TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

400 Chestnut Street Tower II

May 2, 1984

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Chief
Licensing Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Ms. Adensam:

In the Matter of the Application of	)	Docket Nos. 50-390
Tennessee Valley Authority	)	

Westinghouse (W) has developed and applied improved analytical techniques in the reanalysis of the Watts Bar Nuclear Plant (WBN) steamline break transients which allow for the Boron Injection Tank (BIT) to be bypassed, eliminated, or reduced in concentration and the heat tracing system deleted. The enclosed W generated study, "Report for the BIT Concentration Reduction/BIT Elimination Study for Watts Bar Units 1 and 2," (Enclosure 1) provides a detailed description of the criteria and methodology used in the reanalysis and presents the analytical results.

In consideration of the current unit 1 fuel load schedule (June 1984) and the extensive design/hardware modifications required for a complete elimination of the BIT and associated subsystems, TVA, for WBN unit 1, has elected to implement the alternative to reduce the boron concentration in the BIT to below four percent by weight which will in turn eliminate the need for heat tracing. Consequently, the corresponding technical specifications can be deleted. In order to ensure that this issue is resolved in a timeframe consistent with the "proof and review" phase of the unit 1 Technical Specifications, we are forwarding approved FSAR chapter 6, "Engineered Safety Features," and chapter 15, "Accident Analysis," changes (Enclosure 2). These revisions will be reflected in future amendments to the WBN FSAR. Also enclosed are proposed unit 1 Technical Specifications changes (Enclosure 3).

Several FSAR chapter 6 changes resulting from the BIT concentration reduction issue will not be submitted until the resolution of the containment temperature concerns (steam generator tube uncovery/superheat issue), which will ultimately require chapter 6 revisions. It is TVA's opinion however, that the BIT concentration reduction issue is independent of the superheat issue and therefore resolution of the former should proceed at the earliest date possible. (Please note that resolution of the superheat issue will include any contributing effects from the BIT concentration reduction modifications.)

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Director of Nuclear Reactor Regulation May 2, 1984

Additional evaluations will be performed for unit 2 to determine the most beneficial course of action.

If you have any questions concerning this matter, please get in touch with D. B. Ellis at FTS 858-2681.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills

L. M. Mills, Manager
Nuclear Licensing

Sworn to and subscribed before me
this 2nd day of May 1984
Paulette H. White
Notary Public
My Commission Expires 9-5-84

Enclosures (3)
cc: U.S. Nuclear Regulatory Commission (Enclosures)
Region II
Attn: Mr. James P. O'Reilly Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30303

ENCLOSURE 1

WATTS BAR NUCLEAR PLANT

WESTINGHOUSE GENERATED REPORT FOR THE BIT CONCENTRATION REDUCTION/BIT ELIMINATION STUDY

The upper head injection accumulators are located outside containment. Protection is also provided against the release of accumulator gas by accumulator pressure indicators and alarms. These enable the operator to take action promptly as required to maintain plant operation within the requirements of the technical specifications covering accumulator operability.

A complete listing of design parameters for the upper head injection accumulators is presented in Table 6.3-1.

Upper Head Injection Surge Tank

The design parameters for the upper head injection surge tank are given in Table 6.3-1. The tank is designed to accomodate changes in the volume of the injection water stored in the accumulator due to normal temperature changes and to allow for some leakage into the UHI accumulator during normal plant operation. Two independent channels provide level indication in the control room and actuate a common alarm should the level approach the limit of the operating band.

PUMPS

Residual Heat Removal Pumps

Residual heat removal pumps are provided to deliver water from the refueling water storage tank or the containment sump to the Reactor Coolant System should the Reactor Coolant System pressure fall below their shutoff head. Each residual heat removal pump is a single stage, vertical position, centrifugal pump. It has an integral motor-pump shaft, driven by an induction motor. The unit has a self contained mechanical seal cooling system. Component cooling water is the heat exchange medium. The pumps are interlocked to start on receipt of the safety injection signal.

A minimum flow bypass line is provided for the pumps to recirculate through the residual heat exchangers and return the cooled fluid to the pump suction should these pumps be started with their normal flow paths blocked. Once flow is established to the Reactor Coolant System, the bypass line is automatically closed. This line prevents deadheading the pumps and permits pump testing during normal operation.

The residual heat removal pumps are also discussed in Section 5.5.7.

Centrifugal Charging Pumps

These pumps deliver water from the refueling water storage tank through the boron injection tank to the Reactor Coolant System at the prevailing Reactor Coolant System pressure. Each centrifugal charging pump is a multistage, diffuser design, barrel type casing with vertical suction and discharge nozzles. The pump is driven through a speed increaser connected to an induction motor. The unit has a self contained lubrication system, and mechanical seal cooling system. Component cooling water is the normal heat exchange medium for the mechanical seal. The pumps are interlocked to start on receipt of the safety injection signal.

A minimum flow bypass line is provided on each pump discharge to recirculate flow to the volume control tank after cooling in the seal water heat exchanger, if required, to protect the pumps at the shutoff head. The charging pumps may be tested during normal operation through the use of the minimum flow bypass line. The centrifugal charging pumps are also discussed in Section 9.3.4.

Safety Injection Pumps

The safety injection pumps deliver water from the refueling water storage tank after the Reactor Coolant System pressure is reduced below their shutoff head. Each high head safety injection pump is a multistage, centrifugal pump. The pump is driven directly by an induction motor. The unit has a self

A separate motor driven gag is provided on each isolation valve to manually lock the valve in the closed position during plant operations at low pressure. The gag is manually removed and the valve then moved to the open position following plant startup.

Butterfly Valves

Each main residual heat removal line has an air-operated butterfly valve which is normally open and is designed to fail in the open position. These valves are left in the full open position during normal operation to maximize flow from this system to the Reactor Coolant System during the injection mode of the Emergency Core Cooling System operation.

Piping

All piping joints are welded except for the pump and butterfly valve flanged connections.

Weld connections for pipes sized 2-1/2 inches and larger are butt welded. Reducing tees are used where the branch size exceeds one-half of the header size. Branch connections of sizes that are equal to or less than one-half of the header size conform to the ANSI code. Branch connections 1/2 inch through 2 inches are attached to the header by means of full penetration welds, using pre-engineered integrally reinforced branch connections.

Minimum piping and fitting wall thicknesses as determined by ANSI B31.1.0-1967 Ed. formula are increased to account for the manufacturer's permissible tolerance of minus 12-1/2 percent on the nominal wall and an appropriate allowance for wall thinning on the external radius during any pipe bending operations in the shop fabrication of the subassemblies.

System Operation

The operation of the Emergency Core Cooling System, following a loss-of-coolant accident, can be divided into two distinct modes:

1. The injection mode in which any reactivity increase following the postulated accidents is terminated, initial cooling of the core is accomplished, and coolant lost from the primary system in the case of a loss-of-coolant accident is replenished, and

2. The recirculation mode in which long term core cooling is provided during the accident recovery period.

A discussion of these modes follows.

Break Spectrum Coverage

The principal mechanical components of the Emergency Core Cooling System which provide core cooling immediately following a loss-of-coolant accident are the accumulators, the safety injection pumps, the centrifugal charging pumps, the residual heat removal pumps, refueling water storage tank, and the associated valves, and piping.

For large pipe ruptures, the Reactor Coolant System would be depressurized and voided of coolant rapidly, and a high flow rate of emergency coolant is required to quickly cover the exposed fuel rods and limit possible core damage. This high flow is initially provided by the upper head accumulators discharging into the reactor vessel head followed by the passive cold leg accumulators, the charging pumps, safety injection pumps, and the residual heat removal pumps discharging into the cold legs of the Reactor Coolant System. The residual heat removal and safety injection pumps deliver into the accumulator injection lines, between the two check valves, during the injection mode. The charging pumps deliver coolant to the cold legs during the injection mode.

