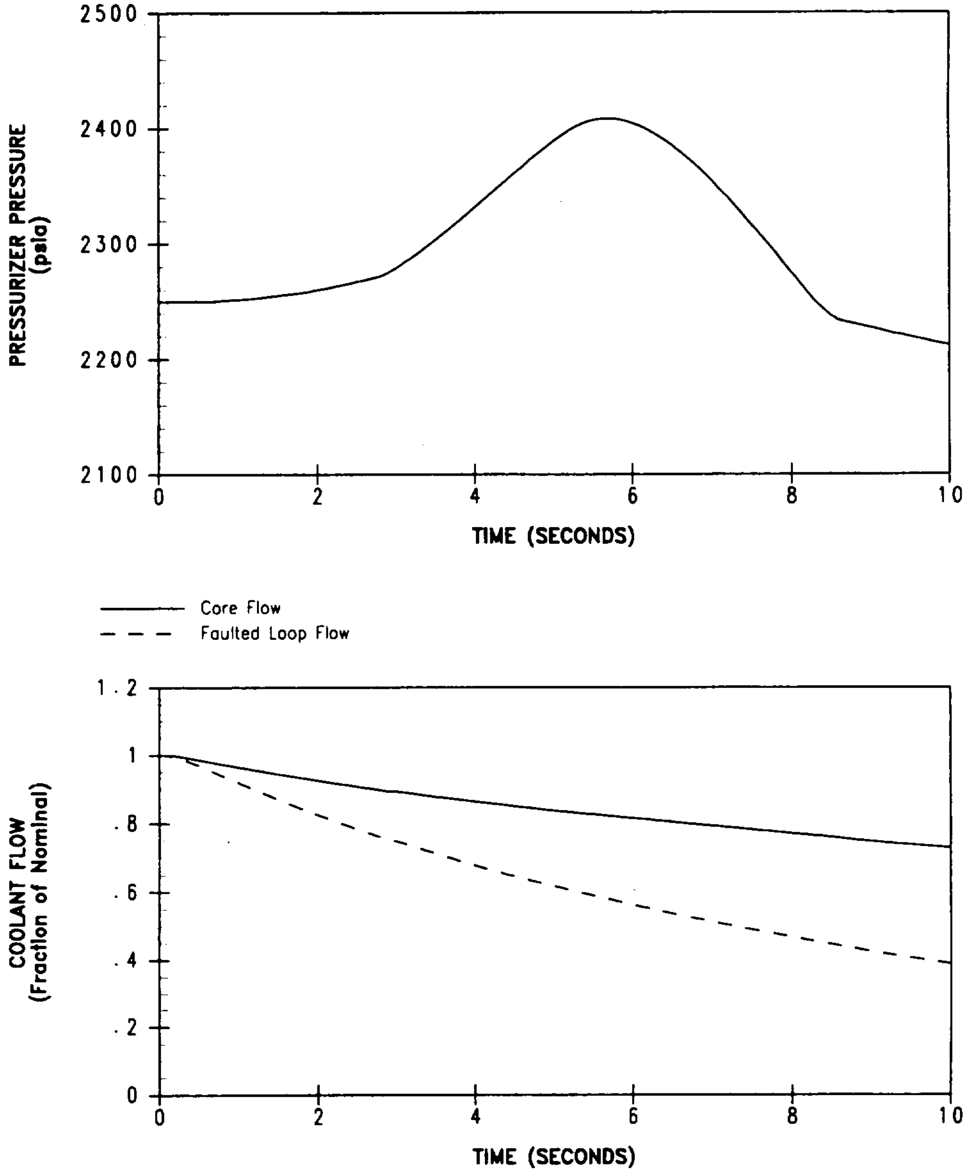


Figure 6.2.10-1

Partial Loss of Forced Reactor Coolant Flow

Pressurizer Pressure and Coolant Flow versus Time

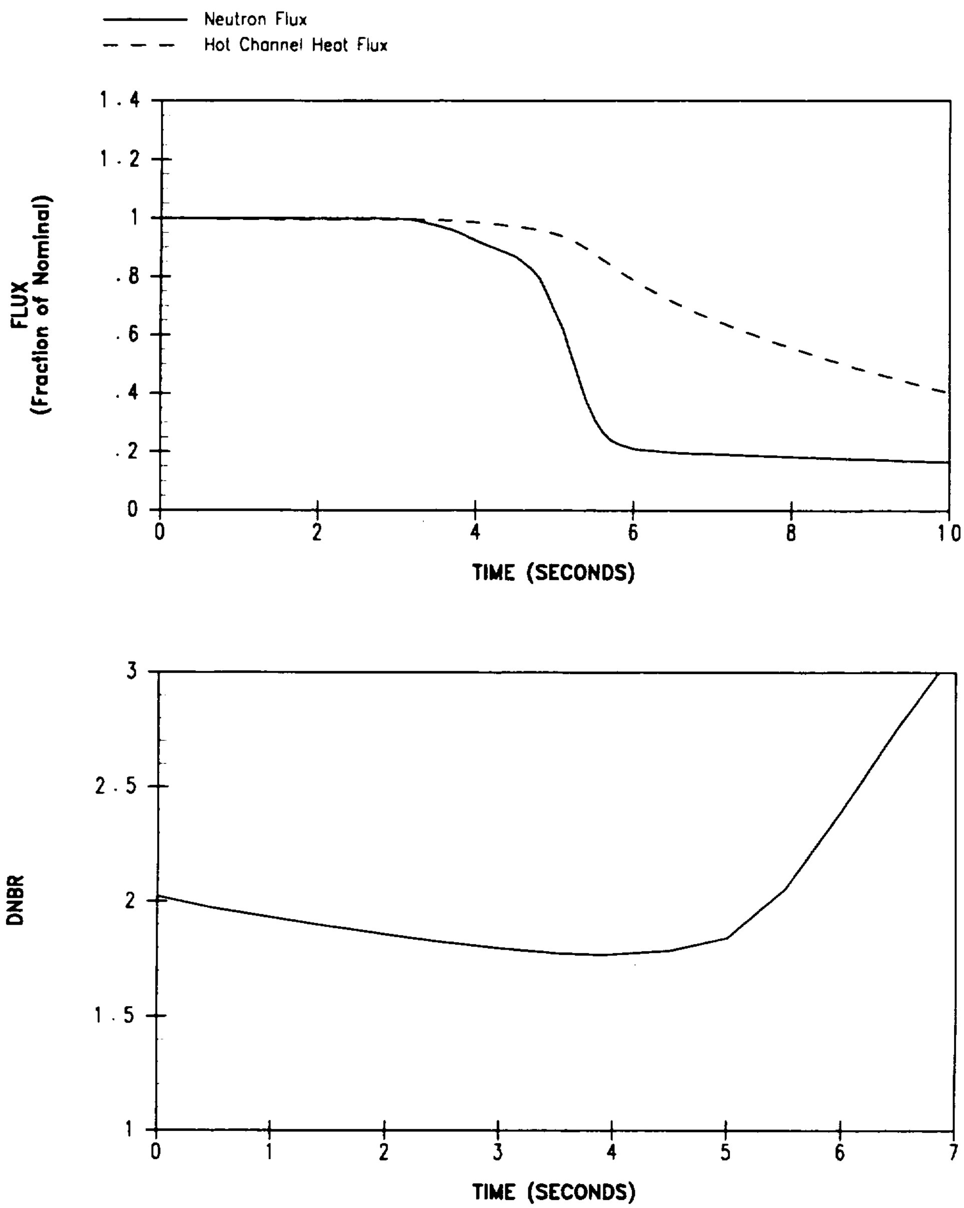


Figure 6.2.10-2

Partial Loss of Forced Reactor Coolant Flow

Heat Flux and DNBR versus Time

6.2.11 Complete Loss of Forced Reactor Coolant Flow

6.2.11.1 Introduction

A complete loss of forced reactor coolant flow accident may result from simultaneous loss of electrical power or a reduction in supply frequency to all reactor coolant pumps (RCPs). If the reactor is at power at the time of the event, the immediate effect is a rapid increase in coolant temperature. This increase in coolant temperature could result in departure from nucleate boiling (DNB), with subsequent fuel damage, if the reactor is not promptly tripped.

The following signals provide protection against a complete loss of forced reactor coolant flow incident:

- Reactor coolant pump power supply undervoltage or underfrequency;
- Low primary reactor coolant loop flow.

The reactor trip on RCP undervoltage is provided to protect against conditions which can cause a loss of voltage to all reactor coolant pumps, i.e., loss of offsite power. The reactor trip on RCP underfrequency is provided to protect against frequency disturbances on the power grid. Both the undervoltage and underfrequency trip functions are blocked below approximately 10 percent power (Permissive P-7).

The reactor trip on low primary coolant loop flow provides protection against loss of flow conditions that affect individual reactor coolant loops and serves as a backup for the undervoltage and underfrequency trip functions. The reactor trip on low primary coolant loop flow is generated by two-out-of-three low flow signals per reactor coolant loop. Above Permissive P-8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (Permissive P-7) and the power level corresponding to Permissive P-8, low flow in any two loops will actuate a reactor trip. Reactor trip on low flow is blocked below Permissive P-7.

6.2.11.2 Input Parameters and Assumptions

This accident is analyzed using the Revised Thermal Design Procedure (Reference 1). Initial core power (consistent with uprated power conditions) and reactor coolant pressure are assumed to be at their nominal values for steady-state, full-power operation. Reactor coolant temperature is assumed to be at the nominal value for the High T_{avg} Program plus a 1.5°F bias. Uncertainties in initial conditions are included in the departure from nucleate boiling ratio (DNBR) limit as described in Reference 1.

The analysis is performed assuming a moderator temperature coefficient (MTC) of 0 pcm/°F and a conservatively large (absolute value) Doppler-only power coefficient. The use of a 0 pcm/°F MTC is consistent with the analysis initial condition assumptions and corresponds to the applicable MTC limit at hot full power (HFP). The HFP analysis results using a 0 pcm/°F MTC bound those for part-power initial conditions with a positive MTC at the licensed allowable MTC limit. The negative reactivity from control rod insertion/scram is conservatively based on 4.0% Δk trip reactivity from HFP.

Normal reactor control systems and engineered safety systems (e.g., Safety Injection) are not required to function. No single active failure in any system or component required for mitigation will adversely affect the consequences of this event.

6.2.11.3 Description of Analysis

The following complete loss of forced reactor coolant flow cases were analyzed for the plant uprate.

1. Complete loss of power to all RCPs with four loops in operation
2. RCP power supply frequency decay event

Case 1 assumes that the RCPs begin to coastdown upon reaching an undervoltage trip setpoint (modeled to occur at t=0 seconds in this analysis). Rod motion following the undervoltage trip is modeled to occur at t=1.5 seconds, reflecting an undervoltage trip time delay of 1.5 seconds. For the frequency decay event (Case 2), a frequency decay rate of 5 Hz/sec is assumed to begin at t=0 seconds, decreasing pump speed, and thus flow to all loops. At t=1.2 seconds, the underfrequency trip setpoint of 54.0 Hz is reached. Rod motion occurs at t=1.8 seconds, following a 0.6 second underfrequency trip time delay.

These transients were analyzed by three digital computer codes. First, the LOFTRAN code (Reference 2) was used to calculate the loop and core flow transients, nuclear power transient, and primary system pressure and temperature transients. This code simulates a multi-loop system, neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, the steam generators, and main steam safety valves. The flow coastdown analysis performed by LOFTRAN is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, and the as-built pump characteristics and is based on conservative system pressure loss estimates.

The FACTRAN code (Reference 3) was then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code (References 4 and 5) was used to calculate the DNBR during the transient based on the heat flux from FACTRAN and the flow from LOFTRAN. The DNBR results are based on the minimum of the typical and thimble cells.

6.2.11.4 Acceptance Criteria

A complete loss of forced reactor coolant flow incident is classified by the American Nuclear Society (ANS) as a Condition III event. However, since a Condition II loss of offsite power event could lead to a Condition III complete loss of flow event, the incident is analyzed to meet the more restrictive Condition II criteria to bound the complete loss of flow following a loss of offsite power event.

The immediate effect from a complete loss of forced reactor coolant flow is a rapid increase in reactor coolant temperature and subsequent increase in RCS pressure. The primary acceptance criterion for this event is that the critical heat flux should not be exceeded. This is ensured by demonstrating that the minimum DNBR does not go below the applicable safety analysis limit at any time during the transient.

The analysis results also demonstrate that pressure in the reactor coolant and main steam systems remains below 110% of the respective design pressures to ensure that the applicable Condition II pressure criteria are met.

6.2.11.5 Results

For the Byron/Braidwood uprate, both the undervoltage and frequency decay transients were analyzed. The THINC-IV (Reference 5) analyses for these scenarios confirmed that the minimum DNBR values are greater than the safety analysis limit.

The analysis of the complete loss of flow event also demonstrates that the peak Reactor Coolant System and Main Steam System pressures are well below their respective limits.

The sequence of events for the more limiting complete loss of flow case, the frequency decay transient, is presented in Table 6.2.11-1. The transient results for this case are presented in Figures 6.2.11-1 and 6.2.11-2.

6.2.11.6 Conclusions

The analysis of the undervoltage and frequency decay cases, performed at uprated conditions, demonstrates that the DNBR does not decrease below the safety analysis limit at any time during the transients; thus, the integrity of the fuel is maintained. The peak primary and secondary system pressures remain below their respective limits at all times. Therefore, all applicable acceptance criteria are met.

6.2.11.7 References

1. Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A, April 1989.

2. Burnett, T.W.T, et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-proprietary), April 1984.

3. Hargrove, H.G., "FACTRAN -- A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

4. Hochreiter, L. E., Chelemer, H., "Application of THINC IV Program to PWR Design," WCAP-8054-P-A, February 1989.

5. Friedland, A. J., Ray, S., "Improved THINC IV Modeling for PWR Core Design," WCAP-12330-A, September 1991.

Table 6.2.11-1 Sequence of Events-Complete Loss of Forced Reactor Coolant Flow Event		
Case	**Event**	**Time (sec)**
Complete loss of forced reactor coolant flow (Frequency Decay)	Frequency decay begins	0.0
	Underfrequency trip setpoint reached	1.2
	Rods begin to drop	1.8
	Minimum DNBR occurs	3.9

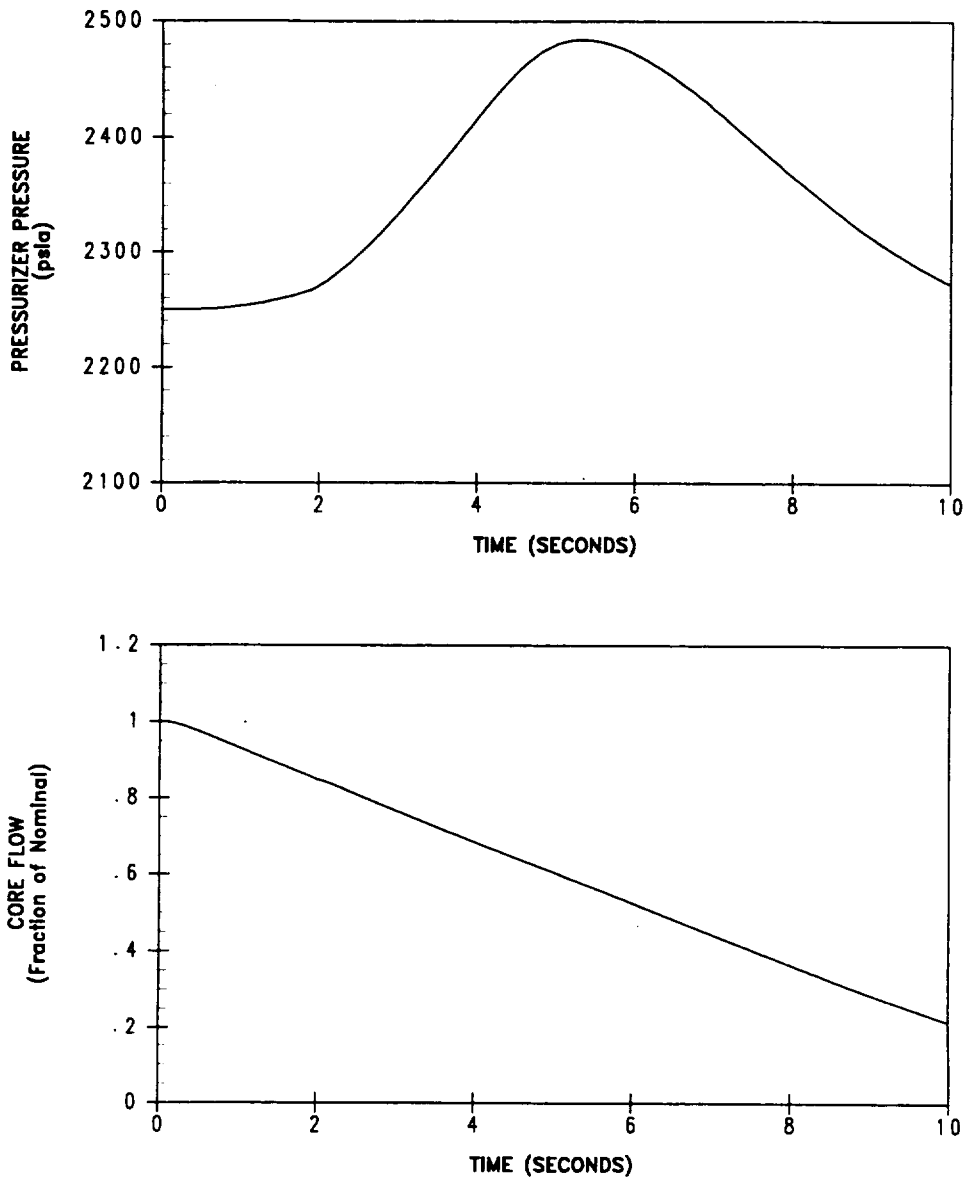


Figure 6.2.11-1

Complete Loss of Forced Reactor Coolant Flow

Pressurizer Pressure and Coolant Flow versus Time

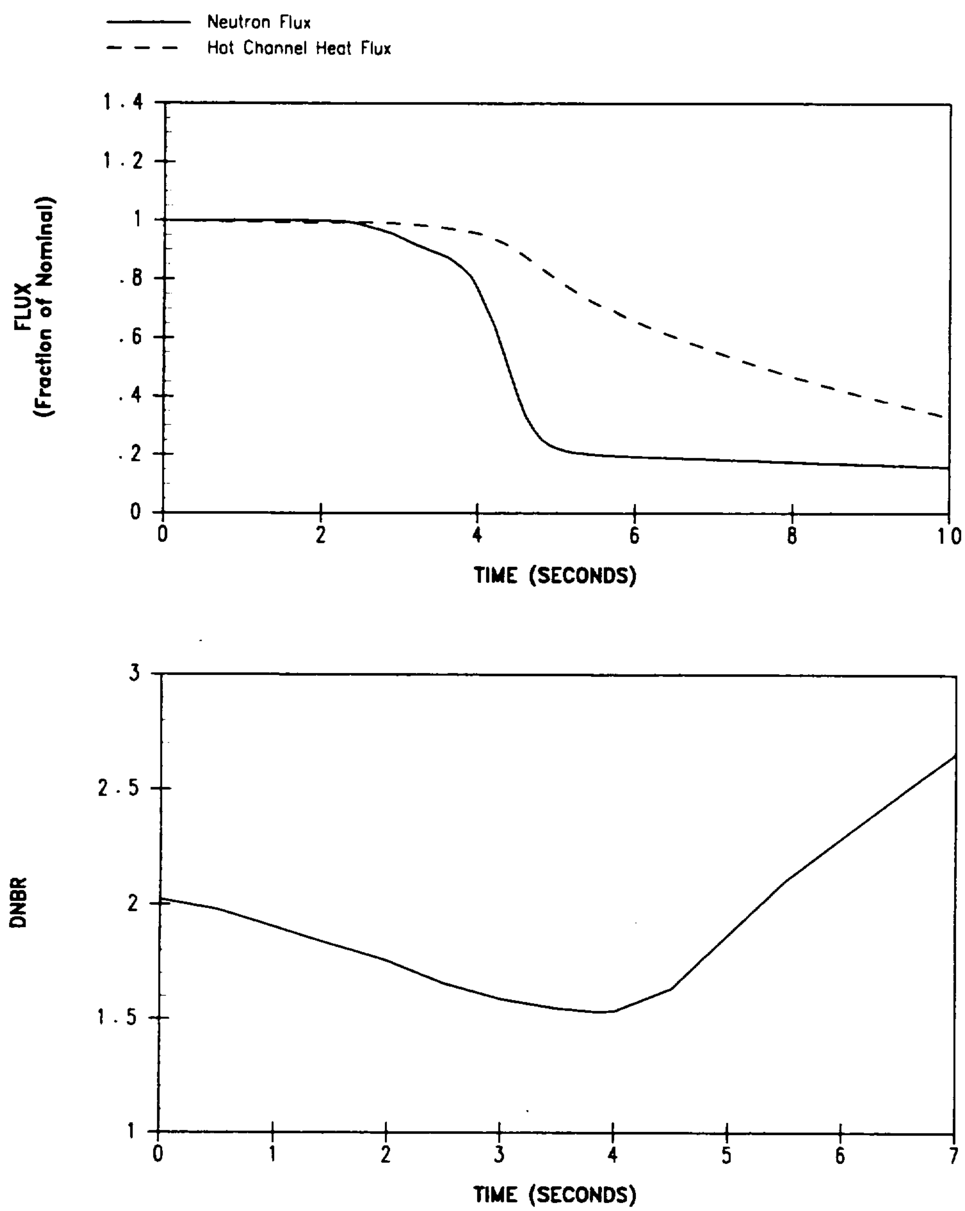


Figure 6.2.11-2

Complete Loss of Forced Reactor Coolant Flow

Heat Flux and DNBR versus Time

6.2.12 Single Reactor Coolant Pump Locked Rotor/Shaft Break

6.2.12.1 Introduction

The event postulated is an instantaneous seizure of a reactor coolant pump (RCP) rotor or the sudden break of a RCP shaft. Flow through the affected reactor coolant loop is rapidly reduced, leading to initiation of a reactor trip on a low reactor coolant loop flow signal.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell-side of the steam generators is reduced; first because the reduced primary flow results in a decreased tube-side film coefficient, and secondly because the reactor coolant in the tubes cools down while the shell-side temperature increases (turbine steam flow is reduced to zero upon plant trip due to turbine trip on reactor trip). The rapid expansion of coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the Reactor Coolant System (RCS). The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are designed for reliable operation and would be expected to function properly during the event. However, for conservatism, their pressure-reducing effect, as well as the pressure-reducing effect of the pressurizer spray, is not included in the analysis.

The consequences of a locked rotor (i.e., an instantaneous seizure of a pump shaft) are very similar to those of a pump shaft break. The initial rate of reduction in coolant flow is slightly greater for the locked rotor event. However, with a broken shaft, the impeller could conceivably be free to spin in the reverse direction. The effect of reverse spinning is to decrease the steady-state core flow when compared to the locked rotor scenario. The analysis considers the most-limiting combination of conditions for the locked rotor and pump shaft break events.

6.2.12.2 Input Parameters and Assumptions

Two cases are evaluated in the analysis. Both assume one locked RCP rotor/shaft break with a total of four loops in operation.

The first case is analyzed to evaluate the RCS pressure and fuel clad temperature transient conditions. This case is analyzed using the Standard Thermal Design Procedure. Initial core power, reactor coolant temperature, and pressure are assumed to be at their maximum values consistent with the uprated full-power conditions including allowances for calibration and instrument errors. This assumption results in a conservative calculation of fuel clad temperature transient conditions and of the coolant insurge into the pressurizer, which in turn results in a maximum calculated peak RCS pressure. In addition, a 7% loop-to-loop RCS flow asymmetry was modeled, with the faulted loop having the highest flow, thus delaying the low reactor coolant loop flow reactor trip.

The second case is an evaluation of DNB in the core during the transient. This accident is analyzed using the Revised Thermal Design Procedure (Reference 1). Initial core power (consistent with uprated power conditions) and reactor coolant pressure are assumed to be at their nominal values for steady state, full-power operation. Reactor coolant temperature is assumed to be at the nominal value for the High T_{avg} Program plus a 1.5°F bias. Uncertainties in initial conditions are included in the departure from nucleate boiling ratio (DNBR) limit as described in Reference 1. This case also considers a maximum RCS loop-to-loop flow asymmetry of 7%.

The analysis is performed assuming a moderator temperature coefficient (MTC) of 0 pcm/°F and a conservatively large (absolute value) Doppler-only power coefficient. The use of a 0 pcm/°F MTC is consistent with the analysis initial condition assumptions and corresponds to the applicable MTC limit at hot full power (HFP). The HFP analysis results using a 0 pcm/°F MTC bound those for part-power initial conditions with a positive MTC at the licensed allowable MTC limit. The negative reactivity from control rod insertion/scram is conservatively based on 4.0% Δk trip reactivity from HFP.

Normal reactor control systems and engineered safety systems (e.g., Safety Injection) are not required to function. No single active failure in any system or component required for mitigation will adversely affect the consequences of this event.

6.2.12.3 Description of Analysis

The following locked rotor/shaft break cases were analyzed in support of operation at uprated conditions.

1. Peak RCS pressure/Fuel Clad Temperatures resulting from a locked rotor/shaft break
2. Number of rods-in-DNB resulting from a locked rotor/shaft break

The first case is analyzed using two digital computer codes. The LOFTRAN code (Reference 2) is used to calculate the resulting loop and core flow transients following pump seizure, time of reactor trip based on the loop flow transients, nuclear power following reactor trip, and peak RCS pressure. The reactor coolant flow coastdown analysis performed by LOFTRAN is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, the as-built pump characteristics, and is based on conservative system pressure loss estimates. The thermal behavior of the fuel located at the core hot spot is then investigated using the FACTRAN code (Reference 3) which uses the core flow and the nuclear power values calculated by LOFTRAN. The FACTRAN code includes a film-boiling heat transfer coefficient.

The case analyzed to evaluate core DNB uses LOFTRAN, FACTRAN and the THINC code (References 4 and 5). The LOFTRAN and FACTRAN codes are used in the same manner as in the previous case. The THINC code is used to calculate the DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN.

For the peak RCS pressure evaluation, the initial pressure is conservatively estimated as 43 psi above the nominal pressure of 2250 psia to allow for errors in pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. The pressure response reported in Table 6.2.12-2 is that at the point in the RCS having the maximum pressure, i.e., the outlet of the RCP.

For a conservative analysis of fuel rod behavior, the hot spot evaluation assumes that DNB occurs at initiation of the transient and continues throughout the event. This assumption reduces heat transfer to the coolant and results in conservatively high hot spot temperatures.

Evaluation of the Pressure Transient

After pump seizure, coolant flow in the loop with the faulted RCP decreases rapidly and RCS temperature and pressure increases. A reactor trip signal is generated when the flow in the affected loop reaches 85.1 percent of nominal flow. Rod motion begins one second later and the neutron flux is rapidly reduced by control rod insertion. As RCS pressure increases, no credit is taken for the pressure-reducing effect of pressurizer power-operated relief valves or pressurizer spray, nor is credit taken for steam dump or controlled feedwater flow after plant trip. Although these systems are expected to function and would result in a lower peak pressure, an additional degree of conservatism is provided by not including their effect.

Upon actuation of the pressurizer safety valves at 2549.4 psia (including 1% allowance for drift and 1% for pressure shift), purge of the water in the safety valve loop seal occurs and full valve relief capacity is achieved within 1 second.

Evaluation of DNB in the Core During the Event

For this event, DNB is assumed to occur in the core; therefore, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this "hot spot" condition represent the upper limit with respect to clad temperature and zirconium-water reaction. In the evaluation, the rod power at the hot spot conservatively considers an F_Q of 2.60. The number of rods-in-DNB is conservatively calculated for use in dose consequence evaluations.

Other Locked Rotor/Shaft Break analysis specific method and modeling assumptions are described in detail in Section 15.3.3 of the Byron/Braidwood UFSAR (Reference 6).

6.2.12.4 Acceptance Criteria

The RCP locked rotor accident is classified by the American Nuclear Society (ANS) as a Condition IV event. The following items summarize the criteria associated with this event.

- Fuel cladding damage, including melting, due to increased reactor coolant temperatures must be prevented. This is precluded by demonstrating that the maximum clad temperature at the core hot spot remains below 2700°F, and the zirconium-water reaction at the core hot spot is less than 16 weight percent.

- Pressure in the reactor coolant system should be maintained below that which would cause stresses to exceed the faulted condition stress limits.

- Rods-in-DNB should be less than or equal to that assumed in the radiological dose analyses for the Locked Rotor/Shaft Break event.

6.2.12.5 Results

The calculated sequence of events is presented in Table 6.2.12-2 for the Locked Rotor event. With respect to peak RCS pressure, peak clad temperature, zirconium-steam reaction, and maximum predicted rods-in-DNB, the analysis demonstrates that there is no significant impact on the transient results due to operation at uprated conditions. The results of the calculations (peak pressure, peak clad temperature, and zirconium-steam reaction) are summarized in Table 6.2.12-1. The number of rods-in-DNB is less than that supported by the radiological dose analysis. Hence, the rods-in-DNB criterion has also been met for the Locked Rotor/Shaft Break event. The transient results for the peak pressure/hot spot case are provided in Figures 6.2.12-1 and 6.2.12-2.

6.2.12.6 Conclusions

The analysis performed at uprated conditions demonstrates that, for the Locked Rotor/Shaft Break event, the peak clad temperature calculated for the hot spot during the worst transient remains considerably less than 2700°F and the amount of zirconium-water reaction is small. Under such conditions, the core will remain in place and intact with no loss of core cooling capability.

The analysis also confirms that the peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. The rods-in-DNB design criterion is also met.

The protection features described in Section 6.2.12.1 provide mitigation for a locked rotor/shaft break transient such that the above criteria are satisfied.

6.2.12.7 References

1. Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A, April 1989. (Proprietary).

2. Burnett, T.W.T, et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-proprietary), April 1984.

3. Hargrove, H.G., "FACTRAN -- A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

4. Hochreiter, L. E., Chelemer, H., "Application of THINC IV Program to PWR Design," WCAP-8054-P-A, February 1989.

5. Friedland, A. J., Ray, S., "Improved THINC IV Modeling for PWR Core Design," WCAP-12330-A, September 1991.

6. "Byron & Braidwood Station, Updated Final Safety Analysis Report," Revision 7, Docket Nos. STN-454/455/456/457, as amended through December 1998.

Table 6.2.12-1 Summary of Results for the Locked Rotor/Shaft Break Transient		
Criteria	**Analysis Value**	**Limit**
Maximum Clad Temperature at Core Hot Spot, °F	1954	2700
Maximum $Zr\text{-}H_2O$ Reaction at Core Hot Spot, wt. %	0.54	16
Maximum RCS Pressure, psia	2736	Faulted Condition Stress Limits

Table 6.2.12-2 Sequence of Events-Locked Rotor/Shaft Break Transient	
Event	**Time (sec)**
Rotor on one pump locks/shaft breaks	0.0
Low flow reactor trip setpoint reached	0.04
Rods begin to drop	1.04
Maximum RCS pressure occurs	3.6
Maximum clad temperature occurs	3.7

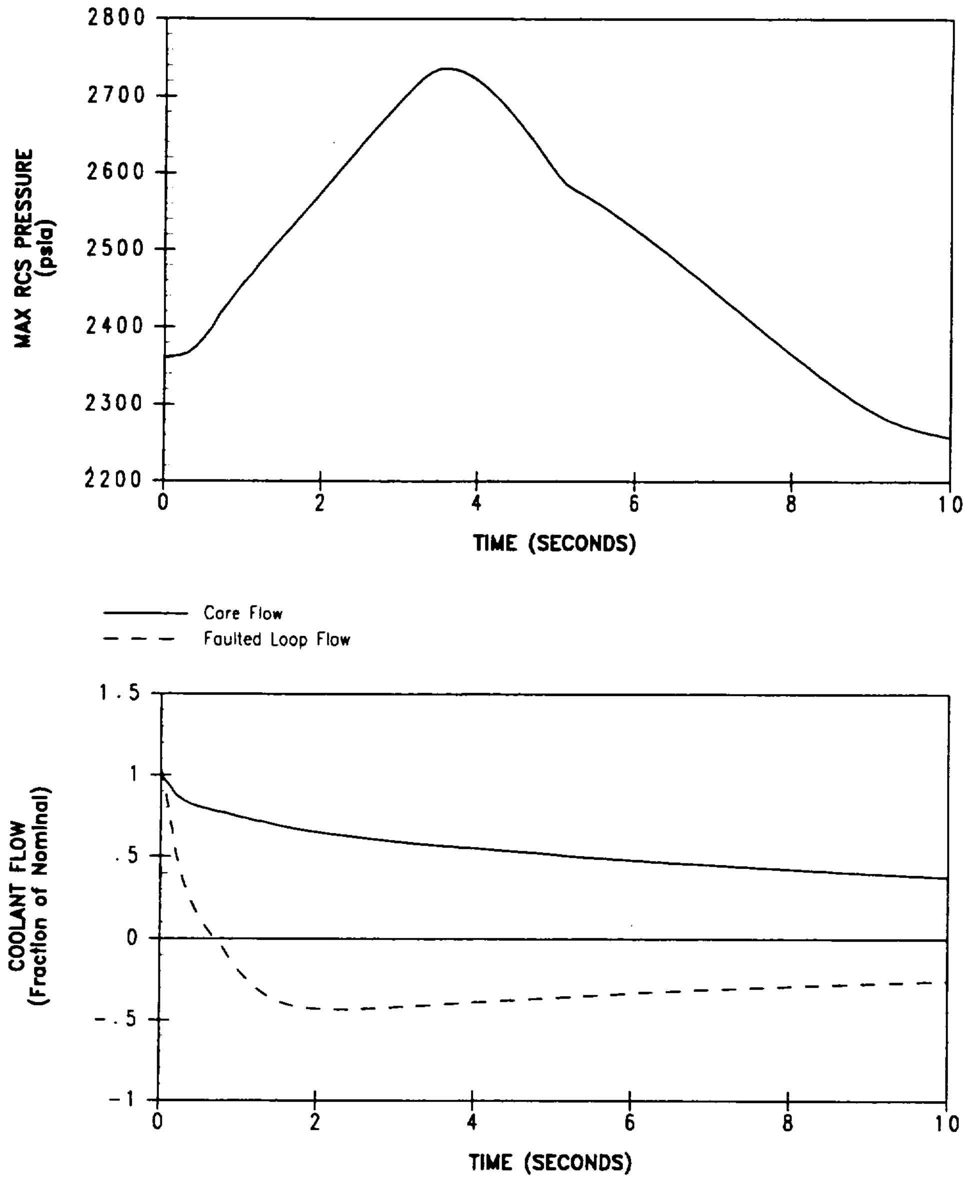


Figure 6.2.12-1

Single Reactor Coolant Pump Locked Rotor/Shaft Break

Maximum RCS Pressure and Coolant Flow versus Time

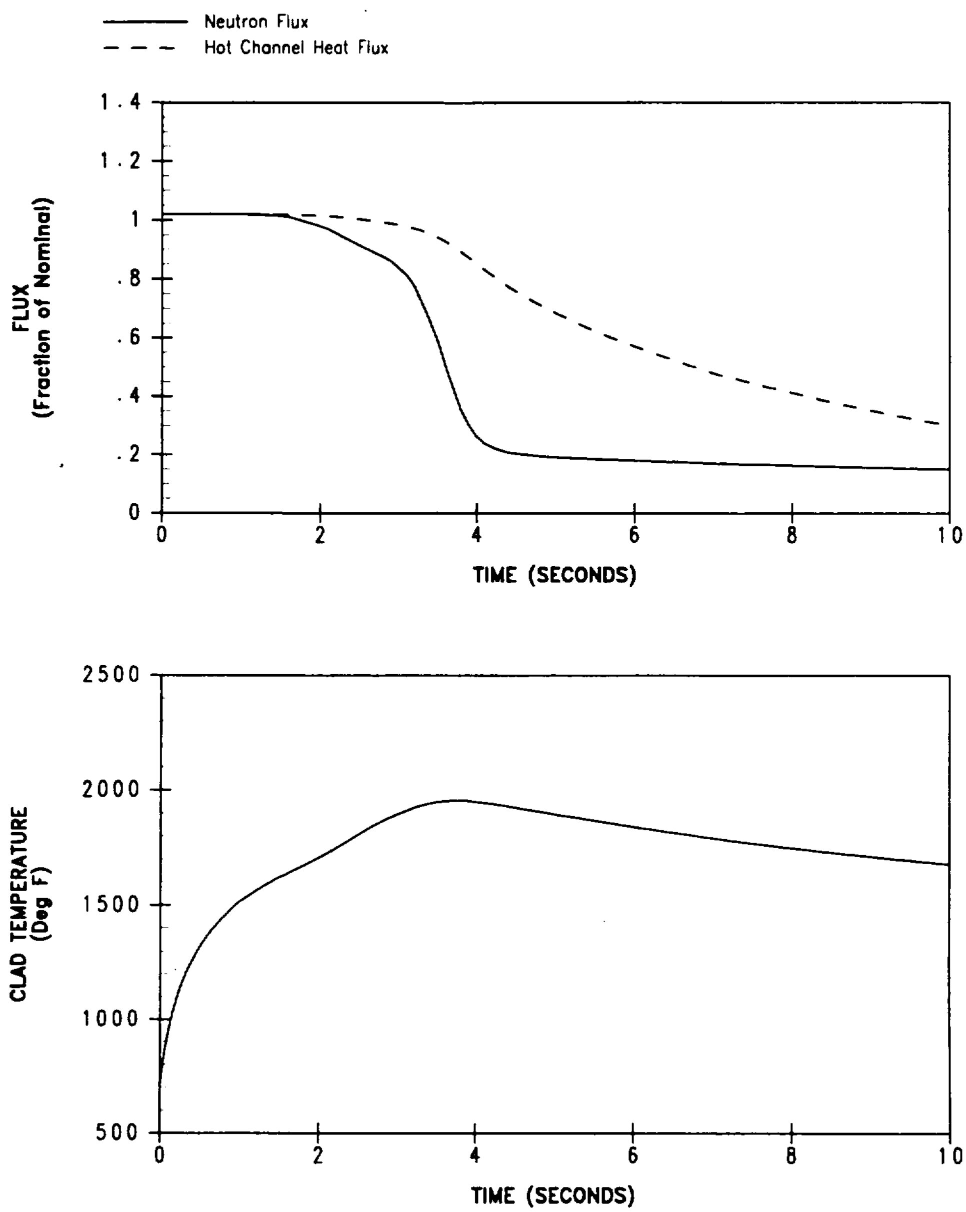


Figure 6.2.12-2

Single Reactor Coolant Pump Locked Rotor/Shaft Break

Heat Flux and Clad Temperature versus Time

6.2.13 Uncontrolled RCCA Withdrawal From a Subcritical Condition

6.2.13.1 Introduction

A Rod Cluster Control Assembly (RCCA) withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of control rods resulting in a power excursion. While the probability of a transient of this type is extremely low, operator action or a malfunction of the reactor control rod drive system could cause such a transient. This could occur with the reactor either subcritical or at power. The at-power case is discussed in Section 6.2.14.

Reactivity is added at a prescribed and controlled rate in bringing the reactor from shutdown to low power during startup by RCCA withdrawal or by reducing the reactor coolant boron concentration. RCCA motion can cause much faster changes in reactivity than can result from changing the boron concentration.

The RCCA drive mechanisms are wired into pre-selected bank configurations that remain the same throughout reactor life. These circuits prevent the RCCAs from being automatically withdrawn in other than their respective banks. Power supplied to the banks is controlled such that no more than two banks can be withdrawn at the same time and in their defined withdrawal sequence. The RCCA drive mechanisms are of the magnetic latch type and coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is analyzed in the detailed plant analysis assuming simultaneous withdrawal of the combination of the two sequential control banks having the maximum combined worth at maximum speed. The maximum reactivity insertion rate, even with these assumptions, is well within the capability of the reactor protection system to prevent core damage.

Should a continuous RCCA withdrawal be initiated, the following automatic features of the reactor protection system will terminate the transient:

a. Source range neutron flux reactor trip - actuated when either of two independent source range channels indicate above a pre-selected, manually adjustable setpoint. This trip function may be manually bypassed only after either of two intermediate range flux channel indicate above a specified level. It is automatically reinstated when both intermediate channels indicate below a specified level.

b. Intermediate range neutron flux reactor trip - actuated when either of two independent intermediate range channels indicate above a pre-selected, manually adjustable setpoint. This trip function may be manually bypassed only after two of four power range channels indicate above approximately 10 percent full-power. It is automatically reinstated when three of four channels indicate below this value.

c. Power range high neutron flux reactor trip (low setting) - actuated when two of four power range channels indicate above approximately 25 percent full-power. This trip function may be manually bypassed when two of four power range channels indicate above approximately 10 percent full-power. It is automatically reinstated when three of four channels indicate below this value.

d. Power range high neutron flux reactor trip (high setting) - actuated when two of four power range channels indicate above a preset setpoint. This trip function is always active.

e. High neutron flux rate reactor trip - actuated when the positive rate of change in neutron flux on two of four power range channels exceeds the preset setpoint. This trip function is always active.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast initial increase terminated by the reactivity feedback effect of the negative Doppler power coefficient. This self-limitation of the initial power increase is of prime importance since it limits nuclear power to an acceptable level prior to protection system action. After the initial increase, the nuclear power is momentarily reduced and then, if the incident is not terminated by a reactor trip, the nuclear power increases again, but at a much slower rate.

Termination of the startup transient by the above protection channels prevents core damage. In addition, control rod stops on high intermediate range flux level (one of two) and high power range flux level (one of four) serve to halt rod withdrawal and prevent the need to actuate the intermediate range flux level trip and power range flux level trip, respectively.

6.2.13.2 Input Parameters and Assumptions

The accident analysis employs the Standard Thermal Design Procedure (STDP). To obtain conservative results for the analysis, the following assumptions are made concerning initial reactor conditions:

a. Since the magnitude of the nuclear power peak reached during the initial part of the transient, for any given rate of reactivity insertion, is strongly dependent on the Doppler power reactivity coefficient, a conservatively low (least negative) value is used. The Doppler reactivity defect is determined as a function of power level using a one-dimensional steady-state computer code with a Doppler-weighting factor of 1.0. The Doppler defect assumed is 965 pcm.

b. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because heat transfer time between the fuel and moderator is much longer than nuclear flux response time. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. Accordingly, the most-positive moderator temperature coefficient is assumed since this yields the maximum rate of power increase.

c. The analysis assumes the reactor to be at hot zero power conditions with a nominal temperature of 557°F. This assumption is more conservative than that of a lower initial system temperature (i.e., shutdown conditions) because it yields a larger fuel-to-moderator heat transfer coefficient, a larger specific heat of both moderator and fuel, and a less-negative (smaller absolute magnitude) Doppler coefficient. The less-negative Doppler coefficient reduces the Doppler feedback effect, thereby increasing the neutron flux peak. The high neutron flux peak combined with a high fuel specific heat and larger heat transfer coefficient yields a larger peak heat flux. The analysis also assumes the initial effective multiplication factor (K_{eff}) to be 1.0 since this results in the maximum neutron flux peak.

d. Reactor trip is assumed on power range high neutron flux (low setting). The most adverse combination of instrumentation error, setpoint error, delay for trip signal actuation, and delay for control rod assembly release is taken into account. The analysis assumes a 10 percent uncertainty in power range flux trip setpoint (low setting), raising it from the nominal value of 25 percent to 35 percent. During the transient, the rise in nuclear power is so rapid that the effect of error in the trip setpoint on the actual time of rod release is negligible. In addition, total reactor trip reactivity is based on the assumption that the highest worth control rod assembly is stuck in its fully withdrawn position.

e. The maximum positive reactivity insertion rate assumed is greater than that for simultaneous withdrawal of the two sequential control banks having the greatest combined worth at maximum speed (45 in/min, which corresponds to 72 steps/min).

f. The DNB analysis assumes the most limiting axial and radial power shapes associated with having the two highest combined worth banks in their high worth position.

g. The analysis assumes initial power to be below that expected for any shutdown condition (10^{-9} fraction of nominal power). The combination of highest reactivity insertion rate and low initial power produces the highest peak heat flux.

h. The analysis assumes two RCPs in operation (Mode 3 Technical Specification allowed operation). This is conservative with respect to the DNB transient.

6.2.13.3 Description of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcriticality is performed in three stages. First, a spatial neutron kinetics computer code, TWINKLE (Reference 1), is used to calculate the core average nuclear power transient, including various core feedback effects, i.e., Doppler and moderator reactivity. Next, the FACTRAN computer code (Reference 2) uses the average nuclear power calculated by TWINKLE and performs a fuel rod transient heat transfer calculation to determine average heat flux and temperature transients. Finally, the average heat flux calculated by FACTRAN is used in the THINC-IV computer code (References 3 & 4) for transient DNBR calculations.

6.2.13.4 Acceptance Criteria

The uncontrolled rod cluster control assembly bank withdrawal from subcritical event is considered an ANS Condition II event, a fault of moderate frequency, and is analyzed to ensure that the core and reactor coolant system are not adversely affected. This is demonstrated by showing that the minimum DNBR remains above the applicable safety analysis limit and that peak hot spot fuel and clad temperatures remain within acceptable limits.

6.2.13.5 Results

The results of the uncontrolled RCCA bank withdrawal analysis performed at the uprated conditions show that the minimum DNBR remains above the safety analysis limit at all times and that peak fuel centerline temperature remains below that at which fuel melt occurs. The calculated sequence of events is shown in Table 6.2.13-1. The neutron flux transient, thermal flux transient and the fuel and clad temperature transients for this accident are shown in Figures 6.2.13-1 through 6.2.13-4, respectively.

6.2.13.6 Conclusions

In the event of an RCCA withdrawal incident from the subcritical condition, the core and RCS are not adversely affected since the combination of thermal power and coolant temperature results in a minimum DNBR greater than the safety analysis limit. Furthermore, since the maximum fuel temperatures predicted to occur during this event are much less than those required for fuel melting (4800°F), no fuel damage is predicted as a result of this transient at uprated conditions. Clad damage is also precluded.

6.2.13.7 References

1. Barry, R. F., Jr. and Risher, D. H., "TWINKLE, a Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A, January 1975 (Proprietary) and WCAP-8028-A, January 1975 (Non-proprietary).

2. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

3. Hochreiter, L. E., Chelemer, H., "Application of THINC IV Program to PWR Design," WCAP-8054-P-A, February 1989.

4. Frieland, A. J., Ray, S., "Improved THINC IV Modeling for PWR Core Design," WCAP-12330-A, September 1991.

Table 6.2.13-1

Sequence of Events-Uncontrolled Rod Withdrawal from Subcritical Event

Event	Time (Sec)
Power range high neutron flux low setpoint reached	11.1
Peak nuclear power occurs	11.3
Rod begins to fall into core	11.6
Peak heat flux occurs	13.6
Minimum DNBR occurs	13.6
Peak average clad temperature occurs	13.9
Peak average fuel temperature occurs	14.0

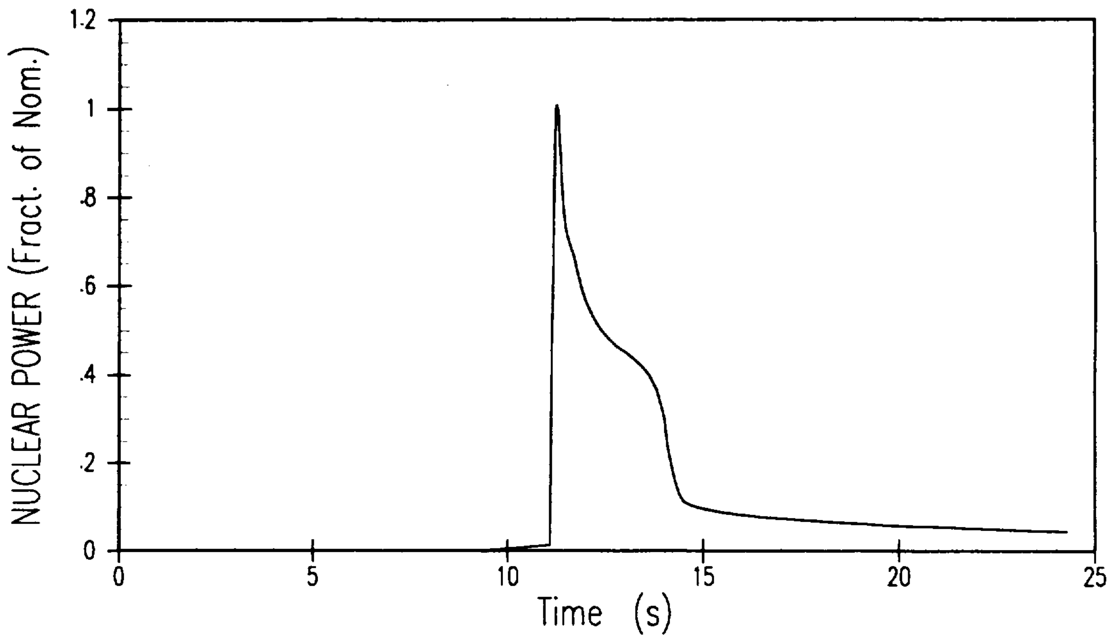


Figure 6.2.13-1

Neutron Flux Transient for Uncontrolled Rod Withdrawal from a Subcritical Condition

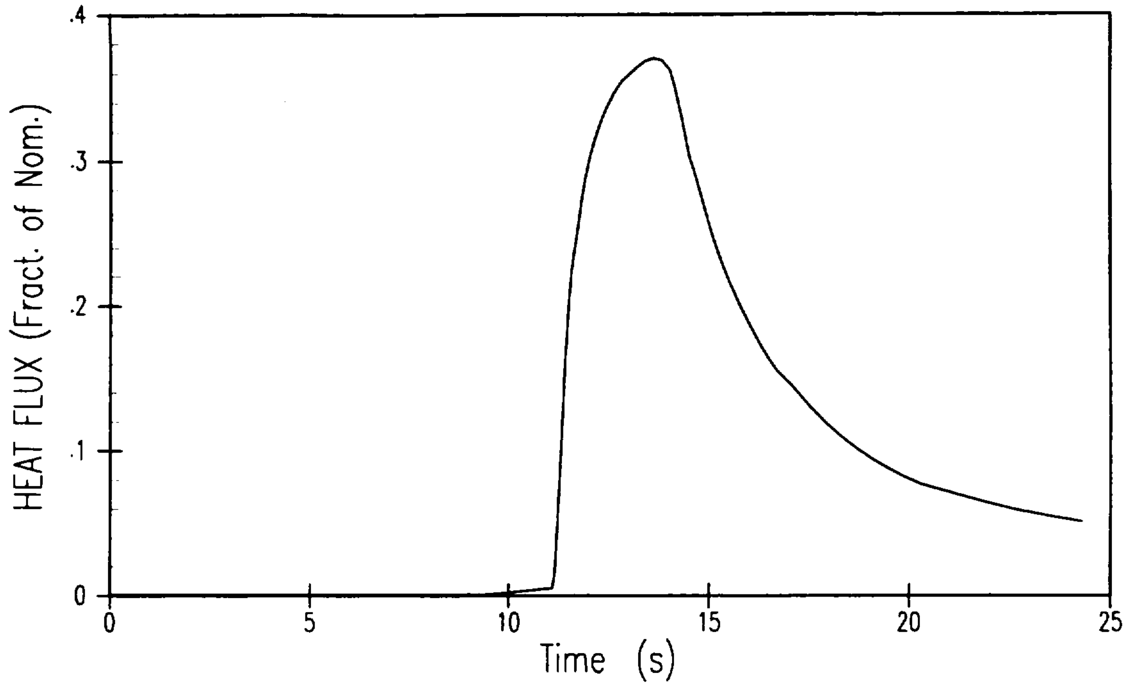


Figure 6.2.13-2

Thermal Flux Transient for Uncontrolled Rod Withdrawal from a Subcritical Condition

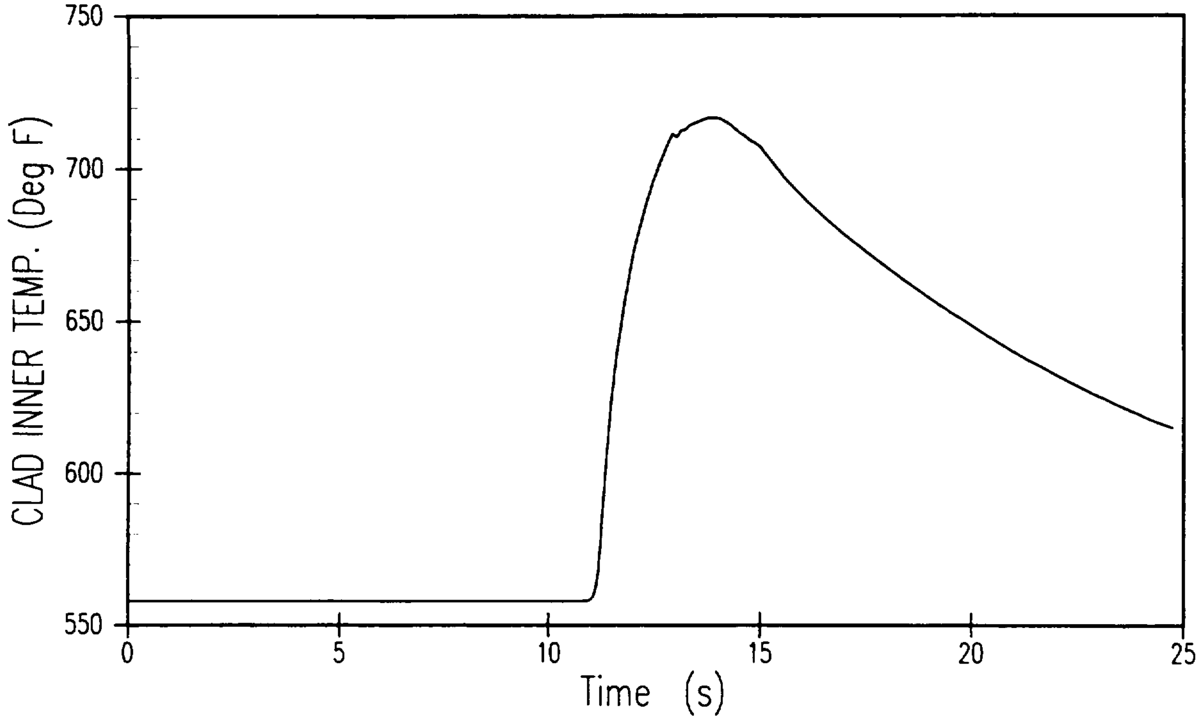


Figure 6.2.13-3

Clad Inner Temperature Transient for Uncontrolled Rod Withdrawal from a Subcritical Condition

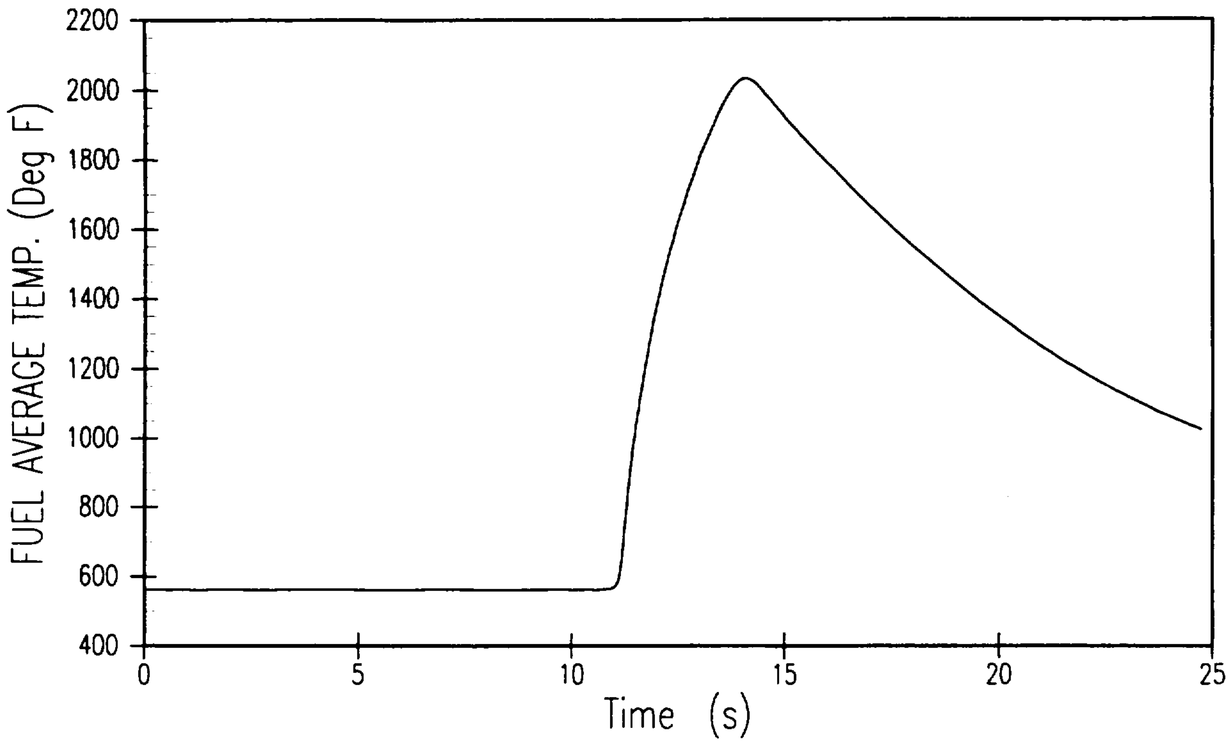


Figure 6.2.13-4

Fuel Average Temperature Transient for Uncontrolled Rod Withdrawal from a Subcritical Condition

6.2.14 Uncontrolled RCCA Bank Withdrawal At Power

6.2.14.1 Introduction

An uncontrolled Rod Cluster Control Assembly (RCCA) bank withdrawal at power that causes an increase in core heat flux may result from faulty operator action or a malfunction in the rod control system. Immediately following initiation of the accident, the steam generator (SG) heat removal rate lags behind the core power generation rate until SG pressure reaches the setpoint of the SG relief or safety valves. This imbalance between heat removal and heat generation rate causes the reactor coolant temperature to rise. Unless terminated, the power mismatch and resultant coolant temperature rise could eventually result in departure from nucleate boiling (DNB) and/or fuel centerline melt. Therefore, to avoid damage to the core, the reactor protection system (RPS) is designed to automatically terminate any such transient before the departure from nucleate boiling ratio (DNBR) falls below the safety analysis limit or the fuel rod linear heat generation rate (kW/ft) limit is exceeded.

