50-247

INDIAN POINT UNIT NOS. 2 & 3 PLANT SPECIFIC INTERPOLATION STUDY (ATTACHMENT 2)
JULY, 1977

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ATTACHMENT 2

Indian Point Unit Nos. 2 and 3
Plant Specific Interpolation Study

(by use of interpolation of the results of the Generic Reactor Coolant System Pressure Transient Analysis).

July, 1977

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Abstract

The generic analyses which appears in the report entitled "Pressure Mitigating System Transient Analysis Results - July 1977" presented the results of pressure transient analyses for the Reactor Coolant System (RCS) of a Pressurized Water Reactor during low temperature operation (water solid) for cases of either MASS INPUT or HEAT INPUT to the system. The most conservative features were selected from among the member plants participating in this program, and then additional conservatisms were also applied in the generic transient analysis. Even for this generic bounding case it was demonstrated that a single power operated relief valve (PORV) was capable of mitigating the worst possible MASS or HEAT INPUT transients.

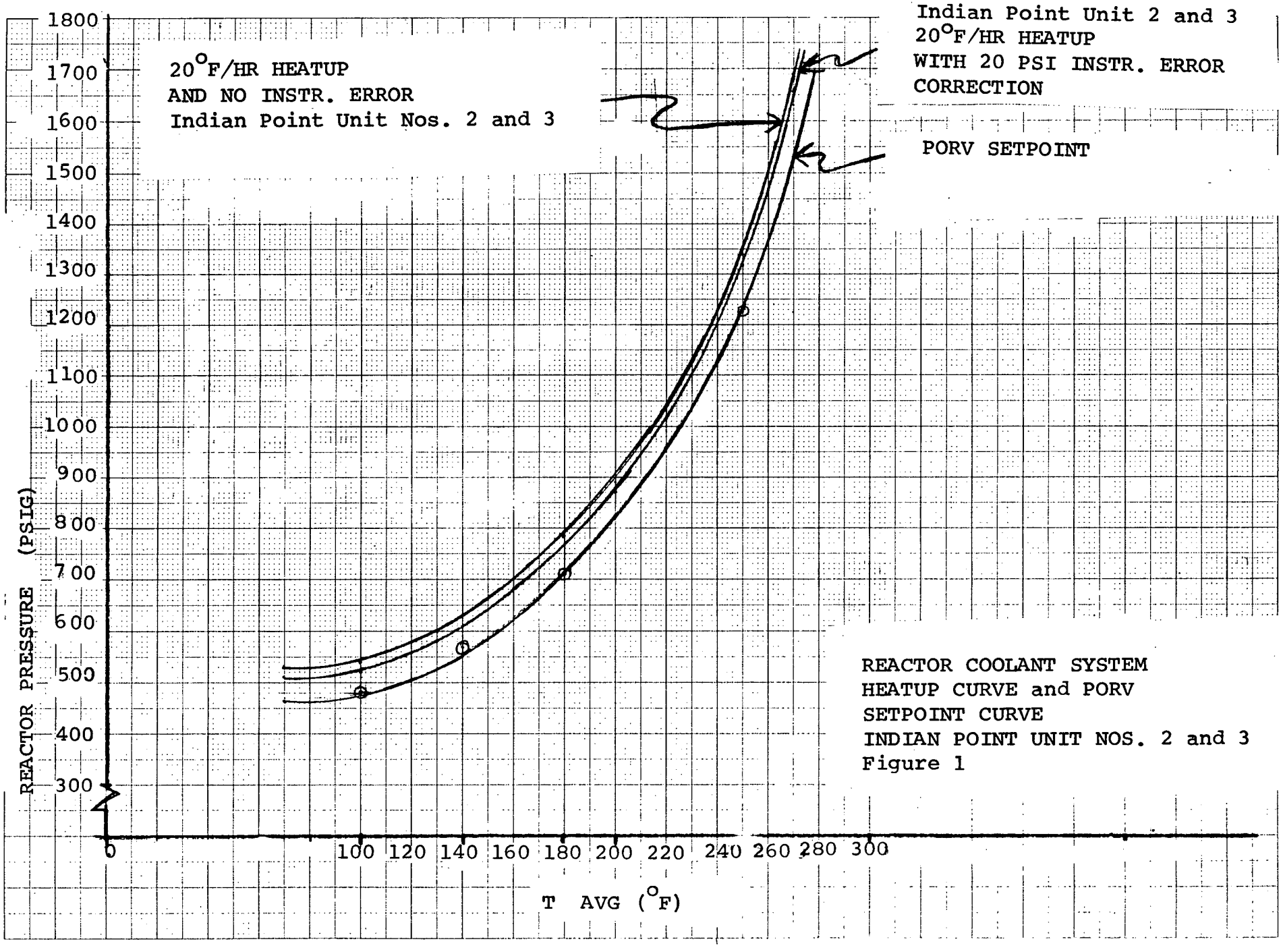
The purpose of the plant specific study (for Indian Point Unit Nos. 2 and 3) is to remove the excessive conservatisms of the generic study and to approximate the transient conditions which would apply to the Indian Point Unit Nos. 2 and 3 plants.

It should be noted that this plant specific study removed some of the excess conservatisms of the generic study, but did not remove all of the conservatism. Although the results of this plant specific study more truly represent the realistic cases that could occur at the Indian Point Unit Nos. 2 and 3 plants, (as compared to the generic analyses), more exact modelling would be expected to yield even more favorable results. Table 1 compares the parameters that were used in the generic study versus those parameters that are present in the Indian Point Unit Nos.

2 and 3 plants. Table 2 lists the conservatisms that remain in this plant specific study, even through the extrapolation from generic to plant specific has been performed.

The results of this plant specific study show that the PORV overshoot (above the PORV setpoints) varies from 39 PSI to 93 PSI for RCS temperatures of 100°F and 250°F, respectively. Figure 1 shows the setting of the PORV, which is sufficiently below the 10 CFR 50 Appendix G curve, such that the PORV overshoots do not result in peak RCS pressures exceeding the 10 CFR 50 Appendix G curve (20°F/HR Heatup curve with 20 PSI correction for instrument error).

The interpolation calculations and use of the factors (which appear in the equations and the graphs) are shown in the following pages of this plant specific study for both the MASS INPUT and HEAT INPUT cases.



REACTOR COOLANT SYSTEM
HEATUP CURVE and PORV
SETPOINT CURVE
INDIAN POINT UNIT NOS. 2 and 3
Figure 1

Table 1

Generic Parameters versus Plant Specific Parameters (Indian Point Unit Nos. 2 and 3)

Power Operated Relief Valves (PORV)

	Generic	IP2 and 3
Quantity (operating)	1	1 of 2
size (nominal)	2 inch	3 inch body, 2 inch internals
C_v	50.	50.
Opening time	3.0 seconds	1.5 seconds
Valve characteristics	0.6 seconds delay and 2.4 seconds linear stroke	(0.3 delay and 1.2 seconds linear stroke, based on operating experience)
Closing time	5.0 and 20.0 seconds	1.5 seconds
Valve characteristics	time delay is 16% of total closing time	(0.3 seconds delay and 1.2 seconds linear stroke, based on operating experience)

RCS SYSTEM VOLUMES

	Generic	IP2 and 3
Cubic Feet	6000 and 13,000	12,224

MASS INPUT

Limiting Mass Input Flow(1)	1 S.I. Pump	1 S.I. Pump
S.I. Pump Flowrate, lb/sec(2)	113	73
PORV Opening time, seconds	3.0	1.5

Table 1 Continued

Generic Parameters versus Plant Specific

Parameters (Indian Point Unit Nos. 2 and 3)

HEAT INPUT

	Generic	IP2 and 3
Steam Generator, Heat Transfer Surface Area, Square Feet	58,000	44,430
Temperature Difference (max) Steam Generators/ RCS, °F	50	50
PORV Opening time, seconds (3)	3.0	3.0 (Actual time is 1.5 sec)

(1) A summary of the plant administrative procedures for the defeat of automatic SI has been submitted to the Commission on page 1 of Attachment 1 to the Indian Point Unit 2 and 3 letters of October 25, 1976 from Mr. W. Cahill, Jr. of Con Edison to Mr. R. Reid, of the NRC. The Con Edison responses to NRC questions concerning defeat of automatic SI appear in Responses 9a through 9e of Attachment 2 to the Indian Point Unit 2 and 3 letters of February 28, 1977 from Mr. W. Cahill, Jr. of Con Edison to Mr. R. Reid of the NRC.

Double administrative controls are employed for the SI Accumulator valves in order to prevent an inadvertent opening of these valves and a subsequent flow of the accumulator contents into a water solid reactor coolant system.

Because of these controls and procedures it is concluded that the postulated worst case for MASS Input flow would be a single S.I. pump.

(2) Refer to Figure 2.3.2 of the generic report entitled "Pressure Mitigating Systems Transient Analysis Results - July 1977". The 113 lb/sec flowrate corresponds to 835 GPM at 600 PSIG (PORV Setpoint) for the Generic SI pump (curve C). The 73 lb/sec flowrate corresponds to 545 GPM at 450 PSIG (PORV Setpoint) for the Indian Point 2 and 3 SI pumps (curve D).

(3) No credit was taken for the 1.5 second PORV for the generic Heat Input study. Thus, the use of the 3.0 second PORV for the Heat Input Study will result in higher values of maximum RCS pressure, and this is conservative. Future studies will be performed in order to evaluate the techniques for extrapolating for the 1.5 second PORV for the Heat Input cases. (The F_z factor for interpolating for the 1.5 second PORV has been developed for the MASS Input cases at the present time).

