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March 11, 1985

VIRGINIA POWER

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. James R. Miller, Chief
Operating Reactors Branch No. 3
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Serial No. 85-086
E&C/JOE:acm
Docket Nos. 50-338
50-339
License Nos. NPF-4
NPF-7

Gentlemen:

VIRGINIA POWER
AMENDMENT TO OPERATING LICENSES NPF-4 AND NPF-7
NORTH ANNA POWER STATION UNIT NOS. 1 AND 2
PROPOSED TECHNICAL SPECIFICATION CHANGES

Pursuant to 10CFR50.90, Virginia Power requests an amendment, in the form of changes to the Technical Specifications, to Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station Units 1 and 2. The proposed changes and the supporting safety evaluation are enclosed.

There are many problems associated with the high boric acid concentration which must be maintained in the boron injection tanks and concentrated boric acid system. During normal operation reactor coolant letdown is concentrated and recycled to the boric acid tanks via the Boron Recovery System resulting in high radiation levels. These high radiation levels compound the maintenance problems caused by the high boric acid concentrations since the boric acid is a highly corrosive fluid. Leakage has led to the degradation of carbon steel components and the failure of heat tracing which is required to maintain solution solubility. Failure of heat tracing results in additional maintenance problems such as boron plateout and potential line plugging as the solution temperature drops. The increased maintenance causes increased personnel radiation exposures. A reduction in boric acid concentration would reduce maintenance requirements and the associated exposures to plant personnel.

This submittal is part of Virginia Power's ongoing effort to reduce the maintenance requirements and the radiation exposure to plant personnel. A similar program for our Surry Unit Nos. 1 and 2 resulted in the NRC's approval (NRC letter dated February 24, 1984) for removal of the Boron Injection Tank (BIT) at those two units.

Attachment 1 provides the detailed justification for a proposed reduction in the minimum boron concentration at North Anna in the BIT and in the boric acid system from 11.5 wt % to 7.4 wt %. A general description of the current design of the Boron Injection Tank and Boric Acid System is given. The proposed changes are described and operational and maintenance

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Rec'd w/check $150.00

VIRGINIA POWER

benefits are discussed. There are no physical changes required to any of the affected systems to implement this change.

Analyses have been performed to determine the impact of the proposed changes on the appropriate North Anna licensing bases, including a reanalysis of the steamline break accidents discussed in Chapter 15 of the North Anna Updated Final Safety Analysis Report (UFSAR). The analysis has been performed by Virginia Power, using the RETRAN Computer Code and the reactor system transient analysis methodoloby described in our topical report which was transmitted by letter dated April 14, 1981 (Serial No. 215). The methodology, assumptions and results of the analysis are discussed in detail in Attachment 1. This documentation will be incorporated into the North Anna UFSAR at the next annual update following NRC approval.

The changes to the Technical Specifications associated with the proposed boron concentration reduction are provided in Attachments 2 and 3.

This request has been reviewed and approved by the Station Nuclear Safety and Operating Committee and the Safety Evaluation and Control staff. It has been determined that this request does not involve any unreviewed safety questions as defined in 10CFR50.59 or a significant hazards consideration as defined in 10CFR50.92.

We have evaluated this request in accordance with the criteria in 10CFR170.12. A check in the amount of $150, is enclosed as an application fee.

Very truly yours,

W. L. Stewart

W. L. Stewart

Enclosures:

(1) Safety Evaluation for Boron Reduction
(2) Proposed Technical Specification Changes, Unit 1
(3) Proposed Technical Specification Changes, Unit 2
(4) Voucher Check for $150

VIRGINIA POWER

cc: Dr. J. Nelson Grace
Regional Administrator
Region II

Mr. Leon B. Engle
NRC Project Manager - North Anna
Operating Reactors Branch No. 3
Division of Licensing

Mr. M. W. Branch
NRC Resident Inspector
North Anna Power Station

Mr. Charles Price
Department of Health
109 Governor Street
Richmond, Virginia 23219

COMMONWEALTH OF VIRGINIA)
)
CITY OF RICHMOND)

The foregoing document was acknowledged before me, in and for the City and Commonwealth aforesaid, today by W. L. Stewart who is Vice President - Nuclear Operations, of the Virginia Electric and Power Company. He is duly authorized to execute and file the foregoing document in behalf of that Company, and the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 11 day of March, 19 85.

My Commission expires: 12/13, 19 88.

David T. Greene
Notary Public

(SEAL)

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ATTACHMENT 1

SAFETY EVALUATION FOR REDUCTION IN BORON CONCENTRATION IN THE BORON INJECTION TANK AND CONCENTRATED BORIC ACID SYSTEM

NORTH ANNA POWER STATION

UNITS 1 AND 2

A. INTRODUCTION

1. Objective

The nuclear industry has experienced problems with the high boron concentrations which have traditionally been maintained in various station systems, such as the Boron Injection Tank (BIT), associated Safety Injection (SI) lines and the concentrated Boric Acid System. Boric Acid is a corrosive fluid and the leakage of boric acid can lead to the degradation of carbon steel components (such as supports and bolts, etc.) and the failure of line heat tracing which is required to keep the solution temperature above the solubility limit. The failure of heat tracing can in turn lead to boron precipitation and line plugging as the solution temperature drops. At many power stations, these problems have resulted in extensive maintenance and excessive radiation exposures to station personnel. At North Anna Power Station the radiation levels associated with the BIT and Concentrated Boric Acid System have resulted from the concentrating and recycling of reactor coolant letdown to the Boric Acid Tanks (BAT) via the Boron Recovery System.

Like several other utilities, Vepco has been looking at ways to reduce the boric acid concentrations to help alleviate the maintenance and ALARA concerns. The following sections describe the design functions of the Boron Injection Tank and the concentrated Boric Acid System and provide the justification for reducing the minimum boron concentration requirements in

order to reduce these maintenance problems and thus also reduce the associated personnel radiation exposures. The proposed reduction consists of a change in the Boron Injection Tank (BIT) and Boric Acid System concentration from a range of 11.5% - 13% to 7.4% - 9.0% boric acid solution (weight percent). The reduction in BIT concentration can be achieved for the steam line break accident analyses while still meeting all applicable acceptance criteria. The reduction in boric acid system concentration can be accomplished by increasing the minimum allowable Boric Acid Tank inventory associated with each unit from 4450 gallons to 6000 gallons, thereby preserving the capability for cold safe shutdown at any time in life with the most reactive control rod assembly withdrawn from the core.

Section A.2 provides a general description of the current design of the Boron Injection Tank and Boric Acid System and describes the proposed setpoint changes to each system. The operational and maintenance benefits of the proposed changes are discussed in Section A.3.

Evaluations and analyses were performed to assess the impact of the proposed boron concentration reduction upon the existing North Anna accident analyses. It was determined that a boron concentration reduction in the BIT only affects the results of the steamline break transient, the accidental depressurization of the main steam system and the spurious operation of the SI system at power. A detailed discussion of

the supporting analyses and evaluations performed for these transients is provided in section B of this attachment. The proposed boron concentration reduction in the BAT does not impact any of the conclusions of the accident analyses presented in Chapter 15 of the UFSAR.

Section C presents an evaluation of the impact of the proposed plant modifications on plant operations and the results of a review of the North Anna UFSAR.

A.2 Changes to Current System Operations

A.2.1 Boron Injection Tank

The Boron Injection Tank (BIT) is a 900 gallon carbon steel tank which is internally clad with stainless steel and is part of the Safety Injection System; it contains boric acid solution at a concentration of 11.5% to 13.0% by weight. Redundant tank heaters and line heat tracing are provided to maintain a minimum solution temperature at the Technical Specifications limit of 145 degrees F, thus preventing boron precipitation. Recirculation from the BIT to the Boric Acid Tank is maintained continuously via a Boric Acid Transfer Pump to ensure that the BIT is full of concentrated boric acid at all times and to prevent cold spots and stratification within the tank. The BIT is isolated from the Reactor Coolant System and the Charging Pumps during normal plant operation by two sets of parallel isolation valves. Figure 1 illustrates the system design as described above.

The purpose of the BIT is to provide injection of highly concentrated boric acid to the Reactor Coolant System to mitigate the positive reactivity addition resulting from the RCS cooldown during a main steam line break accident. Operation of the BIT, which takes place upon actuation of a Safety Injection Signal, does not impact any of the accident analysis results presented in Chapter 15 of the Updated Final Safety Analysis Report other than the steamline break, accidental depressurization of the main steam system and the spurious SI transients.

