ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-89-26)

LIST OF AFFECTED PAGES

<u>Unit 1</u>

VII
3/4 1-11
3/4 1-12
3/4 5-1
3/4 5-7
3/4 5-11
3/4 5-12
3/4 5-13
B 3/4 1-3
B 3/4 5-2
B 3/4 5-3

<u>Unit 2</u>

VII
3/4 1-11
3/4 1-12
3/4 5-1
3/4 5-7
3/4 5-11
3/4 5-12
3/4 5-13
B 3/4 1-3
B 3/4 5-2
B 3/4 5-3

9001260065 900112
PDR ADOCK 05000327
P PDC

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

This change was submitted in TS change 89-25.

R120

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum contained borated water volume of 2175 gallons,

2. Between 20,000 and 22,500 ppm of boron, and

3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 35,443 gallons,

2. A minimum boron concentration of ~~2000~~ 2500 ppm, and

3. A minimum solution temperature of 60°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume, and

3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A boric acid storage system and associated heat tracing with:

   1. A minimum contained borated water volume of 7176 ~~6542~~ gallons,

   2. Between 20,000 and 22,500 ppm of boron, and

   3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

   1. A contained borated water volume of between 370,000 and 375,000 gallons,

   2. Between 2500 ~~2000~~ and 2700 ~~2100~~ ppm of boron,

   3. A minimum solution temperature of 60°F, and

   4. A maximum solution temperature of 105°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable and being used as one of the above required borated water sources, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

These changes are associated with UHI removal, and justification has been submitted by TS change 89-25.

a. The isolation valve open,

b. A contained borated water volume of between 7615 ~~7857~~ and 8094 ~~8071~~ gallons of borated water,

c. Between 2400 ~~1900~~ and 2700 ~~2100~~ ppm of boron, and

d. A nitrogen cover-pressure of between 600 ~~385~~ and 683 ~~447~~ psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 12 hours.

c.# With one pressure or water level channel inoperable per accumulator, return the inoperable channel to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. R128

d.# With more than one channel (pressure or water level) inoperable per accumulator, immediately declare the affected accumulator(s) inoperable.

---

*Pressurizer pressure above 1000 psig.

#Actions c and d are in effect until the restart of Unit 2 from the Unit 2 Cycle 4 refueling outage. R128

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection signal:

   a) Centrifugal charging pump

   b) Safety injection pump

   c) Residual heat removal pump

f. By verifying that each of the following pumps develops the indicated discharge pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1.	Centrifugal charging pump	Greater than or equal to 2400 psig
2.	Safety Injection pump	Greater than or equal to 1407 psig
3.	Residual heat removal pump	Greater than or equal to 165 psig

g. By verifying the correct position of each mechanical stop for the following Emergency Core Cooling System throttle valves:

   1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

   2. At least once per 18 months.

Charging Pump ~~Boron~~ Injection Throttle Valves	Safety Injection Cold Leg Throttle Valves	Safety Injection Hot Leg Throttle Valves
Valve Number	Valve Number	Valve Number
1. 63 - 582	1. 63 - 550	1. 63-542
2. 63 - 583	2. 63 - 552	2. 63-544
3. 63 - 584	3. 63 - 554	3. 63-546
4. 63 - 585	4. 63 - 556	4. 63-548

3/4.5.4 ~~BORON INJECTION SYSTEM~~ DELETED

Delete

BORON INJECTION TANK

LIMITING CONDITION FOR OPERATION

3.5.4.1 The boron injection tank shall be OPERABLE with:

a. A minimum contained borated water volume of 900 gallons,

b. Between 20,000 and 22,500 ppm of boron, 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.

EMERGENCY CORE COOLING SYSTEMS (ECCS)

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.

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained borated water volume of between 370,000 and 375,000 gallons,

b. A boron concentration of between 2500 ~~2000~~ and 2700 ~~2100~~ ppm of boron,

c. A minimum solution temperature of 60°F, and

d. A maximum solution temperature of 105°F. R16

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and
2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

6042

~~5400~~ gallons of 20,000 ppm borated water from the boric acid storage tanks or ~~64,160~~ 82,082 gallons of ~~2000~~ 2500 ppm borated water from the refueling water storage tank.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F, is sufficient to provide a SHUTDOWN MARGIN of 1% delta k/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 635 gallons of 20,000 ppm borated water from the boric acid storage tanks or 9,690 gallons of ~~2000~~ 2500 ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.5 and 9.5 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. BR

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod misalignment on associated accident analyses. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

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

Delete

~~The OPERABILITY of the 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 system 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 usuable 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 21000 ppm boron.~~

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the 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 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 7.5 and 9.5 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. B

Add

Additionally, the OPERABILITY of the RWST 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 system cooldown.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

This change was submitted in TS change 89-25.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and at least one associated heat tracing system with:

   1. A minimum contained borated water volume of 2175 gallons,

   2. Between 20,000 and 22,500 ppm of boron, and

   3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

   1. A minimum contained borated water volume of 35,443 gallons,

   2. A minimum boron concentration of ~~2000~~ 2500 ppm, and

   3. A minimum solution temperature of 60°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1. Verifying the boron concentration of the water,

   2. Verifying the contained borated water volume, and

   3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A boric acid storage system and at least one associated heat tracing system with:

   1. A minimum contained borated water volume of 7176 ~~6542~~ gallons,

   2. Between 20,000 and 22,500 ppm of boron, and

   3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

   1. A contained borated water volume of between 370,000 and 375,000 gallons,

   2. Between 2500 ~~2000~~ and 2700 ~~2100~~ ppm of boron, and

   3. A minimum solution temperature of 60°F.

   4. A maximum solution temperature of 105°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable and being used as one of the above required borated water sources, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

These changes are associated with UHI removal, and justification has been submitted by TS change 89-25.

a. The isolation valve open,

b. A contained borated water volume of between 7615 ~~7857~~ and 8094 ~~8071~~ gallons of borated water,

c. Between 2400 ~~1900~~ and 2700 ~~2100~~ ppm of boron, and

d. A nitrogen cover-pressure of between 600 ~~385~~ and 683 ~~447~~ psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 12 hours.

c.# With one pressure or water level channel inoperable per accumulator, return the inoperable channel to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. R113

d.# With more than one channel (pressure or water level) inoperable per accumulator, immediately declare the affected accumulator(s) inoperable.

*Pressurizer pressure above 1000 psig.

#Actions c and d are in effect until the restart of Unit 2 from the Unit 2 Cycle 4 refueling outage. R113

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection signal:

   a) Centrifugal charging pump

   b) Safety injection pump

   c) Residual heat removal pump

f. By verifying that each of the following pumps develops the indicated discharge pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1.	Centrifugal charging pump	Greater than or equal to 2400 psig
2.	Safety Injection pump	Greater than or equal to 1407 psig
3.	Residual heat removal pump	Greater than or equal to 165 psig

g. By verifying the correct position of each mechanical stop for the following ECCS throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

2. At least once per 18 months.

Charging Pump ~~Boron~~ Injection Throttle Valves	Safety Injection Cold Leg Throttle Valves	Safety Injection Hot Leg Throttle Valves
Valve Number	Valve Number	Valve Number
1. 63 - 582	1. 63 - 550	1. 63-542
2. 63 - 583	2. 63 - 552	2. 63-544
3. 63 - 584	3. 63 - 554	3. 63-546
4. 63 - 585	4. 63 - 556	4. 63-548

3/4.5.4 ~~BORON INJECTION SYSTEM~~ DELETED

Delete

BORON INJECTION TANK

LIMITING CONDITION FOR OPERATION

3.5.4.1 The boron injection tank shall be OPERABLE with:

a. A minimum 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.

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained borated water volume of between 370,000 and 375,000 gallons,

b. A boron concentration of between ~~2000~~ 2500 and ~~2100~~ 2700 ppm of boron, R2

c. A minimum solution temperature of 60°F, and

d. A maximum solution temperature of 105°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

BORATION SYSTEMS (Continued)

provide a SHUTDOWN MARGIN from expected operating conditions of 1.6% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires ~~5400~~ 6042 gallons of 20,000 ppm borated water from the boric acid storage tanks or ~~64,160~~ 82,082 gallons of ~~2000~~ 2500 ppm borated water from the refueling water storage tank.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% delta k/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 835 gallons of 20,000 ppm borated water from the boric acid storage tanks or 9,690 gallons of ~~2000~~ 2500 ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.5 and 9.5 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. BR

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod misalignment on associated accident analyses. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

ECCS SUBSYSTEMS (Continued)

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

Delete

~~The OPERABILITY of the 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 system 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 21,000 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

REFUELING WATER STORAGE TANK (Continued)

RWST and the RCS water 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 7.5 and 9.5 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. BR

Add

Additionally, the OPERABILITY of the RWST 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 system cooldown.

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-89-26)

DESCRIPTION AND JUSTIFICATION FOR
BORON INJECTION TANK DEACTIVATION

ENCLOSURE 2

Description of Change

Tennessee Valley Authority proposes to modify the Sequoyah Nuclear Plant (SQN) Units 1 and 2 technical specifications (TSs) to reflect the effects of the boron injection tank deactivation. The refueling water storage tank boron concentration will be changed in Limiting Condition for Operation (LCO) 3.1.2.5. The volume of the boric acid storage system and the boron concentration of the refueling water storage tank will be changed in LCO 3.1.2.6. In Surveillance Requirement 4.5.2.g.2, the reference to boron injection throttle valves will be changed to charging pump injection throttle valves. TSs 3/4 5.4.1 and 3/4 5.4.2 for the boron injection system are being deleted. LCO 3.5.1.1 will be revised with a new boron concentration for the cold leg injection accumulators, and LCO 3.5.5 will be revised with a new boron concentration for the refueling water storage tank.

Reason for Change

The boron injection tank is a component of the safety injection system whose sole function is to provide concentrated boric acid to the reactor coolant to mitigate the consequences of postulated steamline break accidents. In order to verify that the criteria for radiation releases are met, TSs are applied to the boron injection tank and associated equipment. Specifically, the TSs currently ensure that the boric acid concentration is maintained in excess of 20,000 parts per million (ppm), approximately a 12 weight percent solution. Heat tracing is necessary to maintain the tank and associated piping at a sufficiently high temperature so that the minimum concentration requirements may be met. Furthermore, the safety-related nature of the boric acid system requires that the heating systems be redundant.

The required solubility temperature imposes a continuous load on the heaters, and the potential for low-temperature alarm actuation and heater burnout exists. Violation of the TS on concentration in the boron injection tank poses availability problems in that recovery is required within a very short time. If the concentration is not restored within one hour, the plant must be taken to the hot standby condition and borated to the equivalent of 1 percent delta k/k at 200 degrees Fahrenheit. Thus, this requirement has a potentially serious impact on plant availability. In addition, the high boric acid concentration makes recovery from a spurious safety injection signal (which results in injection of the boron injection tank fluid into the reactor coolant system) time consuming and costly.

These potential difficulties unfavorably affecting plant availability, operability, and maintainability can be drastically reduced in severity or eliminated by the boron injection tank deactivation.

Justification for Change

The only accident analyses that are significantly affected by boron reduction, boron injection tank removal, or bypassing are the steamline break transients. These transients are affected with respect to both core integrity and mass and energy release to containment.

The following steamline break cases were considered in the core integrity analysis for SQN: (1) "hypothetical" steamline break, with and without offsite power available, for the largest double-ended rupture of a steam pipe upstream of the flow restrictor (4.6 square feet); (2) "hypothetical" steamline break, with and without offsite power available, for the largest double-ended rupture of a steampipe downstream of the flow restrictor (1.4 square feet); and (3) "credible" steamline break, with offsite power available, for the largest single failed open steam generator relief, safety, or steam dump valve. (Both uniform and nonuniform cases were analyzed; uniform refers to an equal blowdown from all four steam generators; and nonuniform refers to a blowdown from only one steam generator.)

For the hypothetical breaks, the same criteria were applied as are applied in the Final Safety Analysis Report (FSAR). That is, for the most severe Condition IV break, the analyses show that the radiation releases are within the requirements of 10 CFR 100 by demonstrating that the departure from nucleate boiling design basis is met. The steamline break dose calculations performed for the FSAR use a conservative fuel failure level of one percent, although the core analyses show that no consequential fuel failures are anticipated.

The credible steamline break analysis was performed using a new criterion whereby the plant may return to criticality but no damage may occur to the fuel. This constitutes a relaxation of the conservative internal Westinghouse Electric Corporation criterion for Class II events. This relaxed criterion is in compliance with the criteria used by NRC, which require that releases during steamline break accidents remain within the limits set forth in 10 CFR 20. This limit is met with a return to criticality if it is assured that there is no consequential fuel damage.

For SQN, the system was analyzed assuming that the boron injection tank remains installed, without heat tracing, and with the boric acid concentration reduced to zero ppm. This combination provides the most limiting case for the analyses. The analyses for the hypothetical cases show that the departure from nucleate boiling design basis is met, and that no consequential fuel failures are anticipated. The analysis for the credible break shows a return to criticality, but the departure from nucleate boiling design basis is met and no fuel failures are predicted.

The mass and energy analysis considered two cases: (1) large or double-ended steamline ruptures and (2) small or split steamline ruptures. The small break mass and energy calculations were proven to be the limiting case because of the higher containment temperatures reached. Assuming the boron injection tank remains installed, without heat tracing, and with the boric acid concentration reduced to zero ppm, the temperatures and pressures reached in the small break calculations fall below the containment design limits.

Increasing the refueling water storage tank boron concentration is proposed to address the future need (beyond Cycle 4) for a boron concentration increase, which was identified when the Cycle 4 reload safety evaluations were performed. In fact, the Unit 2 Cycle 4 reload safety evaluation stipulated that the boron injection tank needed to remain in operation during Cycle 4. For future fuel reloads, with or without Vantage 5 Hybrid fuel, the boron concentration needs to be increased to accommodate the higher enrichments resulting from extending the fuel cycles (in the process of going from 12 to 18 months) and decreasing the number of fresh fuel assemblies (of the 193 total assemblies, instead of changing out 72 to 80 new assemblies, changing out 60 to 68).

In performing this evaluation, the strategy employed was to select the highest boron concentration possible that would accommodate the removal of the boron injection tank (approximately 55 ppm), accommodate removal of upper head injection (approximately 45 ppm), meet the post-loss of coolant accident sump potential hydrogen-ion activity (pH) requirements specified in the FSAR and TSs, and be acceptable to NRC in order to provide the maximum margin available for future fuel reloads.

The evaluations performed to support boron injection tank deactivation accommodate the effects from the following modifications planned for the Cycle 4 outages for each unit:

1. Resistance temperature detector bypass elimination
2. Eagle 21 digital protection system implementation
3. Upper head injection removal
4. Vantage 5 Hybrid fuel implementation
5. Use of new steamline break protection
6. Reactor trip on steam flow/feed flow mismatch elimination

In summary, plant specific analyses have been performed for SQN's steamline break transients. These analyses have shown that the boron injection tank may be bypassed, eliminated, or reduced in boron concentration and the heat tracing system removed. Additionally, the analyses performed for SQN require an increase in the minimum and maximum boron concentrations for both the refueling water storage tanks and the cold leg accumulators. This increase is necessary to meet the boron requirements in the postaccident sump. Also, to meet the increased boron requirements associated with future core reloads, the volume of the boric acid storage system will increase.