Table 2

Conservatisms in Generic Analyses,

which are present in the plant specific study

The following conservatisms appear in the generic analyses and are discussed in section 5 of the report entitled, "Pressure Mitigating System Transient Analysis Results-July 1977." The plant specific study for Indian Point Unit Nos. 2 and 3 did not remove the conservatisms which are listed below. Where applicable, the parametric value is shown for the Indian Point Unit Nos. 2 and 3 plants.

MASS Input

	Generic	IP 2 and 3
Expansion for Reactor Vessel	-	-
S.I. Pump Starting Time, seconds	1.64	3.7 (calculated)

HEAT Input

R.C. Pump Starting Time, seconds	9-10	15-20 (based on estimate of plant personnel)
PORV Opening time	see note (3) Table 1	
Steam Generator Secondary Water Volume, FT^3	3,580	2,447

	Conservative	Realistic
Steam Generator Heat Transfer Coefficient (applicable to the Generic Model for this case). $\frac{BTU}{HR\ {}^{o}F\ FT^2}$	Equal to h sec only	Includes h pri, h sec and tube wall conductivity

MASS Input
Calculation of Maximum RCS Pressure when a single S.I. Pump is started at either Indian Point Unit No. 2 or 3.[4]

The basic equations for calculating the maximum RCS pressure and PORV overshoot (above the PORV setpoint) appear below:

$$P_{\substack{MAX\\RCS}} = P_{\substack{PORV\\SETPOINT}} + \Delta P \qquad \text{(EQ 1)}$$

$$\Delta P = \Delta P_{REF}(X) * F_V * F_Z * F_S \qquad \text{(EQ 2)}$$

The calculations for the Indian Point Unit Nos. 2 and 3 follow:

$\Delta P_{REF}(X)$ = 100. @ 73 lb/sec, Figure 2

F_V = 0.540 @ 12,224 FT^3, Figure 3

F_Z = 0.600 @ 1.5 seconds, Figure 4

Assume $P_{\substack{PORV\\SETPOINT}}$ = 450 PSIG @ 100°F (1st trial)

then F_S = 1.20 @ 450 PSIG, Figure 5

substitute into equation 2,

$$\Delta P = 100 * .54 * .60 * 1.20 = 38.9 \approx 39 PSI$$

substitute into equation 1,

$$P_{\substack{MAX\\RCS}} = 450 + 39 = 489 \text{ PSIG.}$$

This value of the maximum RCS pressure is below the 10 CFR 50 Appendix G curve (see figure 1; 520 PSIG @ 100°F). Thus the setpoint of 450 PSIG for the power operated relief valve is acceptable for the MASS Input cases. (a setpoint of 481 PSIG could be utilized at 100°F without exceeding the 10 CFR 50 Appendix G curve as is shown in Figure 1).

The MASS Input cases were analyzed at a RCS temperature of 100°F. The compressibility of water is a minimum at approximately 100°F,

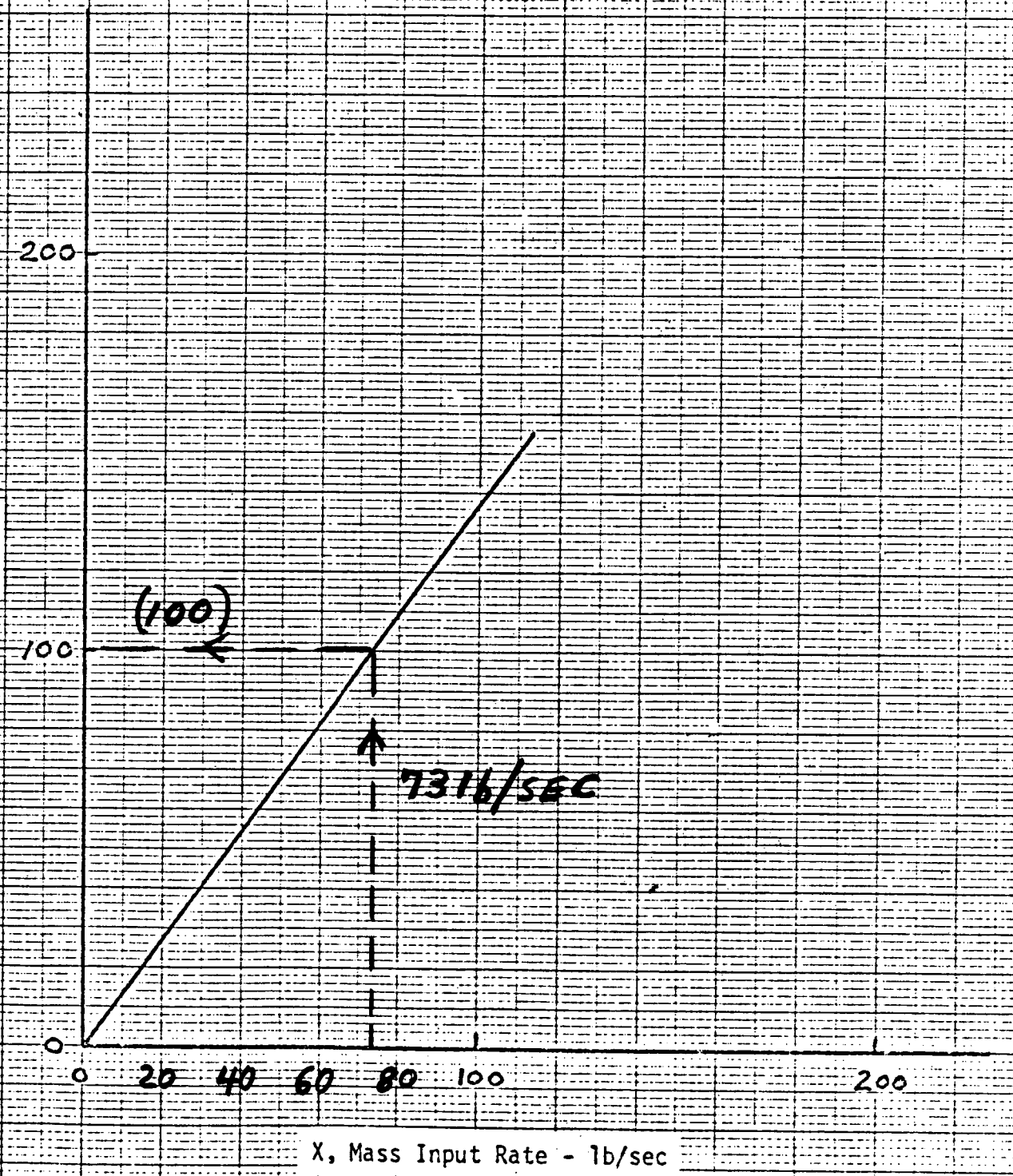
and the 100°F temperature is the most conservative for the MASS Input analyses. (i.e., the maximum value of peak RCS pressure will occur at approximately 100°F). Thus no further calculations will be performed at other RCS initial temperatures for the MASS Input cases.

[4]The equations and detailed explanation of the calculations appear in section 4 "Instructional Guide for Setpoint/Overshoot Determination", of the generic report entitled "Pressure Mitigating System Transient Analysis Results - July 1977".