The automatic RPS features that prevent core damage in a RCCA bank withdrawal incident at power by actuating a reactor trip include the following.

a. Any two-out-of-four power range high neutron flux channels exceed an overpower (OP) setpoint.

b. Any two-out-of-four positive neutron flux rate channels exceed a high rate setpoint.

c. Any two-out-of-four ΔT channels exceed an overtemperature (OT) ΔT setpoint. This setpoint is automatically varied with axial power distribution, coolant average temperature, and coolant average pressure to protect against DNB.

d. Any two-out-of-four ΔT channels exceed an overpower (OP) ΔT setpoint. This setpoint is automatically varied with coolant average temperature so that the allowable heat generation rate (kW/ft) is not exceeded.

e. Any two-out-of-four high pressurizer pressure channels exceed a fixed setpoint. This setpoint is less than the set pressure for the pressurizer safety valves.

f. Any two-out-of-three high pressurizer water level channels exceed a fixed setpoint.

Besides the above listed reactor trips, there are several RCCA bank withdrawal blocks that are not credited in the accident analyses but would serve to limit the severity of this event. These are:

a. High neutron flux (one-out-of-four power range channels)

b. Overpower ΔT (two-out-of-four channels)

c. Overtemperature ΔT (two-out-of-four channels)

6.2.14.2 Input Parameters and Assumptions

A number of cases were analyzed assuming a range of reactivity insertion rates for both minimum and maximum reactivity feedback conditions at various power levels. The cases presented in Section 6.2.14.5 are representative for this event.

Since Byron/Braidwood Units 1 and 2 operate with different steam generator models, this accident is analyzed with input modeling of both steam generator designs for the limiting cases with minimum DNBR, maximum primary pressure, and maximum secondary pressure, to ensure that conservative results are calculated for each criterion.

For an uncontrolled RCCA bank withdrawal at power accident, the following conservative assumptions are made:

a. This accident is analyzed with the Revised Thermal Design Procedure (Reference 1). Therefore, initial reactor power, pressurizer pressure, and reactor coolant system (RCS) temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the DNBR limit.

b. For reactivity coefficients, two cases are analyzed.

   1. Minimum Reactivity Feedback: A +7 pcm/°F moderator temperature coefficient and a least-negative Doppler-only power coefficient form the basis for the beginning-of-life (BOL) minimum reactivity feedback assumption.

   2. Maximum Reactivity Feedback: A conservatively-large positive moderator density coefficient of 0.54 k/g/cm^3 (corresponding to a large negative moderator temperature

coefficient) and a most-negative Doppler-only power coefficient form the basis for the end-of-life (EOL) maximum reactivity feedback assumption.

c. The reactor trip on high neutron flux is actuated at a conservative value of 118% nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors, with maximum delay for trip signal actuation.

d. The RCCA trip insertion characteristic is based on the assumption that the highest-worth rod cluster control assembly is stuck in its fully withdrawn position.

e. A range of reactivity insertion rates is examined. The maximum positive reactivity insertion rate is greater than that which would be obtained from the simultaneous withdrawal of the two control rod banks having the maximum combined worth at a speed of 45 inches/minute or 72 steps/minute.

f. Initial power levels of 10%, 60% and 100% are considered.

g. The impact of a full-power RCS T_{AVG} window was considered for the uncontrolled RCCA bank withdrawal at power analysis. A conservative calculation modeling the high end of the T_{AVG} window was explicitly analyzed since this is limiting with respect to the DNBR results.

6.2.14.3 Description of Analysis

The purpose of this analysis is to demonstrate how the protection functions described previously actuate for various combinations of reactivity insertion rates and initial conditions. Insertion rate and initial conditions determine which trip function actuates first.

The rod withdrawal at power event is analyzed with the LOFTRAN computer code (Reference 2). The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and main steam safety valves. The program computes pertinent plant variables including temperatures, pressures, power level, and DNBR.

6.2.14.4 Acceptance Criteria

Based on its frequency of occurrence, the uncontrolled RCCA bank withdrawal at power accident is considered a Condition II event as defined by the American Nuclear Society (ANS). The following items summarize the main acceptance criteria associated with this event.

The critical heat flux should not be exceeded. This is ensured by demonstrating that the minimum DNBR does not go below the safety analysis limit value at any time during the transient.

Pressure in the reactor coolant and main steam systems should be maintained below 110% of design pressures. With respect to peak pressure, the uncontrolled RCCA bank withdrawal at power accident is bounded by the loss of load/turbine trip analysis described in Section 6.2.6.

The protection features presented in Section 6.2.14.1 provide mitigation of the uncontrolled RCCA bank withdrawal at power transient such that the above criteria are satisfied.

6.2.14.5 Results

Figures 6.2.14-1 through 6.2.14-6 show the transient response for a rapid uncontrolled RCCA bank withdrawal incident (50 pcm/s) starting from 100% power with minimum feedback. Reactor trip on high neutron flux occurs shortly after the start of the accident. Because of the rapid reactor trip with respect to the thermal time constants of the plant, the small changes in T_{AVG} and pressure result in the margin to DNB being maintained.

The transient response for a slow uncontrolled RCCA bank withdrawal (0.3 pcm/s) from 100% power with minimum feedback is shown in Figures 6.2.14-7 through 6.2.14-12. Reactor trip on OTΔT occurs after a much longer period, and the temperature rise is consequently larger, than for a rapid RCCA bank withdrawal. Again, the minimum DNBR is greater than the safety analysis limit.

Figure 6.2.14-13 shows the minimum DNBR as a function of reactivity insertion rate from 100% power for both minimum and maximum reactivity feedback conditions. The high neutron flux and OTΔT reactor trip functions provide DNB protection over the range of reactivity insertion rates. The minimum DNBR is never less than the safety analysis limit.

Figures 6.2.14-14 and 6.2.14-15 show the minimum DNBR as a function of reactivity insertion rate for RCCA bank withdrawal incidents starting at 60% and 10% power, respectively. The results are similar to the 100% power case. However, as the initial power level decreases, the range over which the OTΔT trip is effective increases. In no case does the DNBR fall below the safety analysis limit.

The resultant minimum DNBRs when modeling the Westinghouse D5 steam generator (Units 2) are less than those calculated when modeling the BWI steam generator (Units 1).

The calculated sequence of events for the two cases illustrated in the figures is shown in Table 6.2.14-1. With the reactor tripped, the plant eventually returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

RCS pressures less than the limit are obtained for reactivity insertion rates less than 50 pcm/second.

6.2.14.6 Conclusions

The high neutron flux and OTΔT reactor trip functions provide adequate protection over the entire range of possible reactivity insertion rates, i.e., the minimum value of DNBR is always larger than the safety analysis limit. The reactor coolant and main steam systems are maintained below 110% of their design pressures. Therefore, the results of the analysis show that an uncontrolled RCCA withdrawal at power does not adversely affect the core, RCS, or main steam system, and all applicable acceptance criteria are met.

6.2.14.7 References

1. Friedland, A.J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary), WCAP-11397-A (Non-proprietary), April 1989.

2. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary),WCAP-7907-A (Non-proprietary), April 1984.

Table 6.2.14-1 Sequence of Events-Uncontrolled RCCA Bank Withdrawal at Power Analysis		
Case	**Event**	**Time(s)**
100% Power, Minimum Feedback, Rapid RCCA Withdrawal (50 pcm/sec)	Initiation of Uncontrolled RCCA Withdrawal	0.0
	Power Range High Neutron Flux - High Setpoint Reached	2.3
	Rods Begin to Fall	2.8
	Minimum DNBR Occurs	3.8
100% Power Minimum Feedback, Slow RCCA Withdrawal (0.3 pcm/sec)	Initiation of Uncontrolled RCCA Withdrawal	0.0
	OTΔT Setpoint Reached	81.7
	Rods Begin to Fall	89.7
	Minimum DNBR Occurs	90.1

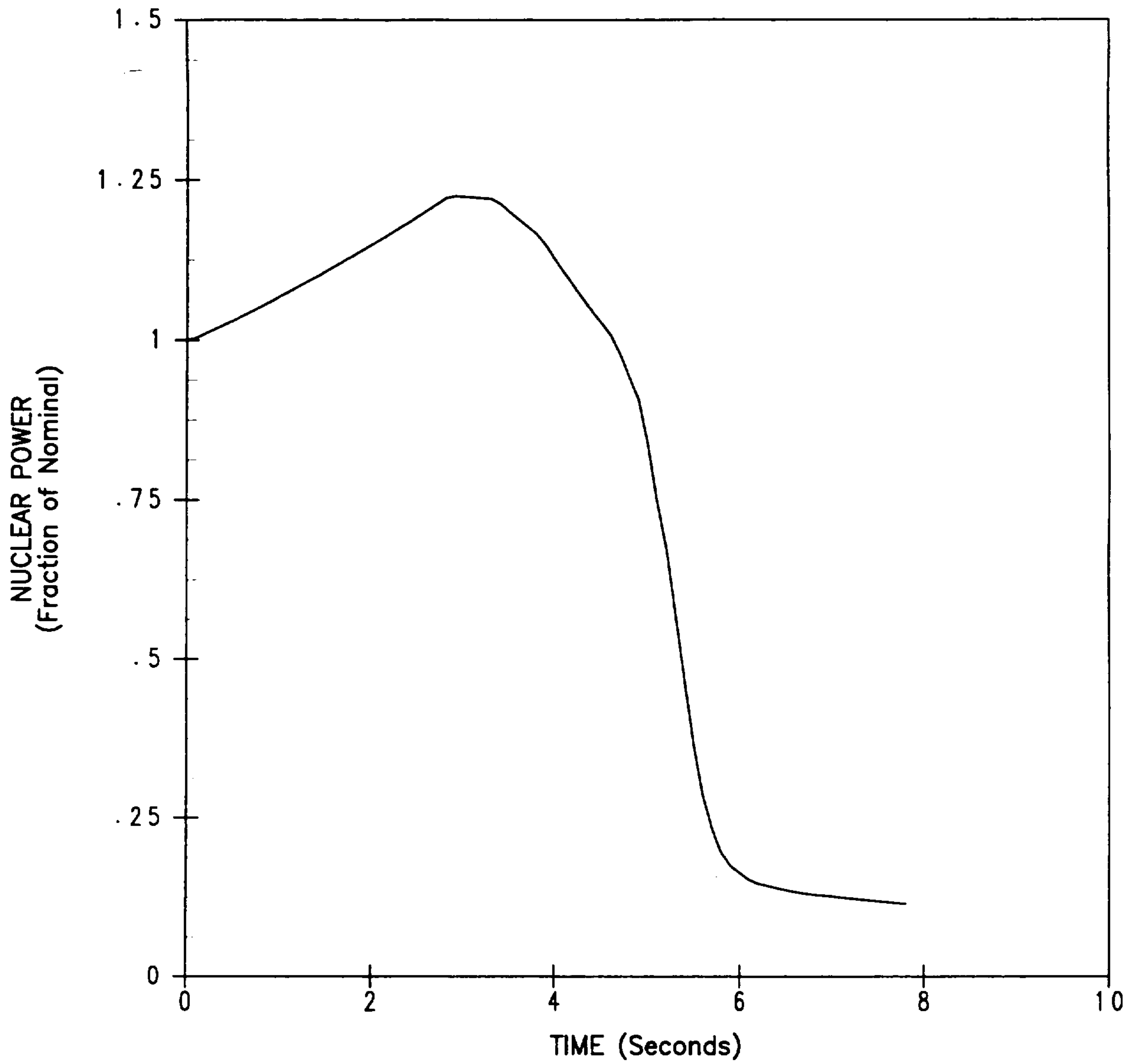


Figure 6.2.14-1

Uncontrolled RCCA Bank Withdrawal at Power, Withdrawal Rate of 50 pcm/second,100% Power, Minimum Reactivity Feedback

Nuclear Power vs. Time

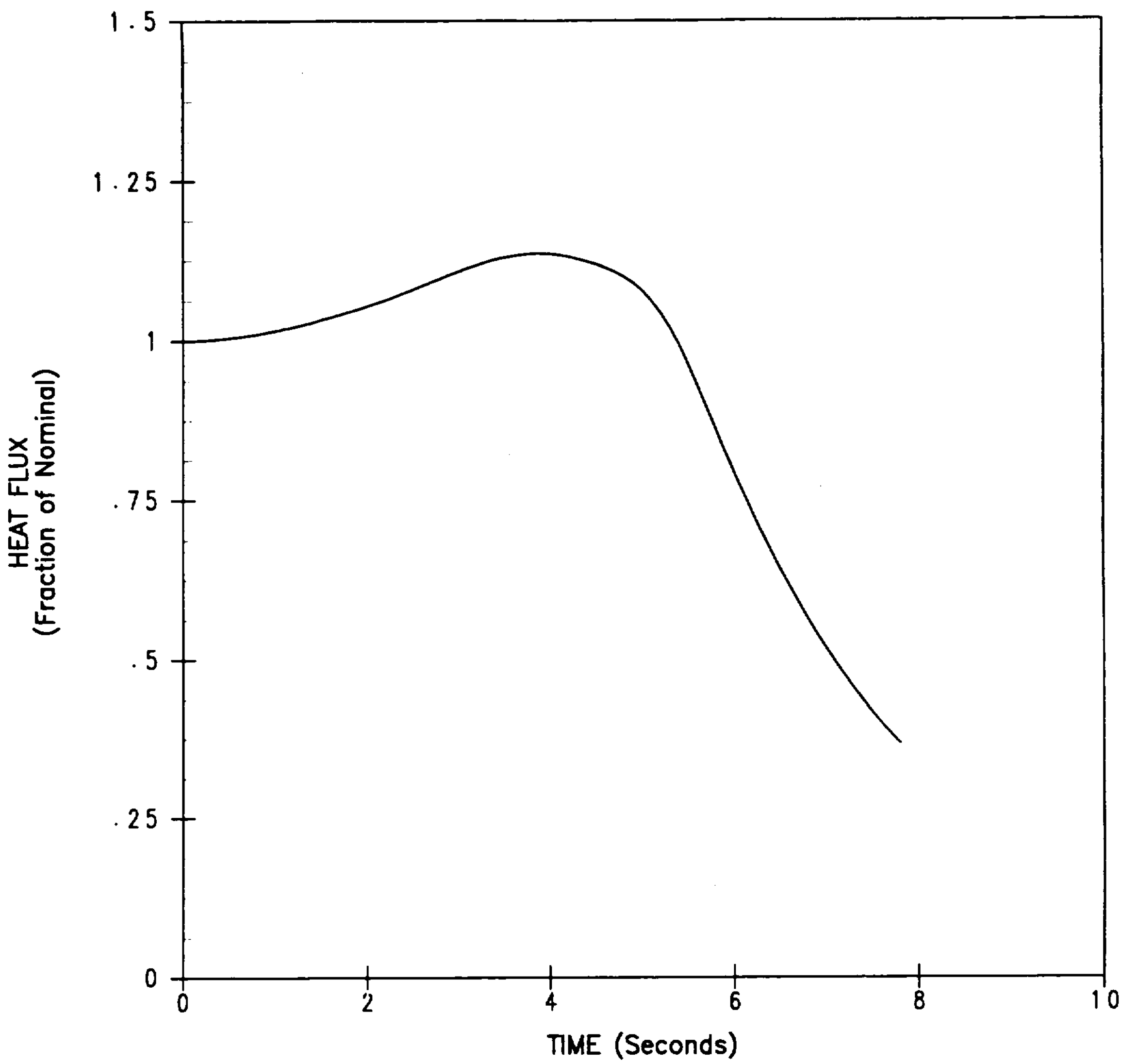


Figure 6.2.14-2

Uncontrolled RCCA Bank Withdrawal at Power, Withdrawal Rate of 50 pcm/second, 100% Power, Minimum Reactivity Feedback Core Heat Flux vs. Time

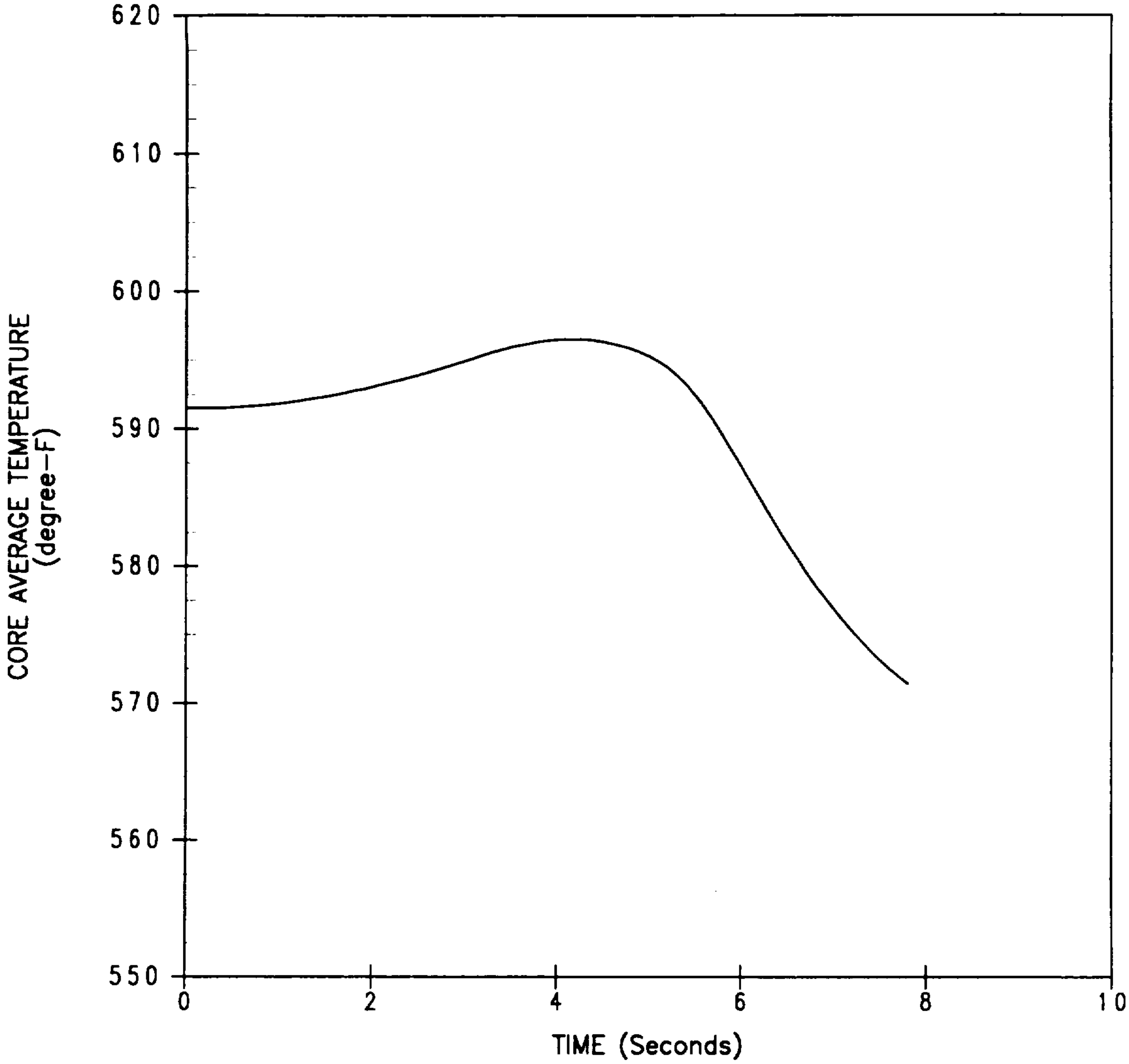


Figure 6.2.14-3

Uncontrolled RCCA Bank Withdrawal at Power, Withdrawal Rate of 50 pcm/second, 100% Power, Minimum Reactivity Feedback Core Average Temperature vs. Time

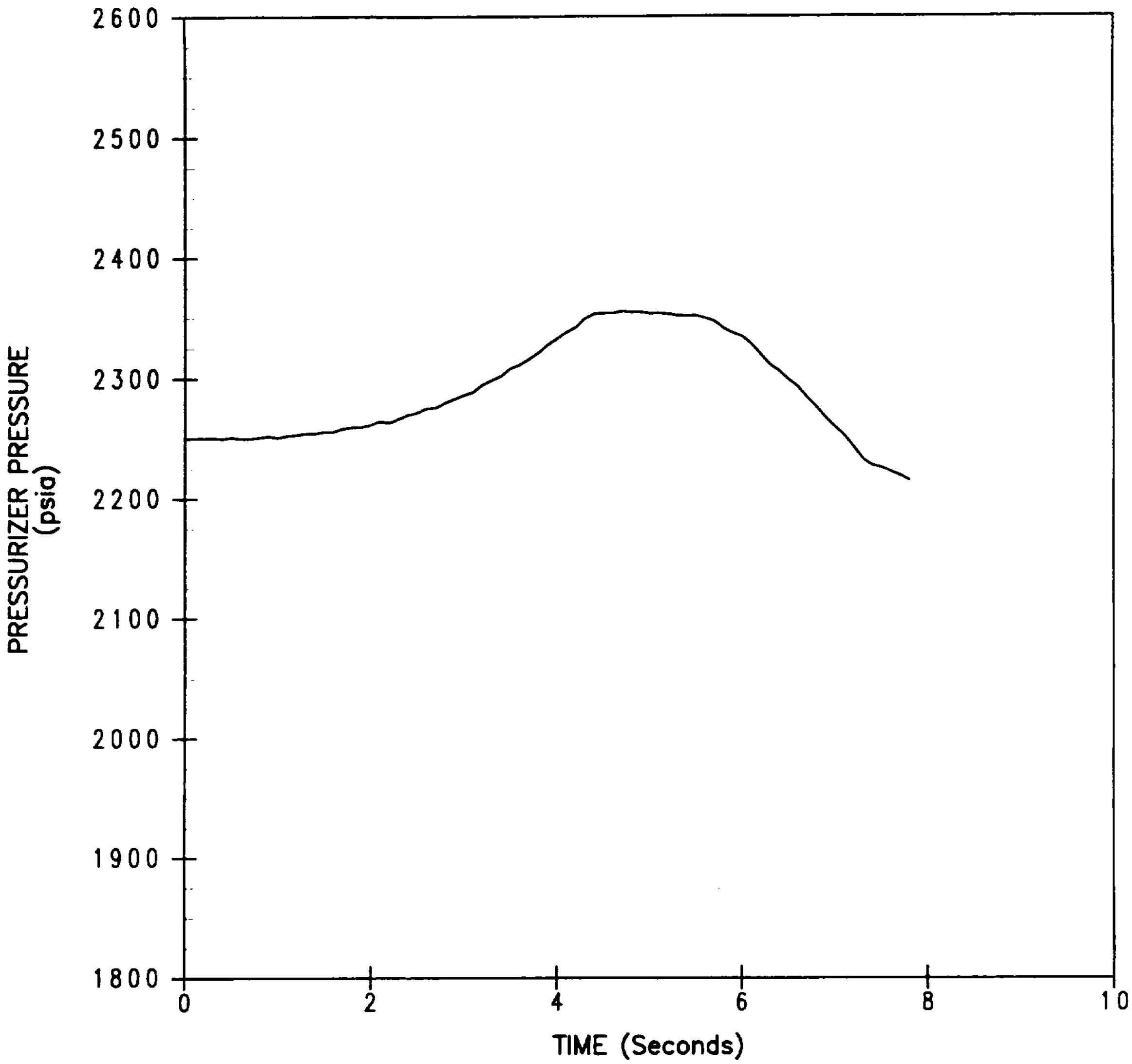


Figure 6.2.14-4

Uncontrolled RCCA Bank Withdrawal at Power, Withdrawal Rate of 50 pcm/second, 100% Power, Minimum Reactivity Feedback Pressurizer Pressure vs. Time

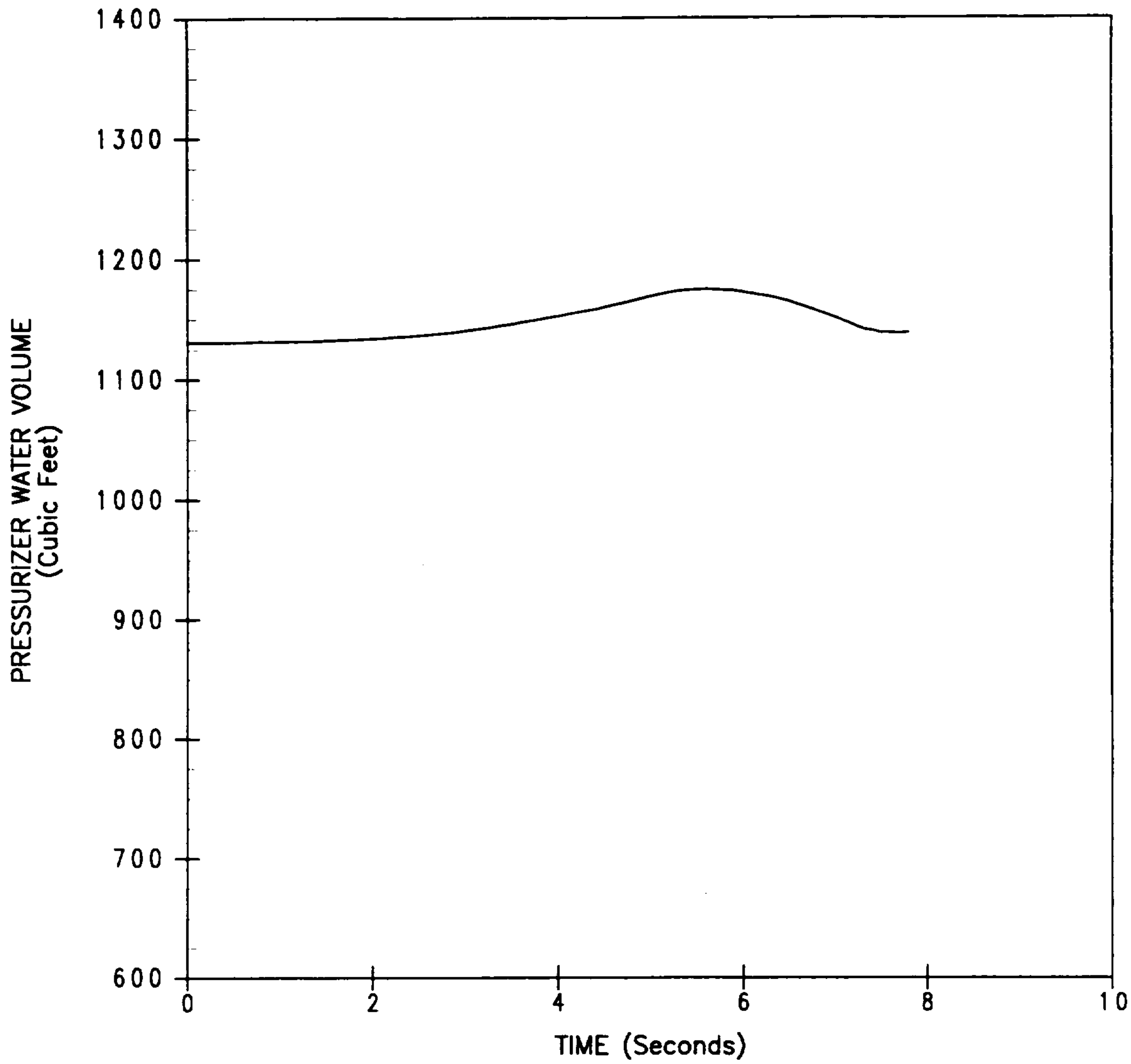


Figure 6.2.14-5

Uncontrolled RCCA Bank Withdrawal at Power, Withdrawal Rate of 50 pcm/second, 100% Power, Minimum Reactivity Feedback Pressurizer Water Volume vs. Time

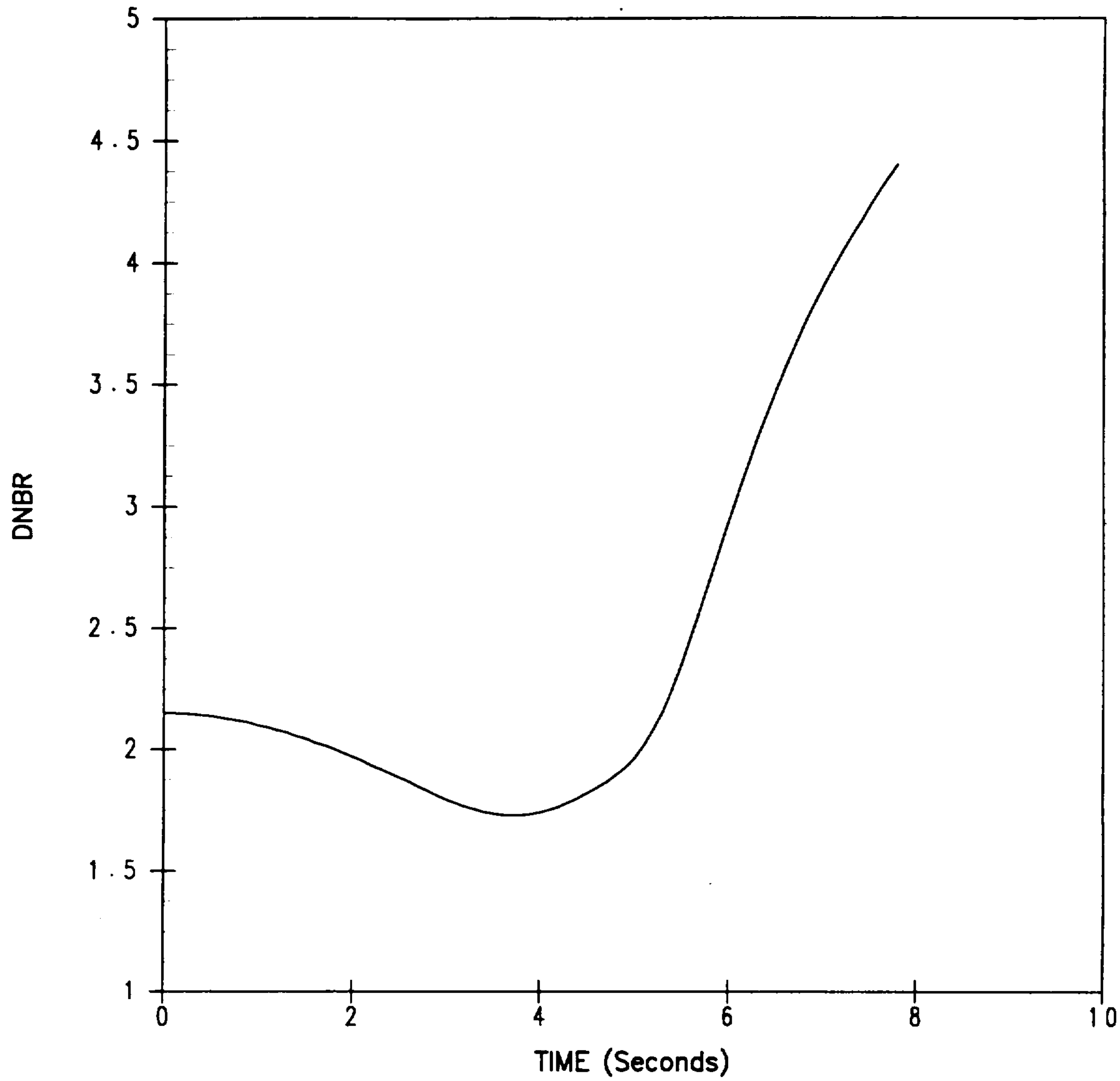


Figure 6.2.14-6

Uncontrolled RCCA Bank Withdrawal at Power, Withdrawal Rate of 50 pcm/second, 100% Power, Minimum Reactivity Feedback

DNBR vs. Time

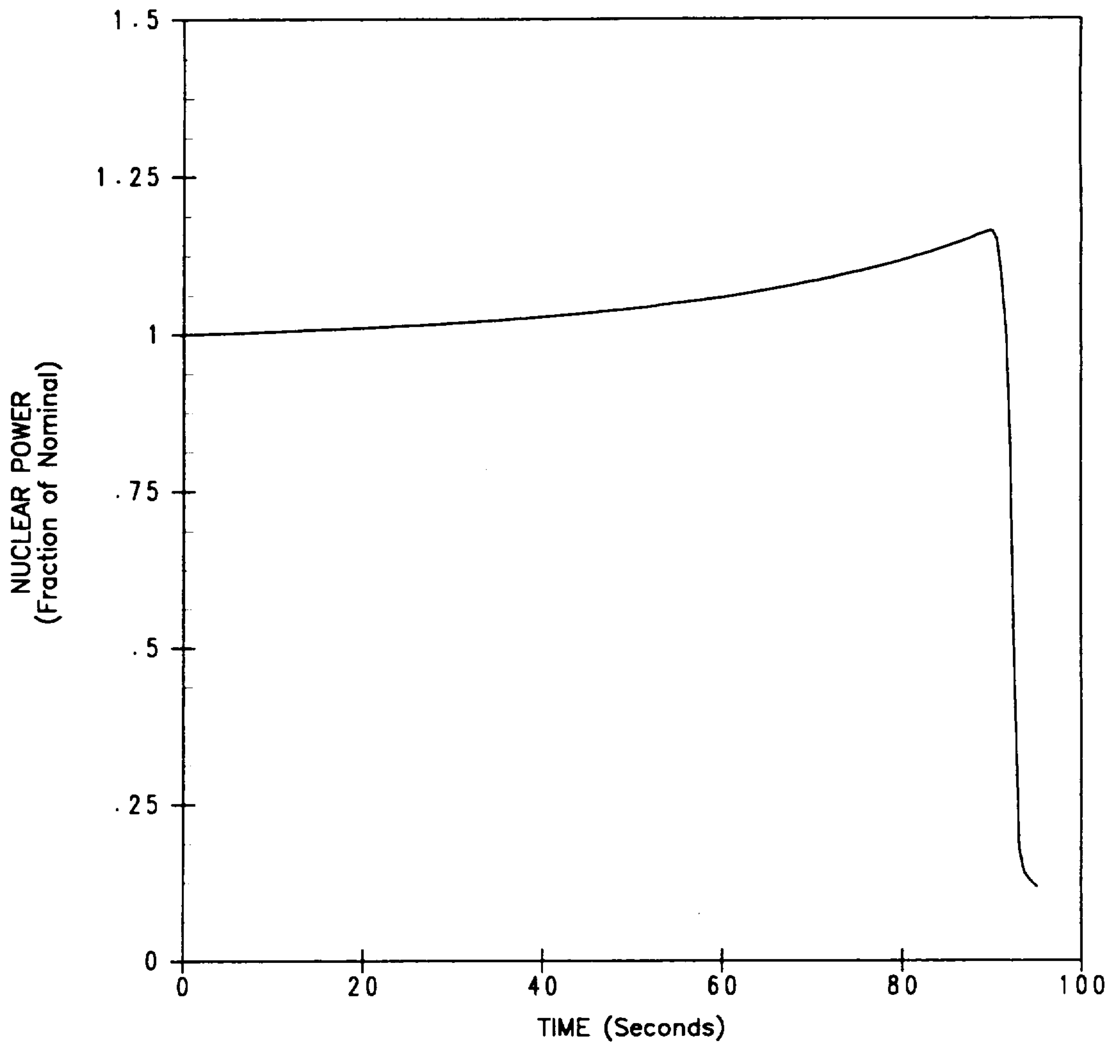


Figure 6.2.14-7

Uncontrolled RCCA Bank Withdrawal at Power, Withdrawal Rate of 0.3 pcm/second, 100% Power, Minimum Reactivity Feedback

Nuclear Power vs. Time

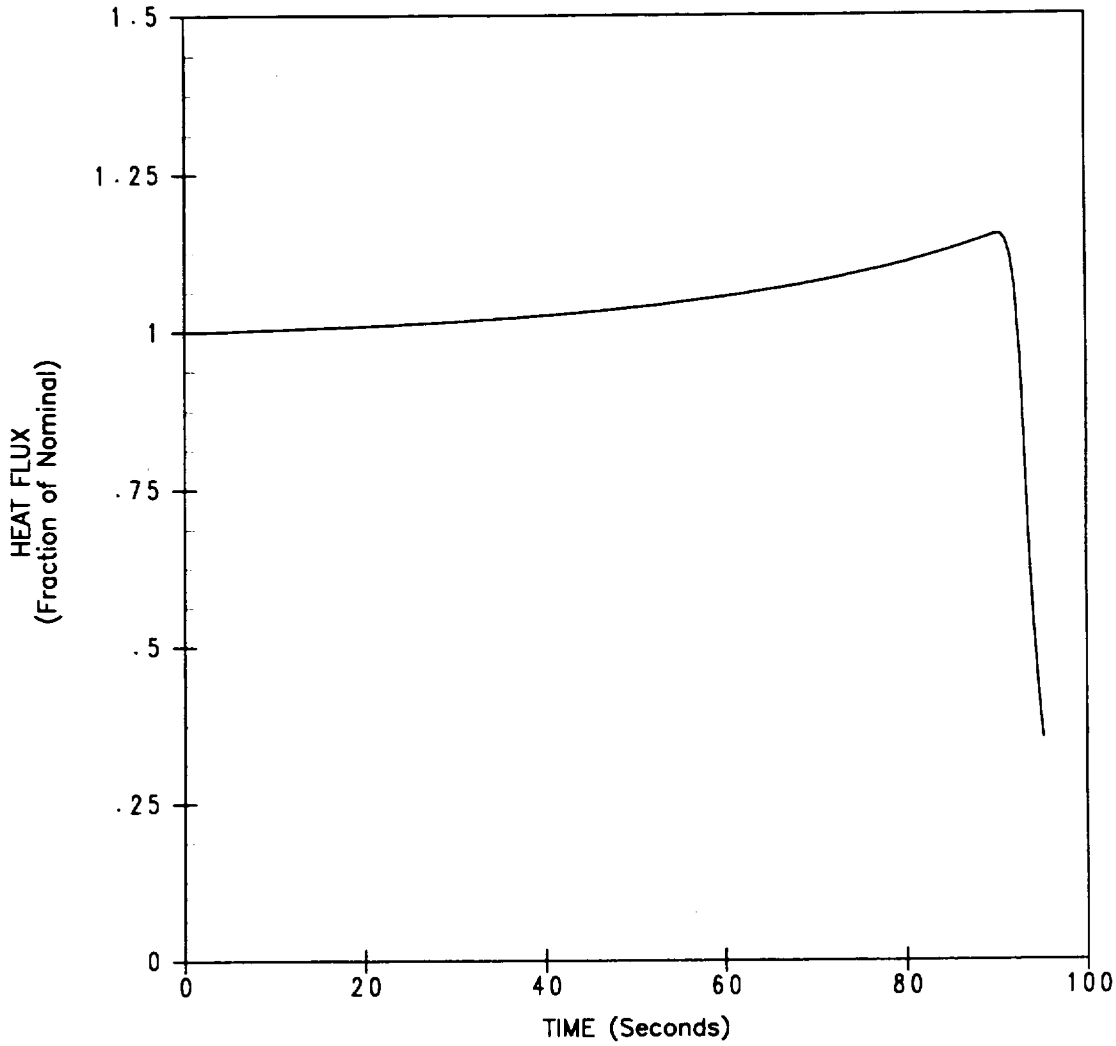


Figure 6.2.14-8

Uncontrolled RCCA Bank Withdrawal at Power, Withdrawal Rate of 0.3 pcm/second, 100% Power, Minimum Reactivity Feedback

Core Heat Flux vs. Time

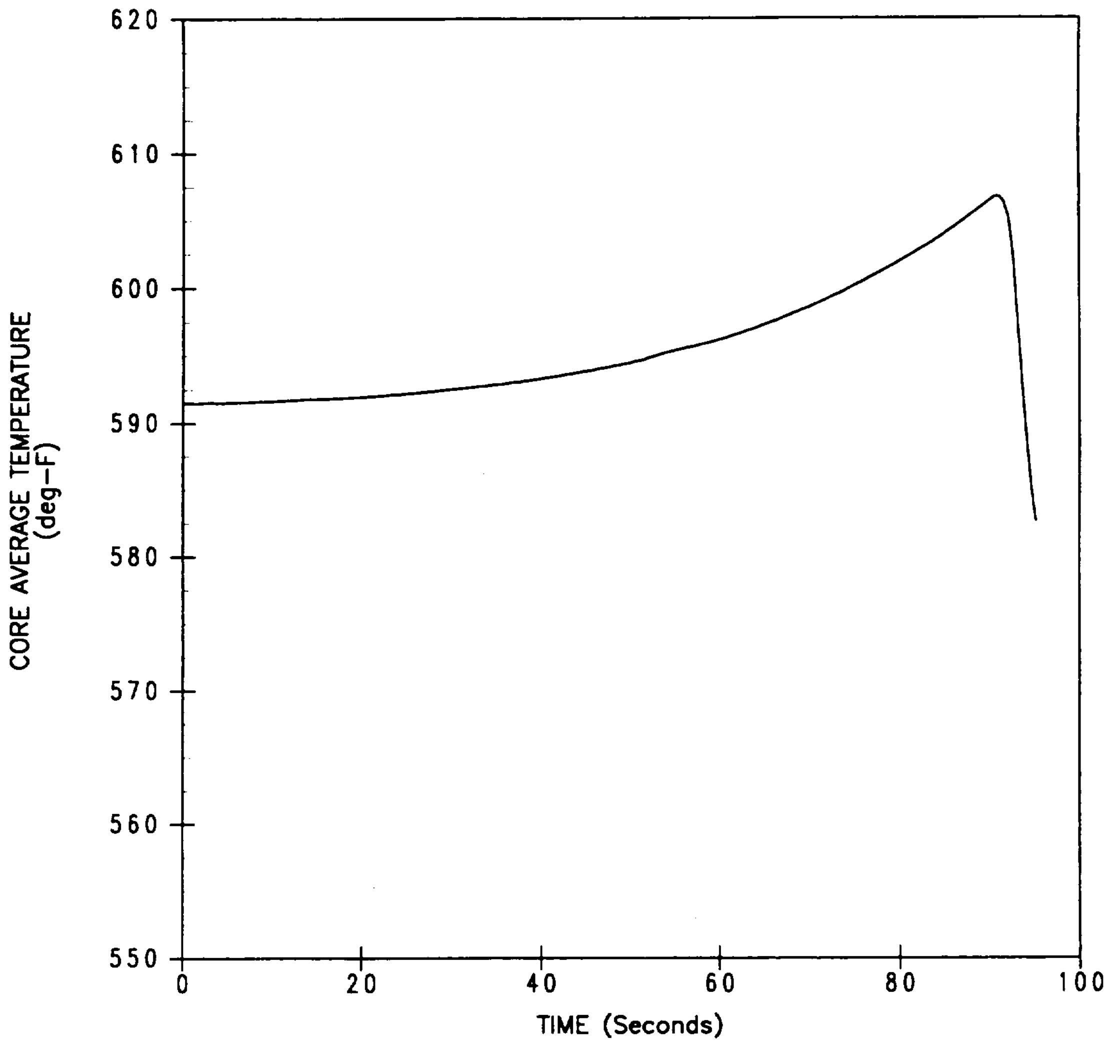


Figure 6.2.14-9

Uncontrolled RCCA Bank Withdrawal at Power, Withdrawal Rate of 0.3 pcm/second, 100% Power, Minimum Reactivity Feedback

Core Average Temperature vs. Time

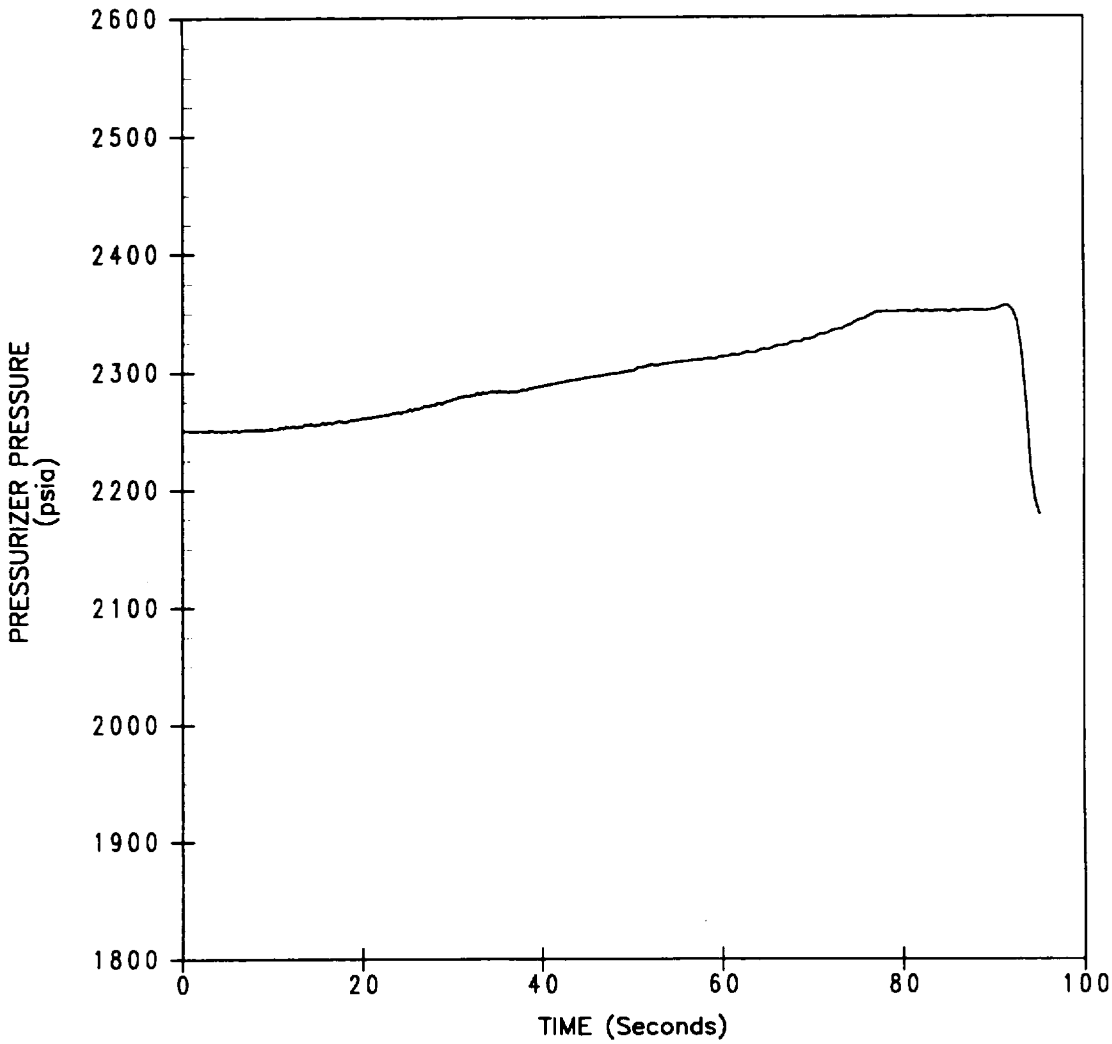


Figure 6.2.14-10

Uncontrolled RCCA Bank Withdrawal at Power, Withdrawal Rate of 0.3 pcm/second, 100% Power, Minimum Reactivity Feedback Pressurizer Pressure vs. Time

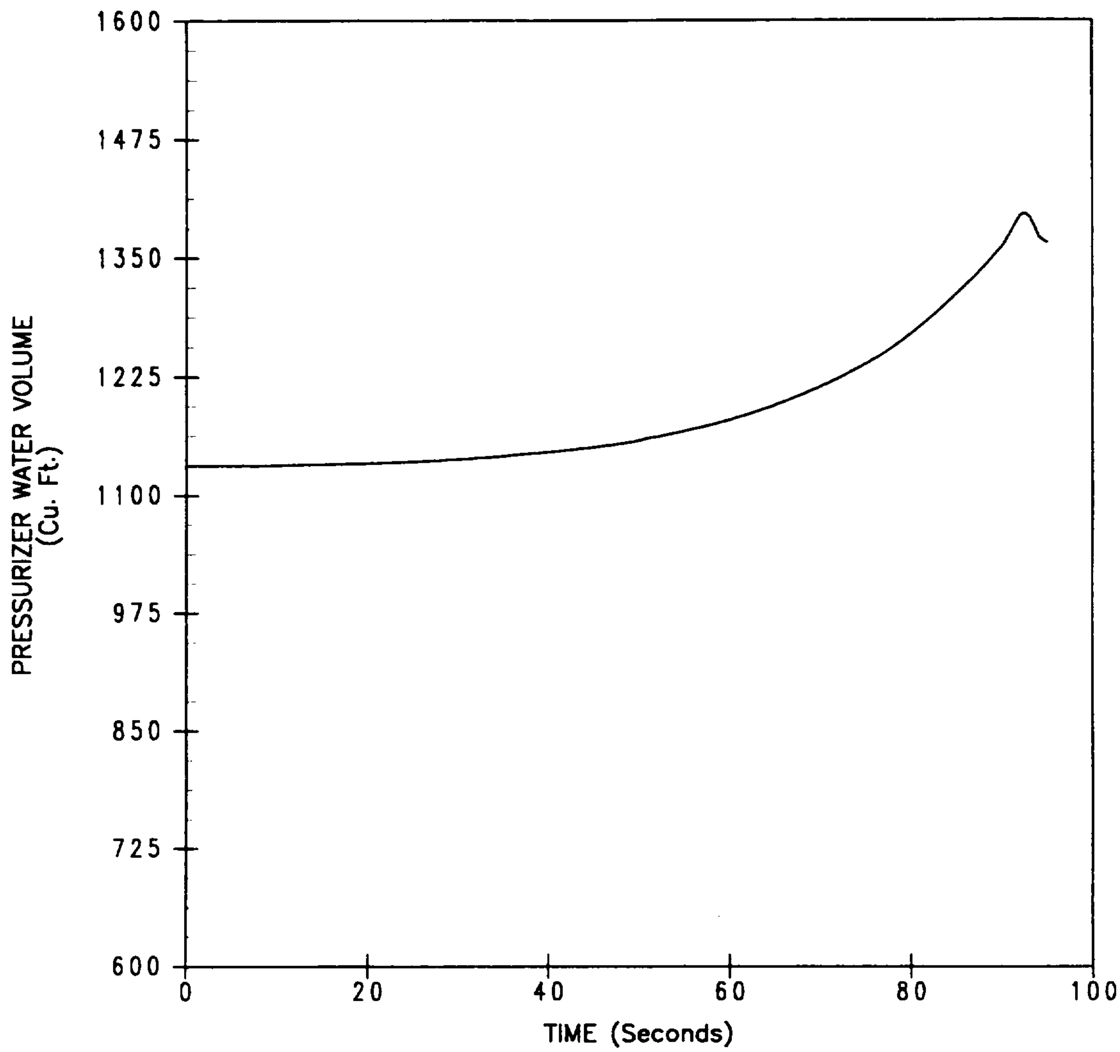


Figure 6.2.14-11

Uncontrolled RCCA Bank Withdrawal at Power, Withdrawal Rate of 0.3 pcm/second, 100% Power, Minimum Reactivity Feedback Pressurizer Water Volume vs. Time

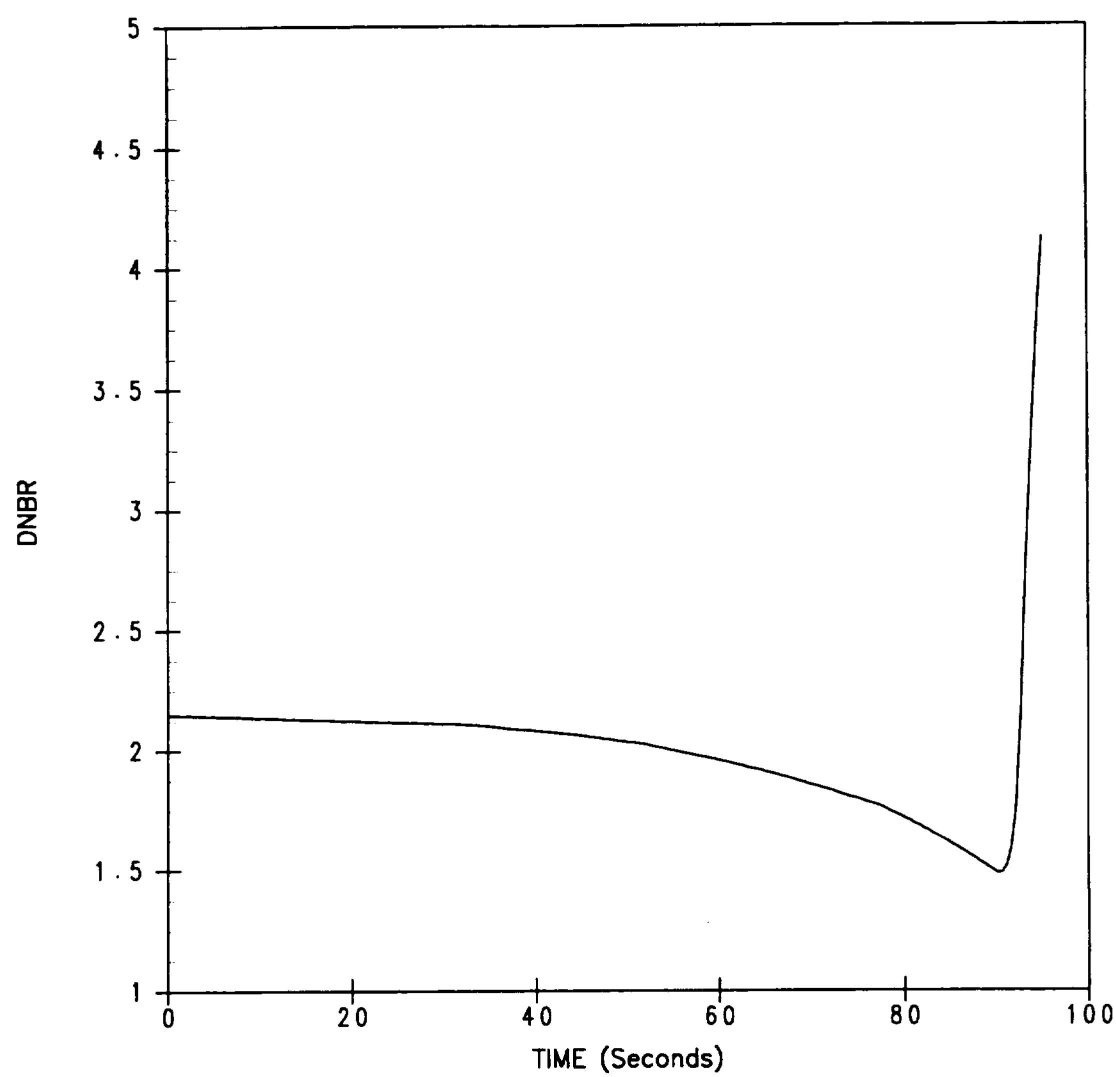


Figure 6.2.14-12

Uncontrolled RCCA Bank Withdrawal at Power, Withdrawal Rate of 0.3 pcm/second, 100% Power, Minimum Reactivity Feedback DNBR vs. Time

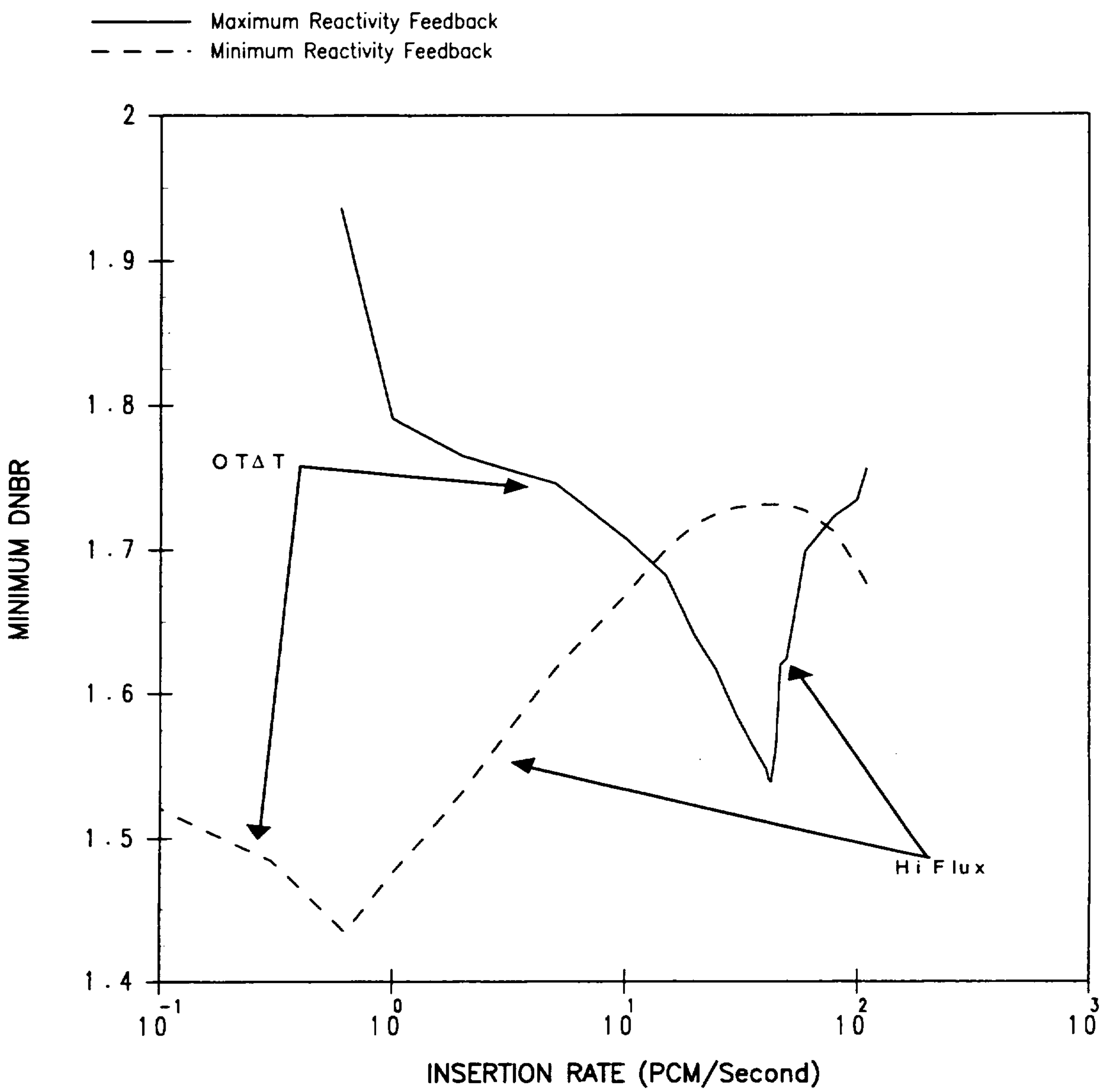


Figure 6.2.14-13

Uncontrolled RCCA Bank Withdrawal at 100% Power

Minimum DNBR vs. Reactivity Insertion Rate

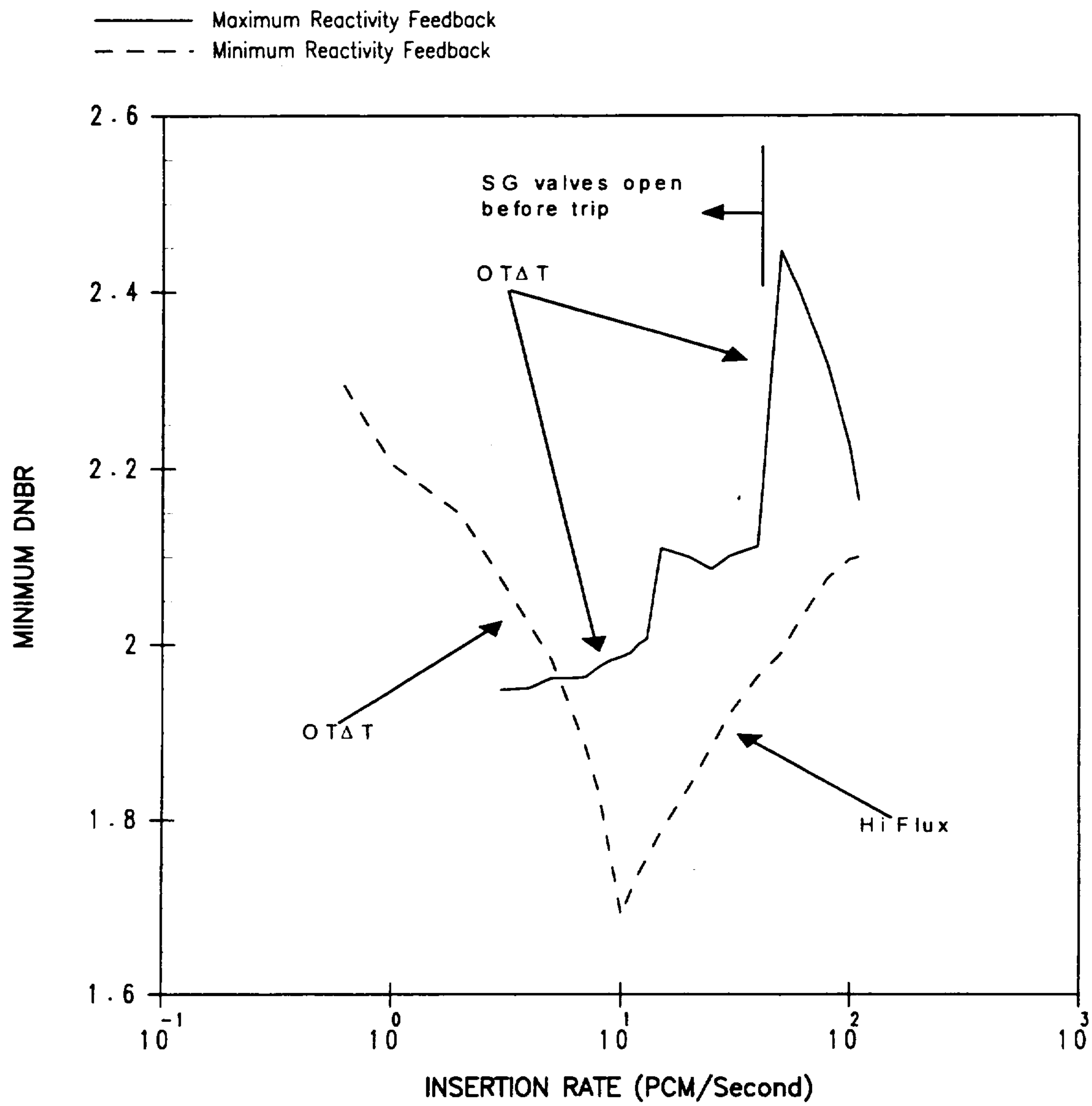


Figure 6.2.14-14

Uncontrolled RCCA Bank Withdrawal at 60% Power

Minimum DNBR vs. Reactivity Insertion Rate

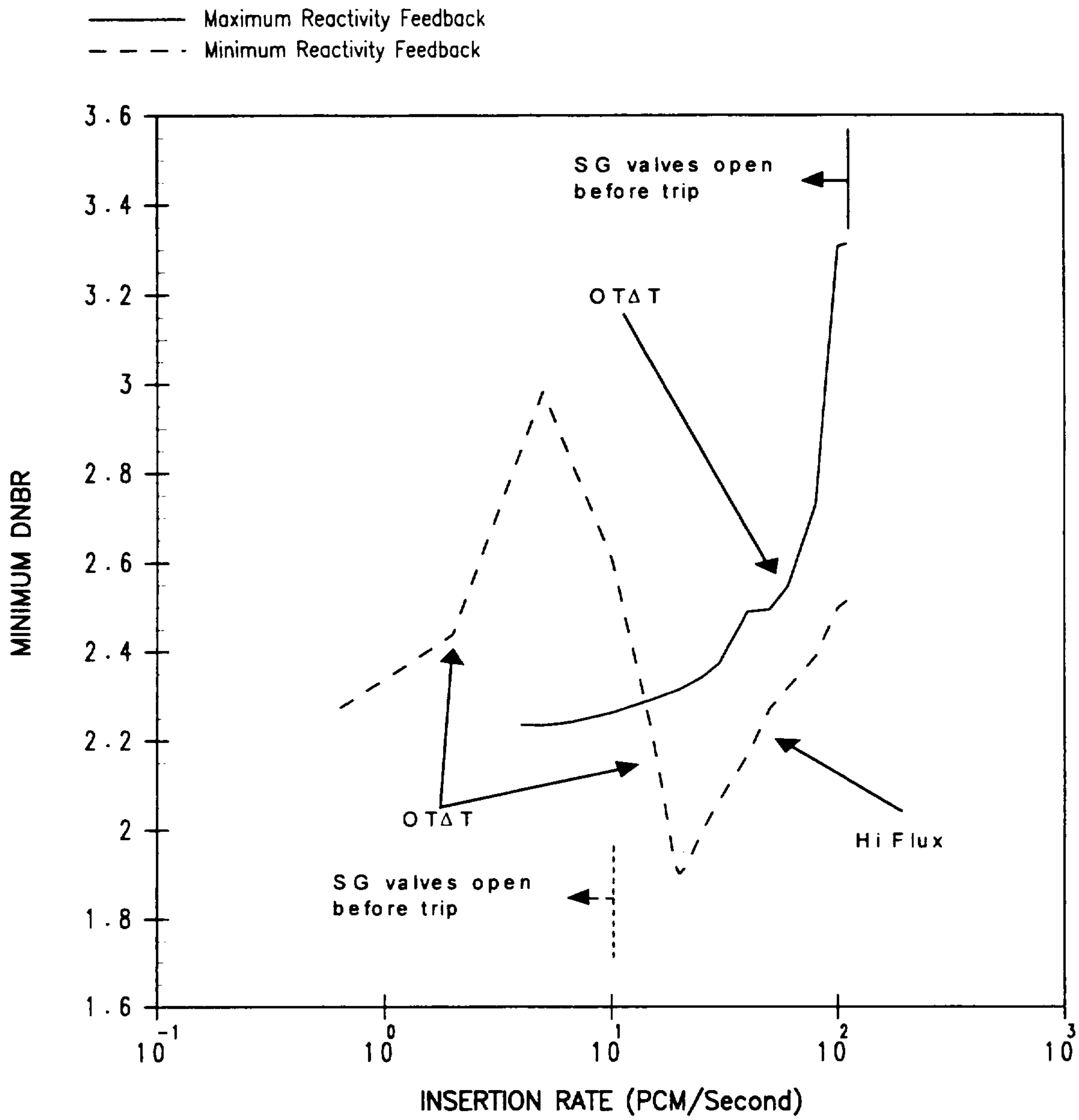


Figure 6.2.14-15

Uncontrolled RCCA Bank Withdrawal at 10% Power

Minimum DNBR vs. Reactivity Insertion Rate

6.2.15 Rod Cluster Control Assembly Misoperation

6.2.15.1 Introduction

Rod Cluster Control Assembly (RCCA) misoperation accidents include the following:

a. One or more dropped RCCAs within the same group,

b. A dropped RCCA bank,

c. A statically misaligned RCCA, and

d. Withdrawal of a single RCCA.