Environmental Impact Evaluation

The proposed change request does not involve an unreviewed environmental question because operation of SQN Units 1 and 2 in accordance with this change would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the Staff's testimony to the Atomic Safety and Licensing Board, supplements to the FES, environmental impact appraisals, or decisions of the Atomic Safety and Licensing Board.

2. Result in a significant change in effluents or power levels.

3. Result in matters not previously reviewed in the licensing basis for SQN that may have a significant environmental impact.

ENCLOSURE 3

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-89-26)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS

ENCLOSURE 3

Significant Hazards Evaluation

TVA has evaluated the proposed TS change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of SQN in accordance with the proposed amendment will not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

The deactivation of the boron injection tank affects the steamline break transients with respect to core integrity and mass and energy release to containment. With the assumption that the boron injection tank remains installed without heat tracing and with boric acid concentration reduced to zero ppm, analyses show that the departure from nucleate boiling design basis is met and no consequential fuel failures are anticipated. Additionally, temperatures and pressures reached in containment would fall below the containment design limits. Therefore, no significant increase in the probability or consequences of a previously analyzed accident would occur.

(2) Create the possibility of a new or different kind of accident from any previously analyzed.

The boron injection tank is a component of the safety injection system whose sole function is to provide concentrated boric acid to the reactor coolant to mitigate the consequences of postulated steamline break analysis. The deactivation of the boron injection tank will therefore affect the steamline break transients, but it will not create the possibility of a new or different type of accident.

(3) Involve a significant reduction in a margin of safety.

The analyses performed for the deactivation of the boron injection tank indicate that the departure from nucleate boiling design basis continues to be met. Additionally, the temperatures and pressures reached in containment would fall below the containment design limits. Since the design bases contain the required margins of safety, no significant reductions in margins of safety will occur.

ENCLOSURE 4

Final Safety Analysis Report
Chapter 15
Analyses Expected Changes

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 (RCS) causes a reduction of coolant temperature and pressure. In the presence of a negative moderato 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 and a single failure in the Engineered Safety Features, ~~there will be no return to criticality~~ * 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. * the DNB design basis will be met

Revise as shown

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 pressurizer pressure.

    b. High differential pressure signals between steam lines.

2. The overpower reactor trips (neutron flux and ΔT) 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 following reactor trip, a safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the feedwater isolation valves.

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.

Revise as shown

1. A full plant digital computer simulation, LOFTRAN ~~MARVEL~~ (Reference 4 ~~7~~) code, to determine RCS temperature and pressure during cooldown.

Revise as shown

2. An evaluation ~~analysis~~ to determine that the reactor DNBR design basis is met. ~~does not return critical.~~ 5

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

DCN No. M0533A
Page ____

1. End of life shutdown margin at no load, equilibrium xenon conditions, and with the most reactive 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 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 15.2.13-1.

3. Minimum capability for injection of high concentration boric acid solution corresponding to the most restrictive single failure in the Safety Injection System. The injection curve assumed is shown in Figure 15.2.13-2. 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 which must be swept from the safety injection lines downstream of the RWST ~~boron injection tank isolation valves~~ prior to the delivery of high concentration boric acid (~~20,000~~ 1950 ppm) to the reactor coolant loops. Revise as Shown

4. The case studied is an initial total steam flow of 228 lbs/second at 1015 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 RCS 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 RCS are reached. After the additional stored energy is removed, cooldown proceeds 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 RCS cooldown are less for steam line breaks occurring at power. 5

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

DCN No. M01533A
Page ____

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

7. The upper head injection system (UHI) is simulated. As stated in WCAP-8185 the significant effect of UHI is to retard the pressure decrease of the RCS. This in turn, reduces the flow of borated water from the Safety Injection System. The potentially detrimental effect is compensated by boration provided by the UHI.

Results

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 15.2.13-3 shows the transients arising as the result of a steam release having an initial steam flow of 228 lbs/second at 1015 psia with steam release from one safety valve. The assumed steam release is the maximum capacity of any single steam dump or safety valve. Safety Injection is conservatively assumed to be initiated by low pressurizer pressure although steam line differential pressure would provide a more rapid signal. Operation of only one centrifugal charging pump is considered. Boron solution at ~~20,000~~ 19,950 ppm enters the RCS providing sufficient negative reactivity to maintain the ~~reactor well below criticality~~ * . The reactivity transient for the case shown in Figure 15.2.13-3 is more severe than that of a failed steam generator safety or relief valve which is terminated by steam line differential pressure or a failed condenser dump valve which is terminated by low pressurizer pressure and level. 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 or the energy stored in the other steam generators. Since the transient occurs over a period of above five minutes, the neglected ~~store~~ stored energy is likely to have a significant effect in slowing the cooldown.

Revise as shown

Revise as shown

* minimum DNBR above the limit value.

15.2.13.3 Conclusions

The analysis ~~has shown that the criteria stated earlier in this section is satisfied.~~ Since the ~~reactor does not return to critical the possibility of a DNBR less than 1.30 does not exist.~~ ▲

▲ minum DNBR remains above the limit value, no consequential damage to the core or RCS occur.

~~15.2.14 Spurious Operation of the Safety Injection System at Power~~ *

* NOTE: THIS ANALYSIS OF RECORD INVOLVES THE BORON INJECTION TANK and provides bounding results.

Revise as shown

~~15.2.14.1 Identification of Causes and Accidents Descriptions~~

Spurious SIS operation at power could be caused by operator error or a false electrical actuating signal. A spurious signal in any of the following channels could cause this incident.

DCN No. M01533A
Page ____

1. High containment pressure

2. Low pressurizer pressure

3. High steam line differential pressure

4. High steam line flow coincident with either low average coolant temperature or low steam line pressure.

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 ~~(BIT)~~ from the charging pumps and ~~the valves isolating the BIT from~~ the injection header then automatically open . The charging pumps then ~~force highly concentrated~~ provide RWST water (~~20,000~~ 2500 ppm) ~~boric acid solution from the BIT,~~ through the header and injection line and into the cold legs of each loop. The safety injection pumps also start automatically but provide no flow when the RCS is at normal pressure. The passive injection system and the low ~~head system also provide no flow at normal RCS pressure.~~ Even though the BIT function removed, the Westinghouse analysis conservatively assumes that the BIT is still intact with 20,000 ppm boron in this case. An SIS signal normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates the SIS will also produce a reactor trip. Therefore, two different courses of events are considered. | 6 | 7

Revise as shown

Case A Trip occurs at the same time spurious injection starts

Case B The reactor protection system produces a trip later in the transient.

For Case A the operator should determine if the spurious signal was transient or steady state in nature, i.e., an occasional occurrence or a definite fault. The operator must also determine if the safety injection system must be defeated for repair. For the former case the operator would stop the safety injection and bring the plant to the hot shutdown conditions. If the safety injection system must be disabled for repair, boration should continue through the normal boration mode and the plant brought to cold shutdown.

For Case B the reactor protection system does not produce an immediate trip and the reactor experiences a negative reactivity excursion causing a decrease in reactor power. The power unbalance causes a drop in T_{avg} and consequent coolant shrinkage. Pressurizer pressure and level drop. Load will decrease due to the effect of reduced steam pressure on load after the electro-hydraulic governor fully opens the turbine throttle valve. If automatic rod control is used, these effects will be lessened until the rods have moved out of the core. The transient is eventually terminated by the reactor protection system low pressure trip or by manual trip.

The time to trip is affected by initial operating conditions including core burnup history which affects initial boron concentration, rate of change of boron concentration, Doppler and moderator coefficients. | 5

Recovery from this incident for case B is made in the same manner described for case A. The only difference is the lower T_{avg} and pressure associated with the power unbalance during the transient. The time at which reactor trip occurs is of no concern for this accident. At lower loads coolant contraction will be slower, resulting in a longer time to trip.

DCN No. MD-5334
Page____

15.2.14.2 Analysis of Effects and Consequences

Method of Analysis

The spurious operation of the SIS system is analyzed by employing the detailed digital computer program LOFTRAN (Reference 4). The code simulates the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and the effect of the safety injection system. The program computes pertinent plant variables including temperatures, pressures, and power level.

Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits. Analysis of several cases shows the results are relatively independent of time to trip.

A transient is presented representing conditions at beginning of core life. Results at end of life are similar except that moderator feedback effects result in a slower transient.

The assumptions are:

1. Initial Operating Conditions - the initial reactor power and Reactor Coolant System temperatures are assumed at their maximum values consistent with the steady state full power operation including allowances for calibration and instrument errors.

2. Moderator and Doppler Coefficients of Reactivity - A low beginning of life moderator temperature coefficient was used. A low absolute value Doppler power coefficient was assumed.

3. Reactor Control - The reactor was assumed to be in manual control.

4. Pressurizer Heaters - Pressurizer heaters were assumed to be nonoperable in order to increase the rate of pressure drop.

Revise as Shown

5. ~~Boron~~ Injection - At time zero two charging pumps inject 20,000 ppm borated RWST water into the cold legs of each loop. Even though the BIT is functionally removed from Unit 2, (W) analysis conservatively assumes that the BIT is still intact. | 1

TABLE 15.1.2-2 (Sheet 2)
(Continued)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES *

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED MODERATOR(1) TEMPERATURE (Δk/°F)	MODERATOR(1) DENSITY (Δk/gm/cc)	DOPPLER(2)	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED (MWt)
CONDITION II (Continued)					
Loss of Normal Feedwater	BLKOUT	---	NA	NA	3577
Loss of Off-Site Power to the Plant Auxiliaries (Plant Blackout)	BLKOUT	---	NA	NA	3423
Excessive Heat Removal Due to Feedwater System Malfunctions	MARVEL	---	0.43	Lower	0 and 3423
Excessive Load Increase	LOFTRAN	---	0 and 0.43	Lower	3423
Accidental Depressurization of the Reactor Coolant System	LOFTRAN	---	0	Upper	3423
Accidental Depressurization of the Main Steam System	REVISE ~~MARVEL~~ LOFTRAN	---	Function of Moderator Density See Subsection 15.2.13 (Figure 15.2.13-1)	REVISE ~~-2.2~~ pcm/°F -2.9	0 (Subcritical)
Inadvertent Operation of ECCS During Power Operation	LOFTRAN	---	0	Lower	3423

2

DCN No. MDL533A
Page____

TABLE 15.1.2-2 (Sheet 4)
(Continued)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES *

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED MODERATOR(1) TEMPERATURE (Δk/°F)	MODERATOR(1) DENSITY (Δk/gm/cc)	DOPPLER(2)	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED (MWt)
CONDITION IV (Continued)	REVISE			REVISE	
Major secondary system pipe rupture up to and including double-ended rupture (Rupture of a Steam Pipe)	~~MARVEL~~ LOFTR[illegible] THINC	Function of Moderator Density See 15.4.2 (Figure 15.4.2-1)		~~-2.2~~ pcm/F -2.9	0 (Critical)
Steam Generator Tube Rupture	NA	NA	NA	NA	3577
Single Reactor Coolant Pump Locked Rotor	PHOENIX, LOFTRAN THINC, FACTRAN	--	0	Upper	2396 and 3423
Fuel Handling Accident	NA	NA	NA		3577
Rupture of a Control Rod Mechanism Housing (RCCA Ejection)	TWINKLE, FACTRAN LEOPARD	-1 pcm/°F BOL --- -26 pcm/°F BOL		Consisent with lower limit shown Figure 15.1.6-1	0 and 3423

Notes:

(1) Only one is used in an analysis i.e. either moderator temperature or moderator density coefficient.

