3. Leakage flow from the vessel inlet nozzle directly to the vessel outlet nozzle through the gap between the vessel and the barrel.

4. Flow introduced between the baffle and the barrel for the purpose of cooling these components and which is not considered available for core cooling.

5. Flow in the gaps between the fuel assemblies on the core periphery and the adjacent baffle wall.

The above contributions are evaluated to confirm that the design value of the core bypass flow is met. The design value of core bypass flow is equal to 4.5 percent of the total vessel flow.

Of the total allowance, 2.5 percent is associated with the internals (items 1, 3, 4 and 5 above) and 2.0 percent for the core. Calculations have been performed using drawing tolerances on a worst case basis and accounting for uncertainties in pressure losses. Based on these calculations, the core bypass flow is < 4.5 percent.

Flow model test results for the flow path through the reactor are discussed in Subsection 4.4.2.7.2.

4.4.4.2.2 Inlet Flow Distributions: Data has been considered from several 1/7 scale hydraulic reactor model tests, References 4.4-23, 4.4-24, and 4.4-62, in arriving at the core inlet flow maldistribution criteria used in the THINC analyses (see Subsection 4.4.4.5.1). THINC-I analyses made, using this data, have indicated that a conservative design basis is to consider 5 percent reduction in the flow to the hot assembly, Reference 4.4-63. The same design basis of 5 percent reduction to the hot assembly inlet is used in THINC IV analyses.

The experimental error estimated in the inlet velocity distribution has been considered as outlined in Reference 4.4-18 where the sensitivity of changes in inlet velocity distributions to hot channel thermal performance is shown to be small. Studies[4.4-18] made with the improved THINC model (THINC-IV) show that it is adequate to use the 5 percent reduction in inlet flow to the hot assembly for a loop out of service based on the experimental data in References 4.4-23 and 4.4-24.

The effect of the total flow rate on the inlet velocity distribution was studied in the experiments of Reference 4.4-23. As was expected, on the basis of the theoretical analysis, no significant variation could be found in inlet velocity distribution with reduced flow rate.

4.4.4.2.3 Empirical Friction Factor Correlations: Two empirical friction factor correlations are used in the THINC-IV computer code (described in Subsection 4.4.4.5.1).

The friction factor in the axial direction, parallel to the fuel rod axis, is evaluated using the Novendstern-Sandberg correlation[4.4-64]. This correlation consists of the following:

Insert Item #1 here

TABLE 15.3-2a

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENTS

(With offsite power)

	4 Loops Operating Initially
Maximum Reactor Coolant System Pressure (psia)	2589
Maximum Clad Temperature at Core Hot Spot (°F)	1675 *
Zr-H_2O reaction at Core Hot Spot (% by weight)	.169

54

* An increase of 10°F in clad average temperature is projected due to the effects of the core inlet flow maldistribution attributed to the RCS Flow Anomaly.

TABLE 15.3-2b

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENTS

(Without offsite power)

	4 Loops Operating Initially
Maximum Reactor Coolant System Pressure (psia)	2589
Maximum Clad Temperature at Core Hot Spot (°F)	1680*
Zr-H_2O reaction at Core Hot Spot (% by weight)	.188

54

* An increase of 10°F in clad average temperature is projected due to the effects of the core inlet flow maldistribution attributed to the RCS Flow Anomaly.

Trip Reactivi y Insertion

The trip reactivity insertion assumed is given in Table 15.4-3 and includes the effect of one stuck RCCA adjacent to the ejected rod. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 seconds after the high neutron flux trip point is reached. This delay is assumed to consist of 0.2 seconds for the instrument channel to produce a signal, 0.15 seconds for the trip breaker to open and 0.15 seconds for the coil to release the rods. A curve of trip rod insertion versus time was used which assumed that insertion to the dashpot does not occur until 2.8 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over one second after the trip point is reached before significant shutdown reactivity is inserted into the core. This conservatism is particularly important for hot full power accidents.

The minimum design shutdown margin available for this plant at hot zero power (HZP) may be reached only at end of life in the equilibrium cycle. This value includes an allowance for the worst stuck rod, adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Physics calculations have shown that the effect of two stuck RCCAs (one of which is the worst ejected rod) is to reduce the shutdown by about an additional one percent Δk. Therefore, following a reactor trip resulting from an RCCA ejection accident, the reactor will be subcritical when the core returns to HZP.

