INDEX

DEFINITIONS

SECTION

1.0 DEFINITIONS

INDEX

DEFINITIONS

SECTION

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

K_{N-1} Insert (A)

1.16 K_{N-1} is the k-effective calculated assuming the fully or partia inserted full-length control element assembly of highest inserted wor fully withdrawn.

Insert (A)

MEMBER(S) OF THE PUBLIC

1.16 7 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.17 8 The OFFSITE DOSE CALCULATION MANUAL shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

1.18 9 A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.19 20 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and cold leg reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.20 1 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PLANAR RADIAL PEAKING FACTOR - F_{xy}

1.21 2 The PLANAR RADIAL PEAKING FACTOR is the ratio of the peak to plane average power density of the individual fuel rods in a given horizontal plane, excluding the effects of azimuthal tilt.

PRESSURE BOUNDARY LEAKAGE

1.22 3 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.2~~3~~4 The PROCESS CONTROL PROGRAM shall contain the provisions to assure that the SOLIDIFICATION of wet radioactive wastes results in a waste form with properties that meet the requirements of 10 CFR Part 61 and of low level radioactive waste disposal sites. The PCP shall identify process parameters influencing SOLIDIFICATION such as pH, oil content, H_2O content, solids content, ratio of solidification agent to waste and/or necessary additives for each type of anticipated waste, and the acceptable boundary conditions for the process parameters shall be identified for each waste type, based on laboratory scale and full-scale testing or experience. The PCP shall also include an identification of conditions that must be satisfied, based on full-scale testing, to assure that dewatering of bead resins, powdered resins, and filter sludges will result in volumes of free water, at the time of disposal, within the limits of 10 CFR Part 61 and of low level radioactive waste disposal sites.

PURGE - PURGING

1.2~~4~~5 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.2~~5~~6 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3800 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.2~~6~~7 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

REPORTABLE EVENT

1.2~~7~~8 A REPORTABLE EVENT shall be any of those conditions specified in Sections 50.72 and 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.2~~8~~9 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. No change in part-length control element assembly position, and

b. All full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.~~29~~30 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOFTWARE

1.~~30~~1 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation, and procedures.

SOLIDIFICATION

1.~~31~~2 SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a homogeneous (uniformly distributed), monolithic, immobilized solid with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

SOURCE CHECK

1.~~32~~3 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.~~33~~4 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.~~34~~5 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.~~35~~6 UNIDENTIFIED LEAKAGE shall be all leakage which does not constitute either IDENTIFIED LEAKAGE or reactor coolant pump controlled bleed-off flow.

UNRESTRICTED AREA

1.~~36~~7 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.3~~7~~8 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.3~~8~~9 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Logarithmic Power Level - High (1)		
a. Startup and Operating	≤ ~~0.798%~~ 0.010% of RATED THERMAL POWER	≤ ~~0.895%~~ 0.011% of RATED THERMAL POWER
b. Shutdown	≤ ~~0.798%~~ 0.010% of RATED THERMAL POWER	≤ ~~0.895%~~ 0.011% of RATED THERMAL POWER
C. Core Protection Calculator System		
1. CEA Calculators	Not Applicable	Not Applicable
2. Core Protection Calculators	Not Applicable	Not Applicable
D. Supplementary Protection System		
Pressurizer Pressure - High	≤ 2409 psia	≤ 2414 psia
II. RPS LOGIC		
A. Matrix Logic	Not Applicable	Not Applicable
B. Initiation Logic	Not Applicable	Not Applicable
III. RPS ACTUATION DEVICES		
A. Reactor Trip Breakers	Not Applicable	Not Applicable
B. Manual Trip	Not Applicable	Not Applicable

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) In MODES 3-4, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) In MODES 3-4, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below ~~1%~~ 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to ~~1%~~ 10^{-4}% of RATED THERMAL POWER.

The approved DNBR limit is 1.231 which includes a partial rod bow penalty compensation. If the fuel burnup exceeds that for which an increased rod bow penalty is required, the DNBR limit shall be adjusted. In this case a DNBR trip setpoint of 1.231 is allowed provided that the difference is compensated by an increase in the CPC addressable constant BERR1 as follows:

$$BERR1_{new} = BERR1_{old} \left[1 + \frac{RB - RB_o}{100} \times \frac{d\ (\%\ POL)}{d\ (\%\ DNBR)}\right]$$

where $BERR1_{old}$ is the uncompensated value of BERR1; RB is the fuel rod bow penalty in % DNBR; RB_o is the fuel rod bow penalty in % DNBR already accounted for in the DNBR limit; POL is the power operating limit; and d (% POL)/d (% DNBR) is the absolute value of the most adverse derivative of POL with respect to DNBR.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - ~~T_{cold} GREATER THAN 210°F~~ ALL CEAs FULLY INSERTED

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to ~~6~~1.0% delta k/k.

APPLICABILITY: MODES ~~1, 2*~~, 3*, ~~and~~ 4*, AND 5 WITH ALL FULL-LENGTH CEAS FULLY INSERTED

ACTION:

With the SHUTDOWN MARGIN less than ~~6~~1.0% delta k/k, immediately initiate and continue boration at greater than or equal to 26 gpm to reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~6~~1.0% delta k/k:

~~a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable as a result of excessive friction or mechanical interference or known to be untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).~~

~~b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.~~

~~c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.~~

* See Special Test Exception 3.10.~~1~~.9

~~d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.~~

a ~~e~~. When in MODE 3 ~~or~~, 4, OR 5 at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1 above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

add to end of page 3/4 1-1

SHUTDOWN MARGIN - ~~T_{cold} LESS-THAN-OR-EQUAL-TO-210°F~~ KN-1 - ANY CEA WITHDRAWN

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 4.0% delta k/k.

APPLICABILITY: MODE 5.

SEE REVISED WRITEUP on next 3 pages

ACTION:

With the SHUTDOWN MARGIN less than 4.0% delta k/k, immediately initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 4.0% delta k/k:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable as a result of excessive friction or mechanical interference or known to be untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

new page 3/4 1-2

SHUTDOWN MARGIN - K_{N-1} - ANY CEA WITHDRAWN

LIMITING CONDITION FOR OPERATION

3.1.1.2

a. The SHUTDOWN MARGIN shall be greater than or equal to that shown in Figure 3.1-1A, ~~and~~

b. For T_{cold} less than or equal to 500 °F, K_{N-1} shall be less than 0.99.

APPLICABILITY: MODES 1, 2*, 3*, 4*, and 5* with any full-length CEA fully or partially withdrawn.

ACTION:

a. With the SHUTDOWN MARGIN less than that in Figure 3.1-1A, immediately initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm of boron or equivalent until the required SHUTDOWN MARGIN is restored, and

b. With T_{cold} less than or equal to 500 °F and K_{N-1} greater than or equal to 0.99, immediately vary CEA positions and/or initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm of boron or equivalent until the required K_{N-1} is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 With any full-length CEA fully or partially withdrawn, the SHUTDOWN MARGIN shall be determined to be greater than or equal to that in Figure 3.1.1A:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable as a result of excessive friction or mechanical interference or known to be untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

* See Special Test Exceptions 3.10.1 and 3.10.9.

new page 1-2a

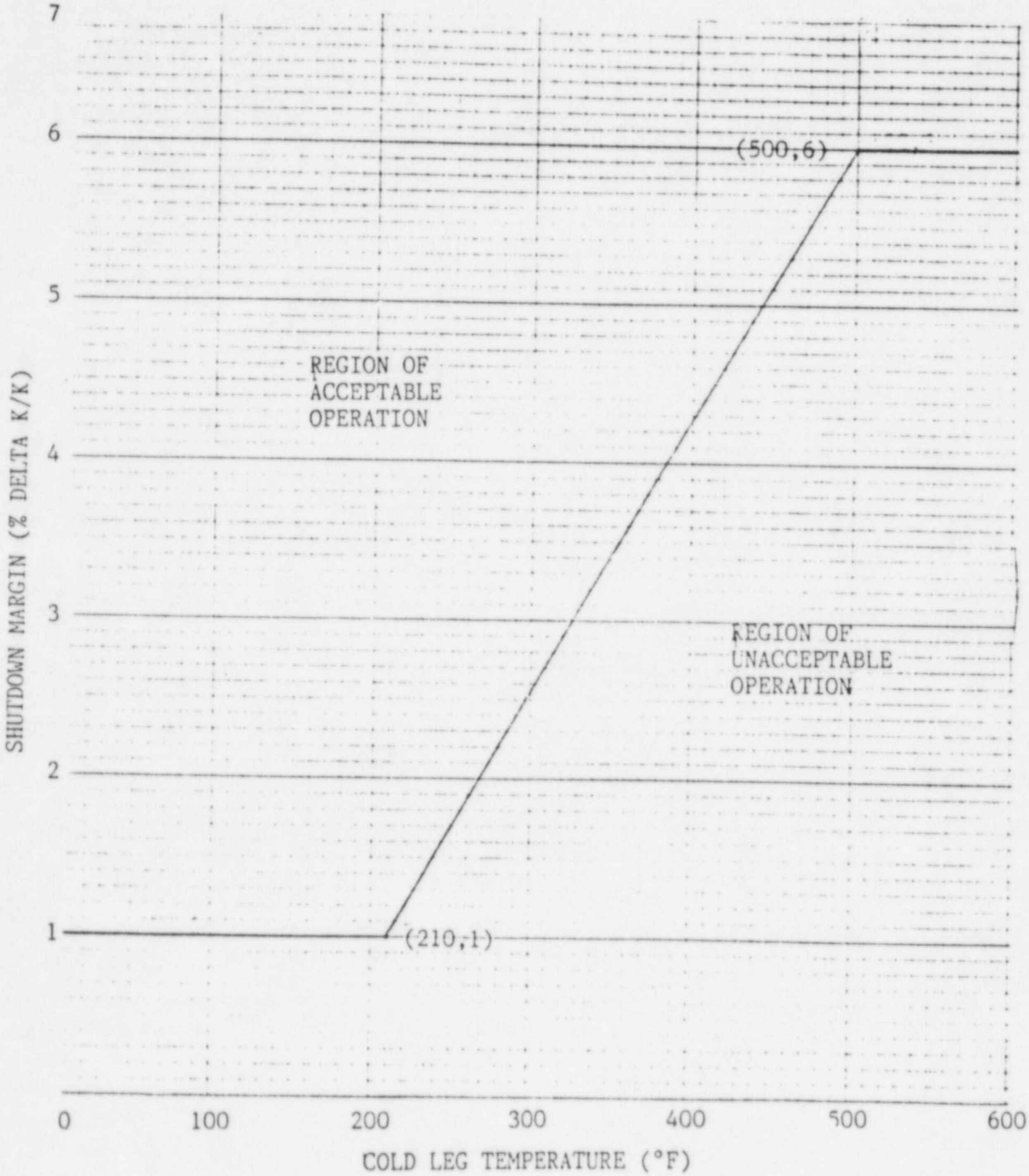


FIGURE 3.1 - 1A

SHUTDOWN MARGIN VERSUS COLD LEG TEMPERATURE new page 1-3

SURVEILLANCE REQUIREMENTS (Continued)

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

e. When in MODE 3, 4, or 5, at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.2.2 When in MODE 3, 4, or 5, with any full-length CEA fully or partially withdrawn, and T_{cold} less than or equal to 500 °F, K_{N-1} shall be determined to be less than 0.99 at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.2.3 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.2.1e or 4.1.1.2.2. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. A gravity feed flow path from either the refueling water tank or the spent fuel pool through CH-536 (RWT Gravity Feed Isolation Valve) and a charging pump to the Reactor Coolant System,

b. A gravity feed flow path from the refueling water tank through CH-327 (RWT Gravity Feed/Safety Injection System Isolation Valve) and a charging pump to the Reactor Coolant System,

c. A flow path from either the refueling water tank or the spent fuel pool through CH-164 (Boric Acid Filter Bypass Valve), utilizing gravity feed and a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY ~~and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210ºF~~ within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2.1 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months when the Reactor Coolant System is at normal operating pressure by verifying that the flow path required by Specification 3.1.2.2 delivers at least 26 gpm for 1 charging pump and 68 gpm for two charging pumps to the Reactor Coolant System.

4.1.2.2.2 The provisions of Specification 4.0.4 are not applicable for entry into Mode 3 or Mode 4 to perform the surveillance testing of Specification 4.1.2.2.b provided the testing is performed within 24 hours after achieving normal operating pressure in the reactor coolant system.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY ~~and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F~~ within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 No additional Surveillance Requirements other than those required by Specification 4.0.5.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 Each of the following borated water sources shall be OPERABLE:

a. The spent fuel pool with:

1. A minimum borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 and 4400 ppm of boron, and
3. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 1, 2,* 3,* and 4*.

ACTION:

a. With the above required spent fuel pool inoperable, restore the pool 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 6% delta k/k at 210°F~~; and restore the above required spent fuel poo to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each of the above required borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

b. At least once per 24 hours by verifying the refueling water tank temperature when the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when irradiated fuel is present in the pool.

*See Special Test Exception 3.10.7.

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

TABLE NOTATIONS

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(c) Trip may be manually bypassed below ~~1%~~ 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to ~~1%~~ 10^{-4}% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) There are four channels, each of which is comprised of one of the four reactor trip breakers, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6.g. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN AND K_{N-1} - CEA WORTH TESTS

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN and K_{N-1} requirement of Specification 3.1.1.~~1~~ 2 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s), or the reactor is subcritical by at least the reactivity equivalent of the highest CEA worth.

APPLICABILITY: MODES 2, 3* and 4*#.

ACTION:

a. With any full-length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 26 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN AND K_{N-1} required by Specification 3.1.1.~~1~~ 2 ~~is~~ are restored.

b. With all full-length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 26 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length and part-length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

4.10.1.3 When in MODE 3 or MODE 4, the reactor shall be determined to be subcritical by at least the reactivity equivalent of the highest estimated CEA worth or the reactivity equivalent of the highest estimated CEA worth is available for trip insertion from OPERABLE CEAs at least once per 2 hours by consideration of at least the following factors:

a. Reactor Coolant System boron concentration,

b. CEA position,

c. Reactor Coolant System average temperature,

d. Fuel burnup based on gross thermal energy generation,

e. Xenon concentration, and

f. Samarium concentration.

---

*Operation in MODE 3 and MODE 4 shall be limited to 6 consecutive hours.

#Limited to low power PHYSICS TESTING at the 320°F plateau.

3/4.10.9 SHUTDOWN MARGIN AND K_{N-1} - CEDMS TESTING

LIMITING CONDITION FOR OPERATION

3.10.9 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 and the SHUTDOWN MARGIN and K_{N-1} requirements of Specification 3.1.1.2 may be suspended for pre-startup tests to demonstrate the OPERABILITY of the control element drive mechanism system provided:

a. No more than one CEA is withdrawn at any time.

b. No CEA is withdrawn more than 7 inches.

c. The K_{N-1} requirement of Specification 3.1.1.2 is met prior to the start of testing.

d. All other operations involving positive reactivity changes are suspended during the testing.

APPLICABILITY: MODES 4 and 5.

ACTION: With any of the above requirements not met, suspend testing and comply with the requirements of Specification 3.1.1.1 or 3.1.1.2, as applicalble.

SURVEILLANCE REQUIREMENTS

4.10.9 Surveillance Requirements 4.1.1.2.1.e and 4.1.1.2.2 shall be conducted within one hour prior to the start of testing, and at least once per 12 hours during testing.

CONTROLLED BY USER

3/4.10.~~9~~10 NATURAL CIRCULATION TESTING PROGRAM

LIMITING CONDITION FOR OPERATION

3.10.~~9~~10 The limitations of Specifications 3.4.1.2, 3.4.1.3, and 3.7.1.6 may be suspended during the performance of the Startup Natural Circulation Testing Program* provided:

a. Operations involving a reduction in boron concentration of the Reactor Coolant System are suspended.

b. Core outlet temperature is maintained at least 10°F below Saturation temperature.

c. A reactor coolant pump shall not be started with one or more of Reactor Coolant System cold leg temperatures less than or equal to 255°F during cooldown, or 295°F during heatup, unless the secondary water temperature (saturation temperature corresponding to steam generator pressure) of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

APPLICABILITY: MODES 3 and 4 during Natural Circulation Testing.

ACTION:

With the Reactor Coolant System saturation margin less than 10°F, place at least one reactor coolant loop in operation, with at least one reactor coolant pump.

SURVEILLANCE REQUIREMENTS

4.10.~~9~~10.1 The saturation margin shall be determined to be within the above limits by continuous monitoring with the saturation margin monitors required by Table 3.3-10 or, by calculating the saturation margin at least once per 30 minutes.

*Startup Natural Circulation Testing Program:

Natural Circulation Cooldown Test at 80% power.

CONTROLLED BY USER

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN AND K_{N-1}

A sufficient SHUTDOWN MARGIN ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits assuming the insertion of the regulating CEAs are within the limits of Specification 3.1.3.6, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SEE REVISED WRITEUP, next page

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{cold}. The most restrictive condition occurs at EOL, with T_{cold} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 6.0% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with the criteria used to establish the power dependent CEA insertion limits and with the assumptions used in the FSAR Safety Analysis.

With T_{cold} less than or equal to 210°F, the reactivity transients resulting from uncontrolled RCS cooldown are minimal and a 4% $\Delta k/k$ SHUTDOWN MARGIN requirement is set to ensure that reactivity transients resulting from an inadvertent single CEA withdrawal event are minimal.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

delete, moved to page: B 3/4 1-1a

NEW PAGE B 3/4 1-1

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN AND K_{N-1}

The function of SHUTDOWN MARGIN is to ensure that the reactor remains subcritical following a design basis accident or anticipated operational occurrence. The function of K_{N-1} is to maintain sufficient subcriticality to preclude inadvertent criticality following ejection of a single control element assembly (CEA). During operation in MODES 1 and 2, with k_{eff} greater than or equal to 1.0, the transient insertion limits of Specification 3.1.3.6 ensure that sufficient SHUTDOWN MARGIN is available.

SHUTDOWN MARGIN is the amount by which the core is subcritical, or would be subcritical immediately following a reactor trip, considering a single malfunction resulting in the highest worth CEA failing to insert. K_{N-1} is a measure of the core's reactivity, considering a single malfunction resulting in the highest worth inserted CEA being ejected.

SHUTDOWN MARGIN requirements vary throughout the core life as a function of fuel depletion and reactor coolant system (RCS) cold leg temperature (T_{cold}). The most restrictive condition occurs at EOL, with T_{cold} at no-load operating temperature,and is associated with a postulated steam line break accident and the resulting uncontrolled RCS cooldown. In the analysis of this accident, the specified SHUTDOWN MARGIN is required to control the reactivity transient and ensure that the fuel performance and offsite dose criteria are satisfied. As (initial) T_{cold} decreases, the potential RCS cooldown and the resulting reactivity transient are less severe and, therefore, the required SHUTDOWN MARGIN also decreases. Below T_{cold} of about 210 °F, the inadvertent deboration event becomes limiting with respect to the SHUTDOWN MARGIN requirements. Below 210 °F, the specified SHUTDOWN MARGIN ensures that sufficient time for operator actions exists between the initial indication of the deboration and the total loss of shutdown margin. Accordingly, with at least one CEA partially or fully withdrawn, the SHUTDOWN MARGIN requirements are based upon these limiting conditions.

Additional events considered in establishing requirements on SHUTDOWN MARGIN that are not limiting with respect to the Specification limits are single CEA withdrawal and startup of an inactive reactor coolant pump.

K_{N-1} requirements vary with the amount of positive reactivity that would be introduced assuming the CEA with the highest inserted worth ejects from the core. In the analysis of the CEA ejection event, the K_{N-1} requirement ensures that the radially averaged enthalpy acceptance criterion is satisfied, considering power redistribution effects. Above T_{cold} of 500 °F, Doppler reactivity feedback is sufficient to preclude the need for a specific K_{N-1} requirement. With all CEAs fully inserted, K_{N-1} and SHUTDOWN MARGIN requirements are equivalent in terms of minimum acceptable core boron concentration.

3/4.1.1 BORATION CONTROL (cont.)

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN AND K_{N-1} (cont.)

Other technical specifications that reference the Specifications on SHUTDOWN MARGIN or K_{N-1} are: 3/4.1.2, BORATION SYSTEMS, 3/4.1.3, MOVABLE CONTROL ASSEMBLIES, 3/4.9.1, REFUELING OPERATIONS- BORON CONCENTRATION, 3/4.10.1, SHUTDOWN MARGIN AND K_{N-1} - CEA WORTH TESTS, and 3/4.10.9, SHUTDOWN MARGIN AND K_{N-1} - CEDMS TESTING.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 552°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, and (3) to ensure consistancy with the FSAR safety analysis.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 552°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, and (3) to ensure consistency with the FSAR safety analysis.

delete; added to previous page (B 3/4 1-1a)

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) an emergency power supply from OPERABLE diesel generators, and (5) the volume control tank (VCT) outlet valve CH-UV-501, capable of isolating the VCT from the charging pump suction line. The nominal capacity of each charging pump is 44 gpm at its discharge. Up to 16 gpm of this may be diverted to the volume control tank via the RCP control bleedoff. Instrument inaccuracies and pump performance uncertainties are limited to 2 gpm yielding the 26 gpm value.

With the RCS temperature above 210°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

EACH SYSTEM IS CAPABLE OF PROVIDING BORATION EQUIVALENT TO A

~~The boration capability of either system is~~ sufficient to provide a SHUTDOWN MARGIN ~~from expected operating conditions~~ of 4% delta k/k after xenon decay and cooldown to 210°F. #1 The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 23,800 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

insert #1 (see next pg)

With the RCS temperature below 210°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 restrictions of one and only one operable charging pump whenever reactor coolant level is below the bottom of the pressurizer is based on the assumptions used in the analysis of the boron dilution event.

insert #2 (see next pg)

~~The boron capability required below 210°F is based upon providing a 4% delta-k/k SHUTDOWN MARGIN after xenon decay and cooldown from 210°F to 120°F.~~ This condition requires 9,700 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

insert #1 THEREFORE, THE BORATION CAPACITY OF EITHER SYSTEM IS MORE THAN SUFFICIENT TO SATISFY THE SHUTDOWN MARGIN AND/OR K_{N-1} REQUIREMENTS OF THE SPECIFICATIONS.

insert #2 EACH SYSTEM IS CAPABLE OF PROVIDING BORATION EQUIVALENT TO A SHUTDOWN MARGIN OF 4% DELTA K/K. THEREFORE, THE BORATION CAPACITY OF THE SYSTEM REQUIRED BELOW 210°F IS MORE THAN SUFFICIENT TO SATISFY THE SHUTDOWN MARGIN AND/OR K_{N-1} REQUIREMENTS OF THE SPECIFICATIONS.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.6 SAFETY INJECTION TANKS

This special test exception permits testing the low pressure safety injection system check valves. The pressure in the injection header must be reduced below the head of the low pressure injection pump in order to get flow through the check valves. The safety injection tank (SIT) isolation valve must be closed in order to accomplish this. The SIT isolation valve is still capable of automatic operation in the event of an SIAS; therefore, system capability should not be affected.

3/4.10.7 SPENT FUEL POOL LEVEL

This special test exception permits loading of the initial core with the spent fuel pool dry.

3/4.10.8 SAFETY INJECTION TANK PRESSURE

This special test exception allows the performance of PHYSICS TESTS at low pressure/low temperature (600 psig, 320°F) conditions which are required to verify the low temperature physics predictions and to ensure the adequacy of design codes for reduced temperature conditions.

3/4.10.9 SHUTDOWN MARGIN AND KN-1 - CEDMS TESTING

THIS SPECIAL TEST EXCEPTION ALLOWS THE PERFORMANCE OF CONTROL ELEMENT DRIVE MECHANISM TESTS PRIOR TO STARTUP, WITHOUT THE OPERATOR HAVING TO BE CONCERNED AS TO WHETHER SPECIFICATION 3.1.1.1 OR 3.1.1.2 IS APPLICABLE AS CEA'S ARE MOVED. THE LOGARITHMIC POWER LEVEL - HIGH TRIP PROVIDES ADDITIONAL PROTECTION AGAINST INADVERTENT CRITICALITY DURING THIS TEST.

DEFINITIONS

SECTION | PAGE

1.0 DEFINITIONS

SECTION

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

K_{N-1} Insert (A)

1.16 K_{N-1} is the k-effective calculated assuming the fully or partially inserted full-length control element assembly of highest inserted worth is fully withdrawn.

DEFINITIONS

Insert (A)

MEMBER(S) OF THE PUBLIC

1.16 7 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.17 8 The OFFSITE DOSE CALCULATION MANUAL shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

1.18 9 A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.19 20 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and cold leg reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.20 1 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PLANAR RADIAL PEAKING FACTOR - F_{xy}

1.21 2 The PLANAR RADIAL PEAKING FACTOR is the ratio of the peak to plane average power density of the individual fuel rods in a given horizontal plane, excluding the effects of azimuthal tilt.

PRESSURE BOUNDARY LEAKAGE

1.22 3 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.234 The PROCESS CONTROL PROGRAM shall contain the provisions to assure that the SOLIDIFICATION of wet radioactive wastes results in a waste form with properties that meet the requirements of 10 CFR Part 61 and of low level radioactive waste disposal sites. The PCP shall identify process parameters influencing SOLIDIFICATION such as pH, oil content, H_2O content, solids content, ratio of solidification agent to waste and/or necessary additives for each type of anticipated waste, and the acceptable boundary conditions for the process parameters shall be identified for each waste type, based on laboratory scale and full-scale testing or experience. The PCP shall also include an identification of conditions that must be satisfied, based on full-scale testing, to assure that dewatering of bead resins, powdered resins, and filter sludges will result in volumes of free water, at the time of disposal, within the limits of 10 CFR Part 61 and of low level radioactive waste disposal sites.

PURGE - PURGING

1.245 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.256 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3800 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.267 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

REPORTABLE EVENT

1.278 A REPORTABLE EVENT shall be any of those conditions specified in Sections 50.72 and 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.289 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. No change in part-length control element assembly position, and

b. All full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.~~29~~30 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOFTWARE

1.~~30~~31 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation, and procedures.

SOLIDIFICATION

1.~~31~~32 SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a homogeneous (uniformly distributed), monolithic, immobilized solid with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

SOURCE CHECK

1.~~32~~33 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.~~33~~34 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.~~34~~35 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.~~35~~36 UNIDENTIFIED LEAKAGE shall be all leakage which does not constitute either IDENTIFIED LEAKAGE or reactor coolant pump controlled bleed-off flow.

UNRESTRICTED AREA

1.~~36~~37 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.3~~7~~8 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.3~~8~~9 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Logarithmic Power Level - High (1)		
a. Startup and Operating	≤ ~~0.798%~~ 0.010% of RATED THERMAL POWER	≤ ~~0.895%~~ 0.011% of RATED THERMAL POWER
b. Shutdown	≤ ~~0.798%~~ 0.010% of RATED THERMAL POWER	≤ ~~0.895%~~ 0.011% of RATED THERMAL POWER
C. Core Protection Calculator System		
1. CEA Calculators	Not Applicable	Not Applicable
2. Core Protection Calculators	Not Applicable	Not Applicable
D. Supplementary Protection System		
Pressurizer Pressure - High	≤ 2409 psia	≤ 2414 psia
II. RPS LOGIC		
A. Matrix Logic	Not Applicable	Not Applicable
B. Initiation Logic	Not Applicable	Not Applicable
III. RPS ACTUATION DEVICES		
A. Reactor Trip Breakers	Not Applicable	Not Applicable
B. Manual Trip	Not Applicable	Not Applicable

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) Trip may be manually bypassed above $10^{-4}\%$ of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to $10^{-4}\%$ of RATED THERMAL POWER.

(2) In MODES 3-4, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) In MODES 3-4, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below ~~1%~~ $10^{-4}\%$ of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to ~~1%~~ $10^{-4}\%$ of RATED THERMAL POWER.

The approved DNBR limit is 1.231 which includes a partial rod bow penalty compensation. If the fuel burnup exceeds that for which an increased rod bow penalty is required, the DNBR limit shall be adjusted. In this case a DNBR trip setpoint of 1.231 is allowed provided that the difference is compensated by an increase in the CPC addressable constant BERR1 as follows:

$$BERR1_{new} = BERR1_{old}\left[1 + \frac{RB - RB_o}{100} \times \frac{d\,(\%\,POL)}{d\,(\%\,DNBR)}\right]$$

where $BERR1_{old}$ is the uncompensated value of BERR1; RB is the fuel rod bow penalty in % DNBR; RB_o is the fuel rod bow penalty in % DNBR already accounted for in the DNBR limit; POL is the power operating limit; and d (% POL)/d (% DNBR) is the absolute value of the most adverse derivative of POL with respect to DNBR.

