

DESIGN FEATURES

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 50 psig and a temperature of 276°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 176 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 136.7 inches and contain a maximum total weight of 3000 grams uranium. The initial core loading shall have a maximum enrichment of 2.99 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.1 weight percent U-235.

5.3.2 Except for special test as authorized by the NRC, all fuel assemblies under control element assemblies shall be sleeved with a sleeve design previously approved by the NRC.

CONTROL ELEMENT ASSEMBLIES

5.3.3 The reactor core shall contain 77 full length and no part length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

- a. In accordance with the code requirements specified in Section 4.2 of the FSAR with allowance for normal degradation pursuant of the applicable Surveillance Requirements,
- b. For a pressure of 2500 psia, and
- c. For a temperature of 650°F, except for the pressurizer which is 700°F.

(taken from Reference (a))

TABLE 1

TABLE 7-1

CALVERT CLIFFS UNIT 2 CYCLE 9
DESIGN BASIS EVENTS CONSIDERED IN THE NON-LOCA SAFETY ANALYSIS

		<u>Analysis Status</u>
7.1	Anticipated Operational Occurrences for which intervention of the RPS is necessary to prevent exceeding acceptable limits:	
7.1.1	Boron Dilution	Not Reanalyzed
7.1.2	Startup of an Inactive Reactor Coolant Pump	(1)
7.1.3	Loss of Load	Reanalyzed ²
7.1.4	Loss of Feedwater Flow	Not Reanalyzed
7.1.5	Excess Heat Removal due to Feedwater Malfunction	Not Reanalyzed
7.1.6	Reactor Coolant System Depressurization	Not Reanalyzed
7.1.7	Excessive Charging Event	Not Reanalyzed
7.2	Anticipated Operational Occurrences for which RPS trips and/or sufficient initial steady state thermal margin, maintained by the LCOs, are necessary to prevent exceeding the acceptable limits:	
7.2.1	Sequential CEA Group Withdrawal	Not Reanalyzed
7.2.2	Loss of Coolant Flow	Not Reanalyzed
7.2.3	Full Length CEA Drop	Not Reanalyzed
7.2.4	Transients Resulting from the Malfunction of One Steam Generator	Not Reanalyzed
7.2.5	Loss of AC Power	Not Reanalyzed
7.2.6	Excess Load	Not Reanalyzed
7.3	Postulated Accidents	
7.3.1	CEA Ejection	Not Reanalyzed
7.3.2	Steam Line Rupture	Evaluated
7.3.3	Steam Generator Tube Rupture	Not Reanalyzed
7.3.4	Seized Rotor	Not Reanalyzed
7.3.5	Feed Line Break	Reanalyzed ²
7.4	Postulated Occurences	
7.4.1	Fuel Handling Incident	Reanalyzed ³

¹Technical Specifications preclude this event during operation.

²These events were reanalyzed to assess the impact of up to 500 plugged U-tubes per steam generator and credited a more realistic model for pressurizer safety valves. The results for Unit 2 Cycle 9 were less limiting than those previously reported.

³The consequences of the Fuel Handling Incident due to increasing the 1-pin burnup to 60,000 MWD/T has been investigated. The results of the investigation demonstrate that the results presented in Reference 8 are not changed by increasing the burnup.

ATTACHMENT 2

(taken from Reference (a))

7.4.1 Fuel Handling Event (Provided in response to NRC commitment, Reference 9)

The consequences of a fuel handling incident due to increasing the 1-pin burnup to 60,000 MWD/T has been investigated for Calvert Cliffs fuel. The results of this investigation demonstrate that the results shown in Reference 8 will not be changed by increasing the burnup.

The Reference 8 gas gap activity is based on the hottest fuel assembly in the core, independent of time (burnup) during the fuel cycle. Since the radioactive fission products which contribute significantly to the dose rate at the site boundary reach maximum concentrations at relatively low burnups, the only significant influence of burnup is the increased release from the fuel pellet for a given fuel temperature, commonly called "enhancement". The fuel temperature, in turn, is principally dependent on linear heat rate.

The maximum linear heat rate as a function of burnup for Calvert Cliffs fuel rods was conservatively represented in calculating the radioactive fission product release to the gas gap. The calculated releases are based on the ANS 5.4 Standard, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel."

The dose rate at the site boundary does not increase because the gas gap inventory remains less than the gas gap inventory shown in Reference 8, other than that for Kr-85. The total energy released per disintegration per second for the isotopes listed in Reference 8 was calculated and found to result in a decrease in energy per disintegration per second. The decrease in energy per disintegration per second is more than sufficient to compensate for the slightly higher Kr-85 inventory, with the net result of a decrease in the whole body dose from that shown in Reference 8. The resulting thyroid dose is also less than the dose shown in Reference 8.

(taken from Reference (a))

3.0 GENERAL DESCRIPTION

The Cycle 9 core will consist of the numbers and types of assemblies and fuel batches as described in Table 3-1. The fuel management will entail the removal of 93 irradiated assemblies: 4 Batch J, 20 Batch J*, 48 Batch H and 21 Batch H*. These assemblies will be replaced by 92 fresh assemblies: 16 unshimmed Batch L assemblies, 20 4-shimmed (B₄C) Batch LX assemblies at 4.30 wt% U-235 enrichment, 24 8-shimmed (B₄C) Batch L/ assemblies at 4.30 wt% U-235 enrichment, 28 12-shimmed (B₄C) Batch L* assemblies at 4.30 wt% U-235 enrichment, and 4 Batch LE Erbium demonstration assemblies; and by one H* assembly which was discharged at the end of Unit 2 Cycle 7.

Figure 3-1 shows the fuel management pattern to be employed in Cycle 9. Figure 3-2 shows the locations of the fuel and poison rods within the K/, K*, LX, L/ and L* shimmed assemblies, and the placement of the U-235 and Erbium bearing fuel rods in the LE assemblies. This fuel management pattern will accommodate Cycle 8 termination burnups from 17,000 MWD/T to 19,000 MWD/T.

The Cycle 9 core loading pattern is 90° rotationally symmetric. That is, if one quadrant of the core were rotated 90° into its neighboring quadrant, each assembly would be aligned with a similar assembly. This similarity includes batch type, number of fuel rods, initial enrichment and burnup.

Figure 3-3 shows the beginning of Cycle 9 assembly burnup distribution for a Cycle 8 termination burnup of 17,000 MWD/T. The initial enrichment of the fuel assemblies is also shown in Figure 3-3. Figure 3-4 shows the end of Cycle 9 assembly burnup distribution. The end of Cycle 9 core average exposure is approximately 34,900 MWD/T and the average discharge exposure is approximately 44,700 MWD/T. The end of cycle burnups are based on a Cycle 8 length of 19,000 MWD/T and a Cycle 9 length of 20,400 MWD/T.

3.1 ERBIUM DEMONSTRATION ASSEMBLIES

Four new demonstration assemblies, designated as Batch LE, will be loaded into Cycle 9 in order to determine their in-core characteristics during a 24-month cycle. These assemblies will contain Erbium as the burnable poison material. Each assembly will consist of 80 standard pins at 4.30 wt% U-235, 52 standard pins at 3.40 wt% U-235, and 44 Erbium bearing pins at 3.40 wt% U-235. This configuration results in an assembly average U-235 enrichment of 3.81 wt%. The fuel arrangement for these demonstration assemblies is shown in Figure 3-2. A further discussion concerning these assemblies is contained in Appendix A.