(2) Reference Figure 15.1.6-1

DCN No. MDL533A
Page ____

4

6937-97

DCN No. [illegible]
Page ___

DELETE and Replace

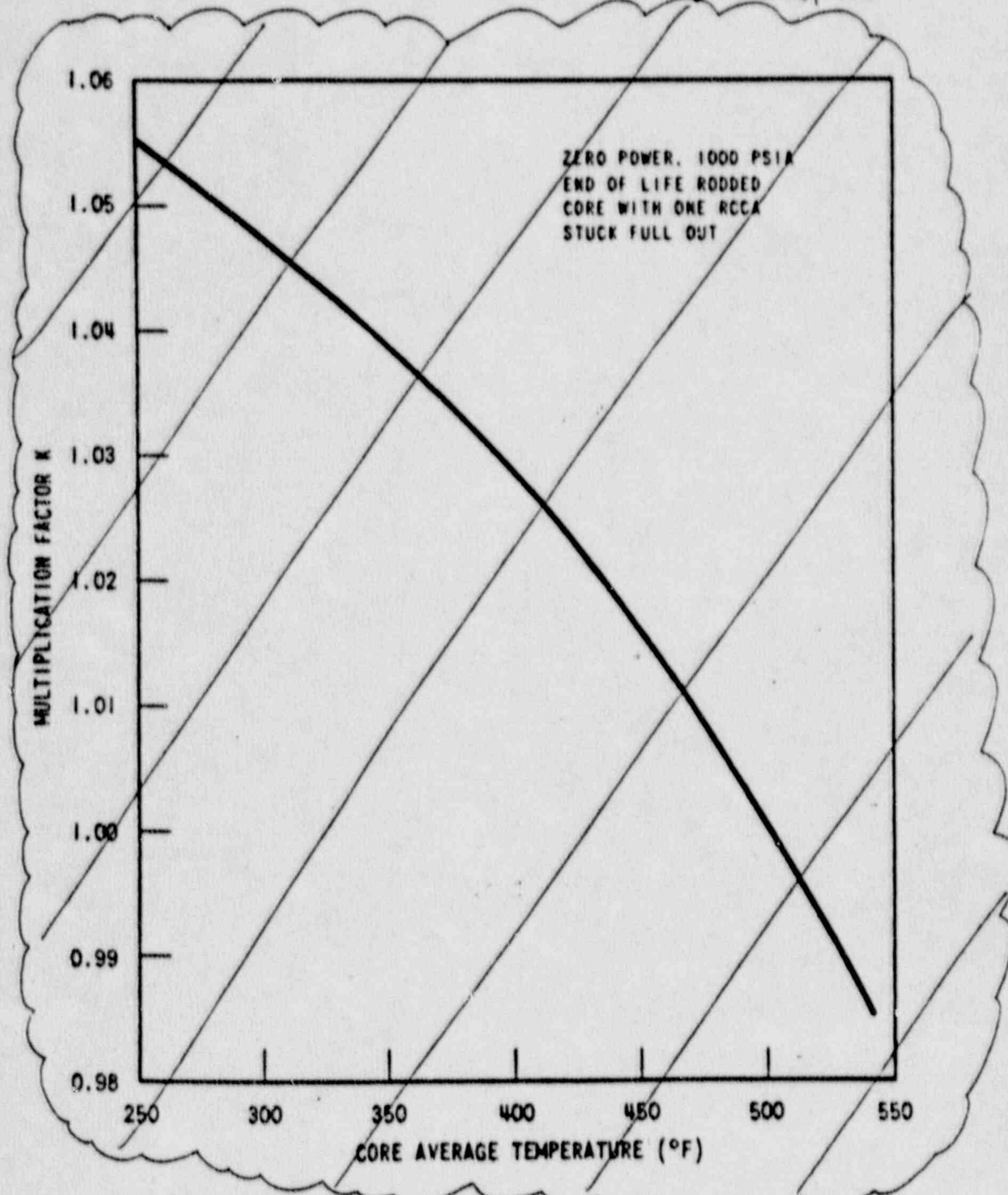


Figure 15.2.13-1 Variation of K_{EFF} with Core Temperature

DCN No. MD15334
Page____

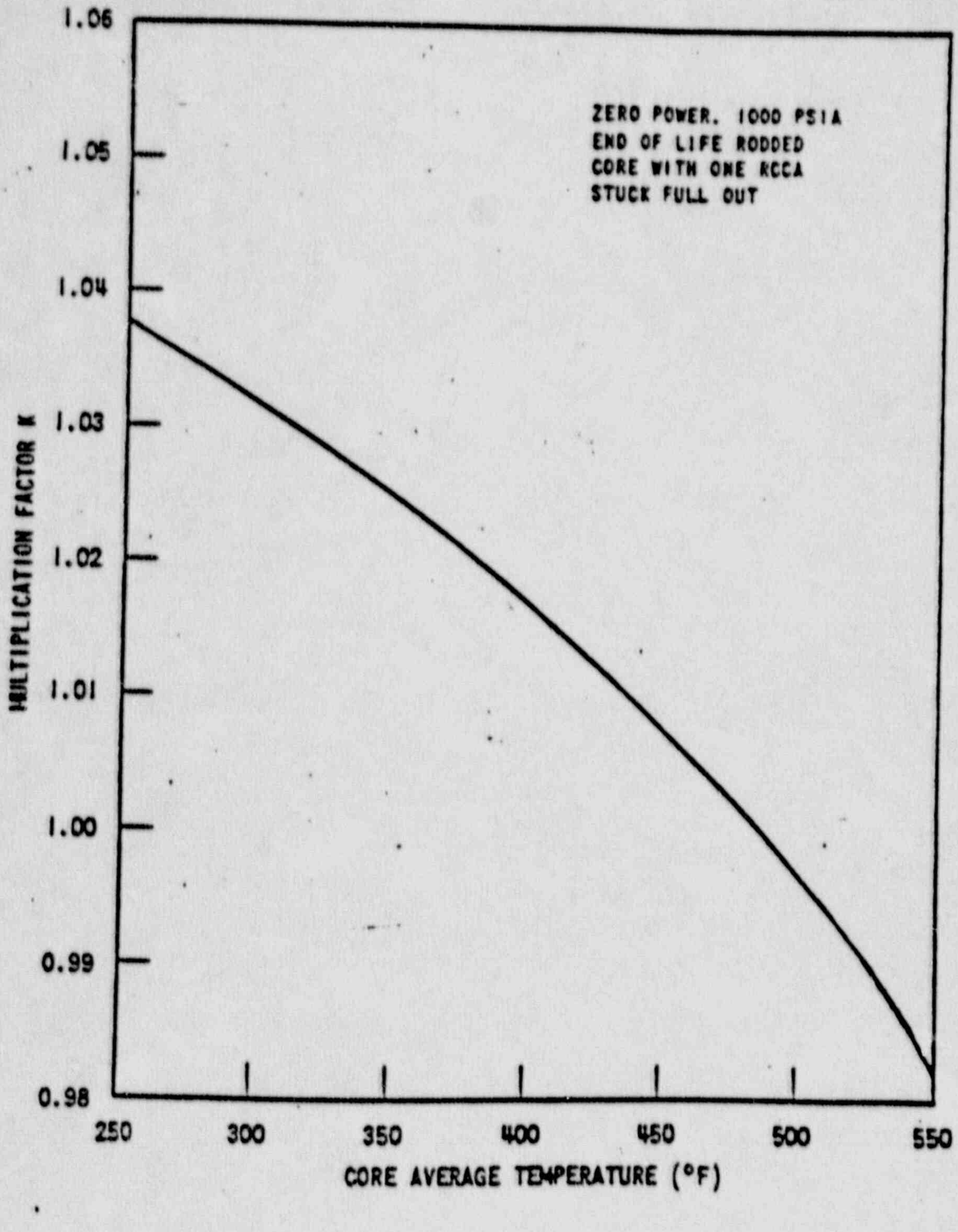


Figure 15.2.13-1 Variation of K_{eff} with Core Temperature

Delete and Replace

7090-4

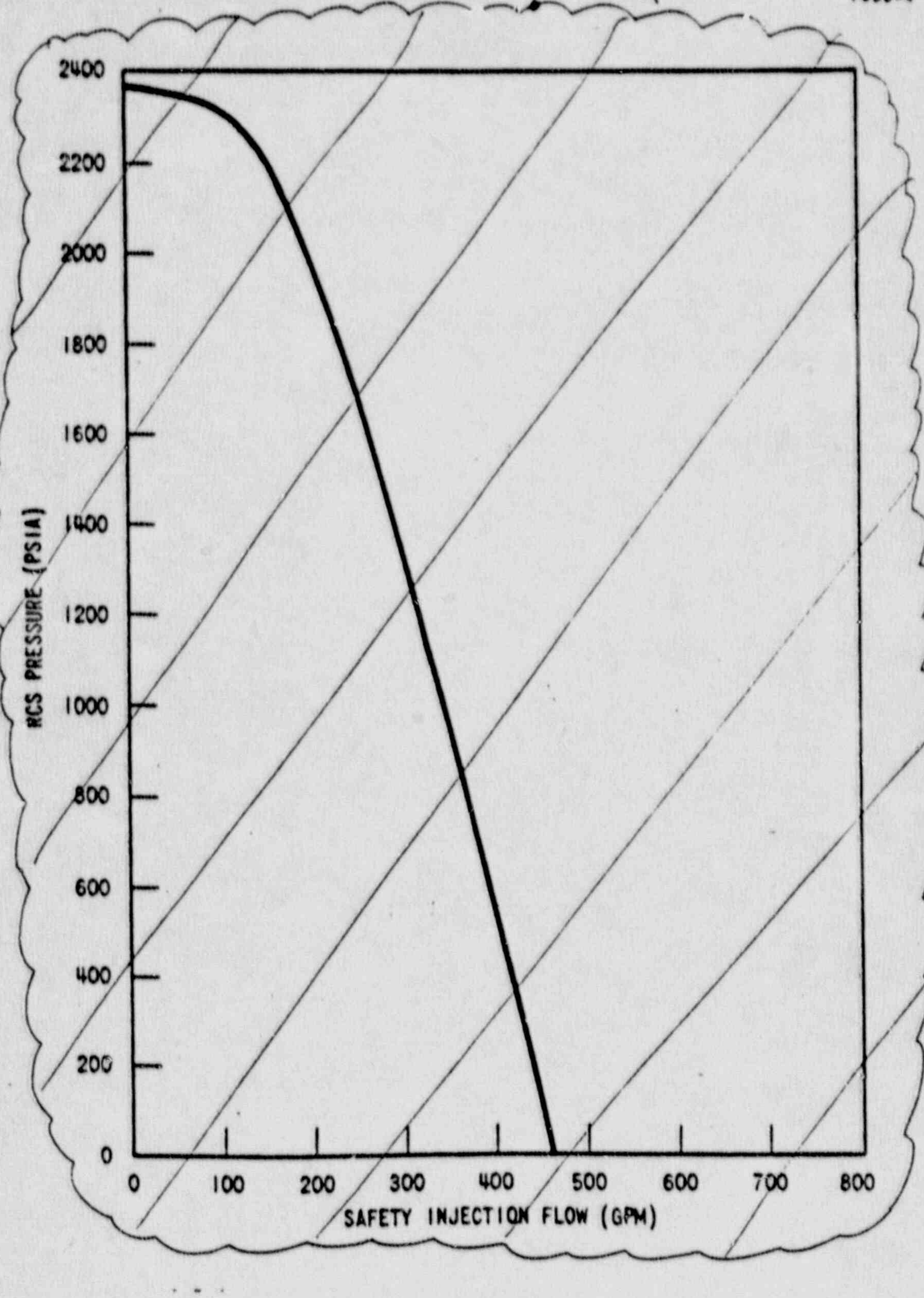


Figure 15.2.13-2 Safety Injection Curve

DCN No. M06533A
Page____

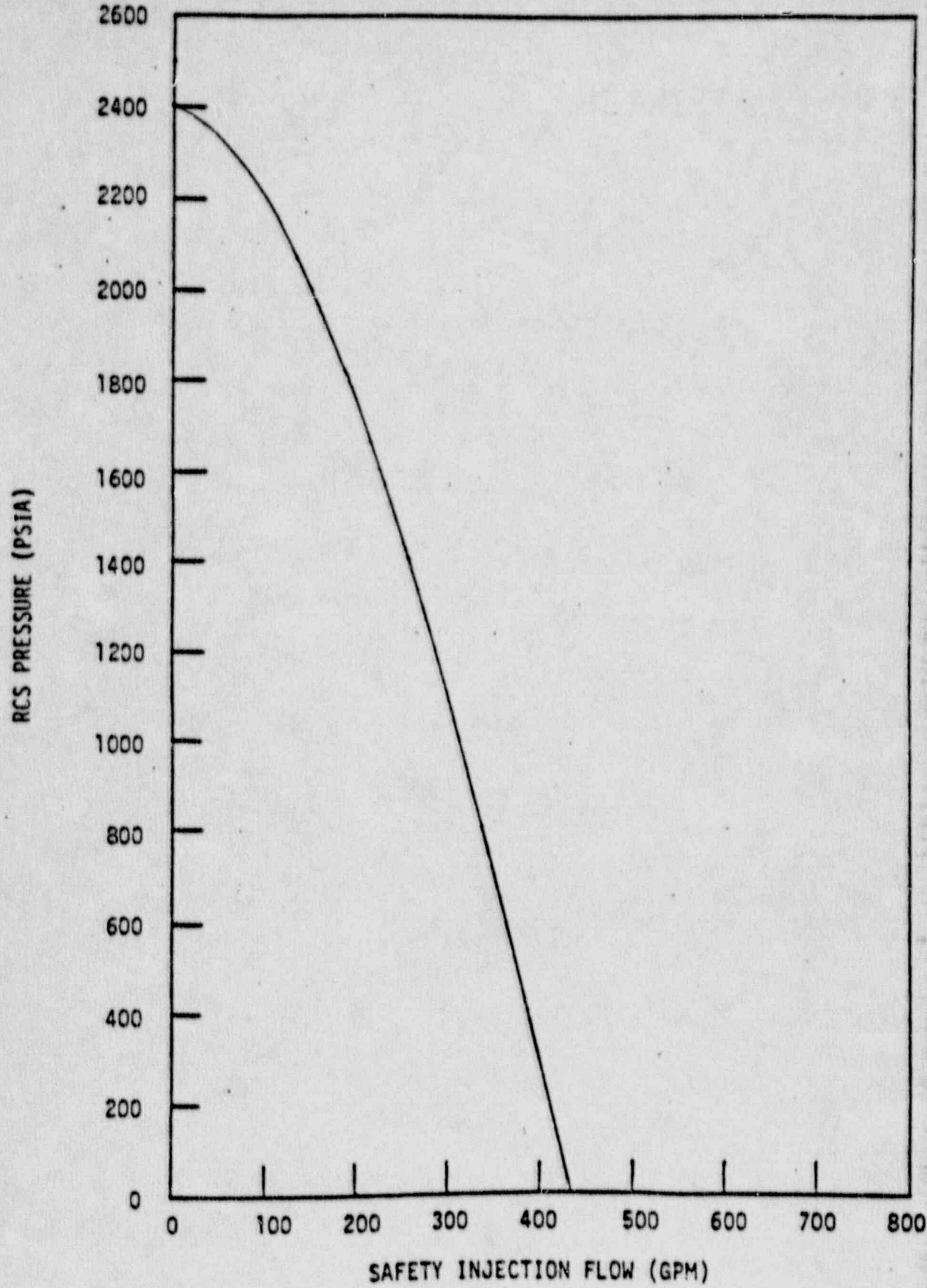


Figure 15.2.13-2 Safety Injection Curve

Delete
and Replace

DCN No. M10153311
Page____

7635-35

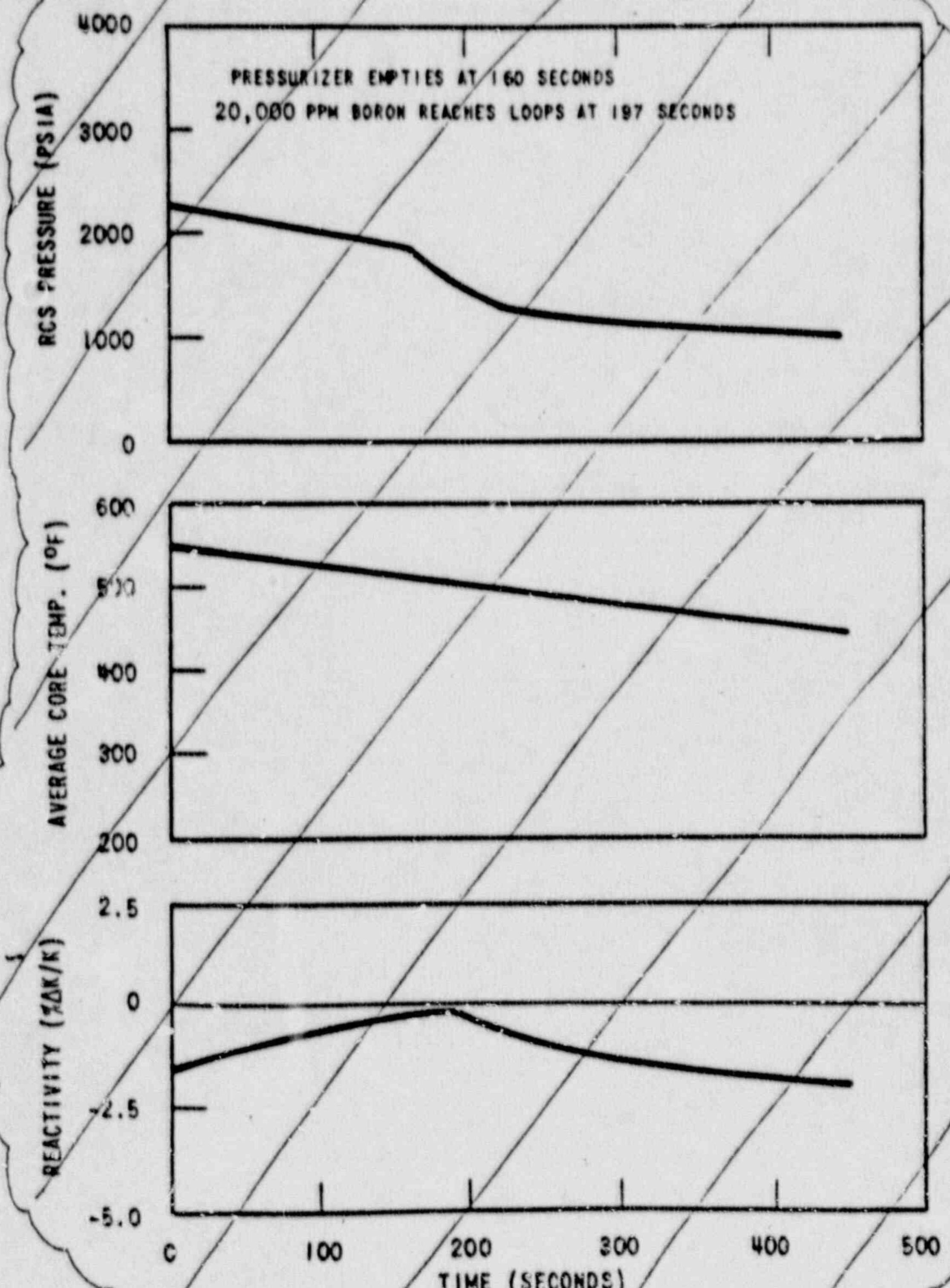


Figure 15.2.13-3 Transient Response for a Steam Line Break Equivalent to 228 Lbs/Sec at 1015 PSIA with Outside Power Available