Each RCCA has a position indicator channel that displays the position of the assembly in a display grouping that is convenient to the operator. Fully inserted assemblies are also indicated by a rod at bottom signal which actuates a local alarm and control room annunciator. Group demand position is also indicated.

RCCAs move in preselected banks, and the banks always move in the same preselected sequence. Each bank of RCCAs consists of two groups. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation) of the stationary gripper, movable gripper, and lift coils of the control rod drive mechanism withdraws the RCCA held by the mechanism. Any single failure that could cause withdrawal would affect a minimum of one group, or four RCCAs. Mechanical failures are in the direction of insertion or immobility. Note that the operator can deliberately withdraw a single RCCA in a control or shutdown bank since this feature is necessary to retrieve an assembly should one drop accidentally.

No single electrical or mechanical failure in the rod control system could cause accidental withdrawal of a single RCCA from the inserted bank at full-power operation. The event analyzed must result from multiple wiring and/or component failures, multiple significant operator errors, or subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is low such that the limiting consequences may include slight fuel damage.

A dropped RCCA or RCCA bank is detected by:

a. Sudden drop in the core power level as seen by the nuclear instrumentation system;

b. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples;

c. Rod at bottom signal;

d. Rod deviation alarm; and/or

e. Rod position indication.

Dropping of a full-length RCCA is assumed to be initiated by a single electrical or mechanical failure which causes any number and combination of rods from the same group of a given control bank to drop to the bottom of the core. The resulting negative reactivity insertion causes nuclear power to rapidly decrease. An increase in the hot channel factor may occur due to the skewed power distribution representative of a dropped rod configuration. For this event, it must be shown that the DNB design basis is met for the combination of power, hot channel factor, and other system conditions which exist following a dropped rod.

Misaligned RCCAs are detected by:

a. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples;

b. Rod deviation alarm; and/or

c. Rod position indication.

The resolution of the rod position indicator channel is ±5 percent of span (±12 steps). Any RCCA can deviate from its group by twice this distance (10 percent of span or 24 steps) and not cause power distributions exceeding design limits. The deviation alarm alerts the operator when any rod deviates from its group position by more than five percent of span. If the rod deviation alarm is not operable, the operator is required to take action as required by the plant Technical Specifications.

If one or more rod position indicator channels are out of service, the operator must follow detailed operating instructions to ensure alignment of the non-indicated RCCAs. The operator is also required to take additional action as required by the plant Technical Specifications.

In the unlikely event of simultaneous electrical failures that could result in single RCCA withdrawal, the plant annunciator will display both the rod deviation and rod control urgent failure alarms, and the rod position indicators will indicate the relative positions of the RCCAs in the bank. The urgent failure alarm also inhibits automatic rod motion of the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, would result in activation of the same alarm and visual indication. Although the OTΔT reactor trip provides automatic protection for this event, it is not always possible to ensure that core safety limits will not be exceeded. Hence, this function is not credited in the analysis for this event.

6.2.15.2 Input Parameters and Assumptions

For one or more dropped RCCA(s) in the same group, or a dropped RCCA bank, the analysis is based on the following conservative assumptions.

a. This accident is analyzed with the Revised Thermal Design Procedure (Reference 1). Therefore, initial reactor power, pressure and RCS average temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the DNBR limit calculated using the referenced methodology.

b. A range of moderator temperature coefficients from 0 pcm/°F to -35 pcm/°F was analyzed, which bounds the limiting time in life.

c. A range of negative reactivity insertions from 100 pcm to 800 pcm is assumed to simulate the Dropped RCCA event.

d. To provide a conservative analysis that minimizes the DNBR, the automatic rod control system is assumed to function. Operation of the automatic rod control system results in increased RCS average temperature and power as it attempts to restore nominal power following the dropped rod(s) event. The resulting RCS average temperature and power overshoot minimize the margin to the DNBR limit.

e. To provide a conservative analysis that minimizes the DNBR, the pressure-reducing functions of the automatic pressure control system are assumed to function. The pressure-reducing functions modeled are the pressurizer power-operated relief valves and spray. Modeling these functions minimize RCS pressure, which is conservative with respect to minimizing DNBR.

For a Statically Misaligned RCCA, the most-severe situations with respect to DNBR at significant power levels arise from cases in which one RCCA is fully inserted or where bank D is fully inserted with one RCCA fully withdrawn. Multiple independent alarms, including a Rod Insertion Limit (RIL) alarm, alert the operator well before the transient approaches the postulated conditions. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the safety analysis limit.

For a single RCCA withdrawal, power distributions are analyzed using appropriate nuclear physics computer codes. The peaking factors are then input to the THINC code to calculate the DNBR. The analysis examines the case of the worst rod withdrawn from bank D, which is initially at the insertion limit with the reactor at full power. The analysis assumes this incident to occur at beginning of life since this results in the minimum moderator temperature coefficient (least negative). This assumption maximizes the power rise and minimizes the tendency of the large moderator temperature coefficient (most negative) to flatten the power distribution.

6.2.15.3 Description of the Analysis

Dropped RCCA(s) and Dropped RCCA Bank

The transient response following a dropped RCCA or dropped bank event is calculated using a detailed digital simulation of the plant. The dropped rod or bank causes a step decrease in reactivity and the resulting core power generation is determined using the LOFTRAN code (Reference 2). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, rod control system, steam generators, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. Since LOFTRAN employs a point neutron kinetics model, a dropped rod event is modeled as a negative reactivity insertion corresponding to the reactivity worth of the dropped rod(s) or bank, regardless of the actual configuration of the rod(s) that drop.

Statepoints are calculated and nuclear models used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met. The transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the dropped rod methodology described in WCAP-11394 (Reference 3).

The Dropped RCCA and dropped bank events were conservatively analyzed at uprated conditions for the Byron/Braidwood Nuclear Plants. The impact on the analysis is primarily due to the revised rod control system parameters discussed in Section 6.2.0. To address the change in the rod control system response, calculations were performed to establish conservative penalties to be applied to the current dropped rod statepoints. The statepoint penalties were then used in the nuclear analysis, which confirmed that the DNB design basis is met.

Statically Misaligned RCCA

For the Statically Misaligned RCCA event, steady-state power distributions are analyzed at uprated power conditions (3586.6 MWt core) using appropriate nuclear physics computer codes. The analysis examines the case of the worst rod withdrawn from bank D inserted at the insertion limit with the reactor initially at full power. The analysis assumes this incident to occur at beginning of life since this results in the minimum value of the moderator temperature coefficient (least negative). This assumption maximizes the power rise and minimizes the tendency of the large moderator temperature coefficient (most negative) to flatten the power distribution.

Single RCCA Withdrawal

The analysis of the single rod withdrawal event considers the following two cases.

a. If the reactor is in manual rod control mode, continuous withdrawal of a single RCCA results in an increase in core power and coolant temperature, and an increase in the local hot channel factor in the area of the withdrawing RCCA. In terms of the overall system response, this case is similar to those presented in Section 6.2.14; however, the increased local power peaking in the area of the withdrawing RCCA results in lower DNBRs than for the withdrawing bank cases. Depending on initial bank insertion and location of the withdrawing RCCA, automatic reactor trip may not occur quickly enough to ensure that the minimum DNBR remains above the safety analysis limit. Evaluation of this case at the power and coolant conditions at which the OTΔT trip would trip the plant shows that an upper limit for the number of fuel rods with a DNBR less than the safety analysis limit value is 5 percent.

b. If the reactor is in automatic rod control mode, the multiple failures that result in the withdrawal of a single RCCA immobilize the other RCCAs in the controlling bank. The transient will then proceed in the same manner as Case A, described above.

For such cases, a reactor trip will ultimately ensue, although not quickly enough in all cases to ensure a minimum DNBR in the core above the safety analysis limit value. Following reactor trip, applicable plant operating procedures are followed to stabilize the plant.

6.2.15.4 Acceptance Criteria

Based on frequency of occurrence, a misaligned or dropped RCCA is considered a Condition II event as defined by the American Nuclear Society. The primary acceptance criterion for these events is that the critical heat flux should not be exceeded. This is demonstrated by precluding Departure from Nucleate Boiling (DNB).

A single RCCA withdrawal event is classified as a Condition III incident. No single electrical or mechanical failure in the rod control system could cause accidental withdrawal of a single RCCA from the inserted bank at full power operation. The probability of the combination of conditions required to cause this event is considered low such that the limiting consequences may include slight fuel damage. The limit of fuel damage is set at five percent of the total fuel rods present in the core.

6.2.15.5 Results

<u>Dropped RCCA(s) or Bank</u>

Following a dropped RCCA(s) or bank event, with or without automatic rod withdrawal, the plant will establish a new equilibrium condition. The transient response of a representative case is presented in the FSAR (Reference 4) Figure 15.4-12a. In all cases, the minimum DNBR remains above the limit; therefore, the acceptance criterion is met.

Statically Misaligned RCCA

The most-severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCCA is fully inserted or where bank D is fully inserted with one RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the transient approaches the postulated conditions. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the safety analysis limit.

The insertion limits in the Technical Specifications may vary over time depending on several limiting criteria. Therefore, it is preferable to analyze the misaligned RCCA case at full power for a position of the control bank as deeply inserted as the criterion on minimum DNBR and power peaking factor will allow. The full-power insertion limits on control bank D must then be chosen to be above that position and will usually be dictated by other criteria. Detailed results will vary from cycle to cycle depending on fuel arrangements.

For RCCA misalignment with bank D inserted to its full-power insertion limit and one RCCA fully withdrawn, DNBR does not fall below the safety analysis limit. The analysis of this case assumes that initial reactor power, pressure, and RCS temperature are at their nominal values, with the increased radial peaking factor associated with the misaligned RCCA.

For RCCA misalignment with one RCCA fully inserted, the DNBR does not fall below the safety analysis limit. The analysis of this case assumes that initial reactor power, pressure, and RCS temperatures are at their nominal values, with the increased radial peaking factor associated with the misaligned RCCA.

Single RCCA Withdrawal

For accidental withdrawal of a single RCCA, with the reactor in automatic or manual control mode and initially operating at full power with bank D at the insertion limit, the analysis at uprated core power conditions of 3586.6 MWt shows that the number of fuel rods experiencing a DNBR below the safety analysis DNBR limit is less than five percent of the total fuel rods in the core.

For both cases discussed, the indicators and alarms mentioned would alert the operator to the malfunction before DNB would occur. For Case B, the insertion limit alarms (low and low-low alarms) would serve in this regard.

6.2.15.6 Conclusions

Following a dropped RCCA(s) or dropped bank event the plant will return to a stabilized condition. Results of the analysis show that a dropped RCCA or dropped bank event, with or without a reactor trip, does not adversely affect the uprated core since the DNBR remains above the limit for a range of dropped RCCA worths.

DNB does not occur for the RCCA misalignment incident at uprated power conditions; thus, there is no reduction in the ability of the primary coolant to remove heat from the fuel rod. After identifying an RCCA group misalignment condition, the operator must take action as required by the plant Technical Specifications and operating instructions.

The results presented above for the single RCCA withdrawal event demonstrate that the number of fuel rods experiencing a DNBR below the safety analysis limit is less than five percent of the total fuel rods in the core. Hence, it is concluded that the applicable acceptance criterion for this event is met at uprated power conditions for both cases of the accidental withdrawal of a single RCCA considered.

6.2.15.7 References

1. Friedland, A.J., and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary), WCAP-11397-A (Non-Proprietary), April 1989.

2. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984.

3. Haessler, R.L., et al, "Methodology for the Analysis of the Dropped Rod Event," WCAP-11394-P-A (Proprietary) and WCAP-11395 (Non-Proprietary), April 1987.

4. "Byron & Braidwood Station, Updated Final Safety Analysis Report," Revision 7, Docket Nos. STN-454/455/456/457, as amended through December 1998.

6.2.16 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

The Technical Specifications require that all four reactor coolant pumps be operating for reactor power operation. Therefore, operation with an inactive loop is precluded by the Technical Specifications.

This event was originally included in the FSAR licensing basis when operation with a loop out of service was considered. Based on the Technical Specifications, which prohibit at-power operation with an inactive loop, and previous changes to the Technical Specifications that deleted all references to three-loop operation, this event has been deleted from the UFSAR.

6.2.16.1 Reference

1. "Byron & Braidwood Station, Updated Final Safety Analysis Report," Revision 7, Docket Nos. STN-454/455/456/457, as amended through December 1998.

6.2.17 Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant

6.2.17.1 Introduction

The principal means of causing inadvertent boron dilution are opening of the primary water makeup control valve and failure of the blend system, either by controller or mechanical failure. The CVCS and RMCS are designed to limit, even under various postulated failure modes, the potential rate of dilution to values which, with indication by alarms and instrumentation, will allow sufficient time for operator response to terminate the dilution. An inadvertent dilution from the RMCS may be terminated by closing primary water makeup control valve CV-111A. All expected sources of dilution may be terminated by closing isolation valves in the CVCS, LCV-112B and -112C. The lost shutdown margin (SDM) may be regained by the opening of isolation valves to the RWST, LCV-112D and -112E, thus allowing the addition of 2300 ppm borated water to the RCS.

The addition rate of unborated water to the RCS is limited to 205 gpm by the capacity of the primary water makeup pumps and two centrifugal charging pumps.

Generally, to dilute, the operator must perform two distinct actions:

a. Switch control of the RMCS from automatic makeup mode to dilute mode, and

b. Set RMCS control switch to start.

Failure to carry out either of the above actions prevents initiation of dilution. Also, during normal operation the operator may add borated water to the RCS by blending boric acid from the boric acid storage tanks with primary grade water. This requires the operator to determine the concentration of the addition and setting of the blended flow rate and boric acid flow rate. The makeup controller will then limit the sum of the boric acid flow rate and primary grade water flow rate to the blended flow rate, i.e., the controller determines the primary grade water flow rate, after the RMCS control switch is set to start.

The status of the RCS makeup is continuously available to the operator by:

a. Indication of boric acid and blended flow rates,

b. CVCS, AB, and PW pump status lights,

c. Deviation alarms if the boric acid or blended flow rates deviate by more than 10 percent from the preset values,

d. Source range neutron flux - when reactor is subcritical;

   1. high flux at shutdown alarm,

   2. indicated source range neutron flux count rates,

   3. audible source range neutron flux count rate, and

   4. source range neutron flux - doubling alarm.

e. With the reactor critical

   1. Axial flux difference alarm (reactor power ≥ 50 percent RTP),

   2. Control rod insertion limit low and low-low alarms,

   3. Overtemperature ΔT alarm (at power),

   4. Overtemperature ΔT turbine runback (at power), and

   5. Overtemperature ΔT reactor trip.

f. Power Range Neutron Flux - High, both high and low setpoint reactor trips.

This event is classified as an ANS Condition II incident (a fault of moderate frequency) as described in Subsection 6.2.0.

6.2.17.2 Input Assumptions and Description of Analysis

To cover all phases of plant operation, boron dilution during refueling, cold shutdown, hot shutdown, hot standby, startup, and power modes of operation are considered in this analysis.

Conservative values for necessary parameters were used, i.e., high RCS critical boron concentrations, high boron worth, minimum shutdown margins, and lower than actual RCS volumes. These assumptions result in conservative determinations of the time available for operator or system response after detection of a dilution transient in progress.

Conservative analysis methods are used to analyze a CVCS malfunction that results in a decrease in boron concentration in the reactor coolant. Minimum reactor coolant volumes and maximum dilution flow rates are conservatively assumed for each case analyzed. The result is a logarithmic decrease in coolant boron concentration according to the equation:

$$dC_B/dt = - [Q_{in}/V]\ C_B$$

where

C_B = Boron concentration in the RCS,

Q_{in} = Maximum dilution flow rate, and

V = Active volume in RCS.

This equation can be solved for the time at which the core would become critical or all shutdown margin would be lost. The rate of reactivity insertion due to the dilution can be calculated from the dilution rate and the differential boron worth. The results of this analysis are conservative for all cases analyzed.

A comprehensive review of the primary system showed that a single failure in the CVCS system in the cold shutdown condition would not result in a boron dilution of the reactor coolant system, and that the CVCS malfunction represents the most limiting potential source of dilution. Based on this review, the analysis results presented herein bound all potential sources of inadvertent dilution under all modes of operation.

6.2.17.2.1 Dilution During Refueling

An uncontrolled boron dilution transient cannot occur during this mode of operation. Inadvertent dilution is prevented by administrative controls that isolate the RCS from potential sources of unborated water. Unborated water source isolation valves, specified in Technical Specification 3.9.2, will be verified closed and secured in position by mechanical stops or by removal of air or electrical power. These valves block all flow paths that could allow unborated makeup water to reach the RCS. Any makeup that is required during refueling will be borated water supplied from the RWST by the low-head safety injection pumps.

6.2.17.2.2 Dilution During Cold Shutdown

The analysis for this mode of operation will be supplied in a separate submittal.

6.2.17.2.3 Dilution During Hot Shutdown

The analysis for this mode of operation will be supplied in a separate submittal.

6.2.17.2.4 Dilution During Hot Standby

The analysis for this mode of operation will be supplied in a separate submittal.

6.2.17.2.5 Dilution During Startup

Startup is a transitory mode of operation. In this mode, the plant is being taken from one long-term mode of operation, hot standby, to another, power operation. The plant is maintained in the startup mode only for the purpose of startup testing at the beginning of each cycle. During this mode of operation the plant is in manual control, i.e., T_{avg}/rod control is in manual. All normal actions required to change power level either up or down require operator initiation. The Technical Specifications require a SDM of 1.3 percent $\Delta K/K$ and four reactor coolant pumps operating. Other conditions assumed are:

a. Dilution flow is limited by the reactor makeup water system. The maximum anticipated flow rate of unborated primary water to the RCS is 205 gpm.

b. A minimum RCS water volume of 10,039.51 ft^3. This is a conservative estimate of the active RCS volume, minus the pressurizer volume.

c. The C_B for criticality (ARI-1) is assumed to be 1730 ppm.

6.2.17.2.6 Dilution During Full Power Operation

The plant may be operated at power two ways, automatic T_{avg}/rod control and under operator control. The Technical Specifications require an available trip reactivity of 1.3 percent ΔK/K and four reactor coolant pumps operating. With the plant at power and the RCS at pressure, the dilution rate is limited by the capacity of the reactor makeup water system with two centrifugal charging pumps in operation (analysis is performed assuming two charging pumps are in operation even though normal operation is with one pump). Conditions assumed for this mode are:

a. For manual and automatic reactor control at full power conditions, the dilution flow is limited by the reactor makeup water system (205 gpm).

b. A minimum RCS water volume of 10,039.51 ft3. This is a conservative estimate of the active RCS volume, minus the pressurizer volume.

c. The CB for criticality (ARI-1) is assumed to be 1730 ppm.

6.2.17.3 Acceptance Criteria

A CVCS malfunction is classified as an ANS Condition II event, a fault of moderate frequency. Criteria established for Condition II events are as follows.

- The critical heat flux should not be exceeded. This is ensured by demonstrating that the minimum DNBR does not go below the limit value at any time during the transient.
- Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures.
- Fuel temperature and fuel clad strain limits should not be exceeded. The peak linear heat generation rate should not exceed a value that would cause fuel centerline melt.

This event is analyzed to ensure that there is sufficient time for mitigation of an inadvertent boron dilution prior to the complete loss of shutdown margin. A complete loss of plant shutdown margin results in a return of the core to the critical condition, causing an increase in the RCS temperature and heat flux. This could violate the safety analysis DNBR limit and challenge fuel and fuel cladding integrity. A complete loss of plant shutdown margin could also result in a return of the core to the critical condition, causing an increase in RCS pressure. This could challenge the pressure design limit for the reactor coolant system.

If the minimum allowable shutdown margin is shown not to be lost, the condition of the plant at any point in the transient is within the bounds of those calculated for other Condition II transients. By showing that the above criteria are met for those Condition II events, it can be concluded that they are also met for the boron dilution event. Operator action is relied upon to preclude a complete loss of plant shutdown margin.

6.2.17.4 Results

6.2.17.4.1 Dilution During Refueling

Dilution during this mode has been precluded through administrative control of valves in the possible dilution flow paths, see Subsection 6.2.17.2.1.

6.2.17.4.2 Dilution During Cold Shutdown

The analysis for this mode of operation will be supplied in a separate submittal.

6.2.17.4.3 Dilution During Hot Shutdown

The analysis for this mode of operation will be supplied in a separate submittal.

6.2.17.4.4 Dilution During Hot Standby

The analysis for this mode of operation will be supplied in a separate submittal.

6.2.17.4.5 Dilution During Startup

This mode of operation is a transitory mode when going to power and is the operational mode in which the operator intentionally dilutes and withdraws control rods to take the plant critical. During this mode, the plant is in manual control with the operator required to maintain a very high awareness of the plant status. For a normal approach to criticality, the operator must manually initiate a limited dilution and subsequently manually withdraw the control rods, a process that takes several hours. The plant Technical Specifications require that the operator determine the estimated critical position of the control rods prior to approaching criticality, thus assuring that the reactor does not go critical with the control rods below the insertion limits. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the source range reactor trip (nominally at 10^5 cps) after receiving P-6 from the intermediate range. Too fast a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly, leaving insufficient time to manually block the source range reactor trip. Failure to perform this manual action results in a reactor trip and immediate shutdown.

However, in the event of an unplanned approach to criticality or dilution during power escalation while in startup mode, the plant status is such that minimal impact will result. The plant will slowly escalate in power to a reactor trip on the power range neutron flux - high, low setpoint (nominally 25 percent RTP). After reactor trip, there are more than 15 minutes for operator action prior to return to criticality. The required operator action is to initiate and continue boration until adequate shutdown margin is restored, and to terminate the dilution.

6.2.17.4.6 Dilution During Full Power Operation

With the reactor in manual control and no operator action taken to terminate the transient, the power and temperature rise will cause the reactor to reach the overtemperature ΔT or Power Range High Neutron Flux trip setpoint resulting in a reactor trip. After reactor trip, there are more than 15 minutes for operator action prior to return to criticality. The required operator action is to initiate and continue boration until adequate shutdown margin is restored, and to terminate the dilution. The boron dilution transient in this case is essentially equivalent to an uncontrolled rod withdrawal at power. The maximum reactivity insertion rate for a boron dilution transient is conservatively estimated to be 2.02 pcm/sec, and is within the range of insertion rates analyzed for uncontrolled rod withdrawal at power. It should be noted that prior to reaching the overtemperature ΔT reactor trip, the operator would have received an alarm on overtemperature ΔT turbine runback.

With the reactor in automatic rod control, a boron dilution will result in a power and temperature increase such that the rod controller will attempt to compensate by slow insertion of the control rods. This action by the controller will result in at least one of three alarms to the operator:

a. rod insertion limit - low level alarm,

b. rod insertion limit - low-low level alarm if insertion continued after (a), and

c. an axial flux difference alarm (ΔI outside of the target band).

Given the many alarms, indications, and the inherent slow process of dilution at power, the operator has sufficient time for action. For example, the operator has more than 15 minutes from the rod insertion limit low-low alarm until 1.3 percent $\Delta K/K$ is inserted at beginning-of-life. The time would be significantly longer at end-of-life, due to the low initial boron concentration, when shutdown margin is a concern.

The above results demonstrate that, in all modes of operation, an inadvertent boron dilution is precluded, responded to by automatic functions, or sufficient time is available for operator action to terminate the transient. Following termination of dilution flow and initiation of boration, the reactor is in a stable condition with the operator regaining the required shutdown margin.

6.2.17.5 Conclusions

If an unintentional dilution of boron in the reactor coolant system does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition rate due to the dilution is slow enough to allow the operator sufficient time to determine the cause and take corrective action before shutdown margin is lost. The acceptance criteria as specified in Section 6.2.17.3 are met.

6.2.17.6 References

None.

6.2.18 Inadvertent Loading of a Fuel Assembly into an Improper Position

6.2.18.1 Introduction

Fuel and core loading errors can arise from inadvertent loading of one or more fuel assemblies into improper positions in the core, loading of a fuel rod during manufacture with one or more pellets of the wrong enrichment, or loading of a full fuel assembly during manufacture with pellets of the wrong enrichment. These misloads will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core loading errors is the inadvertent loading of one or more fuel assemblies that require burnable absorber rods into a new core without burnable absorber rods.

6.2.18.2 Input Parameters and Assumptions

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes which are more peaked than those calculated with the correct enrichments. There is a five percent uncertainty margin included in the design value of the power peaking factor assumed in the analysis of Condition I and Condition II transients. The incore system of moveable flux detectors, used to verify power shapes at the start of life, is capable of revealing any assembly enrichment error or loading error which causes power peaking in excess of the design value.

To reduce the probability of core loading errors, each fuel assembly is marked with an identification number and loaded in accordance with a core loading diagram. During core loading, the identification number is checked before each assembly is moved into the core. These identification numbers are subsequently recorded onto the loading diagram, after loading is completed, as a further check that the assemblies are properly placed.

6.2.18.3 Description of Analysis

Power distribution in the x-y plane of the core and resulting thermal hydraulic conditions are analyzed with the steady-state computer programs briefly described in Chapter 4 of the UFSAR (Reference 1). A discrete representation is used wherein each individual fuel rod is described by a mesh interval. The assembly power distributions in the x-y plane for a correctly loaded core, based on associated enrichments, are also given in Chapter 4.

6.2.18.4 Acceptance Criteria

The power distortion due to any combination of misplaced fuel assemblies would significantly raise peaking factors and be readily observable with incore flux monitors. In addition to the flux monitors, thermocouples are located at the outlet of about one-third of the fuel assemblies in the core. There is a high probability that these thermocouples would also indicate any abnormally high coolant enthalpy rise. Incore flux measurements taken during the startup subsequent to every refueling operation must indicate that the power peaking factor is less than the design limit.

6.2.18.5 Results

The analysis presented in Section 15.4.7 of the UFSAR was reviewed with respect to plant operation at uprated power conditions. The evaluation concluded that operation at uprated power conditions does not affect the ability of the in-core instrumentation to detect the inadvertent loading and subsequent operation with a fuel assembly in an improper position; therefore, the conclusions presented in UFSAR Section 15.4.7 remain valid.

6.2.18.6 Conclusions

Fuel assembly enrichment errors would be prevented by administrative procedures implemented in fabrication.

In the event that a single pin or pellet has a higher enrichment than the nominal value, the consequences in terms of reduced DNBR and increased fuel and clad temperatures will be limited to the incorrectly loaded pin or pins.

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, analyses in this section confirm that the resulting power distribution effects will either be readily detected by the incore moveable detector system or cause a sufficiently small perturbation to be within the uncertainties allowed between nominal and design power shapes.

6.2.18.7 References

1. "Byron & Braidwood Station, Updated Final Safety Analysis Report," Revision 7, Docket Nos. STN-454/455/456/457, as amended through December 1998.

6.2.19 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

6.2.19.1 Introduction

This accident is defined as a mechanical failure of a control rod drive mechanism pressure housing resulting in the ejection of the Rod Cluster Control Assembly (RCCA) and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. The resultant core thermal power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by reactor trip actuated by high nuclear power signals.

A failure of a control rod drive mechanism housing sufficient to allow a control rod to be rapidly ejected from the core is not considered credible for the following reasons.

a. Each full-length mechanism housing is completely assembled and shop tested at 4100 psig.

b. The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head and checked during the hydrotest of the completed Reactor Coolant System.

c. Stress levels in the mechanism are not affected by anticipated system transients at power or by thermal movement of the coolant loops. Moments induced by the design earthquake can be accepted within the allowable primary working stress ranges specified in the ASME Code, Section III, for Class I components.

d. The latch mechanism housing and rod travel housing are each a single length of forged type-304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered.

A significant margin of strength in the elastic range, together with the large energy absorption capability in the plastic range, gives additional assurance that gross failure of the housing will not occur. The joints between the latch mechanism housing and rod travel housing are threaded joints reinforced by canopy-type rod welds.

In general, the reactor is operated with the rod cluster control assemblies inserted only far enough to control design neutron flux shape. Reactivity changes caused by core depletion are compensated for by boron changes. Further, the location and grouping of control rod banks are selected during the nuclear design to lessen the severity of a rod cluster control assembly ejection accident. Therefore, should a rod cluster control assembly be ejected from its normal position during full-power operation, only a minor reactivity excursion, at worst, could be expected to occur. The position of all rod cluster control assemblies is continuously indicated in the control room. An alarm will occur if a bank of rod cluster control assemblies approaches its insertion limit or if one control rod assembly deviates from its bank. There are low and low-low level insertion alarm circuits for each bank. The control rod position monitoring and alarm systems are described in Reference 1.

6.2.19.2 Input Parameters and Assumptions

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The most important parameters are discussed below. Table 6.2.19-1 lists the parameters used in this analysis.

Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using either three-dimensional static methods or a synthesis of one- and two-dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux-flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. The analysis assumes adverse xenon distributions to provide worst-case results.

Appropriate margins are added to the ejected rod worth and hot channel factors to account for any calculational uncertainties, including an allowance for nuclear power peaking due to fuel densification.

Power distributions before and after ejection for a "worst case" can be found in Reference 1. During plant startup physics testing, ejected rod worths and power distributions have been measured in the zero- and full-power configurations and compared to the values used in the analysis. Experience has shown that the ejected rod worth and power peaking factors are consistently overpredicted in the analysis.

Delayed Neutron Fraction

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values of approximately 0.70 percent at beginning of life and 0.50 percent at end of life. The ejected rod accident is sensitive to β_{eff} if the ejected rod worth is equal to or greater than β_{eff}, as in the zero-power transients. To allow for future fuel cycle flexibility, conservative estimates of β_{eff} of 0.55 percent at beginning of cycle and 0.44 percent at end of cycle are used in the analysis.

Reactivity Weighting Factor

The largest temperature rises, and hence the largest reactivity feedbacks, occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple single-channel analysis. Physics calculations have been performed for temperature changes with a flat temperature distribution and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which, when applied to single-channel feedbacks, account for the effective whole-core feedbacks for the appropriate flux shape.

In this analysis, a one-dimensional (axial) spatial kinetics method is employed; thus axial weighting is not necessary if the initial condition is made to match the ejected rod configuration. In addition, no weighting is applied to moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature, as a function of time, accounting for the missing spatial dimension. These weighting factors have also been shown to be conservative when compared to three-dimensional analysis (Reference 1).

Moderator and Doppler Coefficient

The critical boron concentrations at the beginning of life and end of life are adjusted in the nuclear code in order to obtain moderator density coefficient curves which are conservative when compared to the actual design conditions for the plant. As discussed above, no weighting factor is applied to these results. The resulting moderator temperature coefficient is at least

+7 pcm/°F at the appropriate zero- or full-power nominal average temperature for the beginning-of-life cases.

The Doppler reactivity defect is determined as a function of power level using a one-dimensional steady-state computer code with a Doppler weighting factor of 1.0. The Doppler weighting factor will increase under accident conditions, as discussed above.

Heat Transfer Data

The FACTRAN (Reference 2) code, used to determine the hot spot transient, contains standard curves of thermal conductivity versus fuel temperature. During a transient, the peak centerline fuel temperature is independent of gap conductance during the transient. The cladding temperature is, however, strongly dependent on gap conductance and is highest for high gap conductance. For conservatism, a high gap heat transfer coefficient of 10,000 Btu/hr-ft^2-°F has been used during transients. This value corresponds to a negligible gap resistance and a further increase would have essentially no effect on the rate of heat transfer.

Coolant Mass Flow Rates

When the core is operating at full power, all four coolant pumps will always be operating. For zero-power conditions, the system is conservatively assumed to be operating with two pumps. The principal effect of operating at reduced flow is to reduce the film-boiling heat transfer coefficient. This results in higher peak cladding temperatures, but does not affect peak centerline fuel temperature. Reduced flow also lowers the critical heat flux. However, since DNB is always assumed at the hot spot and the heat flux rises very rapidly during the transient, this produces only second-order changes in the cladding and centerline fuel temperatures. All zero-power analyses for both average core and the hot spot have been conducted assuming two pumps in operation.

Trip Reactivity Insertion

The trip reactivity insertion is assumed to be 4% ΔK from hot full power and 2% ΔK from hot zero power, including the effect of one stuck RCCA. These values are also reduced by the ejected rod. The shutdown reactivity is simulated by dropping a rod of the required worth into the core. The start of rod motion occurs 0.5 seconds after reaching the power range high neutron flux trip setpoint. It is assumed that insertion to dashpot occurs 2.7 seconds after the rods begin to fall. The time delay to full insertion, combined with the 0.5 second trip delay, conservatively delays insertion of shutdown reactivity into the core.

The minimum design shutdown margin available for this plant at hot zero power (HZP) may only occur at end of life in the equilibrium cycle. This value includes an allowance for the worst stuck rod, an adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Physics calculations have shown that two stuck RCCAs (one of which is the worst ejected rod) reduce the shutdown margin by about an additional 1% ΔK. Therefore, following a reactor trip resulting from an RCCA ejection accident, the reactor will be subcritical when the core returns to HZP.

6.2.19.3 Description of Analysis

This section describes the models used in the analysis of the rod ejection accident. Only the initial few seconds of the power transient are discussed, since the long-term considerations are the same as for a loss of coolant accident.

The calculation of the RCCA ejection transient is performed in two stages, first an average core channel calculation and then a hot region calculation. The average core calculation uses spatial neutron-kinetics methods to determine average power generation versus time including the various total core feedback effects; i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients at the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and by performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is conservatively assumed to exist throughout the transient. A detailed discussion of the method of analysis can be found in Reference 1.

<u>Average Core Analysis</u>

The spatial-kinetics computer code, TWINKLE (Reference 3) is used for the average core transient analysis. This code solves the two-group neutron diffusion theory kinetic equation in one, two or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multi-region, transient fuel-clad-coolant heat transfer model for calculation of point-wise Doppler and moderator feedback effects. This analysis uses the code as a one-dimensional axial kinetics code since it allows a more-realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described below) for calculating the ejected rod worth and hot channel factor.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal heat flux times the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. Therefore, the assumption is made that the hot spot conditions before and after ejection are coincident. This is very conservative since the peak nuclear power after ejection will occur in or adjacent to the assembly with the ejected rod, whereas prior to ejection the power in this region will be depressed.

The average core energy addition, calculated as described above, is multiplied by the appropriate hot channel factors. The hot spot analysis uses the detailed fuel and clad transient heat transfer computer code, FACTRAN (Reference 2). This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod and the heat flux at the surface of the rod, using as input the nuclear power versus time and local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A conservative pellet radial power distribution is assumed within the fuel rod.

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandberg-Tong correlation (Reference 4) to determine the film-boiling coefficient after DNB. The Bishop-Sandberg-Tong correlation is conservatively used assuming zero bulk fluid quality. The DNB heat flux is not calculated; instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted to force the full-power, steady-state temperature distribution to agree with fuel heat transfer design codes.

Reactor Protection

The protection for this accident, as explicitly modeled in the analysis, is provided by the power range high neutron flux trip (high and low settings) and high positive rate neutron flux trip.

These protection functions are part of the reactor trip system. No single failure of the reactor trip system will negate the protection functions required for the rod ejection accident, or adversely affect the consequences of the accident.

6.2.19.4 Acceptance Criteria

Due to the extremely low probability of a rod cluster control assembly ejection accident, this event is classified as an ANS Condition IV event. As such, some fuel damage could be considered an acceptable consequence.

The Idaho Nuclear Corporation (Reference 5) has carried out comprehensive studies of the threshold of fuel failure and the threshold of significant conversion of the fuel thermal energy to mechanical energy as part of the SPERT project. Extensive tests of UO_2 zirconium-clad fuel rods representative of those present in pressurized water reactor-type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design exhibited failure as low as 225 cal/gm. These results differ significantly from the TREAT (Reference 6) results, which indicated a failure threshold of 280 cal/gm. Limited results have indicated that this threshold decreased 10 percent with fuel burnup. The clad failure mechanism appears to be melting for unirradiated (zero burnup) rods and brittle fracture for irradiated rods. The conversion ratio of thermal to mechanical energy is also important. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure (large fuel dispersal, large pressure rise), even for irradiated rods, did not occur below 300 cal/gm.

The real physical limits of this accident are that the rod ejection event and any consequential damage to either the core or the Reactor Coolant System must not prevent long-term core cooling and any offsite dose consequences must be within the guidelines of 10 CFR 100. More specific and restrictive criteria are applied to ensure fuel dispersal in the coolant, gross lattice distortion or severe shock waves will not occur. In view of the above experimental results, and the conclusions of WCAP-7588, Rev. 1-A (Reference 1) and Reference 7, the limiting criteria are:

a. Average fuel pellet enthalpy at the hot spot must be maintained below 225 cal/gm for unirradiated and 200 cal/gm for irradiated fuel (the 200 cal/gm limit is applied);

b. Peak reactor coolant pressure must be less than that which could cause RCS stresses to exceed the faulted-condition stress limits; and

c. Fuel melting is limited to less than 10 percent of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of Criterion a.

6.2.19.5 Results

A summary of the parameters used in the Rod Ejection analyses, and the analysis results, are listed in Table 6.2.19-1. For both hot full power (HFP) cases, control bank D is assumed at its insertion limit. For both hot zero power (HZP) cases, control bank D is assumed fully inserted with banks B and C at their insertion limits.

The nuclear power and hot spot fuel and clad temperature transients for the worst cases, beginning of life hot full power (BOL HFP) and end of life hot full power (EOL HFP), are shown in Figures 6.2.19-1 through 6.2.19-4. The sequence of events for the worst cases are listed in Table 6.2.19-2.

For the worst cases, peak hot spot average enthalpy is less than the acceptance criteria limit of 200 cal/gm (maximum). Peak fuel centerline temperature exceeded the conservative assumed temperature for fuel melt (4900°F at BOL; 4800°F at EOL), but predicted fuel melt is less than the acceptance criteria limit of 10% fuel pellet volume (maximum) at the hot spot.

A detailed calculation of the pressure surge for an ejected rod worth of one dollar at beginning of life, hot full power, indicates that the peak pressure does not exceed that which would cause reactor pressure vessel stress to exceed the faulted condition stress limits (Reference 1). 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 adverse effects to the RCS.

6.2.19.6 Conclusions

Despite the conservative assumptions, the analyses indicate that the described fuel and clad limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the RCS. The analyses demonstrate that the fission product release as a result of fuel rods entering DNB is limited to less than 10 percent of the fuel rods in the core.

6.2.19.7 References

1. Risher, D. H., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors using Special Kinetics Methods," WCAP-7588; Rev. 1A, January 1975

2. Hargrove, H. G., "FACTRAN, a FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod," WCAP-7908-A, December 1989

3. Risher, D. H. and Barry, R. F., Jr., "TWINKLE, a Multi-dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A, January 1975 (Proprietary) and WCAP-8028-A, January 1975 (Non-Proprietary)

4. Bishop, A. A., Sandberg, R. O. and Tong, L. S., "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," ASME 65-HT-31, August 1965

5. Taxebius, T. G., ed., "Annual Report - SPERT Project, October 1968 - September 1969," IN-1370 Idaho Nuclear Corporation, June 1970

6. Liimatainen, R. C. and Testa, F. J., "Studies in TREAT of Zircaloy 2-Clad, UO2-Core Simulated Fuel Elements," ANL-7225, P 177, November 1966

7. Letter from W. J. Johnson of Westinghouse Electric Corporation to Mr. R. C. Jones of the Nuclear Regulatory Commission, Letter Number NS-NRC-89-3466, "Use of 2700°F PCT Acceptance Limit in Non-LOCA Accidents," October 23,1989

Table 6.2.19-1
Results of the Rod Cluster Control Assembly Ejection Accident Analysis

	Beginning of Cycle HFP	Beginning of Cycle HZP	End of Cycle HFP	End of Cycle HZP
Power level, percent	102	0	102	0
Ejected rod worth, percent K	0.20	0.765	0.25	0.80
Delayed neutron fraction, percent	0.55	0.55	0.44	0.44
Feedback reactivity weighting	1.30	2.07	1.3	3.55
Trip reactivity, percent K	4.0	2.0	4.0	2.0
F_q before rod ejection	2.60	--	2.60	--
Ejected rod F_q	6.1	11.5	6.4	23.0
Number of operational pumps	4	2	4	2
Max fuel pellet average temperature, °F	4128	3123	4044	3056
Max fuel centerline temperature, °F	4968	3616	4872	3479
Max clad average temperature, °F	2434	2348	2369	2337
Max fuel stored energy, cal/gm	181	130	177	127
Fuel melt at the hot spot, percent	6.39	0	7.39	0

Table 6.2.19-2 Sequence of Events - RCCA Ejection Accident		
Case	**Event**	**Time (sec)**
BOL, full power	Initiation of Rod Ejection	0.0
	Power Range High Neutron Flux Setpoint Reached	0.05
	Peak Nuclear Power Occurs	0.13
	Rods Begin to Fall	0.55
	Peak Fuel Average Temperature Occurs	2.28
	Peak Clad Temperature Occurs	2.38
	Peak Heat Flux Occurs	2.39
EOL, full power	Initiation of Rod Ejection	0.0
	Power Range High Neutron Flux Setpoint Reached	0.04
	Peak Nuclear Power Occurs	0.13
	Rods Begin to Fall	0.54
	Peak Fuel Average Temperature Occurs	2.31
	Peak Clad Temperature Occurs	2.40
	Peak Heat Flux Occurs	2.41

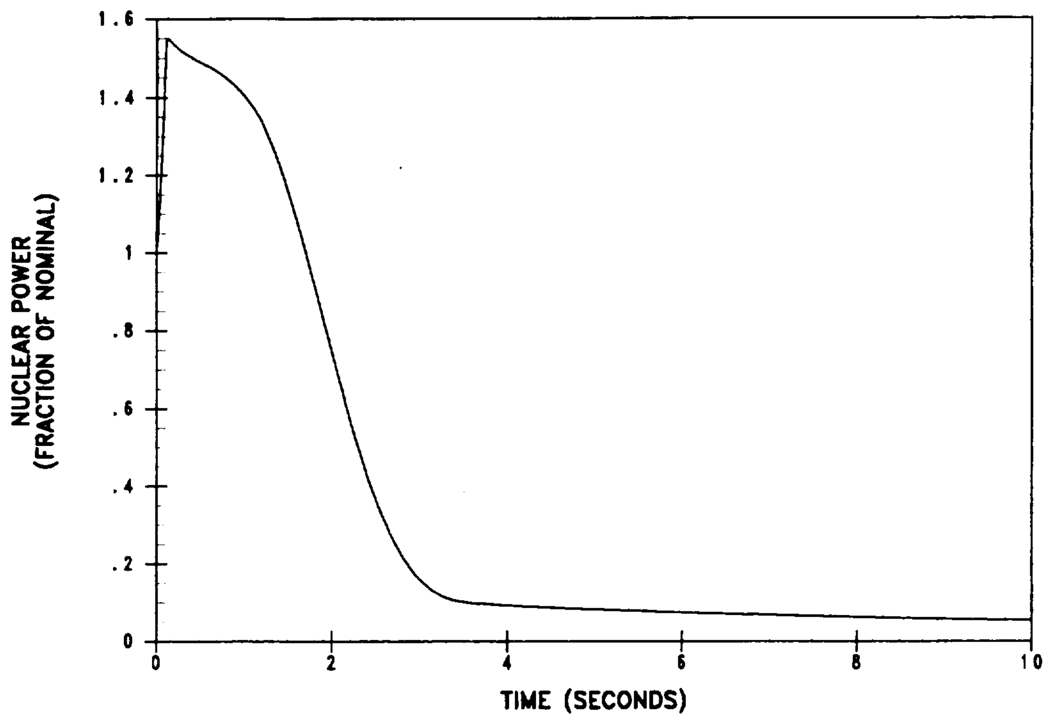


Figure 6.2.19-1

BOL HFP RCCA Ejection, Nuclear Power versus Time

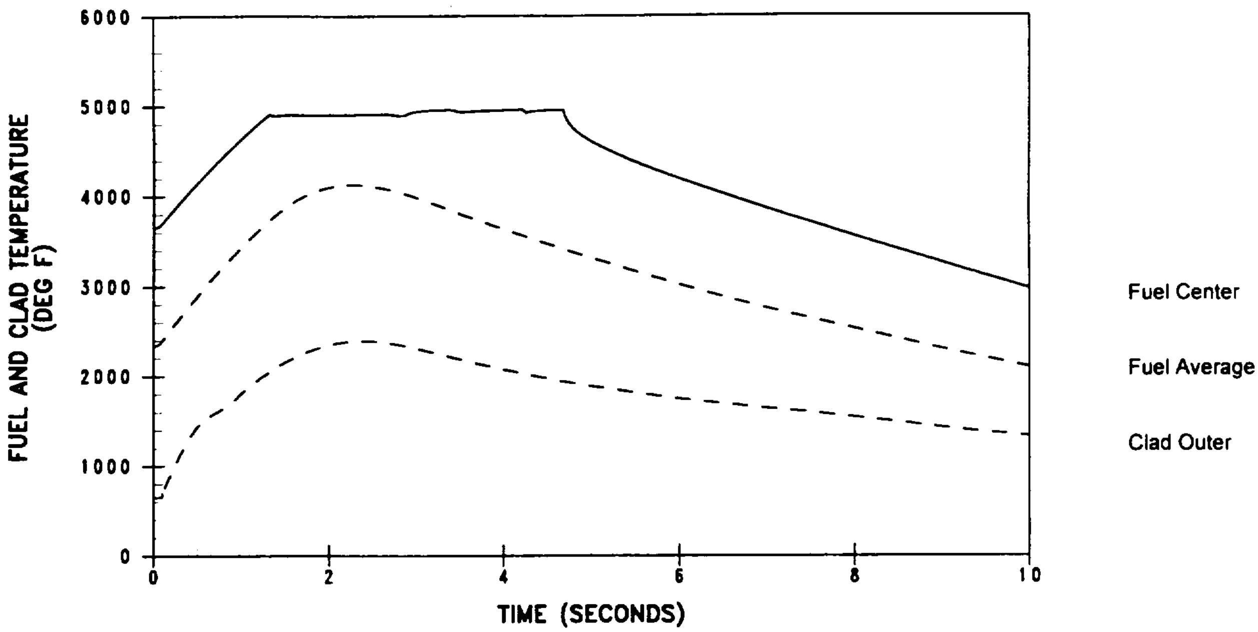


Figure 6.2.19-2

BOL HFP RCCA Ejection, Hot Spot Fuel and Clad Temperatures versus Time

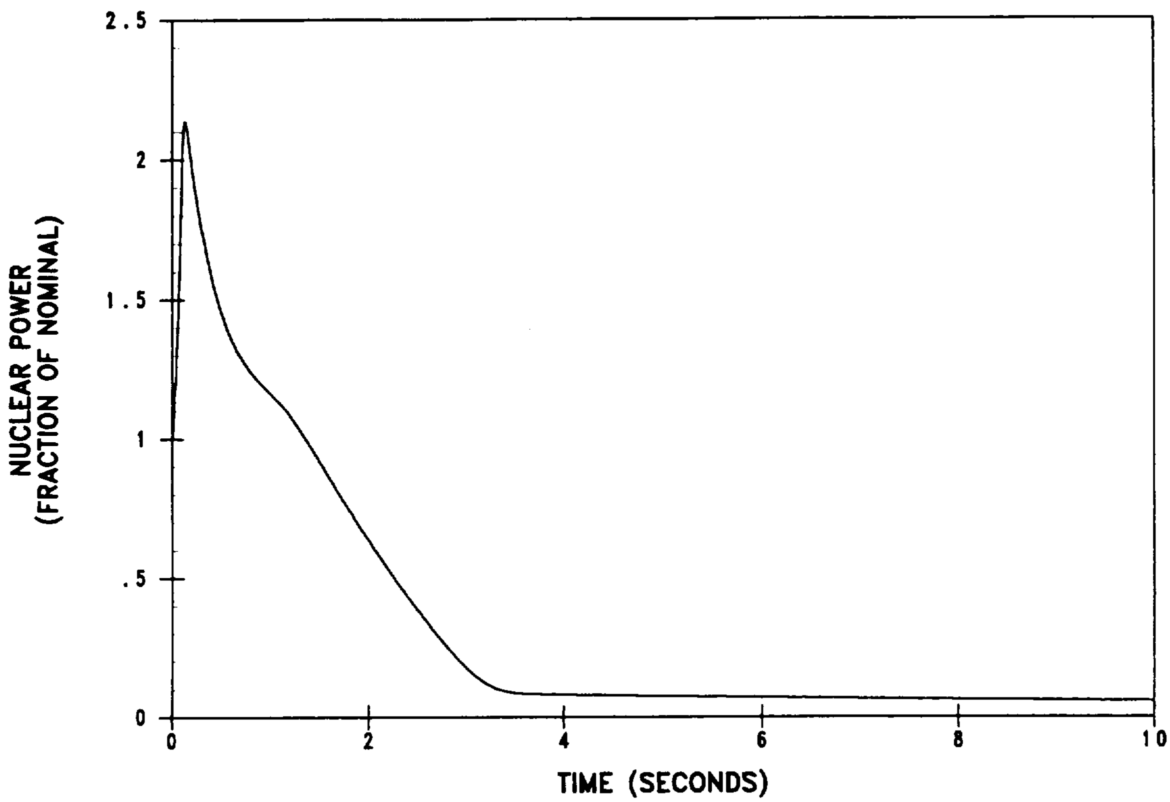


Figure 6.2.19-3

EOL HFP RCCA Ejection, Nuclear Power versus Time

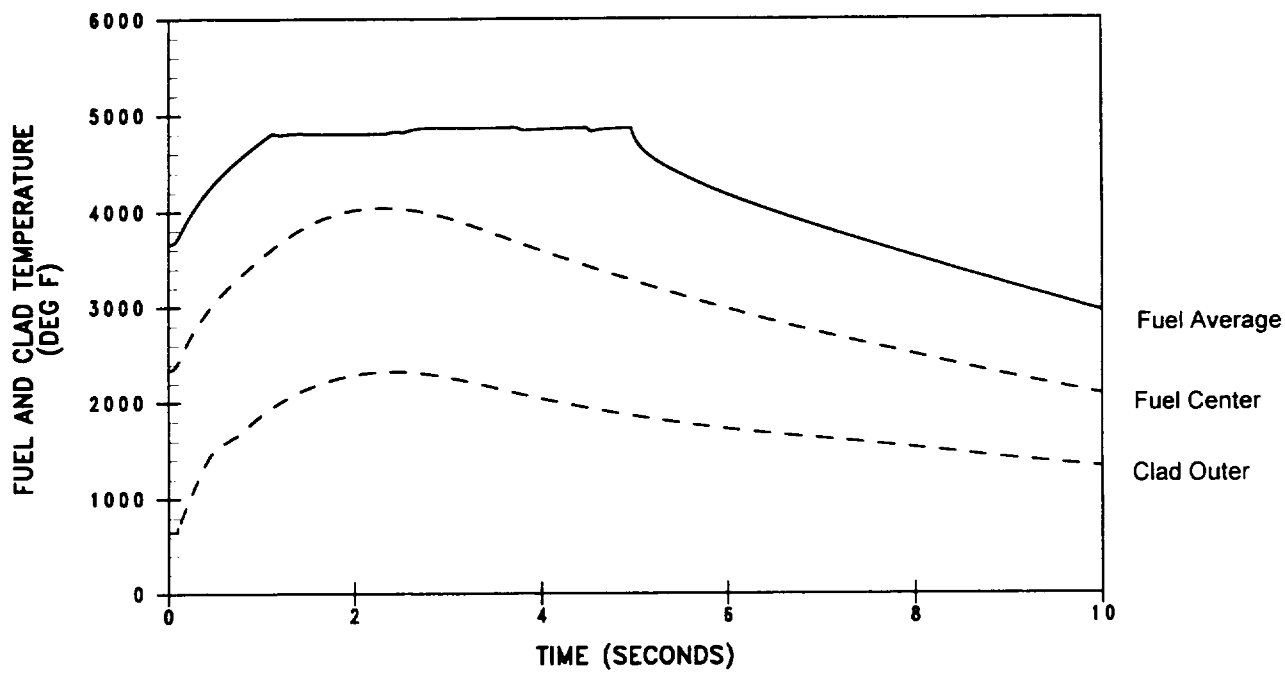


Figure 6.2.19-4

EOL HFP RCCA Ejection, Hot Spot Fuel and Clad Temperatures versus Time

6.2.20 Inadvertent Operation of the Emergency Core Cooling System (ECCS) During Power Operation

6.2.20.1 Identification of Causes and Accident Description

Inadvertent operation of the Emergency Core Cooling System (ECCS) at power could be caused by operator error, test sequence error, or a false electrical actuation signal. A spurious signal initiated after the logic circuitry in one solid-state protection system train for any of the following Engineered Safety Feature (ESF) functions could cause this incident by actuating the ESF equipment associated with the affected train.

a. High containment pressure
b. Low pressurizer pressure
a. Low steamline pressure

Following the actuation signal, the suction of the charging pumps diverts from the volume control tank to the refueling water storage tank (RWST). Simultaneously, the valves isolating the charging pumps from the injection header automatically open and the normal charging line isolation valves close. The charging pumps force borated water from the RWST through the pump discharge header and injection line into the cold leg of each loop. The passive accumulator tank safety injection and low head safety injection systems are available, but do not provide flow when the reactor coolant system (RCS) is at normal pressure.