FIGURE 2

Mass Input

Reference Overshoot

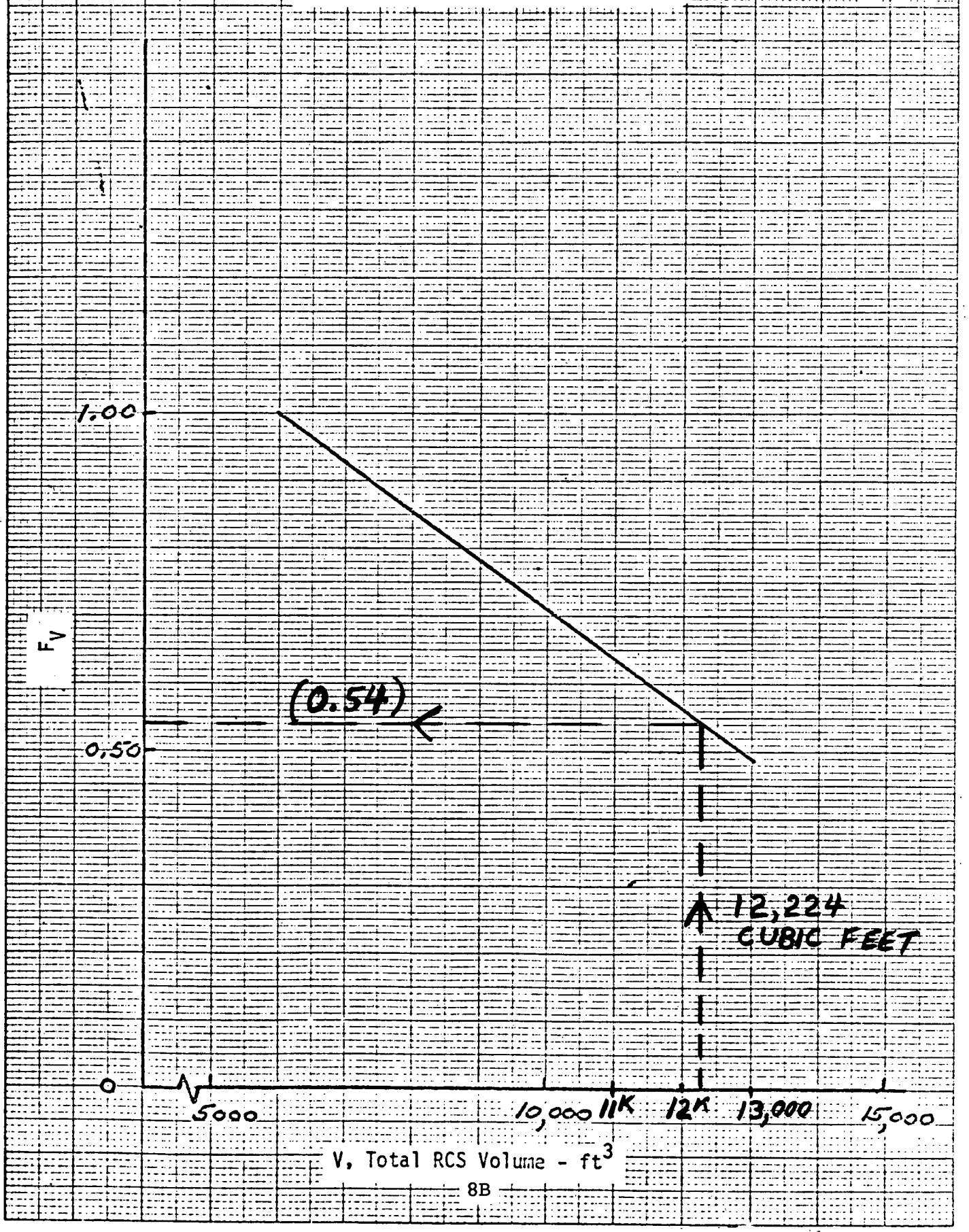


46 1320
K+E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

FIGURE 3

Mass Input

F_V - RCS Volume Factor



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Mass Input

F_Z - Relief Valve Opening Time Factor

FIGURE 4

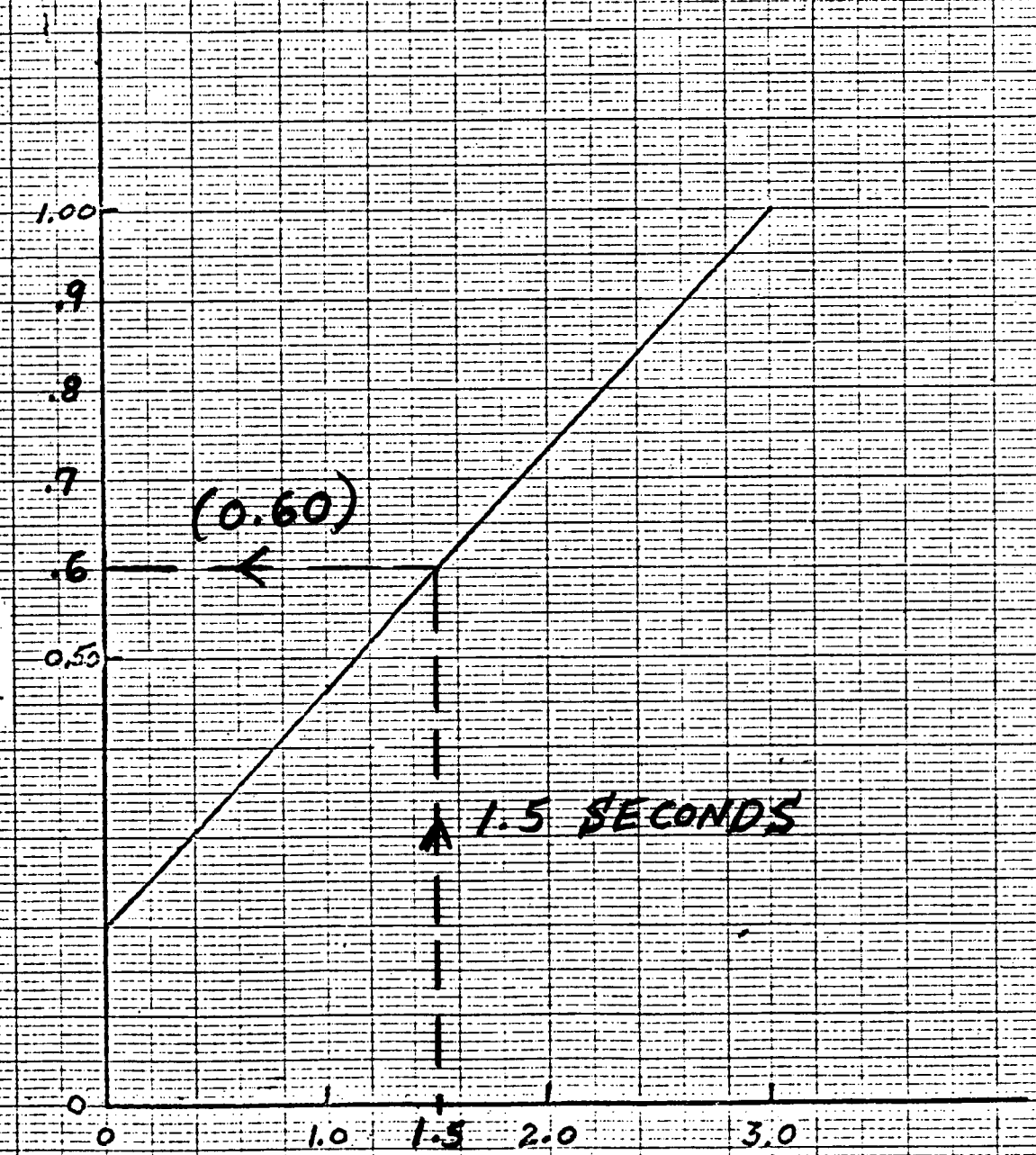

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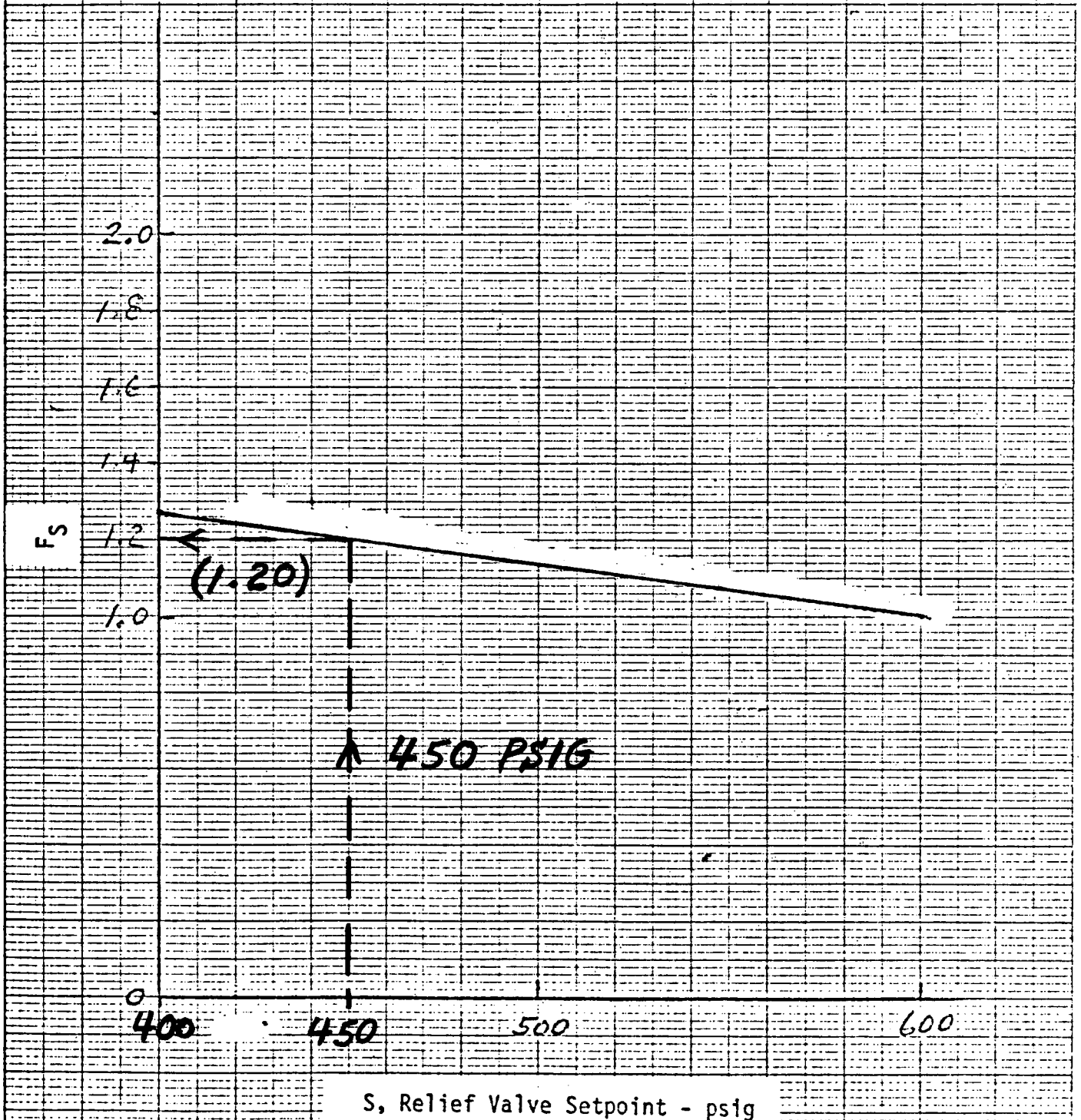
FIGURE 5

Mass Input

F_S - Relief Valve Setpoint Factor



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HEAT INPUT

Calculation of Maximum RCS Temperature when a single Reactor Coolant Pump is started at either Indian Point Unit No. 2 or 3.[4]

The Heat Input cases will be calculated for various initial RCS temperature of 100°F, 140°F, 180°F and 250°F. The basic equations for calculating the maximum RCS pressure and PORV overshoot (above the PORV setpoint) appear below:

$$P_{\substack{MAX \\ RCS}} = P_{\substack{PORV \\ setpoint}} + \Delta P'_{VRCS} \qquad \text{(EQ 3)}$$

$$\Delta P'_{VRCS} = - \frac{(V_{RCS} - 6000)}{(13{,}000 - 6{,}000)} * (\Delta P'_{6k} - \Delta P'_{13k}) + \Delta P'_{6k} \qquad \text{(EQ 4)}$$

The calculations for the Indian Point Unit Nos. 2 and 3 follow: (Four sets of figures 6, 7 and 8 are provided; one set of figures 6, 7 and 8 for each isotherm analyzed).