Emergency cooling is provided for small ruptures primarily by the high need[1] injection pumps and upper head accumulators. Small

[1]The charging pumps and safety injection pumps are commonly referred to as 'high head pumps' and the residual heat removal pumps as 'low head pumps.' Likewise, the term 'high head injection' is used to denote charging pump and safety injection pump injection and 'low head injection' refrs to residual heat removal pum injection.

ruptures are those, with an equivalent diameter of 6 inches or less, which do not immediately depressurize the Reactor Coolant System below the accumulator discharge pressure. The centrifugal charging pumps are designed to deliver borated water at the prevailing Reactor Coolant System. During the injection mode, the charging pumps will take suction from the refueling water storage tank.

The safety injection pumps also take suction from the refueling water storage tank and deliver borated water to the cold legs of the Reactor Coolant System. The safety injection pumps begin to deliver water to the Reactor Coolant System after the pressure has fallen below the pump shutoff head.

Core protection is afforded with the minimum engineered safety feature equipment. The minimum engineered safety feature equipment is defined by consideration of the single failure criteria as discussed in Sections 6.3.1.4 and 3.1. The minimum design case will ensure the entire break spectrum is accounted for and core cooling design bases of Section 6.3.1 are met. The analyses for this case are presented in Sections 15.3 and 15.4.

For large Reactor Coolant System ruptures, the accumulators and the active high head and low head pumping components serve to complete the core refill. One residual heat removal pump is required for long term recirculation.

If the break is small (6 inch equivalent diameter or less) the accumulators with one charging pump and one safety injection pump ensure adequate cooling during the injection mode. Long term recirculation requires one residual heat removal pump and components of the auxiliary heat removal systems which are required to transfer heat from the Emergency Core Cooling System (e.g., Component Cooling System and Essential Raw Cooling Water System). The loss of coolant analyses are presented in Section 15.3 and 15.4.

Certain modifications (i.e., reduced component availability) to the normal operating status as given in Table 6.3-4 of the Emergency Core Cooling System are permissible without impairing the ability of the Emergency Core Cooling System to provide adequate core cooling capability. Accordingly, Technical Specifications have been established (see Chapter 16) to cover these modifications.

The Technical Specifications permit one cold leg accumulator to be isolated for check valve leakage testing. They also permit various pumps of the Emergency Core Cooling System to be inoperable during power operation and for an additional time period while the reactor is at hot shutdown, provided that the

1. Starts the diesel generators and, if all other sources of power are lost, aligns them to the 6.9-kV shutdown boards.

2. Starts the charging pumps, the safety injection pumps, and the residual heat removal pumps.

3. Aligns the charging pumps for injection by:

   a. Closing the valves in the charging pump discharge line to the normal charging line.

   b. Opening the valves in the charging pumps suction line from the refueling water storage tank.

   c. Closing the valves in the charging pump normal suction line from the volume control tank.

   d. Opening the boron injection tank inlet and discharge line isolation valves.

Remotely operated valves for the injection mode which are under manual control (i.e., valves which normally are in their ready position and do not require a safety injection signal) have their positions indicated on a common portion of the control board. If a component is out of its proper position, its monitor light will indicate this on the control panel. At any time during operation when one of these valve is not in the ready position for injection, this condition is shown visually on the board, and an audible alarm is sounded in the control room.

The injection mode continues until the low level is reached in the refueling water storage tank and the recirculation mode is initiated.

<u>Recirculation Mode</u>

The injection mode continues until the RHR pumps have been realigned to the recirculation mode. During the injection mode all pumps take suction from the refueling water storage tank (RWST) until a low level signal from the RWST in conjunction with the 's' signal and a high sump level signal aligns the residual heat removal pumps to take suction from the containment sump. The RHR block valves (FCV-74-3 and 21) are automatically closed coincident with the opening of the sump isolation valves (FCV-63-72 and 73). The automatic positioning of these valves is initiated only in the event that actuation signals are generated by the safeguards protection logic ('s' signal), two of four RWST low level protection logic signals, and two out of four sump high level signals'. It has been determined that the RHR pumps

6.3.3.3 Alternate Analysis Methods

Small Pipe Break

The small pipe break analysis is used to evaluate the initial core thermal transient for a spectrum of pipe rupture from 3/8 inch up to and including the rupture of a six inch diameter pipe. For breaks 3/8 inch or smaller, the charging system can maintain the pressurizer level at the Reactor Coolant System operating pressure and the Emergency Core Cooling System would not be actuated.

The results of the small pipe break analysis indicate that the limits on core behavior are adequately met, as shown in Section 15.3.

Main Steam System Single Active Failure

Analyses of reactor behavior following any single active failure in the main steam system which results in an uncontrolled release of steam are included in Section 15.2. The analyses assume that a single valve (largest of the safety, relief, or bypass valves) opens and fails to close, which results in an uncontrolled cooldown of the Reactor Coolant System.

Results indicate that if the incident is initiated at the hot shutdown condition, which results in the worst reactivity transient, there is no return to criticality. Thus, the Emergency Core Cooling System provides adequate protection for this incident.

Steam Line Rupture

Following a steam line rupture the Emergency Core Cooling System is automatically actuated to deliver borated water from the boron injection tank to the Reactor Coolant System. The response of the Emergency Core Cooling System following a steam line break is identical to its response during the injection mode of operation following a loss-of-coolant accident.

This accident is discussed in detail in Section 15.4. The limiting steam line rupture is a complete line severance.

In the case of a steam line rupture when offsite power is not assumed lost, credit is taken for the uninterrupted availability of power for the Emergency Core Coolant System components.

The results of the analysis in Section 15.4 indicate that the design basis criteria are met. Thus, the Emergency Core Cooling System adequately fulfills its shutdown reactivity addition function.

The safety injection actuation signal initiates identical actions as described for the injection mode of the loss-of- coolant accident, even though not all of these actions are required following a steam line rupture, e.g., the residual heat removal pumps are not required since the Reactor Coolant System pressure will remain above their shutoff head.

The delivery of the concentrated boric acid from the charging pump results in a negative reactivity change to counteract the increase in reactivity caused by the system cooldown. The charging pumps continue to deliver borated water from the refueling water storage tank, until enough water has been added to the Reactor Coolant System to make up for the shrinkage due to cooldown. The safety injection pumps also deliver borated water from the refueling water storage tank for the interval when the Reactor Coolant System pressure is less than the shutoff head of the safety injection pumps. After pressurizer water level has been restored, the operator will verify that the criteria for 'Safety Injection Termination' as defined in the Emergency Operating Instructions are satisfied before manually terminating injection flow.

The sequence of events following a postulated steam line break is described in Section 15.4.

6.3.3.4 Fuel Rod Perforations

Discussions of peak clad temperature and metal-water reactions appear in Sections 15.3.1 and 15.4.1. Analyses of the radiological consequences of RCS pipe ruptures also are presented in Section 15.4.1.

6.3.3.5 Evaluation Model

Does not apply to this plant (BWRs only).

6.3.3.6 Fuel Clad Effects

Does not apply to this plant (BWRs only).

6.3.3.7 ECCS Performance

Does not apply to this plant (BWRs only).

6.3.3.8 Peak Factors

Does not apply to this plant (BWRs only).

6.3.3.9 Fuel Rod Perforations

Does not apply to this plant (BWRs only).

6.3.3.10 Conformance with Interim Acceptance Criteria

Does not apply to this plant (BWRs only).

6.3.3.11 Effects of ECCS Operation on the Core

The effects of the ECCS on the reactor core are discussed in Sections 15.3 and 15.4.

6.3.3.12 Use of Dual Function Components

The Emergency Core Cooling System contains components which have no other operating function as well as components which are shared with other systems and perform normal operating functions. Components in each category are as follows:

1. Components of the Emergency Core Cooling System which perform no other functions are:

   a. Two upper head injection accumulators which discharge borated water into the upper head of the reactor vessel.

   b. One accumulator for each loop which discharges borated water into its respective cold leg of the reactor coolant loop piping.

   c. Two safety injection pumps which supply borated water for core cooling to the Reactor Coolant System.

   d. Associated piping, valves and instrumentation.