During Safety Injection, the suction of the high head safety injection/charging pumps is diverted from the normal suction at the Volume Control Tank (VCT) to the Refueling Water Storage Tank (RWST). The Safety Injection flow path through the BIT is established by the opening of the redundant parallel isolation valves upon a Safety Injection signal. Concurrent with the opening of the BIT isolation valves is the closing of the redundant isolation valves in the recirculation line to the BAT. Flow from the safety injection/charging pumps is introduced into the BIT via a sparger type inlet that distributes the incoming boric acid solution in a 360 degree fan as it enters the tank. This prevents channeling and also ensures radial homogeneity of the boric acid solution.

Current plans are to reduce the required boric acid concentration in the BIT to a range of 7.4 to 9.0% with no physical modifications to the plant.

A.2.2 Boric Acid System

The concentrated boric acid system is a part of the Chemical and Volume Control System described in Section 9.3.4 of the Updated Final Safety Analysis Report. The purpose of the system is to provide an inventory of concentrated boric acid for (1) chemical shim reactivity control, (2) providing makeup to the Reactor Coolant System, Refueling Water Storage Tank, spent fuel pit and refueling cavity as necessary and (3) recirculation of boric acid through the BIT via the boric acid transfer pumps. The system is shared between the two North Anna units. It consists of three Boric Acid Tanks, four boric acid transfer pumps, one batch tank, boric acid filters and associated piping, valves, heat tracing, controls and instrumentation. The Boric Acid Tanks are sized to provide sufficient boric acid to bring both reactors to cold shutdown conditions assuming a stuck control rod. A simplified schematic of the system is shown in Figure 2.

The three Boric Acid Tanks (BAT), which serve as the reservoirs for boric acid inventory, are 7500 gallon stainless steel tanks and are designed for atmospheric pressure. A boric acid solution of 11.5% to 13% by weight is currently maintained at all times. The upper concentration limit of 13% is established to ensure concentrations low enough to remain soluble at a 145 degrees F minimum temperature, which is maintained by redundant tank immersion heaters and line heat tracing.

During normal operation, boric acid is supplied to the boric

acid tanks from the Boric Acid Batching Tank via the Boric Acid Transfer Pumps or from the Boron Recovery System to maintain a minimum volume of 4450 gallons dedicated to each unit.

A reduction in the boron concentration requirements for the Boric Acid System to a range of 7.4 to 9.0% will require increasing the minimum volume of boric acid stored for each operating unit to 6000 gallons. This can be accomplished by resetting the existing level instrumentation and alarms for the new minimum low level. With this increase in volume requirement for each unit, the capability to bring the units to cold shutdown conditions, with the most reactive control rod assembly withdrawn from the core, is preserved. BAT heater controls and the system heat tracing controls will be reset to maintain a minimum of >115°F to maintain solution solubility.

In summary, the proposed reduction in boric acid concentration will require the following:

1. Reset Boric Acid Tank Level instrumentation and alarms for a minimum volume of 6000 gallons.

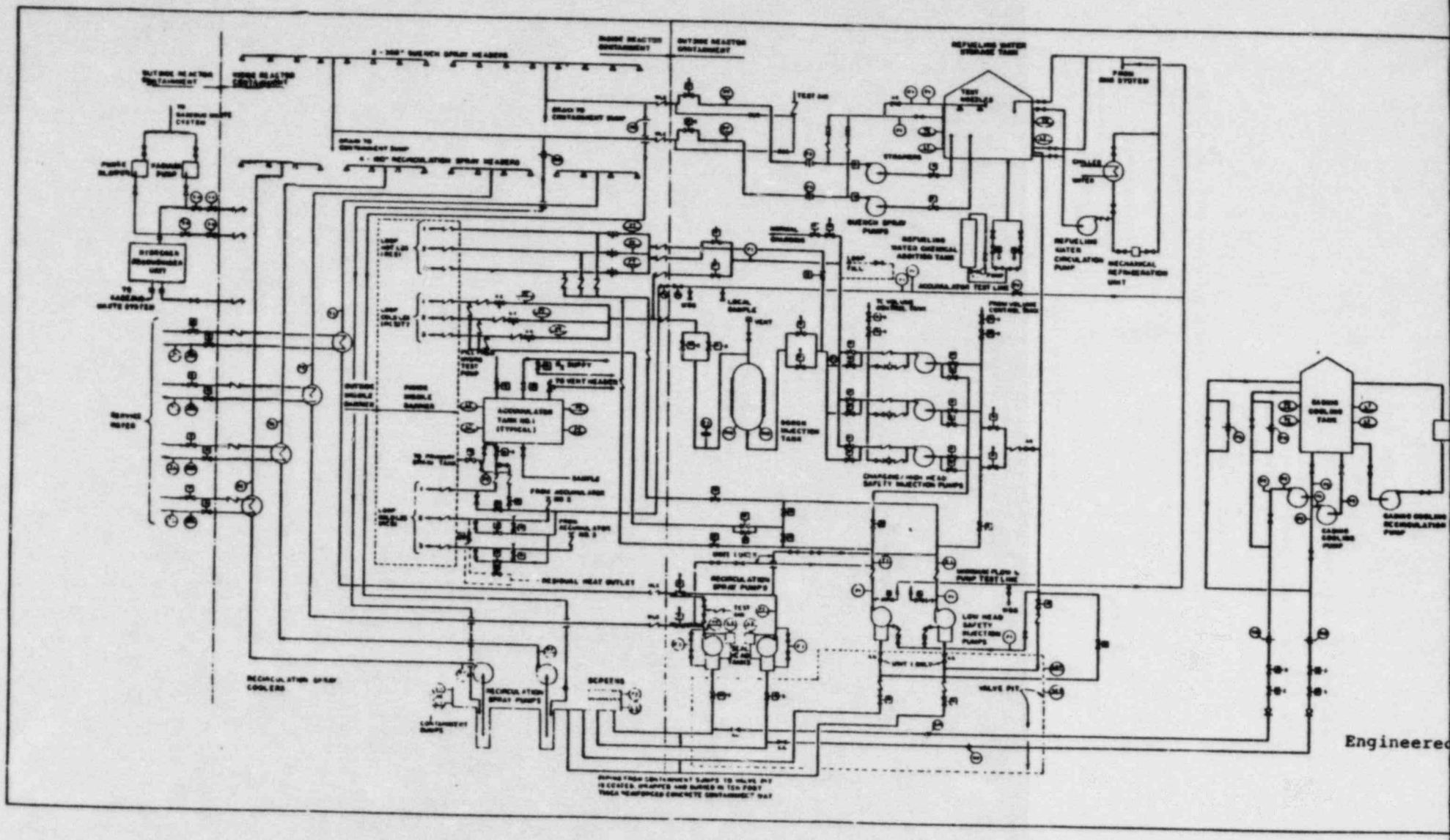
2. Reset Tank heater and heat tracing controls to maintain a minimum solution temperature of >115°F

A.3 Benefits

Reducing the minimum required boric acid concentration for the BIT and the Boric Acid System will improve heat tracing system performance by reducing the temperature that must be maintained to

prevent boron precipitation and by reducing system heat losses. This will result in a reduction in anticipated maintenance requirements (i.e. less heat tracing failures, less line blockages due to boron precipitation), an increase in system reliability and a reduction in future radiation exposures for station personnel. The reduced boron concentrations should also lead to slightly lower radiation levels associated with the BIT and Boric Acid System. The reduction in BIT boron concentration will also reduce the RCS dilution required for a return to power in the event of an inadvertant Safety Injection. This will reduce the amount of letdown which must be processed by the Boron Recovery and Waste Handling systems.

In summary, the proposed reductions in BIT and Boric Acid system concentrations offer significant benefits to Vepco in terms of increased operational reliability, reduced maintenance costs and decreased personnel radiation exposures.