100°F Isotherm (RCS initial Temperature)

(Refer to figures 6, 7 and 8 following the calculations of the 100°F Isotherm)

ΔP_{6k} = 31 PSI @ 6,000 FT^3, Figure 6

ΔP_{13k} = 27 PSI @ 13,000 FT^3, Figure 6

UA_{6k} = 0.082 @ 31 PSI, Figure 7

UA_{13k} = 0.128 @ 27 PSI, Figure 8

Fraction f, Ratios of Steam Generator Surface Areas

$$f = \frac{44{,}430}{58{,}000} = 0.765 \qquad \text{(EQ 5)}$$

[4]The equations and detailed explanation of the calculations appear in section 4 of the report entitled "Pressure Mitigating System Transient Analysis Results - July 1977".

$UA'_{6k} = f^* UA$ (EQ 6)

$UA'_{13k} = f^* UA_{13k}$ (EQ 6)

$UA'_{6k} = 0.765 \times .082 = .06275$

$UA'_{13k} = 0.765 \times .128 = .0980$

$\Delta P'_{6k} = 22$ PSI @ $UA'_{6k} = .06275$, Figure 7

$\Delta P'_{13k} \cong 20$ PSI @ $UA'_{13k} = .0980$, Figure 8

Substitute into equation 4;

$$\Delta P'_{VRCS} = \frac{-(12,224-6000)}{(13,000-6000)} * (22-20) + 22$$

$$\Delta P'_{VRCS} = -1.78 + 22 \cong 20 \text{ PSI}$$

(it should be noted for the MASS Input Case at 100°F, $\Delta P = 39$ PSI, and the MASS Input case is the controlling case; thus the 39 PSI will be utilized to determine the peak RCS pressure for this 100°F Heat Input case).

$$P_{\substack{MAX \\ RCS}} = P_{\substack{PORV \\ setpoint}} + \Delta P = 450 + 39 = 489 \text{ PSIG}$$

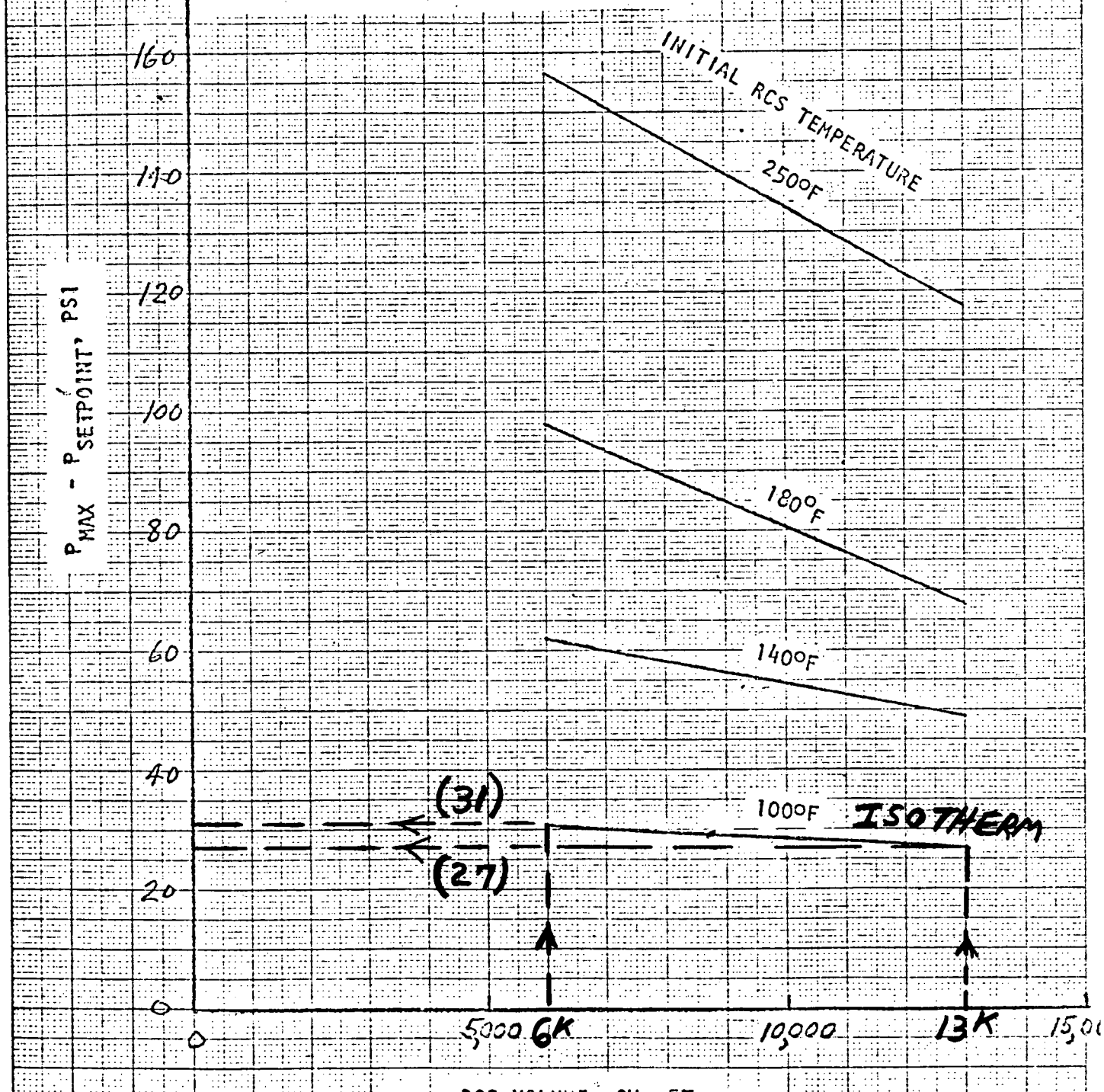
This value of the maximum RCS pressure is below the 10 CFR 50 Appendix G curve (see Figure 1).

Thus the setpoint of 450 PSIG for the power operated relief valve is acceptable for both the MASS and Heat Input cases at 100°F. (a setpoint of 481 PSIG could be utilized at 100°F, as shown in Figure 1, without exceeding the 10 CFR 50 Appendix G curve).

