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

DEC 2 1980

MEMORANDUM FOR: T. N. Novak, Assistant Director for Operating Reactors, DOL

FROM: Paul S. Check, Assistant Director for Plant Systems, DSI

SUBJECT: CALVERT CLIFFS, UNIT 1 - CYCLE 5 FUEL LOAD REVIEW

Plant Name: Calvert Cliffs Unit 1
Docket No: 50-317
TACS NO: 12506
Responsible Branch: ORB 3
and Project Manager: Chang Y. Li
Review Status: Complete

The RSB has evaluated the transients and accidents submitted for Cycle 5 fuel load application for Calvert Cliffs Unit 1. Our evaluation (attached) concludes that the licensee's analysis and results provide an adequate safety margin for all events analyzed except the boron dilution event. The acceptance criterion for this event (SRP 15.4.6) calls for a minimum time allowance of 15 minutes after the operator has been alerted to the boron dilution event until he can act to terminate the transient. The licensee calculated for this event a time of 19.7 minutes from the initiation of the event until the shutdown margin is exhausted and the reactor returns to criticality. The licensee also stated that the operator has no positive means of being alerted to a boron dilution event that may be underway.

It is our position that a positive means of alerting the operator to a boron dilution event be installed in the control room as soon as practical. This alert should permit time intervals for operator actions in compliance with the acceptance criteria of SRP 15.4.6. In order to be able to take credit in the analysis for this positive alert, it must comply with the same requirements applicable to the reactor protection system.

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As stated

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1. Boron Dilution

An inadvertent boron dilution adds positive reactivity, produces power and temperature increases and may cause an approach to both the DNBR and the fuel centerline temperature to melt (CTM) limits.

The boron dilution event was reanalyzed to determine whether sufficient time is available for the operator to identify the cause of and terminate a boron dilution event that may occur during power operation or any other mode of operation (References 1, 4).

In a boron dilution event during power operation the Thermal Margin/Low Pressure (TM/LP) trip, or, for more rapid power excursions, the variable high power level trip will prevent violating the DNBR or the CTM criteria and indicates to the operator that a boron dilution event is in progress. In this event the operator has an adequate time to terminate the boron dilution event.

For a boron dilution event during any other mode of operation, the cold shutdown mode provides the most limiting time from the initiation of the event until the shutdown margin is exhausted and the reactor returns to critical. The following assumptions are made to determine the time to criticality:

(a) Homogeneous Dilution: homogeneous dilution results in a faster boron dilution as compared to heterogeneous dilution.

(b) RCS Active Volume: RCS volume up to the centerline of the hot leg pipes (2778 ft^3) is used for time calculation for the cold shutdown and refueling modes of operation.

(c) Dilution Rate: the maximum dilution rate results from a maximum charging pump capacity of 88 gpm (2 x 44 gpm/pump) during the cold shutdown mode (the technical specifications requires one charging pump to be inoperable during this mode) and 132 gpm (3 x 44 gpm/pump) during the refueling mode.

(d) Critical Boron Concentration: the maximum critical boron concentration of 1900 ppm and 1800 ppm is used for cold shutdown and refueling modes respectively.

(e) Initial Boron Concentration: the minimum initial boron concentration of 2065 ppm and 2300 ppm is used for cold shutdown and refueling modes respectively.

The total time to criticality during the cold shutdown mode is calculated to be 19.7 min. The SRP Section 15.4.6 specifies the minimum time allowed for operator action to be 15 minutes after he is alerted to the ongoing dilution event.

During the course of the review of this event the licensee stated that the operator has no positive means of being alerted to a boron dilution event that may be in progress. Thus the above calculated minimum time of 19.7 minutes may expire and the reactor could become critical before the operator is able to identify and terminate the event. Therefore, it is the staff's position that the operator should be provided with a positive means to alert him to a boron dilution event, and allow him adequate time to terminate the event.

2. Excess Load

The excess load event is an overcooling transient that is caused by the accidental opening of the steam dump and bypass valves. These valves have a total capacity of 45% of the rated steam flow. The excess load event has been analyzed for both full power and hot standby conditions. The excess load event at full power condition was found to be more limiting.

Conservative assumptions were used in the analysis to account for (a) the time of initiation and the capacity of the AFW system, (b) the Doppler coefficient, (c) the moderator temperature coefficient, and (d) the pressurizer pressure control system which was assumed inoperable.