2. Components which also have a normal operating function are as follows:

   a. The residual heat removal pumps and the residual heat exchangers: These components are normally used during the latter stages of normal reactor cooldown and when the reactor is held at cold shutdown for core decay heat removal. However, during all other plant operating

6.3.5 Instrumentation Application

Instrumentation and associated analog and logic channels employed for initiation of Emergency Core Cooling System operation is discussed in Section 7.3. This section describes the instrumentation employed for monitoring Emergency Core Cooling System components during normal plant operation and also Emergency Core Cooling System post accident operation. All alarms are annunciated in the control room.

6.3.5.1 Temperature Indication

Residual Heat Exchanger Inlet Temperature

The fluid temperature at the inlet and outlet of each residual heat exchanger is recorded in the control room.

Refueling Water Storage Tank (RWST) Temperature

Two temperature channels are provided to monitor the RWST temperature. Both are indicated in the Main Control Room.

6.3.5.2 Pressure Indication

Boron Injection Tank Pressure

Boron injection tank pressure is indicated in the control room. A high pressure alarm is provided.

Safety Injection Header Pressure

Safety injection pump discharge header pressure is indicated in the control room.

Cold Leg Accumulator Pressure

Duplicate pressure channels are installed on each cold leg accumulator. Pressure indication in the control room and high and low pressure alarms are provided by each channel.

UHI Isolation Valve Hydraulic Accumulator Pressure

The hydraulic pressure of each of the four hydraulic accumulators is monitored independently with pressure readout and a low pressure alarm light on the hydraulic service panel in the control room and annunciation on the main control board.

Test Line Pressure

A local pressure indicator used to check for proper seating of the accumulator check valves between the injection lines and the Reactor Coolant System is installed on the leakage test line.

Residual Heat Removal Pump Discharge Pressure

Residual heat removal discharge pressure for each pump is indicated in the control room. A high pressure alarm is actuated by each channel.

Upper Head Injection Accumulator Pressure

Duplicate pressure channels are installed on the upper head injection surge tank. Pressure indication, for this accumulator system, is provided in the control room, together with high and low pressure alarms, by each channel.

Alarms are also provided that the operator will manually close and gag the hydraulic isolation valve during plant shutdown at pressure below 1900 psi and manually release the gag and open the valve at pressures in excess of 1900 psi during plant startup.

6.3.5.3 Flow Indication

Charging Pump Injection Flow

Injection flow to the reactor cold legs injector is indicated in the control room.

Residual Heat Removal Pump Injection Flow

Flow through each residual heat removal injection and recirculation header leading to the reactor cold or hot legs is indicated in the control room.

TABLE 6.3-1

Component	Parameters	
Safety Injection Pumps (Con'td)	Maximum Starting Time, Sec.	5
Residual Heat Removal Pumps	Refer to Subsection 5.5.7 for Parameter information	
Residual Heat Exchangers	Refer to Subsection 5.5.7 for parameter information	
Boron Injection Tank	Number	1
	Total Volume, gal	900
	Useable Volume At Operating Conditions (solution), gal.	900
	Design Pressure, psig	2735
	Operating Pressure, psig	70
	Design Temperature, °F	300
Heaters	Number, pr channel	6
	Total	12
	Capacity, each kW	1
	Type	Strip
Upper Head Injection Surge Tank	Number	1
	Design Pressure, psig	1800
	Design Temperature, °F	300
	Operation Temperature, °F	70-100
	Normal Pressure, psia	1050
	Useable Volume, ft^3	35
	Total Volume, ft^3	55
Upper Head Injection Surge Tank (Con't)	Boron Concentration, ppm, min.	1900
Refueling Water Storage Tanks	Number	1

TABLE 6.3-5

EMERGENCY CORE COOLING SYSTEM SHARED FUNCTIONS EVALUATION

Component	Normal Operating Arrangement	Accident Arrangement
Refuling Water Storage Tank	Lined up to suction of safety injection, containment spray, and residual heat removal pumps,	Lined up to suction of centrifugal charging, safety injection, residual heat removal pumps, and containment spray pumps. Valves for realignment meet single failure criteria.
Centrifugal Charging Pumps	Lined up for charging service Suction from volume control tanks	Lined up to inlet of boron injection tank and outlet of RWST. Valves for realignment meet single failure criteria.
Residual Heat Removal Pumps	Lined up to cold legs of reactor coolant piping.	Lined up to cold legs of reactor coolant piping.
Residual Heat Exchangers	Lined up for residual heat removal pump operation.	Lined up for residual heat removal pump operations.
Safety Injection Pumps	Lined up to cold legs of reactor coolant piping	Lined up to cold legs of reactor coolant piping.
Accumulators	Lined up to cold legs of reactor coolant piping.	Lined up to cold legs of reactor coolant piping.
UHI Accumulators	Lined up to upper head of reactor vessel.	Lined up to upper head of reactor vessel.

TABLE 15.1-2 (Continued)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED MODERATOR TEMPERATURE (Δk/°F)	MODERATOR DENSITY (Δk/gm/cc)	DOPPLER	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED* (MWt)
CONDITION II (Cont'd)					
Excessive Heat Removal Due to Feedwater System Malfunctions (2)	MARVEL	---	0.43	lower (1)	0 and 3425
Excessive Load Increase Incident	LOFTRAN	---	Figure 15.1-7 and 0.43	lower (1)	3425
Accidental Depressurization of the Reactor Coolant System	LOFTRAN	---	Figure 15.1-7	upper (1)	3425
Accidental Depressurization of the Main Steam System	LOFTRAN	---	Function of Moderator Density See Section 15.2.13 (Figure 15.2-40)	(3)	0 (subcritical)
Inadvertent Operation of ECCS During Power Operation	LOFTRAN	---	Minimum (2)	lower (1)	3425
CONDITION III					
Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes which Actuates Emergency Core Cooling	WFLASH, LOCTA-IV				3579

TABLE 15.1-2 (Continued)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENT ASSUMED MODERATOR TEMPERATURE ($\Delta k/°F$)	MODERATOR DENSITY ($\Delta k/gm/cc$)	DOPPLER	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED* (MWt)
CONDITION IV (Cont'd)					
Major secondary system pipe rupture up to and including double ended rupture (Rupture of a Steam Pipe)	LOFTRAN, THINC	Function of Moderator Density See Section 15.2.13 (Figure 15.2-40)		(3)	0 (Subcritical)
Steam Generator Tube Rupture	NA	NA	NA	NA	3579
Single Reactor Coolant Pump Locked Rotor	PHOENIX, LOFTRAN THINC, FACTRAN	---	Figure 15.1-7	lower (1)	2397 and 3425
Fuel Handling Accident	NA	NA	NA		3579
Rupture of a Control Rod Mechanism Housing (RCCA Ejection)	TWINKLE, FACTRAN LEO°ARD	Refer to Section 15.4.6	---	Consistent with lower limit shown on Figure 15.1-5	0 and 3425

Notes:

(1) Reference Figure 15.1-5
(2) Reference Figure 15.1-7
(3) Reference Figure 15.4-9

* A minimum of 2 percent margin has to be applied

RCS pressure will stabilize following operator action to terminate flow to the inadvertantly opened valve; normal operating procedures may then be followed. The operating procedures would call for operator action to control RCS boron concentration and pressurizer level using the CVCS and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to stabilize the plant will be in a time frame in excess of ten minutes following reactor trip.

15.2.12.3 Conclusions

The pressurizer low pressure and the overtemperature ΔT Reactor Protection System signals provide adequate protection against this accident, and the minimum DNBR remains in excess of 1.30.

15.2.13 ACCIDENTAL DEPRESSURIZATION OF THE MAIN STEAM SYSTEM

15.2.13.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the Main Steam System are associated with an inadvertent opening of a single steam dump, relief of safety valve. The analyses performed assuming a rupture of a main steam line are given in Section 15.4.2.1.