SQN

JH 1/1/84

DCN No. M-5334

Page ____

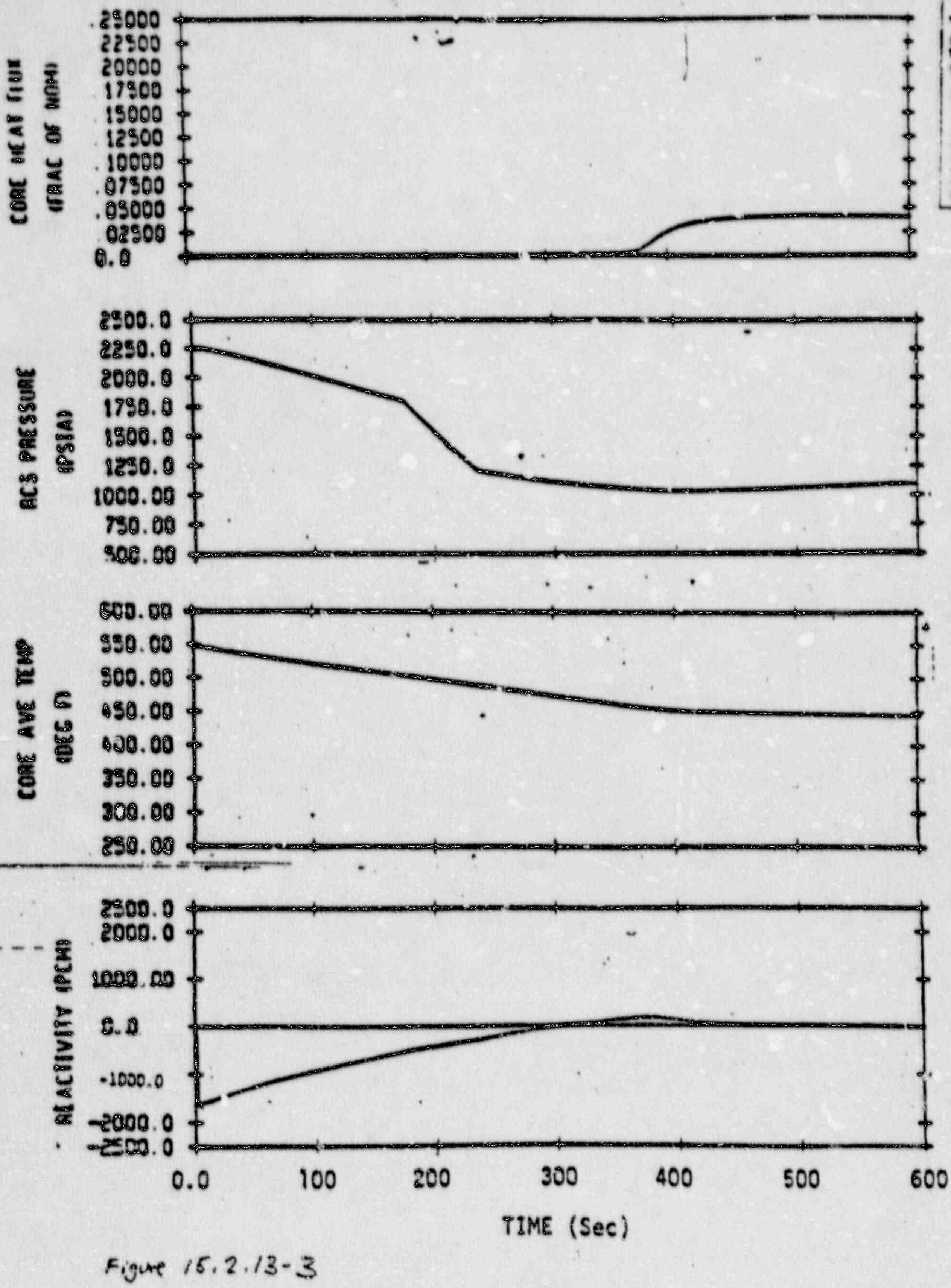


Figure 15.2.13-3

~~FIGURE 6~~ TRANSIENT RESPONSE FOR A STEAMLINE BREAK EQUIVALENT TO 228 LBS./SEC. AT 1015 PSIA WITH OUTSIDE POWER AVAILABLE.

JH 8/8/89

DCN No. [illegible]
Page___

TABLE 15.2-1 (Sheet 6)
(Continued)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time (Sec.)
Excessive Load Increase		
1. Manual Reactor Control (BOL)	10% step load increase	0
	Equilibrium conditions reached (approximate times only)	200
2. Manual Reactor Control (EOL)	10% step load increase	0
	Equilibrium conditions reached (approximate times only)	50
3. Automatic Reactor Control (BOL)	10% step load increase	0
	Equilibrium conditions reached	(3)
4. Automatic Reactor Control (EOL)	10% step load increase	0
	Equilibrium conditions reached (approximate times only)	50
Accidental Depressurization of the Reactor Coolant System	Inadvertent Opening of one RCS Safety Valve	0
	Reactor Trip	29.3
	Minimum DNBR occurs	31.5
Accidental depressurization of the Main Steam Safety System	Inadvertent Opening of one main steam safety or relief valve	0
	Pressurizer Empties	~~160~~ 187
	~~20,000 ppm~~ boron reaches CORE ~~RCS loops~~	~~197~~ 257
	UHI initiation time	~~221~~ 281

REVISE

(3) Did not reach equilibrium within the time scale of Figure 15.2.11-2

3

Revised by Amendment 3

DCN No. ML1533A
Page ___

TABLE 15.2-1 (Sheet 7)
(Continued)

TIME SEQUENCE OF EVENTS FOR
CONDITION II EVENTS

Accident	Event	Time (Sec.)
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
	~~Condition IV events~~	
~~Major Secondary System Pipe Rupture~~		
~~1. Case a~~	~~Steam line ruptures~~	~~0~~
	~~Criticality attained~~	~~18~~
	~~Pressurizer empty~~	~~15~~
	~~20,000 ppm boron reaches loops~~	~~20~~
	~~UHI initiation time~~	~~16~~
~~1. Case b~~	~~Steam line ruptures~~	~~0~~
	~~Criticality attained~~	~~14~~
	~~Pressurizer empty~~	~~17~~
	~~20,000 ppm boron reaches loops~~	~~21~~
	~~UHI initiation time~~	~~25.5~~
~~1. Case c~~	~~Steam line ruptures~~	~~0~~
	~~Criticality attained~~	~~21~~
	~~Pressurizer empty~~	~~16~~
	~~20,000 ppm boron reaches loops~~	~~30~~
	~~UHI initiation time~~	~~17~~
~~1. Case d~~	~~Steam line ruptures~~	~~0~~
	~~Criticality attained~~	~~17~~
	~~Pressurizer empty~~	~~18~~
	~~20,000 ppm boron reaches loops~~	~~32~~
	~~UHI initiation time~~	~~39~~

Delete and Replace w 15.4.1-

~~Revised by Amendment 3~~

3

DCN No. D01533A
Page

Fast-acting isolation valves are provided in each steam line that will fully close within 10 seconds of a large break in the steam line. 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 inside the steam pipes. The nozzles which are of considerably smaller diameter than the main steam pipe are located inside the containment near the steam generators and also serve to limit the maximum steam flow for any break further downstream.

15.4.2.1.2 Analysis of Effects and Consequences

Method of Analysis

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

1. The core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break. The ~~MARVEL~~ LOFTRAN (Reference ~~21~~ 24) code has been used. Revise as shown

2. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital-computer calculation (THINC Code, Paragraph 4.4.3.1) has been used to determine if DNB occurs for the core conditions computed in (1) above. 6

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 assembly 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 the 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 effect of power generation in the core on overall reactivity is shown in Figure 15.4.2-1.

   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

DCN No. M01533A
Page

this method, the reactivity as well as the power distribution was checked for the statepoints shown on Table 15.4.2-1. 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 nonuniform 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 calculation including the above local effects for all statepoints in Table 15.4.2-1. This result verified conservatism, i.e., underprediction of negative reactivity feedback from power generation. 4

3. Minimum capability for injection of high concentration boric acid (approximately ~~20,000~~ 7250 ppm) solution corresponding to the most restrictive single failure in the safety injection system. The injection curve used is shown in Figure 15.2.13-2. 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 which must be swept from the safety injection lines downstream of the ~~boron injection tank isolation valves~~ RWST prior to the delivery of high concentration boric acid to the reactor coolant loops.

Revise as shown

Revise as shown

4. Four combinations of break sizes and initial plant conditions have been considered in determining the core power RCS 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 (upstream of the steam flow measuring nozzle) with the plant initially at no load conditions with offsite power available. 6

   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.~~

Revise as shown

For a steamline break inside containment, with a failure of an MSIV in another steamline to close, the steam generator connected to the MSIV will continue to release steam through any lines or valves that may be open downstream of the MSIVs or upstream of the failed MSIV. Normally, there are open lines to the main steam reheaters, turbine gland seals, main feedwater pumps, and possibly the turbine-driven auxiliary feedwater pump (steam for the auxiliary feedwater pump is drawn from two steamlines upstream of the MSIVs). During the

DCN No. 1101533H
Page____

steamline break, steam flow to the main feedwater pump turbines and the main steam reheaters will be terminated. The flow to the main feedwater pump turbines is terminated by stop valves which actuate automatically on receipt of a safety injection signal. The flow to the reheaters becomes negligibly small because the reheaters are the condensing type. Main steam flow which condenses the reheat steam ceases when the high pressure turbine stop valves close, and the reheaters effectively become a water trap. The remaining steam flow amounts to about 20,000 lbs/hr., or less than .2 percent of nominal steam flow. In order to encompass any additional steam release through unidentified lines and drains, and also to noticeably perturb the steambreak results, this additional steam release was conservatively assumed to be more than 100,000 lbs/hr. Even with this high value for additional steam release, the steambreak analysis results were not significantly affected. The greatest deviation calculated was less than 0.02 percent in the peak core heat flux.

Since the steamline rupture causes the reactor coolant system to cooldown, there would be no reason (or signal) for the power-operated relief valves to open. These are fail-closed valves. Therefore, any postulated malfunction of a power-operated relief valve must be considered an independent failure and inconsistent with a coincident failure anywhere else (MSIVs). The case of spurious opening of a power- [illegible] large steamline break with following a large steamline break with subsequent closure of all MSIVs would be less severe than the steamline break case reported in the FSAR. The spurious opening of a secondary system valve, such as a power-operated relief valve, is considered separately and reported in Section 15.2.13.

The analyses presented do not consider additional steam blowdown from either of these sources. Steam released from open lines and drains on the secondary piping does not significantly affect the analysis results, and the failure of a power-operated relief valve is reported separately.

5. 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 limiting values ~~used for three~~ of the four steamline break accidents analyzed are given in Table 15.4.2-1. 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

* limiting case

Revise as shown

Revise as Shown

parameters ~~used~~ evaluated for each of the ~~three~~ cases correspond to values determined from the respective transient analysis. ~~Five time points are used 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 RCS 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 RCS 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 less for steam line breaks occurring at power.

6. In computing the steam flow during a steam line break, the Moody Curve (Reference 22) for $fL/D = 0$ is used.

7. Perfect moisture separation in the steam generator is assumed. The assumption leads to conservative results since, in fact, considerable water would be discharged. Water carryover would reduce the magnitude of the temperature decrease in the core and the pressure increase in the containment.

8. The Upper Head Injection System (UHI) is simulated. During a design steamline break accident, the reactor coolant system (RCS) pressure decrease may be large enough to actuate upper head injection (UHI). The injection flow rate is a strong function of the RCS pressure--the flow being higher for a lower RCS pressure.

   The UHI flow rates are based on the following model. The pressure drop (ΔP, lbf/ft^2) across a component is given by

$$\Delta P = K_1 \frac{\rho V^2}{2g_c}$$

DCN No. M01533A
Page

Where:

K_1 = loss coefficient (dimensionless)

ρ = fluid density (lbm/ft^3)

V = fluid velocity (ft/second)

g_c = 32.2 lbm-ft/lbf-second^2

Multiplying the right-hand side of equation (1) by $\rho A^2/\rho A^2$ gives

$$\Delta P = \frac{K_1 \rho^2 V^2 A^2}{2\rho\, g_c A^2}$$

4

or using = $\omega = \rho VA$ for mass low rate (lbm/sec), the pressure drop becomes

$$\Delta P = \frac{\omega^2}{K_2 \rho}$$

Where K_2 is a geometrical constant ($lbm\text{-}ft^5/lbf\text{-}sec^2$).

Solving for ω gives the following expression for the UHI system flow rate:

$$\omega = K\sqrt{\rho \Delta P}$$

The pressure drop used in the model is the difference between the UHI gas pressure and the RCS pressure. The density over a given time step is assumed to be constant at the value corresponding to the pressure at the beginning of that time step. The proportionality constant, K, is an input to the code.

The expansion of nitrogen over a time step is assumed to be isentropic. The change in nitrogen volume is calculated as:

$$V_{N2} = V_{N2o} + \frac{\bar{w}}{\rho} \Delta t$$

Revise as Shown

Where V_{N2o} is the volume at the beginning of the timestep and $\bar{W}$ is the average flowrate calculated during the timestep. The pressure is then calculated from:

$$P_N V_{N2}^{\gamma} = P_c V_{N2o}^{\gamma}$$

Where γ is 1.4 for nitrogen

DCN No. MO1533A
Page____

Revise as Shown

Since ~~MARVEL~~ LOFTRAN is not used in the analysis of LOCA, it does not have to deal with the high UHI flow rates induced by the severe depressurization of LOCA. The upper head of the reactor vessel remains full of subcooled water as it receives flow from the UHI accumulator.

