Enclosure

Physics Section Safety Evaluation
Basic Safety Report

Millstone, Unit No. 2
Docket No. 50-336

1.0 SUMMARY OF REPORT (Physics)

This report describes the reactor physics methods used by Westinghouse for the performance analysis of the Millstone Unit 2 reactor built by Combustion Engineering (CE). These methods will be used for the nuclear design of Westinghouse reload fuel for Millstone 2 beginning with Cycle 4. The report addresses the reactor model description, the calculational model verification, and the application of the physics methods to both operating reactor conditions and to reload safety evaluations.

The calculational methods used to analyze the nuclear design of Millstone 2 consist of standard Westinghouse nuclear design procedures, modified to accommodate the differences between Millstone 2 and Westinghouse PWR cores. A description of this revised design procedure is given in the report with the differences from standard Westinghouse design procedures emphasized. The following physics related computer codes are used:

1. LEOPARD-CINDER, linked spectral codes which are used to obtain burnup dependent neutron cross sections for fuel cells.
2. HAMMER-AIM, linked spectral codes which are used to obtain neutron cross sections for the CEA rods.
3. TURTLE, a three-dimensional neutron diffusion-depletion code used to obtain power distributions, fuel depletion, critical boron concentrations, xenon distributions, reactivity coefficients, and control rod worths.

4. PANDA, a one-dimensional diffusion theory code with local feedback which is used to obtain axial power distributions, differential control rod worths, and axial xenon distributions.
5. PALADON, a two-dimensional nodal code which is used to obtain power distributions, fuel depletion, critical boron concentrations, reactivity coefficients, and control rod worths.

The Westinghouse standard nuclear design methods that have been adapted to the Millstone 2 reactor are verified by benchmarking against Millstone 2 measurements over the first three cycles of operation and by comparisons with higher order analysis. An addendum to the BSR (Ref. 1) describes the power peaking factor uncertainty analysis utilized in the nuclear design of Millstone 2 and is also based on measured data from the first three cycles of operation. The following physics parameters are addressed:

1. Control Rod worth comparison to measurement,
2. isothermal temperature coefficient comparison to measurement,
3. power distribution comparisons to measurement, and
4. critical boron concentration comparisons to measurement.

For each parameter addressed the data base is presented, including comparisons between calculations and measurements, and conclusions are drawn regarding the suitability of the model to perform the calculations.

In addition to the calculational methods used, a description of the power distribution control philosophy adopted in Cycle 4, called Relaxed Axial Offset Control (RAOC), is described.

The licensee's analysis of accidents is provided in the BSR in order to demonstrate that Millstone 2 safety criteria are satisfied when the core is reloaded with Westinghouse fuel, and to establish a reference safety analysis for future reloads.

2.0 SUMMARY OF REVIEW (Physics)

2.1 Nuclear Design

We have reviewed the information presented with regard to calculational methods and comparisons of calculations and experiment. Most of the procedures are standard Westinghouse methods which have been used previously and verified against critical experiments and Westinghouse cores. The slight modifications in these procedures due to differences between Millstone 2 and Westinghouse cores have been adequately described in the BSR.

Many of the computer codes used are acceptable industry-wide codes and, therefore, require no additional review. These include the LEOPARD, CINDER, HAMMER, and AIM codes which form the neutron cross section generator. The TURTLE, PANDA, AND PALADON codes have previously been reviewed and approved by the staff.

We have reviewed the comparison of predicted reactivity coefficients (moderator temperature, Doppler, and boron) with measured values from the first three cycles of Millstone 2 and conclude that the Westinghouse model adequately predicts reactivity coefficients for the expected range of operating conditions and burnups.

We have reviewed the comparison of predicted control rod worths with measured values for Cycles 1, 2, and 3 and conclude that the model adequately predicts total and differential worths and trip reactivity and confirms the shutdown margin calculations.

We have reviewed the comparison of predicted power distributions with measured Cycle 1, 2, and 3 values and conclude that the method adequately predicts radial power distributions and peak-to-average distributions for beginning, middle, and end-of-cycle conditions.

Comparisons of power peaking in fuel pins adjacent to CEA water holes using TURTLE (diffusion theory) and KENO (Monte Carlo) have shown an underprediction by diffusion theory, as expected. Due to the unavailability of experimental results on water hole peaking factors, the maximum bias was confirmed by comparisons of TURTLE and INCA results for Cycle 1, 2, and 3 (Ref. 2). We find this water hole peaking correction to be acceptable.