INSIDE REACTOR CONTAINMENT
OUTSIDE REACTOR CONTAINMENT
2 - 100" QUENCH SPRAY HEADERS
4 - 100" RECIRCULATION SPRAY HEADERS
DRAIN TO CONTAINMENT SUMP
TO GASEOUS WASTE SYSTEM
HYDROGEN RECOMBINER UNIT
REFUELING WATER
ACCUMULATOR TANK NO. 1 (TYPICAL)
ACCUMULATOR TANK NO. 2
TO VENT HEADER
SAMPLE
RESIDUAL HEAT OUTLET
RECIRCULATION SPRAY COOLERS
RECIRCULATION SPRAY PUMPS
SCREENS
LOW HEAD SAFETY INJECTION PUMPS
CHARGING/HIGH HEAD SAFETY INJECTION PUMPS
BORON INJECTION TANK
QUENCH SPRAY PUMPS
REFUELING WATER CHEMICAL ADDITION TANK
REFUELING WATER CIRCULATION PUMP
MECHANICAL REFRIGERATION UNIT
GASEOUS COOLING TANK
GASEOUS COOLING PUMP
GASEOUS COOLING RECIRCULATION PUMP
VALVE PIT

Engineered

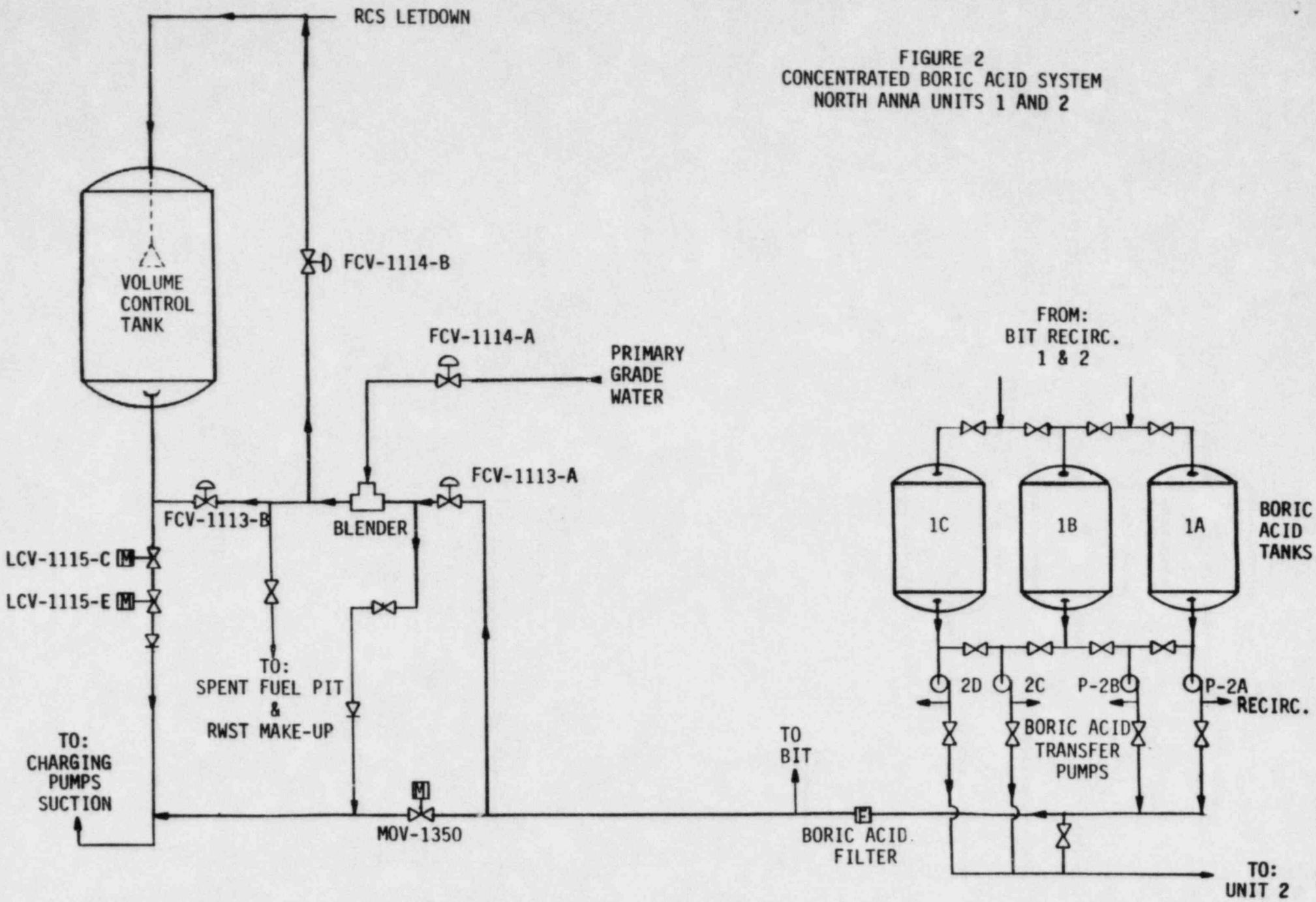


FIGURE 2
CONCENTRATED BORIC ACID SYSTEM
NORTH ANNA UNITS 1 AND 2

B. ACCIDENT ANALYSIS AND EVALUATION

B.1 Accidents Evaluated

The existing North Anna accident analyses presented in the UFSAR were evaluated for potential impact from the proposed boron concentration reduction. Those analyses having a potential impact are the Spurious Operation of the Safety Injection System at Power, the Main Steamline Break and Accidental Depressurization of the Main Steam System.

An evaluation of the NSSS response to an Inadvertant Operation of the SIS at Power was performed for the reduced boron concentration. This evaluation incorporated the assumptions in the exising UFSAR analysis along with the revised boron concentration. It was determined that the lower boron concentration affects only the timing involved for this transient. As noted in the original transient evaluation, this change produces a minor effect on the overall transient by reducing the rate at which the core reactivity is changed. This effect is similar to the effect that would be noted if the conditions chosen for the current analysis were at a different time in the core lifetime which was less responsive to this transient. Thus steady state operation at power remains the bounding condition for this transient.

The proposed boron concentration reduction will impact the existing calculated results for the NSSS response to the range of Main Steamline Breaks. The existing cases of this analysis presented in the UFSAR were reanalyzed. The results are presented in Section B.2 in the UFSAR

format.

The Accidential Depressurization of the Main Steam System analysis in the UFSAR considers two transients; the inadvertent opening of the the largest capacity single steam dump, relief, or safety valve, and a failure of the decay heat release piping. Since the results of these two transients are very similar, only the inadvertent opening of the largest capacity valve was reanalyzed for this set of conditions. Results of this analysis are presented in section B.3. The severity of the results of these two cases is bounded by the results of the main steamline break analyses that were performed.

The containment response following a Main Steamline Break could be impacted if the proposed changes affect the amount of energy released from the steam generators. A conservative evaluation of the mass and energy releases following the limiting steamline break case was performed accounting for the affects of the boron concentration reduction. The containment impact was evaluated by comparing these releases to those in the existing UFSAR analyses. It was concluded that the mass and energy releases, and thus the containment response, of the exising analysis remain bounding.

B.2 Main Steamline Break

Updated UFSAR Accident Analysis Section

15.4.2 MAJOR SECONDARY SYSTEM PIPE RUPTURE

15.4.2.1 Rupture of a Main Steam Line

14.5.2.1.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam pipe would result in an initial increase in steam flow, which decreases 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 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 pipe rupture is a potential problem mainly because of this high power peaking factors that exist, assuming the most reactive rod cluster control assembly 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 pipe rupture is performed to demonstrate that the following criteria are satisfied:

1. Assuming a stuck rod cluster control assembly, with or without offsite power, and assuming a single failure in

the engineered safeguards, there is no consequential damage to the primary system, and the core remains in place and intact.

2. Energy release to containment from the worst steam pipe break does not cause failure of the containment structure.

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

The following functions provide the necessary protection against a steam pipe rupture:

1. Safety injection system actuation from any of the following:

   a. Two out of three low pressurizer pressure signals.

   b. High differential pressure signals between steam lines.

   c. High steam-line flow in two main steam lines (one out of two per line) in coincidence with either low-low reactor coolant system average temperature (two out of three) or low steam-line pressure in any two lines.

   d. Two out of three high containment pressure.

2. The overpower reactor trips (neutron flux and delta 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 that will close the main feedwater valves, 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 main steam trip valves (designed to close in less than 5 sec after receipt of the signal) on

   a. High steam flow in two main steam lines (one out of two per line) in coincidence with either low-low reactor coolant system average temperature or low steam-line pressure in any two lines.

   b. Two out of three intermediate high-high containment pressure.

Each steam line has a fast-acting trip valve with a downstream nonreturn valve. These valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. For example, in the case of a break upstream of the trip valve in one line, closure of either the nonreturn valve in that line or the trip valves in the other lines will prevent blowdown of the other steam generators.

Steam flow is measured by monitoring pressure difference between pressure taps in the steam drum and downstream of the steam-line flow

restrictor nozzles. These nozzles, which are of considerably smaller diameter than the main steam pipe, are located in the steam lines inside the containment near the steam generators to limit the maximum steam flow for any break further downstream.

15.4.2.1.2 Analysis of Effects and Consequences

15.4.2.1.2.1 Method of Analysis. 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 RETRAN 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 calculation, COBRA , has been used to determine if departure from nucleate boiling occurs for the core conditons computed in 1 above.