In ~~MARVEL~~ LOFTRAN, the boron concentration and enthalpy are determined in the following manner:

$$X_{dv} = (WX_A\Delta t + X_{dv}M_{dv})/(W\Delta t + M_{dv})$$

Where X can be replaced by either H or B_c, W is the accumulator flow rate, and M_{dv} is the mass in the dead volume.

As stated in WCAP 8185 the significant effect of UHI is to retard the pressure decrease of the RCS. This in turn reduces the flow of borated water from the Safety Injection System. This potentially detrimental effect is compensated for by the boration provided by the UHI. 6

The RCS depressurizes and cools down as heat is removed via the assumed ruptured steamline. Depending upon the relative rates of temperature and pressure decline, flashing may occur in the RCS at locations other than the pressurizer. In a plant without UHI, the primary coolant system volume in which flashing will occur first is the upper head of the reactor vessel. The temperature in this region tends to be higher than the temperature in other regions, which experience higher coolant flow rates.

Water in the upper head of the Sequoyah reactor vessel does not flash during the steamline rupture. If a steamline is assumed to rupture when the plant is in a hot shutdown condition and the coolant temperature in the reactor vessel upper head is assumed to remain at its initial no-load value (547°F), then the reactor coolant system would have to depressurize to 1020 psia (saturation pressure at 547°F) from 2250 psia before any quality would be observed in the upper head. Meanwhile, the UHI system is conservatively assumed to add cold water into the reactor vessel upper head when the primary system pressure drops below the conservatively assumed 1300 psia setpoint. 6

Revise as shown

The ~~MARVEL~~ LOFTRAN code calculates the mass and energy of the fluid in the upper head based upon the incoming UHI flow and enthalpy, enthalpy and flow from the lower regions of the reactor vessel, and any heat input from the vessel walls. In the ~~MARVEL~~ LOFTRAN model, the UHI water flows into the node representing the upper head of the reactor vessel, where it mixes with the resident reactor coolant (see Figure 15.4.2-6). 4

The UHI system prevents flashing in the reactor coolant system during a steamline break by cooling the upper head region and by adding mass to the primary system, which retards the depressurization.

DCN No. MO 533A
Page

Only the Pressurizer water flashes and this void volume is easily determined from reported plots of the pressurizer water volume history.

Any heat input to the reactor coolant tends to retard the cooldown resulting from the steamline rupture and thereby mitigate the adverse effects of the accident. If the core returns to power, it does not reach as high a power level as it would have reached if the heat input were not accounted for. Heat addition does not significantly diminish the margin of subcooling since it retards the depressurization as well as the cooldown.

Heat transfer from the hot walls to the fluid in the upper head and the pressurizer is very small. Both regions are outside the active circulation path of the coolant. The pressurizer is filled with saturated steam and water. The water remains at the saturation enthalpy as it flows out of the pressurizer during the steambreak cooldown. The water saturation drops as the pressurizer empties.

The total temperature decline is about 50°F based upon the depressurization during the outsurge. On the average, the temperature drop is about 25°F and the heat transfer area is half of the initial area (at no load the water inventory is about 25 percent). Therefore, the heat transfer to water is small, due to the low ΔT and heat transfer area, and this heat input to water, however small, would be used for flashing anyway. This would produce more steam, which tends to retard the depressurization. Once the pressurizer is empty, heat transfer from ~~the steam~~ pressurizer walls to steam is very poor.

The heat transferred to the water from the metal in the rest of the primary system is much greater than the heat contributed by the pressurizer walls.

When UHI is added, the water in the upper head is cooled and remains below the saturation temperature. Heat transfer from the reactor vessel head is greater in this case, but small compared to the cooling effect of the UHI water, and the heat addition to the primary coolant from the steel walls in the other regions of the primary system which are also neglected in the FSAR analysis. When all these heat sources are considered, the cooldown and consequences of the steambreak are significantly reduced.

The maximum UHI flow rate will occur during a major loss of coolant accident and will amount to about 3000 lbs/second. The UHI flow rate calculations are described in WCAP-847939. The maximum UHI flow rate during a steamline break is a small fraction of the UHI flow rate during a LOCA, rarely exceeding 10 percent of the LOCA flow. The average UHI flow during the first 200 seconds of a steam break is about 50 to 60 lbs/second. 6

The UHI flow rate is based upon the pressure drop between the UHI system and the reactor coolant system.

DCN No. W0 5334
Page ____

The core flow rate is a constant volumetric flow, and any void, if present, would affect the mass flow rate through changes in the average coolant density. When the reactor coolant pumps are running, the core flow rate exceeds the maximum UHI flow rate by more than a factor of 10. This is assuming the maximum UHI flow rate during a LOCA. Compared to the average UHI flow during a steam break, the core flow rate is more than 600 times greater.

The upper head ~~injection accumulator pressure may be set between 1185 and 1285 psig.~~ The pressure assumed in this analysis is ~~1300~~ 1185 psia, which is at the ~~high~~ low end of the setpoint range. ~~Therefore, the maximum difference between the UHI setpoint assumed in the analysis and the actual UHI setpoint will be 100 psi.~~ 4

Revise as Shown

Sensitivity studies were performed for the Sequoyah Nuclear Plant to determine the effect of raising or lowering the UHI setpoint assumed in the steam line rupture analysis. A high UHI setpoint results in a relatively early actuation of the UHI system during the reactor ~~coolant system depressurization caused by the steamline rupture.~~ This results in a relatively earlier injection of the UHI boron

Revise as Shown

The UHI addition ~~then would tend~~ tends to retard the reactor coolant system depressurization and thereby reduce the safety injection water delivered, due to the relatively higher backpressure.

New paragraph

The net result is that slightly higher peak power levels are attained following the return to criticality during a steam line rupture cooldown with a lower UHI pressure. Therefore, the assumption that the UHI accumulator pressure is at the low ~~high~~ end of the setpoint range is conservative.

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 15.4.2-2 shows the RCS 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 condition (Case A). The break assumed is the largest break which can occur anywhere outside the containment either upstream or downstream of the isolation valves. Offsite power is assumed available such 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 high differential pressure between any steam line and the remaining steam lines or by high steam flow signals in coincidence with either low-low RCS 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 acting isolation valves in the steam lines by the high steam flow signals in coincidence with either low RCS temperature or low steam line pressure. Even with the failure of one

DCN No. M01533A
Page____

valve, release is limited to no more than 10 seconds for the other steam generators while the one steam generator blows down. The steam line isolation valves are designed to be fully closed in less than 5 seconds after receipt of closure signal with no flow through them.

The steam flow on Figures 15.4.2-2 through 15.4.2-5 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 seconds.~~ Revise as shown

~~The assumption that all steam generators blowdown for the first 10 seconds is conservative with respect to the core reactivity transient and the mass and energy release. The 10 second value represents a very long time delay for the signal generation, transmittal, receipt and subsequent closure of the steamline isolation valves. The long time delay assures a conservatively large energy release.~~ 6

~~When considering possible voiding in the reactor coolant system and flow blockage, a long time delay for closure of the steamline isolation valves is not necessarily conservative. In order to check for possible voiding in the primary coolant system, a steamline break simulation was performed in which the isolation valves were assumed to close two seconds (rather than 10 seconds) after the break. No voiding, and therefore no flow blockage resulted. The degree of subcooling in the primary coolant system was not significantly affected. The calculated core heat flux tended to be lower. The long time delay (10 seconds) is used because the additional mass release has a bigger effect outside the NSSS than the early valve closure time has on the primary coolant temperature.~~ 6

As shown in Figure 15.4.2-2 the core attains criticality with the rod cluster control assemblies inserted (with the design shutdown assuming one stuck assembly) before boron solution of approximately ~~20,000~~ 1950 ppm Revise enters the RCS 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 boron solution can be injected to the RCS 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. 6

Revise

The calculation assumes the ~~20,000~~ 1950 ppm boric acid is mixed with, and diluted by the water flowing in the RCS prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the RCS and in the Safety Injection System. The variation of mass flow rate in the RCS due to water density changes is included in the calculation as is the variation of flow rate from the Safety Injection System, UHI and the accumulator 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. 6

DCN No. MO1533A
Page____

The accumulators provide an additional source ~~of borated water after the RCS pressure decreases to below 447 psia.~~ The* ~~integrated flow rate of borated water from the Safety Injection system~~ for each of the four cases analyzed is shown in Figure 15.4.2-7. * core boron concentration Revise as Shown

Figure 15.4.2-3 shows Case B a steam line rupture at the exit of a steam generator (upstream of the flow measuring nozzles) 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. 6

Figures 15.4.2-4 and 15.4.2-5 show the responses of the salient parameters for cases c and d respectively 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 ~~12~~ a total of 23 seconds to start the emergency diesel generator ~~and 12 seconds~~ to get the safety 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 RCS is reduced by the decreased flow in the RCS. For both these cases the peak core power remains well below the nominal full power value. 6 Revise as Shown

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 which have been sized to cover this condition.

Generic thermal and stress analyses and subsequent fracture mechanics analyses of reactor vessels have been performed for 4-Loop plants. These analyses were applied to a 4-Loop reactor vessel having material properties and end of life (40 years) accumulated fluence similar to the Sequoyah vessel. The fracture mechanics analysis utilized linear elastic fracture mechanics method in the evaluation of the reactor vessel integrity. The fracture mechanics analysis results show that the Reactor Vessel integrity under large Steamline Break conditions would be maintained over the design life of the vessel.

For long term cooling of a steamline break the operator is instructed to use the intact steam generators for the purpose of removing decay heat and plant stored energy. This is done by maintaining the steam generator narrow-range span. 6

Steam pressure from the steam generators is relieved by the steam dump system, secondary system atmospheric safety valves, or secondary system relief valves. The operator is instructed to terminate auxiliary feedwater flow to the faulted steam generator as soon as he determines which steam generator is faulted. As soon as an indicated water level returns to the pressurizer the operator is instructed to turn off the safety injection pumps and restrict the charging pumps as required.

DCN No. M01533A
Page______

can be met by simple switch actions by the operators, i.e., closing auxiliary feed discharge valves and stopping charging pumps and safety injection pumps. Thus, the required simple actions to limit the cooldown and depressurization can be easily recognized, planned and performed within ten minutes. For the longer time requirements for decay heat removal and plant cooldown the operator has time on the order of hours to respond. 1

The worst case condition for long term cooling following a steam line break is loss of offsite power with failure of one emergency power train, since the condition requires the greatest amount of operator action and the longest time to achieve cold shutdown. However, since the plant can be maintained safely at hot standby conditions for extended periods of time, there is no safety requirement which dictates rapid achievement of cold shutdown conditions.

With only onsite power available, the plant can be maintained in a safe hot standby condition using the intact steam generators by supplying feedwater with the auxiliary feedwater system, and venting steam through the secondary side, power-operated relief valves. The relief valves will be controlled to gradually reduce pressure and temperature as the core residual heat decays. If the relief valves are not available, the safety valves will be used for steam dump. In this case, the primary system pressure would be controlled such that adequate subcooling is maintained. 6 Primary system temperature would be maintained at that value necessary to lift the steam generator safety valves as necessary to match the decay heat from the core. This temperature would be approximately 553°F which corresponds to the lowest steam generator safety valve setpoint of 1064 psig. For either means of steam relief, the steam generator water level will be maintained within the span of the narrow range indicators.

The sequence of events is shown in Table 15.4.1-12.

Margin to Critical Heat Flux

Past experience in performing DNB analyses for steamline breaks for W cores has shown that Case B (inside break with offsite ~~power) is always worse than Case A (outside break with offsite power).~~ ~~This can also be seen by examining the state points presented in Table 15.4.2-1~~. Cases A and B generally have very similar temperatures and pressures, but Case B returns to ~~is at~~ a power level from 1-1/2 to 2 times* ~~greater than~~ Case A. ~~It is this higher power which makes case B the worse of the two.~~ 6 1

Revise as Shown

*the power level

ADD ▲→ ~~Generally, only four of the~~ state points ~~presented in Table 15.4.2-1 are~~ subjected to detailed nuclear and thermal-hydraulic analysis. ~~For Case B, the point with the highest power level is analyzed, since past experience has indicated this point is the one which will probably have the lowest DNBR. In addition, either the preceding or succeeding point (depending on the conditions) is analyzed.~~

▲ A complete set of the Steamline break transient statepoints are reviewed to determine the most limiting condition. The LIMITING

Revise as Shown

DCN No. M01533A
Page

~~For Case D (inside break with loss of offsite power), the point most likely to have the lowest DNBR is the point with the highest power/flow ratio. Usually, this point is the one with the highest power. As with Case B, either the preceding or succeeding point is also analyzed. Should any of the points analyzed result in DNBR's near 1.30, additional points may be analyzed to insure that the point with the minimum DNBR condition has been analyzed.~~

The point~~s~~ analyzed for this application had a DNBR~~'s~~ greater than 1.30. Thus, it is concluded that the minimum DNBR for a steam break is greater than 1.30.

Revise as Shown

The maximum linear heat rate for the most limiting steambreak case presented in the FSAR was less than ~~10~~ kW/ft, which is less than the linear heat rate which results in fuel melting. ~~There is no known failure mechanism associated with this peak linear heat rate.~~

15.4.2.2 <u>Major Rupture of a Main Feedwater Pipe</u>

15.4.2.2.1 <u>Identification of Causes and Accident Description</u>

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 shellside 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 feedwater 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. This case is covered by the evaluation in Subsection 15.2.8.)