The steam release as a consequence of this accident results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the Reactor Coolant System causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin.

The analysis is performed to demonstrate that the following criterion is satisfied: Assuming a stuck rod cluster control assembly, with or without offsite power, and assuming a single failure in the Engineered Safety Features there will be no consequent fuel damage after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, relief or safety valve. This criterion is satisfied by verifying the DNB design basis is met.

The following systems provide the necessary protection against an accidental depressurization of the main steam system.

1. Safety Injection System actuation from any of the following:

    a. Two out of three signals of low-low pressurizer pressure

    b. Two out of three high containment pressure signals.

c. Two out of four loops high steam line flow coincident with either two out of four loops low steam line pressure or two out of four loops low-low T_{ave}.

2. The overpower reactor trips (neutron flux and T and the reactor trip occuring in conjunction with receipt of the safety injection signal.

3. Redundant isolation of the main feedwater lines: Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves following reactor trip, a safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

4. Trip of the fast-acting steam line stop valves (Main Stream Isolation Valves) (designed to close in less than 5 seconds) on:

   a. High-high containment pressure.

   b. Two out of four loops high steam line flow conincident with either two out of four loops low steam line pressure or two out of four loops low-low T_{ave}.

c. Two out of three high negative steam pressure rate in any loop (below Permissive P-11).

15.2.13.2 Analysis of Effects and Consequences

Method of Analysis

The following analyses of a secondary system steam release are performed for this section.

1. A full plant digital computer simulation to determine Reactor Coolant System transient conditions during cooldown, and the effect of safety injection [5].

2. Analyses to determine that there is no consequential fuel damage.

The following conditions are assumed to exist at the time of a secondary steam system release.

1. End-of-life shutdown margin at no load, equilibrium xenon conditions, and with the most reactive rod cluster control assembly stuck in its fully withdrawn position. Operation of rod cluster control assembly banks during core burnup is restricted in such a way that addition of positive reactivity in a secondary system steam release accident will not lead to a more adverse condition than the case analyzed.

2. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive rod cluster control assembly in the fully withdrawn position. The variation of the coefficient with temperature and pressure is included. The keff versus temperature at 1000 psi corresponding to the negative moderator temperature coefficient used is shown in Figure 15.2-40.

3. Minimum capability for injection of high concentration boric acid solution corresponding to the most restrictive single failure in the Safety Injection System. This corresponds to the flow delivered by one charging pump delivering its full contents to the cold leg header. The injection curve used is shown in Figure 15.4-10. Low concentration boric acid must be swept from the safety injection lines downstream of the RWST prior to the delivery of high concentration boric acid (1950 ppm) to the reactor coolant loops. This effect has been allowed for in the analysis.

4. The case studies is a steam flow of 247 pounds per second at 1100 psia from one steam generator with offsite power available. This is the maximum capacity of any single steam dump, relief or safety valve. Initial hot shutdown conditions at time zero are assumed since this represents the most conservative initial condition.

   Should the reactor be just critical or operating at power at power at the time of a steam release, the reactor will be tripped by the normal overpower protection when power level reaches a trip point. Following a trip at power, the Reactor Coolant System contains more stored energy than at no load, the average coolant temperature is higher than at no load and there is appreciable energy stored in the fuel.

   Thus, the additional stored energy is removed via the cooldown caused by the steam release before the no load conditions of Reactor Coolant System temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes no load condition at time zero. However, since the initial steam generator water inventory is greatest at no load, the magnitude and duration of the Reactor Coolant System cooldown are greater for steam line release occuring from no load conditions.

5. In computing the steam flow, the Moody Curve for fl/D = 0 is used.

6. Perfect moisture separation in the steam generator is assumed.

<u>Results</u>

The results presented are a conservative indication of the events which would occur assuming a secondary system steam release since it is postulated that all of the conditions described above occur simultaneously.

Figure 15.2-41 shows the transients arising as the result of a steam flow of 247 lbs/second total at 1100 psia with steam release from four steam generators. The assumed steam release is typical of the capacity of any single steam dump relief or safety valve. In this case safety injection is initiated automatically by low pressurizer pressure. Operation of one centrifugal charging pump is considered. Boron solution at 1950 ppm enters the Reactor Coolant System providing sufficient negative reactivity to assure no fuel damage.

The cooldown for the case shown in Figure 15.2-41 is more rapid than the case of steam release from all steam generators through one steam dump, relief, or safety valve. The transient is conservative with respect to cooldown, since no credit is taken for the energy stored in the system metal other than that of the fuel elements.

Following blowdown of the faulted steam generator, the plant can be brought to a stabilized hot standby condition through control of auxiliary feedwater flow and safety injection flow, as described by plant operating procedures. The operating procedures would call for operator action to limit RCS pressure and pressurizer level by terminating safety injection flow, and to control steam generator level and RCS coolant temperature using the auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following safety injection actuation.

15.2.13.3 Conclusions

The analysis shows that the criteria stated earlier in this section are satisfied since a DNBR less than 1.30 does not exist.

15.2.14 INADVERTENT OPERATION OF EMERGENCY CORE COOLING SYSTEM

This analysis was performed before the boron injection tank and associated 900 gal. of 20,000 ppm boron was deleted from the Watts Bar design basis, and therefore it is still referenced through this section. This analysis is however still limiting since the boron injection tank and associated 900 gal. of 20,000 ppm boron lead to a more conservative result.

15.2.14.1 Identification of Causes and Accident Description

Spurious Emergency Core Cooling System (ECCS) operation at power could be caused by operator error or a false electrical actuating signal. Spurious actuation may be assumed to be caused by any of the following:

1. High containment pressure

2. Low pressurizer pressure

3. Low steam line pressure and low steam generator level

4. Manual actuation

Following the actuation signal, the suction of the centrifugal charging pumps is diverted from the volume control tank to the refueling water storage tank. The valves isolating the boron injection tank from the charging pumps and the valves isolating the boron injection tank from the injection header then automatically open. The charging pumps then force highly concentrated (20,000 ppm) boric acid solution from the boron injection tank, through the header and injection line and into

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time (sec.)
Accidental depressurization of the Reactor Coolant System	Inadvertent opening of one RCS safety valve	0
	Reactor Trip	17.6
	Minimum DNBR occurs	19.6
Accidental depressurization of the Main Steam System	Inadvertent opening of one main steam safety or relief valve	0
	Criticality attained	232
	Pressurizer empties	130
	UHI injection	192
	Boron reaches core	194
Inadvertent Operation of ECCS during power Operation	Charging pumps begin injecting borated water	0
	Low pressure trip point reached	51
	Rods begin to drop	53