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

1. End-of-life shutdown margin at no-load, equilibrium xenon conditions, and the most reactive assembly stuck in its fully withdrawn position. Operation of the control rod banks during core burnup is restricted so 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 rod in the fully withdrawn position. The variation of the coefficient with temperature and pressure has been included. The k-effective vs temperature at 1000 psi corresponding to the negative moderator temperature coefficient used plus the Doppler temperature effect, is shown in Figure 15.2-57. The effect of power generation in the core on overall reactivity is shown in Figure 15.4-24.

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. This causes 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 rod cluster control assembly, moderator feedback from the high enthalpy water near the stuck rod cluster control assembly, 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 used in the kinetics analysis

was always larger than the true value for all statepoints in Table 15.4-8, verifying conservatism; i.e., underprediction of negative reactivity feedback from power generation.

3. Minimum capability for injection of high-concentration boric acid (12950 ppm) 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 flow to the cold-leg header. No credit has been taken for the low-concentration boric acid that must be swept from the safety injection lines downstream of the boron injection tank isolation valve prior to the delivery of high-concentration boric acid to the reactor coolant loops.

   The assumed single failure for the steam-line-break analysis is the failure of one safeguards train to function, thus resulting in the maximum delay time for boron to reach the core. Other failures that could affect the severity of the transient are as follows:

   1. Main Steam trip valve.

   2. Feedwater control valve.

   3. Main steam safety valve, atmospheric dump valve, or steam dump valve.

   The failure of any main steam trip valve would result in no more than one steam generator blowing down after line isolation and

would not affect the severity of the transient.

There is a backup feedline isolation valve in series with the feedwater control valve. The failure of either of these valves would not affect the severity of the transient.

The failure of a main steam safety valve, atmospheric dump valve, or main steam dump valve would result in a small increase in steam flow that would be compensated for by full operation of the safety injection system, greatly reducing the delay of boron reaching the core.

4. Four combinations of break sizes and initial plant conditions have been considered in determining the core power and reactor coolant system transients:

   a. Complete severance of a pipe outside the containment, downstream of the steam flow measuring nozzle, with the plant initially at no-load conditions, full reactor coolant flow with offsite power available.

   b. Complete severance of a pipe inside the containment at the outlet of the steam-generator with the plant initially at no-load conditions with offsite power available.

   c. Case (a) above, with loss of offsite power simultaneous with the initiation of the safety injection signal. Loss of offsite power results in coolant pump coastdown.

   d. Case (b) above, with the loss of offsite power simultaneous

with the initiation of the safety injection signal.

5. Power peaking factors corresponding to one stuck rod cluster control assembly 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.

A conservative thermal design flow rate was assumed for the steam-line-break analysis. This flow rate is lower than either the mechanical design flow rate or the measured flow rate. Using a high core flow rate may result in slightly higher peak heat fluxes but would also increase the minimum DNBR.

The values used for three of the four steam-line-break accidents analyzed are given in Table 15.4-8. The three cases are selected on the basis of hot-channel factors, core power, and reactor coolant pressure. The fourth case is less severe relative to DNBR. The core parameters used for each of the three cases correspond to values determined from the respective transient analysis. Five statepoints have been chosen for each case.

All 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 that assumes no-load conditions 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 less for steam-line breaks occurring at power.

6. In computing the steam flow during a steam-line break, the Moody Curve for fL/D = 0 is used.

7. Perfect moisture separation in the steam generator is assumed. The assumption leads to conservative results since, realistically, considerable water would be discharged. Water carryover would reduce the magnitude of the RCS energy removal and system cooldown.

15.4.2.1.2.2 Results. The results presented are a conservative indication of the events that 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 15.4-25 shows the reactor coolant system transient and core heat flux following a main steam pipe rupture (complete severance of a pipe) outside the containment, downstream of the flow- measuring nozzle at initial no-load conditions (case a). The break assumed is the largest break that can occur anywhere outside the containment, either upstream or downstream of the trip valves. Offsite power is assumed available so that full reactor coolant flow exists. The transient shown assumes a 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 high differential pressure between any steam line and the remaining steam lines, or by high steam flow signals in coincidence with either low-low reactor coolant system temperature or low steam-line pressure, will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast-action trip valves in the steam lines by the high steam flow signals in coincidence with either low reactor coolant system temperature or low steam-line pressure. Even with the failure of one valve, release is limited to no more than 10 sec for the other steam generators while the one steam generator blows down. The steam-line trip valves are designed to be fully closed in less than 5 sec after receipt of closure signal with no flow through them. With the high flow existing during a steam-line rupture, the valves will close considerably faster.

The steam flow in Figures 15.4-25 through 15.4-28 represents steam flow from the faulted steam generator only. In addition, all steam generators were assumed to discharge through the break for the first 10 sec.

As shown in Figure 15.4-25, the core attains criticality with the rod cluster control assemblies inserted (with the design shutdown assuming one stuck assembly) before boron solution at 12,950 ppm enters the reactor coolant system from the safety injection system. The delay time consists of the time to receive and actuate the safety injection signal and the time to completely open valve trains in the safety injection lines. The safety injection pumps are then ready to deliver flow. At this stage a further delay time is incurred before 12,950-ppm boron

solution can be injected to the reactor coolant system, due to low-concentration solution being swept from the safety injection lines. A peak core power well below the nominal full-power value is attained.

The calculation assumes that 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 from 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. The accumulators would provide an additional source of borated water when the reactor coolant system pressure decreases to below 550 psia. The integrated flow rate of borated water from the safety injection system for each of the four cases analyzed is shown in Figure 15.4-29.

Figure 15.4-26 shows case b, a steam-line rupture at the exit of a steam generator at no-load. The sequence of events is similar to that described above for the rupture outside the containment, except that criticality is attained earlier due to more rapid cooldown, and a higher peak core average power is attained.

Figures 15.4-27 and 15.4-28 show the response of the salient parameters for cases c and d, which correspond to the cases 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 sec to start the diesel and 10 sec for the safety injection pump to reach 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 core power remains well below 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 main steam safety valves, which have been sized to cover this condition.

A steam-line break assuming an isolated loop is less severe than the case analyzed above. Although operation with an isolated loop results in a reduced primary coolant inventory, this condition is offset by the increased shutdown margin available due to the reduced power defect.

The sequence of events is shown in Table 15.4-9.

The steam-line break analysis adequately addresses the NRC's concerns expressed in IE Bulletin 80-04.

Margin to Critical Heat Flux

Using the transients shown in Figures 15.4-25 through 15.4-28 the Westinghouse W-3 correlation was used in conjunction with the Vepco version of the COBRA core thermal hydraulics code to determine the margin to DNB. Carefully chosen points from each transient were examined and the results showed that all cases had a minimum DNBR greater than 1.30. The power and flow conditions are shown together with pressure and core inlet temperatures in Table 15.4-8 for the three cases most critical to departure from nucleate boiling.

TABLE 15.4-8

CORE PARAMETERS USED IN STEAM BREAK DNB ANALYSIS

Parameter	Case a Time Points 1	2	3	4	5
Reactor vessel inlet temperature to sector nearest affected steam generator, °F	418	411	407	388	365
Reactor vessel inlet temperature to remaining sector, °F	499	495	491	461	428
RCS pressure, psia	1447.	1422.9	1401.0	1279.2	1194.1
RCS, flow %	100	100	100	100	100
Heat flux, %	13.02	13.08	12.80	9.59	9.42
Time, sec	46	58	70.5	156	267

TABLE 15.4-8 (continued)

CORE PARAMETERS USED IN STEAM BREAK DNB ANALYSIS

	Case b				
Reactor vessel inlet temperature to sector nearest affected steam generator, °F	372	363	359	357	355
Reactor vessel inlet temperature to remaining sector, °F	518	513	508	505	500
RCS pressure, psia	1345.7	1162.8	998.8	941.9	909.7
RCS flow, %	100	100	100	100	100
Heat flux, %	23.38	23.72	22.92	22.32	21.80
Time, sec	33.75	45.25	56.75	61.5	70.0

TABLE 15.4-8 (continued)