REACTIVITY CONTROL SYSTEMS

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - ~~T_{cold} GREATER THAN 210°F~~ ALL CEAs FULLY INSERTED

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to ~~6~~1.0% delta k/k.

APPLICABILITY: MODES ~~1, 2*~~, 3*, ~~and~~ 4*, AND 5 WITH ALL FULL-LENGTH CEAs FULLY INSERTED

ACTION:

With the SHUTDOWN MARGIN less than ~~6~~1.0% delta k/k, immediately initiate and continue boration at greater than or equal to 26 gpm to reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~6~~1.0% delta k/k:

~~a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable as a result of excessive friction or mechanical interference or known to be untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).~~

~~b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.~~

~~c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.~~

* See Special Test Exception 3.10.~~1~~9

~~d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.~~

a ~~e~~. When in MODE 3 ~~or~~, 4, OR 5 at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within + 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1 above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

add to end of page 3/4 1-1

KN-1 "ANY" CEA WITHDRAWN

SHUTDOWN MARGIN - ~~T_{cold} LESS THAN OR EQUAL TO 210°F~~

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 4.0% delta k/k.

APPLICABILITY: MODE 5.

SEE REVISED WRITEUP on next 3 pages

ACTION:

With the SHUTDOWN MARGIN less than 4.0% delta k/k, immediately initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 4.0% delta k/k:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable as a result of excessive friction or mechanical interference or known to be untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

new page 3/4 1-2

SHUTDOWN MARGIN - K_{N-1} - ANY CEA WITHDRAWN

LIMITING CONDITION FOR OPERATION

3.1.1.2

a. The SHUTDOWN MARGIN shall be greater than or equal to that shown in Figure 3.1-1A, and

b. For T_{cold} less than or equal to 500 °F, K_{N-1} shall be less than 0.99.

APPLICABILITY: MODES 1, 2*, 3*, 4*, and 5* with any full-length CEA fully or partially withdrawn.

ACTION:

a. With the SHUTDOWN MARGIN less than that in Figure 3.1-1A, immediately initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm of boron or equivalent until the required SHUTDOWN MARGIN is restored, and

b. With T_{cold} less than or equal to 500 °F and K_{N-1} greater than or equal to 0.99, immediately vary CEA positions and/or initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm of boron or equivalent until the required K_{N-1} is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 With any full-length CEA fully or partially withdrawn, the SHUTDOWN MARGIN shall be determined to be greater than or equal to that in Figure 3.1.1A:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable as a result of excessive friction or mechanical interference or known to be untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

* See Special Test Exceptions 3.10.1 and 3.10.9.

new page 1-2a

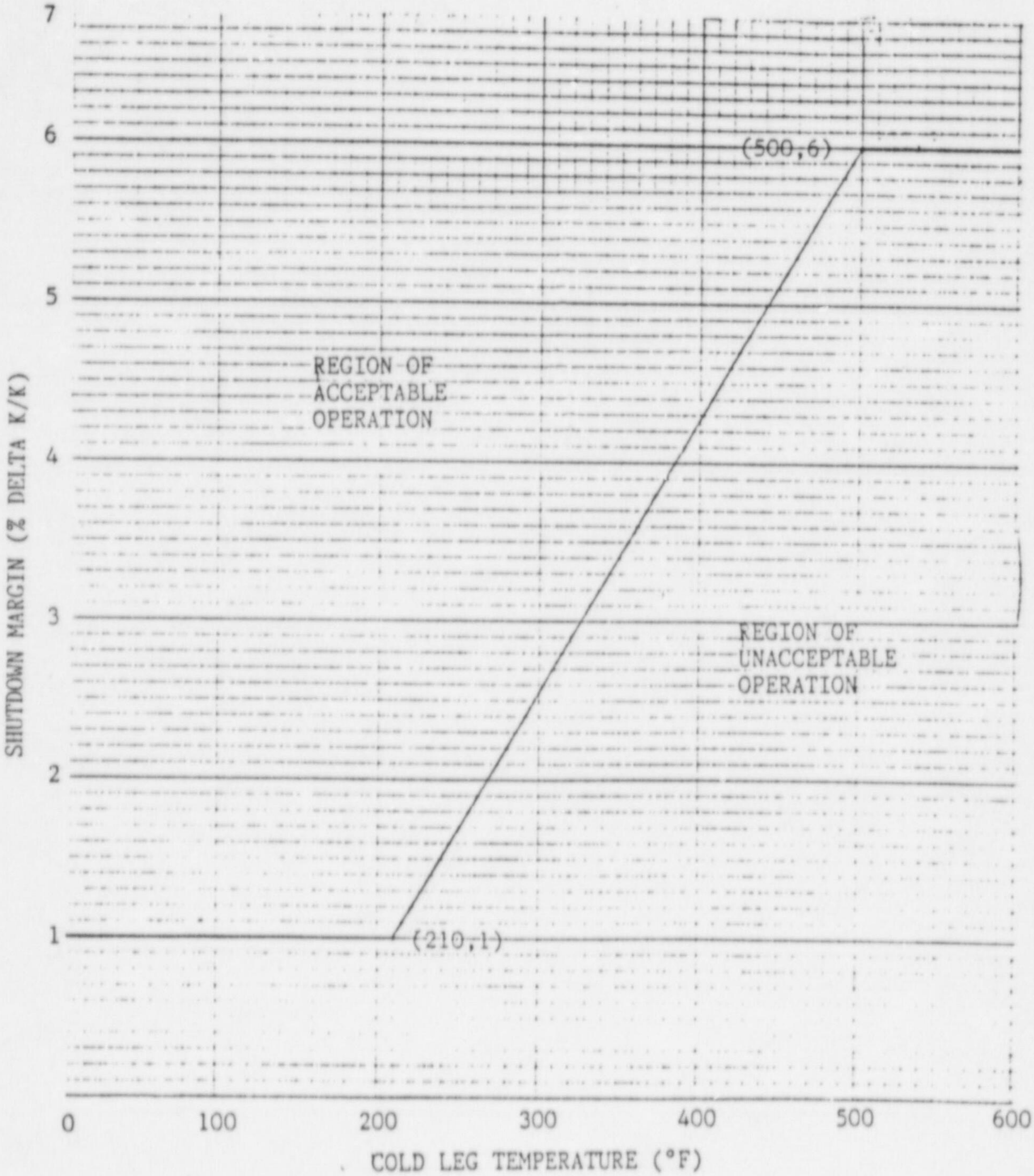


FIGURE 3.1 - 1A

SHUTDOWN MARGIN VERSUS COLD LEG TEMPERATURE new page 1-3

SURVEILLANCE REQUIREMENTS (Continued)

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

e. When in MODE 3, 4, or 5, at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.2.2 When in MODE 3, 4, or 5, with any full-length CEA fully or partially withdrawn, and T_{cold} less than or equal to 500 °F, K_{N-1} shall be determined to be less than 0.99 at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.2.3 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.2.1e or 4.1.1.2.2. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. A gravity feed flow path from either the refueling water tank or the spent fuel pool through CH-536 (RWT Gravity Feed Isolation Valve) and a charging pump to the Reactor Coolant System,

b. A gravity feed flow path from the refueling water tank through CH-327 (RWT Gravity Feed/Safety Injection System Isolation Valve) and a charging pump to the Reactor Coolant System,

c. A flow path from either the refueling water tank or the spent fuel pool through CH-164 (Boric Acid Filter Bypass Valve), utilizing gravity feed and a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY ~~and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F~~ within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2.1 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months when the Reactor Coolant System is at normal operating pressure by verifying that the flow path required by Specification 3.1.2.2 delivers at least 26 gpm for 1 charging pump and 68 gpm for two charging pumps to the Reactor Coolant System.

4.1.2.2.2 The provisions of Specification 4.0.4 are not applicable for entry into Mode 3 or Mode 4 to perform the surveillance testing of Specification 4.1.2.2.b provided the testing is performed within 24 hours after achieving normal operating pressure in the reactor coolant system.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY ~~and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F~~ within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 No additional Surveillance Requirements other than those required by Specification 4.0.5.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 Each of the following borated water sources shall be OPERABLE:

a. The spent fuel pool with:

1. A minimum borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 and 4400 ppm of boron, and
3. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 1, 2,* 3,* and 4*.

ACTION:

a. With the above required spent fuel pool inoperable, restore the pool 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 6% delta k/k at 210°F,~~ and restore the above required spent fuel pool to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each of the above required borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

b. At least once per 24 hours by verifying the refueling water tank temperature when the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when irradiated fuel is present in the pool.

*See Special Test Exception 3.10.7.

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

TABLE NOTATIONS

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above $10^{-4}\%$ of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to $10^{-4}\%$ of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

$10^{-4}\%$

(c) Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

$10^{-4}\%$

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) There are four channels, each of which is comprised of one of the four reactor trip breakers, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6.g. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN AND K_{N-1} · CEA WORTH TESTS

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN and K_{N-1} requirement of Specification 3.1.1.~~1~~ 7 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s), or the reactor is subcritical by at least the reactivity equivalent of the highest CEA worth.

APPLICABILITY: MODES 2, 3* and 4*#.

ACTION:

a. With any full-length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 26 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN AND K_{N-1} required by Specification 3.1.1.~~1~~ 7 ~~is~~ are restored.

b. With all full-length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 26 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length and part-length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

4.10.1.3 When in MODE 3 or MODE 4, the reactor shall be determined to be subcritical by at least the reactivity equivalent of the highest estimated CEA worth or the reactivity equivalent of the highest estimated CEA worth is available for trip insertion from OPERABLE CEAs at least once per 2 hours by consideration of at least the following factors:

a. Reactor Coolant System boron concentration,

b. CEA position,

c. Reactor Coolant System average temperature,

d. Fuel burnup based on gross thermal energy generation,

e. Xenon concentration, and

f. Samarium concentration.

---

*Operation in MODE 3 and MODE 4 shall be limited to 6 consecutive hours.

#Limited to low power PHYSICS TESTING at the 320°F plateau.

3/4.10.9 SHUTDOWN MARGIN AND K_{N-1} - CEDMS TESTING

LIMITING CONDITION FOR OPERATION

3.10.9 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 and the SHUTDOWN MARGIN and K_{N-1} requirements of Specification 3.1.1.2 may be suspended for pre-startup tests to demonstrate the OPERABILITY of the control element drive mechanism system provided:

a. No more than one CEA is withdrawn at any time.

b. No CEA is withdrawn more than 7 inches.

c. The K_{N-1} requirement of Specification 3.1.1.2 is met prior to the start of testing.

d. All other operations involving positive reactivity changes are suspended during the testing.

APPLICABILITY: MODES 4 and 5.

ACTION: With any of the above requirements not met, suspend testing and comply with the requirements of Specification 3.1.1.1 or 3.1.1.2, as applicalble.

SURVEILLANCE REQUIREMENTS

4.10.9 Surveillance Requirements 4.1.1.2.1.e and 4.1.1.2.2 shall be conducted within one hour prior to the start of testing, and at least once per 12 hours during testing.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN AND K_{N-1}

A sufficient SHUTDOWN MARGIN ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits assuming the insertion of the regulating CEAs are within the limits of Specification 3.1.3.6, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SEE REVISED WRITEUP, next page

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{cold}. The most restrictive condition occurs at EOL, with T_{cold} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 6.0% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with the criteria used to establish the power dependent CEA insertion limits and with the assumptions used in the FSAR Safety Analysis.

With T_{cold} less than or equal to 210°F, the reactivity transients resulting from uncontrolled RCS cooldown are minimal and a 4% $\Delta k/k$ SHUTDOWN MARGIN requirement is set to ensure that reactivity transients resulting from an inadvertent single CEA withdrawal event are minimal.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

delete; moved to page B 3/4 1-1a

NEW PAGE B 3/4 1-1

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN AND K_{N-1}

The function of SHUTDOWN MARGIN is to ensure that the reactor remains subcritical following a design basis accident or anticipated operational occurrence. The function of K_{N-1} is to maintain sufficient subcriticality to preclude inadvertent criticality following ejection of a single control element assembly (CEA). During operation in MODES 1 and 2, with k_{eff} greater than or equal to 1.0, the transient insertion limits of Specification 3.1.3.6 ensure that sufficient SHUTDOWN MARGIN is available.

SHUTDOWN MARGIN is the amount by which the core is subcritical, or would be subcritical immediately following a reactor trip, considering a single malfunction resulting in the highest worth CEA failing to insert. K_{N-1} is a measure of the core's reactivity, considering a single malfunction resulting in the highest worth inserted CEA being ejected.

SHUTDOWN MARGIN requirements vary throughout the core life as a function of fuel depletion and reactor coolant system (RCS) cold leg temperature (T_{cold}). The most restrictive condition occurs at EOL, with T_{cold} at no-load operating temperature,and is associated with a postulated steam line break accident and the resulting uncontrolled RCS cooldown. In the analysis of this accident, the specified SHUTDOWN MARGIN is required to control the reactivity transient and ensure that the fuel performance and offsite dose criteria are satisfied. As (initial) T_{cold} decreases, the potential RCS cooldown and the resulting reactivity transient are less severe and, therefore, the required SHUTDOWN MARGIN also decreases. Below T_{cold} of about 210 °F, the inadvertent deboration event becomes limiting with respect to the SHUTDOWN MARGIN requirements. Below 210 °F, the specified SHUTDOWN MARGIN ensures that sufficient time for operator actions exists between the initial indication of the deboration and the total loss of shutdown margin. Accordingly, with at least one CEA partially or fully withdrawn, the SHUTDOWN MARGIN requirements are based upon these limiting conditions.

Additional events considered in establishing requirements on SHUTDOWN MARGIN that are not limiting with respect to the Specification limits are single CEA withdrawal and startup of an inactive reactor coolant pump.

K_{N-1} requirements vary with the amount of positive reactivity that would be introduced assuming the CEA with the highest inserted worth ejects from the core. In the analysis of the CEA ejection event, the K_{N-1} requirement ensures that the radially averaged enthalpy acceptance criterion is satisfied, considering power redistribution effects. Above T_{cold} of 500 °F, Doppler reactivity feedback is sufficient to preclude the need for a specific K_{N-1} requirement. With all CEAs fully inserted, K_{N-1} and SHUTDOWN MARGIN requirements are equivalent in terms of minimum acceptable core boron concentration.

3/4.1.1 BORATION CONTROL (cont.)

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN AND K_{N-1} (cont.)

Other technical specifications that reference the Specifications on SHUTDOWN MARGIN or K_{N-1} are: 3/4.1.2, BORATION SYSTEMS, 3/4.1.3, MOVABLE CONTROL ASSEMBLIES, 3/4.9.1, REFUELING OPERATIONS- BORON CONCENTRATION, 3/4.10.1, SHUTDOWN MARGIN AND K_{N-1} - CEA WORTH TESTS, and 3/4.10.9, SHUTDOWN MARGIN AND K_{N-1} - CEDMS TESTING.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 552°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, and (3) to ensure consistency with the FSAR safety analysis.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 552°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, and (3) to ensure consistency with the FSAR safety analysis.

delete; added to previous page (B 3/4 1-1a)

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) an emergency power supply from OPERABLE diesel generators, and (5) the volume control tank (VCT) outlet valve CH-UV-501, capable of isolating the VCT from the charging pump suction line. The nominal capacity of each charging pump is 44 gpm at its discharge. Up to 16 gpm of this may be diverted to the volume control tank via the RCP control bleedoff. Instrument inaccuracies and pump performance uncertainties are limited to 2 gpm yielding the 26 gpm value.

With the RCS temperature above 210°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

EACH SYSTEM IS CAPABLE OF PROVIDING BORATION EQUIVALENT TO A

~~The boration capability of either system~~ is sufficient to provide a SHUTDOWN MARGIN ~~from expected operating conditions~~ of 4% delta k/k after xenon decay and cooldown to 210°F. #1 The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 23,800 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool. insert #1 (next pg)

With the RCS temperature below 210°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 restrictions of one and only one operable charging pump whenever reactor coolant level is below the bottom of the pressurizer is based on the assumptions used in the analysis of the boron dilution event.

insert #2 (next pg) ~~The boron capability~~ ~~required below 210°F is based upon providing a 4% delta-k/k SHUTDOWN MARGIN after xenon decay and cooldown from 210°F to 120°F.~~ This condition requires 9,700 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

insert #1 THEREFORE, THE BORATION CAPACITY OF EITHER SYSTEM IS MORE THAN SUFFICIENT TO SATISFY THE SHUTDOWN MARGIN AND/OR K_{n-1} REQUIREMENTS OF THE SPECIFICATIONS.

insert #2 EACH SYSTEM IS CAPABLE OF PROVIDING BORATION EQUIVALENT TO A SHUTDOWN MARGIN OF 4% DELTA K/K. THEREFORE, THE BORATION CAPACITY OF THE SYSTEM REQUIRED BELOW 210 °F IS MORE THAN SUFFICIENT TO SATISFY THE SHUTDOWN MARGIN AND/OR K_{n-1} REQUIREMENTS OF THE SPECIFICATIONS.

3/4.10.6 SAFETY INJECTION TANKS

This special test exception permits testing the low pressure safety injection system check valves. The pressure in the injection header must be reduced below the head of the low pressure injection pump in order to get flow through the check valves. The safety injection tank (SIT) isolation valve must be closed in order to accomplish this. The SIT isolation valve is still capable of automatic operation in the event of an SIAS; therefore, system capability should not be affected.

3/4.10.7 SPENT FUEL POOL LEVEL

This special test exception permits loading of the initial core with the spent fuel pool dry.

3/4.10.8 SAFETY INJECTION TANK PRESSURE

This special test exception allows the performance of PHYSICS TESTS at low pressure/low temperature (600 psig, 320°F) conditions which are required to verify the low temperature physics predictions and to ensure the adequacy of design codes for reduced temperature conditions.

3/4.10.9 SHUTDOWN MARGIN AND KN-1 - CEDMS TESTING

THIS SPECIAL TEST EXCEPTION ALLOWS THE PERFORMANCE OF CONTROL ELEMENT DRIVE MECHANISM TESTS PRIOR TO STARTUP, WITHOUT THE OPERATOR HAVING TO BE CONCERNED AS TO WHETHER SPECIFICATION 3.1.1.1 OR 3.1.1.2 IS APPLICABLE AS CEA'S ARE MOVED. THE LOGARITHMIC POWER LEVEL-HIGH TRIP PROVIDES ADDITIONAL PROTECTION AGAINST INADVERTENT CRITICALITY DURING THIS TEST.

H. Supporting Analyses for the Technical Specification Amendment Request:

1. STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE CONTAINMENT - MODE 3 OPERATION: T_C <500°F

1.1 Identification of Event and Causes

Refer to CESSAR Section 15.1.4, except that steam line break events during Mode 3 operation for reactor cold leg temperature (T_{cold}) less than 500°F are analyzed to demonstrate the adequacy of the shutdown margin as specified by Technical Specifications 3.1.1.1 and 3.1.1.2, to prevent degradation in fuel performance as a result of post trip return to power. The results show that the shutdown margin is sufficiently large to prevent a post trip return to power. The steam line breaks presented are:

A. A large steam line break inside containment during Mode 3 operation with concurrent loss of offsite power in combination with a single failure and technical specification shutdown margin.

B. A large steam line break inside containment during Mode 3 operation with offsite power available in combination with a single failure and technical specification shutdown margin.

For T_{cold} > 500°F, the shutdown margin specified by Technical Specification 3.1.1.2 is 6% $\Delta\rho$. Below 500°F, the shutdown margin decreases linearly with temperature. Here, cold leg temperatures below 500°F are considered. Hot zero power steam line breaks above 500°F using technical specification shutdown margin are found in PVNGS FSAR Section 15.1.5. The requirements of Technical Specification 3.1.1.1 are less limiting for steam line breaks than are those of Technical Specification 3.1.1.2.

The largest possible steam line break size is the double ended rupture of a steam line upstream of the main steam isolation valve (MSIV). In the PVNGS design, an integral flow restrictor exists in each steam generator outlet nozzle. The largest effective steam blowdown area for each steam line, which is limited by the flow restrictor throat area, is 1.28 square feet.

These two cases are analyzed for end of equilibrium core, self-generated plutonium recycle (SGR) conditions. PVNGS specific minimum safety injection flow rates, feedwater isolation valve closure time, and steam generator differential pressure (Δ P) isolation (lockout) setpoint are employed.

1.2 Sequence of Events and System Operation

Steam line breaks are characterized as cooldown events due to the increased steam flow rate, which causes excessive energy removal from the steam generators and the reactor coolant system (RCS). This results in a decrease in reactor coolant temperatures and in RCS and steam generator pressures. The cooldown causes an increase in core reactivity due to the negative moderator and Doppler reactivity coefficients.

Mode 3 steam line breaks are initiated from a subcritical reactivity condition. Detection of the cooldown is accomplished by the pressurizer and steam generator low pressure alarms, by the high reactor power alarm and by the low steam generator water level alarm. Reactor trip is provided by one of two available reactor trip signals. These are the low steam generator pressure and the high logarithmic power level trips.

For a steam line break that occurs with a concurrent loss of offsite power, termination of feedwater to both steam generators and coastdown of the reactor coolant pumps are assumed to be initiated simultaneously. In general, the depressurization of the affected steam generator results in actuation of a main steam isolation signal (MSIS). This closes the MSIVs, isolating the unaffected steam generator from blowdown, and closes the main feedwater isolation valves (MFIV), terminating main feedwater flow to both steam generators. After the reduction of steam flow that occurs following MSIV closure, the level in the intact steam generator falls below the auxiliary feedwater actuation signal (AFAS) setpoint. The resulting AFAS causes auxiliary feedwater (AFW) flow to be initiated to both steam generators. If the differential pressure between the two steam generators exceeds the setpoint, the AFW logic isolates flow to the affected steam generator and diverts the flow from both AFW pumps to the intact steam generator. The pressurizer pressure may decrease to the point where a safety injection actuation signal (SIAS) is initiated. The isolation of the unaffected steam generator and subsequent emptying of the affected steam generator terminate the cooldown. The introduction of safety injection boron upon SIAS causes core reactivity to decrease. The operator, via the appropriate emergency procedures, may initiate plant cooldown by manual control of the atmospheric steam dump valves, or, in the event that offsite power is available, by using the unaffected steam generator and the turbine bypass valves, any time after the affected steam generator empties. The analyses presented herein conservatively assume operator action is delayed until 30 minutes after event initiation. The plant is then cooled to 350°F and 400 psia, at which point shutdown cooling is initiated.

A parametric study of single failures (see Appendix 15C of CESSAR) that would have an adverse impact on the SLB event has determined that the failure of one of the high pressure safety injection (HPSI) pumps to start following SIAS has the most adverse effect for those cases that result in generation of SIAS. For the two cases presented here, there is no SIAS actuation for the duration of the transient (500 seconds). For these events the most adverse effect is caused by the failure of a MSIV on one of the steam lines from the intact generator to close following MSIS. Consequently, for these cases steam is assumed to continue to be released from the intact steam generator at 1.5% of the design steam rate. This open flow path is represented by an effective flow area for steam blowdown from the intact steam generator of 0.034 square feet.

1.3 Analysis of Effects and Consequences

A. Mathematical Models

The mathematical models and data transfer between codes used in the SLB analysis are presented in PVNGS FSAR Appendix 15.C.

B. Input Parameters and Initial Conditions

The initial conditions assumed in the analysis of the NSSS response to Cases 1 and 2 are presented in Table 1-1. The initial K_{eff} of 0.99 is the highest value allowed by technical specifications for Mode 3 and leaves the least pretrip margin to criticality. There is no effect on the post trip margin to criticality. Above core inlet temperatures of 500°F, the shutdown margin is 6% $\Delta \rho$. Below 500°F the required shutdown margin decreases linearly with temperature. The initial core inlet temperature of 450°F was selected to demonstrate the adequacy of the shutdown margin in the temperature range where its magnitude is decreasing. This is a representative cold leg temperature. Analysis at other initial cold leg temperatures below 500°F will produce results and parameter trends similar to those presented here. Initially two reactor coolant pumps are assumed to be operating, as allowed in Mode 3. The initial pressurizer pressure of 830 psia falls within the range of normal Mode 3 operating procedures. The SIAS setpoint is set at 430 psia, 400 psi below the initial pressurizer pressure, the maximum offset allowed by technical specifications. This and the high initial pressurizer water volume have the effect of delaying SIAS actuation since SIAS generally occurs after the pressurizer empties. The technical specification shutdown margin at 450°F is 5.1% $\Delta \rho$. Since the reactor is 1% subcritical initially, a CEA worth at trip of 4.1% $\Delta \rho$ is assumed. The moderator and Doppler reactivity coefficients corresponding to the end of equilibrium cycle, self-generated plutonium recycle (SGR) are employed. For the purpose of conservatism the moderator reactivity coefficients correspond to the condition of no initial boron in the core.

C. Results

Case 1: Large Steam Line Break During Mode 3 Operation with Concurrent Loss of Offsite Power (SLBM3LOP)

The dynamic behavior of the salient NSSS parameters following the SLBM3LOP is presented in Figures 1-1 through 1-16. Table 1-2 summarizes the major events, times, and results for this transient.

Concurrent with the steam line break, a loss of offsite power occurs. At this time an actuation signal for the emergency diesel generators is initiated. Also at this time, the CEDM coils are assumed to lose power and, after a 0.34 second coil decay delay, the CEAs begin to drop into the core. At 21.3 seconds the steam generator pressure falls below the main steam isolation signal (MSIS) setpoint of 223 psia. This results in the generation of MSIS at 22.3 seconds, which initiates closure of the MSIVs and MFIVs. The MSIVs close by 26.9 seconds. The MFIVs close by 31.9 seconds.

During the first 500 seconds of the transient the pressurizer has not yet emptied and pressurizer pressure remains above the SIAS setpoint of 430 psia. Hence no safety injection flow and no boron reaches the RCS during this time.

AFAS is assumed to be actuated soon after the MSIVs close. Auxiliary feedwater is assumed to enter the steam generators after the level falls below the 80% high level setpoint; i.e., at 118 seconds. The pressure difference between the two steam generators remains below the analysis setpoint of 325 psid during the transient. Hence there is no automatic isolation of auxiliary feedwater to the affected steam generator.

At 500 seconds the transient reactivity is -2.1%, which indicates there is still a significant margin to recriticality. This margin will continue to decrease as the affected steam generator continues to blow down and the RCS continues to cool. After the pressurizer empties the RCS pressure is expected to fall more rapidly resulting in SIAS and subsequent inflow of boron into the RCS. Alternately SIAS may be manually actuated by the operator. In either case, after the inflow of boron into the RCS, the margin to recriticality is expected to increase.

Eventually, the affected steam generator is expected to blow down to atmospheric pressure. This would terminate further RCS cooldown. Even assuming the limiting case, where the affected steam generator has depressurized to atmospheric pressure and no safety injection boron has reached the RCS, the core will remain subcritical with a margin to criticality of no less than -0.4% $\Delta\rho$.

The discontinuity seen in some of the parameter plots at about 470 seconds (e.g., Figures 1-6 and 1-8) is due to safety injection tank (SIT) flow into the RCS for a short period of time. A SIT injection gas cover pressure of 608 psia was used in the analysis. The effect of this is small since no credit was taken for the SIT boron in the analysis.

The minimum DNBR remains above 10 during this transient. At a maximum of 30 minutes, the operator, via the appropriate emergency procedure, initiates plant cooldown by the manual control of the atmospheric dump valves. Shutdown cooling is initiated when the RCS reaches shutdown cooling entry conditions.

Case 2: Large Steam Line Break During Mode 3 Operation with Offsite Power Available (SLBM3)

The dynamic behavior of the salient NSSS parameters following the SLBM3 is presented in Figures 1-17 through 1-32. Table 1-3 summarizes the major event, times, and results for this transient.

At 24.3 seconds after the initiation of the steam line break, the steam generator pressure drops below the low steam generator pressure trip and MSIS setpoint of 223 psia. The reactor trip breakers open at 25.45 seconds. After a 0.34 second coil delay, the CEAs begin to drop into the core at 25.8 seconds. The MSIS initiates closure of the MSIVs and MFIVs. The MSIVs close by 29.9 seconds. The MFIVs close by 34.9 seconds.

During the first 500 seconds of the transient, as in Case 1, the pressurizer has not yet emptied and pressurizer pressure remains above the SIAS setpoint of 430 psia. Hence no safety injection flow and no safety injection boron reaches the RCS during this time.

AFAS is assumed to be actuated soon after the MSIVs close. Auxiliary feedwater is assumed to enter the steam generators after the level falls below the 80% high level setpoint, i.e., at 115 seconds. The pressure difference between the two steam generators stays below the analysis setpoint of 325 psid during the transient. Hence there is no automatic isolation of auxiliary feedwater to the affected steam generator.