EFFECT OF RCS VOLUME ON PRESSURE OVERSHOOT

FIGURE 6

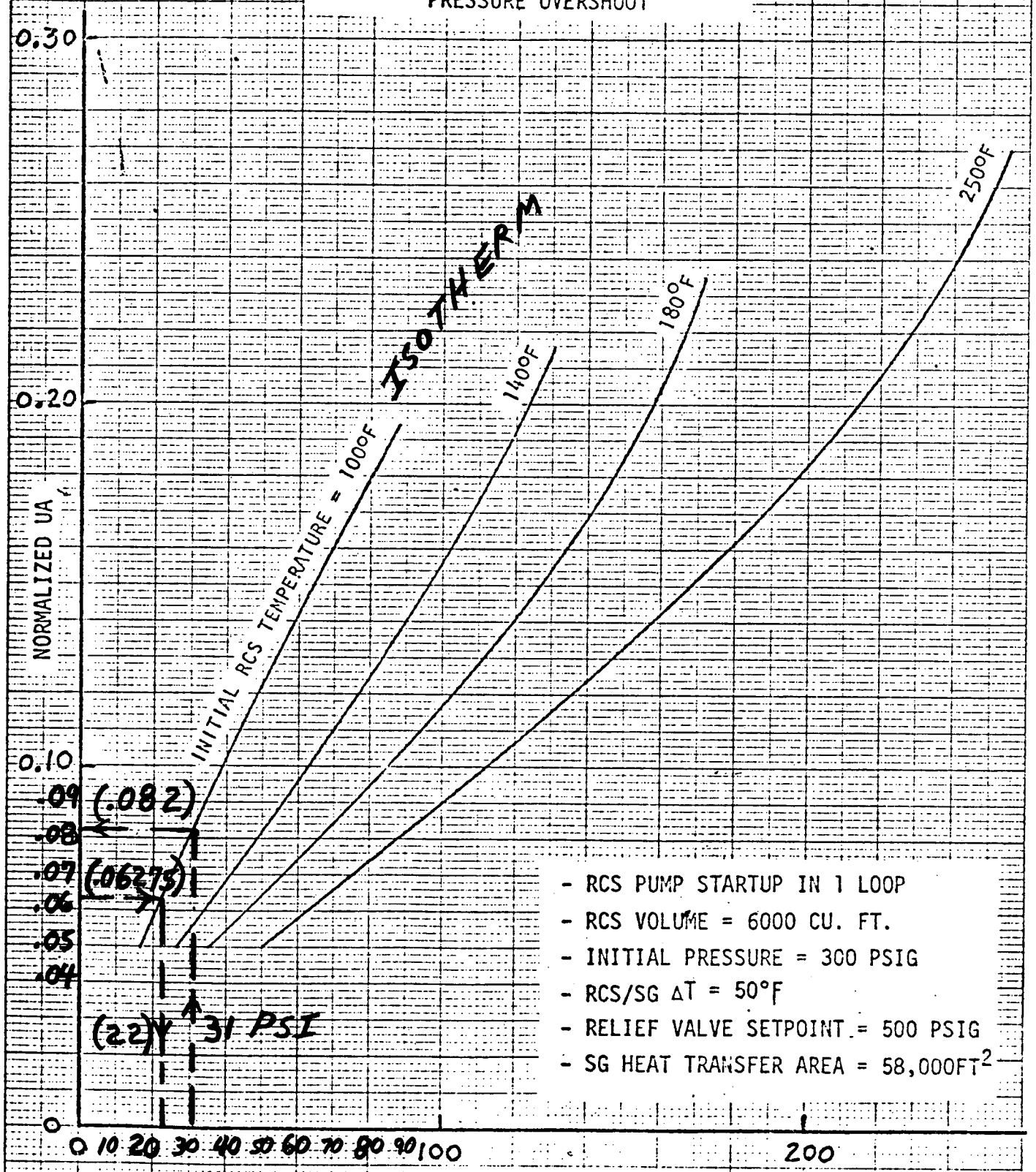
- RCS PUMP STARTUP IN 1 LOOP
- INITIAL PRESSURE = 300 PSIG
- RCS/SG ΔT = 50°F
- RELIEF VALVE SETPOINT = 500 PSIG



461510
K+E 10 X 10 TO THE CENTIMETER 18 X 25 CM. KEUFFEL & ESSER CO. MADE IN U.S.A.

FIGURE 7

EFFECT OF STEAM GENERATOR UA ON PRESSURE OVERSHOOT



- RCS PUMP STARTUP IN 1 LOOP
- RCS VOLUME = 6000 CU. FT.
- INITIAL PRESSURE = 300 PSIG
- RCS/SG ΔT = 50°F
- RELIEF VALVE SETPOINT = 500 PSIG
- SG HEAT TRANSFER AREA = 58,000FT^2

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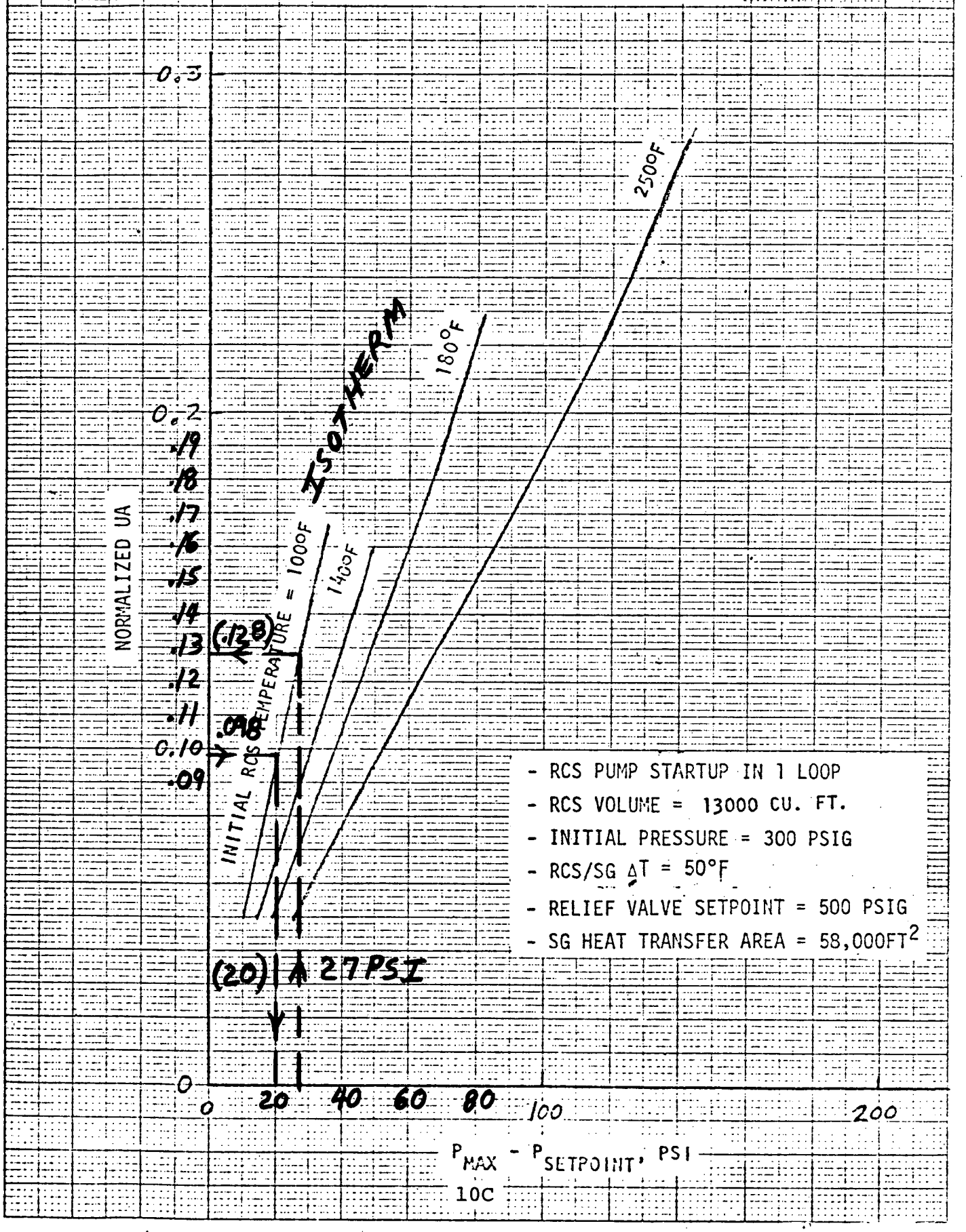
K·E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

FIGURE 8

EFFECT OF STEAM GENERATOR UA ON PRESSURE OVERSHOOT



- RCS PUMP STARTUP IN 1 LOOP
- RCS VOLUME = 13000 CU. FT.
- INITIAL PRESSURE = 300 PSIG
- RCS/SG ΔT = 50°F
- RELIEF VALVE SETPOINT = 500 PSIG
- SG HEAT TRANSFER AREA = 58,000FT^2

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HEAT INPUT
140°F Isotherm (RCS Initial Temperature)

(Refer to figures 6, 7, and 8 following the calculation of the 140°F Isotherm).

ΔP_{6k} = 62 PSI @ 6,000 FT^3, Figure 6

ΔP_{13k} = 49 PSI @ 13,000 FT^3, Figure 6

UA_{6k} = 0.098 @ 62 PSI, Figure 7

US_{13k} = 0.157 @ 49 PSI, Figure 8

$f = \frac{44,430}{58,000} = 0.765$ (substitute into equation 5)

UA'_{6k} = 0.765 x 0.098 = 0.075 (substitute into equation 6)

UA'_{13k} = 0.765 x 0.157 = 0.120 (substitute into equation 6)

$\Delta P'_{6k}$ = 45 PSI @ UA'_{6k} = 0.075, Figure 7

$\Delta P'_{13k}$ = 37 PSI @ UA'_{13k} = 0.120, Figure 8

Substitute into equation 4;

$$\Delta P'_{VRCS} = \frac{-(12,224-6000)}{(13,000-6000)} * (45-37) + 45$$

$$\Delta P'_{VRCS} \cong 38 \text{ PSI}$$

(It should be noted that for the MASS Input Case, ΔP = 39 PSI and the MASS Input case is the controlling case; thus the 39 PSI will be utilized to determine the PORV setpoint for the 140°F isotherm case).

The Indian Point Unit Nos. 2 and 3 power operated relief valves (PORV) will be designed to operate at a variable setpoint (as

opposed to a fixed setpoint). The setpoint will vary as a function of the RCS initial temperature, and the setpoint will be selected so that the peak RCS pressure (at that particular temperature) will not go above the 10 CFR 50 Appendix G limit. Thus by letting the peak RCS pressure, $P_{max_{RCS}}$, equal the Appendix G pressure of 609 PSIG, (at 140°F, Figure 1) the PORV setpoint is calculated from equation 3 as follows:

$$P_{PORV\ setpoint} = P_{MAX\ RCS} - \Delta P'_{VRCS} \ (\text{or } \Delta P_{MASS\ INPUT})$$

$$P_{PORV\ setpoint} = 609 - 39$$

$$P_{PORV\ setpoint} = 570 \text{ PSIG}$$

The PORV setpoints vary as a function of RCS temperature, and Figure 1 shows the PORV curve whose setpoints are below the 10 CFR 50 Appendix G curve, such that the maximum RCS pressure will not exceed the 10 CFR 50 Appendix G limits.

