UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 41 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION UNIT NO. 1

DOCKET NO. 50-293

1. Introduction

By letter dated November 21, 1979,[1] Boston Edison Company (the licensee) has requested an amendment to the Technical Specifications for the Pilgrim Nuclear Power Station. The effect of the amendment would be to allow multiple control blade removal with the reactor in the refuel mode. Analyses supporting this amendment were submitted on December 7, 1979.[2]

2. Discussion

2.1 Motivation

When the reactor mode switch is in the "refuel" position, the refueling interlocks will allow one control blade, but no more than one, to be withdrawn. (When fuel assemblies are being removed or inserted, additional interlocks on the fuel handling equipment will block withdrawal of even one blade.)

The licensee wishes to override this interlock. This will allow scheduled maintenance to be done on several control rod drives simultaneously. In addition to shortened downtime, with a probable economic benefit to the licensee, we note that personnel exposure (man-rem) should be reduced by this change.

2.2 Safety Concern

The safety concern is to ensure that the reactor remains subcritical, i.e., that shutdown margin is preserved. The licensee proposes to do this by removing the four fuel assemblies surrounding each blade to be withdrawn, before overriding the interlock on that particular blade. Although it would seem obvious that removing the fuel surrounding a blade would introduce negative reactivity, it is not necessarily true that fuel removal would more than compensate for removal of the control rod. Therefore, the licensee has calculated the change in shutdown margin for a wide range of configurations and fuel enrichment to demonstrate that, for Pilgrim 1, fuel removal does indeed more than compensate for control blade removal.

3. Evaluation

3.1 Codes Used

The basic methodology used in the analysis of the multiple control rod removal configurations relies on the use of CASMO, a two-dimensional lattice code which calculates macroscopic cross sections for a variety of fuel assembly conditions, and PDQ-7, a fine mesh diffusion theory code used here in a two-dimensional mode to calculate K_{eff} for the various core configurations.

To provide confidence in this methodology, two sets of benchmark calculations were performed. First, a set of CASMO calculations for a broad group of uniform lattices was checked against experiment. Since the output of CASMO forms the basis input to the other calculations, a check of this nature is essential.

Second, the 2D PDQ code was used to calculate the K_{eff} of actual BOC cold critical configurations in the Pilgrim reactor. Because of the two-dimensional nature of these PDQ calculations, the code can only consider fully inserted and fully withdrawn blades. Thus, the K_{eff} of the actual core configurations was based on two PDQ calculations, one with all partially withdrawn rods assumed full out and one with all partially withdrawn rods assumed full in, and the assumption was made that the reactivity worth of the partially withdrawn rods was linearly related to the number of notches of withdrawal.

These benchmark calculations are very limited. The CASMO benchmark calculations were performed only on uniform lattices, and thus were not completely realistic, since a BWR core contains water gaps, channel boxes and control blades. In addition, this assumption of linear incremental rod worth in the PDQ benchmark calculations is very approximate, since actual incremental rod worths are quite non-linear. This error can be considered conservative in the sense that the ultimate use of these codes does not involve partially withdrawn blades, and thus any error estimate derived from the benchmark calculations should overestimate the error in the end use calculations. Because of this, and because of the limited number of benchmark cases, we cannot at this time extend general approval of this code package, even though we consider it adequate for this "trend" analysis, as described below.

3.2 Analysis

The goal of this analysis was to show that the withdrawal of one or more control cells from the Pilgrim 1 core will lead to a more subcritical state. The licensee calculated K_{eff} for the BOC-5 core, then re-calculated

K_{eff} assuming one or more empty control cells. Fifteen cases were run, with the number of empty cells varying from one to 16. The cases studied included a full range of positions of the empty cells within the core and positions of the empty cells with respect to one another. All the calculations assumed a cold xenon-free core, a conservative assumption. In every case, the core became more subcritical.