At 500 seconds the transient reactivity is -1.9%, which indicates there is still a significant margin to criticality. This margin will continue to decrease as the affected steam generator continues to blow down and the RCS continues to cool. After the pressurizer empties the RCS pressure is expected to fall more rapidly resulting in SIAS and subsequent inflow of boron into the RCS. Alternately SIAS may be manually actuated by the operator. In either case, after the inflow of boron into the RCS, the margin to recriticality is expected to increase. Eventually, the affected steam generator is expected to blow down to atmospheric pressure (T=212°F). This would terminate further RCS cooldown. Even assuming the limiting case, where the affected steam generator has depressurized to atmospheric pressure and no safety injection boron has reached the RCS, the core will remain subcritical with a margin to criticality of no less than -0.4% $\Delta\rho$.

The discontinuity seen in some of the parameter plots at about 390 and 450 seconds (e.g., Figures 1-22 and 1-24) is due to safety injection tank (SIT) flow into the RCS for a short period of time. A SIT injection gas cover pressure of 608 psia was used in the analysis. The effect of this is small since no credit was taken for the SIT boron in the analysis.

The minimum DNBR remains above 10 during the transient. At a maximum of 30 minutes, the operator, via the appropriate emergency procedure, initiates plant cooldown. Shutdown cooling is initiated when the RCS reaches shutdown cooling entry conditions.

1.4 Conclusion

For the large steam line break during Mode 3 operation for reactor cold leg temperatures less than 500°F with or without a loss of offsite power, and in combination with a single failure the shutdown margin is sufficient to prevent a post trip return to power.

TABLE 1-1

ASSUMPTIONS AND INITIAL CONDITIONS FOR LARGE STEAM LINE BREAKS DURING MODE 3 OPERATION WITH AND WITHOUT CONCURRENT LOSS OF OFFSITE POWER (SLBM3LOP & SLBM3)

Parameters	Assumed Value
Initial Reactivity	0.99
Initial Core Inlet Coolant Temperature, F	450
Initial Core Mass Flow Rate, 10^6 lbm/hr (2 RCPs)	91.1
Initial Pressurizer Pressure, psia	830
Initial Pressurizer Water Volume, ft^3	1100
Doppler Coefficient Multiplier	1.15
Moderator Coefficient Multiplier	1.10
Axial Shape Index	+.3
CEA Worth at Trip, 10^{-2} $\Delta\rho$	-4.1
Initial Steam Generator Inventory, lbm	311,000
Core Burnup	End of Cycle
Blowdown Fluid	Saturated Steam
Blowdown Area for Each Steam Line, ft^2	1.283

TABLE 1-2

SEQUENCE OF EVENTS FOR A LARGE STEAM LINE BREAK DURING MODE 3 OPERATION WITH CONCURRENT LOSS OF OFFSITE POWER (SLBM3LOP)

Time (Sec)	Event	Setpoint or Value
0.0	Steam line break and loss of offsite power occur. Holding coils lose power.	--
21.3	Steam generator pressure reaches main steam isolation signal (MSIS) analysis setpoint, psia	230
22.3	MSIS generated	--
26.9	MSIVs completely closed	--
31.9	MFIVs completely closed	--
500	Transient reactivity, $10^{-2}\ \Delta\rho$	-2.1
> 500	Pressurizer empties	--
> 500	Pressurizer pressure reaches safety injection actuation signal (SIAS) analysis setpoint, psia	430
> 500	SIAS generated	--
> 500	Voids begin to form in reactor vessel upper head	--
> 500	Safety injection flow begins	--
> 500	Safety injection boron begins to reach reactor core	--
1800	Operator initiates cooldown	--

TABLE 1-3

SEQUENCE OF EVENTS FOR A LARGE STEAM LINE BREAK DURING MODE 3 OPERATION WITH OFFSITE POWER AVAILABLE (SLBM3)

Time (Sec)	Event	Setpoint or Value
0.0	Steam line break occurs	--
24.3	Steam generator pressure reaches main steam isolation signal (MSIS) analysis setpoint and low steam generator pressure trip setpoint, psia	230
25.3	Low steam generator pressure trip signal and MSIS generated	--
25.45	Reactor trip breakers open	--
29.9	MSIVs completely closed	--
34.9	MFIVs completely closed	--
500	Transient reactivity, $10^{-2}\ \Delta\rho$	-1.9
> 500	Pressurizer empties	--
> 500	Pressurizer pressure reaches safety injection actuation signal (SIAS) analysis setpoint, psia	430
> 500	SIAS generated	--
> 500	Voids begin to form in reactor vessel upper head	--
> 500	Safety injection flow begins	--
> 500	Safety injection boron begins to reach reactor core	--
1800	Operator initiates cooldown	--

2. STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE CONTAINMENT - MODE 4 OPERATION

2.1 Identification of Event and Causes

The steam system piping failure event during Mode 4 operation is evaluated to demonstrate the adequacy of the shutdown margin, as specified by technical specifications, to prevent degradation in fuel performance as a result of a post trip return to power.

The double ended rupture of a steam line upstream of the main steam isolation valve (MSIV) is considered. Reactor physics parameters (e.g., moderator and Doppler reactivity coefficients) for the end of equilibrium core, self-generated plutonium recycle (SGR) are assumed. The moderator reactivity coefficient corresponding to the condition of zero initial boron in the core is conservatively assumed. The more likely condition of boron existing in the RCS would cause the moderator reactivity to be less negative or even positive, thus reducing the reactivity increase during a steam line break.

Extrapolation of the results from Section 1 show that the shutdown margin in Mode 4 is sufficiently large to prevent a post trip return to power.

2.2 Sequence of Events and Systems Operation

Steam line breaks result in excessive cooldown of the reactor coolant which causes an increase in core reactivity due to the negative moderator and Doppler reactivity coefficients.

Mode 4 steam line breaks are initiated from a subcritical reactivity condition. The initial cold leg temperature may range from 210°F to 350°F. The sequence of events during a steam line break event will differ depending on the initial cold leg temperature and pressurizer pressure.

Detection of the cooldown is accomplished by the pressurizer and steam generator low pressure alarms and by the low steam generator water level alarm.

Reactor trip is provided by one of two available reactor trip signals. These are the low steam generator pressure and the high logarithmic power level trips. The depressurization of the affected steam generator may result in actuation of a main steam isolation signal (MSIS). This closes the MSIVs, isolating the unaffected steam generator from blowdown. If the level in either steam generator falls sufficiently low, an auxiliary feedwater actuation signal (AFAS) will occur. The pressurizer pressure may decrease to the point where a safety injection actuation signal (SIAS) is initiated. The introduction of safety injection boron upon SIAS causes core reactivity to decrease. Eventually the affected steam generator will depressurize to atmospheric pressure which will terminate the RCS cooldown.

2.3 Analysis of Effects and Consequences

The magnitude of the core reactivity increase during a steam line break during Mode 4 depends on the initial cold leg temperature. A steam line break initiated from 210°F or lower will result in negligible increases in core reactivity. A steam line break initiated from 350°F has the potential for moderate amounts of reactivity increase. Eventually, the affected steam generator will blow down to atmospheric pressure terminating further reactivity increase. If a SIAS is initiated, either automatically or by manual operator action, the core reactivity will decrease subsequent to the inflow of boron into the RCS. Even assuming the limiting case where the affected steam generator has depressurized to atmospheric pressure and no safety injection boron has reached the RCS, the technical specification shutdown margin will prevent a return to core criticality.

2.4 Conclusion

For the large steam line break during Mode 4 operation, the shutdown margin as specified by the technical specifications is sufficient to prevent a return to core criticality.

3. UNCONTROLLED CONTROL ELEMENT ASSEMBLY WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER CONDITION

Refer to CESSAR Section 15.4.1.

4. UNCONTROLLED CONTROL ELEMENT ASSEMBLY WITHDRAWAL FROM MODES 2 AND 3 SUBCRITICAL WITH 4 REACTOR COOLANT PUMPS OPERATING

4.1 Identification of Event and Causes

An uncontrolled sequential withdrawal of CEAs is assumed to occur as a result of a single failure in the Control Element Drive Mechanism (CEDM), Control Element Drive Mechanism Control System (CEDMCS), reactor regulating system, or as a result of operator error. This event is analyzed to justify reduced Technical Specification Shutdown Margin requirements in subcritical modes to below that required for Modes 1 and 2.

4.2 Sequence of Events and Systems Operation

The withdrawal of CEAs from subcritical conditions adds reactivity to the reactor core, causing both the core power level and the core heat flux to increase followed by corresponding increases in reactor coolant temperatures and reactor coolant system (RCS) pressure. The withdrawal of CEAs also produces a time dependent redistribution of core power. These transient variations in core thermal parameters may result in an approach to the specified acceptable fuel design limits (SAFDL), thereby requiring the protective action of the reactor protection system (RPS).

Reactivity Control

The reactivity insertion rate accompanying the uncontrolled CEA withdrawal is dependent primarily upon the CEA withdrawal rate and the CEA worth since, at subcritical conditions, the normal reactor feedback mechanisms do not occur until power generation in the core is large enough to cause changes in the fuel and moderator temperatures. The reactivity insertion rate determines the rate of approach to the fuel design limits. The uncontrolled CEA withdrawal transient from subcritical conditions is terminated by a high logarithmic power level trip or a CPC range trip.

Reactor Heat Removal

Following the cooldown phase in which the steam bypass control system is used, the shutdown cooling system (SCS) is manually actuated when the RCS temperature and pressure have been reduced to 350°F and 400 psia, respectively. This system provides sufficient cooling flow to cool the RCS to cold shutdown.

Primary System Integrity

The RCS pressure remains below the pressurizer pressure safety valve setpoint and remains less than 110% of design pressure. The pressurizer pressure control system and the pressurizer level control system are manually operated to regulate RCS pressure and coolant inventory during the cooldown phase.

Secondary System Integrity

The secondary system pressure increases following reactor trip and is limited by the steam generator safety valves. The atmospheric dump valves are used to cool the plant down to shutdown cooling entry conditions. The feedwater flowrate is in manual mode and is very low because it matches steam flow rates.

Table 4-1 gives the sequence of events for the limiting CEA withdrawal transient from subcritical conditions identified in paragraph 4.3.

4.3 Analysis of Effects and Consequences

A. Mathematical Model

The Nuclear Steam Supply System (NSSS) response to a CEA sequential withdrawal from subcritical or low power conditions was simulated using the CESEC computer program described in PVNGS FSAR Section 15.0. The thermal margin on DNBR in the reactor core was simulated using the TORC computer program described in PVNGS FSAR Section 15.0 with the CE-1 CHF correlation described in PVNGS FSAR Chapter 4.

B. Input Parameters and Initial Conditions

The initial conditions and NSSS characteristics assumed in this analysis have been determined to be the limiting conditions from which a CEA withdrawal could be initiated from subcritical modes. The range of initial conditions which were considered is limited both by Technical Specification LCO's and by the auxiliary trip functions in the CPC's. At 10^{-4} percent power, the CPC zero power bypass is automatically removed. If initial NSSS parameters or core coolant flow, temperature, pressure etc., are beyond specified values the CPC's will cause an immediate reactor trip upon bypass removal. The 10^{-4} percent power level setpoint at which the CPC bypass is automatically removed is well below the setpoint of the high logarithmic power level trip. A trip generated at this power level would cause a decrease in fission power before the point of adding sensible heat flux is reached, thereby precluding a challenge to the SAFDLs.

Parametric studies of initial conditions which would not generate an immediate CPC trip were performed. The initial conditions which resulted in the most adverse transient are presented in Table 4-2.

All control element assemblies are initially assumed to be fully inserted and the initial K_{eff} is 0.91. The reactivity insertion rate is conservatively taken to be 3.23×10^{-4} $\Delta\rho$ /second which is greater than the maximum differential worth of the highest worth CEA bank.

C. Results

The dynamic behavior of important NSSS parameters following a CEA withdrawal from subcritical conditions is presented in Figures 4-1 through 4-5.

The withdrawal of CEA's from subcritical conditions gradually reduces the amount by which the core is shut down. During this time, subcritical multiplication causes core power to increase. The reactor reaches critical at 278 seconds. Following this, core power rises at a rate which increases as the CEA's continue to withdraw.

A reactor trip on high logarithmic power is generated before core power reaches the point of adding sensible heat. Due to the rapid rate of power increase at the time of trip generation and the effect of continued CEA withdrawal until the trip breakers open, a brief power excursion occurs past the point of adding sensible heat.

The CEA's begin dropping into the core at 293.7 seconds terminating the power escalation with a hot channel minimum DNBR greater than 2.0.

4.4 Conclusions

The uncontrolled CEA withdrawal from a subcritical condition event meets general design criteria 25 and 20 as specified in SRP 15.4.1. These criteria require that the specified acceptable fuel design limits are not exceeded and that protection system action is initiated automatically. The transient terminates with a hot channel minimum DNBR greater than 1.19 and the peak fuel centerline temperature during the transient is less than 1230°F.

TABLE 4-1

SEQUENCE OF EVENTS FOR THE SUBCRITICAL CEA WITHDRAWAL EVENT

Time (Sec)	Event	Setpoint or Value
0.0	Withdrawal of CEA's - Initiating Event	--
292.8	Core Power reaches High Logarithmic power level reactor trip analysis setpoint, percent of design power	2.7×10^{-2}
293.2	High Logarithmic Power Level Trip Signal Generated	--
293.4	Trip Breakers Open	--
293.7	Maximum Core Power, % of Design Power	63%
293.9	Maximum Core Average Heat Flux, % of Full Power Heat Flux	7.8%
294.0	Minimum DNBR	2.0

TABLE 4-2

ASSUMPTIONS AND INITIAL CONDITIONS FOR THE SUBCRITICAL CEA WITHDRAWAL ANALYSIS

Parameters	Assumed Value
Initial Fission power level, MWt	2.9×10^{-8}
Core inlet coolant temperature, °F	565.5
Core mass flowrate, 10^{-6} lb_m/h	142.1
Reactor coolant system pressure, psia	1785
One pin 3-D peaking factor, with uncertainty	9.0
Steam generator pressure, psia	1178
Moderator temperature coefficient, 10^{-4} $\Delta\rho$ /°F	+0.5
Doppler coefficient multiplier	.85
CEA reactivity addition rate, 10^{-4} $\Delta\rho$/sec	3.23
CEA Worth on trip, 10^{-2} $\Delta\rho$	3.79
Steam bypass control system	Automatic

5. UNCONTROLLED CONTROL ELEMENT ASSEMBLY WITHDRAWAL FROM MODES 3, 4 AND 5 WITH LESS THAN 4 REACTOR COOLANT PUMPS OPERATING

5.1 Identification of Event and Causes

Refer to Section 3.

5.2 Sequence of Events and Systems Operation

This event proceeds the same as the event in Section 4, the CEAW for 4 Reactor Coolant Pumps Operating. A trip is generated by the CPCs when the zero power bypass is automatically removed at 10^{-4}% power since less than four pumps are in operation. This causes the shutdown of the reactor prior to the point of adding sensible heat flux.

5.3 Analysis of Effects and Consequences

Due to the prompt CPC trip at 10^{-4}% power the consequences of this event are less adverse than for the CEAW presented in Section 3.

6. STARTUP OF AN INACTIVE REACTOR COOLANT PUMP

6.1 Identification of Event and Causes

The Startup of an Inactive Reactor Coolant Pump (SIRCP) is presented here with respect to potential loss of minimum required shutdown margin. This event is also evaluated with respect to RCS pressure and fuel performance criteria.

Administrative procedures govern the starting of RCPs and reduce the effects of RCP starts.

6.2 Sequence of Events and Systems Operation

SIRCP can either raise or lower core average coolant temperature. The average temperature can be lowered by increased heat transfer to the steam generators caused by increased core coolant flow and by colder primary system water in the steam generators being forced into the core. The core average temperature can be raised by increased heat transfer from the steam generators to the RCS as a result of increased core coolant flow and by hotter primary system water in the steam generators being forced into the core.

The SIRCP event which lowers the core average temperature (the cooldown event) combined with a negative isothermal temperature coefficient (ITC) produces a positive reactivity insertion. The SIRCP event which increases core average temperature (the heatup event), combined with a positive ITC produces an increase in RCS pressure and a positive reactivity insertion.

6.3 Analysis of Effects and Consequences

SIRCP can cause either a heatup or cooldown of the primary system depending on the primary to secondary ΔT.

SIRCP was examined in Modes 3 through 6, since plant operation with less than 4 RCPs running is only permitted in these modes.

A. Mathematical Models

The reactivity added to the core during a heatup or cooldown SIRCP event was determined using conservative isothermal temperature coefficients (ITCs) with the maximum uncertainty applied. These ITCs were used with the maximum core temperature increase or decrease to determine the maximum reactivity inserted during SIRCP. This reactivity insertion is compared to the minimum shutdown margin required by the technical specifications.

B. Input Parameters and Initial Conditions

The initial conditions considered for this event ranged from a positive to a negative temperature difference between the secondary and primary system. Primary system temperature higher than the secondary (a positive temperature difference) would result in cooling down the RCS. Secondary system temperature initially higher than the primary temperature (a negative temperature difference) would result in heating up the RCS. Cooling the RCS would increase reactivity if there is a negative ITC. Heating the RCS would increase reactivity and RCS pressure if there is a positive ITC.

To conservatively calculate the reactivity added to the core during SIRCP, the most negative or positive ITCs are used with uncertainties applied in the most conservative direction. The initial core average moderator temperature during SIRCP is assumed to be at the temperature corresponding to the most positive ITC for the heatup event, or the most negative ITC for the cooldown event.

The following assumptions are made:

1) Prior to SIRCP all reactor coolant pumps are off. Normally at least one RCP must be running (or one shutdown cooling train during shutdown cooling operation). The technical specifications allow operation without any pumps running for up to one hour. This assumption maximizes the change in temperature during SIRCP.

2) Following SIRCP the core average temperature either (1) drops to the temperature of the coldest steam generator, for the cooldown event, or (2) increases to the temperature of the hottest steam generator, for the heatup event. This conservatively bounds the maximum change in core temperature that can occur during this event, by ignoring coolant mixing that would occur in the reactor coolant system.

C. Results

The results show that the maximum temperature change during SIRCP when used with the most conservative ITCs does not result in a loss of the minimum required shutdown margin.

When the RCS is above the conditions requiring low temperature over pressure (LTOP) protection, the SIRCP event that results in a heatup of the RCS will not result in a peak pressure greater than 110% of design pressure. While the RCS is in the LTOP mode, the shutdown cooling system (SCS) relief valves will prevent violation of RCS integrity limits. (See PVNGS FSAR Section 5.2 for a general discussion of RCS integrity.)

Since shutdown margin is not lost during the event, there is no increase in heat flux and therefore no decrease in minimum DNBR.

6.4 Conclusions

The SIRCP does not result in a loss of shutdown margin. The increase in pressure during this event will not result in peak pressures above the applicable limits. There is no increase in core heat flux and therefore no decrease in minimum DNBR.

7. INADVERTENT DEBORATION

7.1 Identification of Event and Causes

The Inadvertent Deboration (ID) event is presented here with respect to time available for operator corrective action prior to the loss of minimum required shutdown margin. Fuel integrity is not challenged by this event.

The ID event may be caused by improper operator action or by a failure in the boric acid makeup flow path which reduces the flow of borated water to the charging pump suction. Either cause can produce a boron concentration of the charging flow which is below the concentration of the reactor coolant.

The ID event initiated during each of the six operational modes defined in the technical specifications was evaluated. This evaluation shows that MODE 4 (hot shutdown) results in the least time available for detection and termination of the event. This is because the shutdown margin requirement which will be specified by the technical specifications is at its minimum value in the lower temperature range of MODE 4 and the boron dilution time constant which drives the dilution rate is also small in MODE 4. This combination of a minimum shutdown margin and small time constant results in the fastest dilution rate and, therefore, yields the shortest time to a complete loss of shutdown margin.

Since boron dilution is conducted under strict procedural controls which specify limits on the rate and the magnitude of any required change in boron concentration, the probability of a sustained and erroneous dilution due to operator error is very low.

The indications and/or alarms available to alert the operators that a boron dilution event is occurring in each of the operational modes are outlined below.

1. The following control indications and corresponding pre-trip alarms are available for MODES 1 and 2: a high power or, for some set of conditions, a high pressurizer pressure trip in MODE 1 or a high logarithmic power level trip in MODE 2. Furthermore, a high T_{AVG} alarm may also occur prior to trip.

2. In MODES 3 and 4 with CEAs withdrawn, the high logarithmic power level trip and pre-trip alarm, and a high neutron flux alarm will provide an indication to alert the operator of an inadvertent boron dilution.

3. In MODES 3, 4, and 5 with CEAs fully inserted except the worst rod stuck out and in MODE 6, a high neutron flux alarm on the startup flux channels will provide indication of any boron dilution event.

4. In MODE 5 with the RCS partially drained for system maintenance, the startup flux channel alarm will provide indication of any boron dilution event. In this plant condition, administrative controls would allow operation of only one charging pump at a maximum rate of 44 gpm. Plant operating procedure will require that the power to the other two charging pumps be removed and their breakers locked out. This drained-down case is less limiting than the MODE 4 event presented below.

The operational procedure guidelines, in addition to these indications and/or alarms, will assure detection and termination of the boron dilution event before the shutdown margin is lost.

7.3 Sequence of Events and Systems Operation

The core is initially subcritical with shutdown margin at the minimum value consistent with the technical specification limit. An inadvertent deboration occurs which causes unborated water to be pumped into the RCS. The resulting decrease in RCS boron concentration adds positive reactivity to the core. Assuming dilution continues at the maximum possible rate, 50 minutes would elapse before the core becomes critical.

The success path is as follows:

Reactivity Control:

The operator is alerted to a decrease in the reactor coolant system (RCS) boron concentration either through a high neutron flux alarm on the startup flux channel, sampling, boronometer indications, or boric acid flow rate. The operator turns off the charging pump(s) and closes the letdown control valves in order to halt further dilution. Next, the operator increases the RCS boron concentration by implementing the emergency boration procedure for achieving cold shutdown boron concentration.

7.3 Analysis of Effects and Consequences

A. Mathematical Model

Assuming complete mixing of boron in the RCS, the rate of change of boron concentration during dilution is described by the following equation.

$$M\frac{dC}{dt} = -WC$$

Where: M = RCS mass

C = RCS boron concentration

W = Charging mass flow rate of unborated water

dC/dt is maximized by maximizing W and minimizing M. Assuming:

W = Constant, equal to the maximum possible value,

and choosing:

M = Constant, equal to the minimum value occurring during the boron dilution incident, the solution of Equation (1) can be written

$$C(t) = C(o)e^{-t/\tau} \qquad (2)$$

Where: $\tau = M/W$ = Boron dilution time constant
$C(o)$ = Initial boron concentration

The time T required to dilute to criticality is given by

$$T = \tau \ln \frac{C(o)}{C_{crit}} \qquad (3)$$

Where: C_{crit} = Critical boron concentration

B. Input Parameters and Initial Conditions

It is assumed that the inadvertent deboration proceeds at the maximum possible rate. For this to occur, all charging pumps must be on, the reactor makeup water tank must be aligned with the charging pump suction, a reactor makeup water pump must be on, letdown flow must be diverted from the volume control tank, and a failure in the boric acid makeup water flow path (e.g., flow control valve FV-210Y failing in the closed position) must terminate borated water flow to the charging pump suction.

Evaluation of ID events initiated during each of the six plant operational modes (defined in the technical specifications) shows that MODE 4 (hot shutdown) results in the shortest available time for detection and termination of the event. Therefore, the initial conditions and analysis parameters are chosen for the hot shutdown operational mode to minimize the interval from initiation of dilution to the time at which criticality is reached. The following are the analysis assumptions for the ID event:

1. Complete mixing of boron within the RCS is assumed.

2. The technical specification lower limit on shutdown margin for hot shutdown is assumed. The shutdown margin as specified in the technical specifications can vary as a function of reactor coolant cold leg temperature. The minimum value of shutdown margin at the technical specification lower limit of temperature range of MODE 4, i.e., 210°F, is 1% $\Delta \rho$.

3. The cold reactor coolant system volume, excluding pressurizer and surge line, is 12,016 ft^3. A conservatively low reactor coolant mass was assumed by using the cold RCS internal volume. Assuming the coolant temperature of 350°F, the technical specification upper limit for hot shutdown, the resulting mass is 667,927 lbm.

4. All three charging pumps are assumed to be on at their maximum rate; 44 gpm per pump, for a total of 132 gpm. The corresponding mass flow rate, assuming cold liquid flow, is 18.36 lbm/sec.

5. The critical boron concentration, with all rods in except the highest reactivity worth rod stuck out, and the inverse boron worth are 752 ppm and 65 ppm/% $\Delta \rho$, respectively, including uncertainties for the hot shutdown conditions. The initial subcritical boron concentration for the hot shutdown mode is found by adding the product of the inverse boron worth and the minimum shutdown margin (i.e., one percent) to the critical boron concentration. The resulting minimum initial boron concentration in MODE 4 is 817 ppm. Thus, the change in boron concentration from 1% $\Delta \rho$ subcritical to critical is 65 ppm.

The parameters discussed above are summarized in Table 7-1.

C. Results

Using conservative parameters as described above in Equation (3), the minimum possible time interval to dilute from 1.0% $\Delta\rho$ subcritical to criticality is 50 minutes. Given the numerous indications of improper operation and the high neutron flux alarm on the startup flux channel, sufficient time is available to assure detection of a boron dilution event at least 15 minutes prior to criticality. Boron dilution will then be terminated before loss of shutdown margin by the operator actions discussed in Section 7.2.

7.4 Conclusions

The inadvertent deboration event will result in acceptable consequences. Sufficient time is available for the operator to detect and to terminate an inadvertent deboration event if it occurs. Fuel integrity is not challenged during this event.