A safety injection (SI) signal normally results in a direct reactor trip and turbine trip. However, any single fault that actuates the ECCS will not necessarily produce a reactor trip. If an SI signal generates a reactor trip, the operator should determine if the signal is spurious. If it is, the operator should terminate SI and maintain the plant in hot-standby as determined by appropriate recovery procedures. If repair of the ESF actuation system instrumentation is necessary, future plant operation will be in accordance with the Technical Specifications. If the SI results in discharge of coolant through the pressurizer safety relief valves, the operators will bring the plant to cold shutdown in order to inspect the valves.

If the reactor protection system does not produce an immediate trip as a result of the spurious SI signal, the reactor experiences a negative reactivity excursion due to the injected boron, which causes a decrease in reactor power. The power mismatch causes a drop in T_{avg} and consequent coolant shrinkage. The pressurizer pressure and water level decrease. Load decreases, due to the effect of reduced steam pressure on load, after the turbine throttle valve is fully open. If automatic rod control is used, these effects will lessen until the rods have moved fully out of the core. The transient is eventually terminated by the reactor protection system low pressurizer pressure trip or by manual trip.

The time to trip is affected by initial operating conditions. These initial conditions include the core burnup history that affects initial boron concentration, rate of change in boron concentration, and Doppler and moderator coefficients.

6.2.20.2 Analysis of Effects and Consequences

Method of Analysis

Inadvertent operation of the ECCS is analyzed using the LOFTRAN computer code (Reference 1). The code simulates the core neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, the feedwater system, steam generators, and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures, and power level.

Inadvertent operation of the ECCS at power is classified as a Condition II event, a fault of moderate frequency. The criteria established for Condition II events include the following:

a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values,

b. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the safety analysis limit value derived at a 95% confidence level and 95% probability, and,

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

The inadvertent ECCS actuation at power event is analyzed to determine the maximum RCS pressure encountered throughout the accident. The most limiting case with respect to RCS pressure is an SI at Hot Full Power coincident with a reactor trip. Because of the pressure reduction from the reactor trip, the SI flow is maximized. The SI flow refills the pressurizer until the pressurizer is water solid, and the SI flow results in liquid discharge through the pressurizer safety relief valves. The shut off head of the SI system is less than 110% of the RCS design pressure.

If the Pressurizer Safety relief valves do not reseat, then the transient will proceed and terminate as described in Section 6.2.21, "Inadvertent Opening of a Pressurizer Safety or Relief Valve." This event is also classified as an event of moderate frequency.

The inadvertent ECCS actuation at power event is also analyzed to determine the minimum DNBR value encountered throughout the accident. The most limiting case with respect to DNBR is a minimum reactivity feedback condition with the plant assumed to be in manual rod control. Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits and the minimum calculated DNBR increases throughout the event. Since the minimum DNBR occurs at time zero, the minimum DNBR is identical for both Units 1 and 2. The Unit 1 results presented herein are representative of both units.

The analysis assumptions for the DNBR case are as follows:

a. Initial Operating Conditions

The DNB case is analyzed with the revised thermal design procedure as described in WCAP-11397-P-A (Reference 2). Initial reactor power, RCS pressure, and RCS temperature are assumed to be at their nominal full power values. Uncertainties in initial conditions are included in the DNBR limit as described in Reference 2.

b. Moderator and Doppler Coefficients of Reactivity

A moderator temperature coefficient (MTC) of 0.0 pcm/°F and a low absolute value Doppler power coefficient was assumed for the DNBR cases. A zero MTC is more conservative than a positive MTC (PMTC) at beginning-of-life (BOL) for events that result in a temperature decrease, since the temperature decrease would result in a power decrease with a PMTC. Therefore, the use of a zero MTC conservatively maintains power at the highest possible level until reactor trip.

c. Reactor Control

For the DNB case (without direct reactor trip on SI), it is conservative to assume that the reactor is in manual rod control. If the reactor were in automatic rod control, the control rod banks would move prior to trip and reduce the severity of the transient.

d. Pressurizer Pressure Control

Pressurizer heaters are assumed to be inoperable. This assumption yields a higher rate of pressure decrease, which is conservative with respect to DNBR. Pressurizer spray and PORVs are assumed available to minimize RCS pressure.

e. Boron Injection

At the initiation of the event, two charging pumps inject borated water into the cold leg of each loop. The analysis assumes zero injection line purge volume for calculational simplicity; thus, the boration transient begins immediately in the analysis.

f. ECCS Flow

Maximum SI flow via two High Head Safety Injection (HHSI) charging pumps provides borated water to the RCS. The positive displacement charging pump is assumed to be inoperable at event initiation.

g. Turbine Load

For the DNB case (without direct reactor trip/turbine trip on SI), the turbine load remains constant until the governor drives the throttle valve wide open. After the throttle valve is full open, turbine load decreases as steam pressure drops.

h. Reactor Trip

Reactor trip is initiated by a low pressurizer pressure signal for the DNB case.

i. Decay Heat

Full decay heat is modeled for this event but has no impact on the DNB case (i.e., minimum DNBR occurs prior to reactor trip).

j. Pressurizer Safety Valves

The safety valve setpoints do not impact the DNB case since the PORVs are assumed available to maintain low RCS pressure, which is conservative with respect to DNBR.

k. Auxiliary Feedwater

Auxiliary Feedwater is not credited for the DNB case.

l. Main Steam Safety Valves

The main steam safety valves (MSSVs) are conservatively assumed to open at +5% above their nominal set pressure. No credit for steam dump is assumed in this analysis.

6.2.20.3 Results

The transient responses for the DNB case are shown in Figures 6.2.20-1 through 6.2.20-3. Table 6.2.20-1 shows the calculated sequence of events.

Nuclear power starts decreasing immediately due to boron injection, but steam flow does not decrease until later in the transient when the turbine throttle valve is wide open. The mismatch between load and nuclear power causes T_{avg}, pressurizer water level, and pressurizer pressure to drop. The reactor trips and control rods start moving into the core when the pressurizer pressure reaches the pressurizer low pressure trip setpoint. The DNBR increases throughout the transient.

6.2.20.4 Conclusions

Results of the analysis show that spurious ECCS operation without immediate reactor trip does not present any hazard to the integrity of the RCS with respect to DNBR. The minimum DNBR never falls below the initial value. If the reactor does not trip immediately, the low pressurizer pressure reactor trip will provide protection. This trips the turbine and prevents excess cooldown, which expedites recovery from the incident.

6.2.20.5 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-proprietary), April 1984.

2. Friedland, A. J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary), April 1989.

Table 6.2.20-1 Sequence of Events-Inadvertent ECCS at Power Event		
Case	Event	Time (Sec)
1. DNBR	Spurious SI signal generated; two charging pumps begin injecting borated water	0.0
	Turbine throttle valve wide open, load begins to drop with steam pressure	49.5
	Low pressurizer pressure reactor trip setpoint reached	76.1
	Control Rod motion begins	78.1
	Minimum DNBR occurs	(*)

* Never falls below initial value

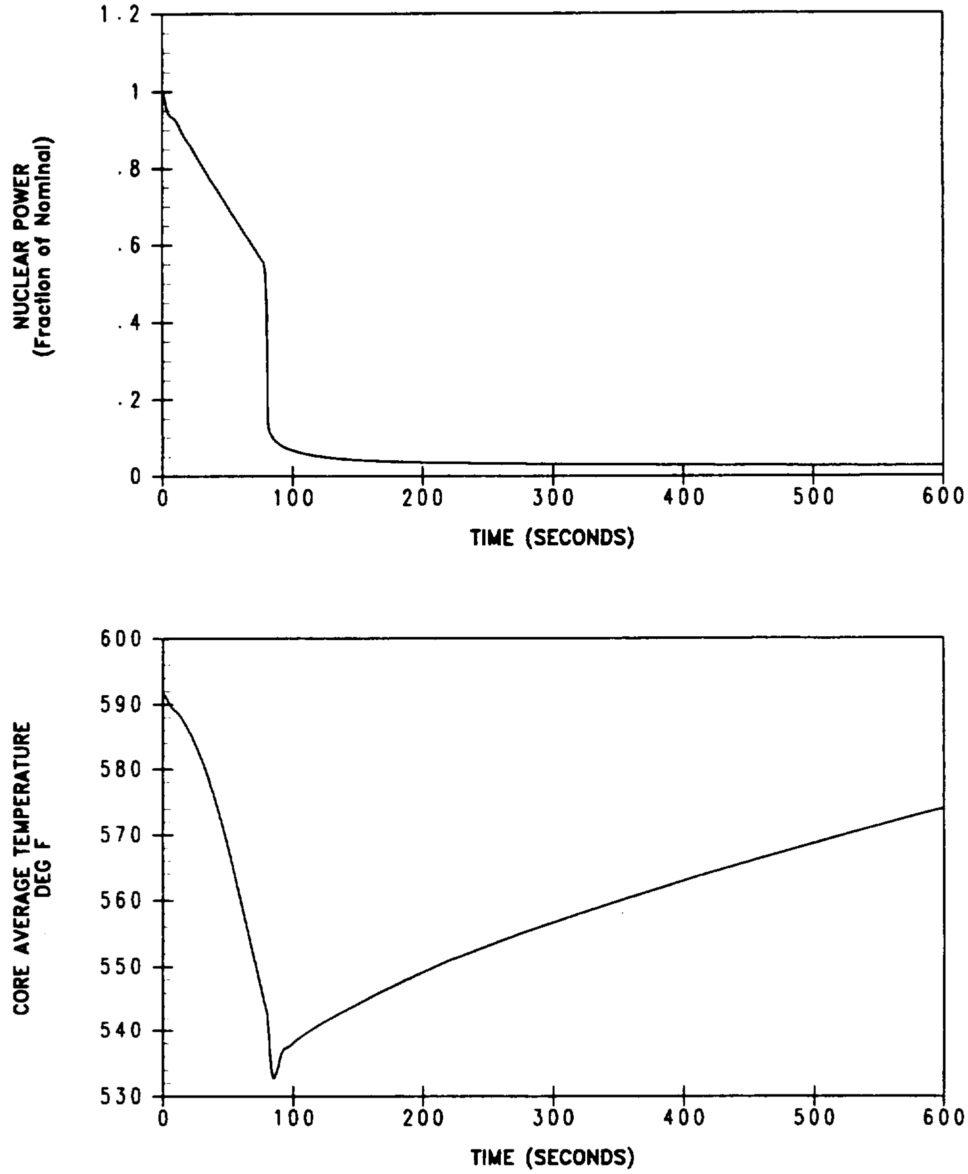


Figure 6.2.20-1

Inadvertent Operation of the ECCS During Power Operation

Nuclear Power and Core Average Coolant Temperature versus Time

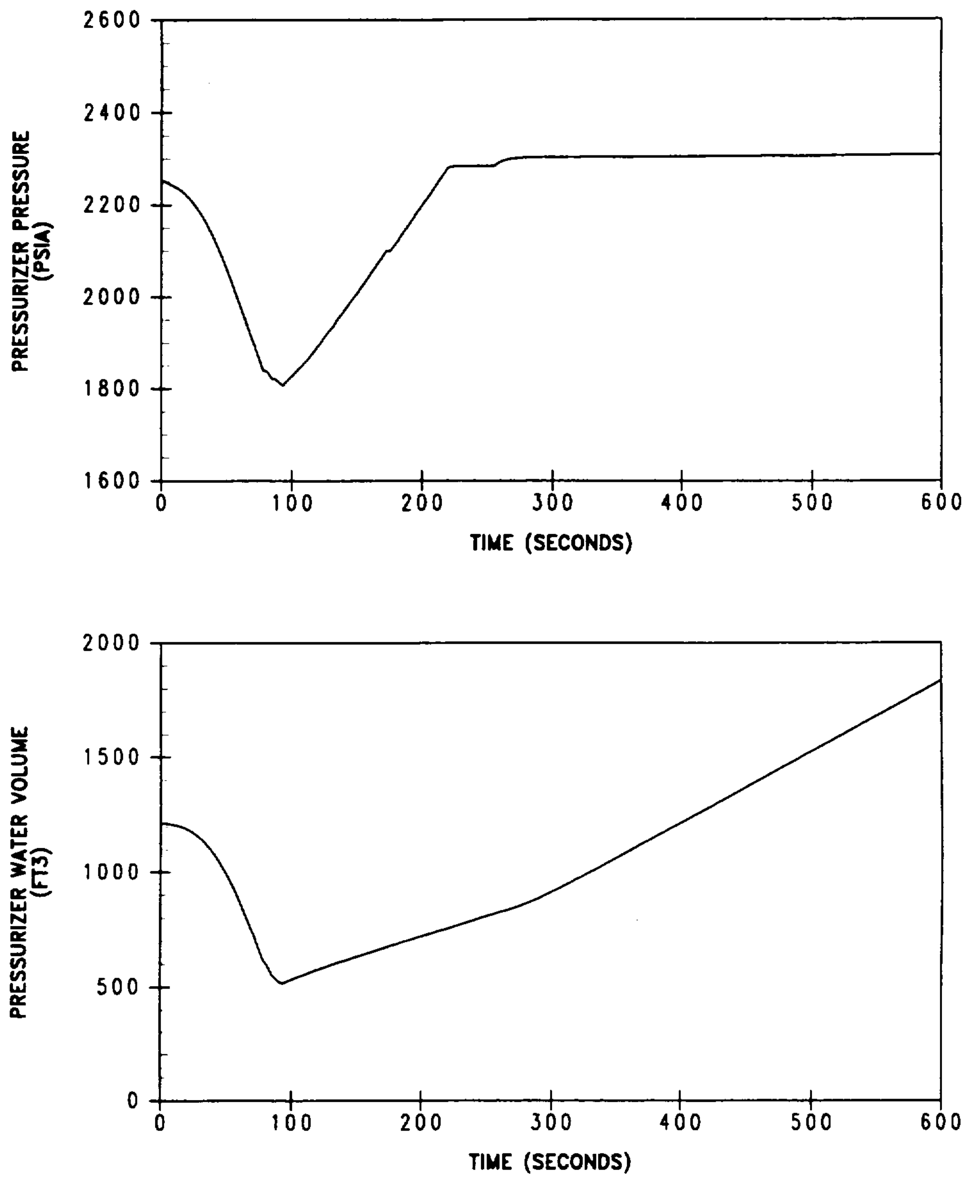


Figure 6.2.20-2

Inadvertent Operation of the ECCS During Power Operation

Pressurizer Pressure and Pressurizer Water Volume versus Time

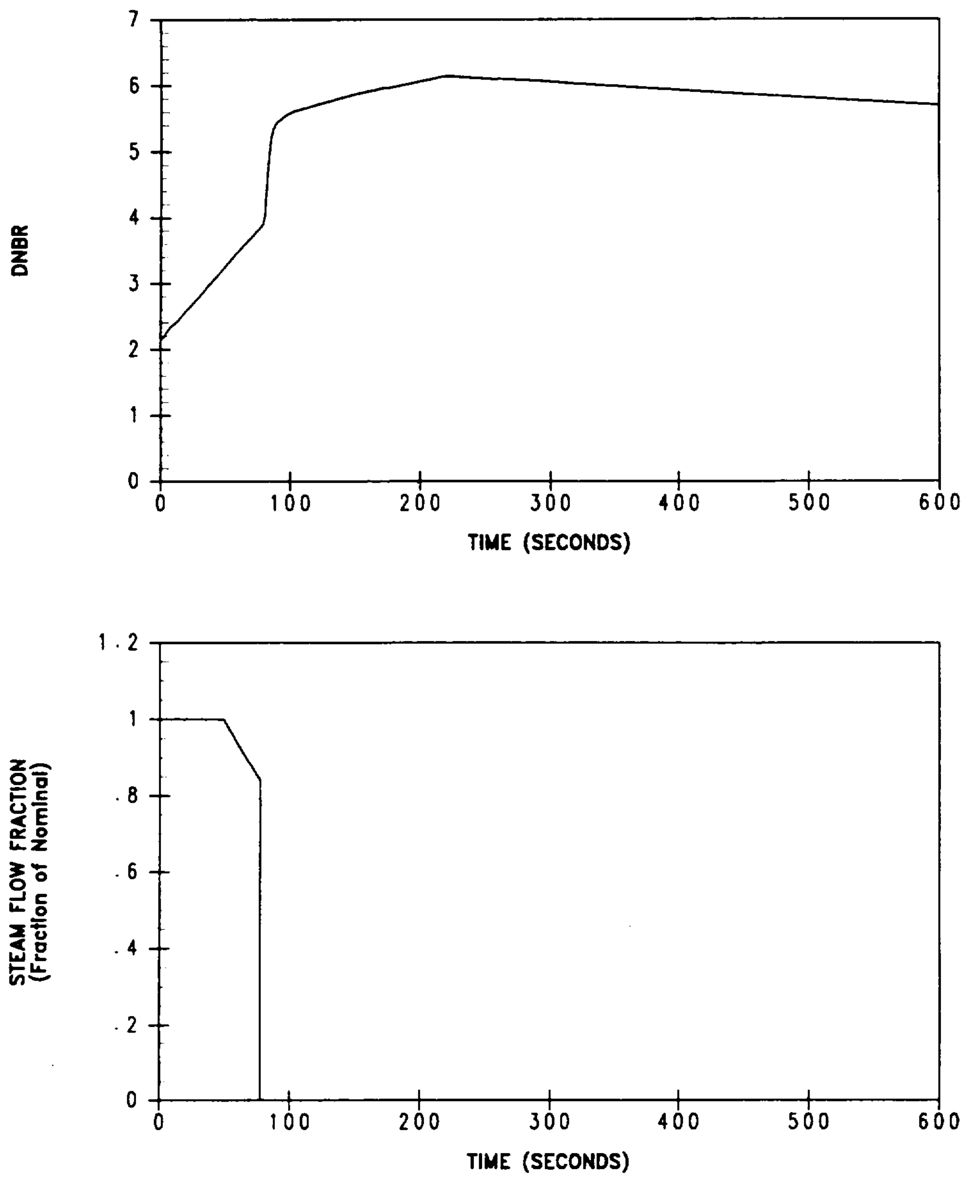


Figure 6.2.20-3

Inadvertent Operation of the ECCS During Power Operation

DNBR and Steam Flow versus Time

6.2.21 Accidental Depressurization of the RCS

6.2.21.1 Introduction

Accidental depressurization of the reactor coolant system (RCS) could occur due to inadvertent opening of a pressurizer relief or safety valve. Since a pressurizer safety valve is sized to relieve approximately twice the steam flowrate of a relief valve, an open safety valve will allow a much more rapid depressurization of the RCS and result in the most-severe core conditions resulting from an accidental depressurization event.

Initially, the event results in a rapidly decreasing RCS pressure, which could reach hot leg saturation conditions without reactor protection system intervention. If saturation conditions are reached, the rate of depressurization is slowed considerably but pressure continues to decrease throughout the event. The effect of the pressure decrease is to increase power via moderator density feedback. However, if the plant is in automatic mode, the rod control system functions to maintain power essentially constant throughout the initial stages of the transient. Average coolant temperature remains approximately the same, but pressurizer level increases until reactor trip because of the decreased reactor coolant density.

The reactor may be tripped by the following reactor protection system signals.

a. Pressurizer low pressure

b. Overtemperature ΔT

6.2.21.2 Input Parameters and Assumptions

To produce conservative results in calculating the DNBR during the transient, the following assumptions are made.

a. Initial reactor power, pressure, and RCS temperatures (consistent with the uprated power conditions) are assumed to be at their nominal values. Uncertainties in initial conditions are included in the DNBR limit as described in WCAP-11397 (Reference 1).

b. A positive moderator temperature coefficient of reactivity (+7 pcm/°F) is assumed. The spatial effect of voids due to local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape.

c. A least negative Doppler-only power coefficient is assumed such that the resultant amount of negative feedback is conservatively low. This maximizes any power increase due to moderator reactivity feedback.

d. Cases assuming both D5 and BWI steam generator models at both minimum and maximum tube plugging levels were analyzed.

6.2.21.3 Description of Analysis

The purpose of this analysis is to demonstrate the manner in which the protection functions described above actuate during RCS depressurization caused by inadvertent opening of a pressurizer safety valve.

The accident is analyzed using the detailed digital computer code LOFTRAN (Reference 2). This code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

6.2.21.4 Acceptance Criteria

Based on its frequency of occurrence, accidental depressurization of the RCS is considered a Condition II event as defined by the American Nuclear Society. The following items summarize the acceptance criteria associated with this event.

Critical heat flux should not be exceeded. This is ensured by demonstrating that the minimum DNBR does not go below the limit at any time during the transient.

Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures. These criteria are not challenged following the subject depressurization of the RCS.

The protection features described in Section 6.2.21.1 mitigate the effects of the accidental depressurization of the RCS transient such that the above criteria are satisfied.

6.2.21.5 Results

The most limiting case (BWI steam generators at maximum tube plugging level) for an inadvertent opening of a pressurizer safety valve is shown in Figures 6.2.21-1 and 6.2.21-2. Figure 6.2.21-1 illustrates the nuclear power and DNBR transients following the depressurization. Nuclear power increases slowly until reactor trip occurs on OTΔT. The DNBR decreases initially, but then increases rapidly following reactor trip. The DNBR remains above the safety analysis limit value throughout the transient. Core average temperature, pressure decay, and pressurizer water volume transients following the accident are presented in Figure 6.2.21-2. The calculated sequence of events is shown in Table 6.2.21-1.

6.2.21.6 Conclusions

The results of the analysis show that the pressurizer low pressure and OTΔT reactor protection system signals provide adequate protection against the RCS depressurization event. Thus, there will be no cladding damage nor release of fission products into the RCS.

6.2.21.7 References

1. A. J. Friedland and S. Ray, "Revised Thermal Design Procedure," WCAP-11397-P-A, April 1989

2. T. W. T. Burnett, et al., "LOFTRAN Code Description," WCAP-7907-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984

Table 6.2.21-1 Sequence of Events - Accidental Depressurization of RCS Event	
Event	**Time (sec)**
Inadvertent opening of one RCS safety valve	0.0
Overtemperature ΔT reactor trip setpoint reached	24.2
Rods begin to drop	32.2
Minimum DNBR occurs	32.8

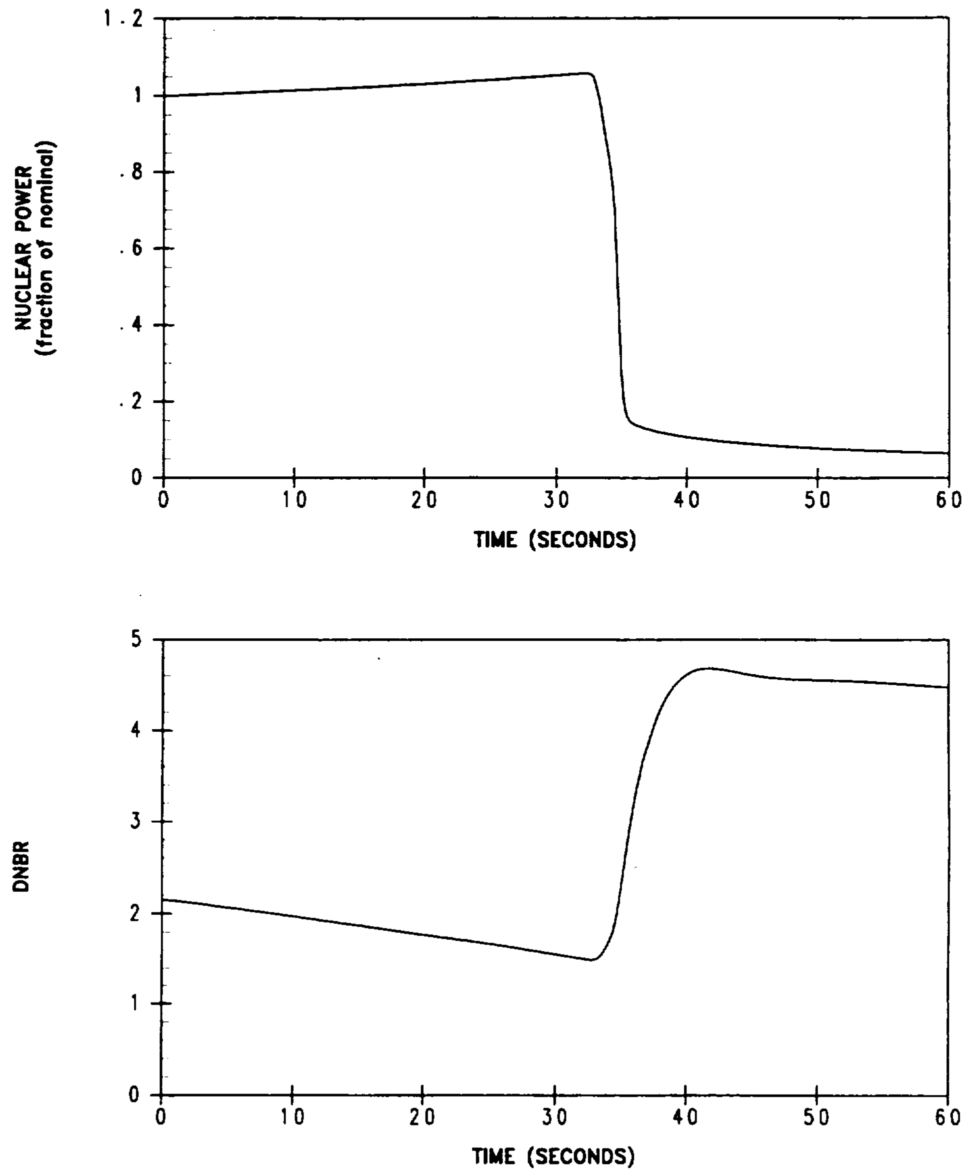


Figure 6.2.21-1

Accidental Depressurization of the RCS, Nuclear Power and DNBR versus Time

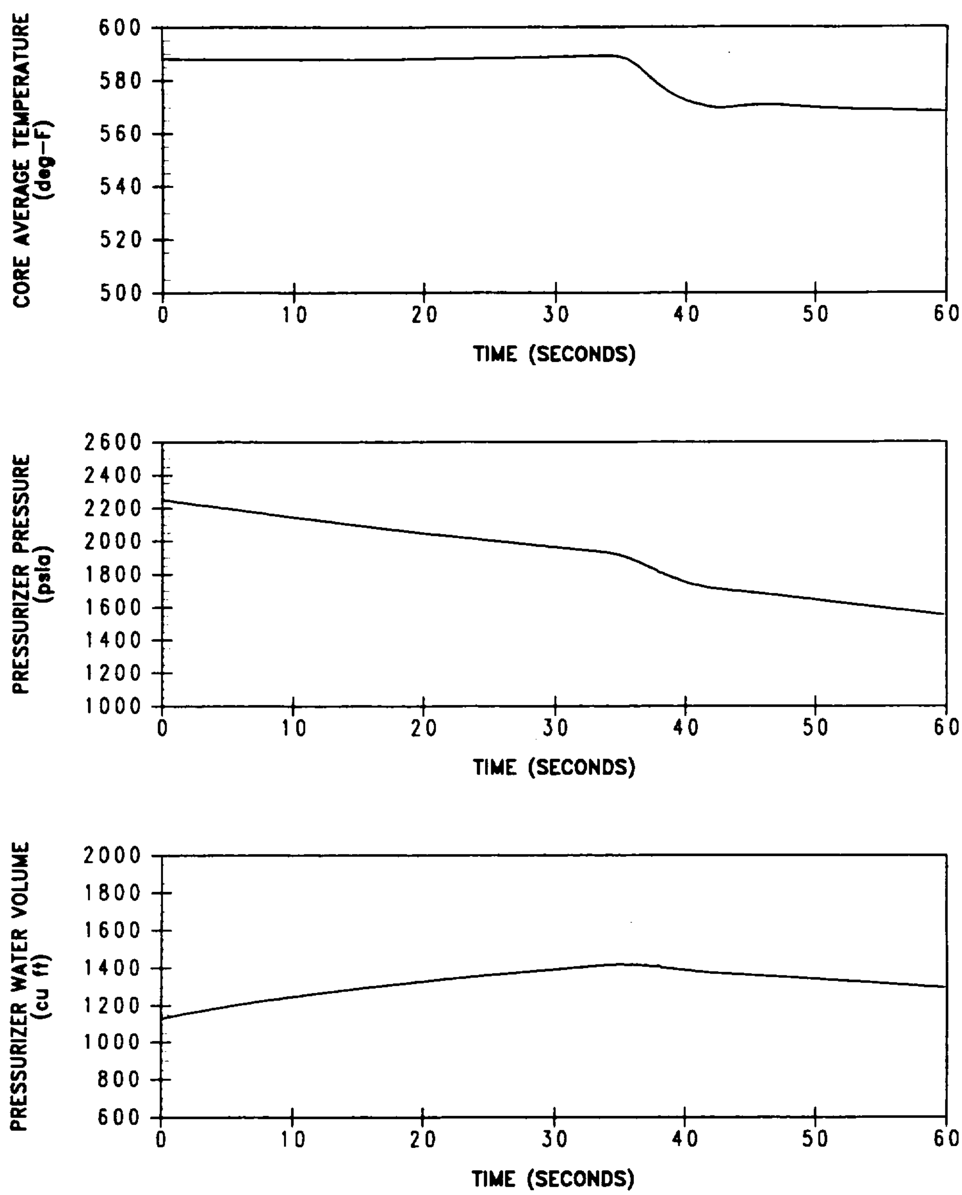


Figure 6.2.21-2

Accidental Depressurization of the RCS, T_{AVG}, Pressurizer Pressure, and Pressurizer Water Volume versus Time

6.3 Steam Generator Tube Rupture Transient

6.3.1 Introduction

In support of the Byron and Braidwood, Units 1 and 2, power uprating program, a Steam Generator Tube Rupture (SGTR) transient analysis has been performed by ComEd per approved methodology (References 1, 2, and 3) to evaluate the two major potential consequences of concern, specifically:

- margin-to-overfill (MTO) case - the potential for overfilling the faulted steam generator before the Auxiliary Feed Water (AFW) can be isolated and the break flow terminated.
- offsite dose case - the potential to release primary system activity through the secondary side in excess of 10 CFR 100 limits. The input parameters specific to the offsite dose case will be discussed in Section 6.7.7.

Two separate analyses were performed, one for Byron and Braidwood Unit 1, and another for Byron and Braidwood Unit 2. This was done because the steam generators have significant design differences.

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at steady-state reactor power with the reactor coolant system activity at the allowable technical specification operating limit. The accident leads to an increase in secondary system activity due to the leakage of radioactive primary system water into the ruptured steam generator. In the event of a coincident loss of offsite power and subsequent failure of the steam dump system, there is a discharge of activity to the atmosphere through the steam generator safety and/or Power-Operated Relief Valves (PORVs).

The response of the operator and his ability to implement recovery actions is critical in mitigating the consequence of an SGTR event. The operator actions assumed in these two analyses are based upon the plant-specific emergency procedures addressing the SGTR accident. The emergency procedures are based upon the latest revision of the Westinghouse Owners Group generic emergency response guidelines and are, therefore, consistent with the latest guidelines regarding SGTR mitigation. The two critical analysis assumption time intervals are 11 minutes to isolate the ruptured steam generator and 31 minutes to complete all other actions required to terminate the break flow and the steam release from the ruptured steam generator. These assumed time intervals are consistent with the current analysis.

6.3.2 Input Parameters and Assumptions

The following analysis assumptions and input parameters were used for both the offsite dose and margin-to-overfill transient cases.

- Loss of offsite power (LOOP) is assumed to occur concurrent with the reactor trip.
- No credit is taken for normal chemical and volume control system charging or letdown flow prior to operator initiation
- The most limiting single failure for the margin to overfill cases is the failure of an atmospheric dump valve to open in one of the intact steam generators.
- Prior to the reactor trip, the SG level control system maintains the faulted steam generator water level essentially constant, compensating for the additional break flow by reducing the feed flow.
- The operator throttles the AFW flow to the intact SGs as necessary to maintain adequate narrow range level.
- During the RCS depressurization, the operator ensures the pressurizer level stays within the limits specified in the E-3 procedure.
- NSSS Power = 3672 MWt (3600.6 MWt with 2% uncertainty)
- Reactor Coolant System (RCS) Average Temperature = 567.4 °F (nominal average temperature of 575°F with 7.6°F uncertainty)
- Thermal Design Flow = 368,000 gpm
- Steam Generator Tube Plugging = 5% for Unit 1

  10% for Unit 2
- Operator response times consistent with current analysis

- Turbine runback to 70% power for the low pressurizer pressure reactor trip case and 95% power for the overtemperature Delta T reactor trip case is used (for MTO only).

- The following initial Steam Generator Masses are used (consistent with power levels above and include mass and level uncertainties):

Case	Unit 1	Unit 2
MTO OTΔT Reactor Trip Case	130,125 lbm	107,312 lbm
MTO Low Pressurizer Pressure Reactor Trip Case	136,611 lbm	125,409 lbm
Offsite Dose Case	110,669 lbm	70,303 lbm

6.3.3 Description of Analyses

RETRAN-02, a thermal hydraulic analysis program received from the Electric Power Research Institute, is used to perform the margin to overfill analysis. Two base cases are analyzed for the margin-to-overfill case. The first case models reactor trip on overtemperature Delta T and the second case models reactor trip on low pressurizer pressure. An additional case models the more limiting of the first two cases while assuming offsite power is available. This case is used to confirm that LOOP conditions remain limiting.

For the tube rupture event, the overtemperature Delta T reactor trip setpoint is reached earlier than the low pressurizer pressure reactor trip setpoint. Since the time from tube rupture to auxiliary feedwater isolation is an assumed operator action time, an early reactor trip results in auxiliary feedwater going to the ruptured steam generator for a longer period of time which leads to a reduction in margin-to-overfill. However, the amount of turbine runback is less (and the initial steam generator mass is smaller) if an overtemperature Delta T reactor trip is assumed instead of a low pressurizer pressure reactor trip.

Prior to power uprating, the feedwater isolation logic is being changed to require a reactor trip and low Tavg for actuation. The feedwater isolation logic associated with an safety injection signal remains unchanged. The base MTO cases assume LOOP coincident with reactor trip. Since the feedwater pumps trip under LOOP conditions, feedwater flow is terminated 6 seconds after reactor trip. These conditions will never generate a feedwater isolation based on reactor trip and low Tave. For the case of offsite power available, the feedwater and reactor coolant pumps continue to run after reactor trip. For this case, feedwater pumps continue to run until the feedwater isolation signal is reached on reactor trip coincident with low Tavg or on an safety injection signal. Therefore, an additional case which assumes offsite power available is performed for the MTO cases.

6.3.4 Acceptance Criteria

There is available margin to steam generator overfill for Byron and Braidwood in the event of a steam generator tube rupture.

6.3.5 Results

The results of the offsite dose cases are discussed in Section 6.7.7.

For Unit 1, the reactor trip on overtemperature Delta T assuming LOOP conditions is the limiting case. At the termination of the transient, there were 22 cubic feet of margin to overfill and a total liquid plus steam mass in the ruptured steam generator of 237,260 lbm. The sequence of events for the margin-to-overfill cases for Unit 1 are shown in Table 6.3-1. For Unit 2, the reactor trip on low pressurizer pressure assuming LOOP conditions is the limiting case. At the termination of the transient, there were 100 cubic feet of margin to overfill and a total liquid plus steam mass in the ruptured steam generator of 267,072 lbm. The sequence of events for the margin-to-overfill cases for Unit 2 are shown in Table 6.3-2. Figures 6.3-1 and 6.3-2 provide the ruptured steam generator water mass as a function of time for Units 1 and 2 respectively. Figures 6.3-3 and 6.3-4 provide the ruptured tube flow as a function of time for Units 1 and 2 respectively. Figures 6.3-5 and 6.3-6 provide the comparison of pressurizer pressure to ruptured steam generator pressure as a function of time for Units 1 and 2 respectively.

6.3.6 Conclusions

The SGTR transient analysis has been completed in support of the Byron and Braidwood power uprating program. Results show that a steam generator tube rupture will cause no subsequent damage to the reactor coolant system or the reactor core. An orderly recovery from the accident can be completed before steam generator overfill occurs even assuming simultaneous loss of offsite power. The changes being made to the plant associated with the power uprating program do not increase the probability or create the possibility of a SGTR transient with steam generator overfill.

6.3.7 References

1. "Revised Steam Generator Tube Rupture Analysis – Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 (TAC Nos. M97315, M97316, M97317, and M97318)," NRC Letter, George F. Dick, Jr. to Oliver D. Kingsley, dated January 28, 1998.

2. "Revised Steam Generator Tube Rupture Analysis – Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 (TAC Nos. M97315, M97316, M97317, and M97318)," NRC Letter, George F. Dick, Jr. to Oliver D. Kingsley, dated March 11, 1998.

3. "Revised Steam Generator Tube Rupture Analysis – Byron Unit 2, and Braidwood Unit 2 (TAC Nos. M97316 and M97318)," NRC Letter, Stewart N. Bailey to Oliver D. Kingsley, dated May 25, 1999.

Table 6.3-1 - Sequence of Events – Unit 1 MTO Cases

System Response/Operator Action	low pzr press. Trip Time (sec)	OTΔT trip Time (sec)
SG Tube Rupture Occurs	0	0
Reactor Trip	466	270
Auxiliary FW Injection	481	270
Safety Injection	481	572
Ruptured SG AFW Isolated	660	660
RCS Cooldown Initiated	1740	1740
RCS Cooldown Terminated	2748	2754
RCS Depressurization Initiated	2868	2874
RCS Depressurization Terminated	2989	2996
ECCS Flow Terminated	3110	3117
70 GPM Charging Flow Established	3230	3237
RCS Letdown Established	3410	3417
Reopen Pressurizer PORV	3650	3658
Reclose Pressurizer PORV	3685	3691

Table 6.3-2 - Sequence of Events – Unit 2 MTO Cases

System Response/Operator Action	low pzr press. Trip Time (sec)	OTΔT trip Time (sec)
SG Tube Rupture Occurs	0	0
Reactor Trip	248	214
Auxiliary FW Injection	257	214
Safety Injection	257	228
Ruptured SG AFW Isolated	660	660
RCS Cooldown Initiated	1740	1740
RCS Cooldown Terminated	2687	2648
RCS Depressurization Initiated	2807	2768
RCS Depressurization Terminated	2904	2867
ECCS Flow Terminated	3025	2988
70 GPM Charging Flow Established	3145	3108
RCS Letdown Established	3325	3288
Reopen Pressurizer PORV	3565	3529
Reclose Pressurizer PORV	3598	3562

Figure 6.3-1 – Ruptured Steam Generator Water Mass – Unit 1

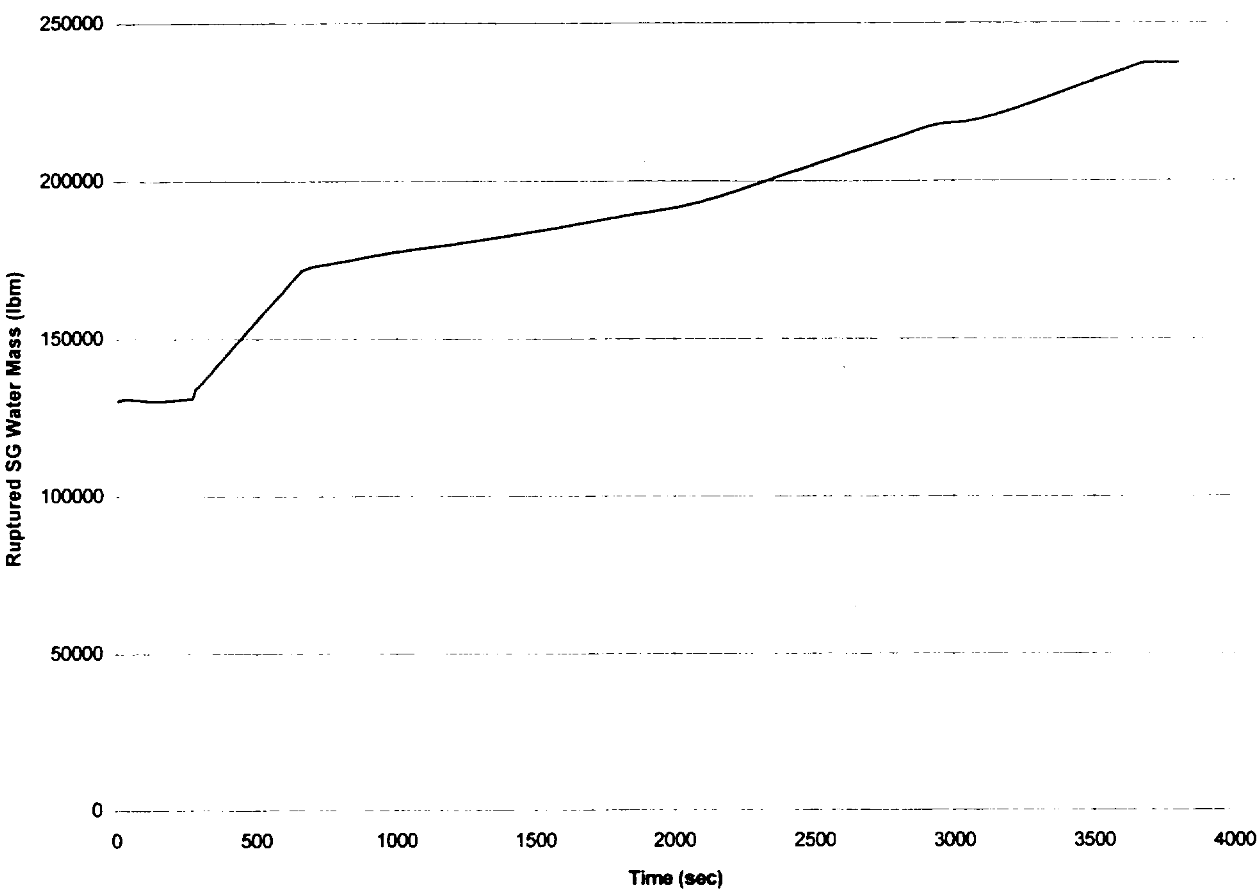


Figure 6.3-2 – Ruptured Steam Generator Water Mass – Unit 2

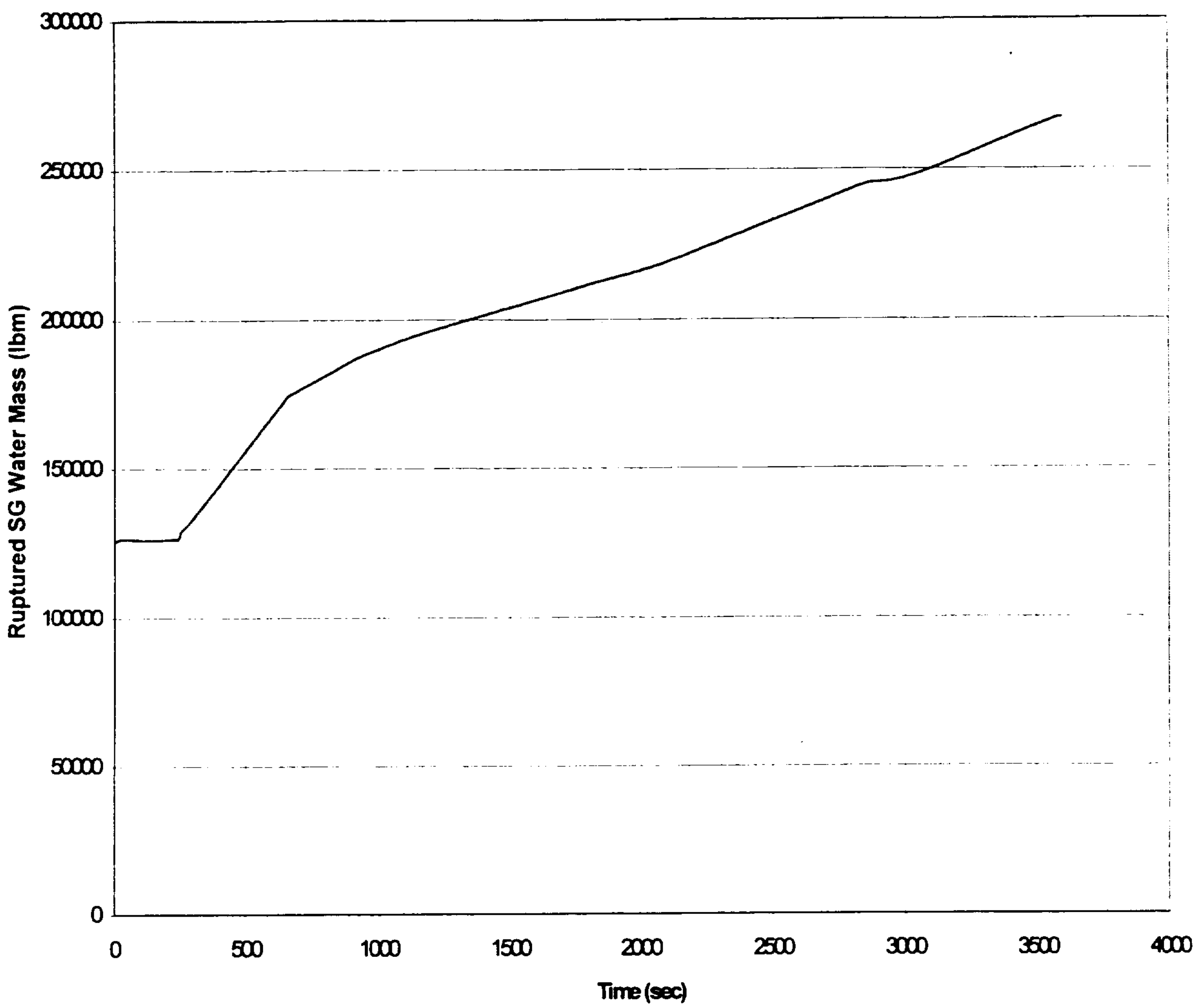


Figure 6.3-3 – Ruptured Tube Flow – Unit 1

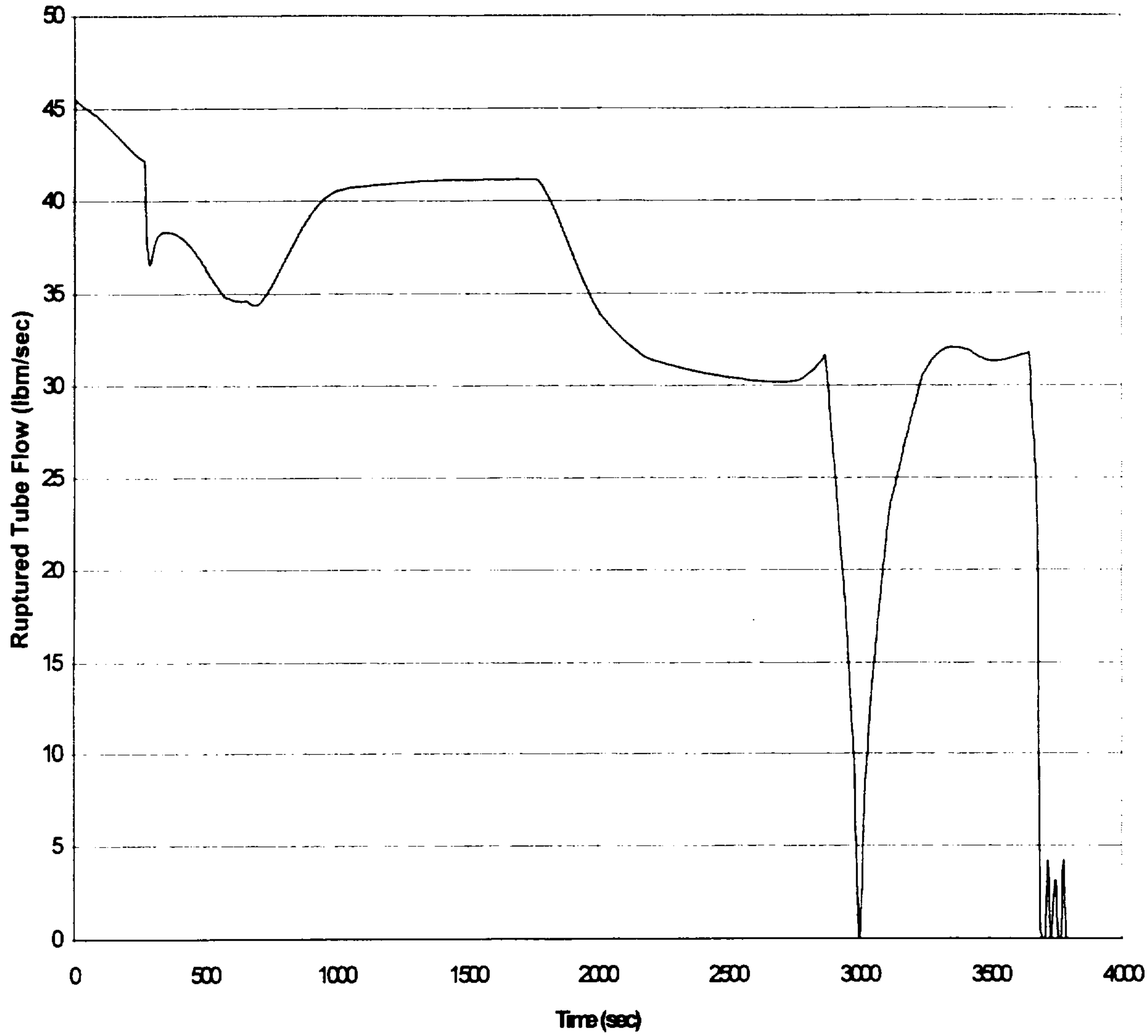


Figure 6.3-4 – Ruptured Tube Flow – Unit 2

Figure 6.3-5 – Pressurizer / Ruptured Steam Generator Pressure Comparison – Unit 1

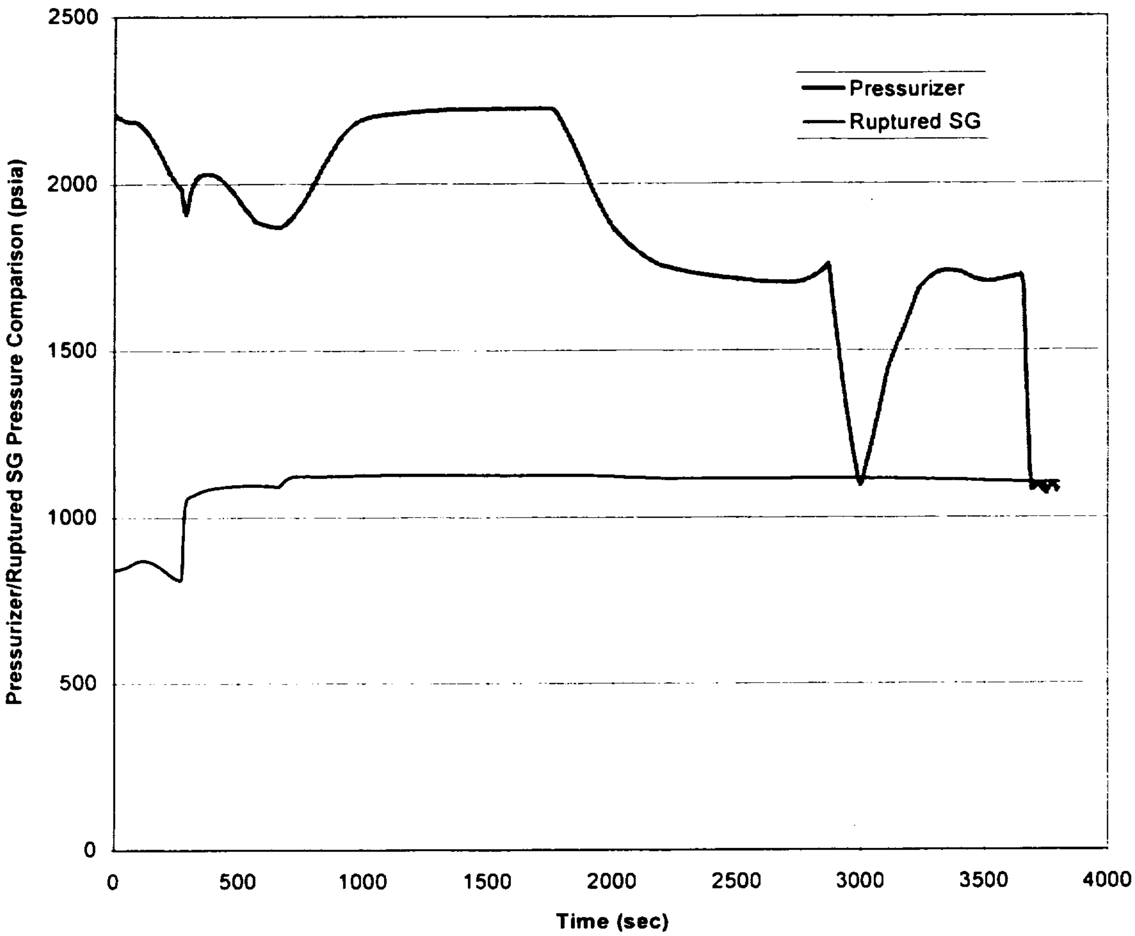
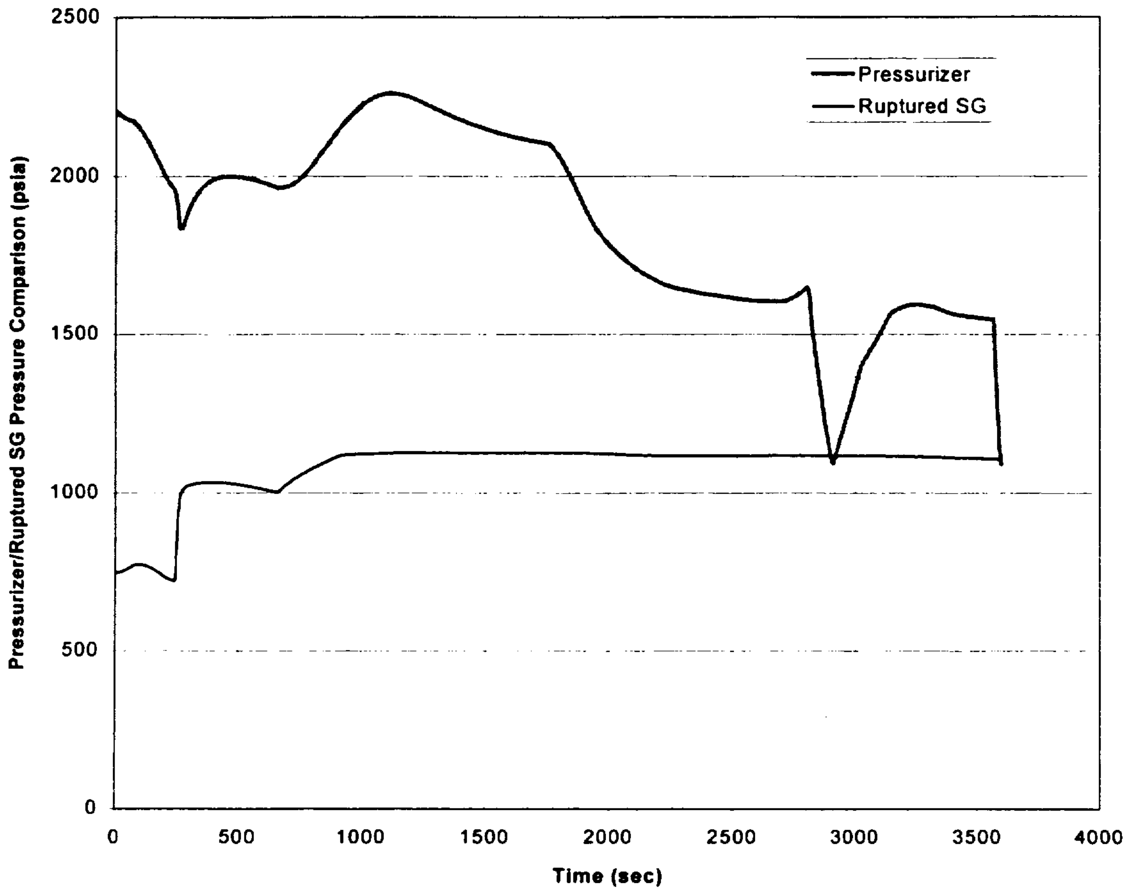


Figure 6.3-6 – Pressurizer / Ruptured Steam Generator Pressure Comparison – Unit 2



6.4 LOCA Containment Integrity

The uncontrolled release of pressurized high temperature reactor coolant, termed a Loss-of-Coolant Accident (LOCA), will result in release of steam and water into the containment. This, in turn, will increase the local subcompartment pressures and the global containment pressure and temperature. Therefore, there are both long- and short-term issues reviewed relative to a postulated LOCA that must be considered at the power uprate conditions for Byron Units 1 and 2 and Braidwood Units 1 and 2.