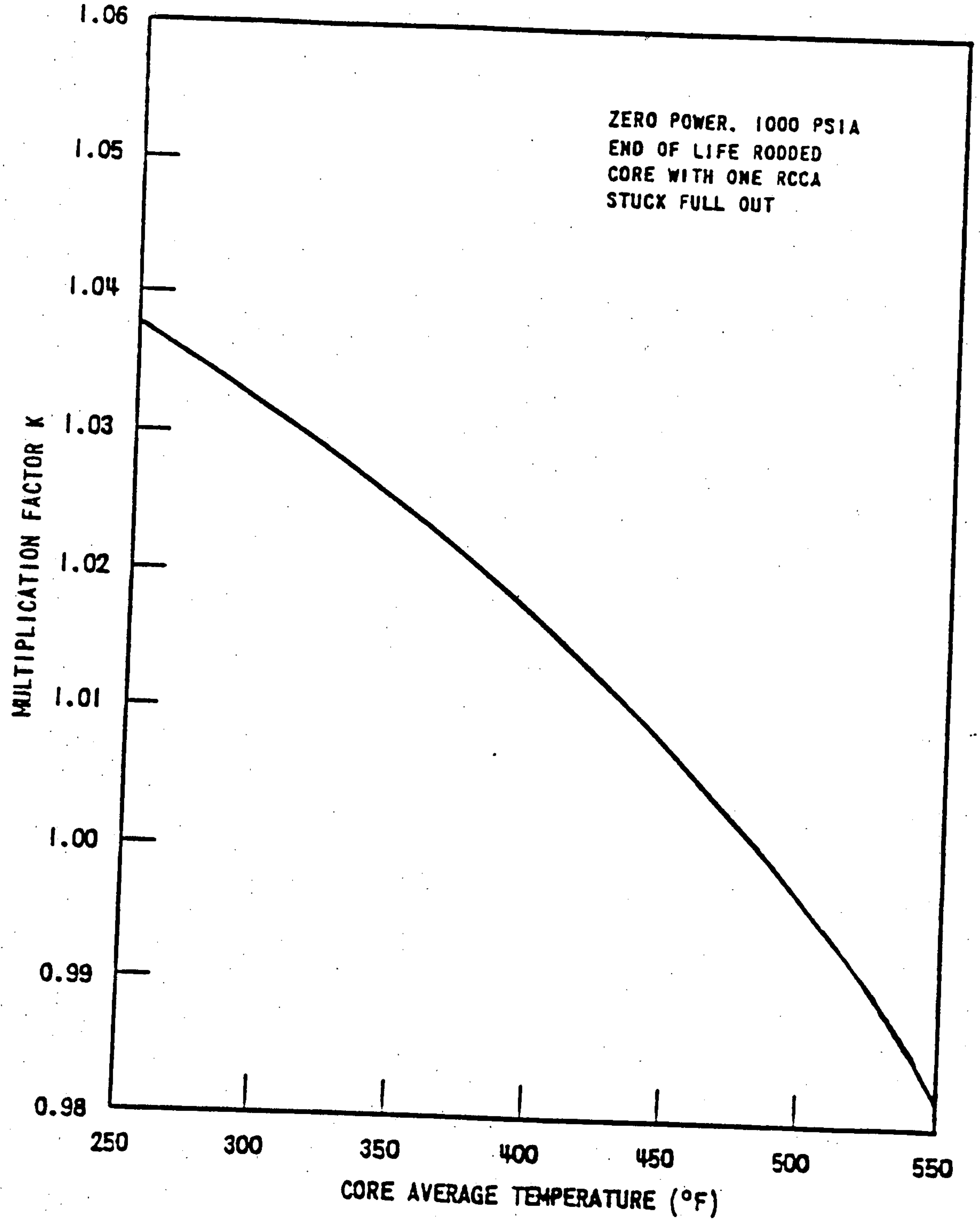

Figure 15.2-40 Variation of K_{eff} with Core Temperature

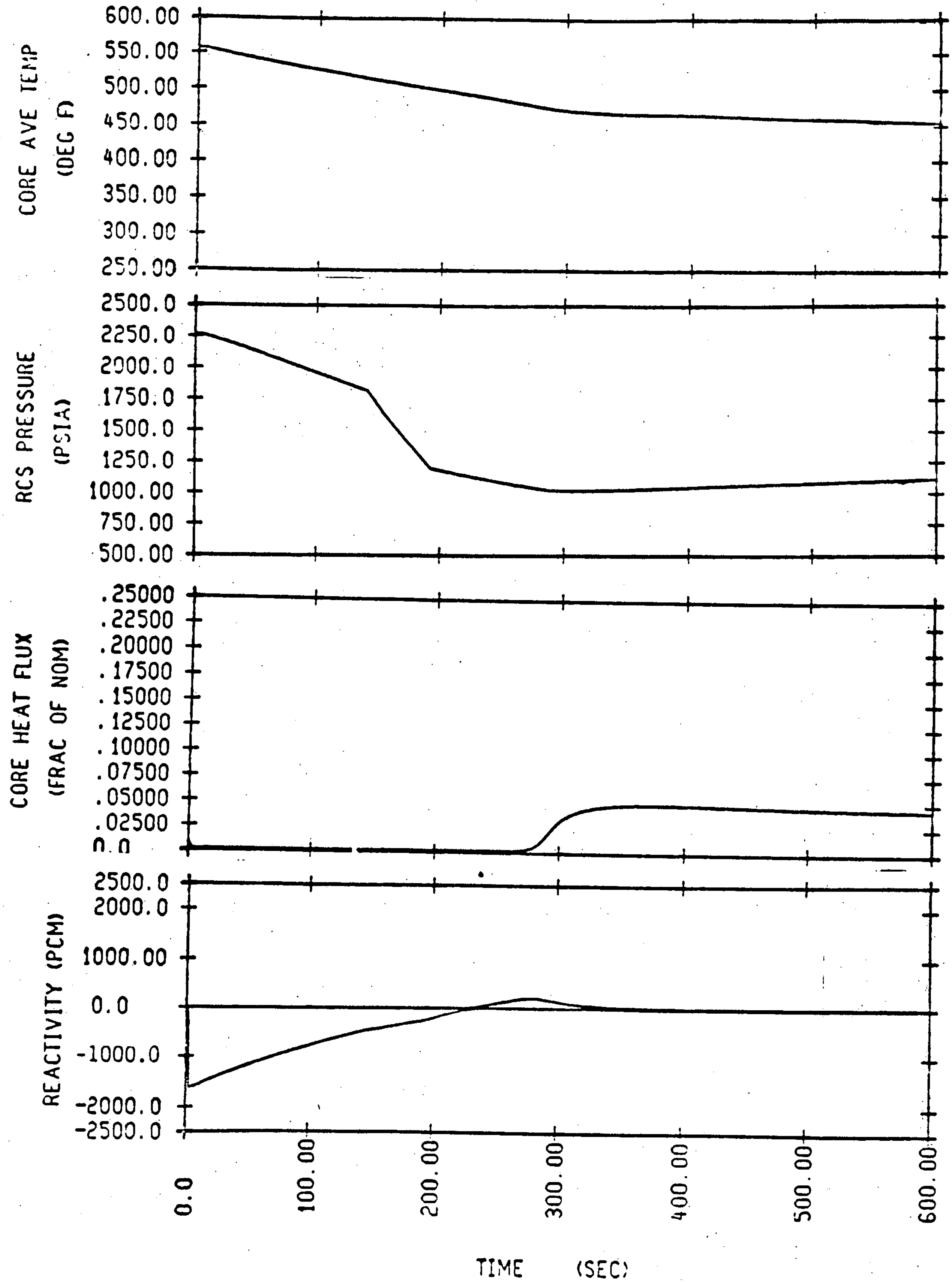

Figure 15.2-41 Transient Response for a Steam Line Break Equivalent to 247 Lbs/Seconds at 1100 PSIA With Outside Power

15.4.2 Major Secondary System Pipe Rupture

15.4.2.1 Major Rupture of a Main Steam Line

15.4.2.1.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam line would result in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the Reactor Coolant System causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam line rupture is a potential problem mainly because of the high power peaking factors which exist assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid injection delivered by the Safety Injection System.

The analysis of a main steam line rupture is performed to demonstrate that the following criterion is satisfied:

Assuming a stuck RCCA with or without offsite power, and assuming a single failure in the engineered safeguards the core remains in place and intact. Radiation doses are not expected to exceed the guidelines of 10CFR100.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive assembly stuck in its fully withdrawn position.

The following functions provide the necessary protection for a steam line rupture:

1. Safety Injection System actuation from any of the following:

   a. Two out of three signals of low pressurizer pressure

   b. Two out of three high containment pressure.

   c. Low steam line pressure in any one loop relative to two out of three other loops.

   d. High steamline flow in two out of four loops coincident with either two out of four loops low steamline pressure or two out of four loops low-low T .

2. The overpower reactor trips (neutron flux and *W*T) and the reactor trip occurring in conjunction with receipt of the safety injection signal.

3. Redundant isolation of the main feedwater lines: Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves, a safety injection signal will rapidly close all feedwater control valves, main feedwater isolation valves, trip the

main feedwater pumps, and close feedwater pump discharge valves.