TABLE 7-1

ASSUMPTIONS FOR THE INADVERTENT DEBORATION ANALYSIS

Parameters	Assumed Value
Cold RCS Volume (excluding pressurizer surge line), ft^3	12,016
RCS Mass (excluding pressurizer and surge line), lbm	667,927
Volumetric Charging Rate, gpm	132
Mass Charging Rate, lbm/sec	18.36
Dilution Time Constant, τ , sec	36,380
Initial Boron Concentration - C(o), ppm	817
Critical Boron Concentration - C_{crit}, ppm	752

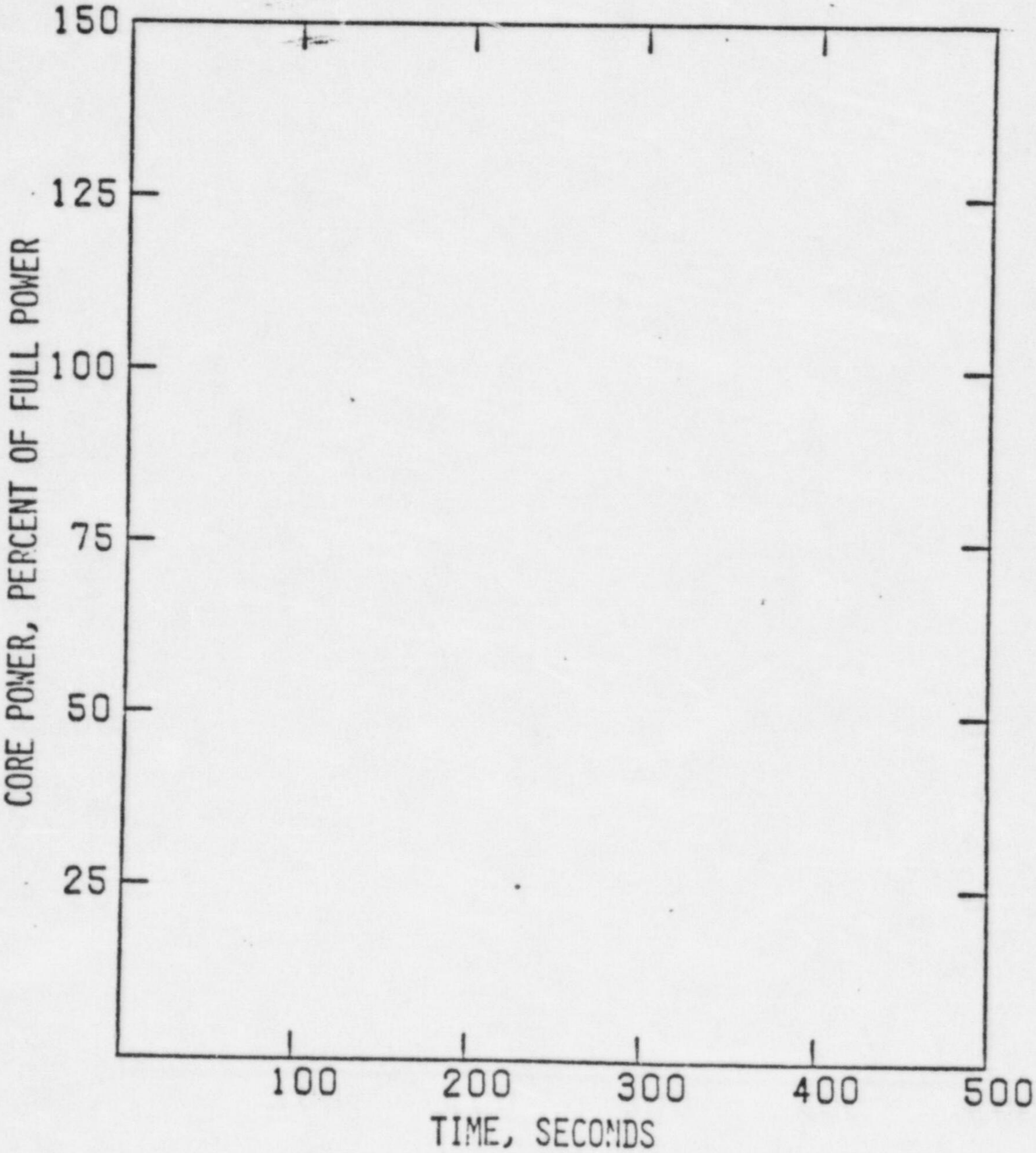
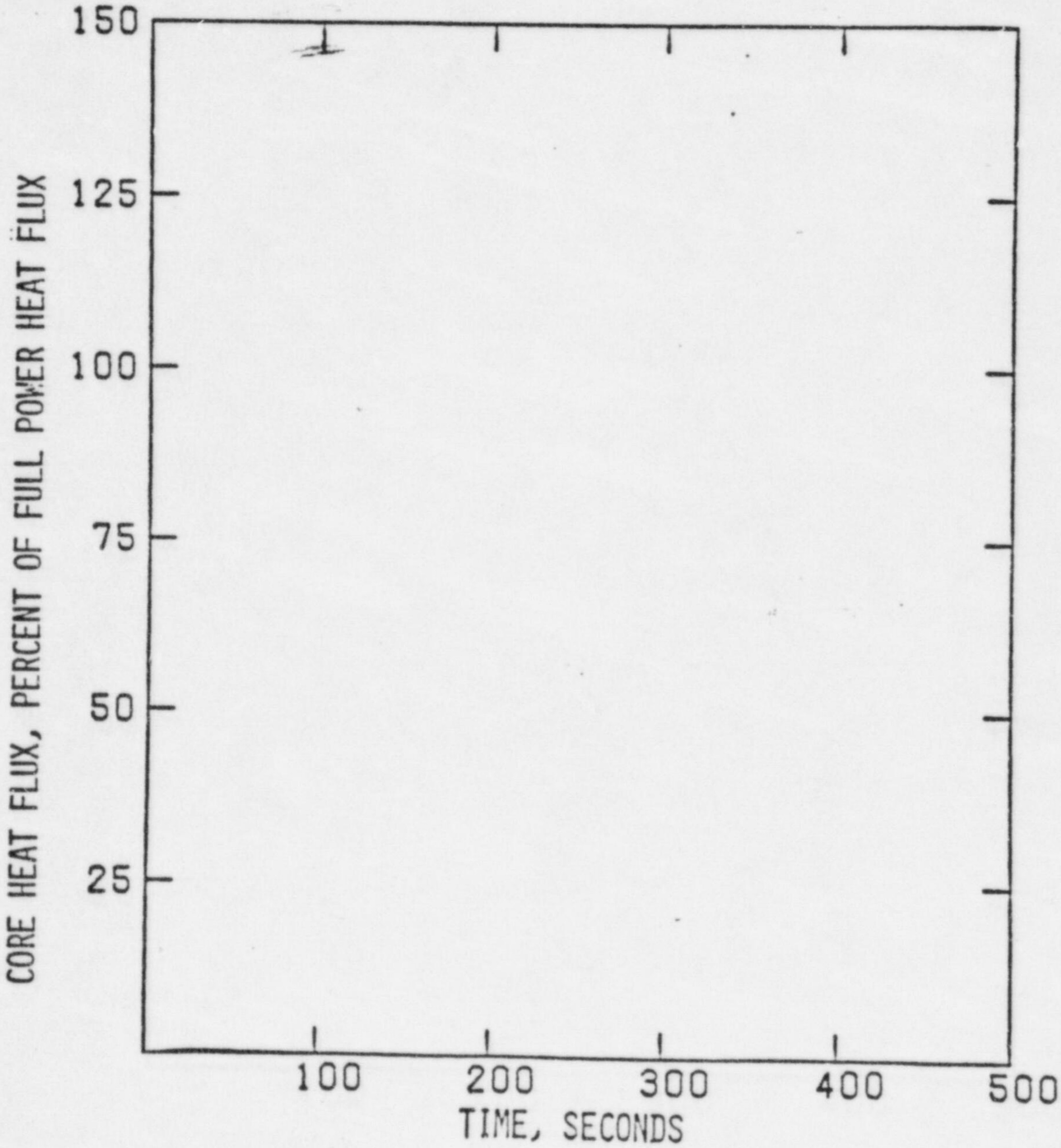

CORE POWER, PERCENT OF FULL POWER
150
125
100
75
50
25
100
200
300
400
500
TIME, SECONDS

Figure 1-1



150
125
100
75
50
25
CORE HEAT FLUX, PERCENT OF FULL POWER HEAT FLUX
100
200
300
400
500
TIME, SECONDS

Figure 1-2

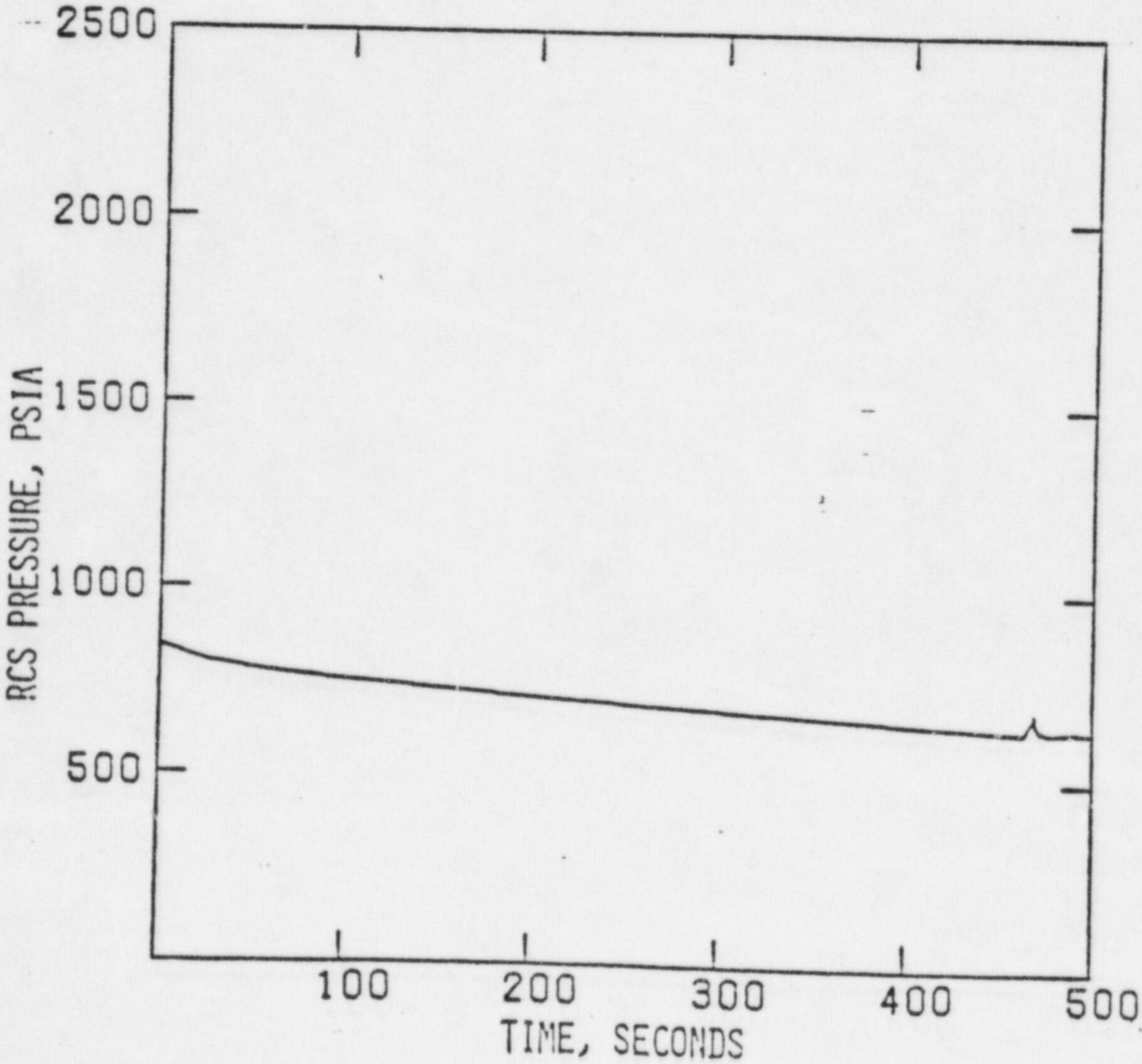

2500
2000
1500
1000
500
RCS PRESSURE, PSIA
100
200
300
400
500
TIME, SECONDS

Figure 1-3

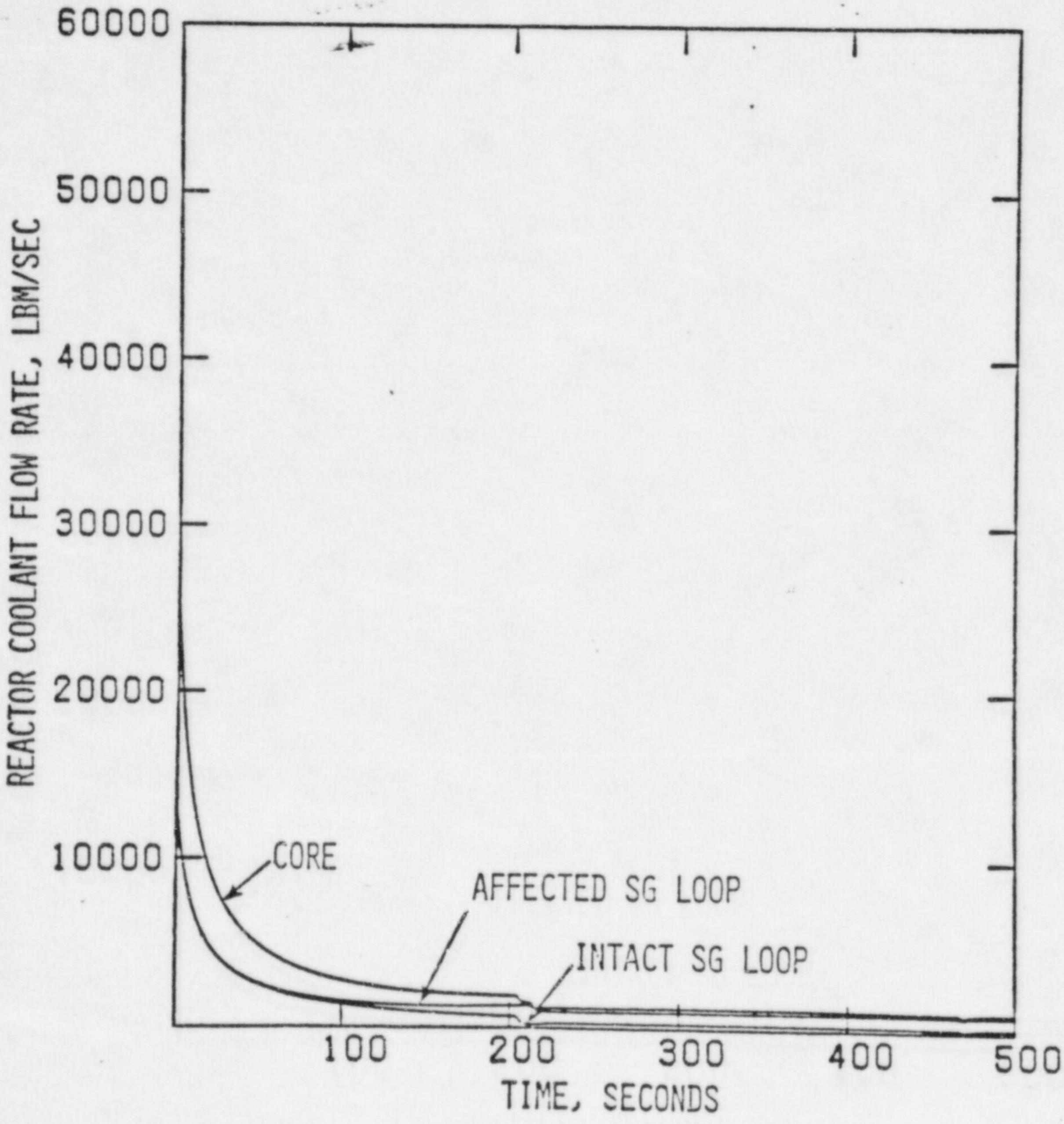
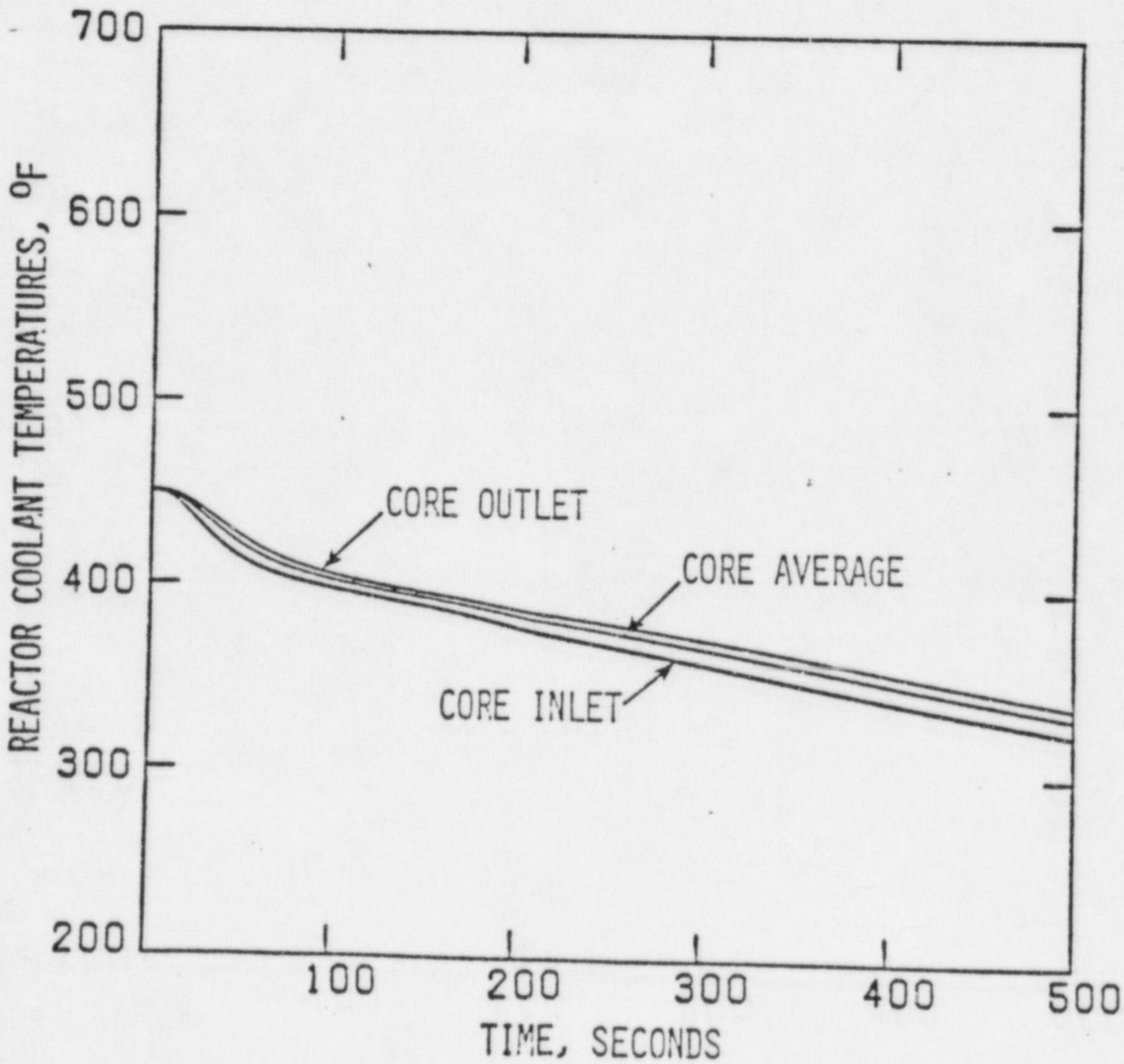
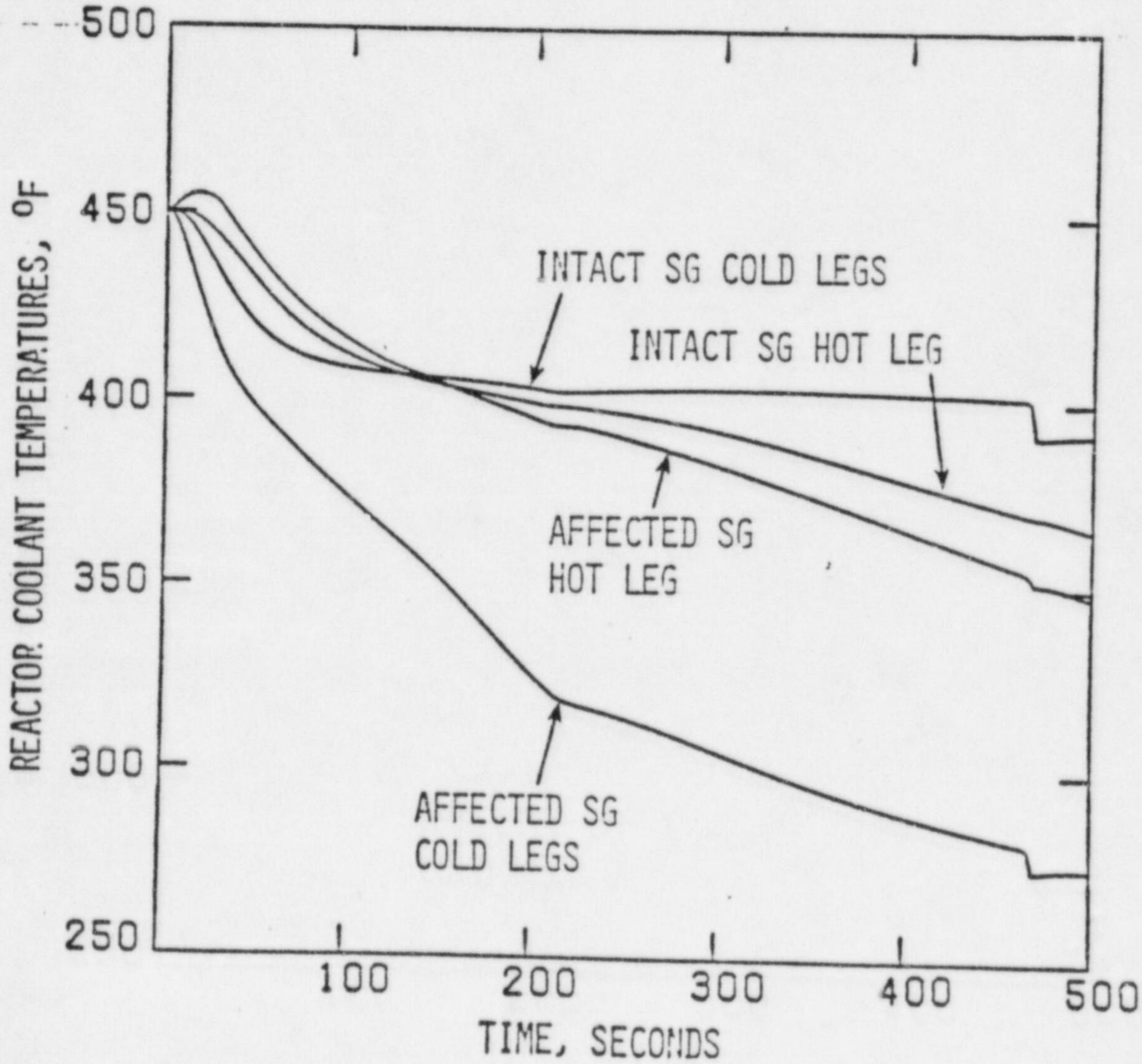

60000
50000
40000
30000
20000
10000
REACTOR COOLANT FLOW RATE, LBM/SEC
CORE
AFFECTED SG LOOP
INTACT SG LOOP
100
200
300
400
500
TIME, SECONDS

Figure 1-4



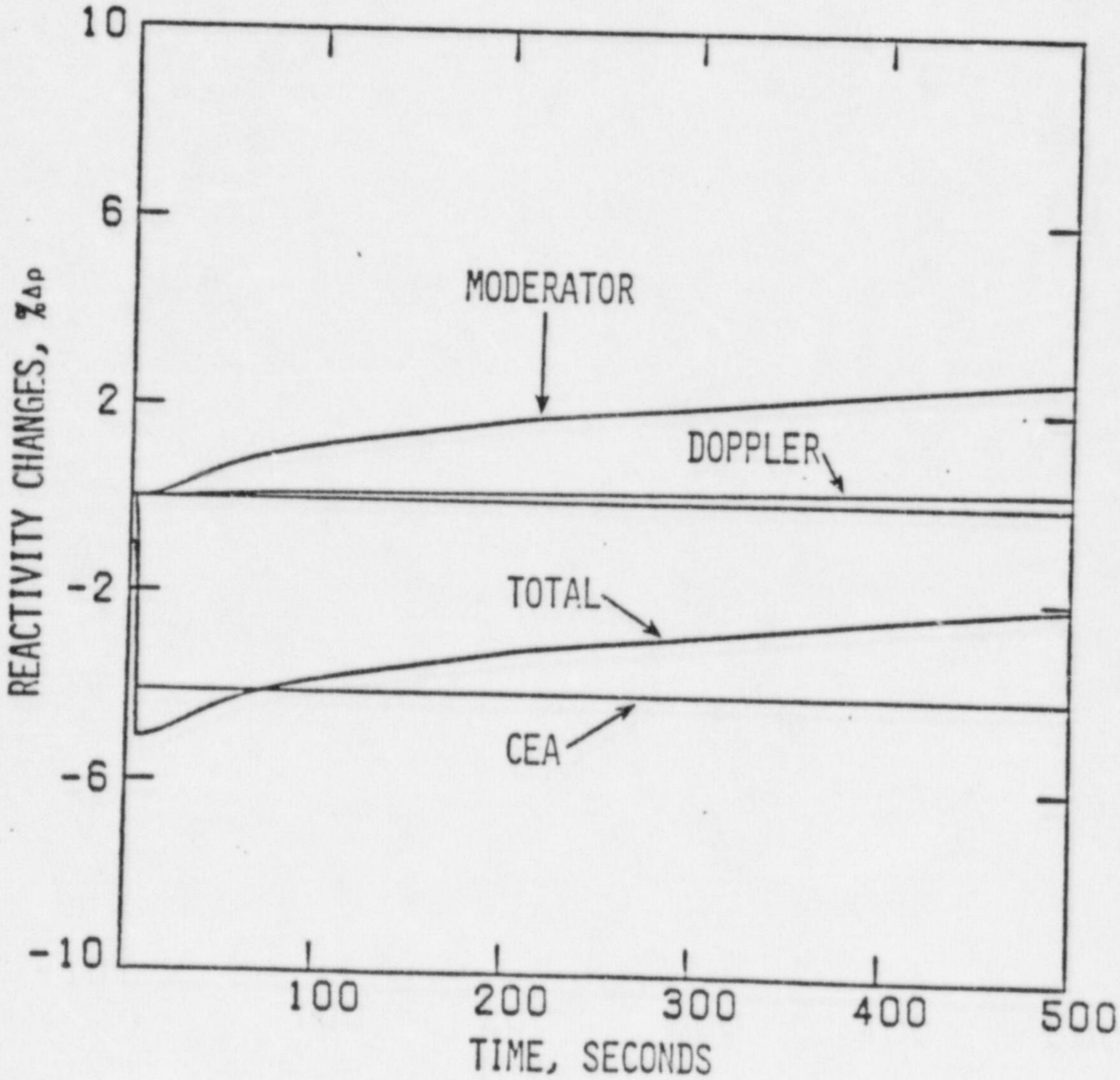
700
600
500
400
300
200
100
200
300
400
500
REACTOR COOLANT TEMPERATURES, °F
TIME, SECONDS
CORE OUTLET
CORE AVERAGE
CORE INLET

Figure 1-5



500
450
400
350
300
250
100
200
300
400
500
REACTOR COOLANT TEMPERATURES, °F
TIME, SECONDS
INTACT SG COLD LEGS
INTACT SG HOT LEG
AFFECTED SG
HOT LEG
AFFECTED SG
COLD LEGS