Because of the gadolinia loading in a BWR core, the most reactive state occurs after BOC. To conservatively bound this effect, the licensee recalculated two of the cases, this time with the gadolinia removed from the fresh fuel. We agree that this should bound the reactivity swing actually experienced during the cycle. Again, the core becomes more subcritical in each case.

The licensee then attempted to bound all possible loadings by calculating a hypothetical core containing assemblies of the maximum enrichment currently available, with the gadolinia removed. This results in a far more reactive core than could be loaded in actual practice. Six cases were studied, and in each case the removal of blade plus fuel made the core reactivity decrease.

Finally, the licensee studied this same highly reactive core (in quarter-core geometry) with one cell containing four highly burned assemblies. Thus, core reactivity is maximized, the rod worth is maximized, and the negative reactivity effect of fuel removal is minimized. Here too the core became (slightly) more subcritical.

We agree that the cases studied should bound any configurations encountered in actual practice. Although we cannot give general approval to the codes used at this time, we do agree that the safety concern of interest here (preservation of shutdown margin) has been adequately addressed because:

- Shutdown margin increased in all cases studied. It is very unlikely that any random error in the calculation of K_{eff} could have led to this result.

- Because all cases involve the differencing of two calculations, most systematic errors should cancel out. The major exception is the cross sections of the water-filled cavity, for which the licensee has provided a separate sensitivity study.

- A completely separate calculation, done by the NRC staff, has shown the same trend.(3)

- Another completely separate calculation, on another docket, also supports this trend.(4)

Therefore, we find these calculations acceptable.

3.3 Technical Specification Implementation .

Shutdown margin, as defined in the plant Technical Specifications and explicitly addressed in Specifications 3.3.A.1 and 4.3.A.1, is required to be maintained with the highest worth control blade stuck out of the core. The calculations described above did not consider the effect of withdrawing a control blade in addition to the blade associated with the water-filled cavity, nor did the licensee's proposed change address this situation. Therefore, we will require that the 8 blades surrounding any such cavity be fully inserted and disarmed. Blades which are inserted and disarmed need not be considered in the search for the strongest rod, consistent with Specifications 4.3.A.1 and 3.3.A.2.b. Thus, with the consent of the licensee, the proposed specifications[1] have been altered as follows:

Replace Proposed Specifications 3.10.D.d and 3.10.D.e with:

d. All control rods in a 3x3 array centered on each of the control rods being removed are fully inserted and electrically or hydraulically disarmed, or have the surrounding four fuel assemblies removed from the core cell.

e. All other control rods are fully inserted.

f. The four fuel assemblies are removed from the core cell surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel.

Replace Proposed Specifications 4.10.D.d and 4.10.D.e by:

d. All control rods in 3x3 array centered on each of the control rods removed or being removed are fully inserted and electrically or hydraulically disarmed, or have the surrounding four fuel assemblies removed.

e. All other control rods are fully inserted.

f. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism that is to be removed from the reactor vessel at the same time are removed from the core and/or reactor vessel.

4. Summary

We have examined the calculational methods used and found them acceptable for this purpose. We have examined the set of cases studied and found it adequate. Therefore, with the change in the amendment detailed in Section 3.3 above, we find this amendment to be acceptable.

5. Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

6. Conclusion

We have concluded based on the considerations discussed above that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7. References

1. Letter, J. E. Howard (Boston Edison) to T. A. Ippolito (NRC), dated November 21, 1979.

2. Letter, G. C. Andognini (Boston Edison) to T. A. Ippolito (NRC), dated December 7, 1979.

3. Memo, L. Kopp (Core Performance Branch/DSS/NRC) to D. Fieno (Core Performance Branch/DSS/NRC), dated July 18, 1979.

4. Technical Specification Change Request No. 51 to Provisional Operating License No. DPR-16, Oyster Creek Nuclear Generating Station, Docket Number 50-219, dated November 19, 1976.

Dated: February 22, 1980