The analysis resulted in a minimum DNBR of 1.36 and a maximum LHGR of 18.3 kw/ft. These values have safety margins of 14% and 13% respectively as compared to the limiting values of 1.195 and 21.0 kw/ft respectively. The analysis results for this transient meet the acceptance criteria of SRP Section 15.1.1.

3. Loss of Load

The Loss of Load (LOL) event is an undercooling transient that results from the accidental sudden closure of the turbine stop valves without a simultaneous reactor trip.

Conservative assumptions were used in the LOL transient analysis to account for (a) the steam dump and bypass valves which were assumed to remain closed, (b) the pressurizer spray and relief valves which were assumed to be closed throughout the transient, (c) a positive MTC of $0.5 \times 10^{-4}\ \Delta K/K.^{o}F$ and a least negative Doppler coefficient with a multiplier of 0.85, and (d) a bottom peaked axial power shape which minimizes the negative reactivity insertion during the initial portion of the scram following a reactor trip and maximizes the time required to mitigate the pressure and heat flux increases.

The LOL transient analysis resulted in a minimum DNBR of 1.35 and a peak reactor coolant pressure of 2550 psia. These values have safety margins of 13% and 200 psia as compared to the limiting criteria of 1.195 and 2750 psia respectively.

The analysis results for this transient meet the acceptance criteria of SRP Section 15.2.1.

4. Loss of Feedwater

The loss of feedwater (LOFW) transient is an undercooling event that is caused by the sudden and total interruption of the main feedwater flow. The safety criteria for this event are that the RCS pressure should not exceed the code safety limit, and the DNBR limit should not be violated.

The last detailed analysis for the LOFW event was conducted for Cycle 2 reload (Ref. 2). Since the key parameters used for Cycle 2 analysis (hereafter "reference cycle") did not change for Cycle 5, therefore the use of Cycle 2 as a reference cycle is justified. In the reference cycle it was shown that the minimum DNBR and the peak RCS pressure obtained during the LOFW event are less limiting than the corresponding values for the loss of load (LOL) event of that cycle. By investigation, based on the licensee's analyses for the reference cycle, we concluded that the LOFW event results had about two times the safety margin (for the minimum DNBR and the peak RCS pressure) as the LOL event results. Consequently, it can be safely assumed that the LOFW event is bounded by the LOL analysis for Cycle 5. The LOL analysis for Cycle 5 concludes that there is at least a 13% DNBR margin throughout the transient, and similarly a 200 psia peak pressure margin exists. The analysis results for this transient meet the acceptance criteria of SRP Section 15.1.1.

5. Excess Heat Removal

The loss of high pressure feedwater heaters is the most adverse feedwater malfunction in terms of Excess Heat Removal (EHR) from the RCS. This event has the same effect on the primary system as a 9% increase in turbine demand which is not matched by an increase in core power. As a result, the DNBR degradation associated with this event is less severe than that associated with the Exc[illegible] (EL) event where there is a larger increase (45%) in turbine demand analyzed. Therefore, the EHR event is bounded by the EL analysis. The EL event analysis for Cycle 5 concludes that there is about 14% DNBR margin throughout the event.

6. RCS Depressurization

The RCS depressurization event is postulated to occur by the accidental opening of the two PORVs at full power operation. Rapid depressurization while at full power causes a corresponding rapid decrease in DNBR. The Thermal Margin/Low Pressure (TM/LP) trip provides protection against violating the DNBR criterion. The last detailed analysis for this event was conducted for Cycle 4 (reference cycle for this event - Ref. 3). None of the key parameters to determine the pressure bias factor (in the TM/LP trip equation) for this event are outside the range of the reference cycle analysis.

The RCS depressurization analysis for the reference cycle produced a pressure bias of 35.0 psia as compared to 62.0 psia for the CEA withdrawal event at full power for the same cycle. Therefore, it can be considered that the CEA withdrawal analysis for Cycle 5 (which produces a pressure bias of 70.0 psia) bounds the RCS depressurization event. That is, the DNBR margin provided by a pressure bias of 70.0 psia will be more than adequate for the RCS depressurization event.