The long-term LOCA mass and energy releases are analyzed to approximately 10^6 seconds, and are utilized as input to the containment integrity analysis, which demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a hypothetical large-break LOCA. The containment safeguards systems must be capable of limiting peak containment pressure to less than the design pressure and the temperature excursion to less than the Environmental Qualification (EQ) acceptance limits. Even though this is a first time application of this methodology for Byron Units 1 and 2 and Braidwood Units 1 and 2, it has been utilized and approved on many plant-specific dockets. Section 6.4.1 discusses the long-term LOCA mass and energy releases generated for this program. The results of this analysis were provided for use in the containment integrity analysis and environmental qualification (EQ) of Section 6.4.3.

The short-term LOCA-related mass and energy releases are used as input to the subcompartment analyses, which are performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than three seconds) accompanying a high-energy-line pipe rupture within that subcompartment. The subcompartments evaluated include the steam generator compartment, reactor cavity region, and pressurizer compartment. For the steam generator compartment and reactor cavity region, the fact that Byron Units 1 and 2 and Braidwood Units 1 and 2 are approved for Leak-Before-Break (LBB) was used to qualitatively demonstrate that any changes associated with the power uprate are offset by the LBB benefit of using smaller Reactor Coolant System (RCS) nozzle breaks, thus demonstrating that the current licensing bases for these subcompartments remain bounding. For the pressurizer compartment, the critical mass flux correlation utilized in the SATAN computer program (Reference 4) was used to conservatively estimate the impact of the changes in RCS temperatures on the short-term releases. The evaluation showed that the releases based on the power uprate conditions were bounded by the releases documented in the Byron/Braidwood Stations Updated Final Safety Analysis Report (UFSAR), Revision 7. Section 6.4.2 discusses the short-term evaluation conducted for this program. The results of this evaluation were provided for use in the pressurizer subcompartment evaluation.

The containment response analysis (Section 6.4.3) demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a LOCA inside containment. The impact of LOCA mass and energy releases on the containment pressure is addressed to assure that the containment pressure remains below its design pressure at the uprated conditions. In support of equipment design and licensing criteria (e.g., qualified operating life), with respect to post accident environmental conditions, long-term containment pressure and temperature transients are generated to conservatively bound the potential post-LOCA containment conditions.

6.4.1 Long-Term LOCA Mass and Energy Releases

The mass and energy release rates described in this section form the basis for the containment pressure calculations presented in Section 6.4.3. Discussed in this section are the long-term LOCA mass and energy releases for the hypothetical double-ended pump suction (DEPS) rupture and double-ended hot leg (DEHL) rupture break cases for Byron and Braidwood Units 1 (BWI replacement steam generator) and Units 2 (original Westinghouse model D5 steam generator).

6.4.1.1 Input Parameters and Assumptions

The mass and energy release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. For example, the RCS operating temperatures are chosen to bound the highest average coolant temperature range of all operating cases, and a temperature uncertainty allowance (+10.0°F) is then added. Nominal parameters are used in certain instances. For example, the RCS pressure in this analysis is based on a nominal value of 2250 psia plus an uncertainty allowance (+43 psi). All input parameters are chosen consistent with accepted analysis methodology.

Some of the most-critical items are the RCS initial conditions, core decay heat, safety injection flow, and primary and secondary metal mass and steam generator heat release modeling. Specific assumptions concerning each of these items are discussed below. Tables 6.4.1-1 through 6.4.1-3 present key data assumed in the analysis.

The core rated power of 3658.3 MWt (including a calorimetric error of +2 percent) was used in the analysis. As previously noted, the use of RCS operating temperatures to bound the highest average coolant temperature range were used as bounding analysis conditions. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures which are at the maximum levels attained in steady state operation. Additionally, an allowance to account for instrument error and deadband is reflected in the initial RCS temperatures. As previously discussed, the initial RCS pressure in this analysis is based on a nominal value of 2250 psia plus an allowance which accounts for the measurement uncertainty on pressurizer pressure. The selection of 2250 psia as the limiting pressure is considered to affect the blowdown phase results only, since this represents the initial RCS pressure. The RCS rapidly depressurizes from this value until it equilibrates with containment pressure.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure. Additionally the RCS has a higher fluid density at the higher pressure (assuming a constant temperature) and subsequently has a higher RCS mass available for releases. Thus, 2250 psia plus uncertainty was selected for the initial pressure as the limiting case for the long-term mass and energy release calculations.

The selection of the fuel design features for the long-term mass and energy release calculation is based on the need to conservatively maximize the energy stored in the fuel at the beginning of the postulated accident (i.e., to maximize the core stored energy). The margin in core stored energy was chosen to be +15 percent. Thus, the analysis very conservatively accounts for the stored energy in the core.

Margin in RCS volume of 3% (which is composed of 1.6% allowance for thermal expansion and 1.4% for uncertainty) is modeled.

A uniform steam generator (SG) tube plugging level of 0% is modeled. This assumption maximizes the reactor coolant volume and fluid release by RCS fluid is present in all SG tubes.

During the post-blowdown period the steam generators are active heat sources, since significant energy remains in the secondary metal and secondary mass and has the potential to be transferred to the primary side. The 0% tube plugging assumption maximizes heat transfer area and therefore the transfer of secondary heat across the SG tubes. Additionally, this assumption reduces the reactor coolant loop resistance, which reduces the Δp upstream of the break for the pump suction breaks and increases break flow. Thus, the analysis very conservatively accounts for the level of steam generator tube plugging.

Regarding safety injection flow, the mass and energy release calculation considered configurations and failures to conservatively bound respective alignments. The cases include: (a) a Minimum Safeguards Case (1 CV, 1 SI, and 1 RH Pumps); and (b) Maximum Safeguards, (2 CV, 2 SI, and 2 RH Pumps).

The following assumptions were employed to ensure that the mass and energy releases are conservatively calculated, thereby maximizing energy release to containment.

1. Maximum expected operating temperature of the reactor coolant system (100% full power conditions)

2. Allowance for RCS temperature uncertainty (+10.0°F)

3. Margin in RCS volume of 3% (which is composed of 1.6% allowance for thermal expansion, and 1.4% for uncertainty)

4. Core rated power of 3586.6 MWt

5. Allowance for calorimetric error (+2 percent of power)

6. Conservative heat transfer coefficient (i.e., steam generator primary/secondary heat transfer and RCS metal heat transfer)

7. Allowance in core stored energy for effect of fuel densification

8. A margin in core stored energy (+15 percent to account for manufacturing tolerances)

9. An allowance for RCS initial pressure uncertainty (+43 psi)

10. A maximum containment backpressure equal to design pressure (50 psig)

11. Minimum RCS loop flow (92,000 gpm/loop)

12. Steam generator tube plugging leveling (0% uniform)

    - Maximizes reactor coolant volume and fluid release
    - Maximizes heat transfer area across the SG tubes
    - Reduces coolant loop resistance, which reduces the Δp upstream of the break for the pump suction breaks and increases break flow

Additionally, there are some differences between Byron & Braidwood Units 1 and Units 2. Units 1 at each site have BWI replacement steam generators, whereas Units 2 at each site have Westinghouse model D5 steam generators. Separate analytical models were generated for each steam generator type and used for the calculations. Mass and energy releases for both steam generator designs are provided herein.

6.4.1.2 Description of Analyses

The evaluation model used for the long-term LOCA mass and energy release calculations is the March 1979 model described in Reference 1. This evaluation model has been reviewed and approved generically by the NRC. The approval letter is included with Reference 1. Even though this is a first time application for Byron Units 1 and 2 and Braidwood Units 1 and 2, it has also been utilized and approved on the plant-specific dockets for other Westinghouse PWRs.

This report section presents the long-term LOCA mass and energy releases generated in support of the Byron Units 1 and 2 and Braidwood Units 1 and 2 power uprate program. These mass and energy releases are then subsequently used in the containment integrity analysis.

6.4.1.3 LOCA Mass and Energy Release Phases

The containment system receives mass and energy releases following a postulated rupture in the RCS. These releases continue over a time, which, for the LOCA mass and energy analysis, is typically divided into four phases.

1. Blowdown - the period from accident initiation (reactor is at steady state operation) to the time that the RCS and containment reach an equilibrium state.

2. Refill - the period when the lower plenum is being filled by accumulator and Emergency Core Cooling System (ECCS) water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, it is assumed that this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Thus, the refill period is conservatively neglected in the mass and energy release calculation.

3. Reflood - begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

4. Post-reflood (Froth) - describes the period following the reflood phase. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators prior to exiting the break as steam. After the broken loop steam generator cools, the break flow becomes two phase.

6.4.1.4 Computer Codes

The Reference 1 mass and energy release evaluation model is comprised of mass and energy release versions of the following codes: SATAN VI, WREFLOOD, FROTH, and EPITOME. These codes were used to calculate the long-term LOCA mass and energy releases for Byron Units 1 and 2 and Braidwood Units 1 and 2.

SATAN VI calculates blowdown, the first portion of the thermal-hydraulic transient following break initiation, including pressure, enthalpy, density, mass and energy flowrates, and energy transfer between primary and secondary systems as a function of time.

The WREFLOOD code addresses the portion of the LOCA transient where the core reflooding phase occurs after the primary coolant system has depressurized (blowdown) due to the loss of water through the break and when water supplied by the ECCS refills the reactor vessel and provides core cooling. The most important feature of WREFLOOD is the steam/water mixing model (see subsection 6.4.1.8.2).

FROTH models the post-reflood portion of the transient. The FROTH code is used for the steam generator heat addition calculation from the broken and intact loop steam generators.

EPITOME continues the FROTH post-reflood portion of the transient from when the secondary equilibrates to containment design pressure through the end of the transient. It also compiles a summary of data on the entire transient, including formal instantaneous mass and energy release tables and mass and energy balance tables with data at critical times.

6.4.1.5 Break Size and Location

Generic studies have been performed with respect to the effect of postulated break size on the LOCA mass and energy releases. The double ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the reactor coolant system loop can be postulated for pipe rupture for any release purposes:

1. Hot leg (between vessel and steam generator)
2. Cold leg (between pump and vessel)
3. Pump suction (between steam generator and pump)

The break locations analyzed for this program are the double-ended pump suction (DEPS) rupture (10.48 ft^2), and the double-ended hot leg (DEHL) rupture (9.18 ft^2). Break mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases of the

LOCA for the DEPS cases. For the DEHL case, the releases were calculated only for the blowdown. The following information provides a discussion on each break location.

The DEHL rupture has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be the highest for this break location, the energy released from the steam generator secondary is minimal because the majority of the fluid which exits the core vents directly to containment bypassing the steam generators. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction or cold leg break locations where the core exit mixture must pass through the steam generators before venting through the break. For the hot leg break, generic studies have confirmed that there is no reflood peak (i.e., from the end of the blowdown period the containment pressure would continually decrease). Therefore only the mass and energy releases for the hot leg break blowdown phase are calculated and presented in this section of the report.

The cold leg break location has also been found in previous studies to be much less limiting in terms of the overall containment energy releases. The cold leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient for the cold leg is, in general, less limiting than that for the pump suction break. During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is reduced. Therefore, the cold leg break is bounded by other breaks and no further evaluation is necessary.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and the addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the RCS in calculating the releases to containment.

6.4.1.6 Application of Single-Failure Criterion

An analysis of the effects of the single-failure criterion has been performed on the mass and energy release rates for each break analyzed. An inherent assumption in the generation of the mass and energy release is that offsite power is lost. This results in the actuation of the emergency diesel generators, required to power the safety injection system. This is not an issue for the blowdown period which is limited by the DEHL break.

Two cases have been analyzed to assess the effects of a single failure. The first case assumes minimum safeguards SI flow based on the postulated single failure of an emergency diesel generator. This results in the loss of one train of safeguards equipment. The other case assumes maximum safeguards SI flow based on no postulated failures that would impact the amount of ECCS flow. The analysis of the cases described provides confidence that the effect of credible single failures is bounded.

6.4.1.7 Acceptance Criteria for Analyses

A large-break loss-of-coolant accident is classified as an ANS Condition IV event, an infrequent fault. To satisfy the Nuclear Regulatory Commission acceptance criteria presented in the Standard Review Plan Section 6.2.1.3, the relevant requirements are as follows:

a. 10 CFR 50, Appendix A
b. 10 CFR 50, Appendix K, paragraph I.A

In order to meet these requirements, the following must be addressed.

1. Sources of Energy
2. Break Size and Location
3. Calculation of Each Phase of the Accident

6.4.1.8 Mass and Energy Release Data

The following subsections apply to both BWI and D5 steam generator designs. Table references list the BWI data table first, followed by the corresponding D5 data table, in the format "Tables 6.4.1-(BWI)/-(D5)" to simplify the text presentation.

6.4.1.8.1 Blowdown Mass and Energy Release Data

The SATAN-VI code is used for computing the blowdown transient. The code utilizes the control volume (element) approach with the capability for modeling a large variety of thermal fluid system configurations. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. A point kinetics model is used with weighted feedback effects. The major feedback effects include moderator density, moderator temperature, and Doppler broadening. A critical flow calculation for subcooled (modified Zaloudek), two-phase (Moody), or superheated break flow is incorporated into the analysis. The methodology for the use of this model is described in Reference 1.

Tables 6.4.1-4/-22 present the calculated mass and energy release for the blowdown phase of the DEHL break. For the hot leg break mass and energy release tables, break path 1 refers to the mass and energy exiting from the reactor vessel side of the break; break path 2 refers to the mass and energy exiting from the steam generator side of the break.

Tables 6.4.1-7/-25 present the calculated mass and energy releases for the blowdown phase of the DEPS break with minimum ECCS flows. Tables 6.4.1-13/-31 present the calculated mass and energy releases for the blowdown phase of the DEPS break with maximum ECCS flows. For the pump suction breaks, break path 1 in the mass and energy release tables refers to the mass and energy exiting from the steam generator side of the break; break path 2 refers to the mass and energy exiting from the pump side of the break.

6.4.1.8.2 Reflood Mass and Energy Release Data

The WREFLOOD code is used for computing the reflood transient. The WREFLOOD code consists of two basic hydraulic models - one for the contents of the reactor vessel, and one for the coolant loops. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena such as pumped safety injection and accumulators, reactor coolant pump performance, and steam generator release are included as auxiliary equations which interact with the basic models as required. The WREFLOOD code permits the capability to calculate variations during the core reflooding transient of basic parameters such as core flooding rate, core and downcomer water levels, fluid thermodynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flow rates through the primary system. The code permits hydraulic modeling of the two flow paths available for discharging steam and entrained water from the core to the break; i.e., the path through the broken loop and the path through the unbroken loops.

A complete thermal equilibrium mixing condition for the steam and ECCS injection water during the reflood phase has been assumed for each loop receiving ECCS water. This is consistent with the usage and application of the Reference 1 mass and energy release evaluation model in recent analyses, e.g., D.C. Cook Docket (Reference 2). Even though the Reference 1 model credits steam/water mixing only in the intact loop and not in the broken loop; the justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (Reference 2). Moreover, this assumption is supported by test data and is further discussed below.

The model assumes a complete mixing condition (i.e., thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two-phase interaction with condensation of steam by cold ECCS water. The second is a single-phase mixing of condensate and ECCS water. Since the steam release is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that need be considered. (Any spillage directly heats only the sump.)

The most applicable steam/water mixing test data has been reviewed for validation of the containment integrity reflood steam/water mixing model. This data was generated in 1/3-scale tests (Reference 3), which are the largest scale data available and thus most clearly simulates the flow regimes and gravitational effects that would occur in a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

A group of 1/3-scale tests corresponds directly to containment integrity reflood conditions. The injection flowrates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in Reference 1. For all of these tests, the data clearly indicates the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the containment integrity reflood calculation is therefore wholly supported by the 1/3-scale steam/water mixing data.

Additionally, the following justification is also noted. The post-blowdown limiting break for the containment integrity peak pressure analysis is the pump suction double ended rupture break. For this break, there are two flowpaths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the reactor coolant pump. Steam which is not condensed by ECCS injection in the intact RCS loops passes around the downcomer and through the broken loop cold leg and pump in venting to containment. This steam also encounters ECCS injection water as it passes through the broken loop cold leg, complete mixing occurs and a portion of it is condensed. It is this portion of steam which is condensed that is taken credit for in this analysis. This assumption is justified based upon the postulated break location, and the actual physical presence of the ECCS injection nozzle. A description of the test and test results are contained in References 1 and 3.

Tables 6.4.1-8/-26 and 6.4.1-14/-32 present the calculated mass and energy releases for the reflood phase of the pump suction double-ended rupture, minimum safeguards, and maximum safeguards cases, respectively.

The transient response of the principal parameters during reflood are given in Tables 6.4.1-9/-27 and 6.4.1-15/-33 for the DEPS cases.

6.4.1.8.3 Post-Reflood Mass and Energy Release Data

The FROTH code (Reference 4) is used for computing the post-reflood transient. The FROTH code calculates the heat release rates resulting from a two-phase mixture present in the steam generator tubes. The mass and energy releases that occur during this phase are typically superheated due to the depressurization and equilibration of the broken loop and intact loop steam generators. During this phase of the transient, the RCS has equilibrated with the containment pressure, but the steam generators contain a secondary inventory at an enthalpy much higher than the primary side. Therefore, significant reverse heat transfer occurs. Steam is produced in the core due to core decay heat. For a pump suction break, a two-phase fluid exits the core, flows through the hot legs and becomes superheated as it passes through the steam generator. Once the broken loop cools, the break flow becomes two phase. During the FROTH calculation, ECCS injection is addressed for both the injection phase and the recirculation phase. The FROTH code calculation stops when the secondary side equilibrates to the saturation temperature (T_{sat}) at the containment design pressure, after this point the EPITOME code completes the SG depressurization (see subsection 6.4.1.8.5 for additional information).

The methodology for the use of this model is described in Reference 1. The mass and energy release rates are calculated by FROTH and EPITOME through containment depressurization. After containment depressurization (14.7 psia), the mass and energy release available to containment is generated directly from core boiloff/decay heat.

Tables 6.4.1-10/-28 and 6.4.1-16/-34 present the two-phase post-reflood mass and energy release data for the pump suction double-ended cases, minimum and maximum ECCS assumptions, BWI and D5 steam generators.

6.4.1.8.4 Decay Heat Model

On November 2, 1978, the Nuclear Power Plant Standards Committee (NUPPSCO) of the American Nuclear Society approved ANS Standard 5.1 (Reference 5) for the determination of decay heat. This standard was used in the M&E release. Table 6.4.1-40 lists the decay heat curve used in the M&E release analysis, post blowdown, for Byron Units 1 and 2, and Braidwood Units 1 and 2 power uprate program.

Significant assumptions in the generation of the decay heat curve for use in the LOCA M&E releases analysis include the following:

1. Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.

2. Decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.

3. Fission rate is constant over the operating history of maximum power level.

4. The factor accounting for neutron capture in fission products has been taken from Equation 11 of Reference 5, up to 10,000 seconds and from Table 10 of Reference 5, beyond 10,000 seconds.

5. The fuel has been assumed to be at full power for 10^8 seconds.

6. The number of atoms of U-239 produced per second has been assumed to be equal to 70 percent of the fission rate.

7. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.

8. Two-sigma uncertainty (two times the standard deviation) has been applied to the fission product decay.

Based upon NRC staff review, Safety Evaluation Report (SER) of the March 1979 evaluation model (Reference 1), use of the ANS Standard-5.1, November 1979 decay heat model was approved for the calculation of M&E releases to the containment following a LOCA.

6.4.1.8.5 Steam Generator Equilibration and Depressurization

Steam generator equilibration and depressurization is the process by which secondary side energy is removed from the steam generators in stages. The FROTH computer code calculates the heat removal from the secondary mass until the secondary temperature reaches saturation (T_{sat}) at the containment design pressure. After the FROTH calculations, the EPITOME code continues the FROTH calculation for SG cooldown, removing steam generator secondary energy at different rates (i.e., first and second stage rates). The first stage rate is applied until the steam generator reaches T_{sat} at the user specified intermediate equilibration pressure, when the secondary pressure is assumed to reach actual containment pressure. Then the second stage rate is used until the final depressurization, when the secondary reaches saturation at 14.7 psia, or 212°F. The heat removal of the broken loop and intact loop steam generators are calculated separately.

During the FROTH calculations, steam generator heat removal rates are calculated using the secondary side temperature, primary side temperature and a secondary side heat transfer coefficient determined using a modified McAdam's correlation. Steam generator energy is removed during the FROTH transient until the secondary side temperature reaches saturation at the containment design pressure. The constant heat removal rate used during the first heat removal stage is based on the final heat removal rate calculated by FROTH. The SG energy available to be released during the first stage interval is determined by calculating the difference in secondary energy available at the containment design pressure and that at the (lower) user specified intermediate equilibration pressure, assuming saturated conditions. This energy is then divided by the first stage energy removal rate, resulting in an intermediate equilibration time. At this time, the rate of energy release drops substantially to the second stage rate. The second stage rate is determined as the fraction of the difference in secondary energy available between the intermediate equilibration and final depressurization at 212°F, and the time difference from intermediate equilibration to the user specified time of the final depressurization at 212°F. With current methodology, all of the secondary energy remaining after the intermediate equilibration is conservatively assumed to be released by imposing a mandatory cooldown and subsequent depressurization to atmospheric pressure (i.e., 14.7 psia and 212°F) at 3600 seconds.

6.4.1.8.6 Sources of Mass and Energy

The sources of mass considered in the LOCA mass and energy release analyses are given in Tables 6.4.1-5/-23, 6.4.1-11/-29, and 6.4.1-17/-35. These sources are the reactor coolant system, accumulators, and pumped safety injection.

The energy inventories considered in the LOCA mass and energy release analyses are given in Tables 6.4.1-6/-24, 6.4.1-12/-30, and 6.4.1-18/-36. The energy sources include:

1. Reactor Coolant System Water
2. Accumulator Water (all four inject)
3. Pumped Safety Injection Water
4. Decay Heat
5. Core Stored Energy
6. Reactor Coolant System Metal (includes SG tubes)
7. Steam Generator Metal (includes transition cone, shell, wrapper, and other internals)
8. Steam Generator Secondary Energy (includes fluid mass and steam mass)
9. Secondary Transfer of Energy (feedwater into and steam out of the SG secondary)

Energy Reference Points:

1. Available Energy: 212°F; 14.7 psia

   (The current approved methodology assumes that all energies in the system are taken out to these conditions in the first hour of the event. This is the total available energy.)

2. Total Energy Content: 32°F; 14.7 psia

   (This is the reference point for the system energy.)

The mass and energy inventories are presented at the following times, as appropriate:

1. Time zero (initial conditions)
2. End of blowdown time
3. End of refill time
4. End of reflood time
5. Time of broken loop steam generator equilibration to pressure setpoint
6. Time of intact loop steam generator equilibration to pressure setpoint
7. Time of full depressurization (3600 seconds)

In the mass and energy release data presented, no Zirc-water reaction heat was considered because the clad temperature is assumed not to rise high enough for the rate of the Zirc-water reaction heat to be of any significance.

The sequence of events for the LOCA transients are shown in Tables 6.4.1-19/-37 through 6.4.1-21/-39.

6.4.1.9 Conclusions

The consideration of the various energy sources in the long-term mass and energy release analysis, including the BWI and D5 steam generators, provides assurance that all available sources of energy have been included in this analysis. Thus, the review guidelines presented in Standard Review Plan Section 6.2.1.3 have been satisfied. The results of this analysis were provided for use in the containment integrity analysis, Section 6.4.3.

6.4.1.10 References

1. "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," WCAP-10325-P-A, May 1983 (Proprietary), WCAP-10326-A (Nonproprietary).

2. Docket No. 50-315, "Amendment No. 126, Facility Operating License No. DPR-58 (TAC No. 7106), for D. C. Cook Nuclear Plant Unit 1," June 9, 1989.

3. EPRI 294-2, "Mixing of Emergency Core Cooling Water with Steam; 1/3-Scale Test and Summary," (WCAP-8423), Final Report, June 1975.

4. "Westinghouse Mass and Energy Release Data For Containment Design," WCAP-8264-P-A, Rev. 1, August 1975 (Proprietary), WCAP-8312-A (Nonproprietary).

5. ANSI/ANS-5.1 1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

Table 6.4.1-1 System Parameters Initial Conditions For Thermal Uprate		
Parameters	Value	
	BWI SG	D5 SG
Core Thermal Power (MWt)	3586.6	3586.6
Reactor Coolant System Total Flowrate (lbm/sec)	37590.96	37590.39
Vessel Outlet Temperature (°F)	630.3	630.3
Core Inlet Temperature (°F	565.7	565.7
Vessel Average Temperature (°F)	598.0	598.0
Initial Steam Generator Steam Pressure (psia)	1024.	953.0
Steam Generator Tube Plugging (%)	0	0
Initial Steam Generator Secondary Side Mass (lbm)	136617.8	106484.0
Assumed Maximum Containment Backpressure (psia)	64.7	64.7
Accumulator Water Volume (ft^3) per accumulator N_2 Cover Gas Pressure (psia) Temperature (°F)	 1005.9 661.7 130.	 1012.2 661.7 130.
Safety Injection Delay, total (sec) (from beginning of event)	40.0	40.0

Note: Core Thermal Power, RCS Total Flowrate, RCS Coolant Temperatures, and Steam Generator Secondary Side Mass include appropriate uncertainty and/or allowance.

Table 6.4.1-2 Safety Injection Flow Minimum Safeguards	
RCS Pressure (psia)	**Total Flow (gpm)**
Injection Mode (Reflood Phase)	
14.7	5686.1
40.	5609.8
60.	5117.1
80.	4532.4
100.	3666.7
120.	1968.2
130.	1217.4
135.	952.6
180.	935.6
200.	928.1
Injection Mode (Post-Reflood Phase)	
64.7	5077.6
Cold Leg Recirculation Mode	
64.7	994.1

Table 6.4.1-3
Safety Injection Flow
Maximum Safeguards

RCS Pressure (psia)	Total Flow (gpm)
Injection Mode (Reflood Phase)	
14.7	12305.
60.	12288.
135.	12226.
327.	1582.
600.	1460.
1800.	659.4
1840.	542.2
2200.	457.1
3000.	59.50
3035.	1.40
Injection Mode (Post-Reflood Phase)	
64.7	12305.
Cold Leg Recirculation Mode	
64.7	11917.1

Table 6.4.1-4 Double-Ended Hot Leg Break Blowdown Mass And Energy Releases Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
	Break Path No.1*		Break Path No.2**	
Time Seconds	Flow (Lbm/Sec)	Energy (Thousand Btu/Sec)	Flow (Lbm/Sec)	Energy (Thousand Btu/Sec)
.00000	.0	.0	.0	.0
.00112	48435.2	31832.6	48434.9	31831.4
.102	42424.3	28200.0	28354.3	18590.7
.201	37791.6	25119.3	24659.4	16057.4
.301	37237.3	24727.2	22285.4	14313.5
.401	35833.3	23763.4	21100.2	13332.5
.502	35401.5	23453.1	20381.9	12679.1
.601	35318.8	23385.8	19911.8	12222.6
.702	35036.8	23210.6	19515.5	11843.6
.802	34470.4	22872.8	19238.8	11566.9
.902	33863.0	22526.7	18996.0	11331.8
1.00	33390.0	22282.8	18829.8	11159.2
1.10	33192.0	22234.5	18645.7	10988.8
1.20	32886.9	22120.3	18534.8	10870.4
1.30	32408.2	21888.1	18480.6	10791.9
1.40	31828.7	21586.6	18452.1	10733.3
1.50	31289.7	21304.5	18455.9	10697.5
1.60	30888.7	21109.4	18483.3	10678.8
1.70	30515.5	20929.2	18523.8	10670.8
1.80	30025.3	20662.5	18565.0	10666.0
1.90	29447.6	20327.5	18603.5	10662.8
2.00	28880.9	19994.1	18640.5	10661.4
2.10	28400.0	19716.9	18676.1	10662.2
2.20	27972.9	19474.0	18707.4	10663.4
2.30	27491.6	19186.0	18729.5	10661.8
2.50	26504.8	18569.5	18734.5	10643.8
2.60	26064.1	18290.6	18719.2	10628.0
2.70	25642.5	18021.0	18691.6	10607.1
2.80	25251.7	17768.6	18651.3	10580.8
2.90	24871.4	17518.5	18599.0	10549.1
3.00	24495.5	17264.0	18534.8	10512.2
3.10	24148.5	17025.2	18457.6	10469.1

Table 6.4.1-4 (cont.) Double-Ended Hot Leg Break Blowdown Mass And Energy Releases Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
	Break Path No.1*		Break Path No.2**	
Time (Seconds)	Flow (Lbm/Sec)	Energy Thousands (Btu/Sec)	Flow (Lbm/Sec)	Energy Thousands (Btu/Sec)
3.20	23845.6	16815.6	18371.7	10422.0
3.30	23552.6	16608.0	18275.9	10370.3
3.40	23274.1	16405.1	18169.3	10313.1
3.50	23035.8	16228.1	18055.9	10252.7
3.60	22805.5	16052.5	17931.1	10186.3
3.70	22591.6	15883.7	17799.1	10116.1
3.80	22409.9	15735.4	17655.5	10039.8
3.90	22247.5	15598.3	17501.3	9957.5
4.00	22097.4	15466.3	17326.8	9864.0
4.20	21857.8	15239.4	16969.0	9672.3
4.40	21675.8	15046.1	16605.0	9477.6
4.60	21553.5	14891.6	16241.5	9283.1
4.80	21500.3	14783.0	15875.7	9086.8
5.00	21473.1	14693.8	15513.5	8892.3
5.20	21460.4	14620.5	15146.4	8694.7
5.40	21520.2	14587.5	14827.9	8525.3
5.60	21625.8	14571.5	14527.9	8365.4
5.80	21778.6	14579.3	14157.0	8163.8
6.00	21983.9	14610.8	13810.2	7976.7
6.20	22242.8	14663.9	13493.3	7806.8
6.40	22656.5	14794.1	13193.7	7646.5
6.60	12889.1	10455.3	12894.8	7485.9
6.80	16622.1	11934.7	12606.2	7330.7
7.00	16809.8	11990.6	12326.8	7181.0
7.20	17108.6	12042.0	12075.7	7047.6
7.60	17728.8	12221.9	11581.7	6783.2
7.80	18104.0	12351.9	11338.8	6652.8
8.00	18518.9	12538.1	11096.1	6522.2
8.20	18960.3	12702.0	10856.3	6393.0
8.40	19255.0	12794.9	10618.0	6264.5
8.60	20318.2	13312.2	10378.8	6135.5
8.80	23306.0	15125.8	10144.8	6009.6
9.00	29459.7	19011.1	9906.9	5881.8

Table 6.4.1-4 (cont.)
Double-Ended Hot Leg Break
Blowdown Mass And Energy Releases
Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time (Seconds)	Break Path No.1* Flow (Lbm/Sec)	Break Path No.1* Energy Thousands (Btu/Sec)	Break Path No.2** Flow (Lbm/Sec)	Break Path No.2** Energy Thousands (Btu/Sec)
9.20	28128.1	17979.3	9667.5	5753.3
9.40	27697.5	17534.1	9407.6	5613.0
9.60	27566.1	17323.2	9109.0	5451.2
9.80	27482.4	17171.4	8813.8	5294.6
10.0	27406.2	17064.2	8506.5	5133.3
10.2	27273.9	16942.3	8202.1	4976.2
10.4	27099.4	16823.9	7898.4	4821.0
10.6	26904.0	16705.0	7600.2	4670.9
10.8	25740.7	15987.1	7309.7	4526.1
11.0	25305.1	15720.4	7029.6	4388.5
11.2	25198.4	15644.1	6761.0	4257.8
11.4	24995.0	15517.7	6506.7	4135.9
11.6	24332.0	15109.2	6268.0	4022.4
11.8	16451.4	10005.6	6039.7	3914.6
12.0	16396.4	9946.1	5827.7	3815.3
12.2	16468.5	9979.8	5643.8	3731.9
12.4	12371.8	8562.9	5495.7	3665.3
12.6	10886.2	7743.7	5364.3	3602.3
12.8	10434.5	7583.9	5262.5	3550.6
13.0	10686.1	7692.2	5187.0	3507.9
13.2	10792.0	7748.7	5127.7	3470.5
13.4	10810.6	7774.5	5080.0	3436.9
13.6	10783.4	7786.1	5037.6	3405.7
13.8	10721.4	7781.0	4995.3	3375.0
14.0	10646.4	7772.4	4947.9	3343.0
14.2	10474.9	7740.9	4896.2	3311.3
14.4	10187.4	7715.7	4828.8	3273.3
14.6	9960.6	7714.8	4754.3	3234.8
14.8	9611.7	7608.1	4666.9	3192.0
15.0	9198.8	7414.6	4567.4	3145.8
15.2	8691.8	7134.3	4455.7	3095.8
15.4	8077.0	6782.0	4336.1	3044.1
15.6	7299.7	6239.8	4212.4	2993.0

Table 6.4.1-4 (cont.)
Double-Ended Hot Leg Break
Blowdown Mass And Energy Releases
Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time (Seconds)	Break Path No.1* Flow (Lbm/Sec)	Break Path No.1* Energy Thousands (Btu/Sec)	Break Path No.2** Flow (Lbm/Sec)	Break Path No.2** Energy Thousands (Btu/Sec)
15.8	6923.1	5892.2	4078.7	2937.8
16.0	6637.2	5686.3	3938.1	2878.9
16.2	6339.4	5497.7	3779.4	2812.6
16.4	6014.1	5292.6	3597.6	2740.7
16.6	5665.3	5073.1	3394.7	2665.4
16.8	5281.6	4827.4	3166.0	2585.2
17.0	4913.6	4564.7	2917.5	2500.0
17.2	4527.9	4304.9	2657.2	2411.9
17.6	3821.5	3845.3	2152.1	2238.9
17.8	3514.8	3653.2	1938.1	2153.6
18.0	3240.5	3479.9	1763.3	2054.7
18.2	3006.8	3322.8	1632.4	1959.2
18.4	2820.2	3177.4	1520.0	1849.6
18.6	2650.4	3025.0	1432.4	1755.4
18.8	2491.8	2872.8	1359.4	1673.3
19.0	2326.7	2713.6	1290.0	1593.4
19.2	2159.6	2548.2	1229.7	1522.4
19.4	1996.5	2378.3	1175.8	1459.0
19.6	1863.4	2237.6	1123.3	1396.4
19.8	1716.8	2075.3	1078.4	1342.5
20.0	1586.6	1930.7	1033.1	1287.9
20.2	1473.3	1804.6	988.8	1234.1
20.4	1392.8	1719.0	944.6	1180.9
20.6	1338.4	1663.3	904.4	1132.3
20.8	1261.3	1574.3	876.4	1098.3
21.0	1181.4	1477.9	849.9	1066.1
21.2	1104.1	1382.8	817.9	1026.5
21.4	1036.5	1299.0	762.0	957.1
21.6	965.6	1211.1	700.8	881.2
21.8	883.0	1107.9	653.4	822.8
22.0	802.0	1009.0	618.4	779.6
22.2	736.4	928.9	514.7	648.6
22.4	671.6	847.9	463.3	585.5

Table 6.4.1-4 (cont.) Double-Ended Hot Leg Break Blowdown Mass And Energy Releases Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
	Break Path No.1*		Break Path No.2**	
Time (Seconds)	Flow (Lbm/Sec)	Energy Thousands (Btu/Sec)	Flow (Lbm/Sec)	Energy Thousands (Btu/Sec)
22.6	622.6	786.9	368.9	466.7
22.8	580.9	734.5	315.2	399.7
23.0	563.7	713.7	255.3	324.5
23.2	541.5	685.8	228.2	290.5
23.4	516.5	654.0	173.2	220.7
23.6	491.8	622.1	28.9	37.2
23.8	480.2	606.6	105.8	136.5
24.0	459.4	576.6	80.5	104.3
24.2	494.7	621.5	.0	.0
24.4	479.9	600.3	.0	.0
24.6	475.2	592.7	.0	.0
24.8	468.9	584.3	.0	.0
25.0	461.9	575.2	.0	.0
25.2	441.7	549.7	.0	.0
25.4	423.5	527.9	.0	.0
25.6	365.7	456.8	.0	.0
25.8	87.4	113.1	.0	.0
26.0	.0	.0	.0	.0

* mass and energy exiting from the reactor vessel side of the break

** mass and energy exiting from the SG side of the break

Table 6.4.1-5 Double-Ended Hot Leg Break Mass Balance Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
		Mass Balance		
Time (Seconds)		.00	26.00	26.00
		Mass (Thousand lbm)		
Initial	In RCS and ACC	815.56	815.56	815.56
Added Mass	Pumped Injection	0.0	0.0	0.0
	Total Added	0.0	0.0	0.0
***** TOTAL AVAILABLE *****		815.56	815.56	815.56
Distribution	Reactor Coolant	567.41	74.73	106.35
	Accumulator	248.41	187.18	155.55
	Total Contents	815.56	261.91	261.91
Effluent	Break Flow	0.0	553.63	553.63
	ECCS Spill	0.0	0.0	0.0
	Total Effluent	0.0	553.36	553.36
***** TOTAL ACCOUNTABLE *****		815.56	815.53	815.53

Table 6.4.1-6
Double-Ended Hot Leg Break Energy Balance
Byron Unit 1 & Braidwood Unit 1 (BWI SG)

		Energy Balance		
Time (Seconds)		.00	26.00	26.00
		Energy (Million Btu)		
Initial Energy	In RCS, ACC, S GEN	987.63	987.63	987.63
Added Energy	Pumped Injection	0.00	0.00	0.00
	Decay Heat	0.00	8.90	8.90
	Heat From Secondary	0.00	-2.06	-2.06
	Total Added	0.00	6.83	6.83
***** TOTAL AVAILABLE *****		987.63	994.46	994.46
Distribution	Reactor Coolant	341.36	17.42	20.56
	Accumulator	24.68	18.62	15.47
	Core Stored	23.98	8.90	8.90
	Primary Metal	169.14	158.40	158.40
	Secondary Metal	119.11	117.04	117.04
	Steam Generator	309.36	302.81	302.81
	Total Contents	987.63	623.19	623.19
Effluent	Break Flow	0.00	370.67	370.67
	ECCS Spill	0.00	0.00	0.00
	Total Effluent	0.00	370.67	370.67
***** TOTAL ACCOUNTABLE *****		987.63	993.86	993.86

Table 6.4.1-7

Double-Ended Pump Suction Break Minimum Safeguards

Blowdown Mass And Energy Releases

Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time (seconds)	Break Path No. 1*		Break Path No. 2**	
	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
.00000	.0	.0	.0	.0
.00111	84856.4	47801.1	43218.0	24297.4
.101	42559.9	23971.4	22324.8	12538.6
.201	42838.3	24219.8	24793.8	13937.7
.302	43226.5	24560.0	24955.4	14041.7
.401	43625.6	24932.5	24141.9	13597.3
.501	44056.9	25355.3	23023.0	12976.9
.601	44503.5	25816.6	22087.8	12454.5
.701	44766.3	26186.6	21272.3	11997.6
.802	44701.5	26364.6	20677.5	11665.0
.902	44292.7	26328.7	20267.2	11437.4
1.00	43608.3	26111.4	20050.3	11317.9
1.10	42765.6	25787.4	19936.6	11255.7
1.20	41896.4	25437.0	19893.3	11232.6
1.30	41055.3	25098.8	19872.8	11221.6
1.40	40248.4	24776.5	19863.8	11216.4
1.50	39444.5	24452.4	19883.0	11226.8
1.60	38646.6	24126.6	19938.5	11258.0
1.70	37860.2	23806.2	20003.5	11294.4
1.80	37102.9	23500.1	20050.7	11320.5
1.90	36357.1	23200.0	20074.1	11332.8
2.00	35586.0	22884.6	20054.8	11320.9
2.10	34799.4	22559.2	20013.5	11296.7
2.20	34000.5	22225.5	19938.2	11253.1
2.30	33113.7	21832.7	19761.0	11151.2
2.40	32245.6	21447.7	19368.8	10928.1
2.50	31345.3	21035.8	19192.8	10828.3
2.60	30453.4	20621.3	19045.0	10744.4
2.70	29584.5	20210.5	18860.4	10639.2
2.80	28690.2	19769.1	18637.6	10512.4
2.90	27530.7	19124.9	18408.0	10381.9

Table 6.4.1-7 (cont.) Double-Ended Pump Suction Break Minimum Safeguards Blowdown Mass And Energy Releases Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
	Break Path No. 1*		Break Path No. 2**	
Time (seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
3.00	25607.1	17919.5	18209.4	10269.3
3.10	23120.7	16287.3	18024.2	10164.4
3.20	22051.4	15662.0	17818.1	10047.6
3.30	21231.8	15166.6	17602.4	9925.4
3.40	20066.9	14390.2	17413.2	9818.5
3.50	19292.6	13890.7	17245.9	9724.3
3.60	18601.5	13435.7	17081.8	9631.9
3.70	17919.0	12975.5	16915.4	9538.4
3.80	17325.1	12574.4	16766.2	9454.8
3.90	16802.4	12220.7	16629.0	9378.1
4.00	16343.3	11909.3	16495.1	9303.3
4.20	15577.2	11385.5	16242.7	9162.9
4.40	15005.2	10989.1	16016.4	9037.6
4.60	14528.3	10649.9	15795.3	8915.7
4.80	14171.1	10388.9	15598.5	8808.0
5.00	13887.1	10169.8	15399.1	8699.4
5.20	13693.7	10007.1	15196.1	8589.1
5.40	13568.4	9883.6	14996.0	8480.8
5.60	13516.6	9804.5	14808.6	8380.0
5.80	13538.7	9769.6	14635.9	8287.7
6.00	13657.4	9794.4	14732.1	8352.4
6.20	13984.6	9958.1	15713.8	8911.3
6.40	13794.5	9737.4	15933.4	9045.3
5.40	13809.1	9848.1	15772.4	8959.5
6.80	12902.3	9659.8	15405.3	8756.2
7.00	11840.0	9178.9	15211.0	8651.6
7.20	11960.9	9203.7	15011.1	8543.8
7.40	12484.0	9455.8	14733.0	8391.1
7.60	12914.9	9652.3	14588.6	8313.7
7.80	13366.8	9870.2	14402.0	8209.3
8.00	14033.1	10222.6	14135.1	8057.2
8.20	14406.7	10336.2	13925.1	7937.7
8.40	13644.5	9682.5	13768.5	7848.8

Table 6.4.1-7 (cont.) Double-Ended Pump Suction Break Minimum Safeguards Blowdown Mass And Energy Releases Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
	Break Path No. 1*		Break Path No. 2**	
Time (seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
8.60	12315.0	8738.5	13711.6	7816.5
8.80	12067.7	8604.7	13670.9	7790.9
9.00	12358.0	8773.6	13392.2	7627.0
9.20	12074.9	8524.9	13244.5	7542.2
9.40	11572.3	8192.5	13292.5	7572.3
9.60	11604.2	8254.3	13122.5	7472.4
9.80	11759.7	8350.4	12831.4	7302.7
10.0	11455.8	8105.4	12783.4	7277.7
10.2	11056.4	7837.7	12751.4	7260.9
10.4	11096.6	7884.1	12495.3	7110.8
10.6	10801.6	7661.7	12335.7	7018.7
10.8	10120.5	7214.9	12448.4	7086.1
11.0	9932.6	7151.3	12241.2	6963.0
11.2	9736.3	7062.2	12019.1	6832.7
11.4	9415.4	6886.5	12110.4	6887.9
11.6	9254.9	6818.1	11879.4	6753.8
11.8	9069.7	6711.2	11738.2	6672.0
12.0	8903.8	6610.7	11719.4	6662.9
12.2	8765.6	6523.2	11517.4	6546.2
12.4	8602.8	6419.9	11446.1	6505.4
12.6	8455.8	6330.8	11343.0	6446.7
12.8	8289.2	6230.7	11220.0	6375.7
13.0	8140.9	6149.3	11131.3	6325.1
13.2	7987.6	6064.7	11026.5	6265.3
13.4	7845.5	5989.4	10915.8	6202.2
13.6	7709.4	5915.1	10820.1	6148.5
13.8	7578.5	5841.0	10709.3	6086.1
14.0	7459.1	5772.1	10606.6	6029.0
14.2	7339.7	5701.0	10496.9	5968.0
14.4	7220.4	5629.9	10382.5	5905.0
14.6	7092.2	5553.1	10253.8	5834.2
14.8	6951.4	5468.7	10127.8	5766.1
15.0	6809.7	5384.1	9997.1	5695.6

Table 6.4.1-7 (cont.)
Double-Ended Pump Suction Break Minimum Safeguards
Blowdown Mass And Energy Releases
Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time (seconds)	Break Path No. 1*		Break Path No. 2**	
	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
15.2	6668.4	5297.1	9877.0	5631.6
15.4	6540.6	5214.5	9757.5	5567.7
15.6	6430.2	5138.6	9650.3	5510.9
15.8	6335.7	5070.5	9538.5	5451.8
16.0	6251.8	5009.3	9435.7	5398.6
16.2	6172.5	4953.0	9330.4	5345.1
16.4	6094.2	4901.1	9227.6	5294.0
16.6	6014.9	4853.8	9127.3	5245.6
16.8	5932.7	4811.4	9024.7	5197.1
17.0	5846.8	4774.2	8923.6	5150.9
17.2	5755.9	4744.0	8824.6	5107.7
17.4	5661.3	4717.5	8643.2	5018.4
17.6	5576.1	4708.7	8486.6	4951.9
17.8	5499.5	4729.6	8191.0	4822.8
18.0	5388.2	4767.4	7950.3	4725.2
18.2	5210.5	4791.6	7706.4	4617.0
18.4	4926.1	4744.2	7267.7	4386.4
18.6	4574.0	4649.0	6920.9	4185.9
18.8	4221.4	4533.3	6605.1	3945.4
19.0	3892.5	4399.4	6353.0	3705.7
19.2	3596.2	4231.9	6149.1	3468.8
19.6	3095.5	3794.3	5705.3	2990.7
19.8	2868.8	3540.2	5428.0	2768.4
20.0	2657.6	3294.5	5146.4	2573.4
20.2	2461.6	3062.3	4869.4	2395.2
20.4	2279.6	2844.3	4604.1	2232.5
20.6	2116.8	2647.9	4354.2	2086.0
20.8	1962.1	2460.2	4130.1	1959.5
21.0	1825.7	2293.9	3913.0	1843.1
21.2	1698.9	2138.6	3707.2	1738.0
21.8	1418.0	1793.1	2637.7	1098.8
22.0	1336.9	1692.7	2308.0	913.4
22.2	1257.4	1593.8	2202.0	833.0

Table 6.4.1-7 (cont.) Double-Ended Pump Suction Break Minimum Safeguards Blowdown Mass And Energy Releases Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
	Break Path No. 1*		**Break Path No. 2****	
Time (seconds)	**Flow (lbm/sec)**	**Energy (Thousand Btu/sec)**	**Flow (lbm/sec)**	**Energy (Thousand Btu/sec)**
22.4	1175.5	1491.6	2323.7	847.6
22.6	1092.8	1387.7	2500.8	889.4
22.8	1003.9	1276.7	2724.4	953.9
23.0	912.8	1162.3	2564.3	887.7
23.2	832.6	1061.2	2115.6	728.6
23.6	694.1	885.9	1853.4	635.0
23.8	626.7	800.5	1633.6	557.5
24.0	564.8	721.8	1391.4	473.6
24.2	510.6	652.9	1184.6	403.2
24.4	471.6	603.4	1129.7	382.7
24.6	444.1	568.4	1042.6	349.4
24.8	407.6	521.9	718.8	239.6
25.0	359.8	460.9	254.8	85.2
25.2	324.6	416.0	55.2	18.6
25.4	295.3	378.6	99.2	33.8
25.6	278.7	357.5	.0	.0
25.8	251.8	323.1	.0	.0
26.0	212.7	273.0	.0	.0
26.4	128.9	165.7	.0	.0
26.6	105.2	135.4	.0	.0
26.8	76.5	98.5	.0	.0
27.0	15.6	20.2	.0	.0
27.2	.0	.0	.0	.0

* mass and energy exiting the SG side of the break

** mass and energy exiting the pump side of the break

Table 6.4.1-8
Double-Ended Pump Suction Break
Minimum Safeguards
Reflood Mass And Energy Releases
Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time (seconds)	Break Path No. 1*		Break Path No. 2**	
	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
27.2	.0	.0	.0	.0
27.8	.0	.0	.0	.0
28.0	.0	.0	.0	.0
28.1	.0	.0	.0	.0
28.2	.0	.0	.0	.0
28.3	.0	.0	.0	.0
28.4	.0	.0	.0	.0
28.5	53.2	62.7	.0	.0
28.6	20.8	24.5	.0	.0
28.7	11.7	13.8	.0	.0
28.8	11.7	13.8	.0	.0
28.9	13.6	16.1	.0	.0
29.0	22.7	26.8	.0	.0
29.1	29.4	34.6	.0	.0
29.2	37.4	44.1	.0	.0
29.3	42.4	50.0	.0	.0
29.4	45.9	54.1	.0	.0
29.5	49.5	58.4	.0	.0
29.6	53.0	62.4	.0	.0
29.7	56.3	66.3	.0	.0
29.8	59.4	70.1	.0	.0
29.9	62.5	73.7	.0	.0
30.0	63.2	74.5	.0	.0
30.0	65.4	77.1	.0	.0
30.1	68.3	80.5	.0	.0
30.2	71.0	83.8	.0	.0
31.2	95.2	112.3	.0	.0
32.2	115.0	135.7	.0	.0
33.2	132.1	155.9	.0	.0
34.2	147.2	173.7	.0	.0

Table 6.4.1-8 (cont.)

Double-Ended Pump Suction Break

Minimum Safeguards

Reflood Mass And Energy Releases

Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time (seconds)	Break Path No. 1*		Break Path No. 2**	
	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
35.0	180.9	213.6	1003.3	157.5
35.2	250.0	295.5	2175.6	349.0
36.3	362.5	429.3	3530.5	598.9
37.3	362.1	428.8	3523.0	604.2
38.3	356.6	422.3	3467.3	598.1
39.3	350.9	415.4	3408.1	591.2
40.3	345.2	408.6	3348.3	584.0
41.3	339.5	401.9	3288.9	576.7
42.3	334.0	395.3	3230.5	569.5
43.3	328.7	389.0	3173.2	562.3
44.3	361.3	427.8	3558.6	597.2
45.3	356.4	421.9	3508.1	590.8
46.3	351.6	416.3	3458.9	584.5
46.8	349.3	413.5	3434.7	581.4
47.3	347.0	410.8	3410.7	578.3
48.3	342.6	405.5	3363.7	572.3
49.3	338.2	400.3	3317.9	566.4
50.3	334.0	395.3	3272.8	560.5
51.3	329.9	390.4	3228.7	554.9
52.3	325.9	385.7	3185.7	549.3
53.3	322.0	381.0	3143.8	543.8
54.0	319.4	377.9	3115.0	540.1
54.3	318.3	376.6	3102.8	538.5
55.3	314.6	372.2	3062.8	533.3
56.3	311.0	368.0	3023.7	528.2
57.3	307.6	363.9	2985.4	523.2
58.3	304.2	359.9	2948.0	518.3
59.3	300.9	356.0	2911.4	513.5
60.3	297.7	352.2	2875.6	508.8
61.3	294.6	348.5	2840.5	504.2
61.9	292.8	346.3	2819.8	501.5

Table 6.4.1-8 (cont.)
Double-Ended Pump Suction Break
Minimum Safeguards
Reflood Mass And Energy Releases
Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time (seconds)	Break Path No. 1* Flow (lbm/sec)	Break Path No. 1* Energy (Thousand Btu/sec)	Break Path No. 2** Flow (lbm/sec)	Break Path No. 2** Energy (Thousand Btu/sec)
62.3	291.6	344.8	2806.2	499.7
63.3	288.6	341.3	2772.5	495.2
64.3	285.7	337.9	2739.5	490.8
65.3	282.9	334.5	2707.1	486.5
66.3	280.1	331.2	2675.4	482.3
67.3	277.4	328.0	2644.2	478.2
68.3	274.7	324.8	2613.6	474.1
69.3	272.1	321.7	2583.5	470.1
70.3	269.6	318.7	2554.0	466.1
71.3	267.1	315.7	2525.0	462.2
72.3	264.6	312.8	2496.4	458.3
73.3	262.2	310.0	2468.4	454.6
74.3	273.4	323.0	274.7	143.1
75.3	435.2	516.1	333.5	240.2
76.3	433.1	513.5	332.2	238.9
77.3	421.5	499.7	327.9	231.8
78.3	409.8	485.7	323.5	224.6
79.3	398.2	471.8	319.2	217.5
80.3	387.9	459.6	315.4	211.3
81.3	378.3	448.1	311.8	205.5
82.3	369.1	437.2	308.4	200.0
83.3	360.4	426.7	305.2	194.8
84.3	352.0	416.8	302.1	189.9
85.3	344.1	407.3	299.2	185.2
86.3	336.5	398.3	296.4	180.8
87.3	329.2	389.6	293.7	176.7
88.3	322.3	381.4	291.0	172.7
90.3	310.1	366.9	286.1	165.8
92.3	299.0	353.7	281.8	159.6
94.3	288.9	341.6	277.8	154.0
95.2	284.6	336.5	276.1	151.6
96.3	279.6	330.6	274.1	148.9

Table 6.4.1-8 (cont.)
Double-Ended Pump Suction Break
Minimum Safeguards
Reflood Mass And Energy Releases
Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time	Break Path No. 1*		Break Path No. 2**	
(seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
98.3	271.1	320.5	270.8	144.2
100.3	263.3	311.3	267.9	140.0
102.3	256.3	302.9	265.1	136.3
104.3	249.8	295.3	262.7	132.9
106.3	244.0	288.4	260.5	129.8
108.3	238.8	282.1	258.5	127.0
110.3	234.0	276.5	256.7	124.5
112.3	229.8	271.5	255.1	122.3
114.3	225.9	266.9	253.7	120.3
116.3	222.5	262.9	252.4	118.6
117.0	221.4	261.5	252.0	118.0
118.3	219.5	259.2	251.2	117.0
120.3	216.7	256.0	250.2	115.6
122.3	214.3	253.2	249.3	114.4
124.3	212.2	250.7	248.5	113.3
126.3	210.3	248.4	247.8	112.3
128.3	208.7	246.5	247.2	111.5
130.3	207.3	244.8	246.7	110.8
132.3	206.1	243.4	246.2	110.1
134.3	205.0	242.1	245.8	109.6
136.3	204.1	241.1	245.5	109.1
138.3	203.4	240.2	245.2	108.7
140.3	202.7	239.4	245.0	108.4
142.4	202.2	238.8	244.8	108.1
144.3	201.8	238.4	244.6	107.9
146.3	201.5	238.0	244.5	107.7
148.3	201.3	237.7	244.4	107.6
150.3	201.2	237.6	244.3	107.5
152.3	201.1	237.4	244.2	107.4
160.3	201.2	237.6	244.2	107.3
162.3	201.3	237.8	244.2	107.4
164.3	201.5	238.0	244.3	107.4

Table 6.4.1-8 (cont.)

Double-Ended Pump Suction Break

Minimum Safeguards

Reflood Mass And Energy Releases

Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time (seconds)	Break Path No. 1*		Break Path No. 2**	
	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
166.3	201.7	238.2	244.3	107.5
168.3	202.0	238.5	244.4	107.6
169.6	202.1	238.7	244.4	107.7
170.3	202.2	238.8	244.5	107.7
172.3	202.5	239.1	244.5	107.8
174.3	202.8	239.5	244.6	107.9
176.3	203.1	239.9	244.7	108.1
178.3	203.4	240.3	244.8	108.2
180.3	203.8	240.7	244.9	108.3
182.3	205.1	242.3	245.8	109.0
184.3	206.4	243.8	247.5	109.7
186.3	207.8	245.4	249.8	110.6
188.3	209.2	247.0	252.7	111.5
190.3	210.5	248.6	255.8	112.4
192.3	211.7	250.1	259.2	113.3
194.3	212.8	251.4	262.7	114.1
196.3	213.7	252.5	266.2	114.9
198.0	214.4	253.2	269.2	115.5

* mass and energy exiting the SG side of the break

** mass and energy exiting the pump side of the break

Table 6.4.1-9

Double-Ended Pump Suction Break- Minimum Safeguards

Principle Parameters During Reflood

Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time Seconds	Flooding Temp °F	Flooding Rate in/sec	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total (Pounds Mass per Second)	Injection Accum (Pounds Mass per Second)	Spill (Pounds Mass per Second)	Enthalpy Btu/lbm
27.0	183.5	.000	.000	.00	.00	.250	.0	.0	.0	.00
28.0	181.6	17.952	.000	.51	1.14	.000	6019.3	6019.3	.0	99.46
28.2	180.6	19.336	.000	.83	1.04	.000	5981.2	5981.2	.0	99.46
28.4	179.8	17.362	.000	1.14	.93	.000	5943.7	5943.7	.0	99.46
28.7	179.6	1.610	.096	1.31	1.46	.249	5865.9	5865.9	.0	99.46
29.1	179.7	2.711	.156	1.37	2.41	.264	5794.9	5794.9	.0	99.46
30.0	180.0	2.359	.301	1.50	4.38	.328	5642.8	5642.8	.0	99.46
31.2	180.4	2.291	.439	1.66	7.34	.349	5443.1	5443.1	.0	99.46
35.0	182.1	2.737	.621	2.00	15.62	.396	4907.9	4907.9	.0	99.46
36.3	182.7	3.753	.658	2.13	16.12	.542	4437.3	4437.3	.0	99.46
38.3	183.9	3.542	.689	2.33	16.12	.539	4230.9	4230.9	.0	99.46
40.3	185.1	3.381	.705	2.50	16.12	.534	4060.2	4060.2	.0	99.46
43.3	187.1	3.205	.719	2.74	16.12	.526	3832.0	3832.0	.0	99.46
44.3	187.9	3.359	.723	2.82	16.12	.549	4262.8	3651.2	.0	97.82
46.8	189.7	3.251	.729	3.00	16.12	.544	4108.0	3489.4	.0	97.73
54.0	195.3	3.016	.738	3.50	16.12	.530	3723.3	3091.4	.0	97.52
61.9	201.5	2.824	.742	4.00	16.12	.516	3377.1	2735.5	.0	97.28
71.3	208.8	2.644	.745	4.55	16.12	.501	3035.0	2384.7	.0	97.00

Table 6.4.1-9 (cont.)