Figure 1-6



10
6
2
-2
-6
-10
REACTIVITY CHANGES, %Δρ
MODERATOR
DOPPLER
TOTAL
CEA
100
200
300
400
500
TIME, SECONDS

Figure 1-7

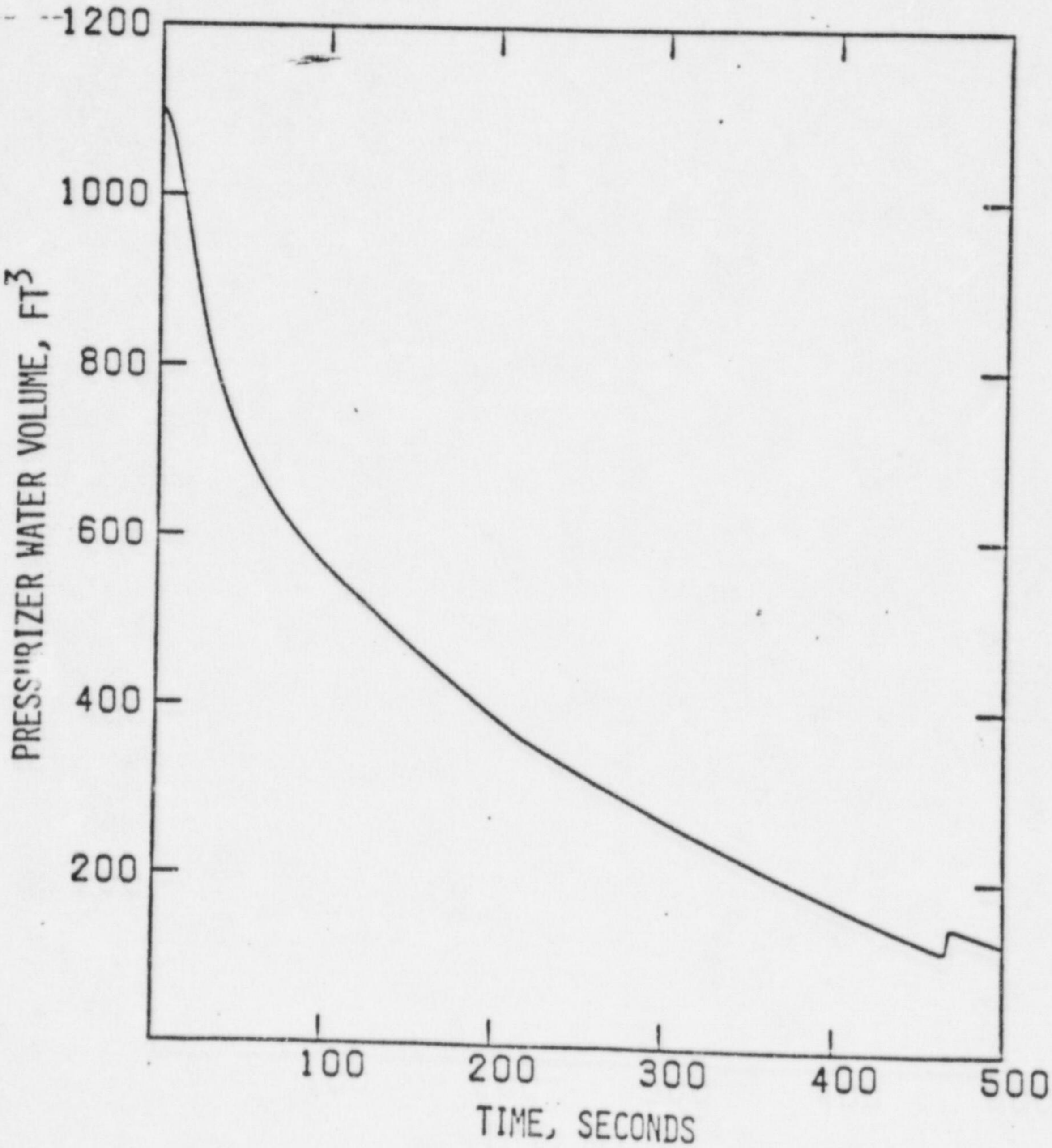
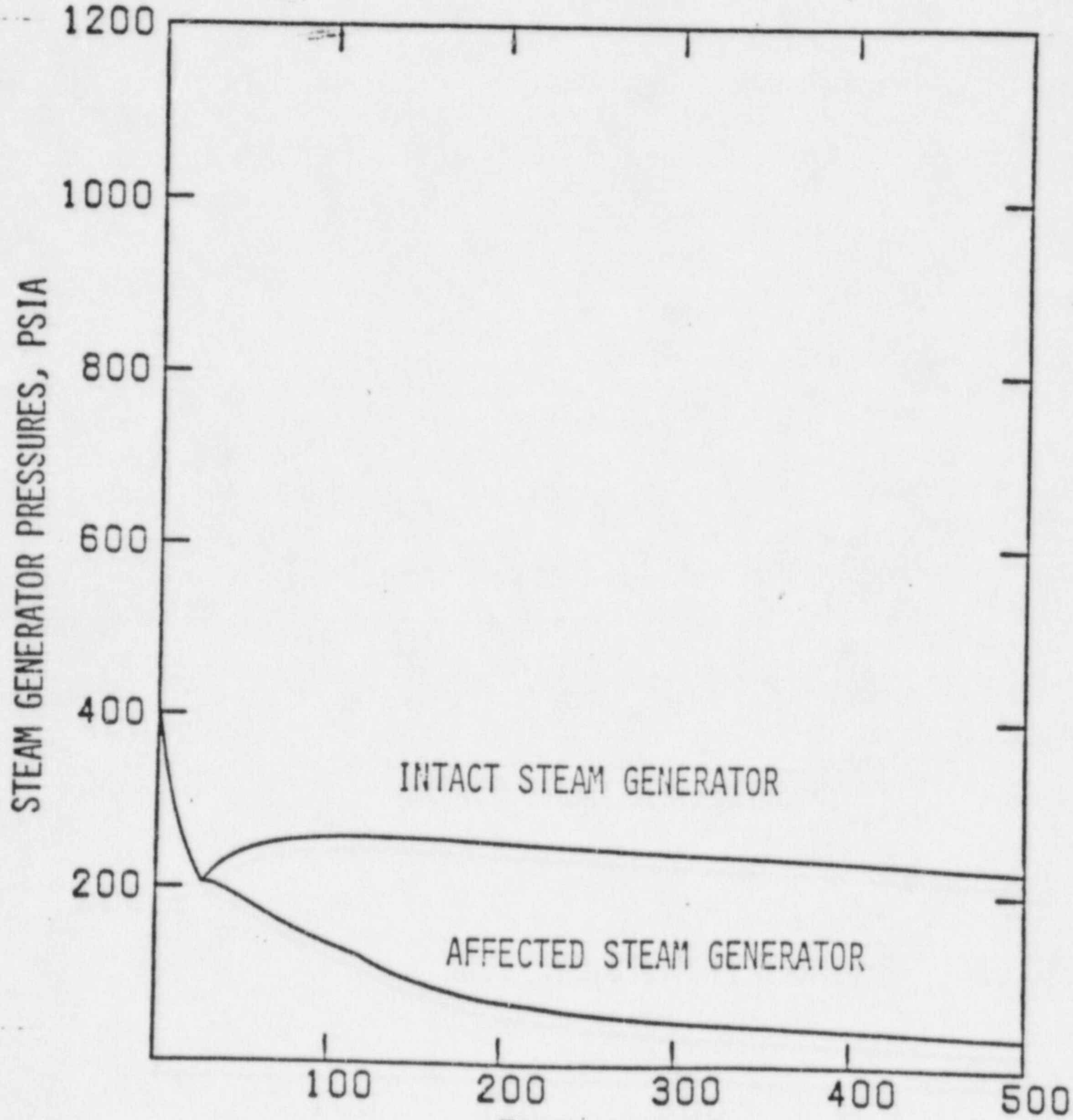
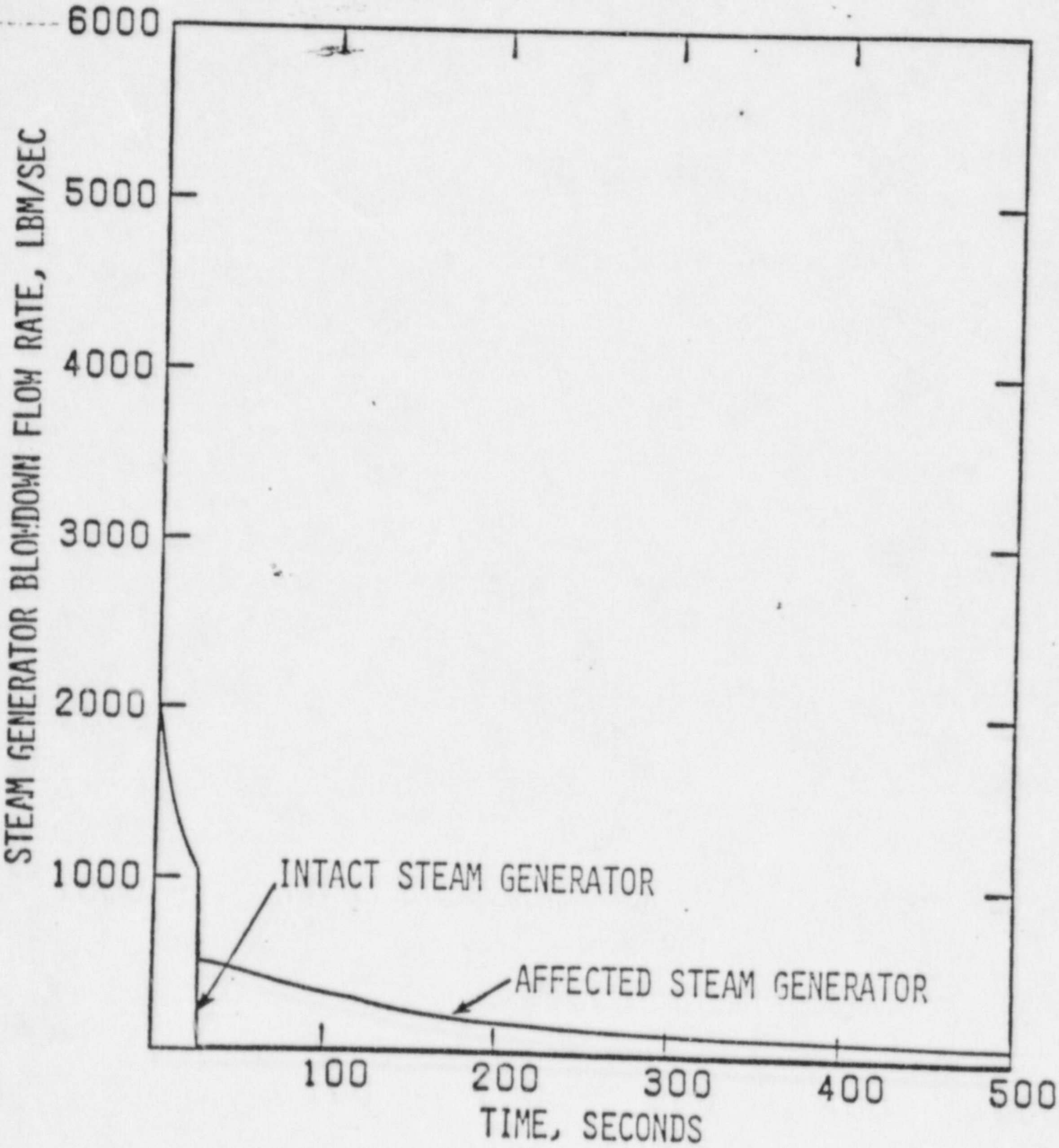

1200
1000
800
600
400
200
PRESSURIZER WATER VOLUME, FT3
100
200
300
400
500
TIME, SECONDS

Figure 1-8



1200
1000
800
600
400
200
STEAM GENERATOR PRESSURES, PSIA
INTACT STEAM GENERATOR
AFFECTED STEAM GENERATOR
100
200
300
400
500
TIME, SECONDS

Figure 1-9



6000
5000
4000
3000
2000
1000
STEAM GENERATOR BLOWDOWN FLOW RATE, LBM/SEC
INTACT STEAM GENERATOR
AFFECTED STEAM GENERATOR
100
200
300
400
500
TIME, SECONDS

Figure 1-10

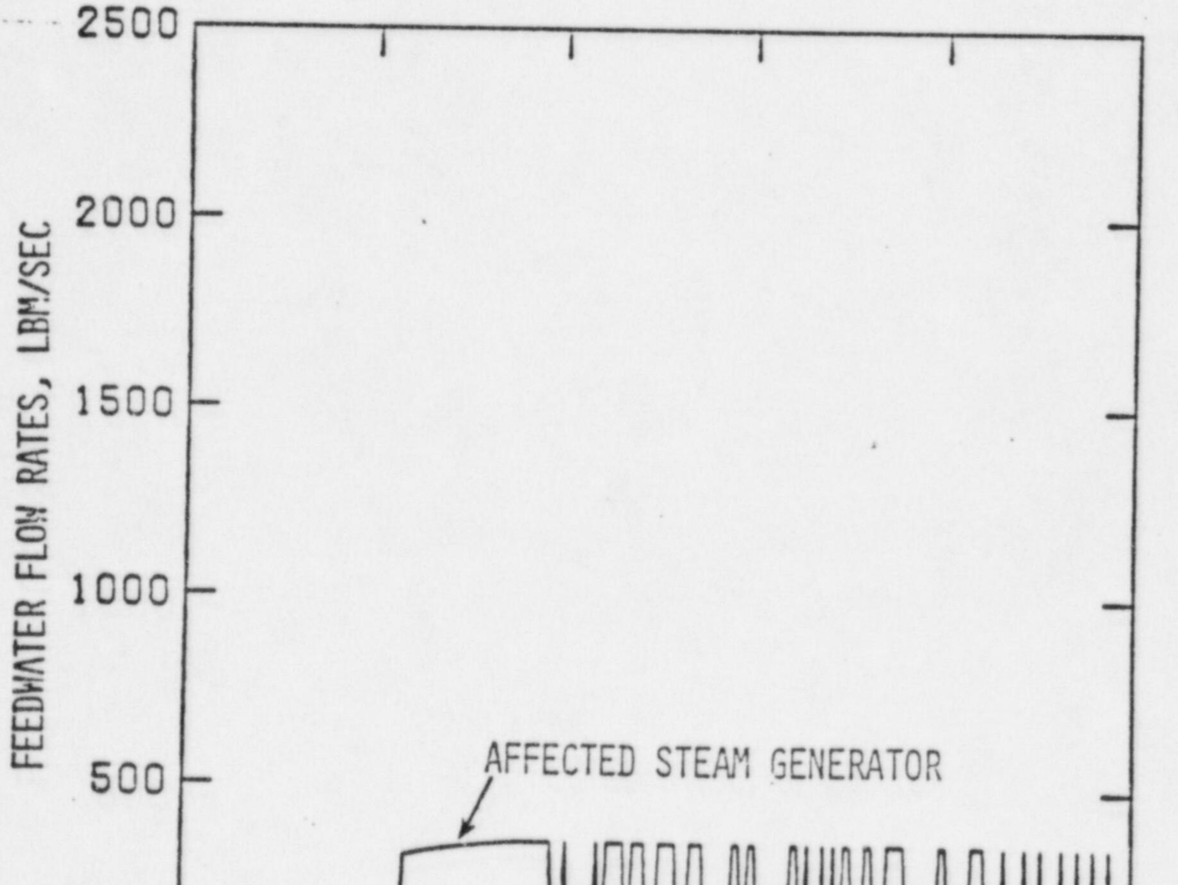

FEEDWATER FLOW RATES, LBM/SEC
2500
2000
1500
1000
500
AFFECTED STEAM GENERATOR
100
200
300
400
500
TIME, SECONDS

Figure 1-11

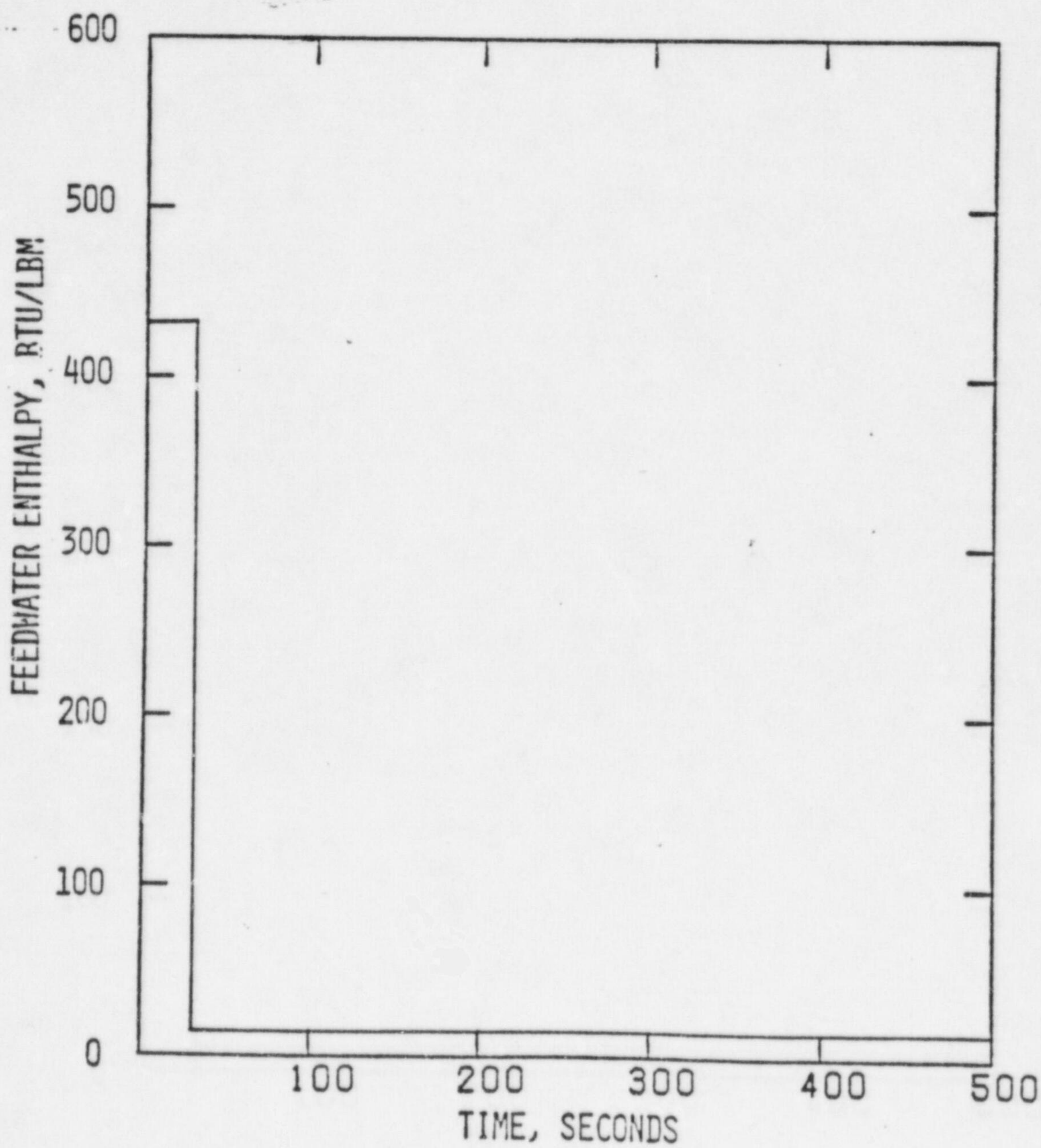

600
500
400
300
200
100
0
FEEDWATER ENTHALPY, BTU/LBM
100
200
300
400
500
TIME, SECONDS

Figure 1-12

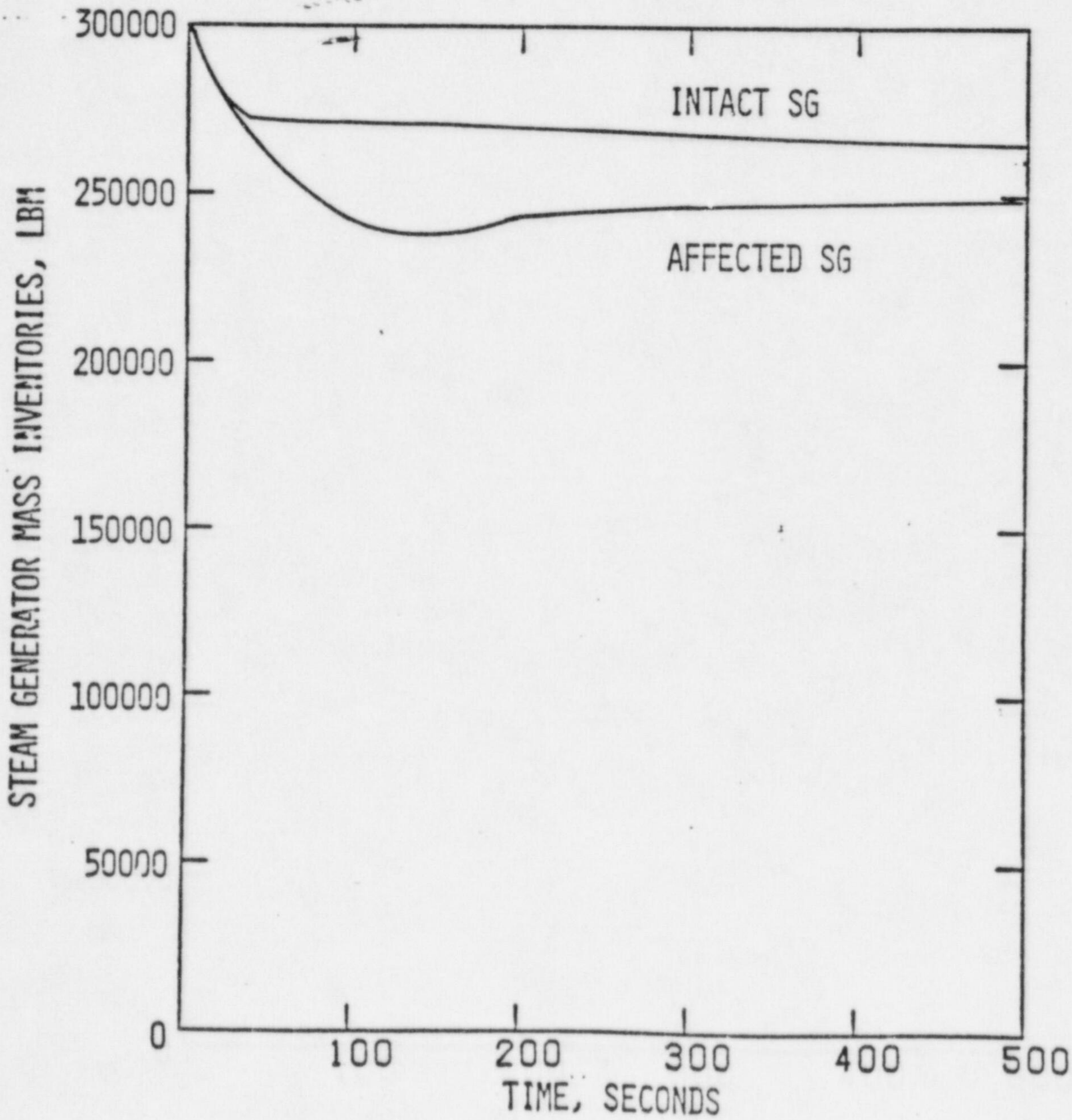

300000
250000
200000
150000
100000
50000
0
STEAM GENERATOR MASS INVENTORIES, LBM
INTACT SG
AFFECTED SG
100
200
300
400
500
TIME, SECONDS