4. Trip of the fast acting steam line stop valves (main steam isolation valves) (designed to close in less than 5 seconds) on:

   a. Two out of four High-High containment pressure.

   b. Two out of four loops high steamline flow coincident with either two out of four loops low steamline pressure or two out of four low-low T in the coolant loops.

Fast-acting isolation valves are provided in each steam line that will fully close within 7 seconds after a steamline isolation signal setpoint is reached. For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would blowdown even if one of the isolation valves fails to close. A description of steam line isolation is included in Chapter 10.

Steam flow is measured by monitoring dynamic head in nozzles located in the throat of the steam generator. The effective throat area of the nozzles is 1.4 square feet, which is considerably less than the main steam pipe and thus the nozzles also serve to limit the maximum steam flow for a break at any location.

Table 15.4-6 lists the equipment required in the recovery from a high energy line rupture. Not all equipment is required for any one particular break, since it will vary depending upon postulated break location and details of initial conditions. Design criteria and methods of protection of safety related equipment from the dynamic effects of postulated piping ruptures are provided in Section 3.6.

15.4.2.1.2 <u>Analysis of Effects and Consequences</u>

<u>Method of Analysis</u>

The analysis of the steam pipe rupture has been performed to determine:

1. The core heat flux and Reactor Coolant System temperature and pressure resulting from the cooldown following the steam line break. The LOFTRAN [11] Code has been used.

2. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital-computer code, THINC, has been used to determine if DNB occurs for the core conditions computed in Item 1 above.

The following conditions were assumed to exist at the time of a main steam line break accident.

1. End-of-life shut down margin at no load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position: Operation of the control rod banks during core burnup is restricted in such a way that addition of positive reactivity in a steam line break accident will not lead to a more adverse condition than the case analyzed.

2. The negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position: The variation of the coefficient with temperature and pressure has been included. The k versus temperature at 1000 psi corresponding to the negative moderator temperature coefficient used is shown in Figure 15.2-40. The effect of power generation in the core on overall reactivity is shown in Figure 15.4-9.

   The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sector were conservatively combined to obtain average core properties for reactivity feedback calculations. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underprediction of the reactivity feedback in the high power region near the stuck rod. To verify the conservatism of this method, the reactivity as well as the power distribution was checked for the statepoints shown on Table 15.4-8. These core analyses considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high water enthalpy near the stuck RCCA, power redistribution and non-uniform core inlet temperature effects. For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the reactivity calculated including the above local effects for all statepoints. The limiting statepoint is presented in Table 15.4-7. These results verified conservatism; i.e., underprediction of negative reactivity feedback from power generation.

3. Minimum capability for injection of high concentration boric acid (1950 ppm) solution corresponding to the most restrictive single failure in the Safety Injection System. The Emergency Core Cooling System consists of four systems: 1) the passive accumulators, 2) the Residual Heat Removal Systems, 3) the Safety Injection System, and 4) the Upper Head Injection Accumulators.

   The actual modeling of the Safety Injection System in LOFTRAN is described in Reference [11]. The injection curve used is shown in Figure 15.4-10. This corresponds to the flow delivered by one charging pump delivering its full flow to the cold leg header. No credit has been taken for the low concentration borated water, which must be swept from the lines downstream of the RWST prior to the delivery of high concentration boric acid to the reactor coolant loops.

   For the cases where offsite power is assumed, the sequence of events in the Safety Injection System is the following. After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the high head safety injection pump starts. In 12 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed. The volume containing the low concentration borated water is swept, of course, before the 1950 ppm reaches the core. This delay, described above is inherently included in the modeling.

   In cases where offsite power is not available, an additional 10 second delay is assumed to start the diesels and to load the necessary safety injection equipment onto them.

4. Design value of the steam generator heat transfer coefficient including allowance for fouling factor.

5. Since the steam generators are provided with integral flow restrictors with a 1.4 square foot throat area, any rupture with a break area greater than 1.4 square feet, regardless of location would have the same effect on the Nuclear Steam Supply System (NSSS) as the 1.4 square foot break. The following cases have been considered in determining the core power and Reactor Coolant System transients:

   a. Complete severance of a pipe, with the plant initially at no load conditions, full reactor coolant flow with offsite power available.

   b. Case a above with loss of offsite power simultaneous with the steam line break and initiation of the safety injection signal. Loss of offsite power results in coolant pump coastdown.

6. Power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures are determined at end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck control assembly during the return to power phase following the steam line break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck assembly. The power peaking factors depend upon the core power, temperature, pressure, and flow, and, thus, are different for each case studied.

   The core parameters used for each of the two cases correspond to values determined from the respective transient analysis. The limiting statepoints for the two cases are presented in Table 15.4-7.

   Both the cases above assume initial hot shutdown conditions at time zero since this represents the most pessimistic initial condition. Should the reactor be just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection system when power level reaches a trip point. Following a trip at power the Reactor Coolant System contains more stored energy than at no load, the average coolant temperature is higher than at no load and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam line break before the no load conditions of Reactor Coolant System temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes no load condition at time zero.

However, since the initial steam generator water inventory is greatest at no load, the magnitude and duration of the Reactor Coolant System cooldown are greater for than steam line breaks occurring from no load conditions.

7. In computing the steam flow during a steam line break, the Moody Curve 9 for fl/D = 0 is used.

Results

The results presented are a conservative indication of the events which would occur assuming a steam line rupture since it is postulated that all of the conditions described above occur simultaneously.

Core Power and Reactor Coolant System Transient

Figure 15.4-11 shows the Reactor Coolant System transient and core heat flux following a main steam line rupture (complete severance of a pipe) at initial no load condition (case a). Offsite power is assumed available so that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs the initiation of safety injection by low steam line pressure coincident with high steamline flow will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast acting isolation valves in the steam lines by high-high containment pressure signals or high steam line flow coincident with either low steam line pressure or low-low T_{ave}. Even with the failure of one valve, release is limited by isolation valve closure for the other steam generators while the one generator blows down. The steam line stop valves are designed to be fully closed in less than 5 seconds from receipt of a closure signal.

As shown in Figure 15.4-11 the core attains criticality with the RCCA's inserted (with the design shutdown assuming one stuck RCCA) before boron solution at 1950 ppm enters the Reactor Coolant System. A peak core power less than the nominal full power value is attained.

The calculation assumes the boric acid is mixed with, and diluted by the water flowing in the Reactor Coolant System prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the Reactor Coolant System and in the Safety Injection System. The variation of mass flow rate in the Reactor Coolant System due to water density changes is included in the calculation as is the variation of flow rate in the Safety Injection System due to changes in the Reactor Coolant System pressure. The Safety Injection System flow calculation includes the line losses in the system as well as the pump head curve.

Figure 15.4-12 shows the responses of the salient parameters for case b which corresponds to the case discussed above with additional loss of offsite power at the time the safety injection signal is generated. The Safety Injection System delay time includes 10 seconds to start the diesel and 12 seconds to bring the injection pump to full speed. In each case criticality is achieved later and the core power increase is slower than in the similar case with offsite power available. The ability of the emptying steam generator to extract heat from the Reactor Coolant System is reduced by the decreased flow in the Reactor Coolant System. For both these cases the peak power remains well below the nominal full power value.

It should be noted that following a steam line break only one steam generator blows down completely. Thus, the remaining steam generators are still available for dissipation of decay heat after the initial transient is over. In the case of loss of offsite power this heat is removed to the atmosphere via the steam line safety valves.

Following blowdown of the faulted steam generator, the plant can be brought to a stabilized hot standby condition through control of auxiliary feedwater flow and safety injection flow as described by plant operating procedures. The operating procedures would call for operator action to limit RCS pressure and pressurizer level by terminating safety injection flow, and to control steam generator level and RCS coolant temperature using the auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following safety injection actuation.

Margin to Critical Heat Flux

A DNB analysis was performed for both of these cases. The limiting statepoint is presented in Table 15.4-7. It was found that all cases had a minimum DNBR greater than 1.30.