Double-Ended Pump Suction Break- Minimum Safeguards

Principle Parameters During Reflood

Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time Seconds	Flooding Temp °F	Flooding Rate in/sec	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Injection Total (Pounds Mass per Second)	Injection Accum (Pounds Mass per Second)	Injection Spill (Pounds Mass per Second)	Enthalpy Btu/lbm
73.3	210.4	2.610	.745	4.66	16.12	.498	2969.6	2317.8	.0	96.94
75.3	212.0	3.561	.752	4.78	15.92	.602	559.9	.0	.0	88.00
78.3	215.0	3.358	.752	5.00	15.23	.598	573.9	.0	.0	88.00
86.3	223.1	2.838	.751	5.51	13.97	.582	617.4	.0	.0	88.00
95.2	232.0	2.468	.750	6.00	13.22	.567	637.5	.0	.0	88.00
106.3	241.2	2.180	.750	6.54	12.86	.551	650.0	.0	.0	88.00
117.0	248.5	2.017	.751	7.00	12.88	.540	656.3	.0	.0	88.00
130.3	256.0	1.911	.754	7.54	13.18	.532	660.1	.0	.0	88.00
142.4	261.8	1.867	.757	8.00	13.59	.529	661.4	.0	.0	88.00
156.3	267.6	1.846	.762	8.52	14.13	.528	661.8	.0	.0	88.00
169.6	272.4	1.841	.767	9.00	14.68	.529	661.6	.0	.0	88.00
170.3	272.6	1.841	.767	9.03	14.71	.529	661.6	.0	.0	88.00
184.3	277.0	1.855	.772	9.52	15.28	.532	660.7	.0	.0	88.00
198.0	280.8	1.879	.778	10.00	15.71	.542	658.4	.0	.0	88.00

Table 6.4.1-10 Double-Ended Pump Suction Break Minimum Safeguards Post-Reflood Mass And Energy Releases Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
	Break Path No. 1*		Break Path No. 2**	
Time (seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
198.1	248.6	311.7	436.7	156.8
203.1	248.6	311.7	436.6	156.6
208.1	248.8	311.9	436.5	156.3
213.1	247.6	310.4	437.7	156.4
218.1	247.6	310.5	437.6	156.1
223.1	246.4	308.9	438.8	156.2
228.1	246.4	309.0	438.8	156.0
233.1	245.2	307.4	440.1	156.1
238.1	245.2	307.4	440.1	155.9
243.1	245.1	307.3	440.1	155.7
248.1	243.8	305.6	441.5	155.8
253.1	243.7	305.5	441.6	155.6
258.1	243.5	305.3	441.7	155.4
263.1	243.4	305.1	441.9	155.2
268.1	241.9	303.3	443.3	155.4
273.1	241.7	303.0	443.5	155.2
278.1	241.4	302.7	443.8	155.0
283.1	241.1	302.3	444.1	154.9
288.1	240.7	301.8	444.5	154.7
293.1	240.3	301.3	444.9	154.6
298.1	239.9	300.8	445.3	154.5
303.1	239.4	300.1	445.8	154.4
308.1	238.9	299.5	446.4	154.3
313.1	238.3	298.7	447.0	154.2
318.1	237.6	297.9	447.6	154.2
323.1	236.9	297.0	448.3	154.1
328.1	236.2	296.1	449.1	154.1
333.1	235.3	295.1	449.9	154.1
338.1	235.5	295.3	449.7	153.8
343.1	234.6	294.1	450.7	153.8
348.1	234.6	294.1	450.7	153.5

Table 6.4.1-10 (cont.)
Double-Ended Pump Suction Break
Minimum Safeguards
Post-Reflood Mass And Energy Releases
Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time (seconds)	Break Path No. 1*		Break Path No. 2**	
	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
353.1	233.4	292.7	451.8	153.6
358.1	233.3	292.5	452.0	153.4
363.1	233.0	292.1	452.3	153.2
368.1	231.6	290.3	453.7	153.4
373.1	231.0	289.7	454.2	153.3
378.1	231.3	290.0	453.9	153.0
388.1	229.5	287.7	455.7	153.0
393.1	229.3	287.4	456.0	152.8
398.1	228.8	286.8	456.5	152.7
403.1	228.1	286.0	457.1	152.6
408.1	227.3	285.0	457.9	152.6
413.1	227.0	284.6	458.2	152.4
418.1	226.4	283.8	458.9	152.3
423.1	226.0	283.4	459.2	152.2
428.1	225.2	282.4	460.0	152.2
433.1	224.5	281.5	460.7	152.1
438.1	224.4	281.4	460.8	151.9
443.1	223.9	280.8	461.3	151.8
448.1	222.7	279.3	462.5	151.9
453.1	222.6	279.2	462.6	151.7
458.1	221.8	278.0	463.5	151.6
463.1	95.5	119.7	589.7	185.2
615.5	95.5	119.7	589.7	185.2
615.6	99.2	123.3	586.1	177.3
618.1	99.1	123.3	586.2	177.2
1108.1	87.8	109.1	597.4	173.9
1110.0	87.8	109.1	155.6	157.0
1486.1	87.8	109.1	155.6	157.0
1486.2	80.8	93.0	53.3	21.5
3000.0	68.6	78.9	65.6	23.8
3000.1	68.6	78.9	65.6	23.6

Table 6.4.1-10 (cont.)

Double-Ended Pump Suction Break

Minimum Safeguards

Post-Reflood Mass And Energy Releases

Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time (seconds)	Break Path No. 1* Flow (lbm/sec)	Break Path No. 1* Energy (Thousand Btu/sec)	Break Path No. 2** Flow (lbm/sec)	Break Path No. 2** Energy (Thousand Btu/sec)
3600.0	64.8	74.5	69.4	24.3
3600.1	51.4	59.2	82.7	8.8
10000.0	37.4	43.0	96.8	10.3
10000.1	37.2	42.8	96.9	9.8
100000.0	19.9	22.9	114.3	11.5
100000.1	19.7	22.7	114.4	10.6
1000000.0	8.5	9.7	125.7	11.7
1000000.1	8.4	9.7	125.7	11.1
10000000.0	2.6	3.0	131.5	11.6

* mass and energy exiting the SG side of the break

** mass and energy exiting the pump side of the break

Table 6.4.1-11
Double-Ended Pump Suction Break Mass Balance
Minimum Safeguards
Byron Unit 1 & Braidwood Unit 1 (BWI SG)

		Mass Balance						
Time (Seconds)		.00	27.20	27.20	198.03	615.58	1486.11	3600.00
		Mass (Thousand lbm)						
Initial	In RCS and ACC	815.56	815.56	815.56	815.56	815.56	815.56	815.56
Added Mass	Pumped Injection	.00	.00	.00	99.83	385.91	775.16	1058.76
	Total Added	.00	.00	.00	99.83	385.91	775.16	1058.76
*** TOTAL AVAILABLE ***		815.56	815.56	815.56	915.39	1201.47	1590.72	1874.32
Distribution	Reactor Coolant	567.41	50.04	80.55	142.66	142.66	142.66	142.66
	Accumulator	248.14	197.38	166.87	.00	.00	.00	.00
	Total Contents	815.56	247.42	247.42	142.66	142.66	142.66	142.66
Effluent	Break Flow	.00	568.12	568.12	761.06	1047.14	1477.06	1760.67
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	568.12	568.12	761.06	1047.14	1477.06	1760.67
TOTAL ACCOUNTABLE		815.56	815.54	815.54	903.72	1189.80	1619.72	1903.33

Table 6.4.1-12

Double-Ended Pump Suction Break Energy Balance

Minimum Safeguards

Byron Unit 1 & Braidwood Unit 1 (BWI SG)

		Energy Balance						
Time (Seconds)		.00	27.20	27.20	198.03	615.58	1486.11	3600.00
		Energy (Million Btu)						
Initial Energy	In RCS, ACC, S GEN	87.63	987.63	987.63	987.63	987.63	987.63	987.63
Added Energy	Pumped Injection	.00	.00	.00	8.79	33.96	69.17	99.44
	Decay Heat	.00	8.83	8.83	30.01	69.81	136.63	263.71
	Heat From Secondary	.00	.04	.04	.04	5.70	15.65	15.65
	Total Added	.00	8.86	8.86	38.83	109.47	221.46	378.80
***** TOTAL AVAILABLE *****		87.63	996.49	996.49	1026.46	1097.09	1209.09	1366.42
Distribution	Reactor Coolant	41.36	11.41	14.45	37.97	37.97	37.97	37.97
	Accumulator	24.68	19.63	16.60	.00	.00	.00	.00
	Core Stored	23.98	13.02	13.02	4.91	4.71	4.31	3.33
	Primary Metal	169.14	160.62	160.62	134.78	98.97	74.06	57.38
	Secondary Metal	119.11	119.15	119.15	109.39	87.74	61.15	47.40
	Steam Generator	309.36	309.47	309.47	280.13	224.67	163.37	129.46
	Total Contents	987.63	633.30	633.30	567.18	54.06	340.86	275.53
Effluent	Break Flow	.00	362.60	362.60	449.20	632.95	858.81	1083.65
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	362.60	362.60	449.20	632.95	858.81	1083.65
***** TOTAL ACCOUNTABLE *****		987.63	995.90	995.90	1016.37	1087.01	1199.67	1359.18

Table 6.4.1-13

Double-Ended Pump Suction Break Maximum Safeguards

Blowdown Mass And Energy Releases

Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time (seconds)	Break Path No. 1*		Break Path No. 2**	
	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
.00000	.0	.0	.0	.0
.00111	84856.4	47801.1	43218.0	24297.4
.101	42559.9	23971.4	22324.8	12538.6
.201	42838.3	24219.8	24793.7	13937.7
.302	43224.9	24558.8	24956.9	14042.5
.402	43629.3	24936.1	24131.6	13591.6
.502	44059.6	25358.0	23018.1	12973.9
.601	44503.7	25817.0	22086.5	12453.8
.701	44766.3	26186.9	21271.1	11996.9
.801	44702.0	26364.3	20677.2	11664.8
.902	44293.3	26328.8	20266.8	11437.2
1.00	43606.9	26110.9	20049.6	11317.6
1.10	42759.5	25784.9	19935.4	11255.1
1.20	41898.8	25438.2	19892.2	11231.9
1.30	41061.4	25101.1	19871.4	11220.8
1.40	40249.1	24776.4	19861.7	11215.1
1.50	39450.6	24454.2	19880.1	11225.2
1.60	38642.2	24124.2	19935.6	11256.3
1.70	37855.4	23803.0	20000.0	11292.4
1.80	37099.3	23496.8	20046.8	11318.2
1.90	36353.5	23197.3	20070.2	11330.6
2.00	35584.0	22881.8	20050.4	11318.4
2.10	34788.4	22551.9	20008.5	11293.7
2.20	33983.5	22215.3	19932.5	11249.8
2.30	33099.2	21822.8	19755.4	11148.0
2.40	32225.5	21434.5	19361.7	10924.0
2.50	31325.8	21022.5	19185.7	10824.3
2.60	30441.3	20611.5	19038.8	10740.8
2.70	29572.6	20200.4	18854.7	10635.9
2.80	28678.3	19758.4	18628.6	10507.2
2.90	27509.7	19107.6	18400.1	10377.3
3.00	25599.5	17911.0	18203.1	10265.6

Table 6.4.1-13 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Blowdown Mass And Energy Releases Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
	Break Path No. 1*		Break Path No. 2**	
Time (seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
3.10	23103.9	16272.7	18016.5	10160.0
3.20	22045.5	15653.9	17811.6	10043.9
3.30	21224.3	15157.2	17595.1	9921.2
3.40	20058.5	14380.3	17405.2	9814.0
3.50	19288.0	13883.3	17237.8	9719.7
3.60	18601.3	13430.8	17074.0	9627.5
3.70	17924.0	12974.3	16909.1	9534.8
3.80	17326.2	12570.4	16750.6	9444.7
3.90	16806.1	12218.4	16610.9	9363.1
4.00	16348.7	11908.1	16483.5	9287.8
4.20	15583.5	11384.6	16246.6	9149.5
4.40	15011.5	10988.0	16030.4	9024.5
4.60	14536.4	10650.0	15823.0	8904.5
4.80	14180.7	10389.6	15640.0	8799.1
5.00	13898.4	10171.3	15460.1	8696.8
5.20	13707.0	10009.6	15287.4	8600.1
5.40	13584.4	9887.5	15108.5	8501.5
5.60	13536.4	9810.5	14928.2	8403.1
5.80	13561.8	9777.2	14753.1	8308.4
6.00	13687.6	9806.4	14588.4	8220.1
6.20	14046.6	9992.4	14163.8	7983.7
6.40	13823.6	9748.1	15264.5	8622.7
6.60	13837.2	9861.2	15698.9	8863.1
6.80	12864.8	9629.3	15551.5	8786.2
7.00	11792.7	9136.4	15477.6	8748.4
7.20	11895.0	9150.7	15256.8	8626.0
7.40	12353.9	9364.7	14994.7	8480.4
7.60	12688.5	9505.4	14864.6	8408.9
7.80	13046.4	9674.0	14692.3	8311.1
8.00	13658.2	10006.0	14424.1	8156.9
8.20	14010.7	10113.0	14196.0	8024.4
8.40	13232.0	9447.0	14053.1	7940.2

Table 6.4.1-13 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Blowdown Mass And Energy Releases Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
	Break Path No. 1*		Break Path No. 2**	
Time (seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
8.60	11978.7	8556.2	14036.9	7928.0
8.80	11808.4	8473.5	13999.9	7902.8
9.00	11990.9	8569.8	13697.4	7726.3
9.20	11695.6	8326.8	13551.6	7639.5
9.40	11386.8	8127.9	13625.3	7678.3
9.60	11504.0	8226.8	13450.5	7575.6
9.80	11557.4	8236.0	13121.3	7385.6
10.0	11241.7	7986.7	13102.2	7372.7
10.2	11049.1	7859.2	13097.0	7367.7
10.4	11077.6	7874.9	12758.1	7173.1
10.6	10562.4	7501.7	12647.7	7109.1
10.8	10036.8	7179.6	12822.0	7207.0
11.0	9906.9	7144.3	12482.2	7011.9
11.2	9631.2	6995.6	12309.8	6912.8
11.4	9388.7	6877.5	12447.3	6989.5
11.6	9227.6	6800.4	12127.3	6805.9
11.8	9023.2	6678.0	12015.1	6741.3
12.0	8902.4	6607.4	12028.7	6747.7
12.2	8743.0	6500.3	11752.6	6589.9
12.4	8585.2	6402.5	11733.5	6578.5
12.6	8447.1	6319.1	11611.2	6508.1
12.8	8267.8	6210.9	11460.5	6422.2
13.0	8129.3	6138.5	11409.1	6392.6
13.2	7969.6	6049.7	11252.9	6303.6
13.4	7828.2	5976.2	11176.3	6260.1
13.6	7694.0	5903.4	11048.1	6187.4
13.8	7562.6	5829.5	10950.2	6132.2
14.0	7444.9	5762.2	10834.4	6067.1
14.2	7324.4	5690.5	10725.0	6005.9
14.4	7207.1	5621.6	10605.1	5939.2
14.6	7077.9	5544.8	10470.7	5864.6
14.8	6939.4	5463.2	10342.1	5793.6
15.0	6795.7	5378.3	10205.2	5718.6

Table 6.4.1-13 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Blowdown Mass And Energy Releases Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
	Break Path No. 1*		Break Path No. 2**	
Time (seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
15.2	6655.2	5293.0	10082.0	5651.7
15.4	6526.2	5210.3	9955.8	5583.7
15.6	6414.4	5134.6	9840.0	5522.0
15.8	6317.2	5065.3	9724.3	5460.9
16.0	6231.6	5003.2	9612.3	5402.4
16.2	6152.1	4946.7	9504.2	5346.8
16.4	6073.7	4894.6	9395.1	5291.4
16.6	5995.1	4847.6	9290.8	5239.6
16.8	5913.8	4805.4	9184.9	5187.8
17.0	5828.1	4768.1	9079.8	5137.7
17.2	5738.5	4738.3	8977.7	5090.8
17.4	5646.2	4712.3	8856.7	5035.5
17.6	5563.9	4706.1	8745.5	4991.5
17.8	5488.5	4731.4	8449.4	4856.2
18.0	5371.3	4768.5	8188.2	4741.8
18.2	5186.1	4789.8	7999.0	4663.5
18.4	4881.6	4726.5	7420.9	4348.4
18.6	4537.8	4639.1	7143.8	4197.8
18.8	4203.4	4537.1	6821.8	3967.7
19.0	3878.9	4404.3	6529.8	3716.0
19.2	3586.7	4234.7	6310.8	3482.1
19.6	3088.1	3788.5	5781.0	2977.3
19.8	2858.5	3528.6	5479.2	2744.9
20.0	2652.5	3288.7	5191.4	2544.2
20.2	2461.0	3062.0	4909.9	2359.5
20.6	2123.5	2656.6	4401.7	2044.1
20.8	1972.2	2472.9	4194.7	1918.7
21.0	1839.8	2311.5	3992.9	1800.1
21.2	1716.5	2160.7	3804.3	1692.1
21.6	1536.2	1940.0	2962.1	1248.6
21.8	1455.5	1840.0	2632.3	1055.6
22.0	1361.9	1723.7	2770.9	1059.7
22.2	1268.5	1607.5	2812.2	1046.2

Table 6.4.1-13 (cont.)
Double-Ended Pump Suction Break Maximum Safeguards
Blowdown Mass And Energy Releases
Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time (seconds)	Break Path No. 1* Flow (lbm/sec)	Break Path No. 1* Energy (Thousand Btu/sec)	Break Path No. 2** Flow (lbm/sec)	Break Path No. 2** Energy (Thousand Btu/sec)
22.4	1176.9	1493.3	2960.1	1081.2
22.6	1090.6	1385.0	3358.3	1205.5
22.8	1000.0	1272.0	3111.1	1102.2
23.0	914.5	1164.5	2545.1	896.8
23.2	834.9	1064.1	2356.6	827.9
23.4	761.4	971.2	2072.6	725.1
23.6	694.4	886.3	1792.1	625.0
23.8	629.7	804.2	1432.6	499.5
24.0	588.9	752.7	1245.3	433.4
24.2	555.5	710.4	1023.9	350.6
24.4	511.1	653.8	714.4	240.5
24.6	468.8	600.0	445.6	148.2
24.8	433.5	555.1	261.8	85.1
25.0	403.2	516.6	.0	.0
25.2	360.3	461.7	.0	.0
25.4	328.9	421.7	.0	.0
25.6	283.5	363.6	.0	.0
25.8	230.0	295.2	.0	.0
26.0	205.3	263.6	.0	.0
26.2	149.9	192.6	.0	.0
26.4	98.1	126.3	.0	.0
26.6	49.9	64.4	.0	.0
26.8	.0	.0	.0	.0

* mass and energy exiting the SG side of the break

** mass and energy exiting the pump side of the break

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Table 6.4.1-14
Double-Ended Pump Suction Break
Maximum Safeguards
Reflood Mass And Energy Releases
Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time (seconds)	Break Path No. 1* Flow (lbm/sec)	Break Path No. 1* Energy (Thousand Btu/sec)	Break Path No. 2** Flow (lbm/sec)	Break Path No. 2** Energy (Thousand (Btu/sec)
26.8	.0	.0	.0	.0
27.4	.0	.0	.0	.0
27.6	.0	.0	.0	.0
27.7	.0	.0	.0	.0
27.8	.0	.0	.0	.0
27.9	.0	.0	.0	.0
28.0	7.4	8.8	.0	.0
28.1	46.9	55.3	.0	.0
28.2	18.7	22.1	.0	.0
28.3	12.6	14.8	.0	.0
28.4	14.4	16.9	.0	.0
28.5	20.0	23.6	.0	.0
28.6	26.6	31.4	.0	.0
28.7	34.5	40.7	.0	.0
28.8	39.5	46.5	.0	.0
28.9	44.7	52.7	.0	.0
29.0	48.5	57.2	.0	.0
29.1	52.1	61.4	.0	.0
29.2	55.5	65.5	.0	.0
29.3	58.8	69.4	.0	.0
29.4	62.0	73.1	.0	.0
29.5	63.5	74.9	.0	.0
29.5	65.1	76.7	.0	.0
29.6	68.0	80.2	.0	.0
29.7	70.9	83.6	.0	.0
29.8	73.7	86.9	.0	.0
30.8	98.0	115.6	.0	.0
31.8	118.0	139.2	.0	.0
32.8	142.5	168.2	422.7	37.2
33.8	181.8	214.6	1079.2	130.1
34.3	432.6	512.9	4323.7	628.5

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Table 6.4.1-14 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass And Energy Releases Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
	Break Path No. 1*		Break Path No. 2**	
Time (seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand (Btu/sec)
34.9	489.2	580.6	4863.7	734.2
35.9	490.5	582.2	4871.6	742.6
36.9	484.9	575.5	4821.3	737.0
37.9	478.6	568.0	4765.0	730.2
38.6	474.2	562.7	4724.2	725.1
38.9	472.3	560.4	4706.6	722.9
39.9	465.9	552.8	4647.8	715.5
40.9	459.7	545.4	4589.5	708.0
41.9	453.6	538.1	4532.2	700.7
42.9	447.7	531.0	4476.1	693.4
43.9	442.0	524.2	4421.3	686.3
44.0	441.4	523.5	4415.8	685.6
44.9	436.5	517.6	4367.8	679.3
45.9	431.1	511.1	4315.7	672.5
46.9	425.9	504.9	4265.1	665.9
47.9	420.9	498.9	4215.7	659.4
48.9	416.0	493.1	4167.7	653.2
49.9	411.3	487.5	4121.0	647.0
50.2	409.9	485.8	4107.2	645.2
50.9	406.7	482.0	4075.4	641.0
51.9	402.3	476.7	4031.1	635.2
52.9	398.0	471.6	3987.8	629.5
53.9	393.8	466.6	3945.7	623.9
54.9	389.7	461.7	3904.6	618.5
55.9	385.8	457.0	3864.4	613.2
56.9	381.9	452.4	3825.3	608.0
57.0	381.6	452.0	3821.4	607.5
57.9	378.2	448.0	3787.0	602.9
58.9	374.6	443.6	3749.6	598.0
59.9	371.0	439.4	3713.1	593.1
60.9	367.5	435.3	3677.3	588.3
61.9	364.1	431.2	3642.3	583.7

Table 6.4.1-14 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass And Energy Releases Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
	Break Path No. 1*		Break Path No. 2**	
Time (seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand (Btu/sec)
62.9	360.8	427.3	3608.1	579.1
63.9	357.6	423.4	3574.6	574.6
64.9	354.4	419.7	3541.8	570.2
65.9	351.3	416.0	3509.6	565.8
66.9	348.3	412.4	3478.1	561.6
67.9	345.3	408.8	3447.1	557.4
68.9	342.4	405.4	3416.8	553.3
69.9	339.6	402.0	3387.0	549.2
70.9	336.8	398.6	3357.8	545.2
71.9	334.0	395.4	3329.1	541.3
72.0	333.8	395.0	3326.3	540.9
72.9	331.3	392.2	3300.9	537.4
73.9	328.7	389.0	3273.2	533.6
74.9	326.1	385.9	3246.0	529.9
75.9	323.5	382.9	3219.2	526.2
76.9	321.0	379.9	3192.9	522.5
77.9	318.5	376.9	3167.0	519.0
78.9	181.5	214.2	1332.4	279.2
79.9	180.8	213.4	1334.0	278.9
80.9	180.5	213.1	1334.6	278.8
81.9	180.2	212.8	1335.2	278.6
82.9	179.9	212.4	1335.8	278.5
83.9	179.7	212.1	1336.5	278.3
84.9	179.4	211.8	1337.1	278.2
85.9	179.2	211.5	1337.8	278.0
86.9	178.9	211.2	1338.5	277.9
87.9	178.6	210.9	1339.1	277.8
89.9	178.1	210.3	1340.5	277.5
91.7	177.7	209.8	1341.7	277.2
91.9	177.6	209.7	1341.8	277.2
93.9	177.1	209.1	1343.2	276.9
95.9	176.6	208.5	1344.6	276.6

Table 6.4.1-14 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass And Energy Releases Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
	Break Path No. 1*		Break Path No. 2**	
Time (seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand (Btu/sec)
97.9	176.1	207.9	1345.9	276.4
99.9	175.6	207.4	1347.3	276.1
101.9	175.2	206.8	1348.7	275.8
103.9	174.7	206.2	1350.1	275.5
105.9	174.2	205.6	1351.5	275.2
107.9	173.7	205.1	1352.9	275.0
109.9	173.2	204.5	1354.3	274.7
111.9	172.8	204.0	1355.7	274.4
113.9	172.3	203.4	1357.0	274.1
114.9	172.1	203.1	1357.7	274.0
115.9	171.8	202.9	1358.4	273.8
117.9	171.4	202.3	1359.8	273.5
119.9	170.9	201.8	1361.1	273.2
121.9	170.5	201.2	1362.5	272.9
123.9	170.0	200.7	1363.8	272.6
125.9	169.5	200.1	1365.1	272.3
127.9	169.1	199.6	1366.5	272.0
129.9	168.6	199.1	1367.8	271.6
131.9	168.2	198.5	1369.1	271.3
133.9	167.8	198.0	1370.4	271.0
135.9	167.3	197.5	1371.6	270.7
137.9	166.9	197.0	1372.9	270.3
139.9	166.4	196.5	1374.2	270.0
140.1	166.4	196.4	1374.3	270.0
141.9	166.0	196.0	1375.4	269.6
143.9	165.6	195.4	1376.7	269.3
145.9	165.1	194.9	1377.9	268.9
147.9	164.7	194.4	1379.2	268.6
149.9	164.3	193.9	1380.4	268.2
151.9	163.9	193.4	1381.6	267.9
153.9	163.4	192.9	1382.8	267.5
155.9	163.0	192.4	1384.0	267.2

Table 6.4.1-14 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass And Energy Releases Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
	Break Path No. 1*		Break Path No. 2**	
Time (seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand (Btu/sec)
157.9	162.6	191.9	1385.2	266.8
159.9	162.2	191.4	1386.4	266.4
161.9	161.8	190.9	1387.6	266.0
163.9	161.4	190.5	1388.8	265.7
165.9	161.0	190.0	1390.0	265.3
167.5	160.6	189.6	1390.9	265.0
167.9	160.5	189.5	1391.2	264.9
169.9	160.1	189.0	1392.3	264.5
171.9	159.7	188.5	1393.5	264.1
173.9	159.3	188.1	1394.6	263.7
175.9	158.9	187.6	1395.8	263.3
177.9	158.5	187.1	1396.9	262.9
179.9	158.2	186.7	1398.1	262.5
181.9	157.8	186.2	1399.2	262.1
183.9	157.4	185.7	1400.3	261.7
185.9	157.0	185.3	1401.4	261.3
187.9	156.6	184.8	1402.5	260.9
189.9	156.2	184.4	1403.6	260.5
191.9	155.9	183.9	1404.7	260.1
193.9	155.5	183.5	1405.8	259.7
195.9	155.1	183.1	1406.9	259.3
197.7	154.8	182.7	1407.9	258.9

* mass and energy exiting the SG side of the break

** mass and energy exiting the pump side of the break

Table 6.4.1-15

Double-Ended Pump Suction Break - Maximum Safeguards

Principle Parameters During Reflood

Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time	Flooding			Core	Downcomer		Total	Injection Accum	Spill	
	Temp	Rate	Carryover	Height	Height	Flow				Enthalpy
Seconds	°F	in/sec	Fraction	(ft)	(ft)	Frac	(Pounds Mass per Second)			Btu/lbm
26.8	180.8	.000	.000	.00	.00	.250	.0	.0	.0	.00
27.6	179.0	8.407	.000	.53	1.18	.000	6203.3	6203.3	.0	99.46
27.8	178.0	9.878	.000	.85	1.08	.000	6162.1	6162.1	.0	99.46
27.9	177.5	9.477	.000	1.01	1.02	.000	6141.8	6141.8	.0	99.46
28.3	177.1	1.765	.101	1.31	1.58	.250	6037.9	6037.9	.0	99.46
28.7	177.2	2.673	.167	1.38	2.55	.281	5961.7	5961.7	.0	99.46
29.5	177.4	2.385	.300	1.50	4.41	.328	5811.9	5811.9	.0	99.46
30.8	177.9	2.318	.446	1.66	7.63	.350	5585.8	5585.8	.0	99.46
34.3	179.4	4.375	.630	2.00	16.04	.584	6269.8	4580.1	.0	96.37
34.9	179.7	4.637	.650	2.09	16.12	.599	6042.0	4352.6	.0	96.26
35.9	180.1	4.473	.675	2.21	16.12	.598	5908.4	4219.1	.0	96.18

TABLE 6.4.1-15 (cont.)
Double-Ended Pump Suction Break - Maximum Safeguards
Principle Parameters During Reflood
Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time Seconds	Flooding Temp °F	Flooding Rate in/sec	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total (Pounds Mass per Second)	Injection Accum (Pounds Mass per Second)	Spill (Pounds Mass per Second)	Enthalpy Btu/lbm
38.6	181.7	4.148	.708	2.51	16.12	.596	5651.4	3962.0	.0	96.03
44.0	185.2	3.795	.731	3.00	16.12	.587	5238.0	3548.4	.0	95.76
50.2	189.7	3.538	.740	3.50	16.12	.578	4858.9	3169.2	.0	95.47
57.0	194.9	3.329	.744	4.00	16.12	.569	4518.3	2828.4	.0	95.17
64.9	200.9	3.136	.747	4.54	16.12	.559	4189.2	2499.1	.0	94.84
72.0	206.1	2.989	.748	5.00	16.12	.551	3936.9	2246.8	.0	94.54
78.9	210.9	2.135	.739	5.42	16.12	.424	1690.7	.0	.0	88.00
80.9	212.1	2.124	.740	5.51	16.12	.424	1690.7	.0	.0	88.00
91.7	219.4	2.074	.742	6.00	16.12	.427	1690.7	.0	.0	88.00
103.9	228.6	2.018	.744	6.54	16.12	.430	1690.7	.0	.0	88.00
114.9	236.9	1.967	.747	7.00	16.12	.432	1690.7	.0	.0	88.00

TABLE 6.4.1-15 (cont.)

Double-Ended Pump Suction Break - Maximum Safeguards

Principle Parameters During Reflood

Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time Seconds	Flooding Temp °F	Flooding Rate in/sec	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total (Pounds Mass per Second)	Injection Accum (Pounds Mass per Second)	Spill (Pounds Mass per Second)	Enthalpy Btu/lbm
127.9	245.5	1.909	.750	7.53	16.12	.436	1690.7	.0	.0	88.00
140.1	252.4	1.856	.753	8.00	16.12	.440	1690.7	.0	.0	88.00
153.9	259.1	1.797	.757	8.52	16.12	.444	1690.7	.0	.0	88.00
167.5	264.8	1.741	.760	9.00	16.12	.448	1690.7	.0	.0	88.00
183.9	270.6	1.675	.764	9.56	16.12	.454	1690.7	.0	.0	88.00
197.7	274.8	1.622	.767	10.00	16.12	.460	1690.7	.0	.0	88.00

Table 6.4.1-16 Double-Ended Pump Suction Break Maximum Safeguards Post-Reflood Mass And Energy Releases Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
	Break Path No. 1 Energy*		Break Path No. 2 Flow**	
Time (seconds)	Flow (lbm/sec)	Flow (Thousand Btu/sec)	Flow (lbm/sec)	Flow (Thousand Btu/sec)
197.8	166.5	209.5	1524.4	257.5
202.8	165.9	208.8	1525.0	257.4
207.8	165.4	208.2	1525.5	257.3
212.8	164.9	207.6	1525.9	257.3
217.8	165.9	208.8	1525.0	256.8
222.8	165.4	208.2	1525.5	256.7
227.8	164.9	207.5	1525.9	256.7
232.8	165.8	208.7	1525.0	256.2
237.8	165.3	208.1	1525.5	256.1
242.8	164.8	207.5	1526.0	256.0
247.8	165.8	208.7	1525.1	255.6
252.8	165.3	208.0	1525.6	255.5
257.8	164.8	207.4	1526.1	255.4
262.8	164.2	206.7	1526.6	255.4
267.8	165.2	207.9	1525.7	254.9
272.8	164.6	207.2	1526.2	254.9
277.8	164.1	206.6	1526.7	254.8
282.8	165.0	207.7	1525.8	254.3
287.8	164.5	207.0	1526.4	254.3
292.8	163.9	206.3	1526.9	254.2
297.8	164.8	207.4	1526.0	253.8
302.8	164.3	206.8	1526.6	253.7
307.8	163.7	206.1	1527.1	253.6
312.8	164.6	207.1	1526.3	253.2
317.8	164.0	206.4	1526.8	253.1
322.8	163.5	205.7	1527.4	253.0
327.8	164.3	206.8	1526.6	252.6
332.8	163.7	206.1	1527.1	252.6
337.8	164.5	207.1	1526.3	252.1
342.8	163.9	206.3	1526.9	252.1
347.8	163.4	205.6	1527.5	252.0
352.8	164.1	206.6	1526.7	251.6
357.8	163.5	205.9	1527.3	251.5
362.8	163.0	205.1	1527.9	251.5

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Table 6.4.1-16 (cont.)
Double-Ended Pump Suction Break Maximum Safeguards
Post-Reflood Mass and Energy Releases
Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time (seconds)	Break Path No. 1 Energy*		Break Path No. 2 Energy**	
	Flow (lbm/sec)	Flow (Thousand Btu/sec)	Flow (lbm/sec)	Flow (Thousand Btu/sec)
367.8	163.7	206.0	1527.1	251.0
372.8	163.1	205.3	1527.7	251.0
377.8	163.8	206.2	1527.0	250.6
382.8	163.2	205.4	1527.6	250.5
387.8	162.6	204.7	1528.2	250.5
392.8	163.3	205.5	1527.5	250.1
397.8	162.7	204.7	1528.2	250.0
402.8	163.4	205.7	1527.4	249.6
407.8	162.9	205.1	1527.9	249.5
412.8	162.5	204.5	1528.4	249.4
417.8	163.2	205.5	1527.6	249.0
422.8	162.7	204.8	1528.1	248.9
427.8	162.2	204.2	1528.6	248.8
432.8	163.0	205.2	1527.8	248.4
437.8	162.5	204.5	1528.4	248.3
442.8	163.2	205.4	1527.6	247.9
447.8	162.7	204.8	1528.2	247.9
452.8	162.2	204.1	1528.7	247.8
457.8	162.8	205.0	1528.0	247.4
462.8	162.3	204.3	1528.5	247.3
467.8	163.0	205.1	1527.9	253.7
472.8	162.4	204.4	1528.4	253.7
477.8	161.8	203.7	1529.0	253.6
482.8	162.5	204.5	1528.4	253.2
487.8	161.9	203.7	1529.0	253.1
492.8	162.5	204.5	1528.4	252.7
497.8	161.9	203.7	1529.0	252.6
502.8	162.4	204.4	1528.4	252.2
507.8	161.8	203.6	1529.1	252.1
512.8	162.3	204.3	1528.5	251.8
517.8	161.7	203.5	1529.2	251.7
522.8	162.1	204.1	1528.7	251.3
527.8	161.5	203.3	1529.4	251.2

Table 6.4.1-16 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Post-Reflood Mass and Energy Releases Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
Time (seconds)	Break Path No. 1 Energy*		Break Path No. 2 Energy**	
	Flow (lbm/sec)	Flow (Thousand Btu/sec)	Flow (lbm/sec)	Flow (Thousand Btu/sec)
532.8	161.9	203.8	1528.9	250.9
537.8	162.3	204.3	1528.5	250.5
542.8	161.6	203.5	1529.2	250.5
547.8	162.0	203.9	1528.8	250.1
552.8	161.3	203.0	1529.5	250.1
557.8	161.6	203.4	1529.2	249.7
562.8	161.9	203.8	1528.9	249.4
567.8	161.2	202.8	1529.7	249.4
572.8	161.4	203.2	1529.4	249.0
577.8	161.6	203.5	1529.2	248.7
582.8	161.8	203.7	1529.0	248.4
587.8	161.0	202.6	1529.9	248.4
592.8	161.1	202.8	1529.7	248.1
597.8	161.2	202.9	1529.6	247.8
602.8	161.3	203.1	1529.5	247.5
607.8	161.4	203.2	1529.4	247.3
612.8	161.5	203.2	1529.4	247.0
617.8	161.5	203.2	1529.4	246.8
622.8	161.4	203.2	1529.4	246.5
627.8	161.3	203.1	1529.5	246.3
632.8	161.2	202.9	1529.6	246.1
637.8	161.0	202.7	1529.8	245.9
642.8	160.8	202.4	1530.1	245.7
647.8	160.5	202.0	1530.4	245.5
652.8	161.0	202.7	1529.8	245.1
657.8	160.6	202.1	1530.3	245.0
662.8	160.9	202.6	1529.9	244.6
667.8	160.3	201.8	1530.5	244.5
672.8	160.5	202.0	1530.3	244.2
677.8	160.5	202.0	1530.3	243.9
682.8	160.4	201.9	1530.4	243.7
687.8	160.9	202.6	1529.9	243.3
687.8	160.9	202.6	1529.9	243.3

Table 6.4.1-16 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Post-Reflood Mass and Energy Releases Byron Unit 1 & Braidwood Unit 1 (BWI SG)				
	Break Path No. 1 Energy*		Break Path No. 2 Energy**	
Time (seconds)	Flow (lbm/sec)	Flow (Thousand Btu/sec)	Flow (lbm/sec)	Flow (Thousand Btu/sec)
692.8	160.5	202.0	1530.3	243.2
695.0	156.7	197.2	1451.6	339.7
700.0	156.9	197.5	1451.3	339.4
705.0	156.9	197.5	1451.3	339.1
710.0	156.7	197.2	1451.6	338.9
715.0	87.0	109.5	1521.3	357.3
946.9	87.0	109.5	1521.3	357.3
947.0	91.9	114.5	1516.3	354.5
950.0	91.8	114.5	1516.4	354.3
1512.8	91.8	114.5	1516.4	354.3
1512.9	81.6	93.9	1526.6	251.8
1600.0	80.6	92.8	1527.6	252.0
1600.1	80.6	92.8	1527.6	271.3
3600.0	65.9	75.9	1542.3	273.9
3600.1	54.4	62.6	1553.8	254.8
7000.0	44.0	50.6	1564.2	256.5
7000.1	42.7	49.1	1565.5	209.8
10000.0	38.4	44.2	1569.8	210.4
10000.1	37.8	43.4	1570.5	182.2
100000.0	20.2	23.2	1588.1	184.2
100000.1	20.0	23.0	1588.2	171.5
1000000.0	8.6	9.9	1599.7	172.8
1000000.1	8.6	9.9	1599.7	169.6
10000000.0	2.7	3.1	1605.6	170.2

* mass and energy exiting the SG side of the break

** mass and energy exiting the pump side of the break

Table 6.4.1-17
Double-Ended Pump Suction Break Mass Balance
Maximum Safeguards
Byron Unit 1 & Braidwood Unit 1 (BWI SG)

		Mass Balance						
Time (Seconds)		.00	26.80	26.80	97.71	47.04	1512.83	3600.00
		Mass (Thousand lbm)						
Initial	In RCS and ACC	815.56	815.56	815.56	815.56	815.56	815.56	815.56
Added Mass	Pumped Injection	.00	.00	.00	280.01	1526.04	2435.96	5792.64
	Total Added	.00	.00	.00	280.01	1526.04	2435.96	5792.64
***** TOTAL AVAILABLE *****		815.56	815.56	815.56	1095.56	2341.60	3251.51	6608.20
Distribution	Reactor Coolant	567.41	56.14	84.14	148.82	148.82	148.82	148.82
	Accumulator	248.14	200.63	172.62	.00	.00	.00	.00
	Total Contents	815.56	256.76	256.76	148.82	148.82	148.82	148.82
Effluent	Break Flow	.00	570.25	570.25	946.54	2192.58	3102.40	6459.09
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	570.25	570.25	946.54	2192.58	3102.40	6459.09
***** TOTAL ACCOUNTABLE *****		815.56	827.01	827.01	1095.36	2341.39	3251.22	6607.90

Table 6.4.1-18
Double-Ended Pump Suction Break Energy Balance
Maximum Safeguards
Byron Unit 1 & Braidwood Unit 1 (BWI SG)

		Energy Balance						
Time (Seconds)		.00	26.80	26.80	197.71	947.04	1512.83	3600.00
		Energy (Million Btu)						
Initial Energy	In RCS, ACC, S GEN	987.47	987.47	987.47	987.47	987.47	987.47	987.47
Added Energy	Pumped Injection	.00	.00	.00	24.64	160.23	298.54	847.36
	Decay Heat	.00	8.75	8.75	29.96	96.96	138.47	263.69
	Heat From Secondary	.00	.03	.03	.03	10.18	15.94	15.94
	Total Added	.00	8.78	8.78	54.64	267.38	452.96	1126.99
***** TOTAL AVAILABLE *****		987.47	996.25	996.25	1042.11	1254.85	1440.43	2114.46
Distribution	Reactor Coolant	341.36	12.17	14.95	38.95	38.95	38.95	38.95
	Accumulator	24.68	19.95	17.17	.00	.00	.00	.00
	Core Stored	23.82	12.94	12.94	4.91	4.71	4.41	3.33
	Primary Metal	169.14	160.64	160.64	132.46	91.34	75.39	57.86
	Secondary Metal	119.11	119.14	119.14	109.53	78.90	62.07	47.96
	Steam Generator	309.36	309.44	309.44	280.29	203.89	165.86	131.12
	Total Contents	987.47	634.28	634.28	566.14	417.79	346.69	279.23
Effluent	Break Flow	.00	362.39	362.39	466.97	828.06	1077.70	1821.64
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	362.39	362.39	466.97	828.06	1077.70	1821.64
***** TOTAL ACCOUNTABLE *****		987.47	996.67	996.67	1033.11	1245.85	1424.40	2100.88

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Table 6.4.1-19

Double-Ended Hot Leg Break

Sequence of Events

Byron Unit 1 & Braidwood Unit 1 (BWI SG)

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.9	Low Pressurizer Pressure SI Setpoint - 1715 psia reached in blowdown
15.3	Broken Loop Accumulator Begins Injecting Water
15.5	Intact Loop Accumulator Begins Injecting Water
25.6	End of Blowdown Phase

Table 6.4.1-20 Double-Ended Pump Suction Break Minimum Safeguards Sequence of Events Byron Unit 1 & Braidwood Unit 1 (BWI SG)	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.6	Low Pressurizer Pressure SI Setpoint - 1715 psia reached in blowdown
18.3	Broken Loop Accumulator Begins Injecting Water
18.6	Intact Loop Accumulator Begins Injecting Water
27.2	End of Blowdown Phase
43.6	Safety Injection Begins
73.72	Broken Loop Accumulator Water Injection Ends
74.22	Intact Loop Accumulator Water Injection Ends
198.02	End of Reflood Phase
1110.0	Cold Leg Recirculation Begins
1.0E+07	Transient Modeling Terminated

Table 6.4.1-21 Double-Ended Pump Suction Break Maximum Safeguards Sequence of Events Byron Unit 1 & Braidwood Unit 1 (BWI SG)	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.61	Low Pressurizer Pressure SI Setpoint - 1715 psia reached in blowdown
18.3	Broken Loop Accumulator Begins Injecting Water
18.6	Intact Loop Accumulator Begins Injecting Water
26.8	End of Blowdown Phase
3.61	Safety Injection Begins
78.01	Broken Loop Accumulator Water Injection Ends
78.46	Intact Loop Accumulator Water Injection Ends
197.7	End of Reflood Phase
695.	Cold Leg Recirculation Begins
1.0E+07	Transient Modeling Terminated

Table 6.4.1-22 Double-Ended Hot Leg Break Blowdown Mass and Energy Releases Byron Unit 2 & Braidwood Unit 2 (D5 SG)				
	Break Path No.1*		Break Path No.2**	
Time Seconds	Flow (Lbm/Sec)	Energy (Thousand Btu/Sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
.00000	.0	.0	.0	.0
.00112	48435.4	31832.7	48435.0	31831.4
.102	42202.1	28050.2	28239.9	18515.2
.201	37779.4	25083.9	24189.2	15763.7
.302	37441.1	24796.2	21581.9	13879.0
.402	36260.0	23998.5	20258.2	12801.6
.502	35567.4	23534.7	19456.3	12082.3
.601	35407.0	23425.5	18899.2	11556.0
.701	35143.7	23267.9	18490.3	11156.0
.801	34573.4	22933.8	18159.7	10833.5
.901	33941.9	22576.9	17943.0	10602.5
1.00	33449.4	22324.3	17870.3	10474.0
1.10	33215.6	22256.4	17860.9	10394.6
1.20	32868.5	22117.2	17949.7	10380.2
1.30	32389.3	21887.4	18060.4	10386.4
1.40	31829.8	21600.8	18186.4	10408.8
1.50	31311.5	21334.7	18316.8	10441.2
1.60	30907.1	21139.3	18437.0	10475.8
1.70	30517.0	20950.1	18541.5	10508.2
1.80	30015.8	20678.0	18615.5	10529.8
1.90	29416.7	20329.3	18657.5	10538.8
2.00	28840.2	19989.7	18673.6	10537.7
2.10	28364.2	19717.8	18669.1	10528.5
2.20	27933.6	19474.9	18647.4	10512.4
2.30	27441.6	19179.5	18605.4	10487.3
2.40	26922.4	18855.5	18546.1	10454.2
2.50	26411.9	18531.2	18473.6	10414.9
2.60	25956.0	18241.2	18393.7	10372.5
2.70	25538.6	17976.7	18305.4	10326.0
2.80	25136.8	17717.8	18212.1	10277.1
2.90	24712.3	17433.8	18110.1	10223.6
3.00	24315.2	17163.7	18002.2	10166.8
3.10	23945.1	16909.0	17892.0	10108.9
3.20	23596.1	16664.2	17779.6	10049.7

Table 6.4.1-22 (cont.)

Double-Ended Hot Leg Break Blowdown Mass and Energy Releases

Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time Seconds	Break Path No.1* Flow (Lbm/Sec)	Break Path No.1* Energy (Thousand Btu/Sec)	Break Path No.2** Flow (lbm/sec)	Break Path No.2** Energy (Thousand Btu/sec)
3.30	23284.6	16443.4	17663.6	9988.4
3.40	22979.9	16222.8	17543.5	9924.7
3.50	22694.4	16010.4	17420.2	9859.3
3.60	22438.8	15816.7	17295.1	9792.8
3.70	22194.5	15626.9	17165.8	9724.0
3.80	21979.4	15455.2	17035.9	9654.9
3.90	21783.0	15294.7	16903.2	9584.3
4.00	21607.0	15146.1	16767.2	9511.9
4.20	21377.9	14934.3	16486.9	9362.8
4.40	21229.8	14778.9	16204.6	9212.6
4.60	21120.5	14652.5	15904.4	9052.6
4.80	21048.5	14553.5	15671.5	8932.1
5.00	21003.7	14465.2	15190.1	8667.7
5.20	21021.0	14393.9	14656.8	8376.1
5.40	21103.6	14361.3	14357.9	8222.4
5.60	21217.0	14337.1	13944.2	8000.9
5.80	21383.7	14333.7	13545.6	7787.7
6.00	21611.7	14363.0	13146.2	7572.9
6.20	21931.5	14443.1	12766.6	7369.1
6.40	23104.8	16032.9	12414.1	7179.9
6.60	16806.7	11975.8	12066.2	6992.2
6.80	16957.9	11962.3	11744.5	6818.5
7.00	17028.4	11905.1	11456.6	6663.2
7.20	17165.0	11930.4	11171.2	6507.3
7.40	17253.6	11914.4	10888.4	6351.8
7.60	17337.8	11880.6	10627.2	6208.1
7.80	17361.8	11830.9	10380.2	6072.0
8.00	17421.0	11810.8	10142.0	5940.4
8.20	17475.1	11784.3	9903.8	5808.3
8.40	17479.0	11732.8	9675.5	5682.0
8.60	17411.7	11664.6	9448.7	5556.4
8.80	17173.9	11458.5	9218.0	5428.7
9.00	17202.9	11413.4	8988.0	5301.9
9.20	17214.7	11362.4	8756.7	5174.6
9.40	17200.2	11299.9	8525.4	5047.9
9.60	17141.7	11216.1	8295.2	4922.3

Table 6.4.1-22 (cont.) Double-Ended Hot Leg Break Blowdown Mass and Energy Releases Byron Unit 2 & Braidwood Unit 2 (D5 SG)				
	Break Path No.1*		Break Path No.2**	
Time Seconds	Flow (Lbm/Sec)	Energy (Thousand Btu/Sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
9.80	17016.3	11097.7	8063.6	4796.5
10.0	16805.9	10936.6	7838.1	4674.6
10.2	16502.4	10727.9	7616.9	4555.5
10.4	16133.1	10486.2	7396.3	4437.2
10.6	15775.7	10254.9	7185.8	4324.9
10.8	15457.5	10047.4	6977.6	4214.1
11.0	15174.5	9861.9	6775.9	4107.3
11.2	14904.5	9686.6	6580.3	4004.2
11.4	14630.4	9512.4	6393.8	3906.3
11.6	14332.8	9328.1	6211.2	3810.8
11.8	14017.3	9137.3	6037.8	3720.4
12.0	13685.3	8940.3	5870.5	3633.2
12.2	13343.4	8740.8	5708.1	3548.9
12.4	12999.0	8543.1	5552.4	3468.3
12.6	12656.8	8349.8	5402.1	3390.6
12.8	12315.6	8160.9	5257.6	3316.2
13.0	11977.3	7979.0	5117.8	3244.4
13.2	11531.2	7777.2	4985.8	3176.8
13.4	10870.2	7544.1	4855.0	3109.8
13.6	10407.0	7375.0	4729.5	3045.8
13.8	9988.0	7205.7	4605.8	2982.8
14.0	9561.4	7016.9	4485.4	2921.8
14.2	9104.0	6804.3	4365.2	2861.6
14.4	8651.0	6590.2	4247.0	2802.9
14.6	8211.6	6386.7	4127.9	2745.0
14.8	7780.5	6194.8	4001.3	2684.9
15.0	7329.5	6001.8	3859.4	2619.7
15.2	6820.0	5720.1	3699.6	2548.7
15.4	6332.9	5391.6	3510.1	2465.2
15.6	5836.9	5083.3	3293.8	2369.1
15.8	5365.5	4803.3	3059.0	2264.9
16.0	4917.1	4548.8	2819.3	2158.9
16.2	4484.8	4300.7	2584.7	2054.1
16.4	4096.4	4061.6	2372.7	1957.7
16.6	3745.4	3842.1	2183.2	1869.6
16.8	3434.8	3646.3	2018.5	1791.2

Table 6.4.1-22 (cont.)

Double-Ended Hot Leg Break Blowdown Mass and Energy Releases

Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time	Break Path No.1*		Break Path No.2**	
Seconds	Flow (Lbm/Sec)	Energy (Thousand Btu/Sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
17.0	3168.1	3471.5	1877.8	1722.8
17.2	2948.8	3306.1	1753.0	1664.7
17.4	2769.2	3148.9	1648.4	1610.2
17.6	2602.9	2989.2	1559.0	1563.8
17.8	2442.1	2832.4	1474.3	1522.9
18.0	2278.0	2672.4	1400.7	1482.4
18.2	2114.6	2509.7	1333.4	1445.9
18.4	1982.4	2376.3	1267.9	1413.7
18.6	1822.3	2205.2	1206.4	1374.3
18.8	1686.8	2055.7	1152.8	1334.2
19.0	1567.9	1924.4	1108.2	1301.9
19.2	1474.8	1823.8	1046.9	1249.4
19.4	1411.3	1755.5	993.5	1198.1
19.6	1340.9	1675.2	946.8	1149.2
19.8	1255.1	1572.4	883.0	1079.5
20.0	1172.9	1472.6	809.6	995.0
20.2	1098.5	1379.0	742.9	916.2
20.4	1023.2	1286.1	684.5	846.1
20.6	955.7	1201.4	653.5	809.5
20.8	901.0	1135.8	539.7	667.9
21.0	843.4	1065.3	487.9	607.2
21.2	787.7	995.3	448.4	558.2
21.4	742.8	938.8	411.5	513.2
21.6	703.7	889.2	376.3	470.1
21.8	405.5	516.1	350.1	437.8
22.0	150.6	191.4	301.6	377.6
22.2	.0	.0	182.4	228.7
22.4	.0	.0	99.7	126.7
22.6	.0	.0	.0	.0

* mass and energy exiting from the reactor vessel side of the break

** mass and energy exiting from the SG side of the break

Table 6.4.1-23

Double-Ended Hot Leg Break Mass Balance

Byron Unit 2 & Braidwood Unit 2 (D5 SG)

		Mass Balance		
Time (Seconds)		.00	22.60	22.60
		Mass (Thousand lbm)		
Initial	In RCS and ACC	760.82	760.82	760.82
Added Mass	Pumped Injection	.00	.00	.00
	Total Added	.00	.00	.00
***** TOTAL AVAILABLE *****		760.82	760.82	760.82
Distribution	Reactor Coolant	511.13	59.47	96.89
	Accumulator	249.69	205.56	168.14
	Total Contents	760.82	265.03	265.03
Effluent	Break Flow	.00	495.77	495.77
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	495.77	495.77
***** TOTAL ACCOUNTABLE *****		760.82	760.80	760.80

Table 6.4.1-24

Double-Ended Hot Leg Break Energy Balance

Byron Unit 2 & Braidwood Unit 2 (D5 SG)

		Energy Balance		
Time (Seconds)		.00	22.60	22.60
		Energy (Million Btu)		
Initial Energy	In RCS, ACC, S GEN	865.38	865.38	865.38
Added Energy	Pumped Injection	.00	.00	.00
	Decay Heat	.00	8.22	8.22
	Heat From Secondary	.00	-.59	-.59
	Total Added	.00	7.64	7.64
***** TOTAL AVAILABLE *****		865.38	873.02	873.02
Distribution	Reactor Coolant	308.94	15.54	19.26
	Accumulator	24.83	20.45	16.72
	Core Stored	23.98	9.54	9.54
	Primary Metal	153.18	144.36	144.36
	Secondary Metal	108.26	107.67	107.67
	Steam Generator	246.19	244.47	244.47
	Total Contents	865.38	542.03	542.03
Effluent	Break Flow	.00	330.39	330.39
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	330.39	330.39
*** TOTAL ACCOUNTABLE ***		865.38	872.42	872.42

Table 6.4.1-25 Double-Ended Pump Suction Break Minimum Safeguards Blowdown Mass and Energy Releases Byron Unit 2 & Braidwood Unit 2 (D5 SG)				
	Break Path No. 1*		Break Path No. 2**	
Time (seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
.00000	.0	.0	.0	.0
.00110	87607.2	49361.2	43219.8	24298.4
.101	42805.0	24152.4	22312.0	12531.3
.201	43760.8	24902.9	24742.7	13908.5
.302	44964.2	25889.5	24946.0	14035.9
.401	46001.0	26848.2	24135.6	13593.8
.502	46551.9	27553.0	23023.5	12977.1
.602	46199.9	27696.9	22097.2	12460.1
.701	44824.7	27177.3	21289.6	12007.5
.802	43056.2	26378.2	20686.6	11670.4
.902	41384.8	25588.1	20272.2	11440.3
1.00	39914.4	24877.6	20049.0	11317.2
1.10	38662.7	24273.8	19922.5	11247.5
1.20	37624.8	23785.6	19866.0	11216.7
1.30	36787.4	23408.7	19835.2	11199.8
1.40	36074.2	23099.6	19811.0	11186.0
1.50	35427.9	22826.3	19797.9	11178.1
1.60	34810.7	22570.8	19812.9	11186.1
1.70	34211.1	22328.9	19844.3	11203.6
1.80	33541.9	22043.6	19862.7	11213.7
1.90	32892.9	21769.2	19847.5	11204.7
2.00	32174.9	21448.6	19784.1	11168.3
2.10	31436.5	21109.7	19707.2	11124.7
2.20	30679.3	20754.8	19611.5	11070.9
2.30	29882.0	20365.4	19460.5	10985.7
2.40	29029.5	19931.6	19008.3	10729.4
2.50	28149.6	19473.5	18762.7	10592.2
2.60	27069.7	18864.2	18575.6	10487.9
2.70	25656.6	18000.8	18364.7	10370.1
2.80	23583.9	16643.6	18136.3	10242.6
2.90	21815.6	15494.5	17905.4	10114.0
3.00	21223.0	15171.4	17674.8	9985.8

Table 6.4.1-25 (cont.) Double-Ended Pump Suction Break Minimum Safeguards Blowdown Mass and Energy Releases Byron Unit 2 & Braidwood Unit 2 (D5 SG)				
Time (seconds)	Break Path No. 1*		Break Path No. 2**	
	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
3.10	20717.6	14866.0	17452.5	9862.5
3.20	20012.5	14401.5	17239.4	9744.6
3.30	19464.3	14047.1	17019.9	9623.1
3.40	18821.8	13614.5	16815.6	9510.4
3.50	18171.0	13170.9	16621.3	9403.6
3.60	17514.7	12718.6	16432.8	9300.1
3.80	16314.1	11885.5	16106.6	9122.4
3.90	15793.5	11521.9	15952.5	9038.7
4.00	15345.6	11209.2	15816.0	8965.2
4.20	14628.9	10703.3	15540.7	8816.6
4.40	14090.9	10317.1	15274.6	8673.4
4.60	13670.1	10005.8	15027.4	8541.1
4.80	13376.5	9779.4	14818.4	8430.3
5.00	13228.2	9649.0	14780.3	8419.5
5.20	13113.5	9537.6	15886.7	9055.6
5.40	13021.8	9442.7	15837.6	9036.5
5.60	12988.7	9387.1	15693.8	8961.9
5.80	12963.7	9337.3	15535.0	8878.7
6.00	12907.6	9270.5	15348.6	8778.7
6.20	12827.3	9191.6	15138.8	8664.5
6.40	12731.9	9106.5	14938.0	8554.5
6.60	12607.6	9007.2	14754.9	8453.2
6.80	12482.2	8913.5	14624.9	8381.1
7.00	12381.3	8833.7	14541.9	8333.7
7.20	12289.1	8754.2	14378.1	8237.6
7.40	12548.8	8922.1	14216.9	8143.1
7.60	12129.5	8633.7	14185.0	8124.5
7.80	11208.7	8478.2	14121.4	8085.3
8.00	9956.1	7940.3	13817.5	7906.3
8.20	9831.6	7834.6	13645.2	7808.1
8.40	9876.9	7784.2	13522.6	7741.0
8.60	9875.6	7703.1	13332.8	7631.1

Table 6.4.1-25 (cont.) Double-Ended Pump Suction Break Minimum Safeguards Blowdown Mass and Energy Releases Byron Unit 2 & Braidwood Unit 2 (D5 SG)				
	Break Path No. 1*		Break Path No. 2**	
Time (seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
8.80	9921.1	7658.1	13103.2	7497.0
9.00	9992.5	7622.5	12969.8	7419.7
9.20	10041.6	7569.4	12789.4	7314.5
9.40	10076.5	7512.5	12595.4	7201.1
9.60	10084.6	7444.7	12456.5	7119.6
9.80	10041.0	7351.9	12278.4	7015.3
10.0	9956.6	7243.8	12127.3	6926.4
10.2	9839.8	7124.5	11977.9	6838.6
10.4	9679.8	6985.8	11826.5	6749.5
10.4	9678.7	6985.0	11825.7	6749.1
10.4	9677.7	6984.1	11824.9	6748.6
10.6	9497.5	6844.2	11692.7	6670.6
10.8	9309.1	6707.4	11548.3	6585.7
11.0	9104.6	6566.2	11418.9	6509.5
11.2	8904.7	6434.2	11284.3	6430.4
11.4	8704.1	6305.3	11154.5	6354.1
11.6	8508.2	6182.6	11025.3	6278.3
11.8	8317.4	6065.1	10898.9	6204.3
12.0	8129.5	5951.1	10772.3	6130.3
12.2	7946.7	5842.3	10647.6	6057.5
12.4	7770.8	5739.3	10523.1	5984.8
12.6	7598.7	5639.3	10399.8	5913.0
12.8	7431.5	5542.3	10278.6	5842.6
13.0	7265.5	5446.2	10158.6	5773.0
13.2	7109.8	5361.1	10042.9	5705.9
13.4	6954.1	5276.6	9917.0	5633.1
13.6	6806.9	5194.5	9804.8	5568.7
13.8	6665.0	5112.1	9676.4	5495.1
14.0	6520.8	5027.6	9548.0	5422.0
14.2	6363.9	4933.6	9394.1	5334.9
14.4	6198.8	4831.3	9245.5	5251.8
14.6	6036.0	4724.1	9097.0	5169.0

Table 6.4.1-25 (cont.) Double-Ended Pump Suction Break Minimum Safeguards Blowdown Mass and Energy Releases Byron Unit 2 & Braidwood Unit 2 (D5 SG)				
Time (seconds)	Break Path No. 1*		Break Path No. 2**	
	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
14.8	5887.4	4620.3	8954.7	5089.1
15.0	5755.1	4524.3	8821.9	5014.5
15.2	5636.1	4436.4	8692.8	4942.2
15.4	5534.8	4361.6	8587.8	4884.6
15.6	5450.0	4302.1	8277.5	4715.5
15.8	5387.2	4287.1	8237.1	4729.8
16.0	5267.4	4280.3	7834.1	4519.6
16.2	5100.1	4263.7	7687.5	4451.4
16.4	4919.1	4245.4	7469.6	4335.1
16.6	4733.5	4227.4	7279.0	4212.3
16.8	4525.2	4193.8	7034.5	4034.5
17.0	4258.8	4116.7	6761.4	3815.6
17.2	3963.3	4014.8	6517.8	3586.3
17.4	3653.4	3882.8	6283.0	3346.5
17.6	3356.3	3727.1	5942.6	3059.5
17.8	3084.2	3554.1	5574.3	2780.9
18.0	2839.5	3367.3	5254.0	2548.4
18.2	2629.6	3185.1	4945.1	2336.3
18.4	2412.2	2956.8	4658.2	2147.0
18.6	2225.9	2747.4	4426.7	1993.3
18.8	2062.7	2557.6	4220.7	1860.0
19.0	1919.3	2387.9	4021.0	1737.1
19.2	1801.2	2247.1	3820.3	1620.9
19.4	1697.7	2122.0	3435.4	1429.7
19.6	1594.4	1996.8	3064.6	1240.5
19.8	1506.5	1889.6	2763.6	1085.3
20.0	1411.8	1773.8	2726.3	1038.8
20.2	1317.3	1657.4	3016.2	1119.4
20.4	1216.2	1532.6	3760.8	1368.8
20.6	1126.8	1421.7	4185.2	1509.5
20.8	1032.2	1304.3	3277.0	1176.5
21.0	953.8	1207.2	2646.5	950.0

Table 6.4.1-25 (cont.)
Double-Ended Pump Suction Break Minimum Safeguards
Blowdown Mass and Energy Releases
Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time (seconds)	Break Path No. 1*		Break Path No. 2**	
	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
21.2	882.7	1118.2	2432.4	873.6
21.4	819.9	1039.7	2149.6	772.1
21.6	761.7	966.5	1663.3	597.6
21.8	684.0	868.5	1049.2	376.9
22.0	598.5	760.5	794.8	275.6
22.2	522.7	664.7	1376.9	429.3
22.4	448.3	570.6	3290.2	952.7
22.6	373.0	475.0	5665.4	1601.2
22.8	298.7	380.7	6508.4	1821.1
23.0	225.2	287.3	6586.2	1831.5
23.2	160.0	204.4	6169.7	1706.7
23.4	97.2	124.4	5401.5	1487.4
23.6	32.2	41.3	4513.5	1238.6
23.8	.0	.0	3724.6	1019.8
24.0	.0	.0	1940.9	530.8
24.2	.0	.0	347.7	95.2
24.4	.0	.0	.0	.0

* mass and energy exiting the SG side of the break

** mass and energy exiting the pump side of the break

Table 6.4.1-26

Double-Ended Pump Suction Break

Minimum Safeguards Reflood Mass and Energy Releases

Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time (seconds)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Energy (lbm/sec)	Energy (Thousand Btu/sec)
24.4	.0	.0	.0	.0
25.0	.0	.0	.0	.0
25.2	.0	.0	.0	.0
25.3	.0	.0	.0	.0
25.4	.0	.0	.0	.0
25.5	.0	.0	.0	.0
25.6	8.2	9.6	.0	.0
25.7	48.0	56.6	.0	.0
25.8	22.7	26.8	.0	.0
25.9	13.9	16.4	.0	.0
26.0	14.4	16.9	.0	.0
26.1	16.6	19.5	.0	.0
26.2	26.2	30.9	.0	.0
26.3	34.2	40.3	.0	.0
26.4	39.5	46.6	.0	.0
26.5	44.8	52.8	.0	.0
26.6	48.9	57.6	.0	.0
26.7	52.7	62.1	.0	.0
26.8	56.4	66.4	.0	.0
26.9	59.9	70.6	.0	.0
27.0	63.2	74.5	.0	.0
27.1	65.6	77.4	.0	.0
27.1	66.4	78.3	.0	.0
27.2	69.6	82.0	.0	.0
27.3	72.6	85.6	.0	.0
27.4	75.5	89.0	.0	.0
27.5	78.3	92.4	.0	.0
28.5	103.2	121.7	.0	.0
29.5	123.7	145.9	.0	.0
30.5	141.3	166.8	.0	.0
31.5	157.0	185.2	.0	.0

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Table 6.4.1-26 (cont.)