Figure 1-13

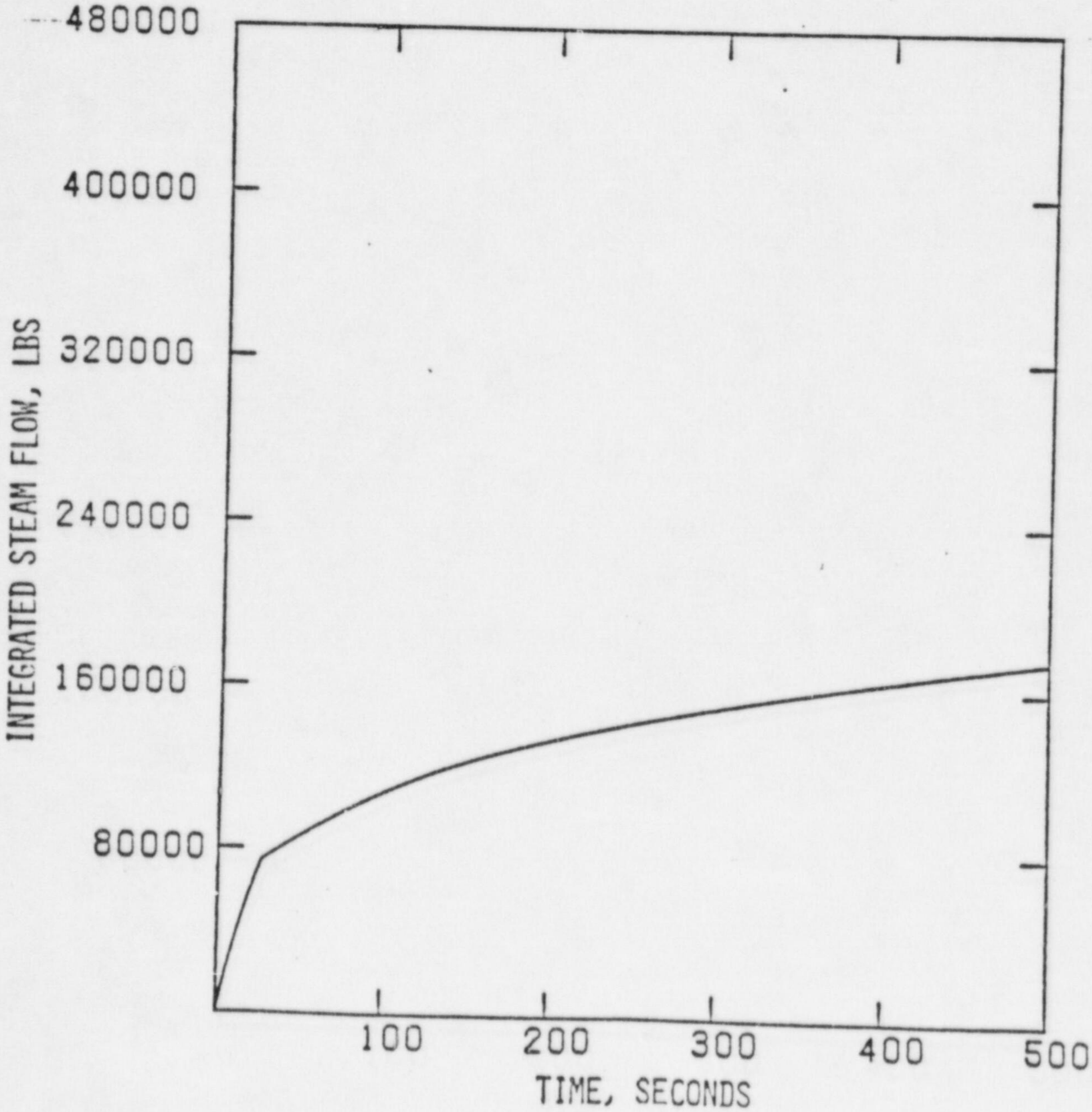

480000
400000
320000
240000
160000
80000
INTEGRATED STEAM FLOW, LBS
100
200
300
400
500
TIME, SECONDS

Figure 1-14

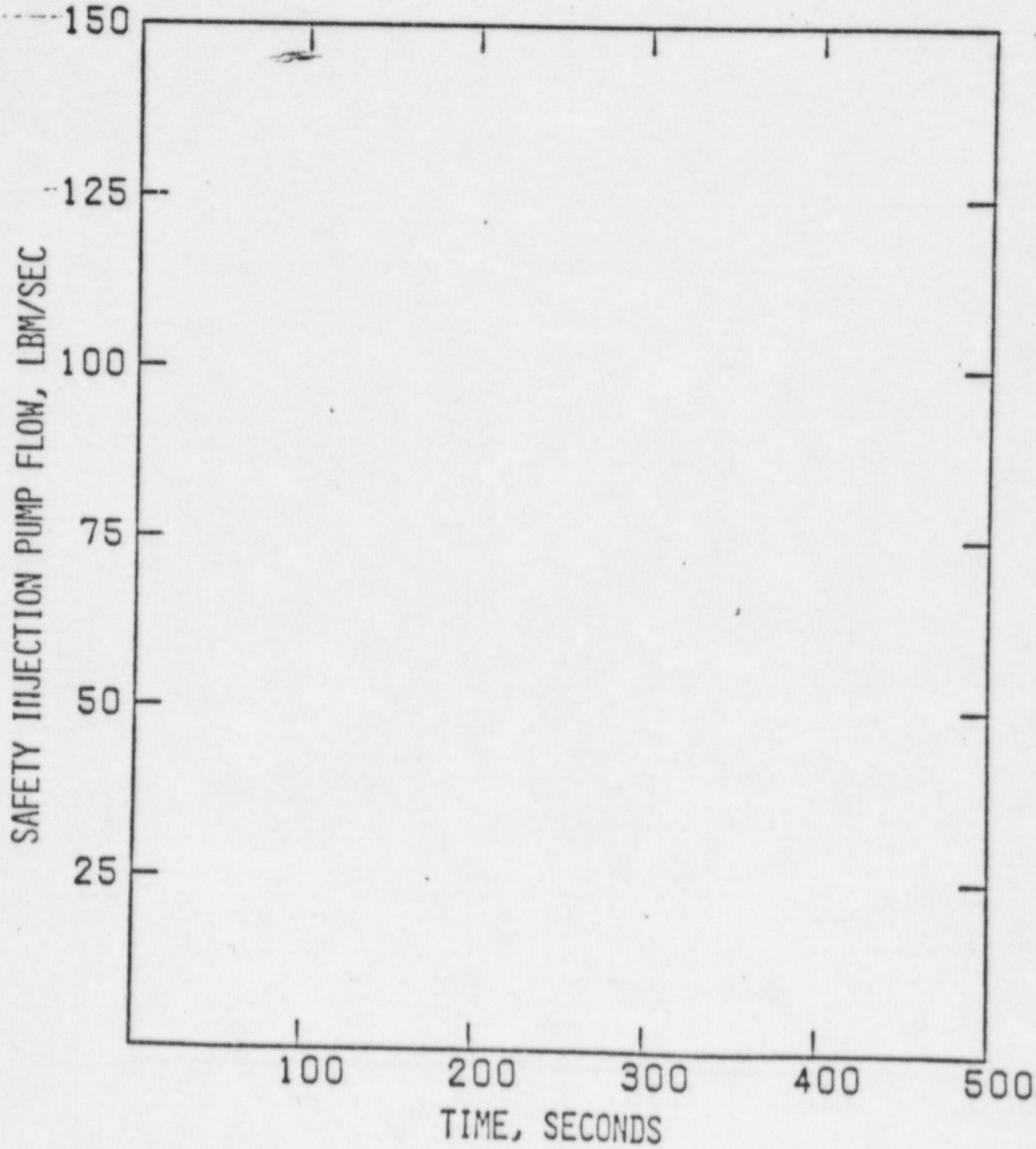
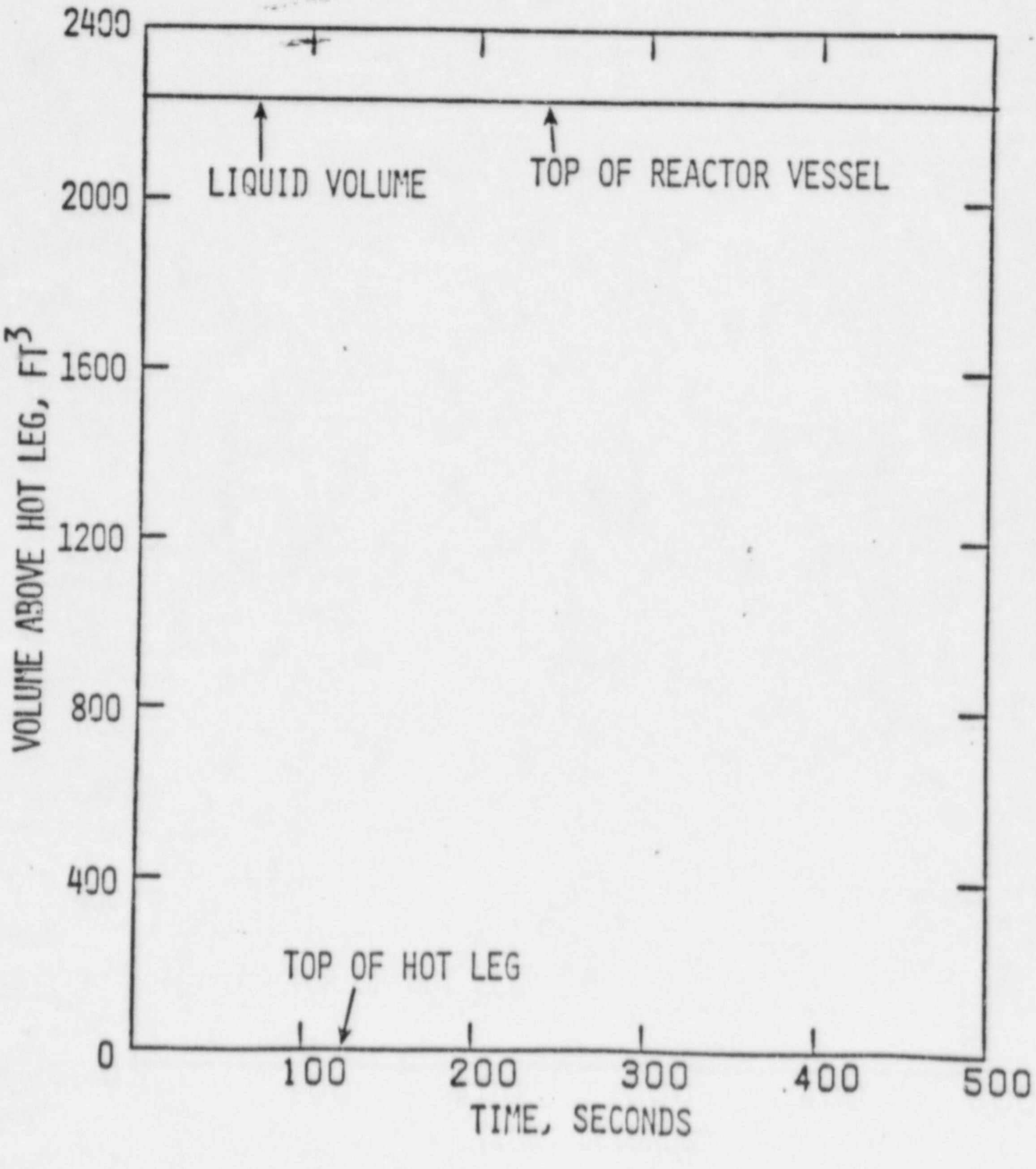
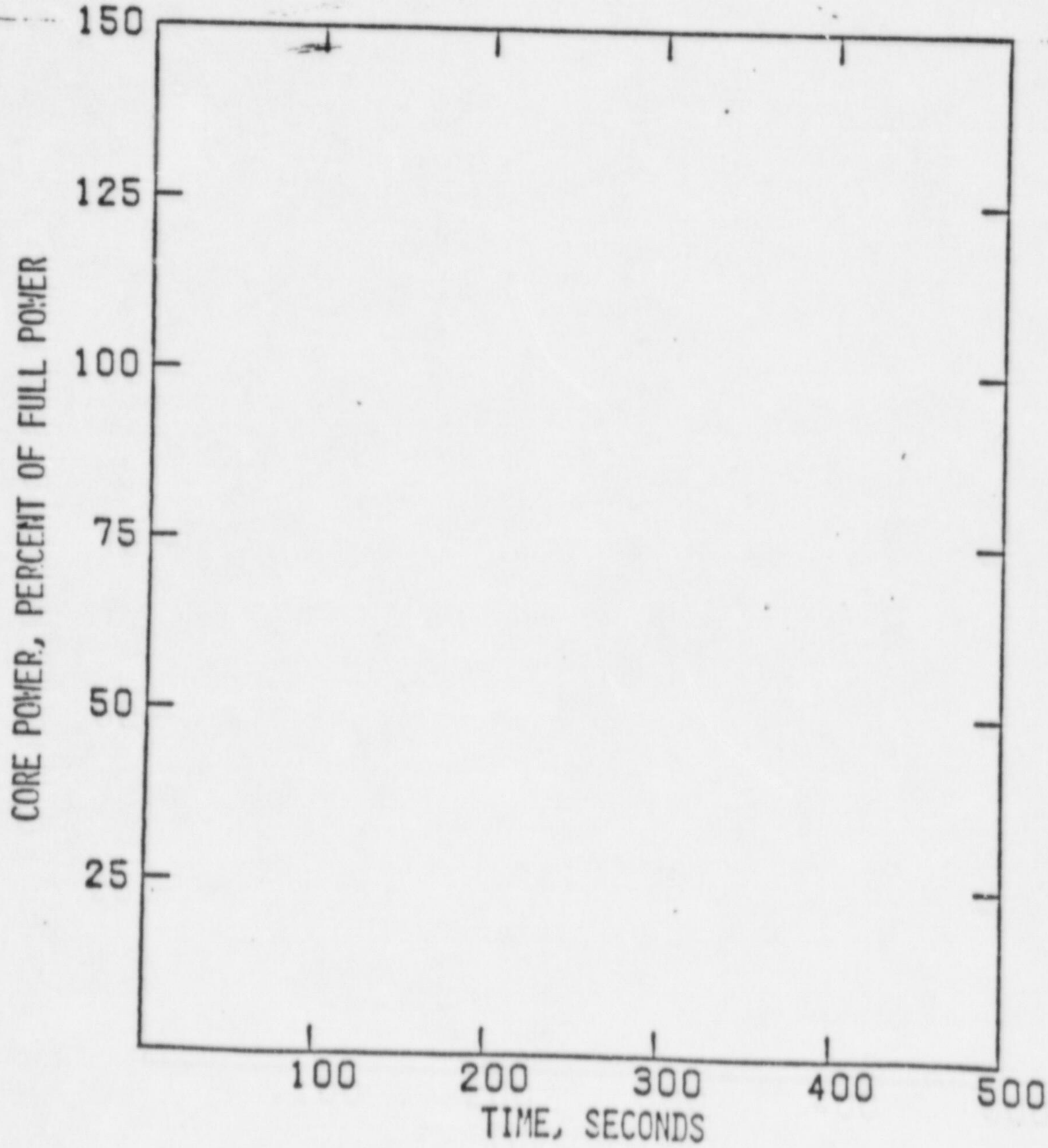

SAFETY INJECTION PUMP FLOW, LBM/SEC
150
125
100
75
50
25
100
200
300
400
500
TIME, SECONDS

Figure 1-15



2400
2000
1600
1200
800
400
0
100
200
300
400
500
VOLUME ABOVE HOT LEG, FT3
TIME, SECONDS
LIQUID VOLUME
TOP OF REACTOR VESSEL
TOP OF HOT LEG

Figure 1-16



150
125
100
75
50
25
CORE POWER, PERCENT OF FULL POWER
100
200
300
400
500
TIME, SECONDS

Figure 1-17

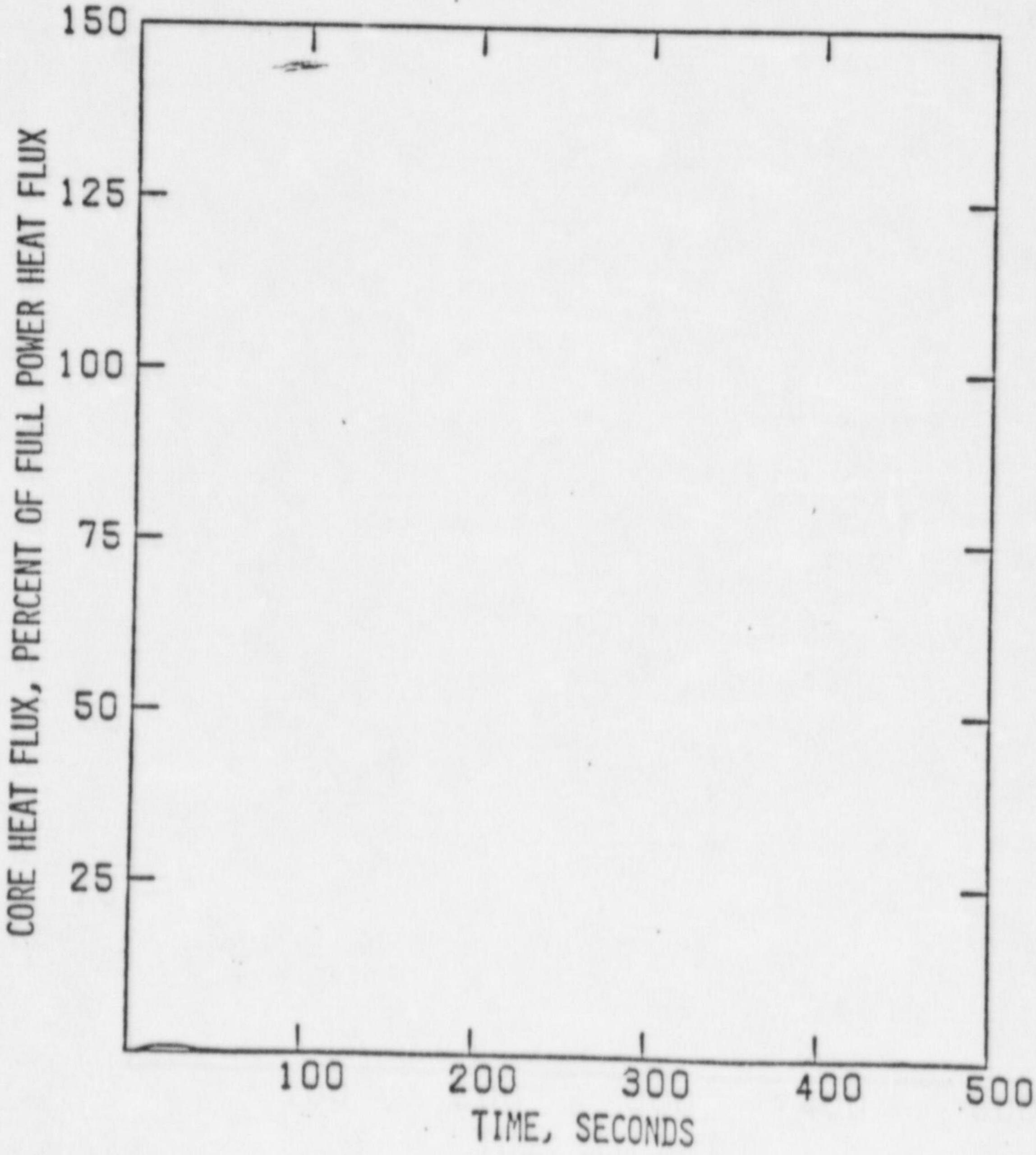

150
125
100
75
50
25
CORE HEAT FLUX, PERCENT OF FULL POWER HEAT FLUX
100
200
300
400
500
TIME, SECONDS

Figure 1-18

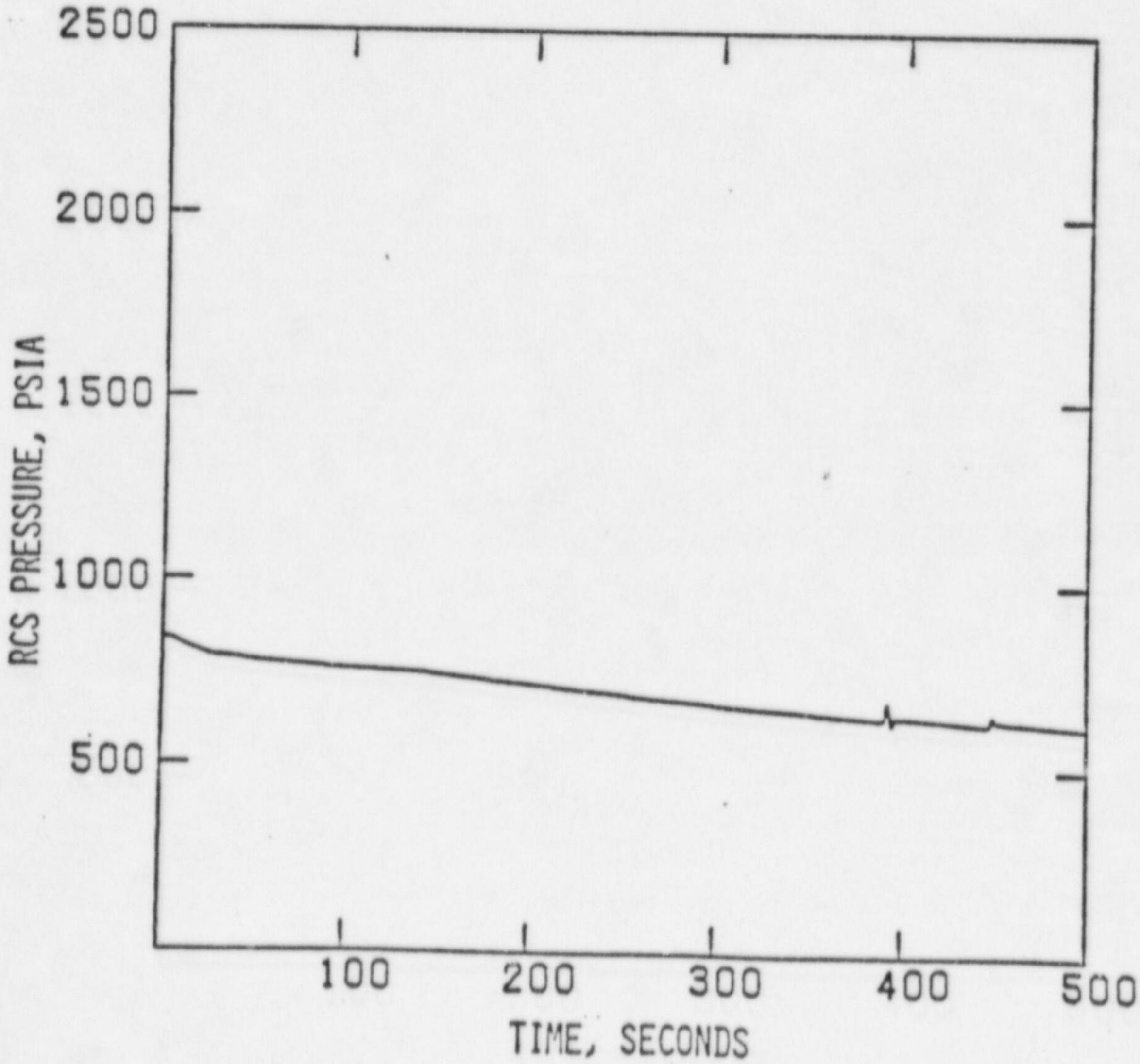
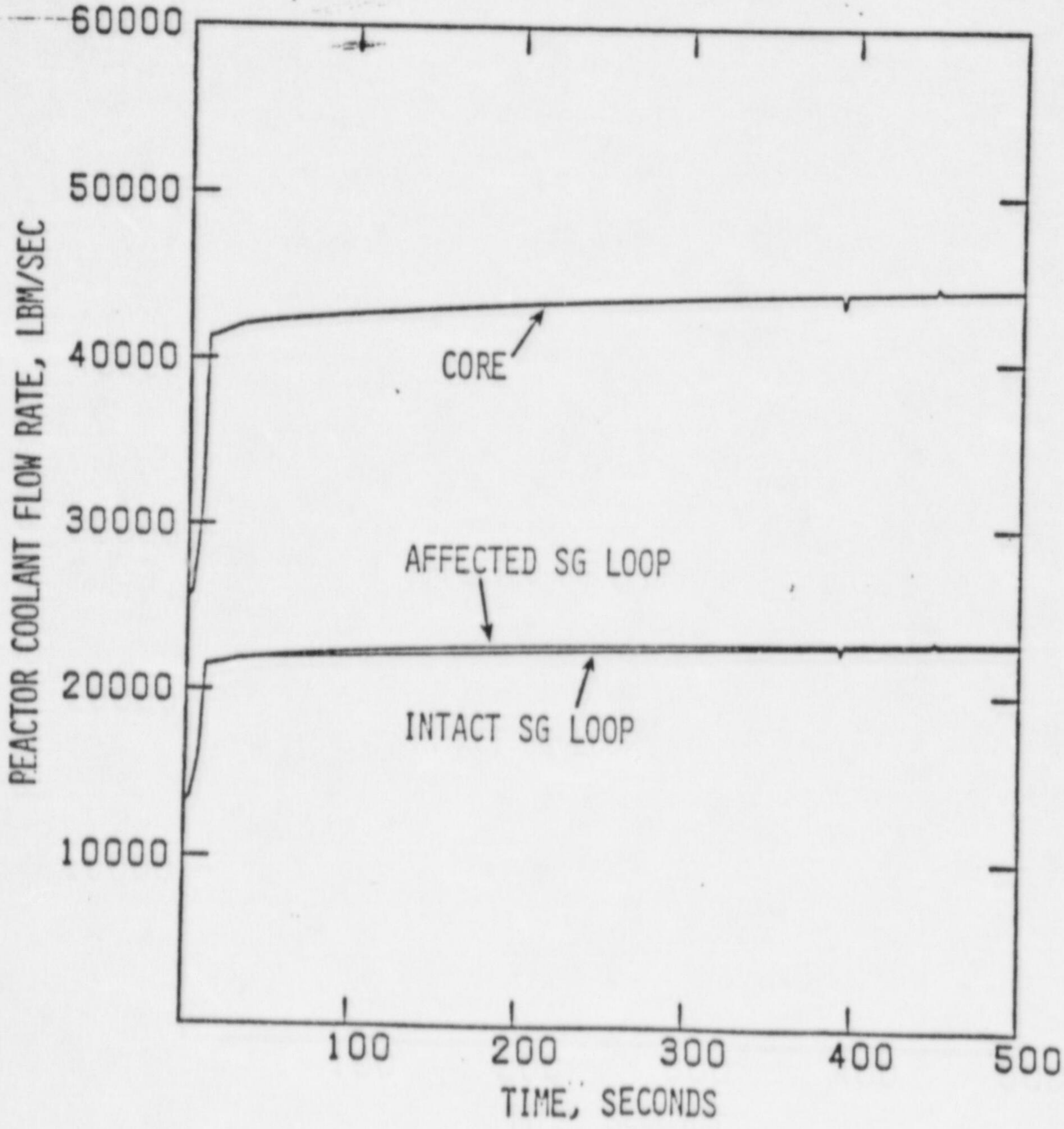

2500
2000
1500
1000
500
RCS PRESSURE, PSIA
100
200
300
400
500
TIME, SECONDS

Figure 1-19



60000
50000
40000
30000
20000
10000
PEACTOR COOLANT FLOW RATE, LBM/SEC
CORE
AFFECTED SG LOOP
INTACT SG LOOP
100
200
300
400
500
TIME, SECONDS

Figure 1-20

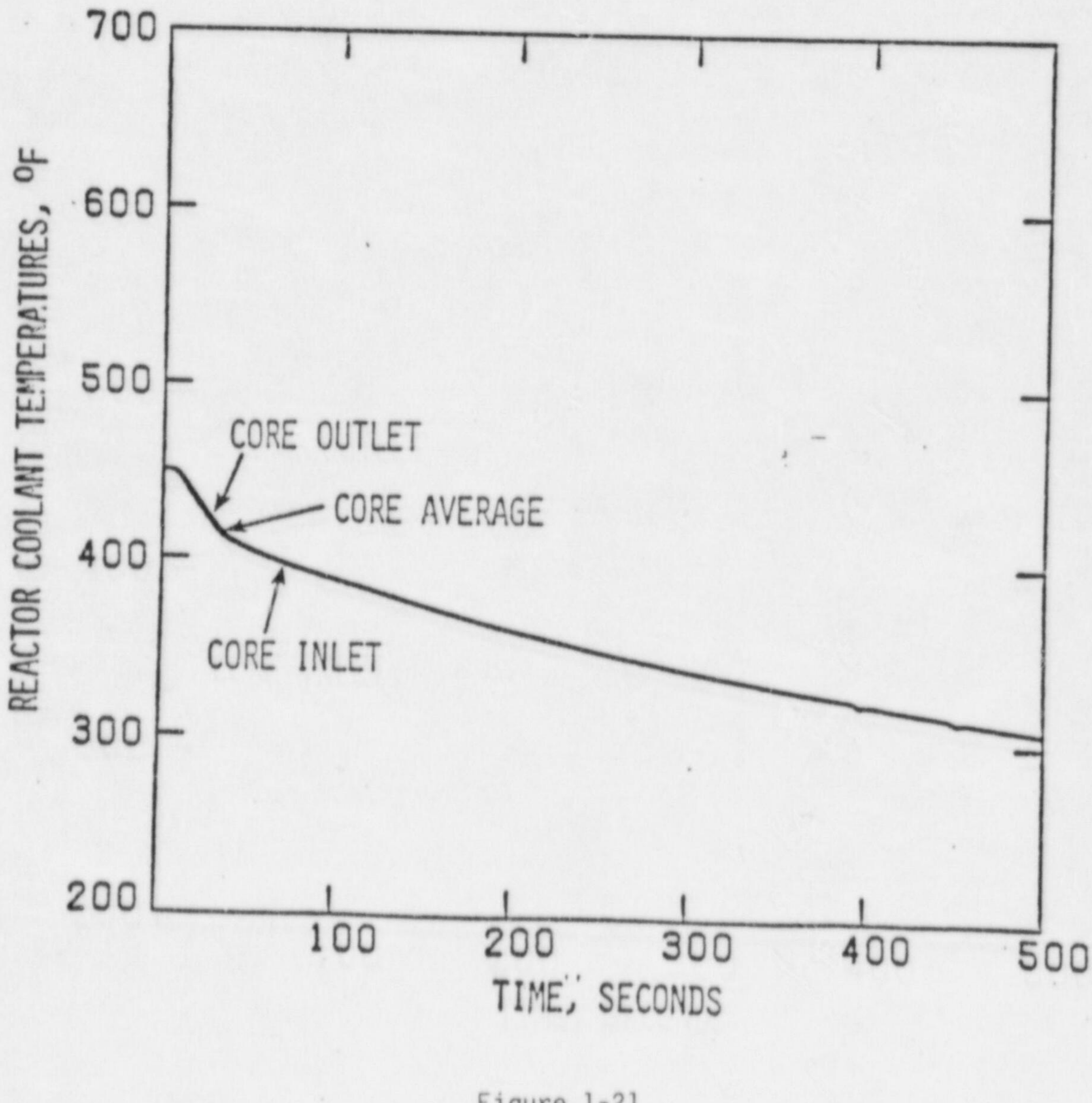

700
600
500
400
300
200
REACTOR COOLANT TEMPERATURES, °F
CORE OUTLET
CORE AVERAGE
CORE INLET
100
200
300
400
500
TIME, SECONDS

Figure 1-21

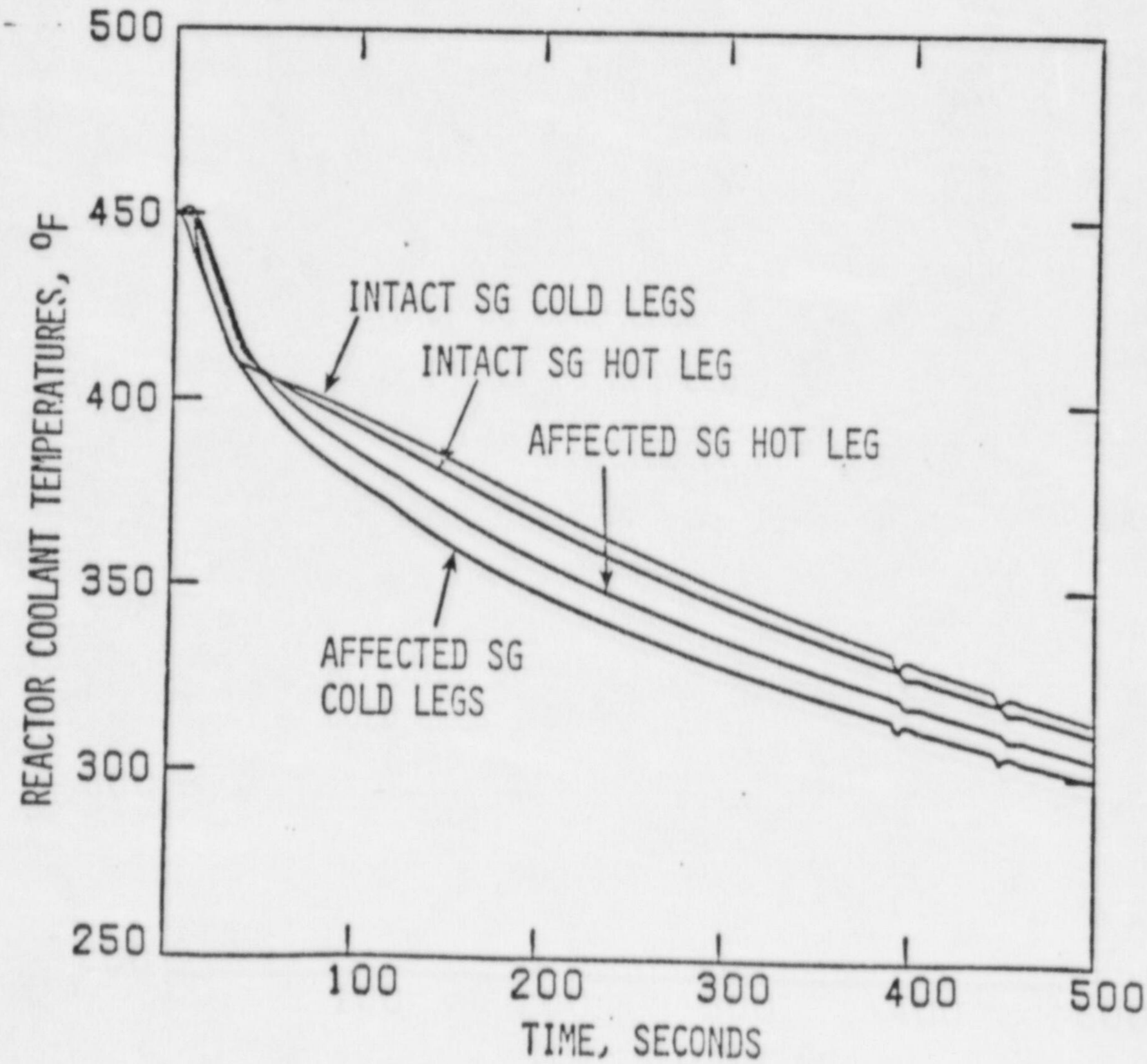
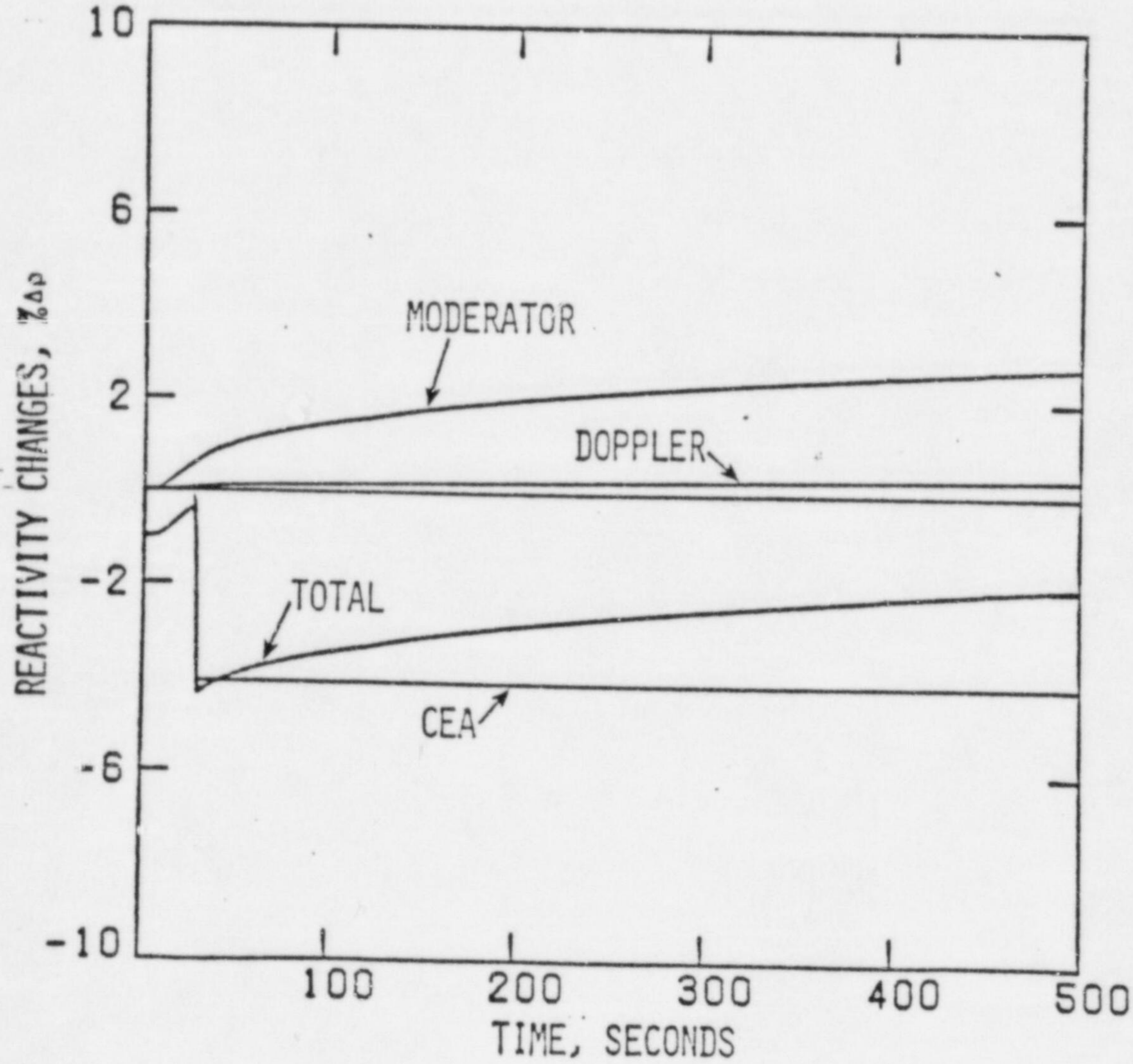