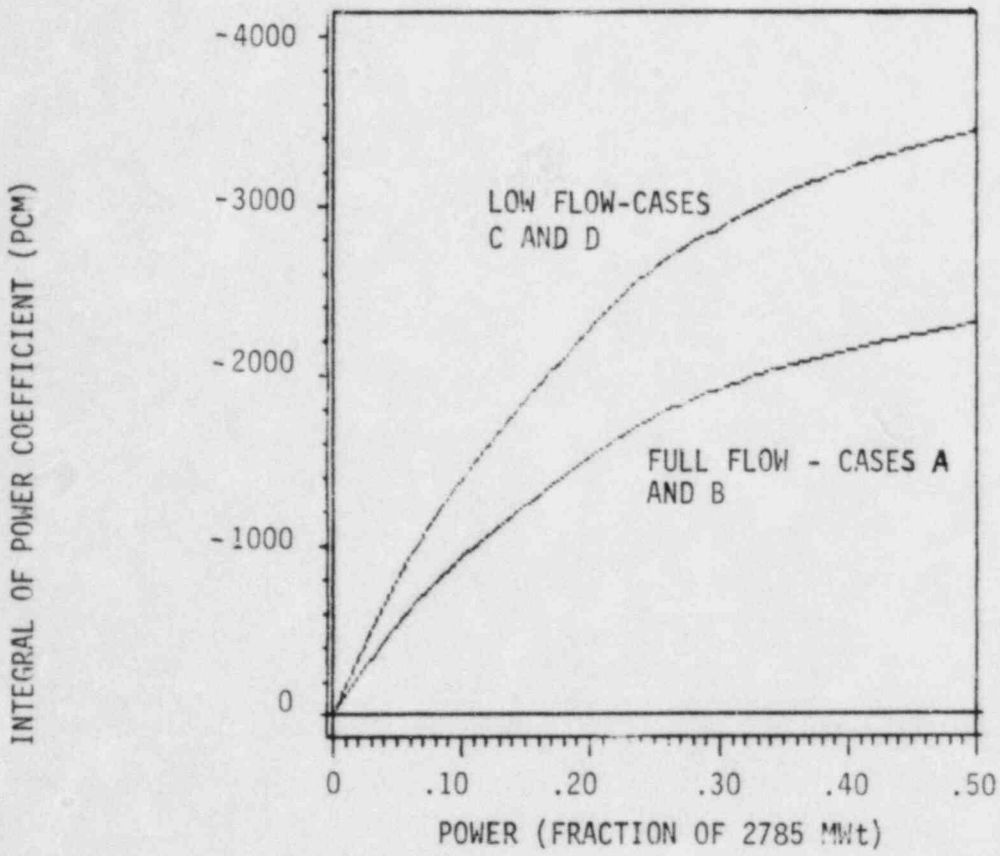
CORE PARAMETERS USED IN STEAM BREAK DNB ANALYSIS

Parameter	Case d 1	2	3	4	5
Reactor vessel inlet temperature to sector nearest affected steam generator, °F	311	244	237	232	226
Reactor vessel inlet temperature to remaining sector, °F	523	514	512	510	508
RCS pressure, psia	1463.9	1393.5	1399.0	1403.2	1407.6
RCS flow, %	18.90	8.94	8.26	7.74	6.99
Heat flux, %	9.456	6.406	6.128	5.691	5.042
Time, sec	50	135	152	168	195

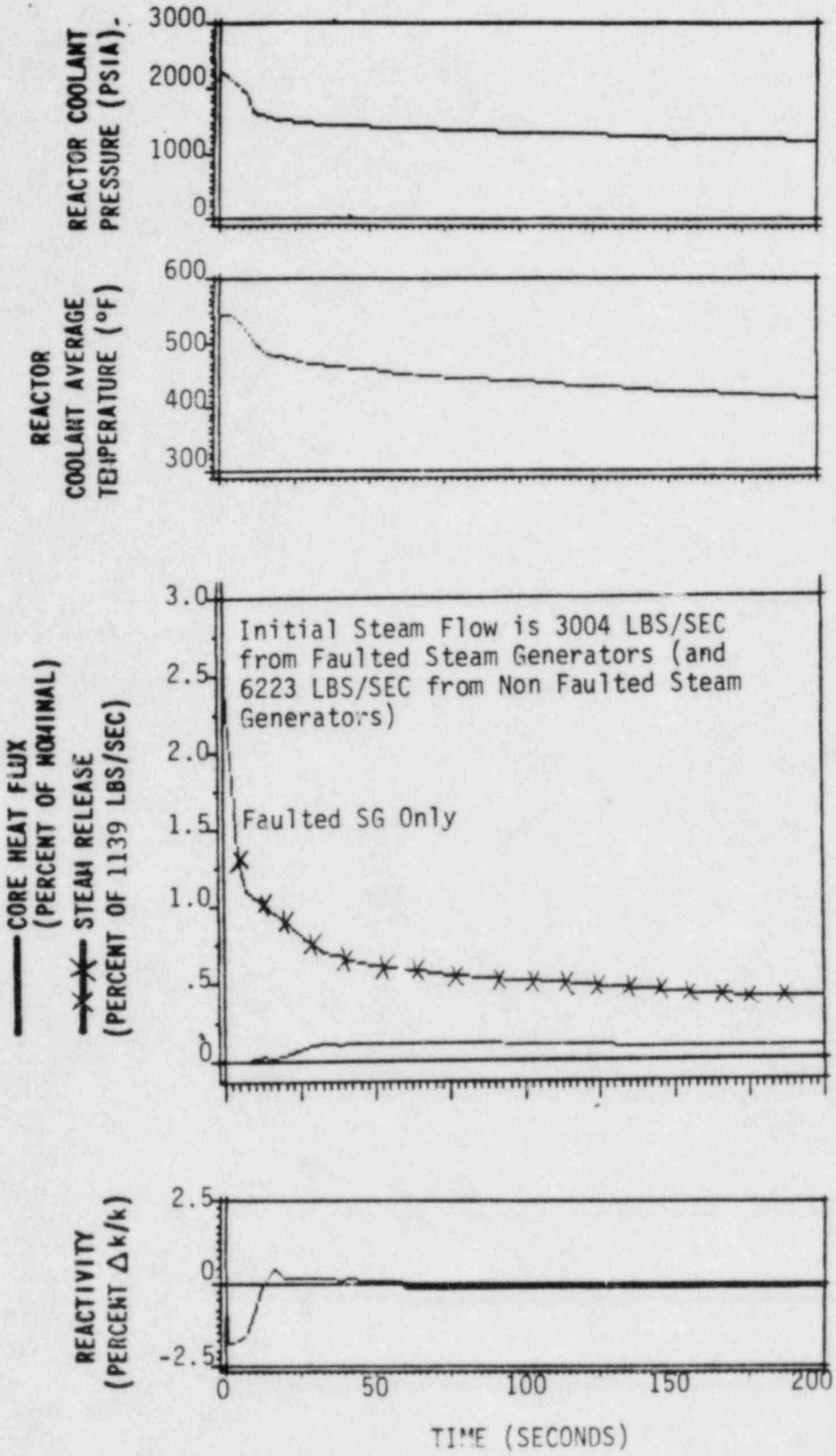
TABLE 15.4-9

TIME SEQUENCE OF EVENTS FOR MAJOR SECONDARY SYSTEM FILE RUPTURE

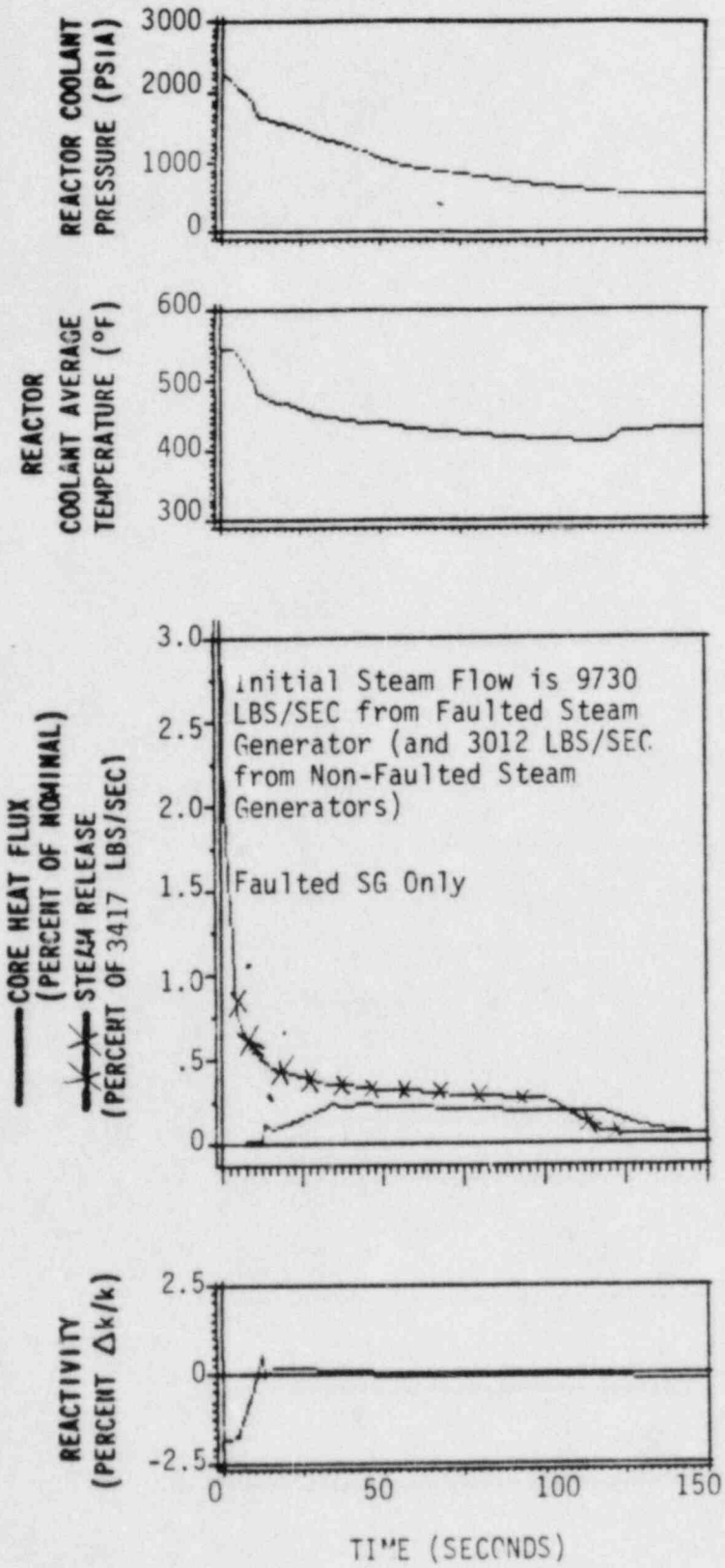
Accident	Event	Time (sec)
Major secondary system pipe rupture		
1. Case a	Steam-line ruptures	0
	Criticality attained	13
	Pressurizer empty	10
	12250 ppm boron reaches loops	32
2. Case b	Steam-line ruptures	0
	Criticality attained	11
	Pressurizer empty	10
	12250 ppm boron reaches loops	32
3. Case c	Steam-line ruptures	0
	Criticality attained	18
	Pressurizer empty	11
	12250 ppm boron reaches loops	41
4. Case d	Steam-line ruptures	0
	Criticality attained	16
	Pressurizer empty	11
	12250 ppm boron reaches loops	41