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a RCS cooldown (by excessive energy discharge through the break), or a RCS heatup. Potential RCS cooldown resulting from a secondary pipe rupture is evaluated in Paragraph 15.4.2.1, "Rupture of a Main Steam Line." Therefore, only the RCS heatup effects are evaluated for a feedline rupture. 6

A feedline rupture reduces the ability to remove heat generated by the core from the RCS because of the following reasons:

1. Feedwater to the steam generators is reduced. Since feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip;

2. Liquid in the steam generator may be discharged through the break, and would then not be available for decay heat removal after trip;

3. The break may be large enough to prevent the addition of any main feedwater after trip.

DCN No. M01533A
Page____

TABLE 15.4.1-12 (Sheet 1)

TIME SEQUENCE OF EVENTS FOR
CONDITION IV EVENTS

Delete and Replace

Accident	Event	Time (Sec)
Major Secondary System Pipe Rupture		
1. Case a	Steam line ruptures	0
	Criticality attained	18
	Pressurizer empty	15
	20,000 ppm boron reaches loops	20
2. Case b	Steam line ruptures	0
	Criticality attained	14
	Pressurizer empty	17
	20,000 ppm boron reaches loops	21
3. Case c	Steam line ruptures	0
	Criticality attained	21
	Pressurizer empty	16
	20,000 ppm boron reaches loops	30
4. Case d	Steam line ruptures	0
	Criticality attained	17
	Pressurizer empty	19
	20,000 ppm boron reaches loops	32

4
4
4
4

~~TABLE 2~~
Table 15.4.1-12

DA 8/7/81

TIME SEQUENCE OF EVENTS

DCN No. M01533A
Page ____

Major Secondary System Pipe Rupture

1. Case a	Steam line ruptures	0
	Pressurizer empty	13.4
	UHI initiation time	21.7
	Criticality attained	30.8
	Boron reaches core	30.8
2. Case b	Steam line ruptures	0
	Pressurizer empty	15.0
	Criticality attained	19.8
	UHI initiation time	23.0
	Boron reaches core	31.8
3. Case c	Steam line ruptures	0
	Pressurizer empty	14.6
	UHI initiation time	24.1
	Criticality attained	35.3
	Boron reaches core	47.3
4. Case d	Steam line ruptures	0
	Pressurizer empty	16.5
	Criticality attained	23.3
	UHI initiation time	28.4
	Boron reaches core	52.3

Accidental depressurization of the Main Steam System

	Inadvertent Opening of one main steam safety or relief valve	0
	Pressurizer empties	161
	Boron reaches core	227
	UHI initiation time	237
	Criticality attained	305

DCN No. M01533A
Page____

SQN

Delete and Replace

TABLE 15.4.2-1 (Sheet 1)

CORE PARAMETERS USED IN STEAM BREAK DNB ANALYSIS

Parameter	Case a, Time Point				
	1	2	3	4	5
Reactor Vessel inlet temperature to sector connected to affected Steam Generator, °F	436.9	434.6	411.4	405.0	399.7
Reactor Vessel inlet temperature to remaining sector, °F	492.8	491.1	486.0	481.3	475.9
RCS pressure, psia	1143.0	1117.0	1077.0	1049.8	1023.5
RCS flow, % 100	100	100	100	100	
Heat flux, % 6.91	7.45	7.22	6.83	6.37	
Time, sec.	32.5	41.0	55.0	65.0	75.0

DCN No. M01533A
Page ____

Delete

TABLE 15.4.2-1 (Sheet 2)
(Continued)

CORE PARAMETERS USED IN STEAM BREAK DNB ANALYSIS

Parameter	Case b, Time Point 1	2	3	4	5
Reactor Vessel inlet temperature to sector connected to affected steam generator, °F	382.6	368.4	363.5	355.1	351.4
Reactor Vessel inlet temperature to remaining sector, °F	521.8	505.1	497.9	482.7	475.4
RCS pressure, psia	1245.0	1107.8	1070.6	984.7	954.7
RCS flow, % 100	100	100	100	100	
Heat flux, % 9.67	10.79	10.98	10.3	9.96	
Time, sec.	30.0	45.0	52.5	67.5	75.0

DCN No. M01533A
Page ____

SQN

Delete

TABLE 15.4.2-1 (Sheet 3)
(Continued)

CORE PARAMETERS USED IN STEAM BREAK DNB ANALYSIS

Parameter	Case d, Time Point 1	2	3	4	5
Reactor Vessel inlet temperature to sector connected to affected steam generator, °F	375.1	350.1	330.3	318.5	305.5
Reactor Vessel inlet temperature to remaining sector, °F	529.7	528.6	528.1	527.5	526.7
RCS pressure, psia	1524.0	1348.6	1277.5	1256.7	1229.0
RCS flow, % 40.6	32.2	27.0	24.2	21.4	
Heat flux, % 5.88	6.83	6.19	5.4	4.67	
Time, sec.	25.0	35.0	45.0	52.5	62.5

CH 10/7/89

5?IV

DCN No. M01533A
Page____

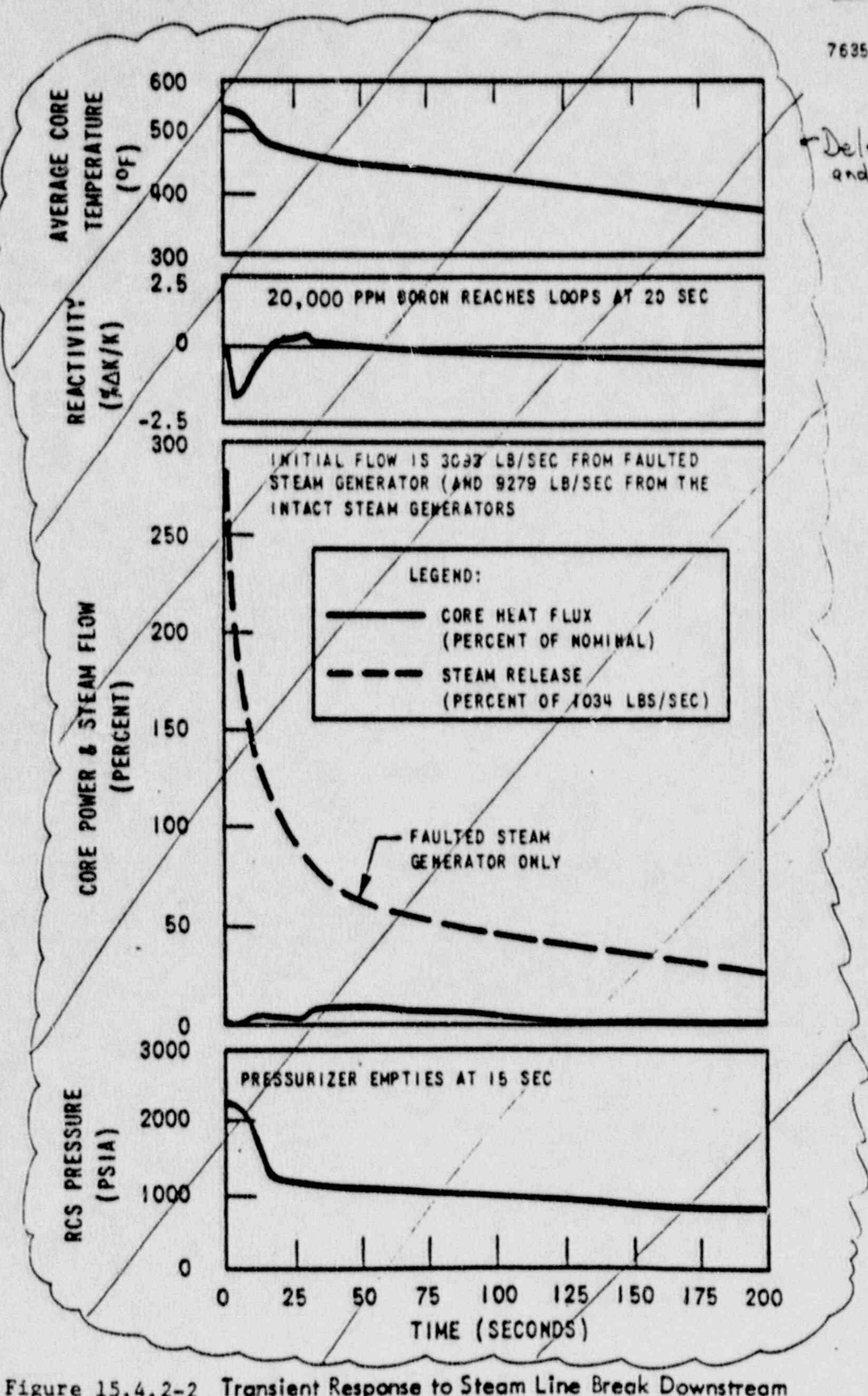
~~TABLE 1~~ SH 8/7/89
Table 15.4.2-1

LIMITING CORE PARAMETERS USED IN STEAM BREAK DNB ANALYSIS

Case	Inside break with power (case b)
Reactor vessel inlet temperature	319.3°F (Faulted SG Loop) 414.2°F (Intact SG Loops)
RCS pressure	798.52 psia
RCS flow	100% (of nominal)
Heat flux	17.60% (of nominal)
Time	212.75 seconds

DCN No. M01533A
Page ______

7635-37



Delete and Replace

Figure 15.4.2-2 Transient Response to Steam Line Break Downstream of Flow Measuring Nozzle with Safety Injection and Off-Site Power (case a)

JH 8/8/89

SQN

DCN No. M01533A
Page______

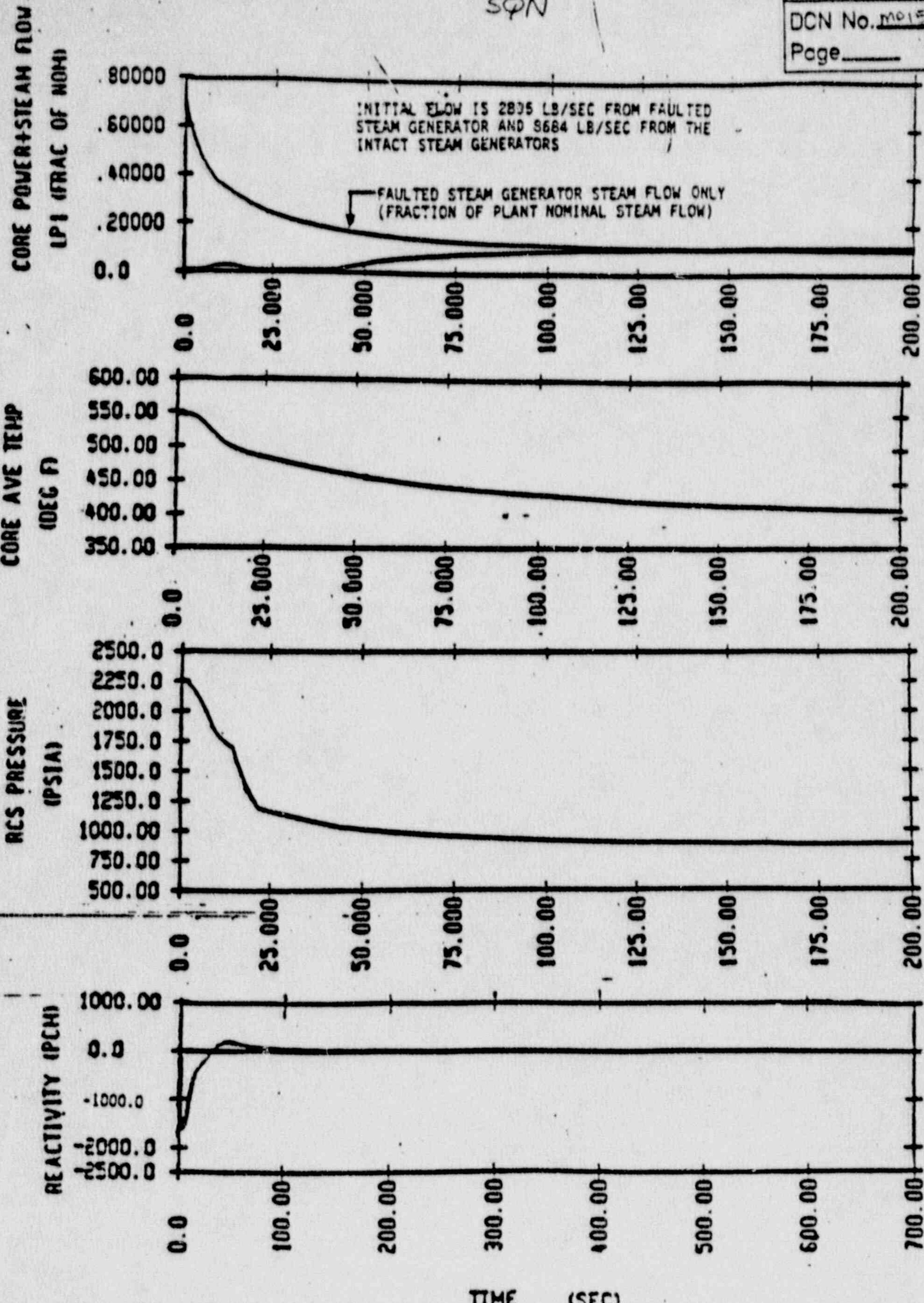


Figure 15.4.2-2

~~FIGURE 1~~ JH 8/7/89 TRANSIENT RESPONSE TO STEAMLINE BREAK DOWNSTREAM OF FLOW MEASURING NOZZLE WITH SAFETY INJECTION AND WITH OFF-SITE POWER (CASE A)

DCN No. M01533A
Page ____

7635-38

Delete and Replace

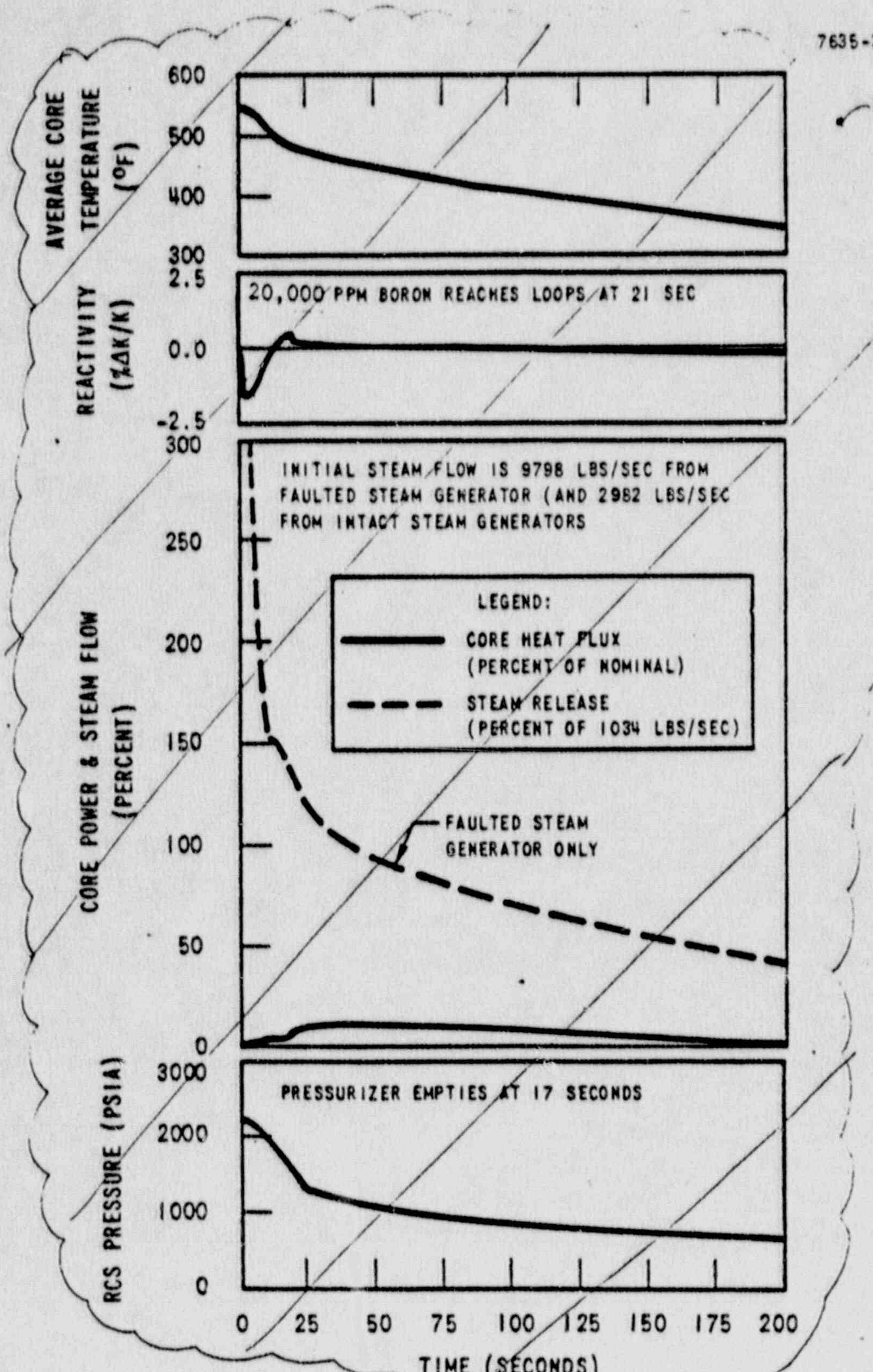


Figure 15.4.2-3 Transient Response to Steam Line Break at Exit of Steam Generator with Safety Injection and Off-Site Power (case b)