The power distribution control philosophy to be used in Millstone 2 in Cycle 4 and beyond is Relaxed Axial Offset Control (RAOC) which is similar to the procedure used for Cycle 3 in most respects. Based on the information presented in the BSR and additional discussions with NNECO and Westinghouse, we find the RAOC procedure acceptable for providing power distribution control limits for Millstone 2 operation.

In addition, the Reload Safety Evaluation Report submitted by NNECO for Cycle 4 operation added to the data base for comparison of calculated and measured physics parameters and further verifies the nuclear design methods.

NNECO has submitted an Addendum to the BSR (Ref. 1) which describes the power peaking factor uncertainty analysis used in the nuclear design of Millstone 2 beginning with cycle 4 operation. The analysis uses measured data from the first 3 cycles and accounts for the error in the Fourier fit for the axial power shape used by INCA and includes a correction for three-dimensional effects on the power distribution. Based on this analysis, we concur that the measurement uncertainties of 6% for F_r and 7% for F_Q are adequate.

2.2 Transient Analysis

2.2.1 CEA Withdrawal

The CEA Withdrawal Event was reanalyzed from both the hot zero power condition and the full power initial condition. For the zero power case, two computer programs were used. WIT-6 was used to calculate the nuclear power (reactivity) transient and FACTRAN was then used to obtain the thermal heat flux transient and the fuel and clad temperatures. The reactor trips on the Variable High Power Trip at 25% power and the nuclear power does not overshoot the full power nominal value. The core and the RCS are not adversely affected since the combination of thermal power and the coolant temperature result in a DNBR greater than the limiting value at 1.30. For the full power case, the LOFTRAN computer program is used. The thermal margin/low pressure trip provides protection for this case and terminates the transient before the DNBR falls below 1.30. We have reviewed the initial conditions, the reactivity coefficients, and the CEA trip insertion characteristics and find the CEA withdrawal analyses and consequences acceptable.

2.2.2 CEA Drop

The CEA drop event was reanalyzed using standard W nuclear design methods to compute steady state power distributions. The peaking factors were then used in the THINC code to calculate the DNBR. LOFTRAN was used for the transient analysis. The results indicate that following the drop of the worst CEA, the reactor may return to full power without exceeding the core thermal limits. We have reviewed the assumptions used for initial system conditions as well as the reactivity feedback coefficients and dropped CEA worths used and find them to be acceptable.

2.2.3 CEA Ejection

The CEA ejection accident was reanalyzed for both full power and zero power initial conditions at BOC and EOC using the TWINKLE code in one-dimension (axial) for the average core channel calculation and the FACTRAN code for the hot fuel rod transient heat transfer calculation. The analysis performed for the more limiting HFP case predicted a maximum fuel stored energy of 172 cal/gm which is well within the Regulatory Guide 1.77 limiting criterion of 280 cal/gm. We have reviewed the analysis assumptions including the Doppler and moderator temperature coefficients, delayed neutron fractions, initial fuel temperatures, ejected rod worths, hot channel factors and trip reactivity insertion and find the analysis to be conservative and the predicted consequences acceptable.

3.0 EVALUATION PROCEDURE (Physics)

We have reviewed the report within the guidelines provided by Section 4.3, 15.4.1, 15.4.2, 15.4.3, and 15.4.8 of the Standard Review Plan. Included in our review were the description of the experimental data base, the calculations performed, and the comparisons made to support the conclusion that the NNECO reactor physics methods are adequate to calculate physics parameters and reactivity transients for Millstone 2 reload cores.

4.0 REGULATORY POSITION (Physics)

We have reviewed the revised Westinghouse reactor physics methods used by NNECO and benchmarked against Millstone 2 measurements over the first three cycles and find them acceptable to be used in Millstone 2 safety related calculations of those quantities described above.

We also find the reanalysis of the reactivity initiated transients described in the BSR adequately defines the reference safety analysis and is valid for all future cycles of Millstone 2 provided that the reload safety analysis input parameters for any given cycle are bounded by these reference analysis values. When a reload parameter is not bounded, further evaluation or a reanalysis will be necessary.

The report may be referenced in licensing submittals by NNECO for the Millstone 2 reactor.

5.0 REFERENCES (Physics)

1. NNECO submittal of a proprietary addendum to the BSR on nuclear uncertainties, W. Counsil to R. Clark, May 28, 1980.

2. NNECO answer to question 10 on power peaking in fuel pins, W. Counsil to R. Clark, July 22, 1980.