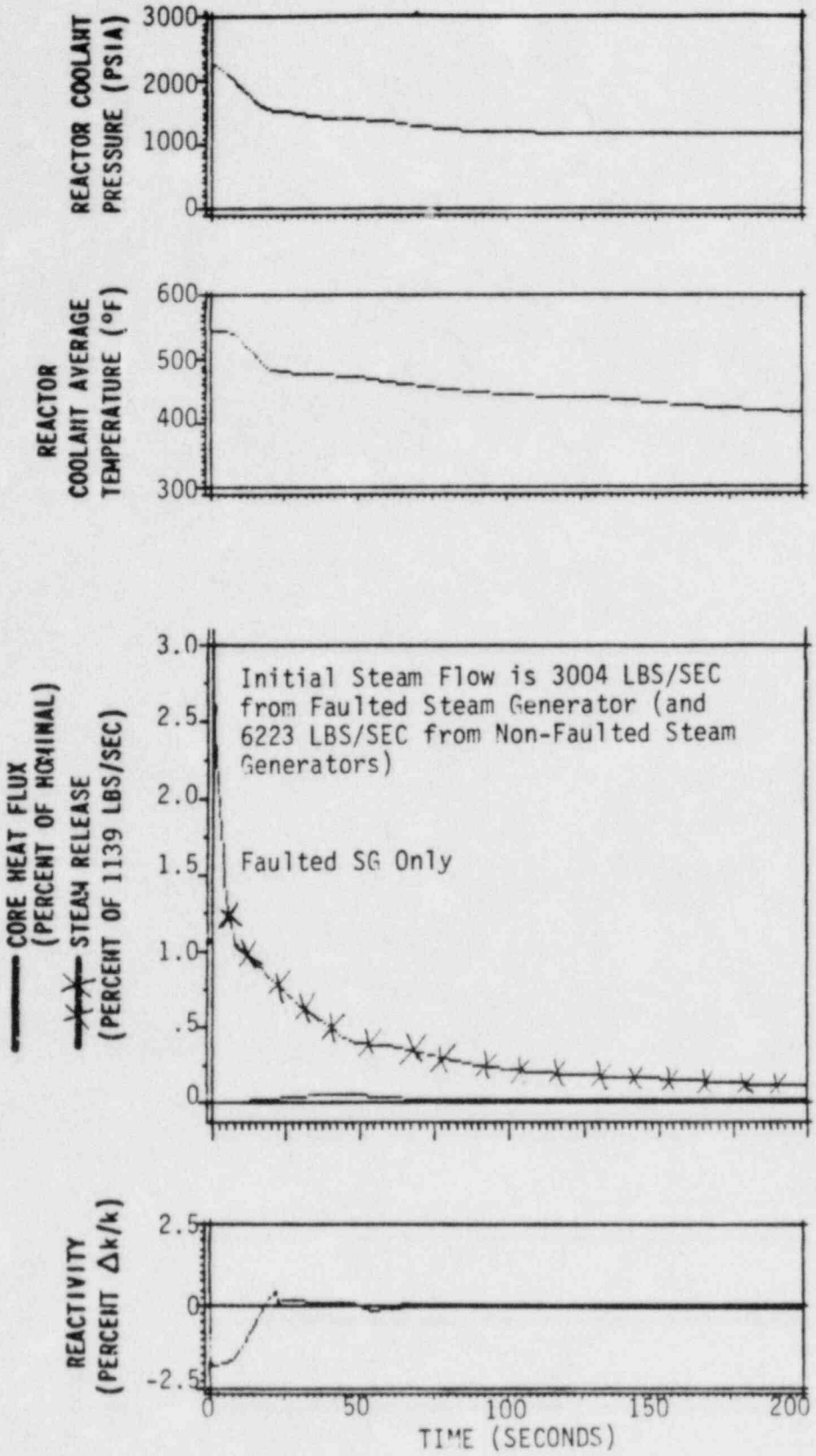
Variation of Reactivity with Power at Constant Core Average Temperature



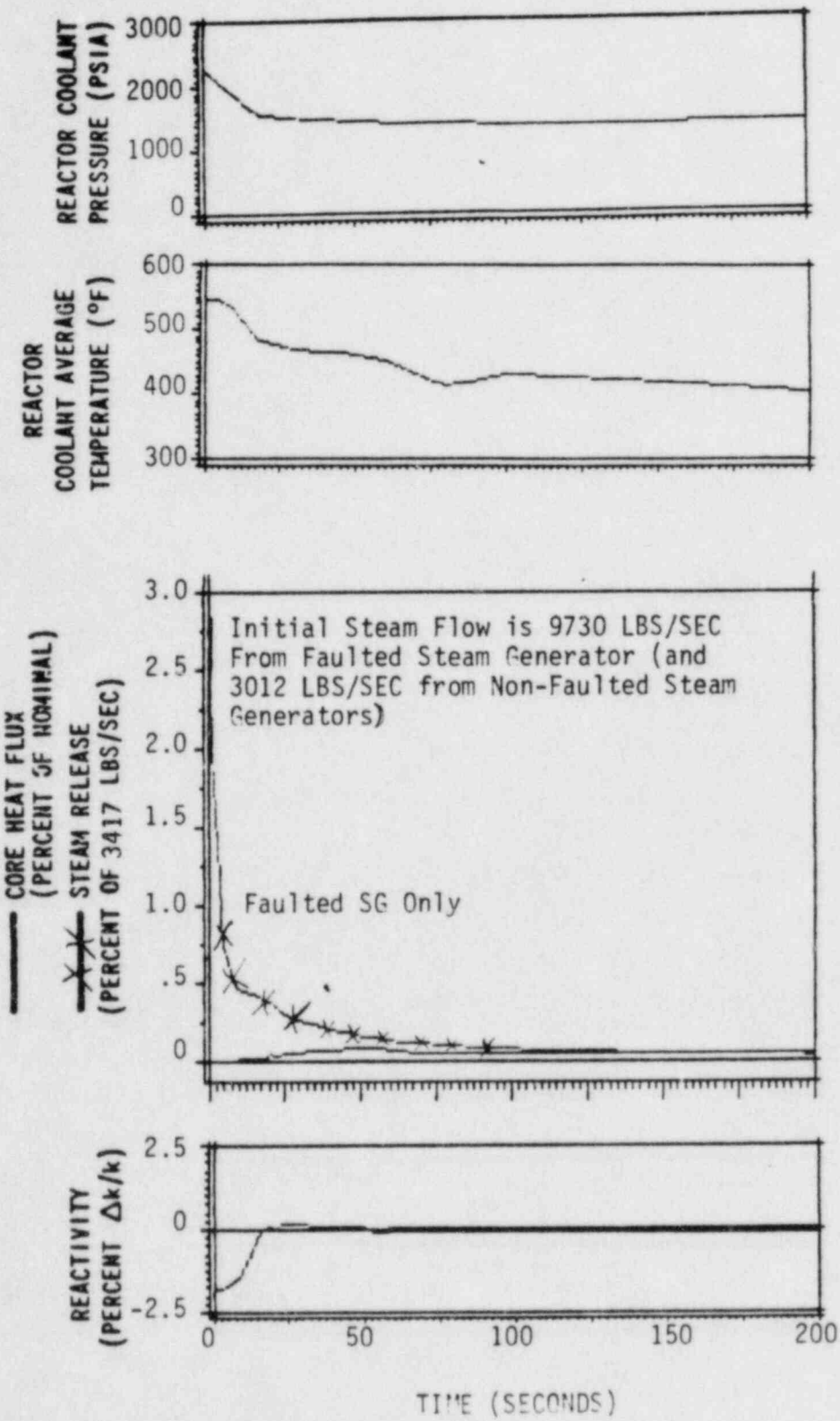
Transient Response to Steam Line Break Downstream of Flow Measuring Nozzle with Safety Injection and Offsite Power (Case a)