Revised by Amendment 2

DCN No.
Page

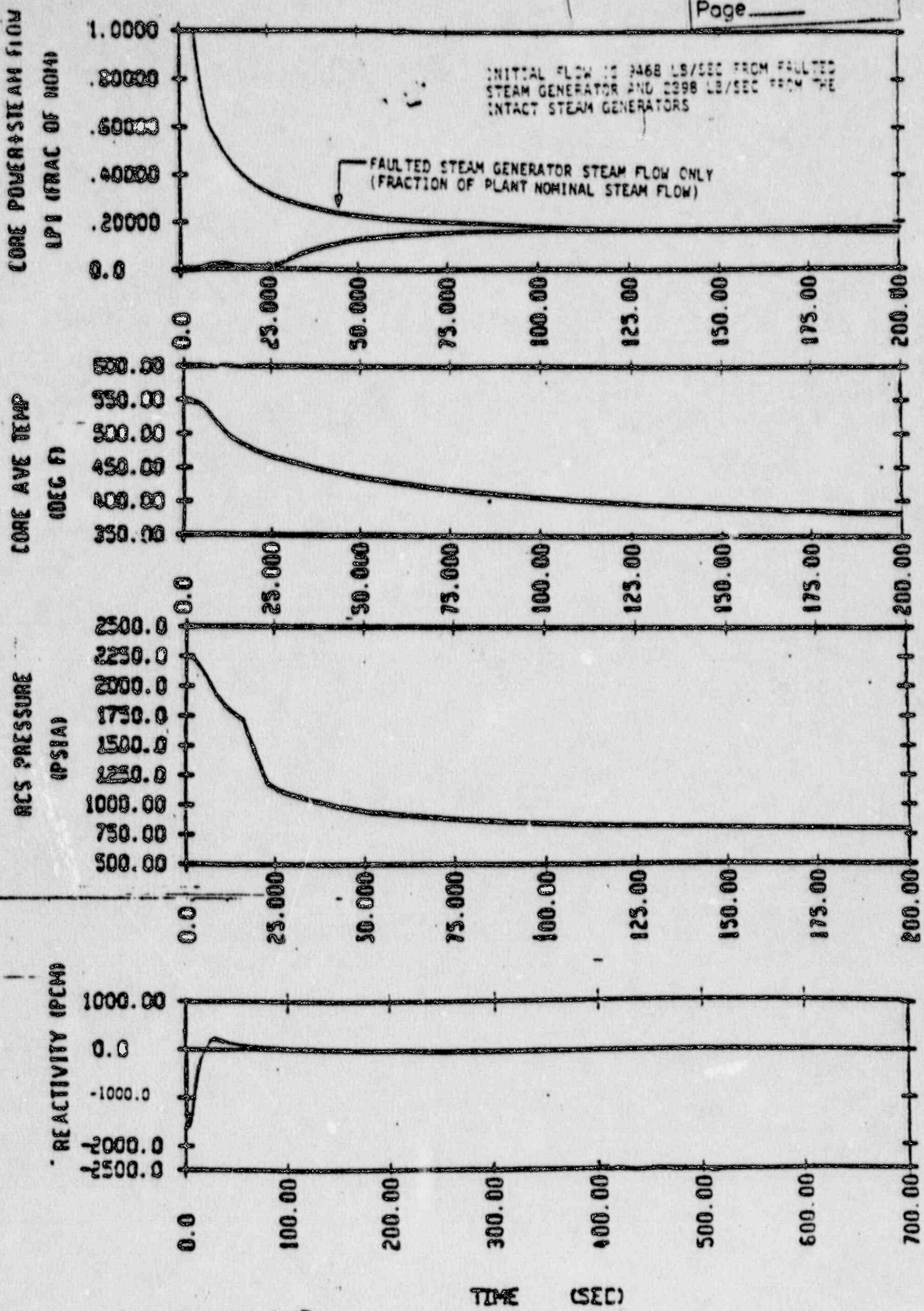


Figure 15.4.2-3

~~FIGURE 2~~ 8/7/89

TRANSIENT RESPONSE TO STEAMLINE BREAK AT EXIT OF STEAM GENERATOR WITH SAFETY INJECTION AND WITH OFF-SITE POWER (CASE B)

DCN No. MC-1533A
Page____

7635-39

Delete and Replace

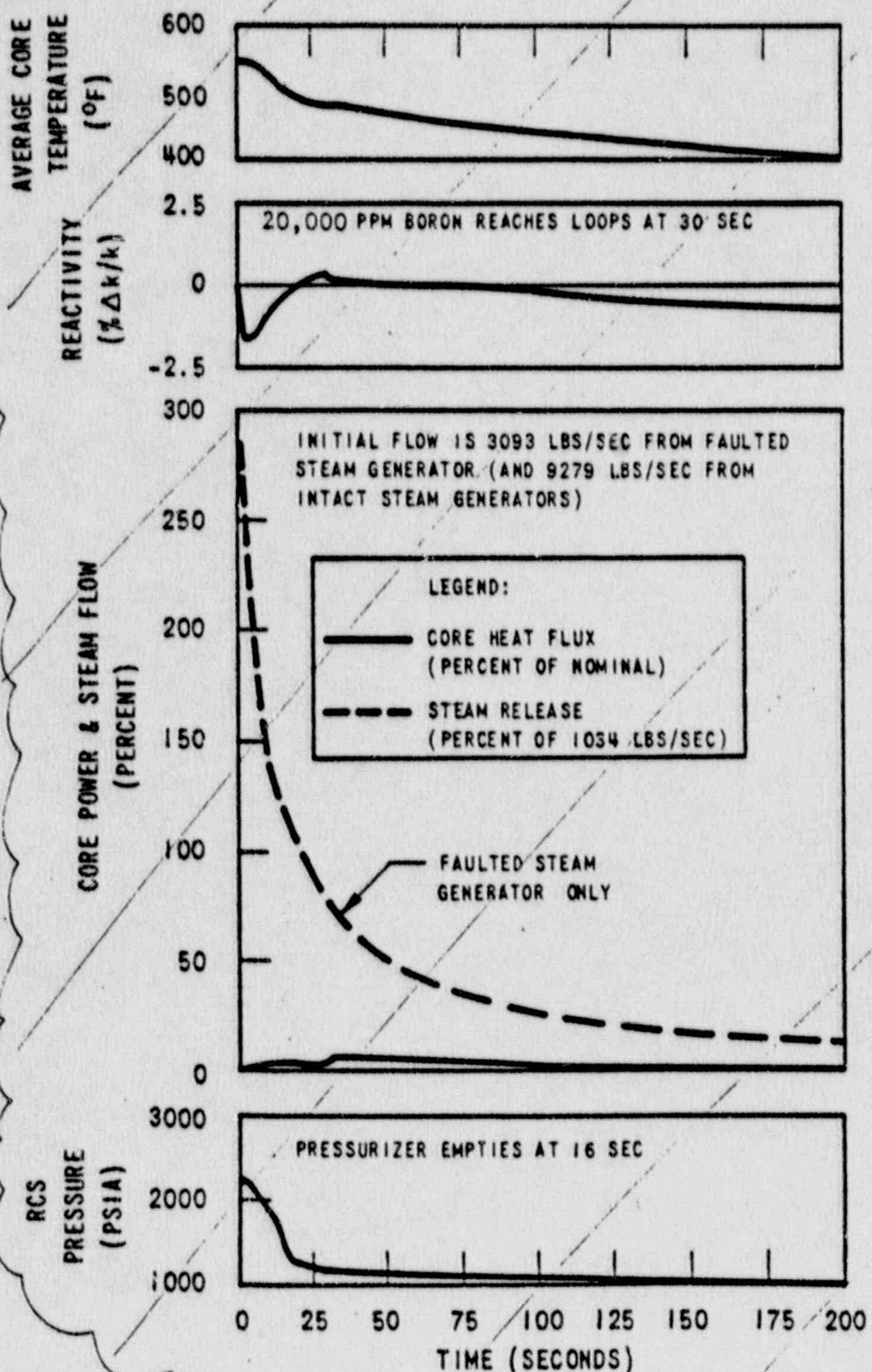


Figure 15.4.2-4 Transient Response to Steam Line Break Downstream of Flow, Measuring Nozzle with Safety Injection and Without Off-Site Power (case c)

Revised by Amendment 2

DCN No. M41533A
Page____

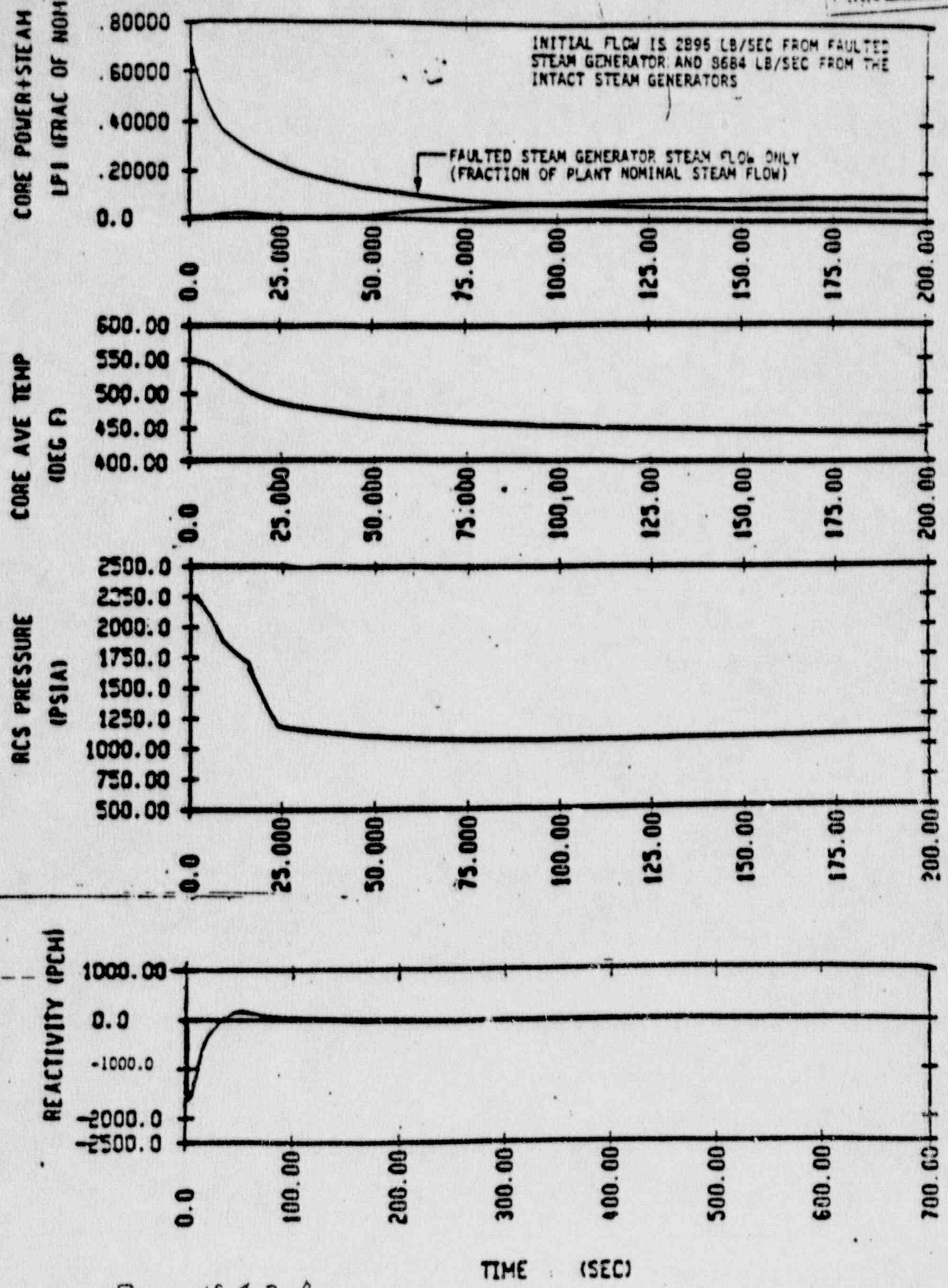


Figure 15.4.2-4

~~FIGURE 3~~
JH 8/17/89

TRANSIENT RESPONSE TO STEAMLINE BREAK DOWNSTREAM OF FLOW MEASURING NOZZLE WITH SAFETY INJECTION AND WITHOUT OFF-SITE POWER (CASE C)

