operation[4] and has been successfully operating in similar cores for a substantial time (Reference Section 4.5 of Reference 1). The new Mark B-4 fuel assemblies, therefore, do not represent any unreviewed or untested change in mechanical design from the reference cycle and are therefore acceptable.

This mechanical design change has been taken into account in the various analyses which are discussed in the following sections. The results of these analyses have shown that this fuel design difference in the TMI-1 core is of negligible effect.

Fuel rod cladding creep collapse analyses were performed for the cycle 3 core. The CROV computer code was used to calculate the time to fuel rod cladding creep collapse[1,5]. The calculational methods, assumptions, and data have been previously reviewed and approved by the NRC staff[6]. The analysis assumed a 2000 hour densification time which maximizes creep; no fission gas production which maximizes differential pressure; and a lower tolerance limit on clad thickness and an upper tolerance limit on cladding ovality, both of which maximize cladding creep deformation.

The batch 3 fuel was found to be more limiting than the batch 4, 5, and 1a fuel due to the lower prepressurization, lower pellet density, and previous power history. The most limiting assembly in batch 3 was found to have a collapse time longer than the maximum projected three-cycle core exposure (24,288 EFPH).

From the viewpoint of cladding stress due to differential pressure, thermal stress due to fuel temperature gradients, and bending stress, neither the yield stress nor the B&W 1% total strain criterion for the cladding is predicted to be exceeded in the cycle 3 core.

The Batch 5 fuel assembly design is based upon established concepts and utilizes standard component materials. Therefore, on the bases of the analyses presented and previously successful operations with equivalent fuel the staff concludes that the fuel mechanical design for cycle 3 operation is acceptable and its application to cycle 3 operation will not endanger the health and safety of the public.

Fuel Thermal Design

The fuel thermal design analysis was conducted with the TAFY-3 computer code, as discussed in reference 7. The analysis considered the effect of a power spike from fuel pellet densification, as modeled in the "Fuel Densification Report"[8]. Modifications to the "Fuel Densification Report" on the fuel pellet void probability, F_g, and fuel grain size distribution, F_k have been previously reviewed and approved by the NRC staff.[9]

Based on the analyses presented in reference 1 and comparison with allowable Linear Heat Generation Rate (LHGR) for fuel centerline melt considerations, the fuel thermal design for the cycle 3 core is acceptable and can be applied with reasonable assurance that the health and safety of the public will not be endangered.

Fuel Material Design

The fuel material design for cycle 3 operation is not significantly different from that of cycle 2 operation. The only difference is that Zircaloy-4 is used as the fuel assembly tubular spacer material in Mark B-4 fuel instead of zirconium dioxide (ZrO_2), which is used in Mark B-3 fuel. This change does not affect the fuel system chemistry. This change has been reviewed and has a substantial amount of previous experience (Section 4.5 of reference 1). Therefore, the fuel material design for TMI cycle 3 operation is acceptable.

Nuclear Design

The TMI-1 reactor has completed two operating cycles and is thus sufficiently close to equilibrium cycle to show only minor changes in physics parameters. The cycle 3 core will consist of four distinct fuel types: fresh batch 5 assemblies located at periphery, once-burned batch 4 assemblies located generally in an intermediate zone and also near the core center, twice-burned batch 3 assemblies located between the periphery and the intermediate zone, and located between the intermediate zone and central zone, plus 13 batch 1a assemblies loaded with the batch 4 assemblies. Thus, although the cycle 3 core is a four batch loading, the physics parameters are

quite close to those of the cycle 2 core. In addition, these parameters will be verified during the startup testing program described later.

The only significant procedural change from the reference cycle (cycle 2) is the specification of axial power shaping rod (APSR) position limits. The APSR position limits will provide additional control of power peaking through an improved definition of the core power distribution.

The calculational methods used by the licensee are the same as were used for cycle 2.[10] Because of this, and because of the verification provided by the physics testing which will be performed during the cycle 3 startup, the staff finds the nuclear design for cycle 3 to be acceptable.

Thermal-Hydraulic Analysis

Major acceptance criteria for the thermal-hydraulic design are specified in the NRC's Standard Review Plan Section 4.4 ("Thermal and Hydraulic Design"). These criteria establish the acceptable limits for DNBR (Departure from Nucleate Boiling Ratio). The thermal-hydraulic analyses for the TMI-1 cycle 3 reload core were made with previously approved models and methods, as stated in the TMI-1 Final Safety Analysis Report[11].

The reactor coolant flow rate was accurately measured during cycle 1 operation and a minimum measured value of 108% of the system design flow was determined. The licensee has taken credit in the cycle 2 & 3 thermal-hydraulic analyses for the fact that the actual system flow is greater than the design flow rate, and has also included uncertainties and conservatisms in this analysis.[1, 10] The new design flow is 106.5% of the cycle 1 design flow.

In the past, a reactor coolant flow penalty had been assumed in the thermal-hydraulic design analysis for TMI-1. This penalty was associated with the potential for a core internal vent valve to be stuck open during normal operation. The core internal vent valves are incorporated into the design of the reactor internals to preclude potential vapor lock during a postulated cold-leg break Loss-of-Coolant Accident (LOCA). The NRC staff has concluded that by application of a surveillance program the vent valve flow penalty may be removed. The surveillance requirements demonstrate that the vent valves are not stuck open and that the vent valves operate freely. A separate review of the Licensee's surveillance program for the vent valves has concluded that the program adequately meets the staff's requirements, and that the vent valve penalty was properly eliminated[12].