Double-Ended Pump Suction Break

Minimum Safeguards Reflood Mass and Energy Releases

Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time (seconds)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Energy (lbm/sec)	Energy (Thousand Btu/sec)
32.0	204.2	241.1	1353.5	212.0
32.5	335.2	396.6	3152.4	512.3
33.5	383.1	453.8	3639.7	619.3
34.5	378.9	448.7	3596.8	616.7
35.5	372.8	441.4	3535.9	609.8
36.5	366.6	434.0	3472.9	602.3
37.1	362.9	429.6	3435.1	597.7
37.5	360.5	426.7	3409.9	594.6
38.5	354.4	419.5	3347.6	587.0
39.5	348.6	412.6	3286.4	579.4
40.5	342.9	405.8	3226.6	571.9
41.5	337.4	399.3	3168.2	564.5
42.5	332.1	392.9	3111.2	557.3
43.5	326.9	386.8	3055.8	550.2
43.6	326.4	386.2	3050.3	549.5
44.6	351.1	415.6	3398.5	576.1
45.6	354.9	420.2	3392.5	579.0
46.6	350.4	414.7	3345.2	572.8
47.6	346.0	409.5	3298.8	566.9
48.6	341.7	404.3	3253.3	561.0
49.6	337.5	399.4	3208.9	555.2
50.6	333.4	394.5	3165.6	549.6
50.8	332.6	393.6	3157.1	548.5
51.6	329.5	389.8	3123.3	544.1
52.6	325.6	385.3	3082.0	538.7
53.6	321.9	380.8	3041.6	533.4
54.6	318.3	376.5	3002.2	528.2
55.6	314.7	372.3	2963.6	523.2
56.6	311.3	368.2	2925.8	518.2
57.6	307.9	364.2	2888.9	513.3
58.6	304.7	360.3	2852.7	508.6
59.6	301.5	356.5	2817.2	503.9

Table 6.4.1-26 (cont.)
Double-Ended Pump Suction Break
Minimum Safeguards Reflood Mass and Energy Releases
Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time (seconds)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Energy (lbm/sec)	Energy (Thousand Btu/sec)
60.6	298.3	352.8	2782.5	499.3
61.6	295.3	349.2	2748.4	494.8
62.6	292.3	345.6	2715.0	490.3
63.6	289.4	342.2	2682.3	485.9
64.6	286.5	338.8	2650.1	481.6
65.6	283.7	335.4	2618.5	477.4
66.6	281.0	332.2	2587.5	473.3
67.6	278.3	329.0	2557.0	469.2
68.6	275.7	325.9	2527.1	465.1
69.6	283.8	335.4	733.3	217.3
70.6	335.5	397.0	573.0	197.2
71.6	342.2	405.0	540.8	193.7
72.6	343.4	406.4	528.0	191.9
73.6	447.4	530.4	334.7	241.1
74.6	461.1	546.8	338.8	249.7
74.8	459.0	544.4	338.0	248.5
75.6	448.7	532.1	334.1	242.1
76.6	435.1	515.8	329.1	233.8
77.6	421.6	499.7	324.1	225.6
78.6	409.7	485.4	319.8	218.5
79.6	398.5	472.1	315.7	211.8
80.6	387.9	459.4	311.8	205.5
81.6	377.8	447.4	308.2	199.6
82.6	368.2	435.9	304.7	194.0
83.6	359.1	425.0	301.4	188.8
84.6	350.4	414.7	298.3	183.8
85.6	342.1	404.9	295.4	179.1
87.6	326.9	386.8	289.5	170.6
89.6	313.9	371.3	284.5	163.4
90.8	306.7	362.7	281.7	159.5
91.6	302.1	357.2	279.9	157.0
93.6	291.3	344.4	275.8	151.1

Table 6.4.1-26 (cont.)

Double-Ended Pump Suction Break

Minimum Safeguards Reflood Mass and Energy Releases

Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time (seconds)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Energy (lbm/sec)	Energy (Thousand Btu/sec)
95.6	281.4	332.7	272.0	145.9
97.6	272.5	322.1	268.7	141.2
99.6	264.4	312.5	265.6	136.9
101.6	257.1	303.9	262.9	133.1
103.6	250.5	296.0	260.4	129.7
105.6	244.6	289.0	258.3	126.7
107.6	239.2	282.6	256.3	124.0
109.6	234.4	277.0	254.5	121.5
111.6	230.2	271.9	253.0	119.4
112.1	229.2	270.7	252.6	118.9
113.6	226.4	267.4	251.6	117.5
115.6	223.0	263.4	250.4	115.8
117.6	220.0	259.9	249.3	114.3
119.6	217.4	256.7	248.3	113.0
121.6	215.1	254.0	247.5	111.9
123.6	213.1	251.6	246.8	110.9
125.6	211.3	249.5	246.1	110.0
127.6	209.8	247.8	245.6	109.2
129.6	208.5	246.2	245.1	108.6
131.6	207.4	244.9	244.7	108.0
133.6	206.5	243.8	244.4	107.6
135.6	205.7	242.9	244.1	107.2
137.3	205.1	242.2	243.9	106.9
137.6	205.1	242.1	243.8	106.8
139.6	204.5	241.5	243.6	106.6
141.6	204.2	241.1	243.5	106.3
143.6	203.9	240.7	243.3	106.2
145.6	203.7	240.5	243.3	106.1
147.6	203.5	240.3	243.2	106.0
149.6	203.5	240.3	243.1	105.9
151.6	203.5	240.3	243.1	105.9
153.6	203.5	240.3	243.1	105.9

Table 6.4.1-26 (cont.)

Double-Ended Pump Suction Break

Minimum Safeguards Reflood Mass and Energy Releases

Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time (seconds)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Energy (lbm/sec)	Energy (Thousand Btu/sec)
155.6	203.6	240.4	243.1	105.9
157.6	203.7	240.6	243.2	105.9
159.6	203.9	240.8	243.2	106.0
161.6	204.1	241.0	243.2	106.0
163.6	204.4	241.3	243.3	106.1
164.7	204.5	241.5	243.3	106.2
165.6	204.6	241.6	243.4	106.2
167.6	204.9	242.0	243.5	106.3
169.6	205.2	242.3	243.5	106.4
171.6	205.5	242.7	243.6	106.6
173.6	205.9	243.1	243.7	106.7
175.6	206.3	243.6	243.8	106.8
177.6	206.6	244.0	243.9	107.0
179.6	207.0	244.4	244.1	107.2
181.6	207.4	244.9	244.2	107.3
183.6	208.7	246.4	244.9	107.9
185.6	210.0	248.0	246.4	108.6
187.6	211.4	249.6	248.6	109.5
189.6	212.8	251.3	251.4	110.4
191.6	214.2	253.0	254.5	111.3
193.3	215.3	254.3	257.3	112.0

* mass and energy exiting the SG side of the break

** mass and energy exiting the pump side of the break

Table 6.4.1-27
Double-Ended Pump Suction Break- Minimum Safeguards
Principle Parameters During Reflood
Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time Seconds	Flooding Temp °F	Flooding Rate in/sec	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total (Pounds Mass per Second)	Injection Accum (Pounds Mass per Second)	Spill (Pounds Mass per Second)	Enthalpy Btu/lbm
24.4	175.1	.000	.000	.00	.00	.250	.0	.0	.0	.00
25.2	173.5	18.467	.000	.53	1.18	.000	6227.7	6227.7	.0	99.46
25.4	172.7	19.950	.000	.85	1.09	.000	6186.9	6186.9	.0	99.46
25.5	172.2	19.551	.000	1.02	1.03	.000	6166.8	6166.8	.0	99.46
25.9	171.9	1.824	.101	1.31	1.52	.254	6068.7	6068.7	.0	99.46
26.3	172.0	2.733	.166	1.38	2.51	.277	5992.9	5992.9	.0	99.46
27.1	172.3	2.424	.302	1.50	4.37	.331	5843.9	5843.9	.0	99.46
28.5	172.9	2.358	.455	1.67	7.78	.355	5606.4	5606.4	.0	99.46
32.0	174.8	2.910	.623	2.00	15.69	.420	5053.7	5053.7	.0	99.46
33.5	175.7	3.862	.666	2.16	16.12	.549	4522.9	4522.9	.0	99.46
34.5	176.4	3.738	.682	2.26	16.12	.547	4415.0	4415.0	.0	99.46
37.1	178.3	3.499	.707	2.50	16.12	.541	4175.3	4175.3	.0	99.46
43.6	183.5	3.149	.730	3.00	16.12	.524	3687.1	3687.1	.0	99.46
44.6	184.3	3.307	.731	3.07	16.12	.547	4066.8	3457.3	.0	97.74
50.8	189.7	3.103	.740	3.50	16.12	.535	3784.5	3154.1	.0	97.55
58.6	196.5	2.903	.744	4.01	16.12	.520	3426.7	2786.1	.0	97.32
67.6	204.2	2.720	.747	4.54	16.12	.506	3082.9	2433.5	.0	97.05
68.6	205.1	2.702	.747	4.60	16.12	.504	3048.2	2398.0	.0	97.02

Table 6.4.1-27 (cont.)

Double-Ended Pump Suction Break- Minimum Safeguards

Principle Parameters During Reflood

Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time Seconds	Flooding Temp °F	Flooding Rate in/sec	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total (Pounds Mass per Second)	Injection Accum (Pounds Mass per Second)	Spill (Pounds Mass per Second)	Enthalpy Btu/lbm
72.6	208.7	3.059	.752	4.84	16.12	.552	1104.5	479.2	.0	92.97
73.6	209.8	3.639	.755	4.91	16.00	.608	562.0	.0	.0	88.00
74.6	211.0	3.688	.755	4.99	15.73	.611	547.7	.0	.0	88.00
74.8	211.2	3.672	.755	5.00	15.68	.610	548.9	.0	.0	88.00
82.6	220.3	3.030	.754	5.53	14.12	.594	604.6	.0	.0	88.00
90.8	229.3	2.598	.753	6.00	13.23	.578	632.7	.0	.0	88.00
101.6	238.9	2.250	.752	6.54	12.72	.560	648.1	.0	.0	88.00
112.1	246.4	2.054	.753	7.00	12.66	.547	655.8	.0	.0	88.00
125.6	254.3	1.923	.755	7.55	12.93	.537	660.5	.0	.0	88.00
137.3	260.2	1.873	.758	8.00	13.32	.534	662.0	.0	.0	88.00
151.6	266.3	1.849	.763	8.53	13.87	.533	662.5	.0	.0	88.00
164.7	271.2	1.843	.768	9.00	14.41	.534	662.4	.0	.0	88.00
165.6	271.5	1.843	.768	9.03	14.45	.534	662.4	.0	.0	88.00
179.6	276.0	1.846	.774	9.53	15.03	.535	661.9	.0	.0	88.00
193.3	279.9	1.878	.779	10.00	15.54	.543	659.8	.0	.0	88.00

Table 6.4.1-28 Double-Ended Pump Suction Break Minimum Safeguards Post-Reflood Mass and Energy Releases Byron Unit 2 & Braidwood Unit 2 (D5 SG)				
Time	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
(seconds)	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
193.3	245.4	305.4	439.8	152.1
198.3	244.6	304.3	440.7	152.1
203.3	243.9	303.5	441.3	152.0
208.3	243.3	302.7	441.9	151.9
213.3	242.7	301.9	442.6	151.8
218.3	242.0	301.1	443.3	151.8
223.3	241.3	300.2	444.0	151.7
228.3	241.3	300.2	443.9	151.4
233.3	240.5	299.3	444.7	151.4
238.3	239.7	298.2	445.5	151.4
243.3	239.6	298.1	445.6	151.1
248.3	238.7	296.9	446.6	151.1
253.3	238.5	296.7	446.8	150.9
258.3	237.4	295.4	447.8	150.9
263.3	237.1	295.0	448.1	150.8
268.3	236.7	294.5	448.5	150.6
273.3	235.5	293.0	449.7	150.7
278.3	235.0	292.3	450.3	150.6
283.3	234.3	291.5	450.9	150.5
288.3	233.6	290.7	451.6	150.4
293.3	233.5	290.5	451.7	150.2
298.3	232.6	289.4	452.6	150.2
303.3	232.3	289.0	453.0	150.0
308.3	231.2	287.6	454.1	150.0
313.3	230.6	286.9	454.7	149.9
318.3	229.8	285.9	455.4	149.9
323.3	229.5	285.6	455.7	149.7
328.3	229.1	285.0	456.2	149.5
333.3	228.4	284.1	456.9	149.5
338.3	227.4	283.0	457.8	149.4
343.3	226.8	282.2	458.4	149.3
348.3	225.9	281.0	459.4	149.3

Table 6.4.1-28 (cont.) Double-Ended Pump Suction Break Minimum Safeguards Post-Reflood Mass and Energy Releases Byron Unit 2 & Braidwood Unit 2 (D5 SG)				
	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (seconds)	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
353.3	225.7	280.8	459.6	149.1
358.3	224.5	279.3	460.8	149.2
363.3	223.8	278.4	461.4	149.1
368.3	223.4	278.0	461.8	148.9
373.3	222.7	277.0	462.6	148.8
378.3	221.7	275.8	463.5	148.8
383.3	221.0	274.9	464.3	148.8
388.3	220.4	274.2	464.8	148.6
393.3	219.6	273.2	465.7	148.6
398.3	98.1	122.0	587.2	180.8
556.7	98.1	122.0	587.2	180.8
556.8	100.1	123.7	585.2	176.8
558.3	100.0	123.6	585.2	176.7
1108.3	86.6	107.0	598.6	167.0
1110.0	86.6	106.9	153.5	154.6
1282.4	86.6	106.9	153.5	154.6
1282.5	82.9	95.4	51.3	21.5
3600.0	63.6	73.1	70.6	25.0
3600.1	53.6	61.6	80.6	11.9
10000.0	39.0	44.8	95.2	14.1
100000.0	20.8	24.0	113.3	16.8
1000000.0	8.9	10.3	125.2	18.5
10000000.0	2.8	3.2	131.4	19.4

* mass and energy exiting the SG side of the break

** mass and energy exiting the pump side of the break

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Table 6.4.1-29
Double-Ended Pump Suction Break Mass Balance
Minimum Safeguards
Byron Unit 2 & Braidwood Unit 2 (D5 SG)

		Mass Balance						
Time (Seconds)		.00	24.40	24.40	193.25	56.83	1282.44	3600.00
		Mass (Thousand lbm)						
Initial	In RCS and ACC	760.82	760.82	760.82	760.82	760.82	760.82	760.82
Added Mass	Pumped Injection	.00	.00	.00	96.28	345.39	747.58	1058.50
	Total Added	.00	.00	.00	96.28	345.39	747.58	1058.50
*** TOTAL AVAILABLE ***		760.82	760.82	760.82	857.10	1106.21	1508.40	1819.32
Distribution	Reactor Coolant	511.13	39.14	76.56	138.21	138.21	138.21	138.21
	Accumulator	249.69	204.88	167.47	.00	.00	.00	.00
	Total Contents	760.82	244.02	244.02	138.21	138.21	138.21	138.21
Effluent	Break Flow	.00	516.78	516.78	707.21	956.32	1376.39	1687.32
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	516.78	516.78	707.21	956.32	1376.39	1687.32
*** TOTAL ACCOUNTABLE ***		760.82	760.80	760.80	845.42	1094.53	1514.60	1825.53

Table 6.4.1-30
Double-Ended Pump Suction Break Energy Balance
Minimum Safeguards
Byron Unit 2 & Braidwood Unit 2 (D5 SG)

		Energy Balance						
Time (Seconds)		.00	24.40	24.40	93.25	556.83	1282.44	3600.00
		Energy (Million Btu)						
Initial Energy	In RCS, ACC, S GEN	865.38	865.38	865.38	865.38	865.38	865.38	865.38
Added Energy	Pumped Injection	.00	.00	.00	8.47	30.39	67.18	113.19
	Decay Heat	.00	8.29	8.29	29.41	64.61	122.09	263.40
	Heat From Secondary	.00	1.21	1.21	1.21	6.14	14.40	14.40
	Total Added	.00	9.50	9.50	39.09	101.14	203.67	390.99
***** TOTAL AVAILABLE *****		865.38	874.88	874.88	904.48	966.52	1069.05	1256.38
Distribution	Reactor Coolant	308.94	9.66	13.38	37.31	37.31	37.31	37.31
	Accumulator	24.83	20.38	16.66	.00	.00	.00	.00
	Core Stored	23.98	12.83	12.83	4.91	4.71	4.38	3.33
	Primary Metal	153.18	145.89	145.89	119.54	87.32	66.98	51.46
	Secondary Metal	108.26	109.48	109.48	98.88	80.21	56.74	43.80
	Steam Generator	246.19	249.88	249.88	220.48	177.51	129.99	102.59
	Total Contents	865.38	548.11	548.11	481.12	387.05	295.40	238.50
Effluent	Break Flow	.00	326.19	326.19	413.31	569.43	755.11	1004.25
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	326.19	326.19	413.31	569.43	755.11	1004.25
**** TOTAL ACCOUNTABLE *****		865.38	874.30	874.30	894.43	956.48	1050.50	1242.74

Table 6.4.1-31

Double-Ended Pump Suction Break Maximum Safeguards

Blowdown Mass and Energy Releases

Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time (seconds)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
.00000	.0	.0	.0	.0
.00110	87607.2	49361.2	43219.8	24298.4
.101	42805.0	24152.4	22312.0	12531.3
.201	43760.8	24902.9	24742.7	13908.5
.302	44964.2	25889.5	24946.0	14035.9
.401	46001.0	26848.2	24135.6	13593.8
.502	46551.9	27553.0	23023.5	12977.1
.602	46199.9	27696.9	22097.2	12460.1
.701	44824.7	27177.3	21289.6	12007.5
.802	43056.2	26378.2	20686.6	11670.4
.902	41384.8	25588.1	20272.2	11440.3
1.00	39914.4	24877.6	20049.0	11317.2
1.10	38662.7	24273.8	19922.5	11247.5
1.20	37624.8	23785.6	19866.0	11216.7
1.30	36787.4	23408.7	19835.2	11199.8
1.40	36074.2	23099.6	19811.0	11186.0
1.50	35427.9	22826.3	19797.9	11178.1
1.60	34810.7	22570.8	19812.9	11186.1
1.70	34211.1	22328.9	19844.3	11203.6
1.80	33541.9	22043.6	19862.7	11213.7
1.90	32892.9	21769.2	19847.5	11204.7
2.00	32174.9	21448.6	19784.1	11168.3
2.10	31436.5	21109.7	19707.2	11124.7
2.20	30679.3	20754.8	19611.5	11070.9
2.30	29882.0	20365.4	19460.5	10985.7
2.40	29029.5	19931.6	19008.3	10729.4
2.50	28149.6	19473.5	18762.7	10592.2
2.60	27069.7	18864.2	18575.6	10487.9
2.70	25656.6	18000.8	18364.7	10370.1
2.80	23583.9	16643.6	18136.3	10242.6
2.90	21815.6	15494.5	17905.4	10114.0

Table 6.4.1-31 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Blowdown Mass and Energy Releases Byron Unit 2 & Braidwood Unit 2 (D5 SG)				
Time (seconds)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
3.00	21223.0	15171.4	17674.8	9985.8
3.10	20717.6	14866.0	17452.5	9862.5
3.20	20012.5	14401.5	17239.4	9744.6
3.30	19464.3	14047.1	17019.9	9623.1
3.40	18821.8	13614.5	16815.6	9510.4
3.50	18171.0	13170.9	16621.3	9403.6
3.60	17514.7	12718.6	16432.8	9300.1
3.70	16893.6	12288.7	16269.1	9210.9
3.80	16314.1	11885.5	16106.6	9122.4
3.90	15793.5	11521.9	15952.5	9038.7
4.00	15345.6	11209.2	15816.0	8965.2
4.20	14628.9	10703.3	15540.7	8816.6
4.40	14090.9	10317.1	15274.6	8673.4
4.60	13670.1	10005.8	15021.6	8536.3
4.80	13376.7	9779.4	14827.7	8426.0
5.00	13229.0	9649.6	14649.6	8325.4
5.20	13113.7	9537.7	15800.5	8988.6
5.40	13015.7	9438.0	15867.1	9021.3
5.60	12953.3	9362.0	15870.0	9025.3
5.80	12915.7	9304.7	15751.2	8958.9
6.00	12842.8	9228.8	15562.9	8852.5
6.20	12729.3	9132.9	15372.3	8744.7
6.40	12589.8	9025.6	15199.1	8646.5
6.60	12421.3	8904.4	15034.5	8552.2
6.80	12271.7	8801.8	14926.5	8489.6
7.00	12138.0	8705.7	14841.7	8438.7
7.20	12018.3	8613.1	14675.8	8340.5
7.40	12225.7	8748.2	14517.0	8246.1
7.60	11917.1	8546.3	14492.1	8229.1
7.80	10995.4	8410.4	14435.4	8192.6
8.00	9788.7	7889.4	14108.0	8001.6
8.20	9673.1	7779.7	13913.2	7887.1

Table 6.4.1-31 (cont.)
Double-Ended Pump Suction Break Maximum Safeguards
Blowdown Mass and Energy Releases
Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
(seconds)	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
8.40	9717.1	7722.2	13821.4	7832.9
8.60	9725.4	7643.2	13643.0	7728.7
8.80	9775.9	7592.9	13369.9	7570.0
9.00	9862.1	7560.5	13238.9	7493.0
9.20	9925.8	7511.0	13080.5	7399.9
9.40	9967.1	7453.1	12851.1	7266.1
9.60	9986.0	7389.1	12713.7	7184.8
9.80	9958.4	7304.6	12550.6	7088.6
10.0	9878.7	7196.7	12367.4	6980.8
10.2	9775.2	7085.5	12235.8	6902.6
10.4	9623.9	6950.4	12071.1	6805.4
10.6	9440.5	6807.0	11931.7	6722.8
10.8	9263.2	6677.9	11794.1	6641.2
11.0	9058.7	6535.7	11644.6	6552.9
11.2	8862.9	6406.7	11524.3	6481.4
11.4	8667.0	6280.8	11376.9	6394.6
11.6	8471.1	6158.0	11251.5	6320.4
11.8	8283.9	6043.0	11120.4	6243.1
12.0	8096.5	5929.6	10986.0	6164.2
12.2	7917.2	5823.4	10864.0	6092.4
12.4	7740.1	5719.7	10729.0	6013.5
12.6	7572.2	5623.2	10606.8	5941.9
12.8	7404.8	5526.3	10477.4	5866.4
13.0	7238.8	5430.6	10357.7	5796.6
13.2	7085.1	5348.0	10231.5	5723.3
13.4	6930.2	5264.0	10105.3	5650.1
13.6	6784.0	5182.2	9987.2	5581.9
13.8	6642.9	5100.8	9854.4	5505.4
14.0	6500.5	5018.2	9724.8	5431.0
14.2	6343.5	4924.9	9560.7	5337.2
14.4	6181.5	4825.5	9418.0	5255.8
14.6	6020.1	4720.0	9258.2	5165.0

Table 6.4.1-31 (cont.)

Double-Ended Pump Suction Break Maximum Safeguards

Blowdown Mass and Energy Releases

Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time (seconds)	Break Path No. 1 Flow* (lbm/sec)	Break Path No. 1 Flow* Thousand (Btu/sec)	Break Path No. 2 Flow** (lbm/sec)	Break Path No. 2 Flow** Thousand (Btu/sec)
14.8	5874.0	4619.6	9120.6	5087.2
15.0	5740.5	4524.1	8974.0	5004.8
15.2	5619.2	4436.0	8845.1	4932.9
15.4	5516.1	4360.8	8728.6	4868.7
15.6	5430.9	4302.9	8568.2	4788.4
15.8	5363.8	4290.3	8384.1	4711.6
16.0	5234.6	4281.0	8060.5	4546.8
16.2	5065.2	4265.0	7997.0	4526.1
16.4	4883.6	4244.9	7600.1	4307.4
16.6	4699.4	4228.3	7519.9	4256.8
16.8	4485.8	4190.6	7145.4	4014.4
17.0	4217.0	4110.6	6926.2	3834.6
17.2	3927.6	4017.6	6656.1	3599.3
17.4	3615.9	3881.0	6346.6	3332.4
17.6	3319.3	3718.1	5947.4	3024.2
17.8	3050.4	3539.2	5574.5	2750.3
18.0	2808.3	3347.8	5242.3	2515.3
18.2	2595.5	3153.4	4928.0	2302.5
18.4	2389.5	2932.9	4626.6	2108.3
18.6	2205.5	2724.1	4414.4	1964.1
18.8	2050.9	2544.3	4203.8	1829.8
19.0	1913.4	2381.2	4004.9	1707.5
19.2	1800.6	2246.5	3734.5	1562.0
19.4	1695.5	2120.0	3334.6	1365.0
19.6	1598.6	2002.2	2996.7	1193.2
19.8	1494.2	1874.3	3298.0	1268.8
20.0	1381.9	1736.6	3954.8	1486.3
20.2	1275.6	1605.4	4455.1	1658.3
20.4	1180.2	1488.0	4292.7	1586.2
20.6	1109.1	1400.0	3143.1	1156.2
20.8	1022.9	1293.2	2807.7	1032.5
21.0	948.9	1201.1	2453.8	901.5

Table 6.4.1-31 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Blowdown Mass and Energy Releases Byron Unit 2 & Braidwood Unit 2 (D5 SG)				
Time	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
(seconds)	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
21.2	880.3	1115.4	2177.7	800.2
21.4	818.6	1038.2	1505.6	552.6
21.6	711.6	902.7	702.6	254.8
21.8	599.3	761.2	1043.2	333.8
22.0	515.7	655.6	4091.7	1150.7
22.2	435.8	554.5	6502.8	1797.9
22.4	378.6	482.0	6906.6	1909.9
22.6	308.9	393.6	7058.2	1954.4
22.8	252.9	322.5	6933.5	1921.3
23.0	188.0	239.9	6311.1	1748.7
23.2	83.9	107.3	4284.0	1185.6
23.4	57.6	73.9	2352.1	648.0
23.6	41.9	53.8	.0	.0
23.8	.0	.0	.0	.0

* mass and energy exiting the SG side of the break

** mass and energy exiting the pump side of the break

Table 6.4.1-32

Double-Ended Pump Suction Break

Maximum Safeguards Reflood Mass and Energy Releases

Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time (seconds)	Break Path No. 1* Flow (lbm/sec)	Break Path No. 1* Energy (Thousand Btu/sec)	Break Path No. 2** Flow (lbm/sec)	Break Path No. 2** Energy (Thousand Btu/sec)
23.8	.0	.0	.0	.0
24.4	.0	.0	.0	.0
24.6	.0	.0	.0	.0
24.7	.0	.0	.0	.0
24.8	.0	.0	.0	.0
24.9	.0	.0	.0	.0
25.0	17.8	20.9	.0	.0
25.1	41.0	48.4	.0	.0
25.2	20.8	24.6	.0	.0
25.3	14.9	17.6	.0	.0
25.4	17.3	20.4	.0	.0
25.5	23.4	27.6	.0	.0
25.6	30.3	35.7	.0	.0
25.7	38.0	44.8	.0	.0
25.8	43.7	51.6	.0	.0
25.9	47.7	56.2	.0	.0
26.0	51.7	61.0	.0	.0
26.1	55.6	65.5	.0	.0
26.2	59.2	69.8	.0	.0
26.3	62.7	74.0	.0	.0
26.4	66.1	77.9	.0	.0
26.5	69.3	81.8	.0	.0
26.6	72.5	85.5	.0	.0
26.7	75.5	89.0	.0	.0
26.8	78.4	92.5	.0	.0
26.9	81.3	95.9	.0	.0
27.9	106.4	125.5	.0	.0
28.9	127.0	149.9	.0	.0
29.9	144.9	170.9	.0	.0
30.9	218.9	258.5	1669.0	213.9
31.2	421.0	498.9	4083.6	590.4

Table 6.4.1-32 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases Byron Unit 2 & Braidwood Unit 2 (D5 SG)				
Time	Break Path No. 1*		Break Path No. 2**	
(seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
31.9	508.5	603.6	4935.4	746.9
32.9	509.1	604.3	4936.5	754.4
33.9	503.0	597.0	4882.4	748.2
34.9	496.3	589.0	4822.6	741.0
35.4	492.9	585.0	4791.9	737.1
35.9	489.6	580.9	4761.1	733.3
36.9	482.8	572.8	4699.4	725.4
37.9	476.2	565.0	4638.5	717.6
38.9	469.8	557.3	4578.6	709.9
39.9	463.6	549.8	4520.0	702.3
40.7	458.7	544.0	4474.2	696.4
40.9	457.5	542.6	4462.9	694.9
41.9	451.7	535.6	4407.3	687.6
42.9	446.0	528.8	4353.2	680.6
43.9	440.6	522.3	4300.6	673.7
44.9	435.3	516.0	4249.3	667.0
45.9	430.2	509.9	4199.5	660.5
46.8	425.7	504.5	4155.8	654.7
46.9	425.2	503.9	4151.0	654.1
47.9	420.4	498.2	4103.8	647.9
48.9	415.7	492.6	4057.8	641.8
49.9	411.2	487.2	4013.0	635.9
50.9	406.8	482.0	3969.3	630.2
51.9	402.5	476.9	3926.8	624.5
52.9	398.4	471.9	3885.2	619.0
53.5	395.9	469.0	3860.8	615.8
53.9	394.3	467.1	3844.7	613.6
54.9	390.4	462.4	3805.1	608.4
55.9	386.6	457.8	3766.4	603.2
56.9	382.8	453.4	3728.6	598.2
57.9	379.2	449.0	3691.6	593.3
58.9	375.6	444.8	3655.5	588.4

Table 6.4.1-32 (cont.)
Double-Ended Pump Suction Break
Maximum Safeguards Reflood Mass and Energy Releases
Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time (seconds)	Break Path No. 1* Flow (lbm/sec)	Break Path No. 1* Energy (Thousand Btu/sec)	Break Path No. 2** Flow (lbm/sec)	Break Path No. 2** Energy (Thousand Btu/sec)
59.9	372.1	440.6	3620.1	583.7
60.9	368.7	436.6	3585.4	579.0
61.9	365.4	432.6	3551.5	574.4
62.9	362.2	428.7	3518.3	569.9
63.9	359.0	424.9	3485.7	565.5
64.9	355.8	421.2	3453.7	561.2
65.9	352.8	417.6	3422.4	556.9
66.9	349.8	414.0	3391.6	552.7
67.9	346.8	410.5	3361.4	548.6
68.4	345.4	408.7	3346.5	546.5
68.9	343.9	407.0	3331.8	544.5
69.9	341.1	403.7	3302.7	540.5
70.9	338.3	400.3	3274.1	536.6
71.9	335.6	397.1	3245.9	532.7
72.9	332.9	393.9	3218.3	528.9
74.0	223.0	263.4	1880.3	360.4
75.0	222.1	262.3	1872.4	359.0
76.0	221.2	261.3	1864.5	357.6
77.0	186.6	220.3	1328.1	279.5
78.0	186.3	220.0	1328.8	279.3
79.0	186.0	219.6	1329.5	279.2
80.0	185.8	219.3	1330.1	279.1
81.0	185.5	219.0	1330.8	278.9
82.0	185.2	218.7	1331.5	278.8
83.0	184.9	218.3	1332.2	278.6
84.0	184.7	218.0	1332.9	278.5
85.0	184.4	217.7	1333.5	278.4
87.0	183.9	217.1	1334.9	278.1
88.1	183.6	216.7	1335.7	277.9
89.0	183.4	216.4	1336.3	277.8
91.0	182.8	215.8	1337.7	277.5
93.0	182.3	215.2	1339.1	277.3

Table 6.4.1-32 (cont.)

Double-Ended Pump Suction Break

Maximum Safeguards Reflood Mass and Energy Releases

Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time (seconds)	Break Path No. 1*		Break Path No. 2**	
	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
95.0	181.8	214.6	1340.5	277.0
97.0	181.3	214.0	1342.0	276.7
99.0	180.8	213.4	1343.4	276.4
101.0	180.3	212.8	1344.8	276.2
103.0	179.8	212.2	1346.3	275.9
105.0	179.3	211.7	1347.7	275.6
107.0	178.8	211.1	1349.1	275.4
109.0	178.3	210.5	1350.6	275.1
111.0	177.8	209.9	1352.0	274.8
111.2	177.8	209.9	1352.1	274.8
113.0	177.4	209.4	1353.4	274.5
115.0	176.9	208.8	1354.8	274.2
117.0	176.4	208.2	1356.2	273.9
119.0	175.9	207.7	1357.5	273.6
121.0	175.4	207.1	1358.9	273.3
123.0	175.0	206.5	1360.2	273.0
125.0	174.5	206.0	1361.6	272.7
127.0	174.0	205.4	1362.9	272.3
129.0	173.6	204.9	1364.2	272.0
131.0	173.1	204.3	1365.6	271.7
133.0	172.7	203.8	1366.9	271.4
135.0	172.2	203.3	1368.2	271.0
136.2	171.9	202.9	1368.9	270.8
137.0	171.8	202.7	1369.4	270.7
139.0	171.3	202.2	1370.7	270.3
141.0	170.9	201.7	1372.0	270.0
143.0	170.4	201.1	1373.2	269.7
145.0	170.0	200.6	1374.5	269.3
147.0	169.5	200.1	1375.7	268.9
149.0	169.1	199.6	1377.0	268.6
151.0	168.7	199.1	1378.2	268.2
153.0	168.2	198.6	1379.4	267.9

Table 6.4.1-32 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases Byron Unit 2 & Braidwood Unit 2 (D5 SG)				
	Break Path No. 1*		Break Path No. 2**	
Time (seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
155.0	167.8	198.1	1380.7	267.5
157.0	167.4	197.6	1381.9	267.1
159.0	166.9	197.0	1383.1	266.7
161.0	166.5	196.5	1384.3	266.4
163.0	166.1	196.0	1385.5	266.0
163.4	166.0	195.9	1385.7	265.9
165.0	165.7	195.5	1386.7	265.6
167.0	165.3	195.1	1387.9	265.2
169.0	164.9	194.6	1389.0	264.8
171.0	164.4	194.1	1390.2	264.4
173.0	164.0	193.6	1391.4	264.0
175.0	163.6	193.1	1392.5	263.6
177.0	163.2	192.6	1393.7	263.2
179.0	162.8	192.2	1394.8	262.8
181.0	162.4	191.7	1396.0	262.4
183.0	162.0	191.2	1397.1	262.0
185.0	161.6	190.8	1398.2	261.6
187.0	161.2	190.3	1399.4	261.2
189.0	160.9	189.8	1400.5	260.8
191.0	160.5	189.4	1401.6	260.4
193.0	160.1	188.9	1402.7	260.0
193.5	160.0	188.8	1403.0	259.9

* mass and energy exiting the SG side of the break

** mass and energy exiting the pump side of the break

Table 6.4.1-33
Double-Ended Pump Suction Break-Maximum Safeguards
Principle Parameters During Reflood
Byron Unit 2 & Braidwood Unit 2 (D5 SG)

	Flooding						Total	Injection Accum	Spill	
Time Seconds	Temp °F	Rate in/sec	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	(Pounds Mass per Second)			Enthalpy Btu/lbm
23.8	176.9	.000	.000	.00	.00	.250	.0	.0	.0	.00
24.6	175.2	8.963	.000	.54	1.23	.000	6431.7	6431.7	.0	99.46
24.8	174.2	0.665	.000	.87	1.13	.000	6387.5	6387.5	.0	99.46
24.9	173.8	0.231	.000	1.04	1.07	.000	6365.7	6365.7	.0	99.46
25.3	173.5	1.954	.108	1.32	1.65	.254	6259.5	6259.5	.0	99.46
25.6	173.6	2.760	.158	1.37	2.41	.265	6198.0	6198.0	.0	99.46
26.4	173.8	2.449	.300	1.50	4.40	.331	6031.5	6031.5	.0	99.46
27.9	174.4	2.386	.462	1.68	8.11	.357	5763.0	5763.0	.0	99.46
31.2	176.1	4.329	.633	2.00	15.99	.572	6456.1	4766.3	.0	96.46
31.9	176.5	4.749	.655	2.10	16.12	.603	6135.9	4446.7	.0	96.30
32.9	177.1	4.580	.679	2.23	16.12	.602	5996.7	4307.5	.0	96.23
35.4	178.7	4.268	.710	2.51	16.12	.600	5747.1	4057.8	.0	96.09
40.7	182.6	3.901	.733	3.00	16.12	.592	5320.2	3630.7	.0	95.82
46.8	187.5	3.634	.743	3.50	16.12	.582	4928.5	3238.8	.0	95.53
53.5	192.9	3.416	.747	4.00	16.12	.573	4576.3	2886.4	.0	95.23
60.9	198.9	3.224	.750	4.52	16.12	.563	4251.6	2561.6	.0	94.90
68.4	204.8	3.059	.751	5.01	16.12	.555	3971.4	2281.3	.0	94.58
77.0	210.8	2.168	.743	5.49	16.12	.428	1690.7	.0	.0	88.00

Table 6.4.1-33 (cont.)

Double-Ended Pump Suction Break-Maximum Safeguards

Principle Parameters During Reflood

Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time Seconds	Flooding Temp°F	Flooding Rate in/sec	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	FlowFrac	Total (Pounds Mass per second)	Injection Accum (Pounds Mass per second)	Spill (Pounds Mass per second)	Enthalph Btu/lbm
78.0	211.5	2.163	.743	5.54	16.12	.428	1690.7	.0	.0	88.00
88.1	218.6	2.115	.745	6.00	16.12	.430	1690.7	.0	.0	88.00
101.0	228.7	2.054	.748	6.57	16.12	.434	1690.7	.0	.0	88.00
111.2	236.6	2.006	.751	7.00	16.12	.436	1690.7	.0	.0	88.00
125.0	245.9	1.942	.754	7.57	16.12	.440	1690.7	.0	.0	88.00
136.2	252.3	1.892	.757	8.00	16.12	.443	1690.7	.0	.0	88.00
151.0	259.6	1.828	.760	8.56	16.12	.448	1690.7	.0	.0	88.00
163.4	264.8	1.776	.763	9.00	16.12	.452	1690.7	.0	.0	88.00
179.0	270.4	1.712	.767	9.53	16.12	.458	1690.7	.0	.0	88.00
193.5	274.9	1.654	.770	10.00	16.12	.464	1690.7	.0	.0	88.00

Table 6.4.1-34 Double-Ended Pump Suction Break Maximum Safeguards Post-Reflood Mass and Energy Releases Byron Unit 2 & Braidwood Unit 2 (D5 SG)				
	Break Path No. 1*		Break Path No. 2**	
Time (seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
193.5	173.0	216.2	1517.9	262.2
198.5	173.1	216.4	1517.8	261.9
203.5	172.4	215.5	1518.4	261.8
208.5	172.8	216.0	1518.1	261.5
213.5	172.2	215.3	1518.6	261.3
218.5	172.6	215.7	1518.3	261.0
223.5	172.0	215.0	1518.8	260.9
228.5	172.3	215.5	1518.5	260.5
233.5	172.7	215.9	1518.2	260.2
238.5	172.1	215.1	1518.8	260.1
243.5	172.4	215.5	1518.4	259.7
248.5	171.8	214.8	1519.0	259.6
253.5	172.1	215.1	1518.7	259.3
258.5	171.5	214.4	1519.4	259.2
263.5	171.8	214.7	1519.1	258.8
268.5	172.0	215.1	1518.8	258.5
273.5	171.4	214.3	1519.4	258.4
278.5	171.6	214.6	1519.2	258.0
283.5	171.9	214.9	1519.0	257.7
288.5	171.2	214.1	1519.6	257.6
293.5	171.4	214.3	1519.4	257.3
298.5	171.7	214.6	1519.2	257.0
303.5	171.0	213.7	1519.9	256.9
308.5	171.2	214.0	1519.7	256.5
313.5	171.3	214.2	1519.5	256.2
318.5	170.6	213.3	1520.2	256.1
323.5	170.8	213.5	1520.1	255.8
328.5	170.9	213.6	1519.9	255.5

Table 6.4.1-34 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Post-Reflood Mass and Energy Releases Byron Unit 2 & Braidwood Unit 2 (D5 SG)				
	Break Path No. 1*		Break Path No. 2**	
Time (seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
333.5	171.0	213.8	1519.8	255.2
338.5	171.1	213.9	1519.7	254.9
343.5	170.3	213.0	1520.5	254.8
348.5	170.4	213.1	1520.4	254.5
353.5	170.5	213.1	1520.4	254.2
358.5	170.5	213.2	1520.3	253.9
363.5	170.5	213.2	1520.3	253.6
368.5	170.5	213.2	1520.3	253.4
373.5	170.5	213.2	1520.3	253.1
378.5	170.5	213.1	1520.4	252.8
383.5	170.4	213.0	1520.4	252.5
388.5	170.3	212.9	1520.5	252.3
393.5	170.2	212.8	1520.6	252.0
398.5	170.1	212.6	1520.8	251.8
403.5	170.0	212.6	1520.8	251.5
408.5	170.0	212.5	1520.8	251.2
413.5	170.0	212.5	1520.9	251.0
418.5	169.9	212.4	1520.9	250.7
423.5	169.8	212.3	1521.0	250.4
428.5	169.7	212.1	1521.2	250.2
433.5	169.5	211.9	1521.3	249.9
438.5	169.3	211.7	1521.5	249.7
443.5	169.1	211.5	1521.7	249.5
448.5	169.6	212.0	1521.2	249.1
453.5	169.3	211.7	1521.5	248.9
458.5	169.0	211.3	1521.8	248.6
463.5	169.3	211.7	1521.5	248.3
468.5	168.9	211.2	1521.9	248.1

Table 6.4.1-34 (cont.)
Double-Ended Pump Suction Break
Maximum Safeguards
Post-Reflood Mass and Energy Releases
Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time (seconds)	Break Path No. 1* Flow (lbm/sec)	Break Path No. 1* Energy (Thousand Btu/sec)	Break Path No. 2** Flow (lbm/sec)	Break Path No. 2** Energy (Thousand Btu/sec)
473.5	169.2	211.5	1521.7	247.7
478.5	169.3	211.7	1521.5	247.4
483.5	168.8	211.0	1522.1	247.3
488.5	168.8	211.1	1522.0	247.0
493.5	168.8	211.0	1522.1	246.7
498.5	168.7	210.9	1522.2	246.4
503.5	169.1	211.4	1521.7	246.0
508.5	168.8	211.1	1522.0	245.8
513.5	168.4	210.6	1522.4	250.3
518.5	168.5	210.7	1522.3	249.9
523.5	168.5	210.7	1522.3	249.6
528.5	168.3	210.4	1522.5	249.4
533.5	168.5	210.7	1522.3	249.0
538.5	168.5	210.7	1522.3	248.7
543.5	168.3	210.4	1522.5	248.4
548.5	168.3	210.4	1522.5	248.1
553.5	168.5	210.6	1522.4	247.7
558.5	168.2	210.3	1522.6	247.4
563.5	168.4	210.5	1522.4	247.1
568.5	91.4	114.3	1599.4	267.3
693.5	87.6	109.5	1603.3	264.0
695.0	87.5	109.4	1520.7	353.2
780.6	87.5	109.4	1520.7	353.2
780.7	94.0	116.6	1514.2	349.5
785.0	93.9	116.5	1514.4	349.2
1254.2	93.9	116.5	1514.4	349.2
1254.3	84.0	96.6	1524.3	239.3
3600.0	64.2	73.9	1544.0	242.8

Table 6.4.1-34 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Post-Reflood Mass and Energy Releases Byron Unit 2 & Braidwood Unit 2 (D5 SG)				
	Break Path No. 1*		Break Path No. 2**	
Time (seconds)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
3600.1	53.6	61.6	1554.7	230.1
10000.0	39.0	44.8	1569.3	232.3
100000.0	20.8	24.0	1587.4	234.9
1000000.0	8.9	10.3	1599.3	236.7
10000000.0	2.8	3.2	1605.4	237.6

* mass and energy exiting the SG side of the break

** mass and energy exiting the pump side of the break

Table 6.4.1-35

Double-Ended Pump Suction Break Mass Balance

Maximum Safeguards

Byron Unit 2 & Braidwood Unit 2 (D5 SG)

		Mass Balance						
Time (Seconds)		.00	23.80	23.80	193.45	780.67	1254.25	3600.00
		Mass (Thousand lbm)						
Initial	In RCS and ACC	60.82	760.82	760.82	760.82	760.82	760.82	760.82
Added Mass	Pumped Injection	.00	.00	.00	276.18	1261.91	2023.55	5796.09
	Total Added	.00	.00	.00	276.18	1261.91	2023.55	5796.09
***** TOTAL AVAILABLE *****		60.82	760.82	760.82	1037.00	2022.73	2784.37	6556.91
Distribution	Reactor Coolant	511.13	42.55	78.23	142.75	142.75	142.75	142.75
	Accumulator	249.69	209.32	173.64	.00	.00	.00	.00
	Total Contents	760.82	251.87	251.87	142.75	142.75	142.75	142.75
Effluent	Break Flow	.00	518.20	518.20	891.84	1877.57	2639.15	6411.68
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	518.20	518.20	891.84	1877.57	2639.15	6411.68
***** TOTAL ACCOUNTABLE *****		760.82	770.07	770.07	1034.60	2020.33	2781.90	6554.44

Table 6.4.1-36
Double-Ended Pump Suction Break Energy Balance
Maximum Safeguards
Byron Unit 2 & Braidwood Unit 2 (D5 SG)

		Energy Balance						
Time (Seconds)		.00	23.80	23.80	193.45	780.67	1254.25	3600.00
		Energy (Million Btu)						
Initial Energy	In RCS, ACC, S GEN	865.38	865.38	865.38	865.38	865.38	865.38	865.38
Added Energy	Pumped Injection	.00	.00	.00	24.30	119.31	232.04	790.37
	Decay Heat	.00	8.17	8.17	29.41	83.57	120.02	263.31
	Heat From Secondary	.00	1.22	1.22	1.22	9.17	14.05	14.05
	Total Added	.00	9.39	9.39	54.93	212.06	366.11	1067.73
*** TOTAL AVAILABLE ***		865.38	874.77	874.77	920.31	1077.44	1231.49	1933.11
Distribution	Reactor Coolant	308.94	10.27	13.81	38.06	38.06	38.06	38.06
	Accumulator	24.83	20.82	17.27	.00	.00	.00	.00
	Core Stored	23.98	12.69	12.69	4.91	4.71	4.48	3.33
	Primary Metal	153.18	145.93	145.93	117.29	81.48	67.88	51.65
	Secondary Metal	108.26	109.48	109.48	99.03	72.88	57.26	44.03
	Steam Generator	246.19	249.88	249.88	220.52	161.58	130.68	102.68
	Total Contents	865.38	549.07	549.07	479.80	358.70	298.35	239.75
Effluent	Break Flow	.00	325.93	325.93	431.33	709.56	913.95	1679.40
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	325.93	325.93	431.33	709.56	913.95	1679.40
*** TOTAL ACCOUNTABLE ***		865.38	875.00	875.00	911.13	1068.26	1212.30	1919.15

Table 6.4.1-37 Double-Ended Hot Leg Break Sequence of Events Byron Unit 2 & Braidwood Unit 2 (D5 SG)	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.8	Low Pressurizer Pressure SI Setpoint - 1715 psia reached in blowdown
14.1	Broken Loop Accumulator Begins Injecting Water
14.3	Intact Loop Accumulator Begins Injecting Water
22.6	End of Blowdown Phase

Table 6.4.1-38

Double-Ended Pump Suction Break

Minimum Safeguards Sequence of Events

Byron Unit 2 & Braidwood Unit 2 (D5 SG)

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
4.4	Low Pressurizer Pressure SI Setpoint - 1715 psia reached in blowdown
16.2	Broken Loop Accumulator Begins Injecting Water
16.6	Intact Loop Accumulator Begins Injecting Water
24.4	End of Blowdown Phase
44.4	Safety Injection Begins
69.25	Broken Loop Accumulator Water Injection Ends
72.85	Intact Loop Accumulator Water Injection Ends
193.25	End of Reflood Phase
1110.	Cold Leg Recirculation Begins
1.0E+07	Transient Modeling Terminated

Table 6.4.1-39 Double-Ended Pump Suction Break Maximum Safeguards Sequence of Events Byron Unit 2 & Braidwood Unit 2 (D5 SG)	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
4.0	Low Pressurizer Pressure SI Setpoint - 1715 psia reached in blowdown
16.3	Broken Loop Accumulator Begins Injecting Water
16.6	Intact Loop Accumulator Begins Injecting Water
23.8	End of Blowdown Phase
44.0	Safety Injection Begins
73.5	Broken Loop Accumulator Water Injection Ends
76.35	Intact Loop Accumulator Water Injection Ends
193.45	End of Reflood Phase
695.0	Cold Leg Recirculation Begins
1.0E+07	Transient Modeling Terminated

Table 6.4.1-40 LOCA Mass And Energy Release Analysis Core Decay Heat Fraction	
Time (sec)	Decay Heat Generation Rate (Btu/hr)
10	0.052293
15	0.049034
20	0.047562
40	0.041504
60	0.038493
80	0.036410
100	0.034842
150	0.032180
200	0.030432
400	0.026664
600	0.024486
800	0.022943
1000	0.021722
1500	0.019483
2000	0.017903
4000	0.014386
6000	0.012684
8000	0.011645
10000	0.010916
15000	0.010130
20000	0.009368
40000	0.007784
60000	0.006976
80000	0.006439
100000	0.006034
150000	0.005336
200000	0.004859
400000	0.003781
600000	0.003212
800000	0.002844
1000000	0.002589

6.4.2 Short-Term LOCA Mass and Energy Releases

6.4.2.1 Purpose

An evaluation was conducted to determine the effect of a power uprate on the short-term LOCA-related mass and energy (M&E) releases that support subcompartment analyses discussed in Chapter 6.2.1.2 of the Byron/Braidwood Stations Updated Final Safety Analysis Report (UFSAR). From the UFSAR (Reference 2) the following licensing position is stated. "Based on the main coolant loop piping leak-before-break (LBB) analysis and the scope outlined in General Design Criterion 4, the containment subcompartments need not be designed for pressurization loads due to postulated primary coolant loop piping breaks. The subcompartments are not necessary to containment function, and therefore, dynamic or non-static pressurization loads need not be considered; however, the evaluation of loads on the containment subcompartments are retained in the UFSAR. The following LOCA breaks are analyzed in the UFSAR for the following regions of the containment, 1) reactor cavity - a break of 150 in^2 at the reactor vessel inlet nozzle, 2) loop compartments - two breaks have been postulated for the loop compartments, these are a doubled-ended cold leg (DECL) guillotine break and a double-ended hot leg (DEHL) guillotine break, and 3) upper pressurizer cubicle - a double-ended pressurizer spray line break. This evaluation addresses the impact of the power uprate and other relevant issues on the current licensing basis for postulated breaks in the containment subcompartments.

6.4.2.2 Discussion and Evaluation

The subcompartment analysis is performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than three seconds) which accompanies a high-energy-line pipe rupture within the subcompartment. The magnitude of the pressure differential across the walls is a function of several parameters, which include the blowdown M&E release rates, the subcompartment volume, vent areas, and vent flow behavior. The blowdown M&E release rates are affected by initial RCS temperature conditions. Since short-term releases are linked directly to the critical mass flux, which increases with decreasing temperatures, the short-term LOCA releases would be expected to increase due to any reductions in RCS coolant temperature conditions. Short-term blowdown transients are characterized by a peak mass and energy release rate that occurs during a subcooled condition, thus the Zaloudek correlation, which models this condition, is currently used in the short-term LOCA mass and energy release analyses with the SATAN computer program.

This calculation was used to conservatively evaluate the impact of the changes in RCS temperature conditions due to the power uprate on the short-term releases. This was accomplished by maximizing the reservoir pressure, maximizing the RCS inlet and outlet temperatures for the current analysis of record, and minimizing the RCS inlet and outlet temperatures for the uprate data. Since this maximizes the change in short-term LOCA mass and energy releases, data representative of the lowest inlet and outlet temperatures with uncertainty subtracted was used for the power uprate evaluation.