500
450
400
350
300
250
REACTOR COOLANT TEMPERATURES, ^{o}F
INTACT SG COLD LEGS
INTACT SG HOT LEG
AFFECTED SG HOT LEG
AFFECTED SG
COLD LEGS
100
200
300
400
500
TIME, SECONDS

Figure 1-22



MODERATOR
DOPPLER
TOTAL
CEA
REACTIVITY CHANGES, %Δρ
10
6
2
-2
-6
-10
100
200
300
400
500
TIME, SECONDS

Figure 1-23

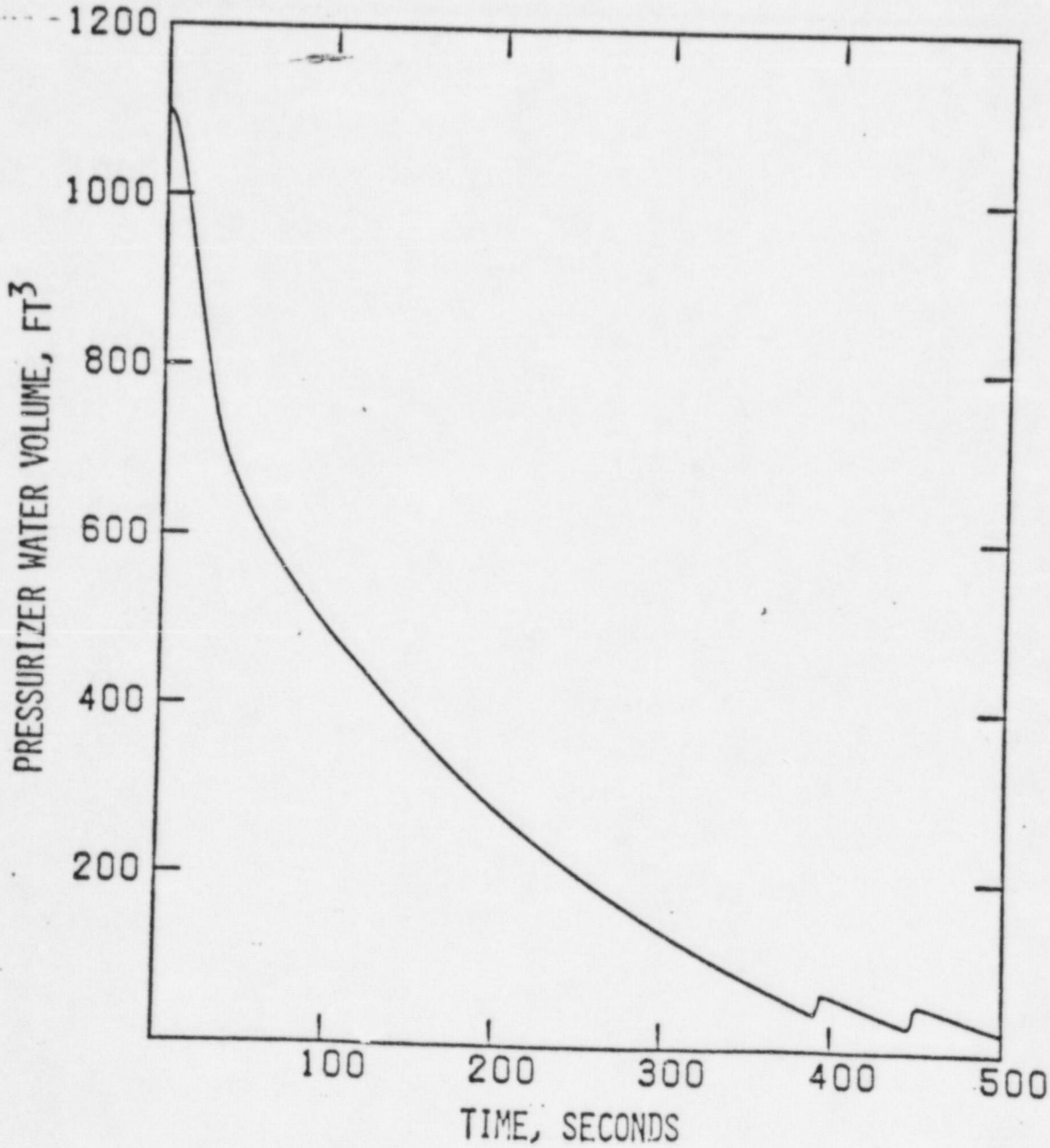

1200
1000
800
600
400
200
PRESSURIZER WATER VOLUME, FT3
100
200
300
400
500
TIME, SECONDS

Figure 1-24

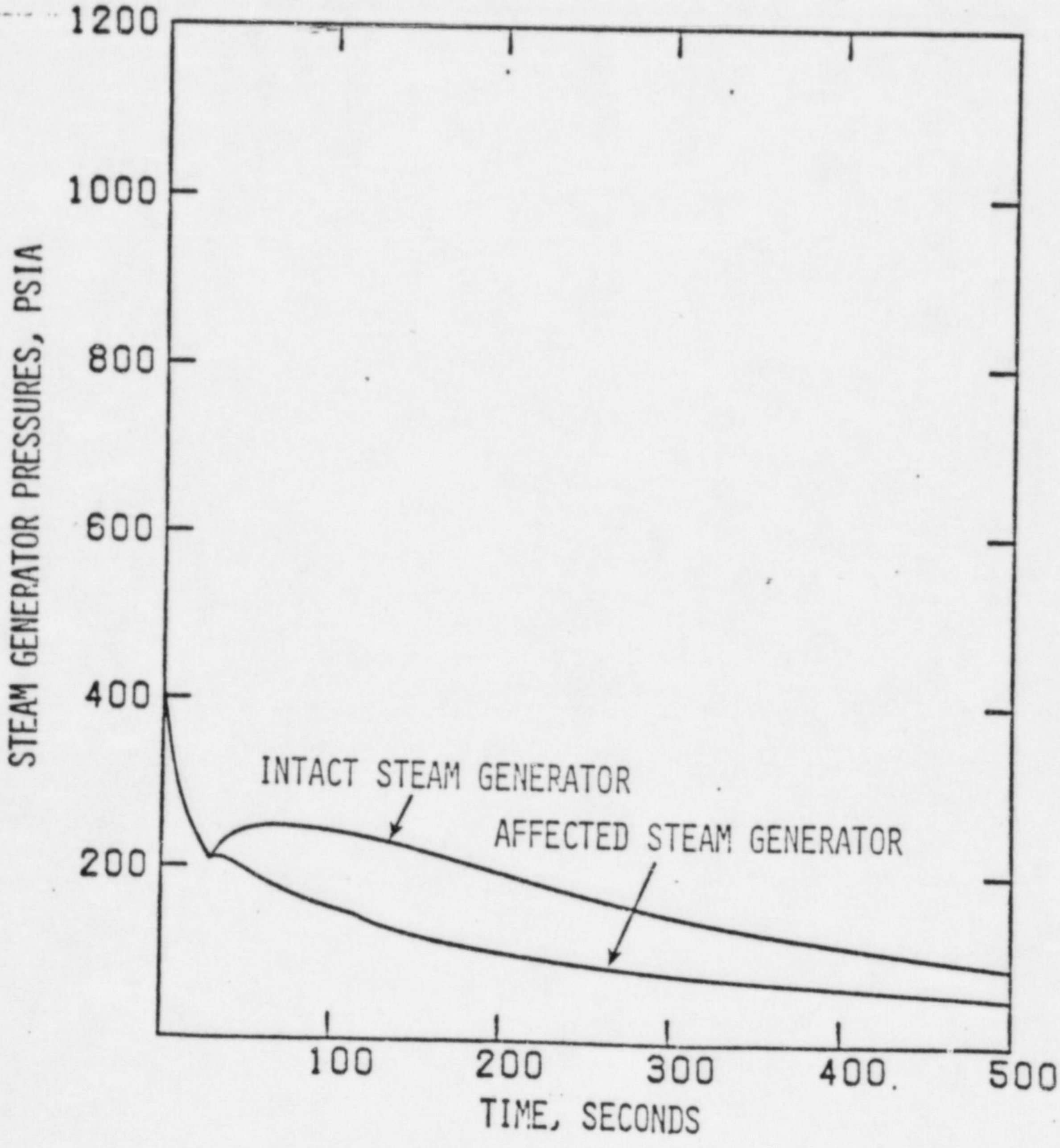
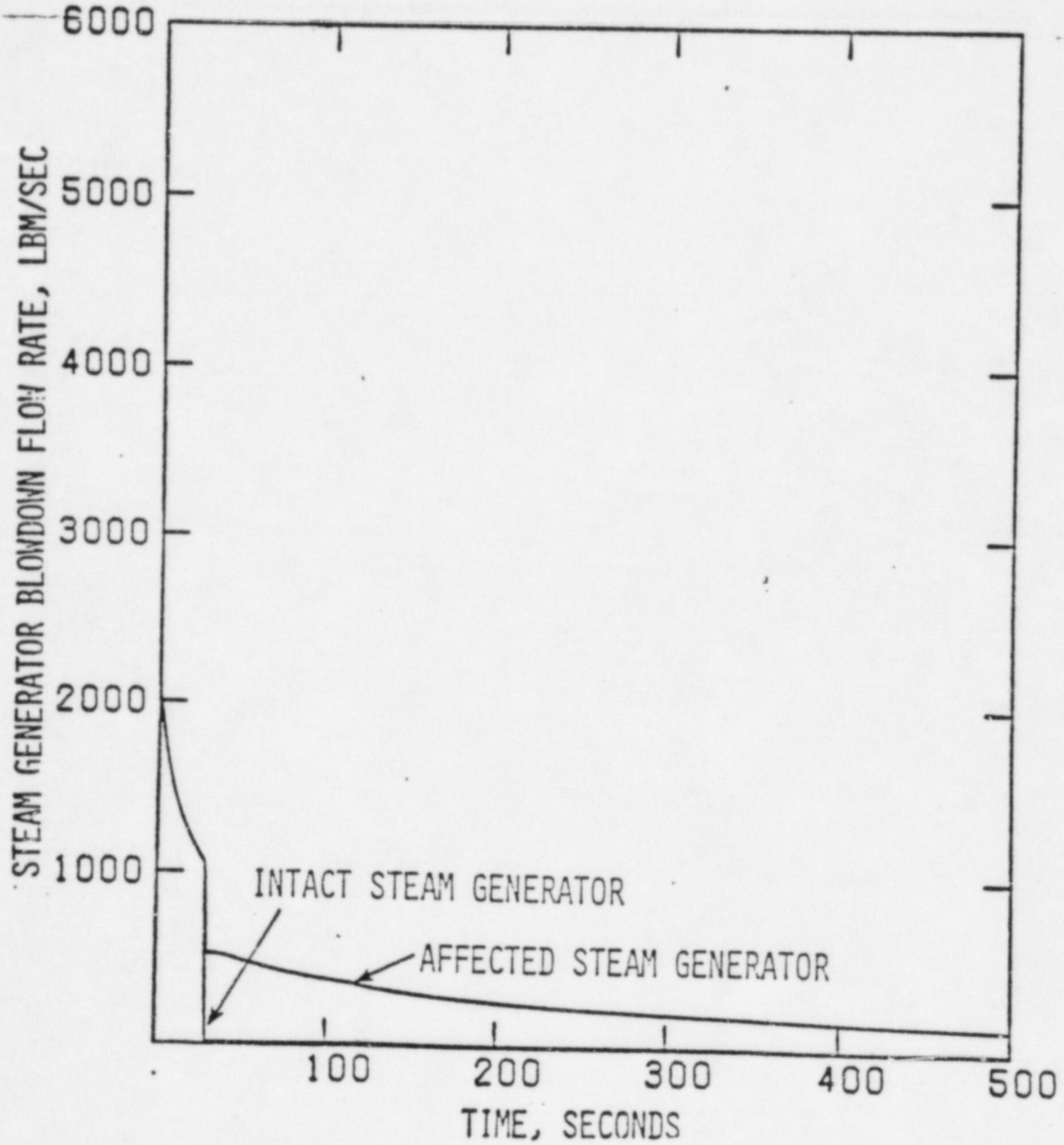

1200
1000
800
600
400
200
STEAM GENERATOR PRESSURES, PSIA
INTACT STEAM GENERATOR
AFFECTED STEAM GENERATOR
100
200
300
400.
500
TIME, SECONDS

Figure 1-25



STEAM GENERATOR BLOWDOWN FLOW RATE, LBM/SEC
6000
5000
4000
3000
2000
1000
INTACT STEAM GENERATOR
AFFECTED STEAM GENERATOR
100
200
300
400
500
TIME, SECONDS

Figure 1-26

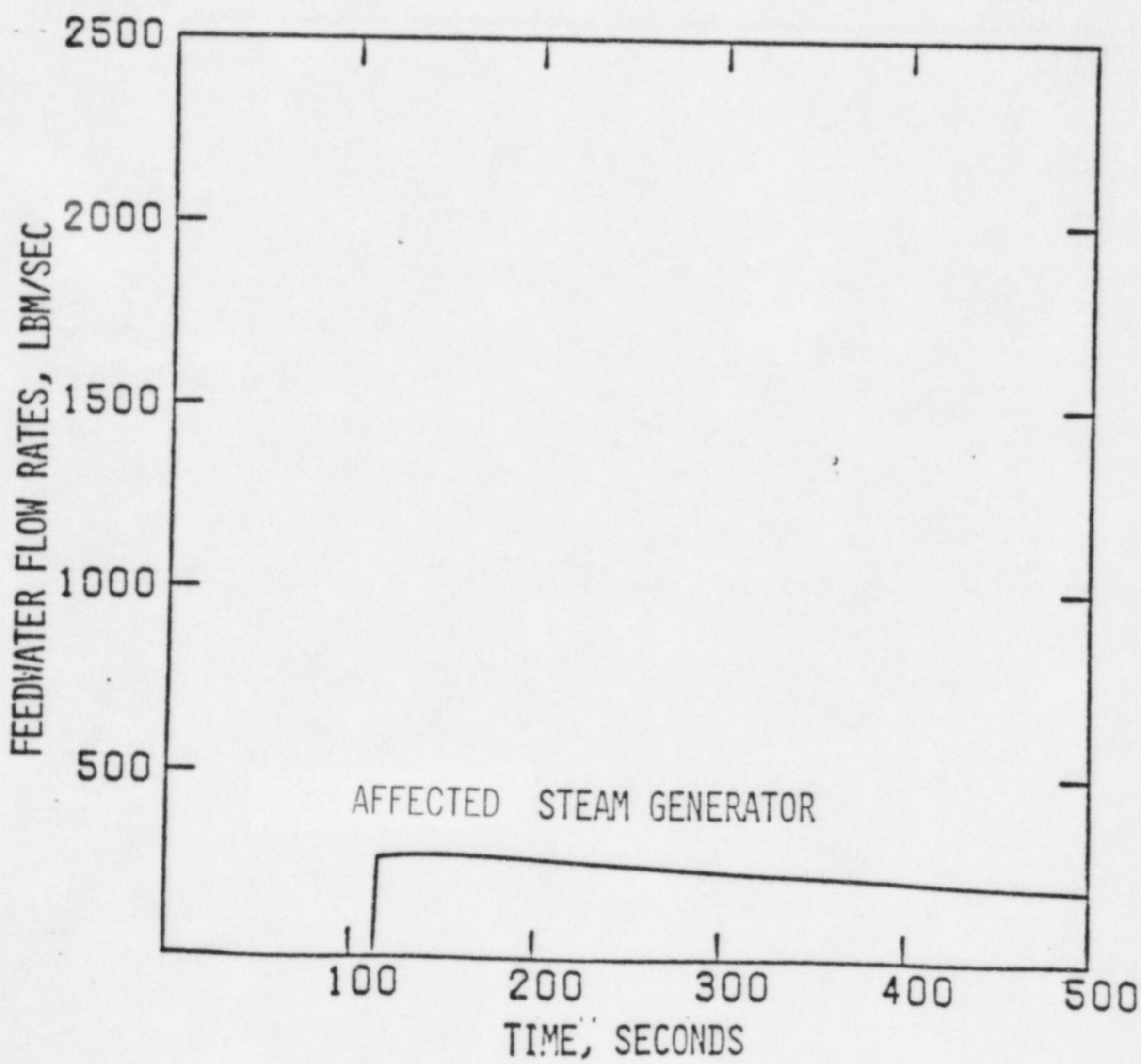
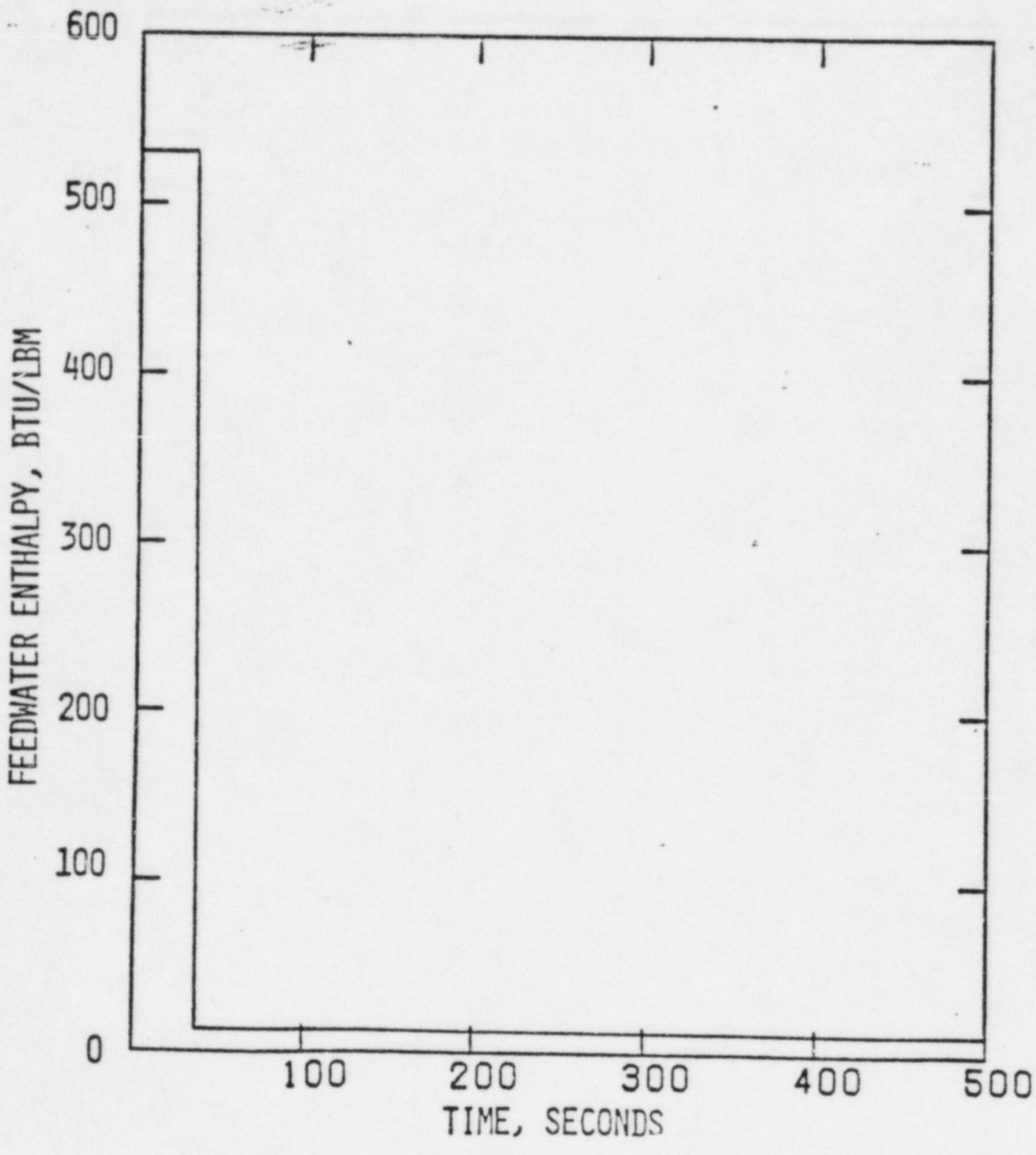

2500
2000
1500
1000
500
FEEDWATER FLOW RATES, LBM/SEC
AFFECTED STEAM GENERATOR
100
200
300
400
500
TIME, SECONDS

Figure 1-27



600
500
400
300
200
100
0
FEEDWATER ENTHALPY, BTU/LBM
100
200
300
400
500
TIME, SECONDS

Figure 1-28

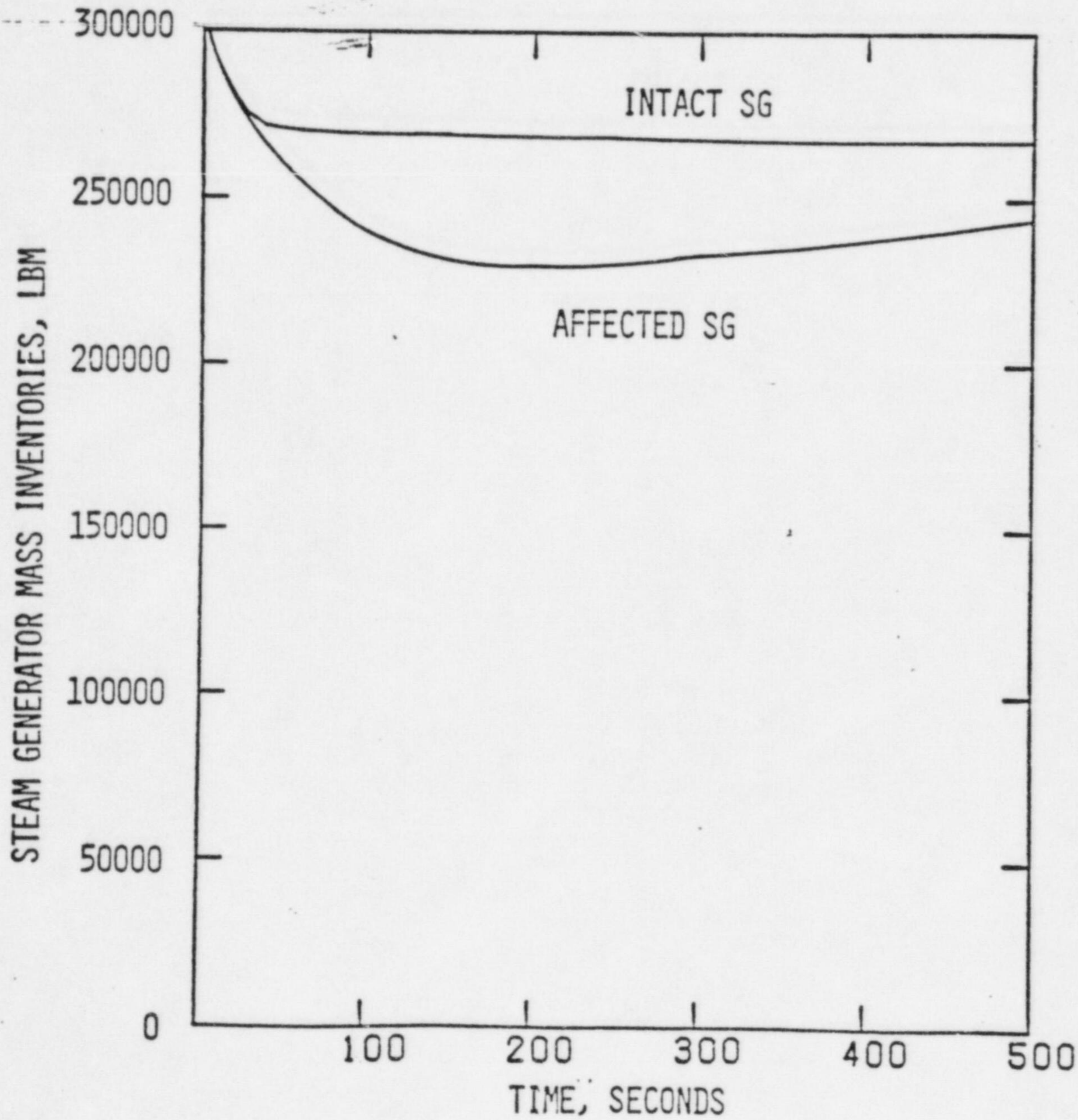

300000
250000
200000
150000
100000
50000
0
100
200
300
400
500
STEAM GENERATOR MASS INVENTORIES, LBM
TIME, SECONDS
INTACT SG
AFFECTED SG