The effect of fuel rod bow was evaluated by the Licensee with consideration given to both the hot channel power spike and the effect on DNBR. This evaluation was also separately reviewed and accepted by the staff[12].

There are differences in the flow resistance between the Mark B-3 fuel assemblies and the Mark B-4 assemblies. The flow resistance for a Mark B-4 fuel assembly is slightly less than that for the Mark B-3 assemblies. For the cycle 3 loading, the highest assembly power always occurs in a Mark B-4 assembly. The cycle 2 analysis[10], also used for cycle 3 reference evaluation[1], assumed the hot assembly to be a Mark B-3 type. This analysis is conservative for cycle 3 because the predicted hot assembly coolant flow rate is less than that of a corresponding Mark B-4 assembly.

Because of the analyses discussed above, we have found the thermal-hydraulic analysis to be acceptable and the proposed Technical Specifications related to the thermal-hydraulic analysis also acceptable.

Accident and Transient Analyses

A generic LOCA analysis for a B&W 177 assembly lowered-loop plant has been performed using the Final Acceptance Criteria ECCS evaluation model[13, 3]. This analysis has been reviewed by the staff[14], and found applicable to the TMI-1 cycle 3 core.

All other accidents and transients (loss of flow, dropped rod, inadvertent bank withdrawal, etc.) have been examined by the licensee for cycle 3 and found to fall within the bounds of the FSAR analyses, as updated for cycle 2 operation. The staff has reviewed

the various input parameters for cycle 3, and has found the licensee's conclusion acceptable.

Startup Program

The licensee has proposed a startup program which will verify:

- Critical boron concentration
- Temperature reactivity coefficient at two points
- Control bank worth by boron swap. More than half of the required shutdown reactivity will be verified
- Control bank worth by bank drop. The remainder of the banks will be checked by this method.
- Ejected rod worth

In addition, during the power escalation phase, the startup program will verify:

- Power distribution at three plateaus.
- Dropped-rod power distribution
- Incore/excore imbalance correlation
- Doppler coefficient at 100% power
- Temperature reactivity coefficient at 100% power

The staff has reviewed this proposed startup program and has found it acceptable.

Technical Specifications

The licensee has proposed revisions to the technical specifications to implement the changes due to the cycle 3 reload[1]. The staff has

reviewed the revised technical specifications and found them acceptable except for the following modifications, which we will require and to which the licensee has agreed:[3]

. Add the following:

3.1.7.2 The moderator temperature coefficient shall be $\leq + 0.5 \times 10^{-4}$ $\Delta k/k/F$ at power levels $\leq$ 95% of rated power.

. Revise 3.5.2.7 to read:

3.5.2.7 A power map shall be taken at intervals not to exceed 30 effective full power days using the incore instrumentation detection system to verify the power distribution is within the limits shown in Figure 3.5-2J.

Conclusion

Based on our evaluation of the application and available reload information as set forth above, and assuming compliance with the requirements set forth above, we conclude that it is acceptable for the licensee to proceed with cycle 3 operation in the manner proposed.

References

1. Letter, R. C. Arnold (Metropolitan Edison) to Director of Nuclear Reactor Regulation, dated January 26, 1977, enclosing Technical Specification Change Request No. 45 and BAW-1442.

2. BAW-1442, Three Mile Island Unit 1 Cycle 3 Reload Report, November, 1976.

3. Letter, R. C. Arnold (Metropolitan Edison) to Director of Nuclear Reactor Regulation, dated March 31, 1977, enclosing responses to Round 1 Questions concerning the TMI-1 cycle 3 Reload Application.

4. SER on Oconee Nuclear Station, Units 1,2,&3, dated June 30, 1976, Amendment Nos. 27, 27, and 23 for License Nos. DPR-38, DPR-47, and DPR-55.

5. BAW-10084P, Rev. 1, Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, October, 1976.

6. Letter, A. Schwencer (NRC) to J. F. Mallay (B&W), dated January 29, 1975.

7. BAW-10044, TAFY - Fuel Pin Temperature and Glass Pressure Analysis, May, 1972.

8. BAW-10055, Rev. 1, Fuel Densification Report, June, 1973.

9. Memorandum from R. Lobel to D. F. Ross, "Present Status of B&W Power Spike Model," July 23, 1974.

10. Three Mile Island Unit 1 - Cycle 2 Reload Report, Rev. 1, July, 1976.

11. Three Mile Island Unit 1 Nuclear Station, Final Safety Analysis Report, USNRC Docket No. 50-289.

12. Safety Evaluation by the Office of Nuclear Reactor Regulation, Amendment No. 25 to Facility Operating License No. DPR-50, dated March 7, 1977.

13. BAW-10103, Rev. 1, ECCS Analysis of B&W's 177-FA Lowered Loop NSS, September, 1975.

14. Letter, D. F. Ross (NRC) to D. B. Vassallo (B&W), Re: Topical Report Evaluation BAW-10104, ECCS Evaluation Model, Revised Nucleate Boiling Lockout Model, dated February 2, 1977.