Reactor Protection

As discussed in Section 15.4.8.1.1, reactor protection for a rod ejection is provided by high neutron flux trip (high and low setting) and high positive neutron flux rate trip. These protection functions are part of the Reactor Trip System (RTS). No single failure of the RTS will negate the protection functions required for the rod ejection accident, or adversely affect the consequences of the accident. | 43 | 57

Results

Cases are presented for both beginning and end of life at zero and full power.

1. Beginning of Cycle, Full Power

   Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.20 percent Δk and 7.10 respectively. The peak hot spot clad average temperature was 2,219°F. The peak hot spot fuel center temperature reached melting, conservatively assumed at 4,900°F. However, melting was restricted to less than ten percent of the pellet. | 18

   Insert Item #2 here

2. Beginning of Cycle, Zero Power

   For this condition, control bank D was assumed to be fully inserted and banks B and C were at their insertion limits. The worst ejected rod is located in control bank D and has a worth of 0.86 percent Δk and a hot channel factor of 13.0. The peak hot spot clad average temperature reached 2,001°F, the fuel center temperature was 3,476°F. | 18

3. End of Cycle, Full Power

Control bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively calculated to be 0.20 percent Δk and 7.10, respectively. This resulted in a peak clad average temperature of 2,096°F. The peak hot spot fuel center temperature reached melting at 4,800°F. However, melting was restricted to less than ten percent of the pellet. |18

Insert Item #3 here

4. End of Cycle, Zero Power

The ejected rod worth and hot channel factor for this case were obtained assuming control bank D to be fully inserted and banks B and C at their insertion limits. The results were 1.0 percent Δk and 20.00, respectively. The peak clad average and fuel center temperatures were 2,422°F and 4,154°F. The Doppler weighting factor for this case is significantly higher than for the other cases due to the very large transient hot channel factor. |18

A summary of the cases presented above is given in Table 15.4-3. The nuclear power and hot spot fuel and clad temperature transients for the worst cases (beginning of life full power and end of life zero power) are presented on Figures 15.4-26 through 15.4-29.

The calculated sequence of events for the worst case rod ejection accidents, as shown on Figures 15.4-26 through 15.4-29, is presented in Table 15.4-1. For all cases, reactor trip occurs very early in the transient, after which the nuclear power excursion is terminated. The reactor will remain subcritical following reactor trip.

The ejection of a RCCA constitutes a break in the RCS, located in the reactor pressure vessel head. The effects and consequences of loss-of-coolant accidents are discussed in Section 15.6.5. Following the RCCA ejection, the operator would follow the same emergency instructions as for any other loss-of-coolant accident to recover from the event.

Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than 10 percent of the rods entered DNB based on a detailed three-dimensional THINC analysis (Ref. 15.4-10). Although limited fuel melting at the hot spot was predicted for the full power cases, it is highly unlikely that melting will occur since the analysis conservatively assumed that the hot spots before and after ejection were coincident.

Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at beginning of life, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits (Ref. 15.4-10). Since the severity of the present analysis does not exceed the "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

TABLE 15.4-3

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

	Beginning of cycle	Beginning of cycle	End of cycle	End of cycle
Power level, %	102	0	102	0
Ejected rod worth, %Δk	0.20	0.86	0.20	1.0
Delayed neutron fraction, %	0.55	0.55	0.44	0.44
Feedback reactivity weighting	1.60	2.50	1.30	4.5
Trip reactivity, %Δk	5.0	2.0	4.0	2.0
F_q before rod ejection	2.50	-	2.50	-
F_q after rod ejection	7.10	13.0	7.10	20.00
Number of operational pumps	4	2	4	2
Max. fuel pellet average temperature, °F	4090	2856	3900	3453
Max. fuel center temperature, °F	4900	3476	4800	4154
Max. clad average temperature, °F	2219*	2001	2096*	2422
Max. fuel stored energy, cal/gm	179	117	169	146

1

* An increase of 10°F in clad average temperature is projected due to the effects of the core inlet flow maldistribution attributed to the RCS Flow Anomaly.

Revisions to STP Unit 1 FSAR

Item (1) Subsequent tests conducted at STP have shown that a further reduction in the inlet flow to the hot assembly may occur when a phenomenon known as the RCS Flow Anomaly is present.

Item (2) An increase of $10^{o}F$ in clad average temperature is projected due to the effects of the core inlet flow maldistribution attributed to the RCS Flow Anomaly.

Item (3) An increase of $10^{o}F$ in clad average temperature is projected due to the effects of the core inlet flow maldistribution attributed to the RCS Flow Anomaly.

ATTACHMENT 4

HL&P CHECK NO. 319118

L4\NRC\ek