Any changes in RCS volume, steam generator liquid/steam mass and volume, or other differences in units, have no effect on the releases because of the short duration of the postulated accident. Any volumetric changes are small and have no impact on the subcompartment model. Therefore, the only change that needs to be addressed for this program is the decreased RCS coolant temperatures.

For this evaluation, a RCS pressure of 2250 psia and a vessel/core inlet temperature of 542.0°F were considered for the uprating. Considering the temperature uncertainty of 10°F and the current licensing basis initial conditions, the following RCS temperature ranges were used for the evaluations herein:

Vessel/Core Inlet	542°F minus 10°F =532°F

A pressure uncertainty of 43 psi was also included in the evaluation.

Based upon the results of the evaluation, the current design basis LOCA-related mass and energy releases have previously been bounded by the Byron/Braidwood Stations T_{hot} Reduction Program, Reference 3.

Reactor Cavity and Loop Compartments

Per Reference 1, Byron Units 1 and 2 and Braidwood Units 1 and 2 are approved for Leak-Before-Break (LBB). LBB eliminates the dynamic effects of postulated primary loop pipe ruptures from the design basis. This means that the current breaks (a double-ended circumferential rupture of the reactor coolant cold leg and hot leg for the loop compartments, and a 150 in^2 reactor vessel inlet break for the reactor cavity region) no longer have to be considered for the short-term effects. Since the RCS piping has been eliminated from consideration, the large branch nozzles must be considered for design verification. This includes the surge line, accumulator line, and Residual Heat Removal (RHR) line. These smaller breaks, which are outside the cavity region, would result in minimal asymmetric pressurization in the reactor cavity region. Additionally, compared to the large RCS double-ended ruptures, the differential loadings are significantly reduced. For example, the peak break compartment pressure can be reduced by a factor of greater than two, and the peak differential across an adjacent wall can be reduced by a factor of greater than three, if the nozzle breaks are considered. Therefore, since Byron Units 1 and 2 and Braidwood Units 1 and 2 are approved for LBB, the decrease in mass and energy releases associated with the smaller RCS nozzle breaks, as compared to the larger RCS pipe breaks, more than offsets the increased releases associated with lower RCS initial coolant temperatures. The current licensing basis subcompartment analyses that consider breaks in the RCS remain bounding.

Upper Pressurizer Cubicle

For the upper pressurizer cubicle, the postulated double-ended spray line rupture was previously evaluated for a vessel/core inlet temperature of 528.3°F and a nominal pressurizer pressure of 2250 psia, as part of the Byron/Braidwood T_{hot} Reduction Program (Reference 3). The power uprate conditions of 532°F and 2283 psia are bounded by the smaller pressurizer break area identifed in Reference 3.

6.4.2.3 Results and Conclusion

The short-term LOCA-related mass and energy releases discussed in Chapter 6.2.1.2 of the UFSAR have been reviewed to assess the effects associated with the power uprate conditions for Byron Units 1 and 2 and Braidwood Units 1 and 2. Results show that the current design basis spray line releases have previously been bounded by the Reference 3 evaluations. Since Byron Units 1 and 2 and Braidwood Units 1 and 2 are approved for LBB, the decrease in mass and energy releases associated with the smaller RCS nozzle breaks, as compared to the larger RCS pipe breaks, more than offsets the increased releases associated with the power uprate conditions. The current licensing basis subcompartment analyses that consider breaks in the primary loop reactor coolant system piping (i.e., steam generator subcompartments and reactor cavity region), therefore, remain bounding.

The results of this evaluation were provided for use in the pressurizer subcompartment structural analysis.

6.4.2.4 References

1. Letter from Ramin R. Assa (NRC) to Ms. Irene Johnson (CECo), "Safety Evaluation (SE) Regarding Leak-Before-Break Analysis - Byron Station Units 1 and 2, and Braidwood Station, Units 1 and 2 (Tac Nos. M95342, M95343, M95344, and M95345)," October 25, 1996.

2. "Byron & Braidwood Station, Updated Final Safety Analysis Report,"Revision 7, Docket Nos. STN-454/455/456/457, as amended through December 1998.

3. WCAP-11387 Rev.2 (Non-Proprietary), "Byron/Braidwood Thot Reduction Final Licensing Report," November 1987

6.4.3 Long Term LOCA Containment Response

6.4.3.1 Accident Description

The Byron Units 1 and 2 and Braidwood Units 1 and 2 containment systems are designed such that for all loss-of-coolant accident (LOCA) break sizes, up to and including the double-ended severance of a reactor coolant pipe, the containment peak pressure remains below the design pressure. This section details the containment response subsequent to a hypothetical LOCA. The containment response analysis uses the long-term mass and energy (M&E) release data from Section 6.4.1.

The containment response analysis demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a LOCA inside containment. The impact of LOCA mass and energy releases on the containment pressure is addressed to assure that the containment pressure remains below its design pressure at the uprated conditions. In support of equipment design and licensing criteria (e.g., qualified operating life), with respect to post accident environmental conditions, long-term containment pressure and temperature transients are generated to conservatively bound the potential post-LOCA containment conditions.

6.4.3.2 Input Parameters and Assumptions

An analysis of containment response to the rupture of the RCS must start with knowledge of the initial conditions in the containment. The pressure, temperature, and humidity of the containment atmosphere prior to the postulated accident are specified in the analysis as shown in Table 6.4.3-1.

Also, values for initial temperature of the Service Water (SW) and Refueling Water Storage Tank (RWST) have been specified, along with Containment Spray (CS) pump flowrate and Reactor Containment Fan Cooler (RCFC) heat removal performance. These values are chosen conservatively, as shown in Table 6.4.3-1. Long-term sump recirculation is addressed via Residual Heat Removal (RHR) System heat exchanger performance. The primary function of the RHR System is to remove heat from the core by way of the Emergency Core Cooling System (ECCS). Table 6.4.3-1 provides the RHR System parameters assumed in the analysis.

A series of cases were performed for LOCA containment response. Section 6.4.1 documented the M&E releases for the minimum and maximum safeguards cases for a double-ended pump suction (DEPS) break and the releases from the blowdown of a double-ended hot leg (DEHL) break for the BWI and D5 steam generators.

For the maximum safeguards DEPS case, a failure of a containment spray pump was assumed as the single failure. This leaves one containment spray pump and four RCFCs available as active heat removal systems. Table 6.4.3-3 provides the performance data for one spray pump in operation. (Note: For the maximum safeguards case, a limiting assumption was made concerning the modeling of the recirculation system, i.e., heat exchangers. Minimum heat exchanger performance data was conservatively used to model the RHR heat exchangers, i.e., two RHR HXs were credited for residual heat removal. Emergency safeguards equipment data is given in Table 6.4.3-1.)

The minimum safeguards case was based upon a diesel train failure (which leaves available as active heat removal systems one containment spray pump and 2 RCFCs). Due to the duration of the DEHL transient (i.e., blowdown only), no containment safeguards equipment is modeled. The calculations for the DEPS minimum safeguards case were performed for 2.592 million seconds (approximately 30 days) and the maximum safeguards case 1.0 million seconds (approximately 11 days). The DEHL cases were terminated soon after the end of the blowdown. The sequence of events for each of these cases is shown in Tables 6.4.3-4 through 6.4.3-6 for the BWI steam generator and 6.4.3-7 through 6.4.3-9 for the Westinghouse D5 steam generator.

The following are the major assumptions made in the analysis. The mass and energy released to the containment are described in Section 6.4.1 for LOCA.

Homogeneous mixing is assumed. The steam-air mixture and water phases each have uniform properties. More specifically, thermal equilibrium between the air and steam is assumed. However, this does not imply thermal equilibrium between the steam-air mixture and water phase.

Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermodynamic properties.

For the blowdown portion of the LOCA analysis, the discharge flow separates into steam and water phases at the breakpoint. The saturated water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the containment. For the post-blowdown portion of the LOCA analysis, steam and water releases are input separately.

The saturation temperature at the partial pressure of the steam is used for heat transfer to the heat sinks and the Reactor Containment Fan Coolers.

6.4.3.3 Description of Analysis

Calculation of containment pressure and temperature is accomplished by use of the digital computer code COCO (Reference 1). COCO is a mathematical model of a generalized containment; the proper selection of various options in the code allows creation of a specific model for particular containment design. The values used in the specific model for different aspects of the containment are derived from plant-specific input data. The COCO code has been used and found acceptable to calculate containment pressure transients for many dry containment plants, most recently including Vogtle Units 1 and 2, Turkey Point Unit 3, Salem Units 1 and 2, Diablo Canyon Units 1 and 2, Indian Point Unit 2, and Indian Point 3. Transient phenomena within the reactor coolant system affect containment conditions by means of convective mass and energy transport through the pipe break.

For analytical rigor and convenience, the containment air-steam-water mixture is separated into a water (pool) phase and a steam-air phase. Sufficient relationships to describe the transient are provided by the equations of conservation of mass and energy as applied to each system, together with appropriate boundary conditions. As thermodynamic equations of state and conditions may vary during the transient, the equations have been derived for possible cases of superheated or saturated steam and subcooled or saturated water. Switching between states is handled automatically by the code.

Passive Heat Removal

The significant heat removal source during the early portion of the transient is the containment structural heat sinks. Provision is made in the containment response analysis for heat transfer through, and heat storage in, both interior and exterior walls. Each wall is divided into a large number of nodes. For each node, a conservation of energy equation expressed in finitedifference form accounts for heat conduction into and out of the node and temperature rise of the node. Table 6.4.3-10 is the summary of the containment structural heat sinks used in the analysis. The thermal properties of each heat sink material are shown in Table 6.4.3-11.

The heat transfer coefficient to the containment structure for the early part of the event is calculated based primarily on the work of Tagami (Reference 2). From this work, it was determined that the value of the heat transfer coefficient can be assumed to increase parabolically to a peak value. In COCO, the value then decreases exponentially to a stagnant heat transfer coefficient which is a function of steam-to-air-weight ratio. The h for stagnant conditions is based upon Tagami's steady state results.

Tagami presents a plot of the maximum value of the heat transfer coefficient, h, as a function of "coolant energy transfer speed", defined as follows:

$$h = \frac{\text{total coolant energy transferred into containment}}{\text{(containment volume) (time interval to peak pressure)}}$$

From this, the maximum heat transfer coefficient of steel is calculated:

$$h_{max} = 75\left(\frac{E}{t_p V}\right)^{0.60} \qquad \text{(Equation 1)}$$

where:

h_{max} = maximum value of h (Btu/hr ft^2 °F).

t_p = time from start of accident to end of blowdown for LOCA, and to steam line isolation for secondary breaks (sec).

V = containment net free volume (ft^3).

E = total coolant energy discharge from time zero to t_p(Btu).

75 = material coefficient for steel.

(Note: Paint is accounted for by the thermal conductivity of the material (paint) on the heat sink structure, not by an adjustment on the heat transfer coefficient.) The basis for the equations is a Westinghouse curve fit to the Tagami data.

The parabolic increase to the peak value is calculated by COCO according to the following equation:

$$h_s = h_{max}\left(\frac{t}{t_p}\right)^{0.5}, 0 \le t \le t_p \qquad \text{(Equation 2)}$$

where:

h_s = heat transfer coefficient between steel and air/steam mixture (Btu/hr ft^2 °F).

t = time from start of event (sec).

For concrete, the heat transfer coefficient is taken as 40 percent of the value calculated for steel during the blowdown phase.

The exponential decrease of the heat transfer coefficient to the stagnant heat transfer coefficient is given by:

$$h_s = h_{stag} + (h_{max} - h_{stag})e^{-0.05(t-t_p)}\ t > t_p \qquad \text{(Equation 3)}$$

where:

h_{stag} = 2 + 50X, 0 < X < 1.4.

h_{stag} = h for stagnant conditions (Btu/hr ft^2 °F).

X = steam-to-air weight ratio in containment.

Active Heat Removal

For a large break, the engineered safety features are quickly brought into operation. Because of the brief period required to depressurize the reactor coolant system or main steam system, the containment safeguards are not a major influence on the blowdown peak pressure; however, they reduce the containment pressure after blowdown and maintain a low long-term pressure and a low long-term temperature.

RWST Injection

During the injection phase of post-accident operation, the Emergency Core Cooling System pumps water from the Refueling Water Storage Tank (RWST) into the reactor vessel. Since this water enters the vessel at RWST temperature, which is less than the temperature of the water in the vessel, it is modeled as absorbing heat from the core until the saturation temperature is reached. Safety injection and containment spray can be operated for a limited time, depending on the RWST capacity.

RHR Sump Recirculation

After the supply of refueling water is exhausted, the recirculation system is operated to provide long-term cooling of the core. In this operation, water is drawn from the sump, cooled in a Residual Heat Removal (RHR) System heat exchanger, then pumped back into the reactor vessel to remove core residual heat and energy stored in the vessel metal. The heat is removed from the RHR heat exchanger by Component Cooling Water (CCW). The RHR and CCW HXs are coupled in a closed loop system, where the ultimate heat sink is the service water cooling to the CCW HX.

Containment Spray

Containment spray (CS) is an active removal mechanism used for rapid pressure reduction and for containment iodine removal. During the injection phase of operation, the containment spray pumps draw water from the RWST and spray it into the containment through nozzles mounted high above the operating deck. As the spray droplets fall, they absorb heat from the containment atmosphere. Since the water comes from the RWST, the entire heat capacity of the spray, from the RWST temperature to the temperature of the containment atmosphere, is available for energy absorption. During the recirculation phase the sprays take suction directly from the sump. However, the spray water is not cooled by the RHR Heat Exchangers.

When a spray droplet enters the hot, saturated, steam-air containment environment, the vapor pressure of the water at its surface is much less than the partial pressure of the steam in the atmosphere. Hence, there will be diffusion of steam to the drop surface and condensation on the droplet. This mass flow will carry energy to the droplet. Simultaneously, the temperature difference between the atmosphere and the droplet will cause the droplet temperature and vapor pressure to rise. The vapor pressure of the droplet will eventually become equal to the partial pressure of the steam, and condensation will cease. The temperature of the droplet will essentially equal the temperature of the steam-air mixture.

The equations describing the temperature rise of a falling droplet are as follows.

$$\frac{d}{dt}(Mu) = mh_g + q \qquad \text{(Equation 4)}$$

where,

M = droplet mass
u = internal energy
m = diffusion rate
h_g = steam enthalpy
q = heat flow rate
t = time

$$\frac{d}{dt}(M) = m \qquad \text{(Equation 5)}$$

where,

q = $h_cA * (T_s - T)$
m = $k_gA * (P_s - Pv)$
A = drop surface area
h_c = coefficient of heat transfer
k_g = coefficient of mass transfer
T = droplet temperature
T_s = steam temperature
P_s = steam partial pressure
P_v = droplet vapor pressure

The coefficients of heat transfer (h_c) and mass transfer (k_g) are calculated from the Nusselt number for heat transfer, Nu, and the Nusselt number for mass transfer, Nu'.

Both Nu and Nu' may be calculated from the equations of Ranz and Marshall (Reference 3).

$$Nu = 2 + 0.6(Re)^{1/2}(Pr)^{1/3} \quad \text{(Equation 6)}$$

where,

Nu = Nusselt number for heat transfer

Pr = Prandtl number

Re = Reynolds number

$$Nu' = 2 + 0.6(Re)^{1/2}(Sc)^{1/3} \quad \text{(Equation 7)}$$

where,

Nu' = Nusselt number for mass transfer

Sc = Schmidt number

Thus, Equations 4 and 5 can be integrated numerically to find the internal energy and mass of the droplet as a function of time as it falls through the atmosphere. Analysis shows that the temperature of the (mass) mean droplet produced by the spray nozzles rises to a value within 99 percent of the bulk containment temperature in less than 2 seconds. Detailed calculations of the heatup of spray droplets in post-accident containment atmospheres by Parsly (Reference 4) show that droplets of the size encountered in the containment spray reach equilibrium in a fraction of their residence time in a typical pressurized water reactor containment. These results confirm the assumption that the containment spray will be 100 percent effective in removing heat from the atmosphere.

Reactor Containment Fan Coolers

The Reactor Containment Fan Coolers (RCFCs) are another means of heat removal. Each RCFC has a fan which draws in the containment atmosphere from the upper volume of the containment via a return air riser. The RCFCs are cooled by the Essential Service Water (ESW) and, during the post-LOCA mode of operation, the fan speed is reduced to compensate for the higher air density inside containment due to the LOCA pressurization. The steam/air mixture is routed through the enclosed RCFC unit, past Essential Service Water cooling coils. The RCFC then discharges the air through ducting containing a check damper. The discharged air is directed at the lower containment volume. See Table 6.4.3-2 for RCFCs heat removal capability assumed for the containment response analyses.

6.4.3.4 Acceptance Criteria

A Loss-Of-Coolant-Accident (LOCA) is an ANS Condition IV event, an infrequent fault. The relevant containment integrity requirements for a design-basis LOCA are as follows.

1. GDC 10 and GDC 49 from the UFSAR Chapter 3.1. In order to satisfy the requirements of GDC 10 and 49, the peak calculated containment pressure should be less than the containment design pressure of 50 psig.

2. UFSAR Chapter 3.1, GDC 52.

3. UFSAR Chapter 15.6.5 requirement that the calculated pressure at 24 hours should be less than 50% of the peak calculated value.

6.4.3.5 Analysis Results

The containment pressure, steam temperature and water (sump) temperature profiles for the Double Ended Pump Suction (DEPS) LOCA cases are shown in Figures 6.4.3-1 through 6.4.3-4 for the BWI replacement steam generator and Figures 6.4.3-7 through 6.4.3-10 for the Westinghouse D5 steam generator. The results of the DEHL break with minimum ECCS flows are shown in Figures 6.4.3-5 through 6.4.3-6 for the BWI replacement steam generator and Figures 6.4.3-11 through 6.4.3-12 for the Westinghouse D5 steam generator. Tables 6.4.3-13 through 6.4.3-15 provide detailed results for the analyses using mass and energy releases based on the BWI replacement steam generator, while Tables 6.4.3-17 through 6.4.3-19 provide similar information based on the Westinghouse D5 steam generator design.

6.4.3.5.1 DEPS Break with Minimum Safeguards

This analysis assumes a loss of offsite power coincidence with a double ended rupture of the RCS piping between the steam generator outlet and the RCS pump inlet (suction). The associated single failure assumption is the failure of a diesel to start, resulting in one train of ECCS and containment safeguards equipment being available. This combination results in a minimum set of safeguards being available. Further, loss of offsite power delays the actuation times of the safeguards equipment due to diesel startup time after receipt of the Safety Injection signal.

Transient Description With BWI Steam Generator

Table 6.4.3-4 provides the sequence of events for this case and the peak pressure result of 41.84 psig is shown in Table 6.4.3-12. The postulated RCS break results in a rapid release of mass and energy to the containment with a resulting rapid rise in both containment pressure and temperature. The rapid rise in containment pressure results in generation of a containment Hi-1 signal at 1.19 seconds and a containment Hi-3 signal at 7.69 seconds.

The containment pressure continues to rise rapidly in response to the release of mass and energy until the end of blowdown at 27.2 seconds, with the pressure reaching a value of 40.4 psig. The end of blowdown marks a time when the initial inventory in the RCS has been exhausted and a slow process of filling the RCS downcomer, in preparation for reflood, has begun. During this period, the RCFCs start at 65.0 seconds. Since the mass and energy release during this period is low and the RCFCs are removing heat, the pressure decreases slightly to 37.7 psig at approximately 74 seconds, when the intact loop accumulators have emptied.

The pressure then starts to slowly rise in response to the loss of steam condensation in the RCS loops and the introduction of the accumulator nitrogen gas to the containment. Containment spray initiation occurs at 88.1 seconds. Reflood continues at a reduced flooding rate due to the buildup of mass in the RCS core, which offsets the downcomer head. This reduction in flooding rate and the continued action of the RCFCs and spray leads to a slowly decreasing pressure out to the end of reflood, which occurs at 198.02 seconds.

At this juncture, by design of the Reference 5 model, energy removal from the SG secondaries begins at a greatly increased rate, resulting in a rise in containment pressure from 198.02 seconds out to 463.1, seconds when energy has been removed from the SG in the faulted loop, bringing the SG secondary pressure down to the containment design pressure of 50 psig. The result of this SG secondary energy release is a containment pressure of 41.84 psig at 460.56 seconds, the ultimate peak pressure for this transient.

After this occurs, the mass and energy released is reduced, due to the large amount of energy already removed from the SGs, and pressure slowly falls out to the cold leg recirculation time of 1110. seconds. At this time, the RHR is realigned for cold leg recirculation, resulting in an increase in the SI temperature due to delivery from the hot sump, and a reduction in steam condensation with a concomitant increase in containment pressure.

By 1500 seconds the SG secondary energy has been greatly reduced and the containment pressure begins a steady decline. Containment spray realignment for sump recirculation occurs at 3778 seconds, resulting in a slight increase in containment pressure as a result of spraying the hot sump water, for about 2000 seconds until the safeguards energy removal once again exceeds decay heat addition at about 6000 seconds. This trend continues to the end of the transient at 2.592 E+06 seconds (~30 days).

Transient Description With Westinghouse D5 Steam Generator

The DEPS minimum ECCS assumptions containment transient, using mass and energy releases developed with the Westinghouse D5 steam generator, follows a similar sequence of events as calculated with the BWI steam generator. Table 6.3.4-7 provides the key sequence of events and Table 6.4.3-16 shows a peak pressure of 37.71 psig at 399 seconds.

6.4.3.5.2 DEPS Break with Maximum Safeguards

The DEPS break with maximum safeguards has a transient history very similar to the minimum safeguards case discussed in Section 6.4.3.5.1. For this case, the single failure is the loss of one spray pump. Table 6.4.3-5 provides the key sequence of events and Table 6.4.3-12 shows that a peak pressure of 40.85 psig @ 22.89 seconds was calculated for the BWI steam generator. Calculational results using mass and energy releases based on the Westinghouse D5 steam generator are found in Table 6.4.3-8 and Table 6.4.3-16 shows a peak pressure of 36.77 psig at 21.01 seconds

6.4.3.5.3 Double Ended Hot Leg Break with Minimum Safeguards

This analysis assumes a loss of offsite power coincident with a double ended rupture of the RCS piping between the reactor vessel outlet nozzle and the steam generator inlet (i.e., A break in the RCS hot leg). The associated single failure assumption is the failure of a diesel to start, resulting in one train of ECCS and containment safeguards equipment being available. This combination results in a minimum set of safeguards being available. Further, loss of offsite power delays the actuation times of the safeguards equipment due to the diesel startup time after receipt of the Safety Injection signal.

Transient Description With BWI Steam Generator

The postulated RCS break results in a rapid release of mass and energy to the containment with a resulting rapid rise in both the containment pressure and temperature. See Table 6.4.3-6 for the sequence of events and Table 6.4.3-12 for the peak pressure and temperature. This rapid rise in containment pressure results in the generation of a containment Hi-1 signal at 1.164 seconds and a containment Hi-3 signal at 7.068 seconds. The containment pressure continues to rise rapidly in response to the release of mass and energy until the end of blowdown at 25.6 seconds, with the pressure reaching a value of 42.77 psig at 22.116 seconds. The end of blowdown marks a time when the initial inventory in the RCS has been exhausted and a process of filling the RCS downcomer in preparation for reflood has begun. Since the reflood for a hot leg break is very fast due to the low resistance to steam venting posed by the broken hot leg, Westinghouse terminates hot leg break mass and energy release transients at end of blowdown. The basis for this is further developed in References 5 and 6.

Transient Description With Westinghouse D5 Steam Generator

The containment transient from a DEHL break with minimum ECCS, using mass and energy releases developed with the Westinghouse D5 steam generator, follows a similar sequence of events as calculated with the BWI steam generator. Table 6.3.4-9 provides the key sequence of events and Table 6.4.3-16 shows a peak pressure of 38.36 psig at 21.079 seconds.

6.4.3.5.4 DEHL Break with Maximum Safeguards

The DEHL break with maximum safeguards was not analyzed since neither the ECCS pumps or containment safeguards start prior to the end of blowdown. Thus, the maximum ECCS case would be identical to the minimum ECCS case discussed in 6.4.3.5.3.

6.4.3.6 Conclusions

The LOCA containment response analyses have been performed as part of the Byron Units 1 and 2 and Braidwood Units 1 and 2 uprate program. The analyses included long-term pressure and temperature profiles for each case, including analyses with mass and energy releases based on the BWI replacement steam generator and the Westinghouse D5 steam generator. As shown by the results in Section 6.4.3-5, the analyzed cases resulted in a peak containment pressure that was less than the containment design pressure of 50 psig. The long-term pressures are well below 50% of the peak value within 24 hours. Based on these results, the applicable LOCA criteria for Byron Units 1 and 2 and Braidwood Units 1 and 2 have been met.

6.4.3.7 References

1 "Containment Pressure Analysis Code (COCO)," WCAP-8327, July 1974 (Proprietary), WCAP-8326, July, 1974 (Non-Proprietary).

2 Takashi Tagami, "Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June 1965," No.1.

3 D. W. Ranz and W. R. Marshall, Jr., "Evaporation for Drops," Chemical Engineering Progress, 48, pp.141-146, March 1952.

4 Parsly, L. F., "Design Consideration of Reactor Containment Spray System. Part VI, The Heating of Spray Drops in Air-Steam Atmospheres," ORNL-TM-2412 Part VI, January 1970.

5 "Westinghouse LOCA Mass and Energy Release Model for Containment Design – March 1979 Version," WCAP-10325-P-A, May 1983 (Proprietary), WCAP-10326-A

6 "Westinghouse Mass and Energy Release Data For Containment Design," WCAP-8264-P-A, Rev. 1, August 1975 (Proprietary), WCAP-8312-A (Nonproprietary)

Table 6.4.3-1 LOCA Containment Response Analysis Parameters	
Service water temperature (°F)	100
RWST water temperature (°F)	120
Initial containment temperature (°F)	120
Initial containment pressure (psia)	15.7
Initial relative humidity (%)	20
Net free volume (ft^3)	2.758×10^6
Reactor Containment Air Recirculation Fan Coolers	
Total	4
Analysis maximum	4
Analysis minimum	2
Containment Hi-1 setpoint (psig)	6.8
Delay time (sec) With Offsite Power Without Offsite Power	 27.0 65.0
Containment Spray Pumps	
Total	2
Analysis maximum	1
Analysis minimum	1
Flowrate (gpm) Injection phase (per pump)- see Table 6.4.3-3 Recirculation phase (total)	 3285
Containment Hi-3 setpoint (psig)	24.8
Delay time (sec) With Offsite Power (delay after High High setpoint) Without Offsite Power (total time from t=0)	 53.1 88.1
ECCS Recirculation Switchover, sec Minimum Safeguards Maximum Safeguards	 1110. 695.
Containment Spray Recirculation Switchover, (sec) Minimum Safeguards Maximum Safeguards	 3778. 3363.

Table 6.4.3-1 (cont.) **LOCA Containment Response Analysis Parameters**	
Emergency Core Cooling System (ECCS) Flows (GPM)	
Minimum ECCS	
Injection alignment	5686.
Recirculation alignment	994.1
Maximum ECCS	
Injection alignment	12305
Recirculation alignment	11917.1
Residual Heat Removal System Heat Exchangers	
Modeled in analysis *	1
Recirculation switchover time, sec	
Minimum Safeguards Maximum Safeguards	1110. 695.
UA, 10^6 * BTU/hr-°F	2.16
Component Cooling (Shell Side) Flow gpm	5000
Component Cooling Water Heat Exchangers	
Modeled in analysis	1
UA, 10^6 * BTU/hr-°F	+4.73
Flows – gpm	
Shellside *	5000.
Tubeside * (service water)	5000.
Additional heat loads, (BTU/hr)	6.8×10^6

* Minimum safeguard data representing 1 EDG

Table 6.4.3-2 Reactor Containment Fan Cooler Performance	
Containment Temperature (°F)	**Heat Removal Rate [Btu/sec] Per RCFC**
100	0.00
110	893.54
130	3181.61
160	8057.82
190	14535.02
220	21896.92
250	29430.06
271	34613.85
300	41813.19
350	54225.24

Table 6.4.3-3 Containment Spray Performance	
Containment Pressure (psig)	With 1 Pump (gpm)
0	3285.
10	3285.
20	3285.
30	3285.
40	3285.
50	3285.

Table 6.4.3-4 Double-Ended Pump Suction Break Sequence of Events for Byron Unit 1 and Braidwood Unit 1 (BWI SG) Minimum Safeguards	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
1.19	Containment HI-1 Pressure Setpoint Reached
3.6	Low Pressurizer Pressure SI Setpoint = 1715 psia Reached (Safety Injection Begins coincident with Low Pressurizer Pressure SI Setpoint)
7.69	Containment HI-3 Pressure Setpoint Reached
18.3	Broken Loop Accumulator Begins Injecting Water
18.6	Intact Loop Accumulator Begins Injecting Water
27.2	End of Blowdown Phase
43.6	Safety Injection Begins
65.0	Reactor Containment Air Recirculation Fan Coolers Actuate
73.72	Broken Loop Accumulator Water Injection Ends
74.22	Intact Loop Accumulator Water Injection Ends
88.1	Containment Spray Pump(s) (RWST) start
198.02	End of Reflood for MIN SI Case
460.56	Peak Pressure and Temperature Occur
1110.	RHR/HHSI/CHRGSI alignment for recirculation
3778.	Containment Spray is aligned for Recirculation
2.592E6	Transient Modeling Terminated

Table 6.4.3-5 Double-Ended Pump Suction Break Sequence of Events for Byron Unit 1 and Braidwood Unit 1 (BWI SG) Maximum Safeguards	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
1.22	Containment HI-1 Pressure Setpoint Reached
3.61	Low Pressurizer Pressure SI Setpoint = 1715 psia Reached (Safety Injection Begins coincident with Low Pressurizer Pressure SI Setpoint)
7.74	Containment HI-3 Pressure Setpoint Reached
18.3	Broken Loop Accumulator Begins Injecting Water
18.6	Intact Loop Accumulator Begins Injecting Water
22.89	Peak Pressure and Temperature Occur
26.8	End of Blowdown Phase
28.22	Reactor Containment Air Recirculation Fan Coolers Actuate
43.61	Safety Injection Begins
60.84	Containment Spray Pump(s) (RWST) start
78.01	Broken Loop Accumulator Water Injection Ends
78.46	Intact Loop Accumulator Water Injection Ends
197.7	End of Reflood for Max SI Case
695.0	RHR/HHSI/CHRG SI aligned for recirculation
3363.	Containment Spray is aligned for Recirculation
1.0E6	Transient Modeling Terminated

Table 6.4.3-6 Double-Ended Hot Leg Break Sequence of Events for Byron Unit 1 and Braidwood Unit 1 (BWI SG) Minimum Safeguards	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
1.164	Containment HI-1 Pressure Setpoint Reached
3.9	Low Pressurizer Pressure SI Setpoint = 1715 psia reached
7.068	Containment HI-3 Pressure Setpoint Reached
15.3	Broken Loop Accumulator Begins Injecting Water
15.5	Intact Loop Accumulator Begins Injecting Water
22.116	Peak Pressure and Temperature Occur
25.6	End of Blowdown Phase
30.0	Transient Modeling Terminated

Table 6.4.3-7 Double-Ended Pump Suction Break Sequence of Events for Byron Unit 2 and Braidwood Unit 2 (D5 SG) Minimum Safeguards	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
1.185	Containment HI-1 Pressure Setpoint Reached
4.4	Low Pressurizer Pressure SI Setpoint = 1715 psia Reached (Safety Injection Begins coincident with Low Pressurizer Pressure SI Setpoint)
8.107	Containment HI-3 Pressure Setpoint Reached
16.2	Broken Loop Accumulator Begins Injecting Water
16.6	Intact Loop Accumulator Begins Injecting Water
24.4	End of Blowdown Phase
44.4	Safety Injection Begins
65.0	Reactor Containment Air Recirculation Fan Coolers Actuate
69.25	Broken Loop Accumulator Water Injection Ends
72.85	Intact Loop Accumulator Water Injection Ends
88.1	Containment Spray Pump(s) (RWST) start
193.25	End of Reflood for MIN SI Case
399.0	Peak Pressure and Temperature Occur
1110.	RHR/HHSI/CHRGSI alignment for recirculation
3778.	Containment Spray is aligned for Recirculation
1.0E7	Transient Modeling Terminated

Table 6.4.3-8
Double-Ended Pump Suction Break
Sequence of Events for Byron Unit 2 and
Braidwood Unit 2 (D5 SG) Maximum Safeguards

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
1.18	Containment HI-1 Pressure Setpoint Reached
4.0	Low Pressurizer Pressure SI Setpoint = 1715 psia Reached (Safety Injection Begins coincident with Low Pressurizer Pressure SI Setpoint)
8.19	Containment HI-3 Pressure Setpoint Reached
16.3	Broken Loop Accumulator Begins Injecting Water
16.6	Intact Loop Accumulator Begins Injecting Water
21.01	Peak Pressure and Temperature Occur
23.8	End of Blowdown Phase
28.18	Reactor Containment Air Recirculation Fan Coolers Actuate
44.0	Safety Injection Begins
61.29	Containment Spray Pump(s) (RWST) start
73.50	Broken Loop Accumulator Water Injection Ends
76.35	Intact Loop Accumulator Water Injection Ends
193.45	End of Reflood for Max SI Case
695.0	RHR/HHSI/CHRG SI aligned for recirculation
3363.	Containment Spray is aligned for Recirculation
1.0E7	Transient Modeling Terminated

Table 6.4.3-9 Double-Ended Hot Leg Break Sequence of Events for Byron Unit 2 and Braidwood Unit 2 (D5 SG) Minimum Safeguards	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
1.21	Containment HI-1 Pressure Setpoint Reached
3.8	Low Pressurizer Pressure SI Setpoint = 1715 psia reached
7.274	Containment HI-3 Pressure Setpoint Reached
14.1	Broken Loop Accumulator Begins Injecting Water
14.3	Intact Loop Accumulator Begins Injecting Water
21.08	Peak Pressure and Temperature Occur
22.6	End of Blowdown Phase
30.0	Transient Modeling Terminated

Table 6.4.3-10 Containment Heat Sinks				
No.	Description	Material	Thickness (ft)	Surface Area (ft^2)
1	Containment Cylinder Wall	Paint Carbon Steel Concrete	8.30E-04 0.0208 0.75	72,741
2	Containment Dome	Paint Carbon Steel Concrete	8.30E-04 0.0208 0.75	17,550
3	Unlined Concrete – Combined from Containment Floor and the Slab at 425'-0"	Concrete (in contact w/water)	0.75	16,037
4	Lined Concrete – Combined from Containment Floor and Reactor Pool Wall	Stainless Steel Concrete	4.15E-02 0.75	848
5	Unlined Concrete – Combined from Reactor Cavity, Outside Reactor Wall, and Reactor Pool Wall	Concrete	1.0	4,803
6	Lined Concrete – Secondary Wall	Paint Carbon Steel Concrete	8.30E-04 0.0766 0.75	7,702
7	Lined Concrete – Slab at 425'-0", lining only	Paint Carbon Steel	8.30E-04 0.0625	422.3
8	Unlined Concrete – Combined from Slabs on Steel Beams at El. 412'-0" and 426'-0", Instrument Access Tunnel, and enclosures for SG, RCPs, etc.	Concrete	0.75	69,541
9	Lined Concrete – Slabs on Steel Beams at El. 412'-0" and 426'-0"	Paint Carbon Steel Concrete	8.30E-04 0.004 0.75	3,852
10	Lined Concrete – Enclosures for SGs. RCPs. etc.	Paint Carbon Steel Concrete	8.30E-04 0.0710 0.75	1,571**
11	Lined Concrete – Refueling Cavity	Stainless Steel Concrete	0.0690 0.75	2,129

Table 6.4.3-10 (cont.) Containment Heat Sinks				
No.	**Description**	**Material**	**Thickness (ft)**	**Surface Area (ft^2)**
12	Miscellaneous Steel Plate, HVAC Hangers, Polar Crane Trolley and Bridge Plates, and NSSS Supports	Paint Carbon Steel	8.30E-04 0.0416	19,792**
13	Miscellaneous Steel Plate, Grating, Press. Relief Tank, Polar Crane Bridge Plates, and Return Air Riser	Paint Carbon Steel	8.30E-04 0.0210	94,672**
14	Polar Crane Trolley and Bridge Plates and Machinery	Paint Carbon Steel	8.30E-04 0.0760	14,085**
15	Combined from Man. Crane Fan, RCFC Fan, and Reactor Cavity Fans	Paint Carbon Steel	8.30E-04 0.0400	21,875
16	Combined from CCFU*Housing, HVAC Hangers, Uninsulated Pipe, Ductwork, and Duct Supports	Paint Carbon Steel	8.30E-04 0.0150	22,528
17	Cable/Conduit Trays	Paint Carbon Steel	8.30E-04 0.0104	27,095
18	Combined from Cable/Conduit Tray Supports, Junction Boxes, and IFME*	Paint Carbon Steel	8.30E-04 8.20E-03	6,385
19	Combined from CFU,*and Miscellaneous Steel Beams and Columns	Paint Carbon Steel	8.30E-04 0.0157	69,856
20	Lined Concrete – Combined from the Instrument Access Tunnel, Reactor Cavity, and Inside Reactor Pool	Stainless Steel Concrete	0.0165 0.75	9,291

* CCFU – Containment Charcoal Filter Unit

IFME – Incore Flux Mapping Equipment

CFU – Charcoal Filter Unit

** These values differ slightly from those in the UFSAR. However, these differences will have no affect on the analysis results or conclusions.

Table 6.4.3-11 Thermophysical Properties of Containment Heat Sinks		
Material	**Thermal Conductivity (Btu/hr-ft - °F)**	**Volumetric Heat Capacity (Btu/ft^3 - °F)**
Paint	0.30	28.00
Carbon Steel	27.00	58.80
Stainless Steel	9.00	53.70
Concrete	0.92	22.62

Table 6.4.3-12

LOCA Containment Response Results For Byron Unit 1 and Braidwood Unit 1 (BWI SG) Loss of Offsite Power Assumed

Case	Peak Press. (psig)	Peak Steam Temp. (°F)	Pressure (psig) @ 24 hours	Steam Temperature (°F) @ 24 hours
DEPS MINSI	41.84 @ 460.56sec	262.30 @ 460.56 sec	9.074@ 24 hrs	174@ 24 hrs
DEPS MAXSI	40.85 @ 22.89 sec	261.56 @ 22.887 sec	5.694@ 24 hrs	147@ 24 hrs
DEHL MINSI	42.77 @ 22.116 sec	264.50 @ 22.116 sec	NA	NA

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Table 6.4.3-13

Double-Ended Pump Suction Break

Byron Unit 1 & Braidwood Unit 1

(BWI SG) Minimum Safeguards

Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
.00100000	1.000	120.00	120.00
.50000000	3.536	141.76	183.24
1.0000000	5.943	161.30	198.84
2.0000000	10.47	191.17	212.98
3.0000000	14.45	210.45	220.59
4.0000000	17.30	219.53	225.11
5.0000000	19.52	223.71	228.47
6.0000000	21.47	225.85	231.32
7.0000000	23.40	227.57	234.02
8.0000000	25.42	232.43	236.32
9.0000000	27.32	236.72	238.54
10.000000	29.02	240.35	240.52
11.000000	30.59	243.53	242.28
12.000000	32.01	246.31	243.82
13.000000	33.34	248.80	245.19
14.000000	34.57	251.06	246.43
15.000000	35.73	253.10	247.55
16.000000	36.79	254.94	248.55
17.000000	37.79	256.63	249.46
18.000000	38.75	258.20	250.28
19.000000	39.67	259.69	250.88
20.000000	40.36	260.79	251.30
21.000000	40.75	261.40	251.65
22.000000	40.93	261.68	251.90
23.000000	40.96	261.73	252.11
24.000000	40.91	261.65	252.27
25.000000	40.79	261.47	252.36
26.000000	40.63	261.20	252.36
27.000000	40.43	260.89	252.35
28.000000	40.21	260.54	252.33
29.000000	40.00	260.21	252.31
30.000000	39.81	259.90	252.30
31.000000	39.63	259.62	252.28
32.000000	39.46	259.35	252.26

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Table 6.4.3-13 (cont.)
Double-Ended Pump Suction Break
Byron Unit 1 & Braidwood Unit 1
(BWI SG) Minimum Safeguards

Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
33.000000	39.31	259.11	252.24
34.000000	39.17	258.88	252.22
35.000000	39.04	258.67	252.17
36.000000	38.93	258.49	251.88
37.000000	38.85	258.35	251.39
38.000000	38.77	258.23	250.91
39.000000	38.71	258.12	250.47
40.000000	38.64	258.01	250.05
41.000000	38.58	257.91	249.65
42.000000	38.52	257.82	249.28
43.000000	38.47	257.73	248.93
44.000000	38.42	257.65	248.58
45.000000	38.38	257.58	248.15
46.000000	38.34	257.51	247.72
47.000000	38.30	257.45	247.31
48.000000	38.27	257.39	246.92
49.000000	38.24	257.34	246.55
50.000000	38.21	257.29	246.19
51.000000	38.18	257.25	245.85
52.000000	38.15	257.20	245.52
53.000000	38.13	257.16	245.21
54.000000	38.11	257.12	244.91
55.000000	38.08	257.09	244.62
56.000000	38.06	257.05	244.35
57.000000	38.05	257.02	244.08
58.000000	38.03	256.99	243.83
59.000000	38.01	256.96	243.59
60.000000	38.00	256.94	243.35
61.000000	37.98	256.91	243.13
62.000000	37.97	256.89	242.91
63.000000	37.96	256.87	242.71
64.000000	37.94	256.85	242.51
65.000000	37.93	256.83	242.32
66.000000	37.91	256.79	242.14
67.000000	37.89	256.76	241.97
68.000000	37.87	256.72	241.80

Table 6.4.3-13 (cont.) Double-Ended Pump Suction Break Byron Unit 1 & Braidwood Unit 1 (BWI SG) Minimum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
69.000000	37.85	256.69	241.64
70.000000	37.83	256.66	241.49
71.000000	37.81	256.62	241.34
72.000000	37.79	256.59	241.20
73.000000	37.78	256.56	241.07
75.000000	37.80	256.55	241.00
76.000000	37.87	256.62	241.01
77.000000	37.94	256.70	241.01
78.000000	38.01	256.78	241.02
79.000000	38.07	256.85	241.03
80.000000	38.13	256.92	241.04
81.000000	38.19	256.98	241.05
82.000000	38.25	257.04	241.06
83.000000	38.31	257.09	241.07
84.000000	38.36	257.15	241.08
85.000000	38.41	257.20	241.09
86.000000	38.46	257.24	241.10
87.000000	38.51	257.29	241.11
88.000000	38.56	257.33	241.12
89.000000	38.60	257.36	241.14
90.000000	38.63	257.38	241.16
91.000000	38.66	257.39	241.19
92.000000	38.69	257.41	241.21
93.000000	38.72	257.42	241.24
95.000000	38.74	257.43	241.30
96.000000	38.74	257.43	241.33
97.000000	38.74	257.43	241.36
98.000000	38.74	257.43	241.39
99.000000	38.74	257.43	241.41
100.00000	38.74	257.43	241.43
101.00000	38.74	257.43	241.46
102.00000	38.74	257.43	241.49
103.00000	38.73	257.42	241.52
104.00000	38.73	257.42	241.54
105.00000	38.73	257.41	241.57
106.00000	38.72	257.40	241.60

Table 6.4.3-13 (cont.) Double-Ended Pump Suction Break Byron Unit 1 & Braidwood Unit 1 (BWI SG) Minimum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
107.00000	38.72	257.39	241.62
108.00000	38.71	257.39	241.65
109.00000	38.71	257.38	241.68
110.00000	38.70	257.37	241.70
111.00000	38.70	257.36	241.73
112.00000	38.69	257.35	241.76
113.00000	38.68	257.34	241.78
114.00000	38.68	257.32	241.81
115.00000	38.67	257.31	241.83
116.00000	38.66	257.30	241.86
117.00000	38.65	257.29	241.88
118.00000	38.65	257.28	241.91
119.00000	38.64	257.26	241.94
120.00000	38.63	257.25	241.96
121.00000	38.62	257.24	241.99
122.00000	38.62	257.22	242.01
123.00000	38.61	257.21	242.04
124.00000	38.60	257.20	242.06
125.00000	38.59	257.18	242.09
126.00000	38.58	257.17	242.11
127.00000	38.58	257.16	242.13
128.00000	38.57	257.14	242.16
129.00000	38.56	257.13	242.18
130.00000	38.55	257.12	242.21
131.00000	38.54	257.10	242.23
132.00000	38.54	257.09	242.26
133.00000	38.53	257.08	242.28
134.00000	38.52	257.06	242.30
135.00000	38.51	257.05	242.33
136.00000	38.51	257.04	242.35
137.00000	38.50	257.03	242.37
138.00000	38.49	257.01	242.40
139.00000	38.48	257.00	242.42
140.00000	38.48	256.99	242.44
141.00000	38.47	256.98	242.47
142.00000	38.46	256.97	242.49

Table 6.4.3-13 (cont.)
Double-Ended Pump Suction Break
Byron Unit 1 & Braidwood Unit 1
(BWI SG) Minimum Safeguards

Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
143.00000	38.46	256.95	242.51
145.00000	38.44	256.93	242.56
146.00000	38.44	256.92	242.58
147.00000	38.43	256.91	242.61
148.00000	38.42	256.90	242.63
149.00000	38.42	256.89	242.65
150.00000	38.41	256.88	242.67
151.00000	38.41	256.87	242.69
152.00000	38.40	256.86	242.72
153.00000	38.40	256.85	242.74
154.00000	38.39	256.84	242.76
155.00000	38.38	256.83	242.78
156.00000	38.38	256.83	242.80
157.00000	38.38	256.82	242.83
158.00000	38.37	256.81	242.85
159.00000	38.37	256.80	242.87
160.00000	38.36	256.80	242.89
161.00000	38.36	256.79	242.91
162.00000	38.35	256.78	242.93
163.00000	38.35	256.77	242.96
164.00000	38.35	256.77	242.98
165.00000	38.34	256.76	243.00
166.00000	38.34	256.76	243.02
167.00000	38.34	256.75	243.04
168.00000	38.33	256.75	243.06
169.00000	38.33	256.74	243.08
170.00000	38.33	256.74	243.10
171.00000	38.33	256.73	243.12
172.00000	38.32	256.73	243.14
173.00000	38.32	256.72	243.16
174.00000	38.32	256.72	243.18
175.00000	38.32	256.72	243.21
176.00000	38.32	256.71	243.23
177.00000	38.31	256.71	243.25
178.00000	38.31	256.71	243.27
179.00000	38.31	256.71	243.29

Table 6.4.3-13 (cont.) Double-Ended Pump Suction Break Byron Unit 1 & Braidwood Unit 1 (BWI SG) Minimum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
180.00000	38.31	256.70	243.31
181.00000	38.31	256.70	243.33
182.00000	38.31	256.70	243.35
183.00000	38.31	256.70	243.37
184.00000	38.31	256.70	243.39
185.00000	38.31	256.70	243.41
186.00000	38.31	256.70	243.43
187.00000	38.31	256.70	243.45
188.00000	38.31	256.70	243.47
189.00000	38.31	256.71	243.49
190.00000	38.31	256.71	243.51
191.00000	38.31	256.71	243.53
192.00000	38.32	256.71	243.55
193.00000	38.32	256.72	243.57
194.00000	38.32	256.72	243.59
195.00000	38.32	256.72	243.61
196.00000	38.33	256.73	243.63
197.00000	38.33	256.73	243.65
198.00000	38.33	256.74	243.67
199.00000	38.34	256.75	243.70
209.00000	38.47	256.96	243.99
219.00000	38.60	257.18	244.30
229.00000	38.73	257.40	244.60
239.00000	38.87	257.61	244.90
249.00000	39.00	257.84	245.19
259.00000	39.14	258.06	245.48
269.00000	39.28	258.28	245.77
279.00000	39.42	258.51	246.04
289.00000	39.56	258.73	246.31
299.00000	39.70	258.96	246.59
309.00000	39.84	259.18	246.85
319.00000	39.98	259.41	247.12
329.00000	40.12	259.63	247.38
339.00000	40.26	259.85	247.64
349.00000	40.40	260.07	247.89
359.00000	40.53	260.28	248.15

Table 6.4.3-13 (cont.) Double-Ended Pump Suction Break Byron Unit 1 & Braidwood Unit 1 (BWI SG) Minimum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
369.00000	40.67	260.50	248.39
379.00000	40.80	260.71	248.64
389.00000	40.94	260.91	248.88
399.00000	41.07	261.12	249.13
409.00000	41.20	261.32	249.36
419.00000	41.33	261.52	249.60
429.00000	41.45	261.72	249.83
439.00000	41.58	261.91	250.06
449.00000	41.70	262.10	250.29
459.00000	41.83	262.29	250.52
469.00000	41.75	262.17	250.79
479.00000	41.64	261.99	251.05
489.00000	41.53	261.81	251.31
499.00000	41.42	261.65	251.56
599.00000	40.57	260.29	253.68
699.00000	39.89	259.17	255.31
799.00000	39.25	258.11	256.58
899.00000	38.65	257.10	257.57
999.00000	38.05	256.08	258.37
1099.0000	37.46	255.06	259.02
1199.0000	38.37	256.57	258.87
1299.0000	39.36	258.20	258.70
1399.0000	40.30	259.71	258.60
1499.0000	41.21	261.14	258.56
1599.0000	39.84	258.94	258.57
1699.0000	38.66	257.00	258.51
1799.0000	37.56	255.15	258.40
1899.0000	36.53	253.37	258.24
1999.0000	35.55	251.63	258.05
2099.0000	34.61	249.92	257.82
2199.0000	33.71	248.26	257.56
2299.0000	32.85	246.62	257.28
2399.0000	32.02	245.00	256.96
2499.0000	31.22	243.42	256.63
2599.0000	30.45	241.86	256.28
2699.0000	29.71	240.32	255.90

Table 6.4.3-13 (cont.) Double-Ended Pump Suction Break Byron Unit 1 & Braidwood Unit 1 (BWI SG) Minimum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
2799.0000	29.00	238.80	255.51
2899.0000	28.31	237.31	255.11
2999.0000	27.64	235.84	254.69
3099.0000	27.00	234.39	254.26
3199.0000	26.38	232.96	253.82
3299.0000	25.79	231.56	253.37
3399.0000	25.22	230.19	252.91
3499.0000	24.67	228.84	252.45
3599.0000	24.14	227.52	251.97
3699.0000	23.37	225.55	251.23
3799.0000	22.56	223.45	250.40
3899.0000	22.79	224.03	249.63
3999.0000	22.92	224.38	248.89
4999.0000	23.10	224.85	242.31
5999.0000	22.66	223.70	236.66
6999.0000	21.94	221.77	231.67
7999.0000	21.03	219.26	227.27
8999.0000	20.05	216.46	223.24
9999.0000	19.06	213.48	219.47
19999.000	14.41	197.39	194.06
29999.000	12.89	191.16	184.25
39999.000	12.04	187.37	179.71
49999.000	11.35	184.14	176.60
59999.000	10.73	181.05	173.83
69999.000	10.10	177.77	171.06
79999.000	9.508	174.52	168.23
89999.000	8.899	171.00	165.34
99999.000	8.338	167.57	162.41
199999.00	7.195	159.85	153.02
299999.00	6.772	156.66	150.40
399999.00	6.387	153.61	147.91
499999.00	6.020	150.55	145.44
599999.00	5.675	147.52	142.98
699999.00	5.347	144.47	140.53
799999.00	5.030	141.41	138.08
899999.00	4.735	138.49	135.64

Table 6.4.3-13 (cont.) Double-Ended Pump Suction Break Byron Unit 1 & Braidwood Unit 1 (BWI SG) Minimum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
999999.00	4.424	135.25	133.21
1099999.0	4.279	133.68	130.57
1199999.0	4.242	133.26	130.40
1299999.0	4.228	133.07	130.27
1399999.0	4.221	132.97	130.14
1499999.0	4.203	132.79	130.02
1599999.0	4.190	132.62	129.89
1699999.0	4.179	132.47	129.77
1799999.0	4.171	132.36	129.64
1899999.0	4.155	132.16	129.51
1999999.0	4.145	132.01	129.39
2099999.0	4.132	131.87	129.26
2199999.0	4.118	131.69	129.14
2299999.0	4.098	131.46	129.03
2399999.0	4.093	131.37	128.89
2499999.0	4.076	131.17	128.77
2592000.0	4.068	131.05	128.65

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Table 6.4.3-14 Double-Ended Pump Suction Break Byron Unit 1 & Braidwood Unit 1 (BWI SG) Maximum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
.00100000	1.000	120.00	120.00
.50000000	3.536	141.76	183.23
1.0000000	5.942	161.28	198.84
2.0000000	10.47	191.07	212.97
3.0000000	14.44	210.26	220.58
4.0000000	17.28	219.23	225.10
5.0000000	19.49	223.31	228.46
6.0000000	21.44	225.37	231.32
7.0000000	23.35	227.43	233.97
8.0000000	25.36	232.28	236.29
9.0000000	27.24	236.54	238.51
10.000000	28.94	240.16	240.49
11.000000	30.49	243.34	242.28
12.000000	31.91	246.12	243.84
13.000000	33.23	248.61	245.23
14.000000	34.46	250.86	246.48
15.000000	35.61	252.90	247.62
16.000000	36.67	254.74	248.63
17.000000	37.66	256.40	249.55
18.000000	38.60	257.96	250.37
19.000000	39.52	259.45	251.00
20.000000	40.20	260.53	251.42
21.000000	40.58	261.14	251.78
22.000000	40.76	261.41	252.03
23.000000	40.80	261.47	252.29
24.000000	40.75	261.40	252.45
25.000000	40.63	261.21	252.51
26.000000	40.46	260.95	252.48
27.000000	40.26	260.63	252.46
28.000000	40.04	260.28	252.45
29.000000	39.82	259.93	252.43
30.000000	39.61	259.59	252.41
31.000000	39.41	259.28	252.39
32.000000	39.23	258.98	252.38

Table 6.4.3-14 (cont.) Double-Ended Pump Suction Break Byron Unit 1 & Braidwood Unit 1 (BWI SG) Maximum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
33.000000	39.06	258.70	252.29
34.000000	38.90	258.44	252.06
35.000000	38.79	258.25	251.36
36.000000	38.70	258.12	250.46
37.000000	38.63	257.99	249.61
38.000000	38.55	257.87	248.80
39.000000	38.48	257.76	248.02
40.000000	38.42	257.65	247.29
41.000000	38.36	257.55	246.58
42.000000	38.30	257.45	245.90
43.000000	38.25	257.36	245.25
44.000000	38.20	257.27	244.63
45.000000	38.15	257.19	244.04
46.000000	38.10	257.11	243.47
47.000000	38.06	257.04	242.92
48.000000	38.01	256.97	242.39
49.000000	37.97	256.90	241.88
50.000000	37.94	256.84	241.40
51.000000	37.90	256.78	240.93
52.000000	37.87	256.72	240.47
53.000000	37.83	256.66	240.03
54.000000	37.80	256.61	239.61
55.000000	37.77	256.56	239.20
56.000000	37.74	256.51	238.81
57.000000	37.72	256.47	238.43
58.000000	37.69	256.42	238.06
59.000000	37.66	256.38	237.70
60.000000	37.64	256.34	237.36
61.000000	37.62	256.30	237.03
62.000000	37.58	256.24	236.72
63.000000	37.55	256.18	236.43
64.000000	37.52	256.13	236.14
65.000000	37.49	256.07	235.87
66.000000	37.46	256.02	235.61
67.000000	37.43	255.97	235.35
68.000000	37.40	255.92	235.10

Table 6.4.3-14 (cont.) Double-Ended Pump Suction Break Byron Unit 1 & Braidwood Unit 1 (BWI SG) Maximum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
69.000000	37.37	255.87	234.86
70.000000	37.34	255.82	234.63
71.000000	37.31	255.78	234.40
72.000000	37.29	255.73	234.18
73.000000	37.26	255.68	233.97
74.000000	37.24	255.64	233.76
75.000000	37.24	255.60	233.56
76.000000	37.24	255.56	233.36
77.000000	37.24	255.52	233.17
78.000000	37.24	255.48	232.99
79.000000	37.23	255.43	232.89
80.000000	37.21	255.35	232.93
81.000000	37.18	255.27	232.97
82.000000	37.16	255.20	233.01
83.000000	37.14	255.12	233.05
84.000000	37.12	255.05	233.09
85.000000	37.09	254.97	233.13
86.000000	37.07	254.90	233.16
87.000000	37.05	254.83	233.20
88.000000	37.03	254.76	233.23
89.000000	37.02	254.69	233.27
90.000000	37.00	254.62	233.30
91.000000	36.98	254.55	233.33
92.000000	36.96	254.48	233.36
93.000000	36.95	254.42	233.39
94.000000	36.92	254.35	233.42
95.000000	36.88	254.28	233.45
96.000000	36.84	254.21	233.48
97.000000	36.80	254.14	233.50
98.000000	36.76	254.08	233.53
99.000000	36.72	254.01	233.56
100.00000	36.68	253.94	233.58
101.00000	36.65	253.88	233.60
102.00000	36.61	253.81	233.63
103.00000	36.57	253.75	233.65
104.00000	36.54	253.69	233.67

Table 6.4.3-14 (cont.) Double-Ended Pump Suction Break Byron Unit 1 & Braidwood Unit 1 (BWI SG) Maximum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
105.00000	36.50	253.62	233.69
106.00000	36.46	253.56	233.71
107.00000	36.43	253.50	233.73
108.00000	36.39	253.44	233.75
109.00000	36.36	253.38	233.77
110.00000	36.32	253.32	233.79
111.00000	36.29	253.26	233.81
112.00000	36.26	253.20	233.82
113.00000	36.22	253.14	233.84
114.00000	36.19	253.08	233.85
115.00000	36.16	253.02	233.87
116.00000	36.12	252.96	233.88
117.00000	36.09	252.90	233.90
118.00000	36.06	252.85	233.91
119.00000	36.03	252.79	233.92
120.00000	35.99	252.73	233.93
121.00000	35.96	252.68	233.94
122.00000	35.93	252.62	233.95
123.00000	35.90	252.57	233.96
124.00000	35.87	252.51	233.97
125.00000	35.84	252.46	233.98
126.00000	35.81	252.40	233.99
127.00000	35.78	252.35	234.00
128.00000	35.75	252.30	234.00
129.00000	35.72	252.24	234.01
130.00000	35.69	252.19	234.01
131.00000	35.66	252.14	234.02
132.00000	35.63	252.08	234.02
133.00000	35.60	252.03	234.03
134.00000	35.57	251.98	234.03
135.00000	35.54	251.93	234.04
136.00000	35.51	251.88	234.04
137.00000	35.49	251.83	234.04
138.00000	35.46	251.78	234.04
139.00000	35.43	251.73	234.04
140.00000	35.40	251.68	234.04