DCN No. [illegible]1533A
Page____

7635-40

-Delete and Replace

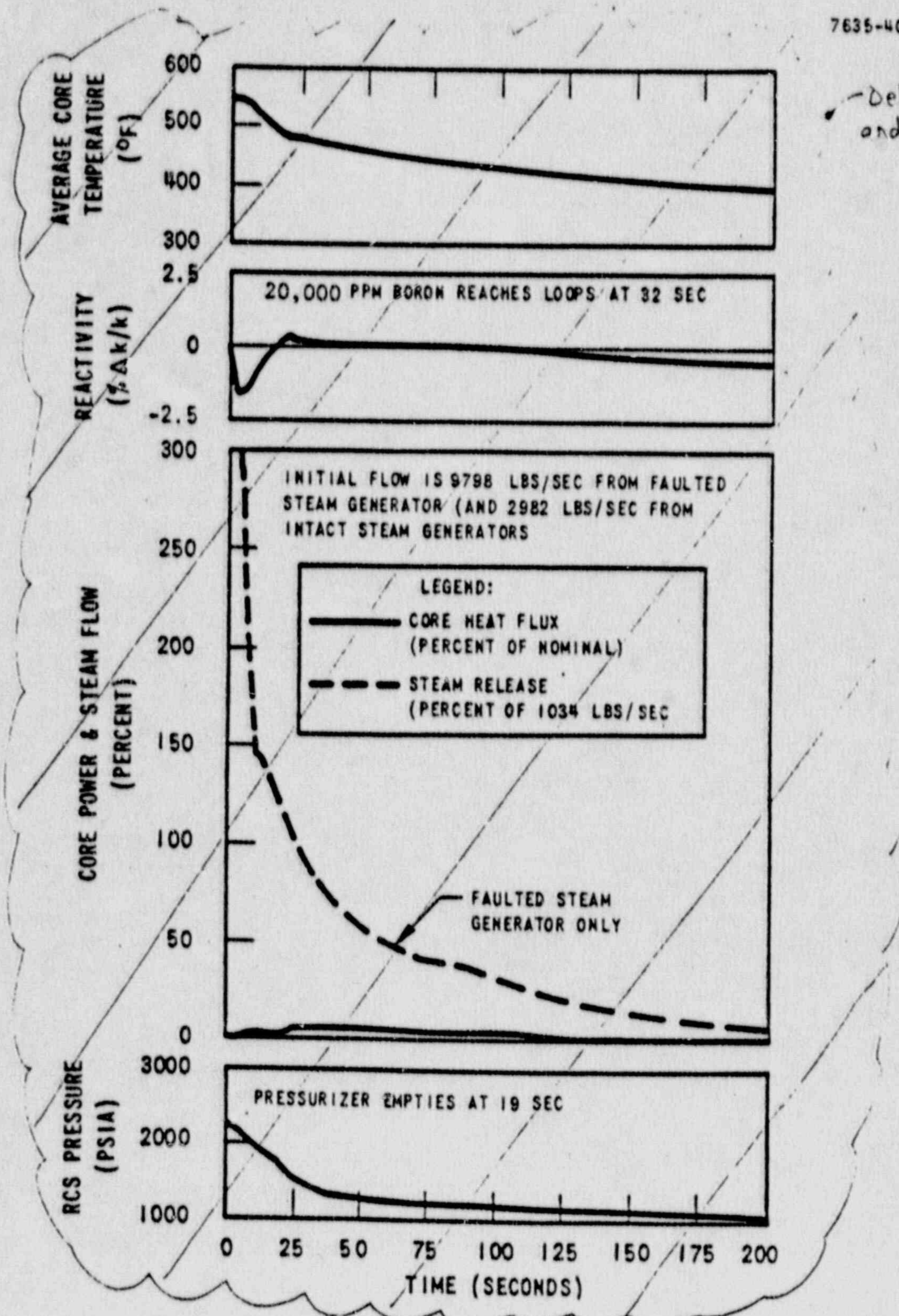


Figure 15.4.2-5 Transient Response to Steam Line Break at Exit of Steam Generator with Safety Injection and Without Off-Site Power (case d)

Revised by Amendment 2

SPN

JH 8/7/87

DCN No. M01533A
Page

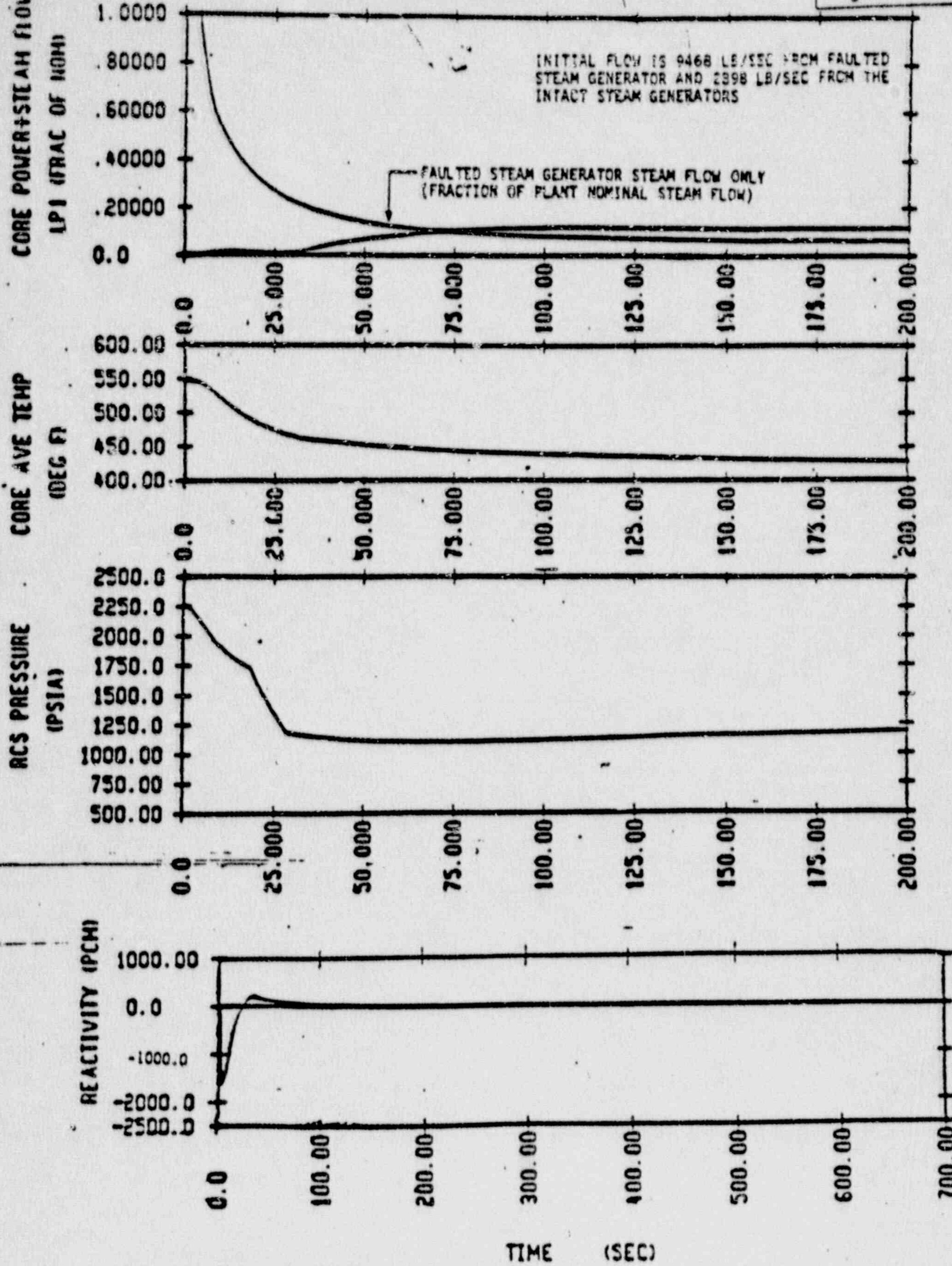


Figure 15.4.2-5

~~FIGURE 4~~ TRANSIENT RESPONSE TO STEAMLINE BREAK AT EXIT OF STEAM GENERATOR WITH SAFETY INJECTION AND WITHOUT OFF-SITE POWER (CASE D)

7635-41

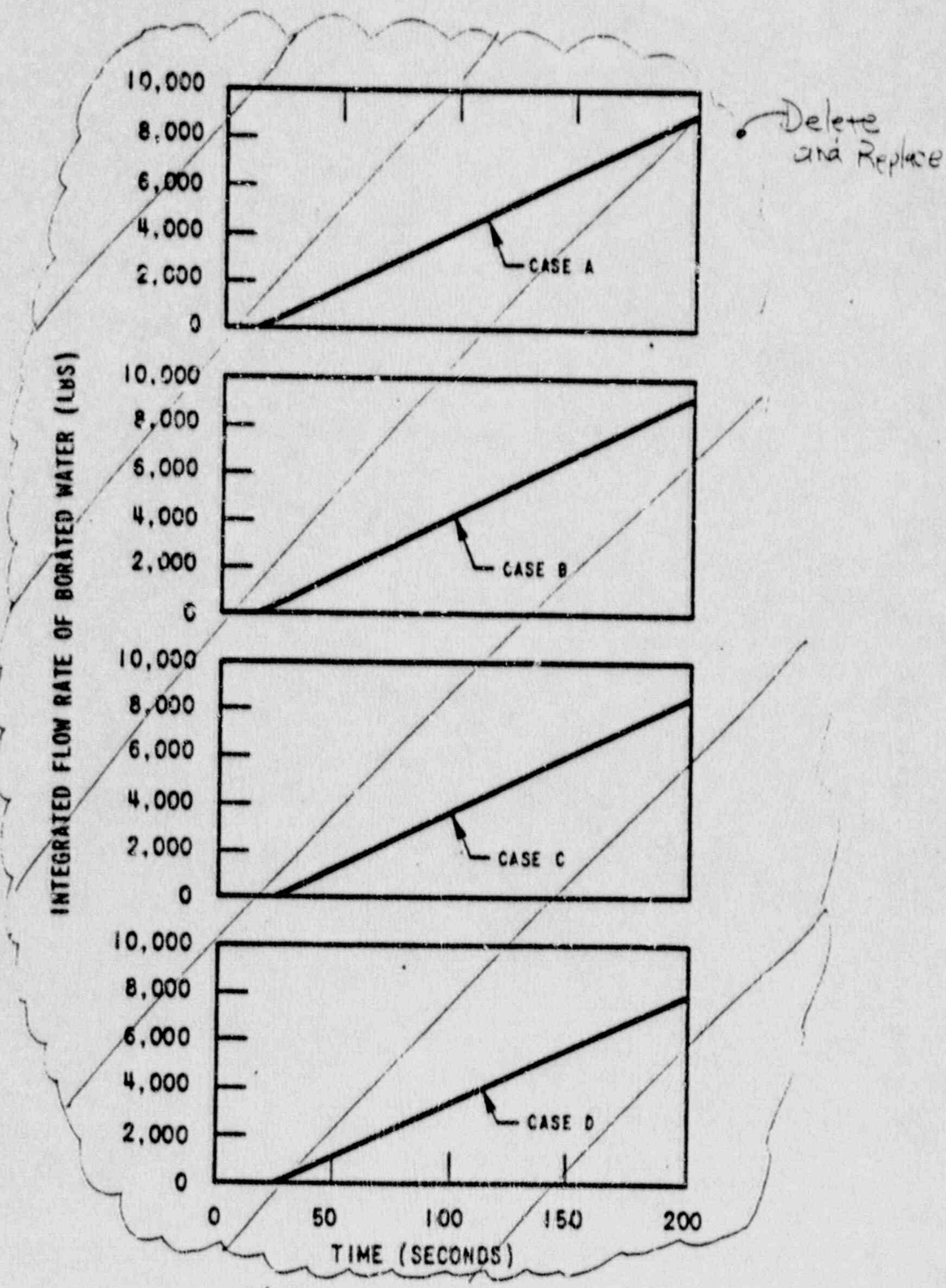


Figure 15.4.2-7 Integrated Flow of Borated Water versus Time

SQN

JH 8/7/89

DCN No. W15334
Page____

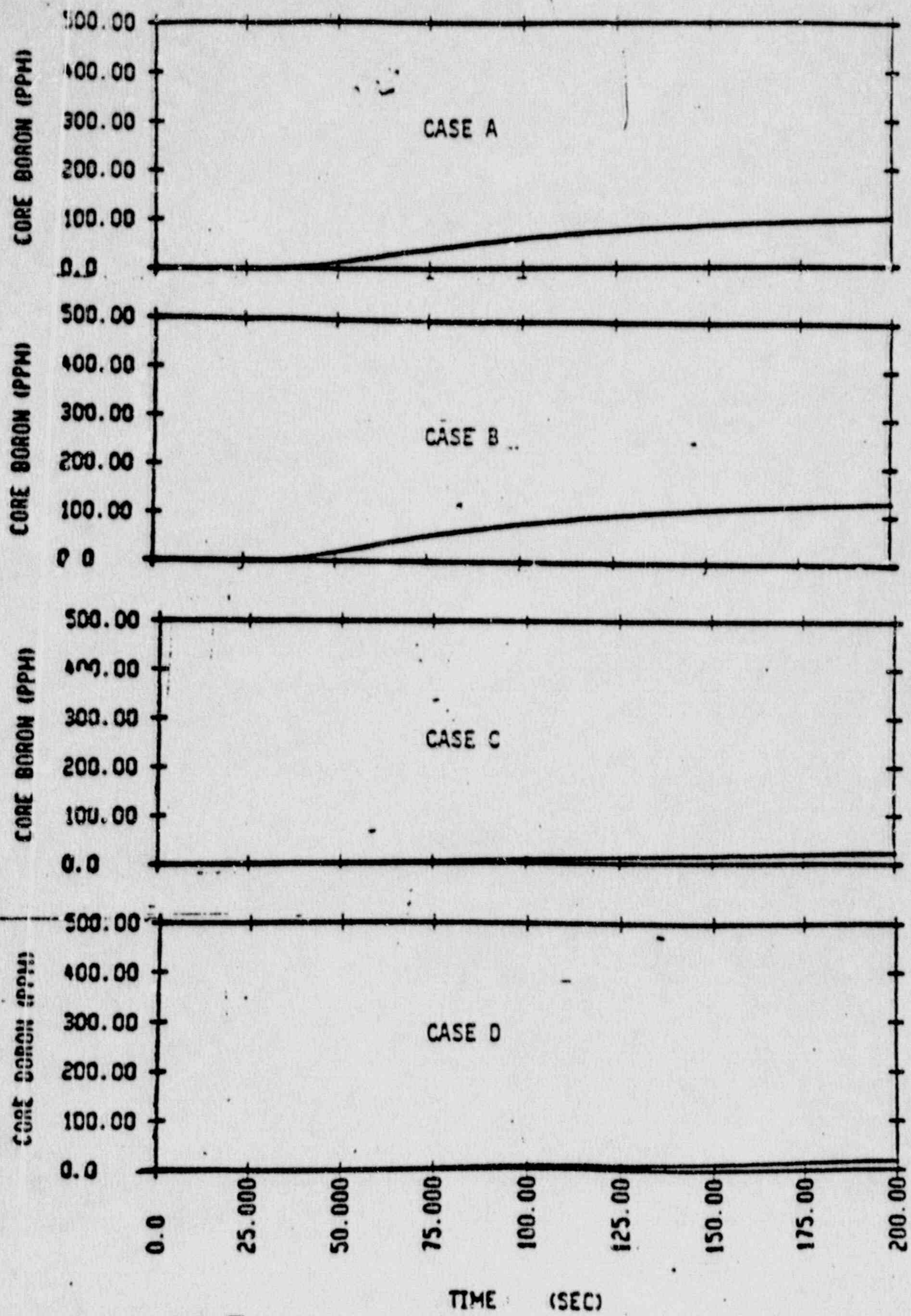


Figure 15.4.2-7

~~FIGURE 5~~ CORE BORON CONCENTRATION VERSUS TIME
JH 8/7/89