15.4.2.1.3 Conclusions

The analysis has shown that the criteria stated earlier in this section are satisfied.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable and not precluded in the criterion, the above analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive RCCA stuck in its fully withdrawn position.

If it is assumed that there is leakage from the Reactor Coolant System to the secondary system in the steam generators and that offsite power is lost following the steam line break, radioactivity will be released to the atmosphere through the relief or safety valves. Parameters recommended for use in determining the amount of radioactivity released are given in Table 15.4-8.

15.4.2.2 Major Rupture of a Main Feedwater Pipe

15.4.2.2.1 Identification of Causes and Accident Description

A major feedwater line rupture is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. Further, a break in this location could preclude the subsequent addition of auxiliary feed-water to the affected steam generator. (A break upstream of the feedline check valve would affect the Nuclear Steam Supply System only as a loss of feedwater).

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a reactor coolant system cooldown (by excessive energy discharge through the break), or a reactor coolant system heatup. Potential reactor coolant system cooldown resulting from a secondary pipe rupture is evaluated in "Major Rupture of a Main Steam Line" in the FSAR. Therefore, only the reactor coolant system heatup effects are evaluated for a feedline rupture.

Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 15.4-12 and includes the effect of one stuck RCCA. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 seconds after the high neutron flux trip point was reached. This delay is assumed to consist of 0.2 seconds for the instrument channel to produce a signal, 0.15 seconds for the trip breaker to open and 0.15 seconds for the coil to release the rods. The curve of trip rod insertion versus time shown in Figure 15.1-2 was used. The rod ejection transient was evaluated at both thermal design and mechanical design flow rate with their corresponding rod drop times. Analysis has shown that rod ejection transients initiated at thermal design flow and mechanical design flow and their corresponding scram times bound the consequences of transients initiated at intermediate flow rates.

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

Depressurization calculations have been performed for a typical four-loop plant assuming the maximum possible size break (2.75 inch diameter) located in the reactor pressure vessel head. The results show a rapid pressure drop and a decrease in system water mass due to the break. The Safety Injection System is actuated on low pressurizer pressure within one minute after the break. The Reactor Coolant System pressure continues to drop and reaches saturation (1100 to 1300 psi depending on the system temperature) in about two to three minutes. Due to the large thermal inertia of primary and secondary system, there has been no significant decrease in the Reactor Coolant System temperature below no-load by this time, and the depressurization itself has caused an increase in shutdown margin by about 0.2 percent k due to the pressure coefficient. The cooldown transient could not absorb the available shutdown margin until more than 10 minutes after the break. The addition of highly borated (1950 ppm) safety injection flow starting one minute after the break is much more than sufficient to ensure that the core remains sub-criti-

TABLE 15.4-1 (Continued)

TIME SEQUENCE OF EVENTS FOR
CONDITION IV EVENTS

Accident	Event	Time (Seconds)
Major Secondary System Pipe Rupture		
1. Case A		
	Event	**Time (Seconds)**
	Steam Line Ruptures	0.0
	High Steam Flow/Low Steam Pressure Setpoint Reached	0.65
	UHI Injection	9.4
	Criticality Attained	10.2
	Boron Reaches Core	16.5
	Pressurizer Empties	6.6
2. Case B		
	Event	**Time (Seconds)**
	Steam Line Ruptures	0.0
	High Steam Flow/Low Steam Pressure Setpoint Reached	0.65
	UHI Injection	10.0
	Criticality Attained	10.6
	Boron Reaches Core	27.5
	Pressurizer Empties	6.6
Single Reactor Coolant Pump Locked Rotor		
1. Four loops in operation, one locked rotor		
	Rotor on one pump locks	0
	Low flow trip oint reached	0.03
	Rods begin to drop	1.03

TABLE 15.4-6

EQUIPMENT REQUIRED FOLLOWING A HIGH ENERGY LINE BREAK

SHORT TERM (REQUIRED FOR MITIGATION OF ACCIDENT)	HOT STANDBY	REQUIRED FOR COOLDOWN
Reactor trip and safeguards actuation channels including sensors, circuitry, and processing equipment (the protection circuits used to trip the reactor on undervoltage, underfrequency, and turbine trip may be excluded).	Auxiliary Feedwater System including pumps, water supply, and system valves and piping (this system must be placed in service to supply water to operable steam generators no later than 10 minutes after the incident).	Steam generator power-operated relief valves (can be manually operated locally)
		Controls for defeating automatic safety injection actuation during a cooldown and depressurization.
Safety Injection System including the pumps, the refueling water storage tank, and the systems valves and piping.	Reactor containment ventilation cooling units.	Residual Heat Removal System including pumps, heat exchanger, and system valves and piping necessary to cool and maintain the Reactor Coolant System in a cold shutdown condition.
	Capability for obtaining a Reactor Coolant System sample.	
Diesel generators and emergency power distribution equipment.		
Essential Raw Cooling Water System.		
Containment safeguards cooling equipment.		
Main feedwater control valves* (trip closed feature).		

TABLE 15.4-7

LIMITING CORE PARAMETERS USED IN STEAM BREAK

DNB ANALYSIS

CASE	A	
Reactor Vessel inlet temperature	388.3°F 469.0°F	(Faulted SG Loop) (Intact SG Loops)
RCS pressure	906.87	psia
RCS flow	100%	(of nominal)
Heat flux	16.7%	(of nominal)
Time	97	seconds