EFFECT OF RCS VOLUME ON PRESSURE OVERSHOOT

FIGURE 6

- RCS PUMP STARTUP IN 1 LOOP
- INITIAL PRESSURE = 300 PSIG
- RCS/SG ΔT = 50°F
- RELIEF VALVE SETPOINT = 500 PSIG

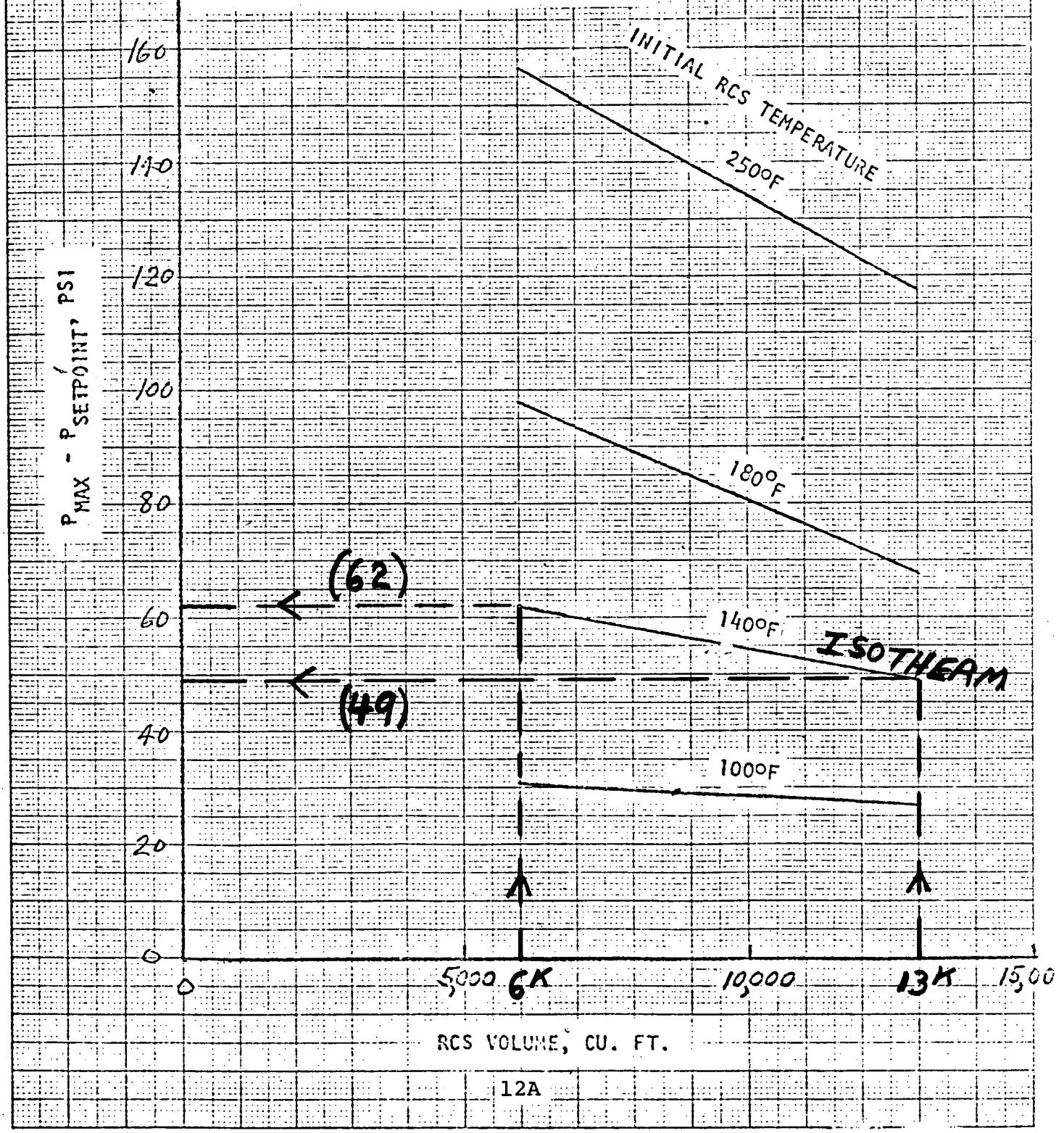

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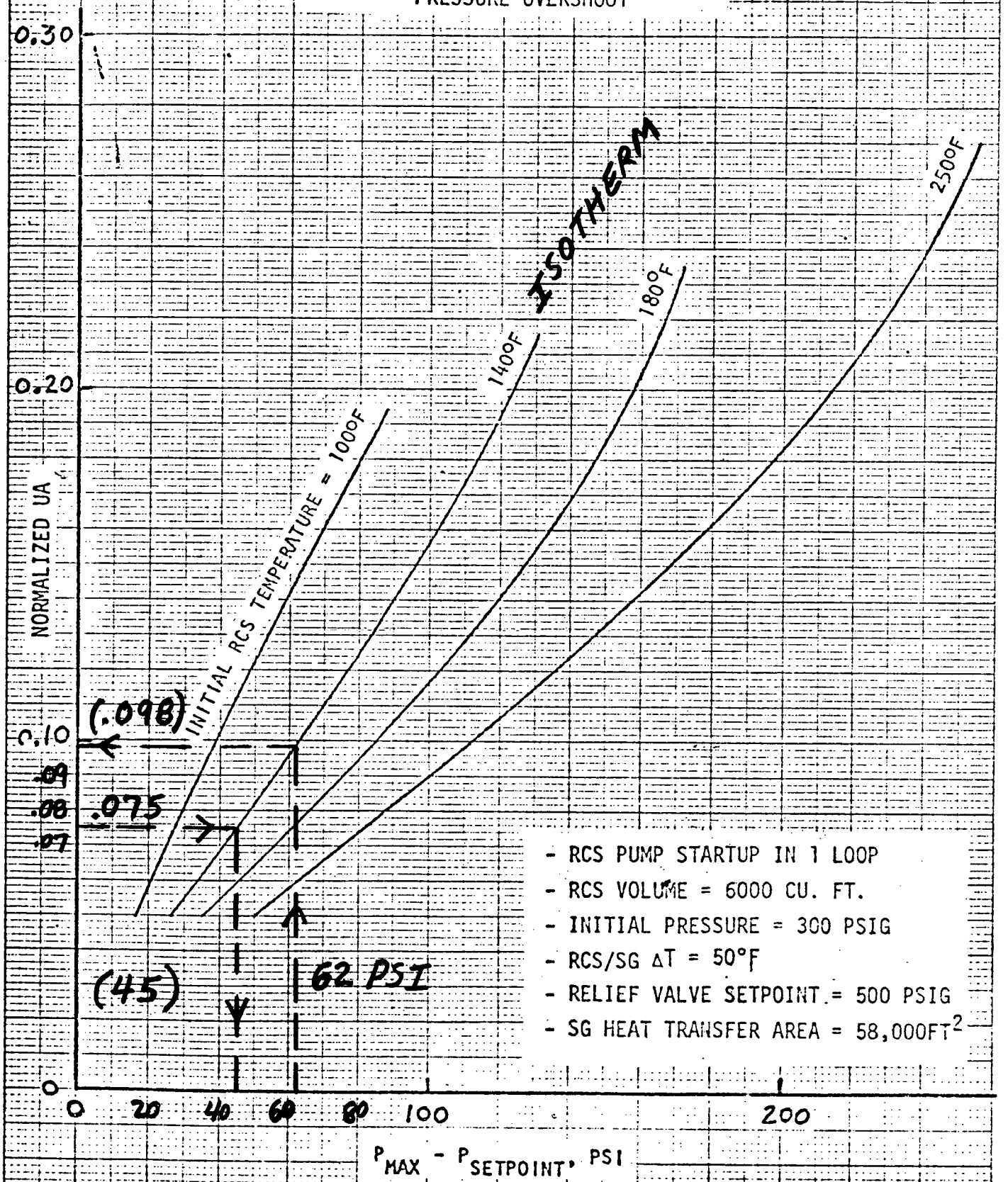
FIGURE 7

EFFECT OF STEAM GENERATOR UA ON PRESSURE OVERSHOOT



- RCS PUMP STARTUP IN 1 LOOP
- RCS VOLUME = 6000 CU. FT.
- INITIAL PRESSURE = 300 PSIG
- RCS/SG ΔT = 50°F
- RELIEF VALVE SETPOINT = 500 PSIG
- SG HEAT TRANSFER AREA = 58,000FT^2