Figure 1-29

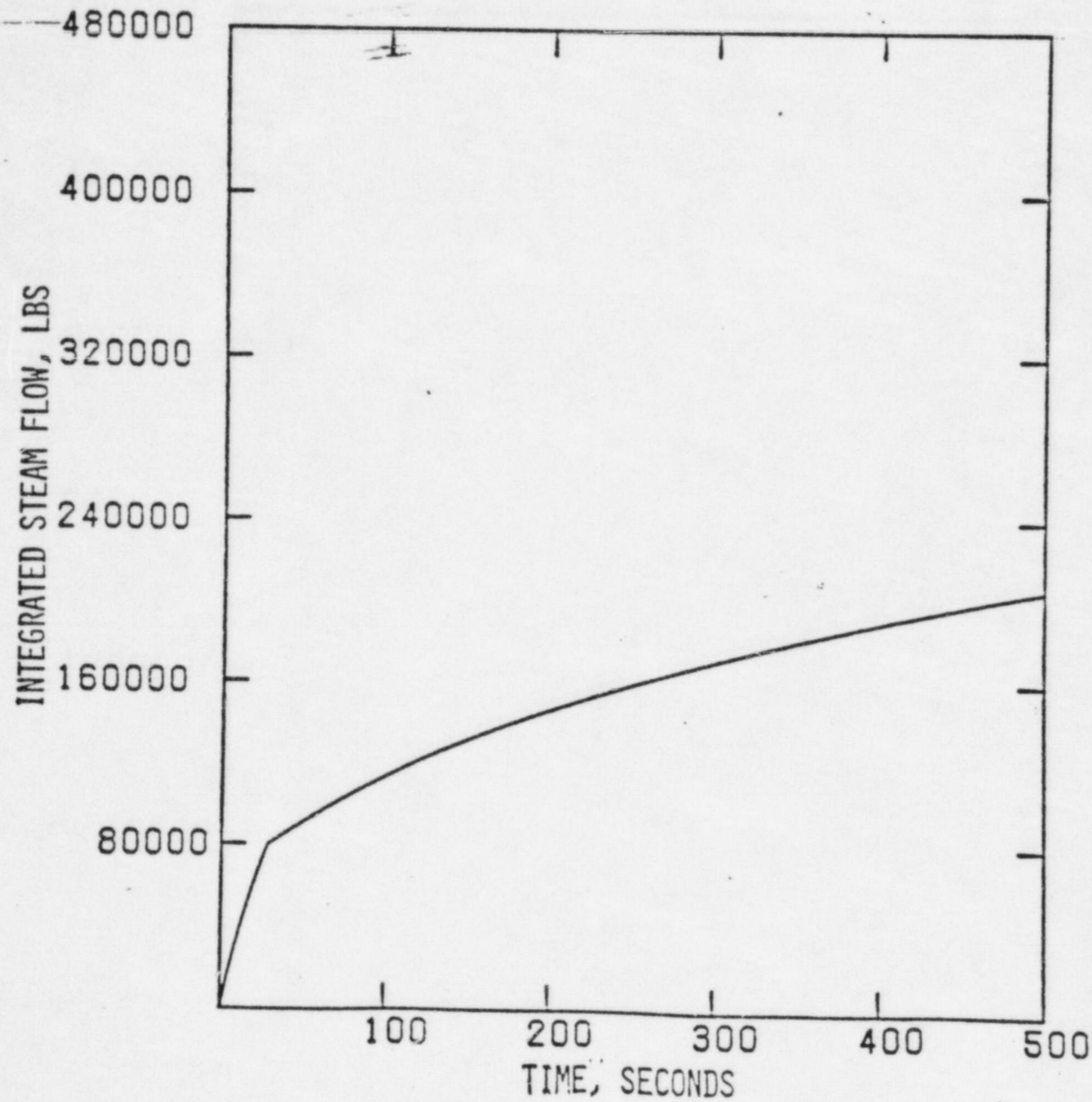

480000
400000
320000
240000
160000
80000
INTEGRATED STEAM FLOW, LBS
100
200
300
400
500
TIME, SECONDS

Figure 1-30

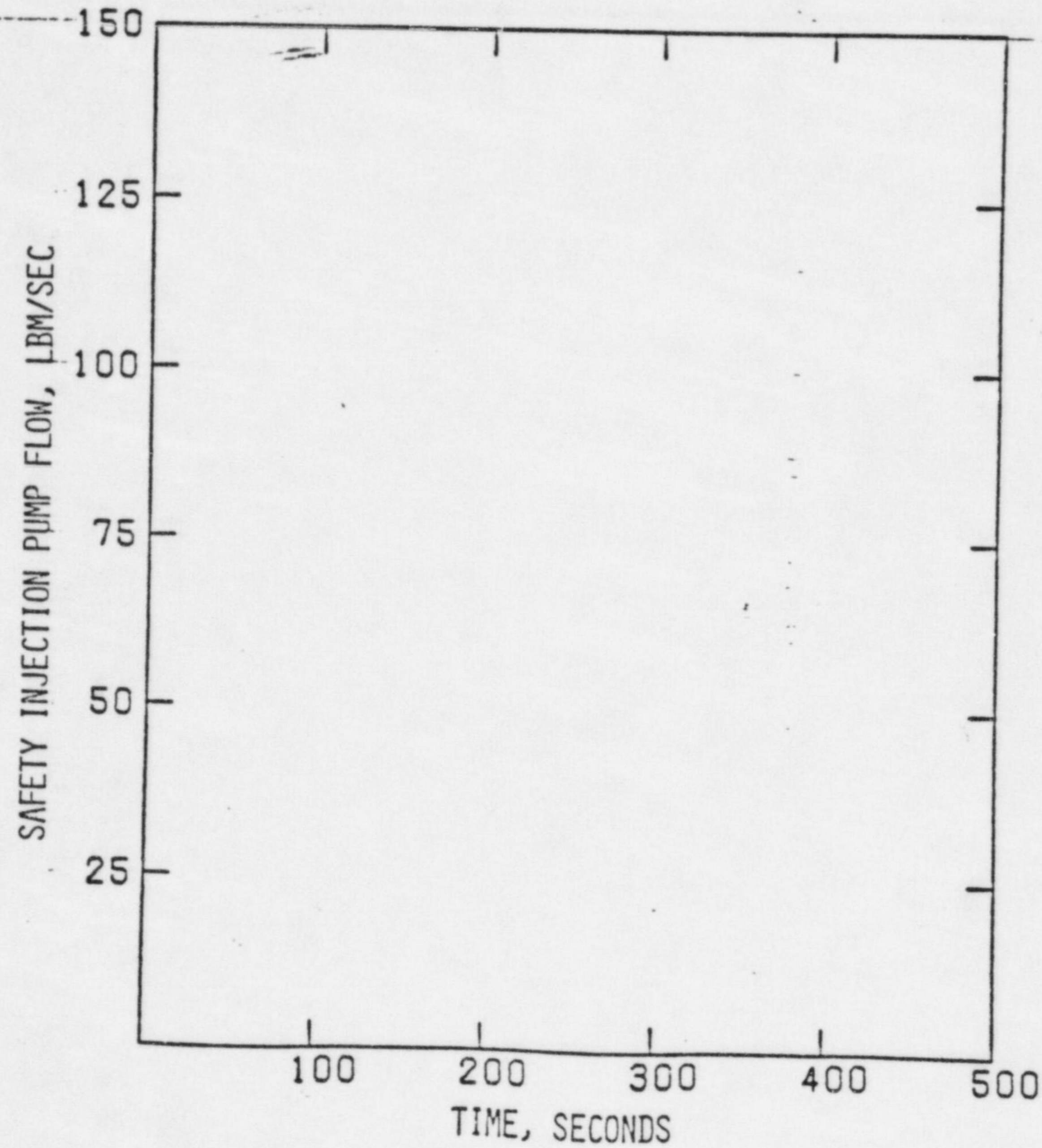
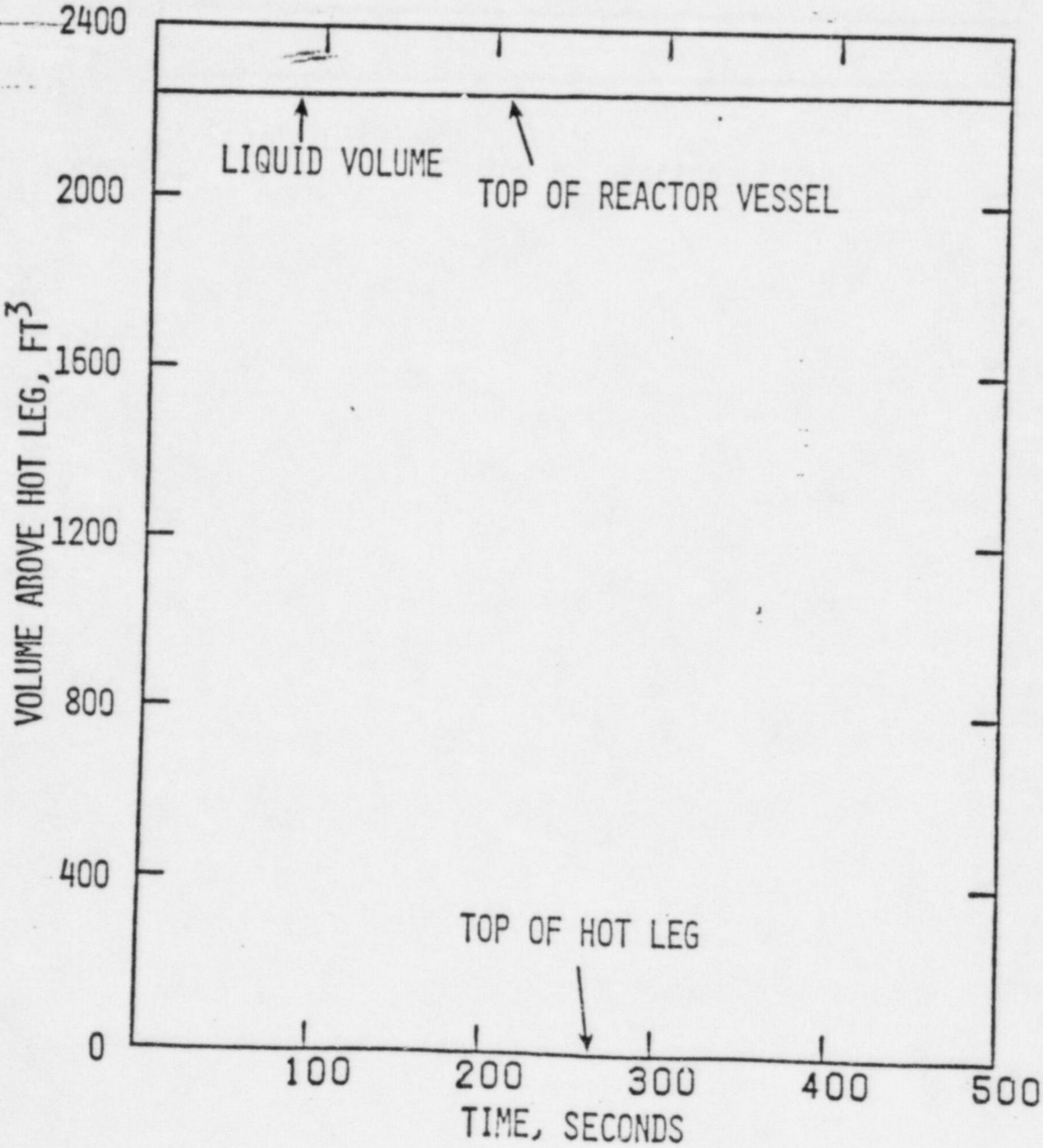

150
125
100
75
50
25
SAFETY INJECTION PUMP FLOW, LBM/SEC
100
200
300
400
500
TIME, SECONDS

Figure 1-31



2400
2000
1600
1200
800
400
0
VOLUME ABOVE HOT LEG, FT3
LIQUID VOLUME
TOP OF REACTOR VESSEL
TOP OF HOT LEG
100
200
300
400
500
TIME, SECONDS

Figure 1-32

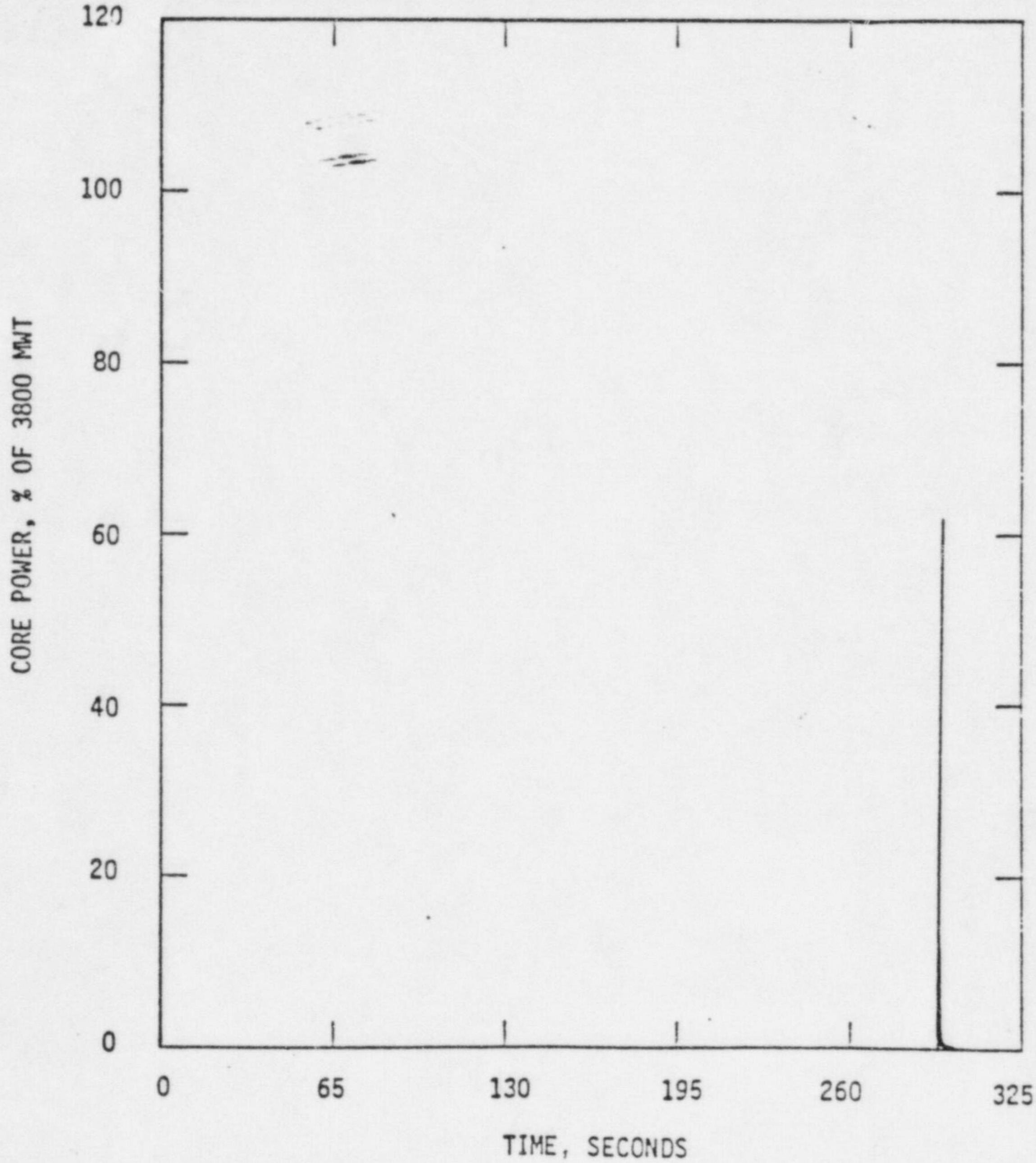
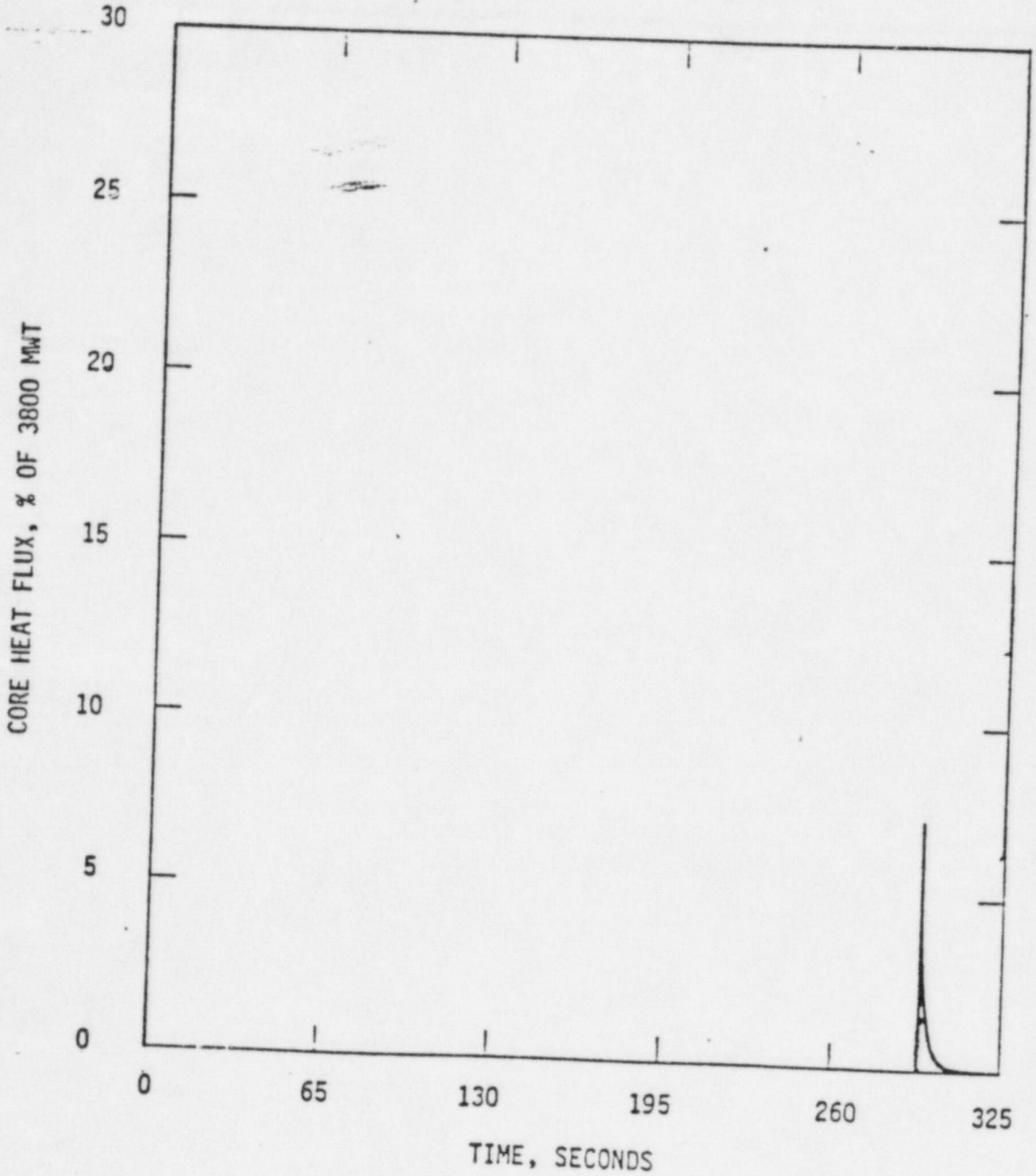
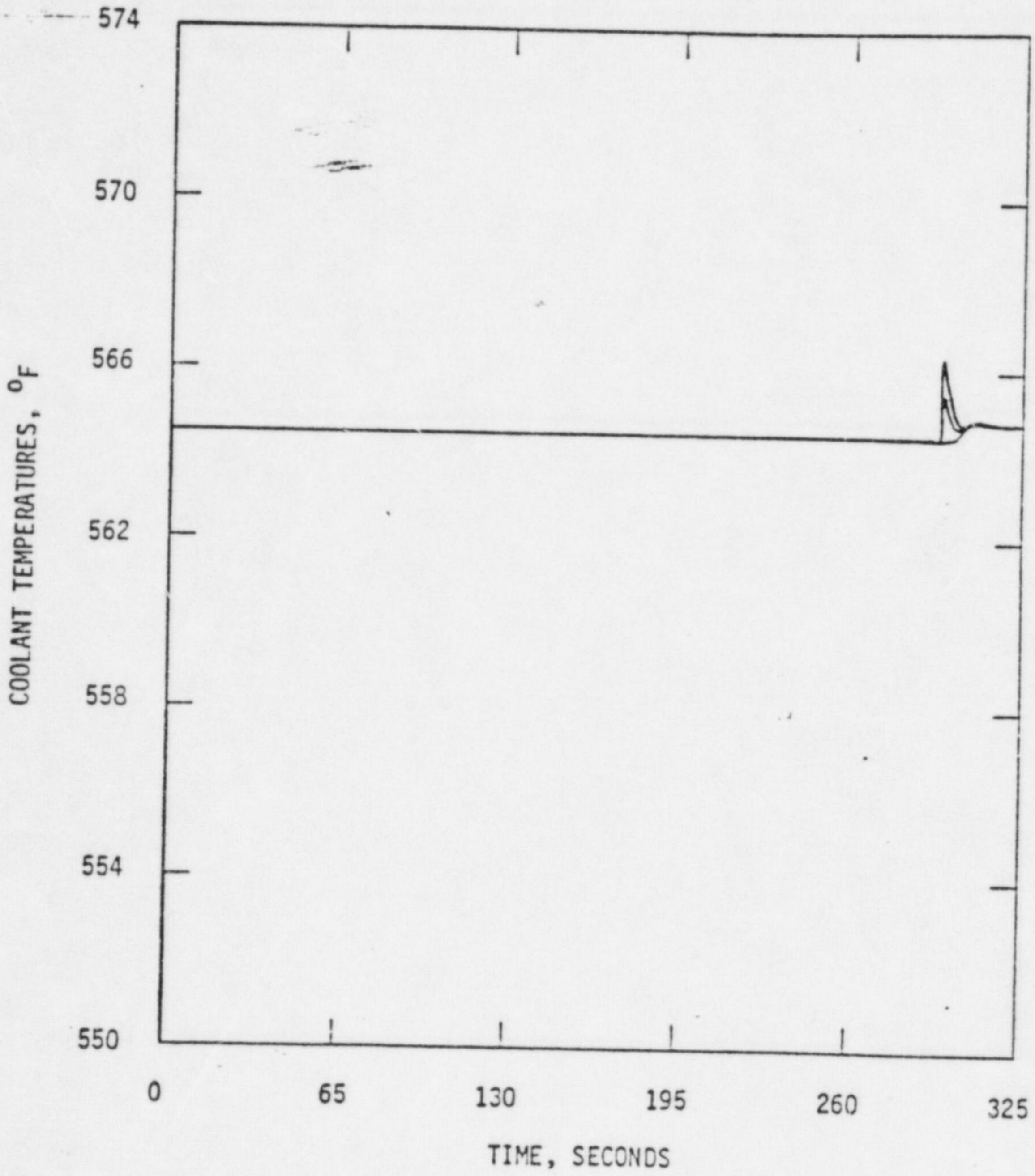

120
100
80
60
40
20
0
CORE POWER, % OF 3800 MWT
0
65
130
195
260
325
TIME, SECONDS

Figure 4-1



CORE HEAT FLUX, % OF 3800 MWT
30
25
20
15
10
5
0
0
65
130
195
260
325
TIME, SECONDS

Figure 4-2



574
570
566
562
558
554
550
COOLANT TEMPERATURES, ^{o}F
0
65
130
195
260
325
TIME, SECONDS

Figure 4-3

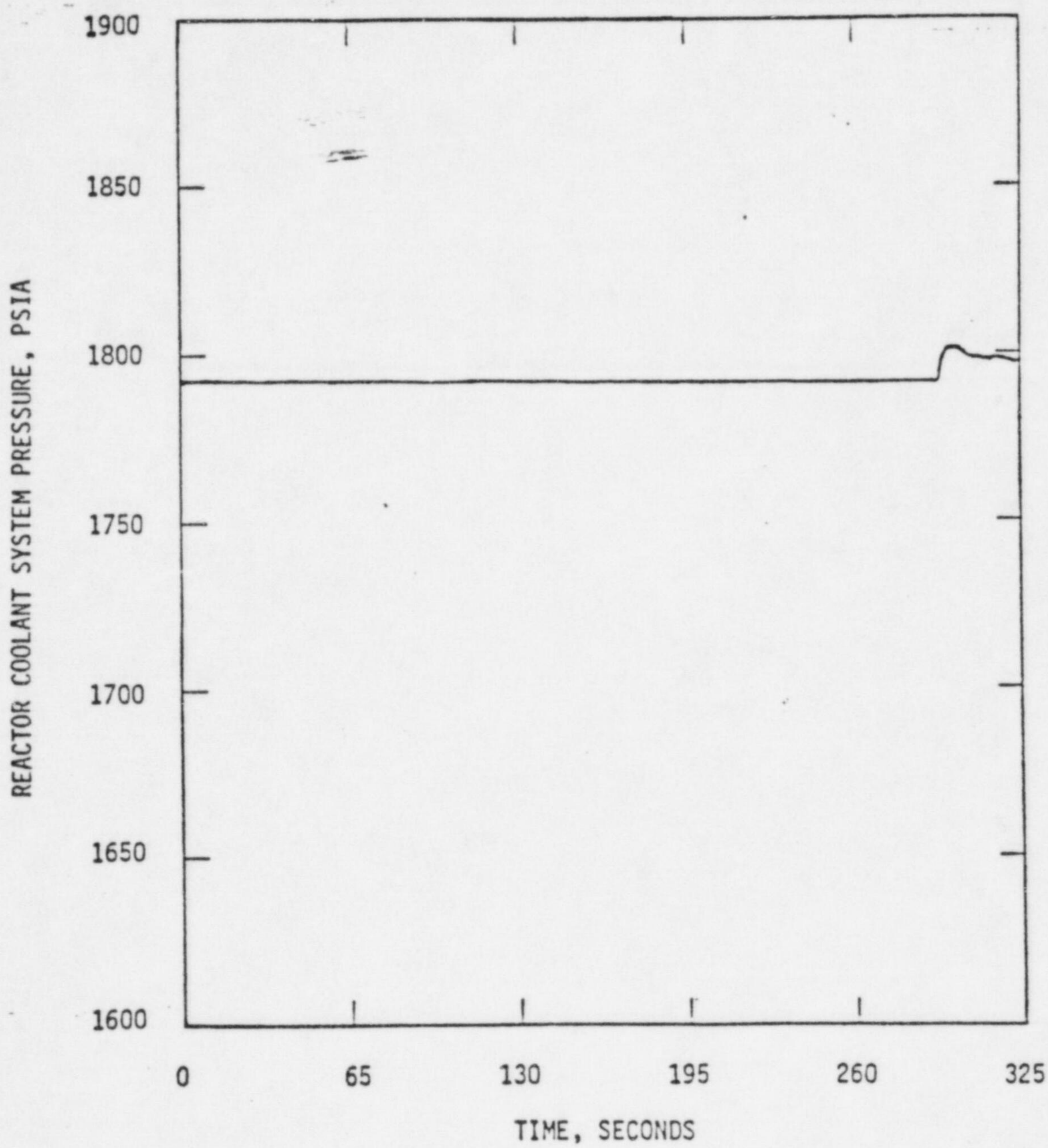
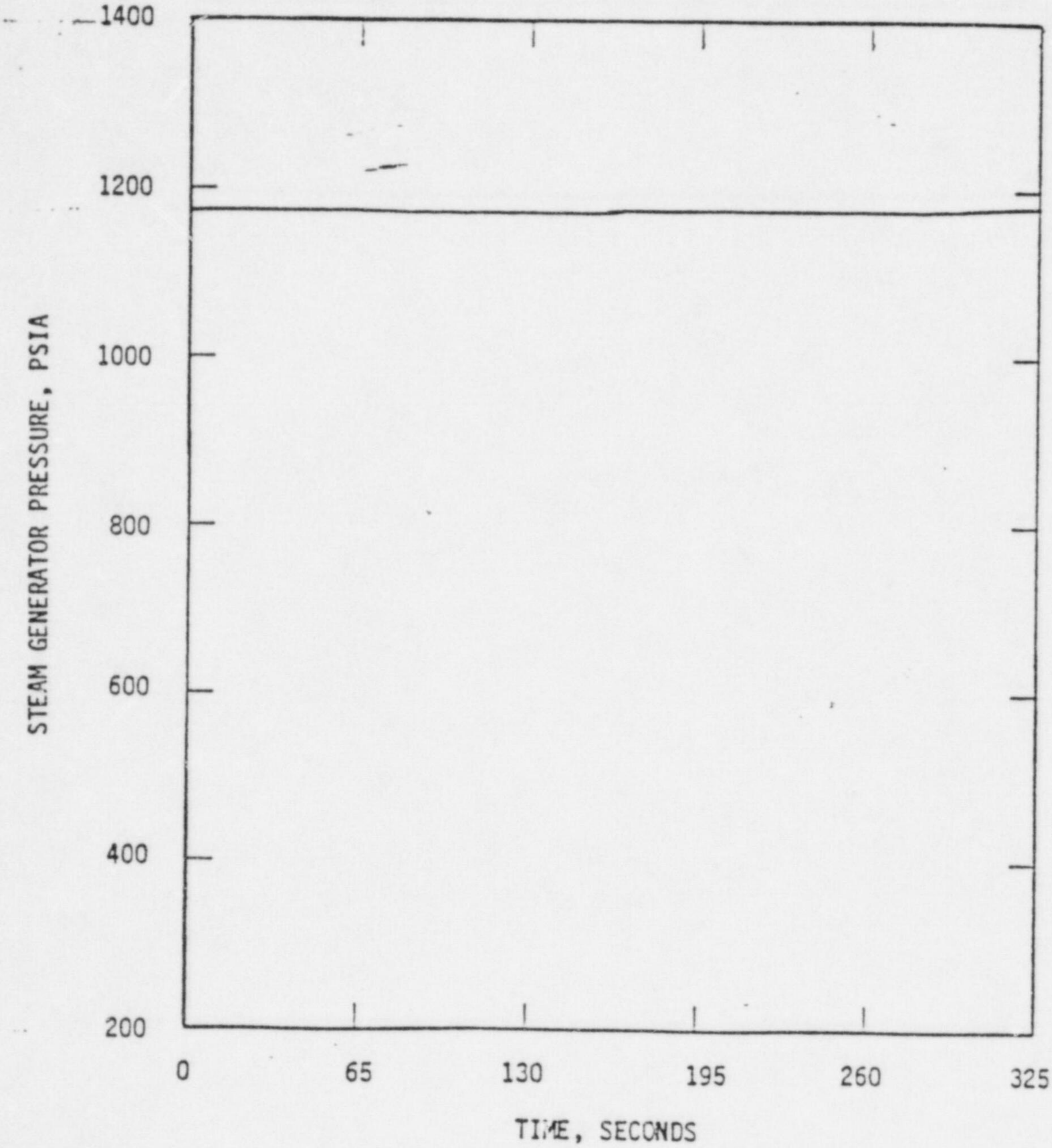

1900
1850
1800
1750
1700
1650
1600
REACTOR COOLANT SYSTEM PRESSURE, PSIA
0
65
130
195
260
325
TIME, SECONDS

Figure 4-4



1400
1200
1000
800
600
400
200
0
65
130
195
260
325
STEAM GENERATOR PRESSURE, PSIA
TIME, SECONDS

Figure 4-5