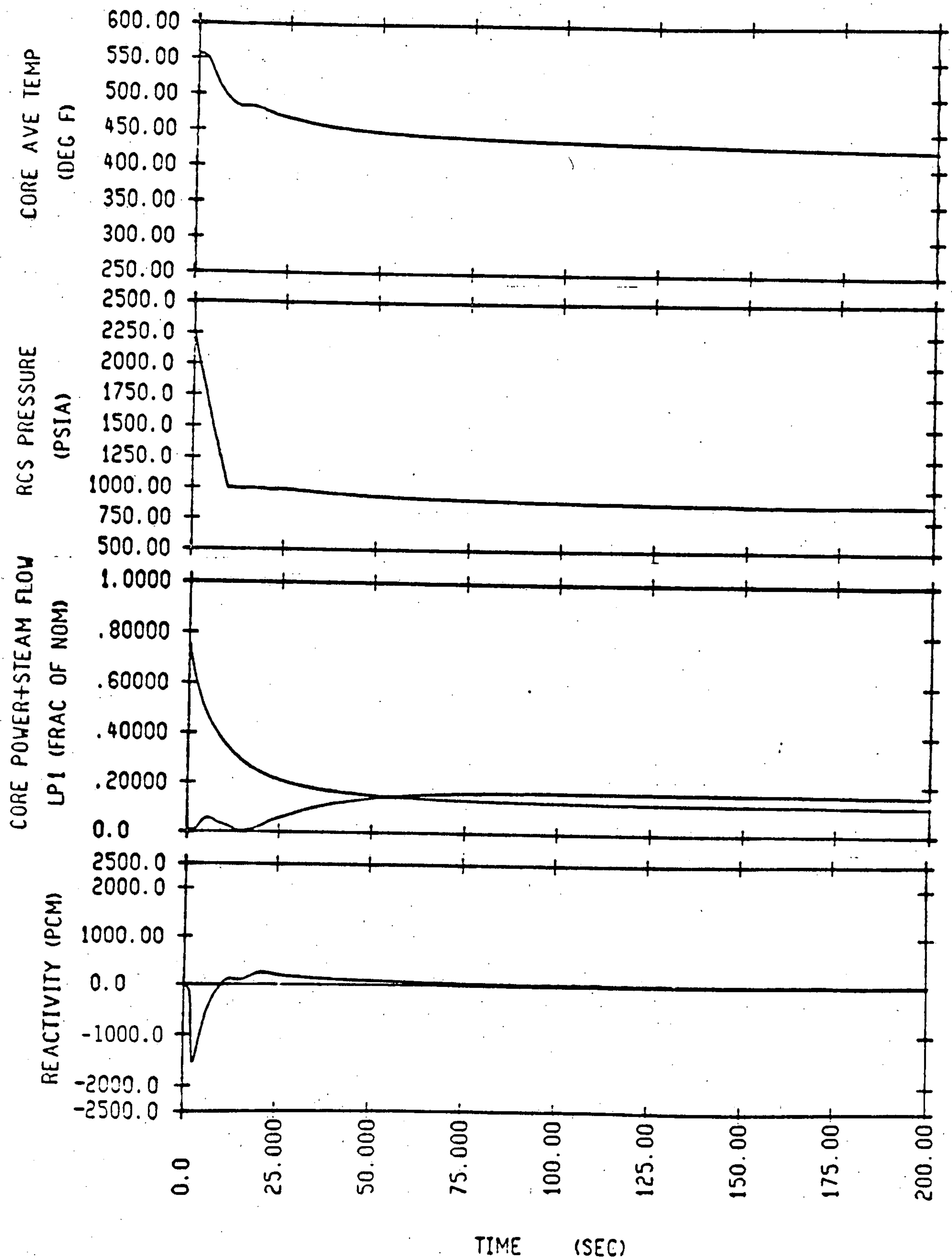

Figure 15.4-11. Transient Response to Steam Line Break with Safety Injection and Offsite Power (case a).

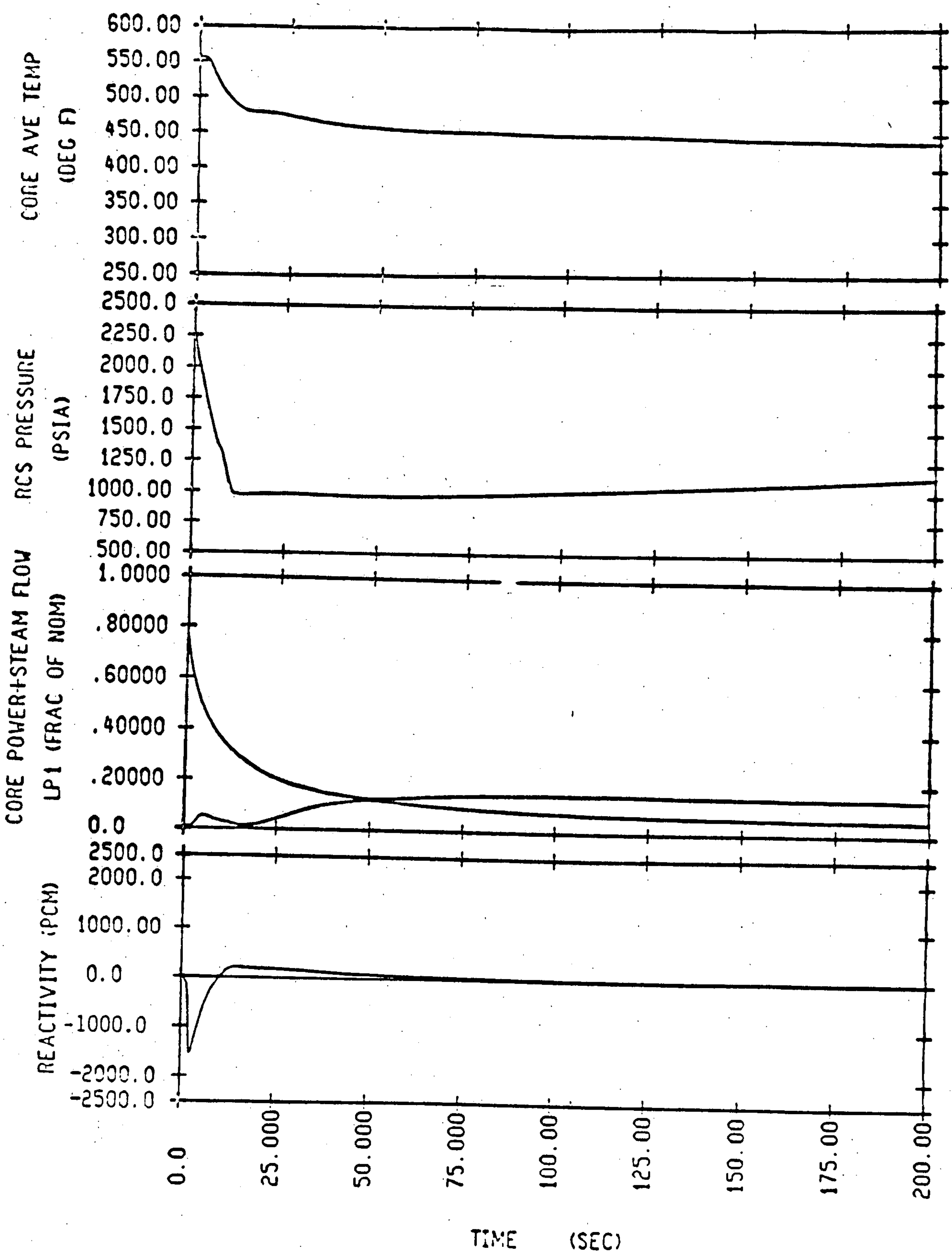

Figure 15.4-12. Transient Response to Steam Line Break with Safety Injection and Without Offsite Power (case b).

ENCLOSURE 3

WATTS BAR NUCLEAR PLANT UNIT 1

PROPOSED MODIFICATIONS TO THE TECHNICAL SPECIFICATIONS
BIT CONCENTRATION REDUCTION/HEAT TRACING DELETION

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 BORON INJECTION SYSTEM

DELETE

BORON INJECTION TANK

LIMITING CONDITION FOR OPERATION

3.5.4.1 The boron injection tank shall be OPERABLE with:

a. A contained borated water volume of 900 gallons,

b. A boron concentration of between 20,000 and 22,500 ppm, and

c. A minimum solution temperature of 145°F.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the boron injection tank inoperable, restore the tank to OPERABLE status within 1 hour or be in HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% delta k/k at 200°F within the next 6 hours; restore the tank to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.1 The boron injection tank shall be demonstrated OPERABLE by:

a. Verifying the contained borated water volume at least once per 7 days,

b. Verifying the boron concentration of the water in the tank at least once per 7 days, and

c. Verifying the water temperature at least once per 24 hours.

DELETE

HEAT TRACING

LIMITING CONDITION FOR OPERATION

3.5.4.2 At least two independent channels of heat tracing shall be OPERABLE for the boron injection tank and for the heat traced portions of the associated flow paths.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one channel of heat tracing on either the boron injection tank or on the heat traced portion of an associated flow path OPERABLE, operation may continue for up to 30 days provided the tank and flow path temperatures are verified to be greater than or equal to 145°F at least once per 8 hours; otherwise, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.2 Each heat tracing channel for the boron injection tank and associated flow path shall be demonstrated OPERABLE:

a. At least once per 31 days by energizing each heat tracing channel, and

b. At least once per 24 hours by verifying the tank and flow path temperatures to be greater than or equal to 145°F. The tank temperature shall be determined by measurement. The flow path temperature shall be determined by either measurement or recirculation flow until establishment of equilibrium temperatures within the tank.

ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump and one Safety Injection pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and Safety Injection pumps except the required OPERABLE charging pump to be inoperable below 310°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

~~3/4.5.4 BORON INJECTION SYSTEM~~

ADD To BASES of 3/4.5.5

Additionally, ~~T~~the OPERABILITY of the RWST ~~Boron Injection System~~ as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident, or a steam line rupture.

~~The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the steam line break analysis are met. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.~~

~~The OPERABILITY of the redundant heat tracing channels associated with the Boron Injection System ensure that the solubility of the boron solution will be maintained above the solubility limit of 135°F at 22,500 ppm boron.~~

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water

BASES

REFUELING WATER STORAGE TANK (Continued)

volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

INSERT INDICATED PARAGRAPH FROM 3/4.5.4