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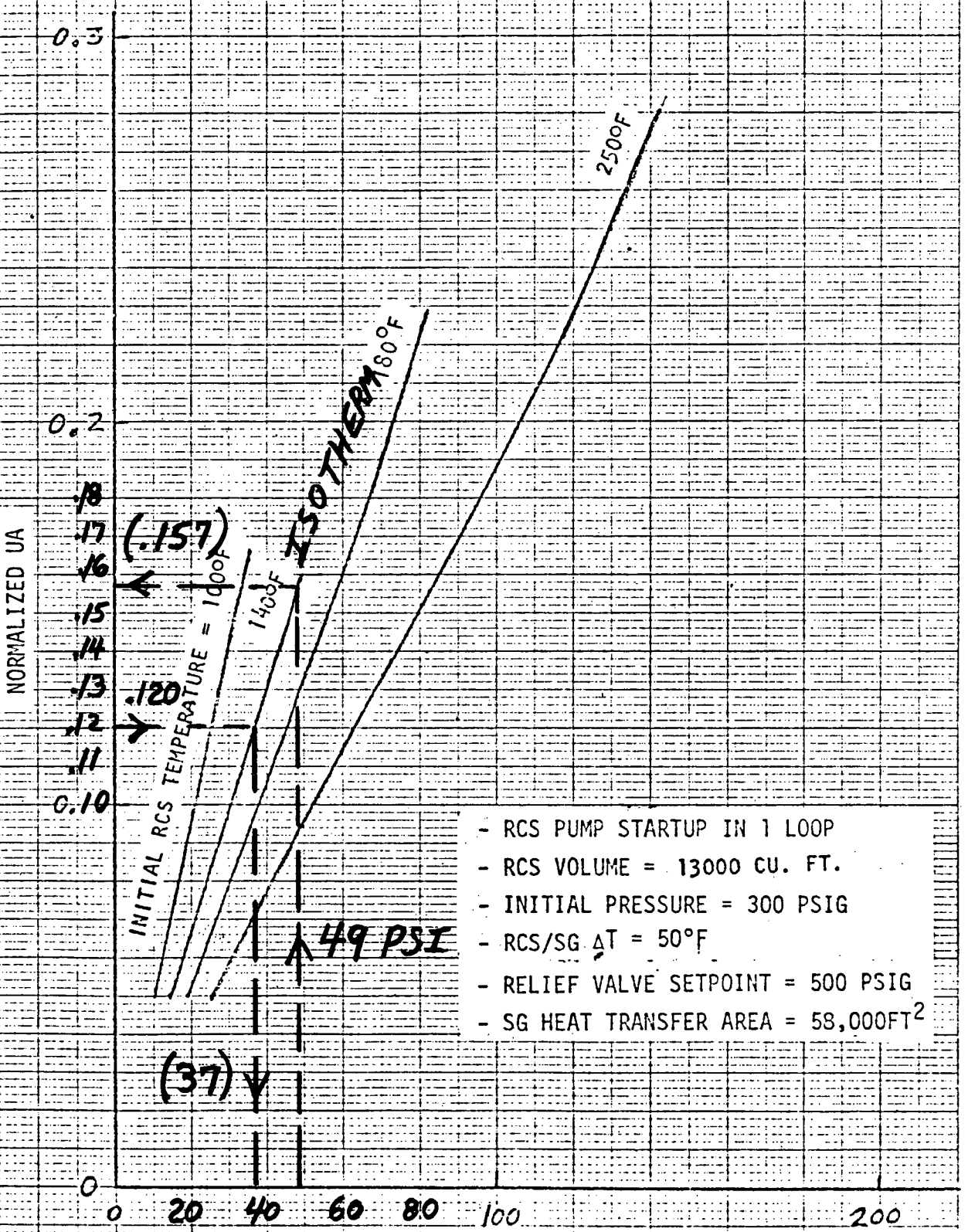
FIGURE 8

EFFECT OF STEAM GENERATOR UA ON PRESSURE OVERSHOOT



- RCS PUMP STARTUP IN 1 LOOP
- RCS VOLUME = 13000 CU. FT.
- INITIAL PRESSURE = 300 PSIG
- RCS/SG ΔT = 50°F
- RELIEF VALVE SETPOINT = 500 PSIG
- SG HEAT TRANSFER AREA = 58,000FT^2

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Heat Input
180°F Isotherm (RCS Initial Temperature)

(Refer to figures 6, 7, and 8 following the calculation of the 180°F isotherm).

ΔP_{6k} = 98 PSI @ 6,000 FT^3, Figure 6

ΔP_{13k} = 68 PSI @ 13,000 FT^3, Figure 6

UA'_{6k} = 0.115 @ 98 PSI, Figure 7

UA'_{13k} = 0.184 @ 68 PSI, Figure 8

$f = \frac{44,430}{58,000} = 0.765$ (substitute into equation 5)

UA'_{6k} = .765 x .115 = .088 (substitute into eq. 6)

UA'_{13k} = .765 x .184 = .141 (substitute into eq. 6)

$\Delta P'_{6k}$ = 73 PSI @ UA'_{6k} = .088, Figure 7

$\Delta P'_{13k}$ = 53 PSI @ UA'_{13k} = .141, Figure 8

substitute into equation 4;

$$P'_{VRCS} = -\frac{(12,224-6000)}{(13,000-6000)} * (73 - 53) + 73$$

$\Delta P'_{VRCS} \doteq$ 55 PSI > ΔP = 39 PSI (MASS INPUT)

From the 10 CFR 50 App G curve (Figure 1), $P_{MAX\ RCS}$ = 765 PSIG @ 180°F

$$P_{PORV\ setpoint} = P_{MAX\ RCS} - \Delta P'_{VRCS}$$

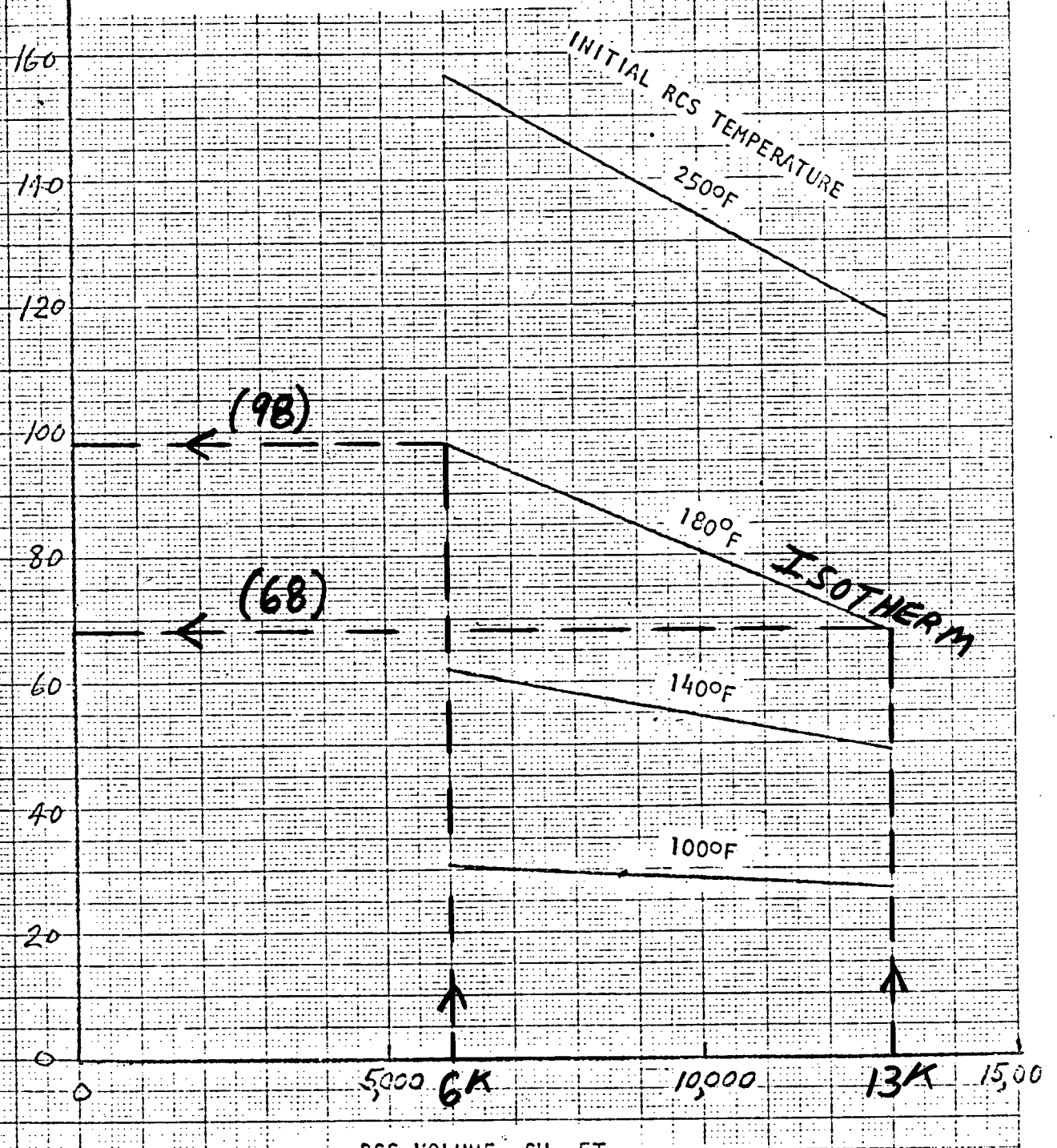
$$P_{PORV\ setpoint} = 765 - 55$$

$$P_{PORV\ setpoint} = 710 \text{ PSIG}$$

FIGURE 6

EFFECT OF RCS VOLUME ON PRESSURE OVERSHOOT

- RCS PUMP STARTUP IN 1 LOOP
- INITIAL PRESSURE = 300 PSIG
- RCS/SG ΔT = 50°F
- RELIEF VALVE SETPOINT = 500 PSIG



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K+E 10 X 10 TO THE CENTIMETER 18 X 25 CM.
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FIGURE 7