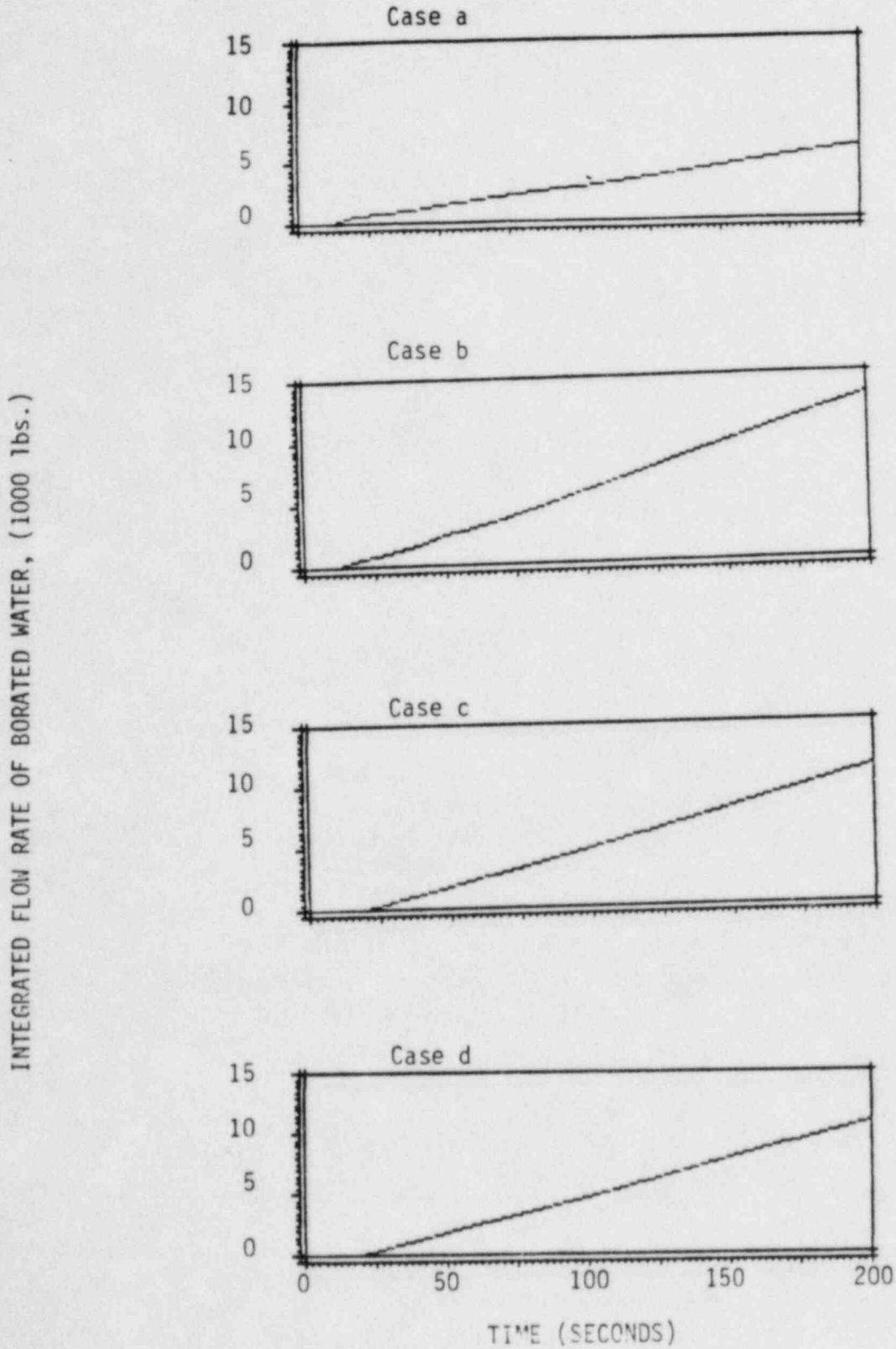
Transient Response to Steam Line Break at Exit of Steam-Generator with Safety Injection and Offsite Power (Case b)



Transient Response to Steam Line Break Downstream of Flow Measuring Nozzle with Safety Injection and without Offsite Power (Case c)



Transient Response to Steam Line Break at Exit of Steam Generator with Safety Injection and without Offsite Power (Case d)


Case a
Case b
Case c
Case d
15
10
5
0
0
50
100
150
200
INTEGRATED FLOW RATE OF BORATED WATER, (1000 lbs.)
TIME (SECONDS)

B.3 Accidental Main Steam System Depressurization

Updated UFSAR Accident Analysis Section

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, or safety valve. The analyses performed assuming a rupture of a main steam pipe are given in Section 15.4.

The steam release as a consequence of this accident results in an initial increase in steam flow that decreases 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 current analysis is performed to demonstrate that the following criterion is satisfied: assuming a stuck rod cluster control assembly and a single failure in the engineered safety features, there will be no departure from nucleate boiling in the core 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.

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 low-low pressurizer pressure signals.

   b. High differential pressure signals between steam lines.

2. The overpower reactor trips (neutron flux and delta 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 that 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.

15.2.13.2 Analysis of Effects and Consequences

15.2.13.2.1 Method of Analysis

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

1. The core heat flux and reactor coolant system temperature and pressure resulting from the cooldown following the inadvertent opening of the largest capacity valve listed above. The RETRAN code has been used.

2. The thermal and hydraulic behavior of the core during this event. A detailed thermal and hydraulic digital-computer calculation, COBRA , has been used to determine if departure from nucleate boiling occurs for the core conditions computed in 1 above.

The following conditions are assumed to exist at the time of a secondary-system-break accident:

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 so that addition of positive reactivity in a secondary-system-break 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 plus the Doppler temperature effect is shown in Figure 15.2-57.

3. Minimum capability for injection of high-concentration boric acid solution corresponding to the most restric-

tive 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. No credit has been taken for the low-concentration boric acid (2000 ppm) that must be swept from the safety injection lines downstream of the boron injection tank isolation valves prior to the delivery of high-concentration boric acid (12,950 ppm) to the to the reactor coolant loops.

4. The case studied is an initial total steam flow of 262 lb/sec at 1020 psia from all steam generators, with offsite power available. This is the maximum capacity of any single steam dump or safety valve. Initial hot shutdown conditions at time zero are assumed, since this represents the most pessimistic initial condition.

   Should the reactor be just critical or operating 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-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 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 less for steam-line breaks occurring at power.

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

6. Perfect mositure separation in the steam-generator is assumed.

7. In the original analysis of the steam-line break incident, which is a depressurization transient, credit was taken for coincident low pressurizer pressure and level for safety injection actuation following a credible break (accidental depressurization of the main steam system). Since that analysis was performed, the low-level coincidence requirement has been removed from the plant protection circuitry. Thus, safety injection actuation can occur on a low pressurizer pressure.

15.2.13.2.2 **Results**

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

Figure 15.2-59 shows the transients arising as the result of a steam release with an initial steam flow of 262 lb/sec at 1020 psia with steam release from one condensor dump valve. The assumed steam release is typical of the capacity of any single steam dump 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 12,950 ppm enters the reactor coolant system, providing sufficient negative reactivity to limit the return to power to a level below 4% of the rated nominal power. With the reactor coolant pumps still providing full flow, the minimum departure from nucleate boiling ratio is well above the limit for Condition II acceptance criteria. The reactivity transient for the case shown in Figure 15.2-59 is more severe than that of a faulted steam-generator safety or relief valve, which is terminated by steam-line differential pressure. The transient is quite 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. Since the transient occurs over a period of about 5 min, the neglected stored energy is likely to have a significant effect in slowing the cooldown.