7. Loss of Non-Emergency A-C Power (LOAC)

The LOAC event involves the simultaneous interruption of steam flow to the turbine or the condenser, of the main feedwater supply to both steam generators, and of the forced reactor coolant flow. In order to meet the safety criteria, the licensee relies on the reactor trip concurrent with the LOAC, and on manual operator action. The operator ~~action~~ is required to initiate the Auxiliary feedwater flow to the steam generators within 10 minutes, and to manipulate the atmospheric dump valves in order to cool the RCS to 300°F where the RHR may begin.

The licensee referenced the Loss of Coolant Flow (LOCF) -see section 8- transient analysis to describe the DNBR degradation for (LOAC). By inspection of the reference cycle (Cycle 2 - Ref. 2) analyses we concur with the licensee's approach. Therefore, the LOCF analysis for Cycle 5 is considered representative of the LOAC transient for the same cycle.

By reference to the LOCF analysis we conclude that the DNBR criterion will be met for the LOAC event.

8. <u>Loss of Coolant Flow</u>

The loss of coolant flow (LOCF) event (4-pump coastdown) was reanalyzed for Cycle 5 to determine the minimum initial margin that must be maintained by the limiting condition for operations (LCOs) such that in conjunction with the RPS (low flow trip), the DNBR limit will not be exceeded.

The 4-pump LOCF produces a rapid approach to the DNBR limit due to the rapid decrease in the core coolant flow.

Aside from the basic assumption of 4-pump LOCF without a simultaneous reactor trip, other conservative assumptions were used in the LOCF transient analysis to reflect the following initial conditions: (a) the Technical Specification LCOs, and (b) an axial shape index (ASI) of 0.0. The choice of axial power shape with ASI of 0.0 over a positive ASI is explained as follows. If the transient is initiated from a positive ASI condition there will be a delay in terminating the DNBR decrease due to control rod insertion from the top in a bottom peaked flux shape. On the other hand, if the transient is initiated from a 0.0 ASI condition the initial steady state DNBR will be low. The analyses of both cases concluded that the 0.0 ASI condition is more conservative for the LOCF type of events (Ref.5). The analysis for this transient resulted in a minimum DNBR of 1.195 and an RCS peak pressure of 2302 psia as compared to the safety criteria of 1.195 and 2750 psia respectively.

The analysis results for this transient meet the acceptance criteria of SRP Section 15.3.1.

9. Steam Generator Tube Rupture

The steam generator tube rupture event has two effects of interest from the safety standpoint: (1) the RCS depressurization and the corresponding DNBR degradation, and (2) the radioactive release through the atmospheric steam dump and safety valves and its effect on the site boundary doses.

The RCS depressurization resulting from a double ended rupture of one steam generator tube is bounded by the analysis for the RCS depressurization resulting from the accidental opening of Two PORVs (see Section 6). The radioactive release and site boundary doses are being reviewed by the Accident Evaluation Branch.

10. Seized RCP Rotor

The seized rotor (SR) event was reanalyzed for Cycle 5 to demonstrate that the RCS upset pressure limit of 2750 psia will not be exceeded and only a small fraction of fuel pins are predicted to fail during this event.

The SR event is postulated to occur as a result of a mechanical failure. In this hypothetical event the RCS flow rapidly decreases to the three-pump value. A reactor trip is initiated by a low coolant flow rate as determined by a reduction in the sum of the steam generator hot to cold leg pressure drops. This signal is compared with a setpoint which is a function of the initial number of operating RCPs.

The analysis for this transient used an initial ASI value of -0.16. The reason for choosing this initial ASI valve is similar to the reason given in the section addressing the LOCF transient. The analysis of this transient concludes that 3% of the fuel pins are expected to fail, and that the RCS pressure will reach a peak value of 2292 psia.

The analysis results of this transient meet the acceptance criteria of SRP Section 15.3.3.

References

1. Letter, A. E. Lundvall, BG&E, to ONRR, NRC, September 22, 1980 with attachment.

2. Letter, A. E. Lundvall, BG&E, to B. C. Rusche, Director, ONRR, NRC, October 1, 1976, with attachment.

3. Letter, A. E. Lundvall, BG&E, to ONRR, NRC, February 23, 1979, with attachment.

4. Letter, A. E. Lundvall, BG&E, to ONRR, NRC, November 11, 1980.

5. Letter, A. E. Lundvall, BG&E, to ONRR, NRC, November 19, 1980.