TABLE 15.2-1 (continued)

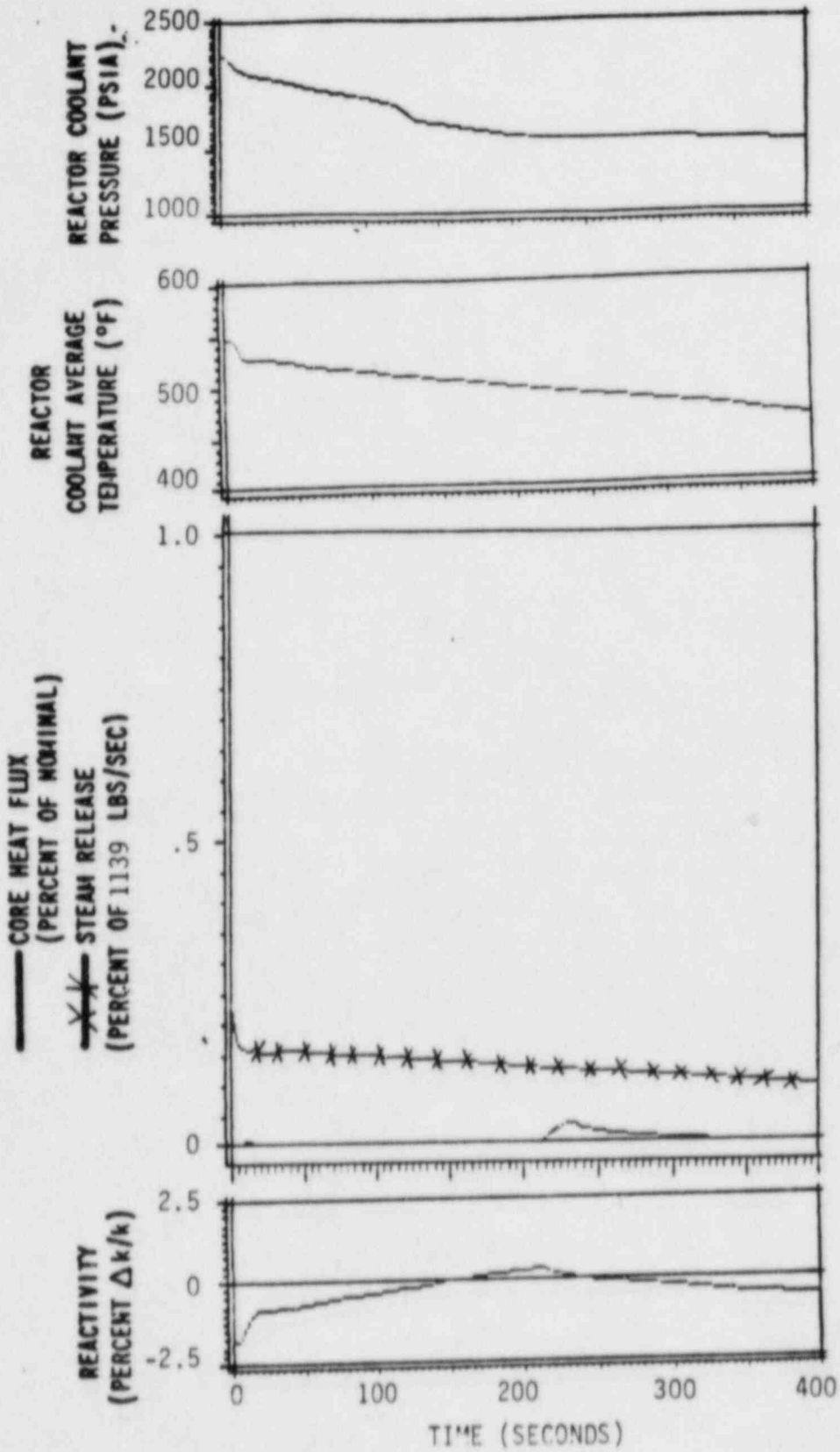
TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time (sec)
Accident depressurization of the reactor coolant system	Inadvertent opening of one RCS safety valve	0
	Reactor trip	21.6
	Minimum DNBR occurs	24.0
Accidental depressurization of the main steam system	Inadvertent opening of one steam safety or relief valve	0
	Pressurizer empties	106
	12950 ppm boron reaches core	233
Decay heat release line break		
Hot zero-power break	Break occurs	0.0
	Pressurizer empties	150
	Safety injection starts	172
	20,000 ppm reaches core	215
	Minimum shutdown margin reached	215

TABLE 15.2-1 (continued)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time (sec)
Hot full-power break	Low-low steam-generator level trip	0.0
	Auxiliary feedwater starts	60
	Pressurizer empties	564
	Safety injection starts	566
	20,000 ppm reaches core	613
	Minimum shutdown margin reached	617
Inadvertent operation of ECCS during power operation.	Charging pumps begin injecting borated water	0
	Low pressure trip point reached	64
	Rods begin to drop	66



TRANSIENT RESPONSE FOR A STEAM LINE BREAK EQUIVALENT TO 252 LB/SEC AT 1020 PSIA WITH OUTSIDE POWER AVAILABLE

C. OPERATIONS/FSAR REVIEW

1. Operations Impact

An evaluation of effects on plant operations was made to determine all positive or negative implications of reducing boron concentrations in the concentrated Boric Acid System and in the Boron Injection Tank. A summary of those impacts is provided below:

a. Increase in time to borate under normal and emergency operating conditions-Station Curve Book nomographs for boron addition will be revised for the decreased minimum concentration of 7.4%. All increases in times required to borate or makeup were found to be satisfactory from a plant operational standpoint.

b. Increase in minimum volume of boric acid. The increase in the minimum volume of boric acid stored for each unit to 6000 gallons (previously 4450) was evaluated relevant to overflow considerations. It was determined that sufficient tank capacity is available to replenish the tank in standard batch volumes without overflowing the tank.

c. The boron evaporators will be operated to produce a nominal boric acid concentrate of 8% if reuse is desired or 12% if disposal (via solid waste) is desired.

d. Setpoint, chemistry, and operating procedure changes have been identified and will be revised in accordance with approved procedures.

2. UFSAR Review

The Updated Final Safety Analysis Report for North Anna has been reviewed and the required changes, other than the changes in the accident analysis section previously addressed, have been identified. Upon approval of this submittal, these changes will be submitted with the normal yearly UFSAR update.

D. CONCLUSIONS

The reduction in boric acid concentration for the Boron Injection Tank and the Concentrated Boric Acid System offers significant benefits to Virginia Electric and Power Company. These benefits include increased operational reliability, reduced maintenance costs and decreased personnel radiation exposure.

Additionally, a detailed operational review was conducted; it has been concluded that the plant can continue to be operated in a safe and efficient manner following the change.

Analyses of the Main Steam Pipe Rupture and Accidental Depressurization of the Main Steam System incorporating the proposed changes have been performed to demonstrate that these transients meet the Condition II transient acceptance criteria. As such, it can be concluded that the change in boron concentration will not cause any safety limits to be exceeded for any incident and consequently no unreviewed safety questions as defined in 10CFR50.59 exist as a result of these proposed changes. The results of this evaluation can be stated as follows.

1. No increase in the probability of occurrence or consequences of an accident will result from these proposed changes. The systems will undergo no physical changes for the reduction in boron concentration and therefore no change in the associated probabilities is expected.

2. Since the proposed changes cause no other system changes (e.g., alterations in plant configuration), and given that the effects upon system accident response are fully described by the parameters evaluated, operation with these proposed changes does not create the possibility of an accident of different type than any evaluated previously in the Safety Analysis Report.

3. The margin of safety as defined in the basis for the Technical Specifications is not reduced. The calculated safety parameters for the affected transients are all well within the allowable limits for the acceptance criteria for Condition II transients.

It has also been determined that the reduction of the boron concentration in the boron injection tank and the boric acid tanks does not pose a significant hazards consideration. This is based on Example vi of those examples of amendments that are considered not likely to involve significant hazards considerations. Example vi partially states "A change which either may result in some increase

to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin , but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan." The analyses do show that the reduction of the boron concentration in the boron injection tank and the boric acid tanks allow a slight increase in the accident consequences (i.e., a small return-to-power for the accidental depressurization of the main steam system). The results of these analyses clearly show that all of the acceptance criteria for these types of transients are met as shown in Chapter 15 of the North Anna UFSAR and the Standard Review Plan.