EFFECT OF STEAM GENERATOR UA ON PRESSURE OVERSHOOT

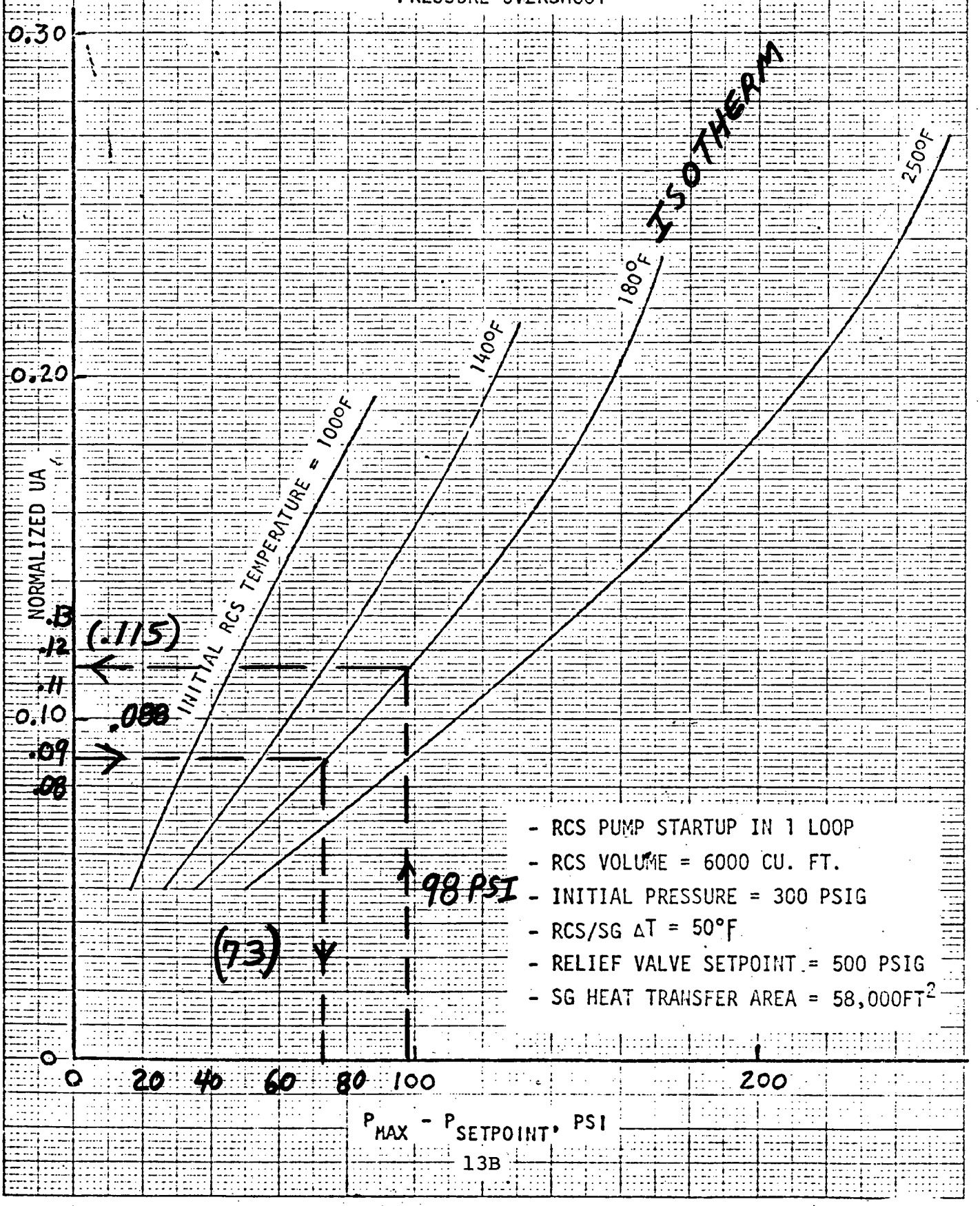

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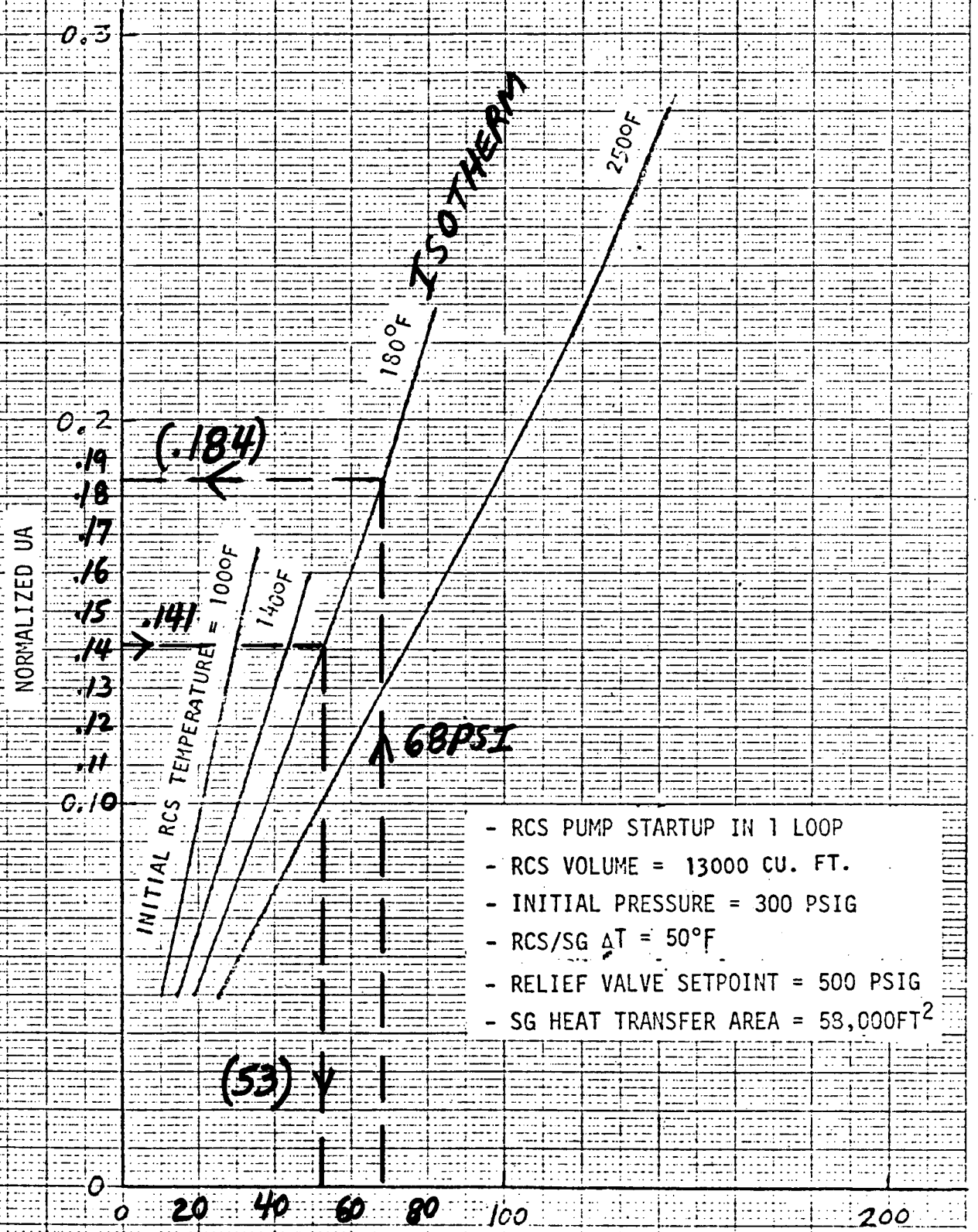
FIGURE 8

EFFECT OF STEAM GENERATOR UA ON PRESSURE OVERSHOOT



- RCS PUMP STARTUP IN 1 LOOP
- RCS VOLUME = 13000 CU. FT.
- INITIAL PRESSURE = 300 PSIG
- RCS/SG ΔT = 50°F
- RELIEF VALVE SETPOINT = 500 PSIG
- SG HEAT TRANSFER AREA = 58,000FT^2

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Heat Input
250°F Isotherm (RCS Initial Temperature)

(Refer to figures 6, 7, and 8 following the calculations of the 250°F Isotherm).

ΔP_{6k} = 156.5 PSI @ 6000 FT^3, Figure 6

ΔP_{13k} = 117.5 PSI @ 13,000 FT^3, Figure 6

UA_{6k} = 0.138 @ 156.5 PSI, Figure 7

UA_{13k} = 0.220 @ 117.5 PSI, Figure 8

$f = \frac{44,430}{58,000} = 0.765$ (substitute into eq. 5)

UA'_{6k} = .765 x .138 = .1057 (substitute into eq. 6).

UA'_{13k} = .765 x .220 = .1684 (substitute into eq. 6).

$\Delta P'_{6k}$ = 118 PSI @ UA'_{6k} = .1057, Figure 7.

$\Delta P'_{13k}$ = 90 PSI @ UA'_{13k} = .1684, Figure 8.

substitute into equation 4;

$$P'_{VRCS} = \frac{-(12,224-6000)}{(13,000-6000)} * (118-90) + 118$$

$P'_{VRCS} \cong$ 93 PSI > 39 PSI (Mass Input).

From the 10 CFR 50 App G curve (figure 1), $P_{MAX\ RCS}$ = 1318 PSIG @ 250°F

$$P_{PORV\ setpoint} = P_{MAX\ RCS} - \Delta P'_{VRCS}$$

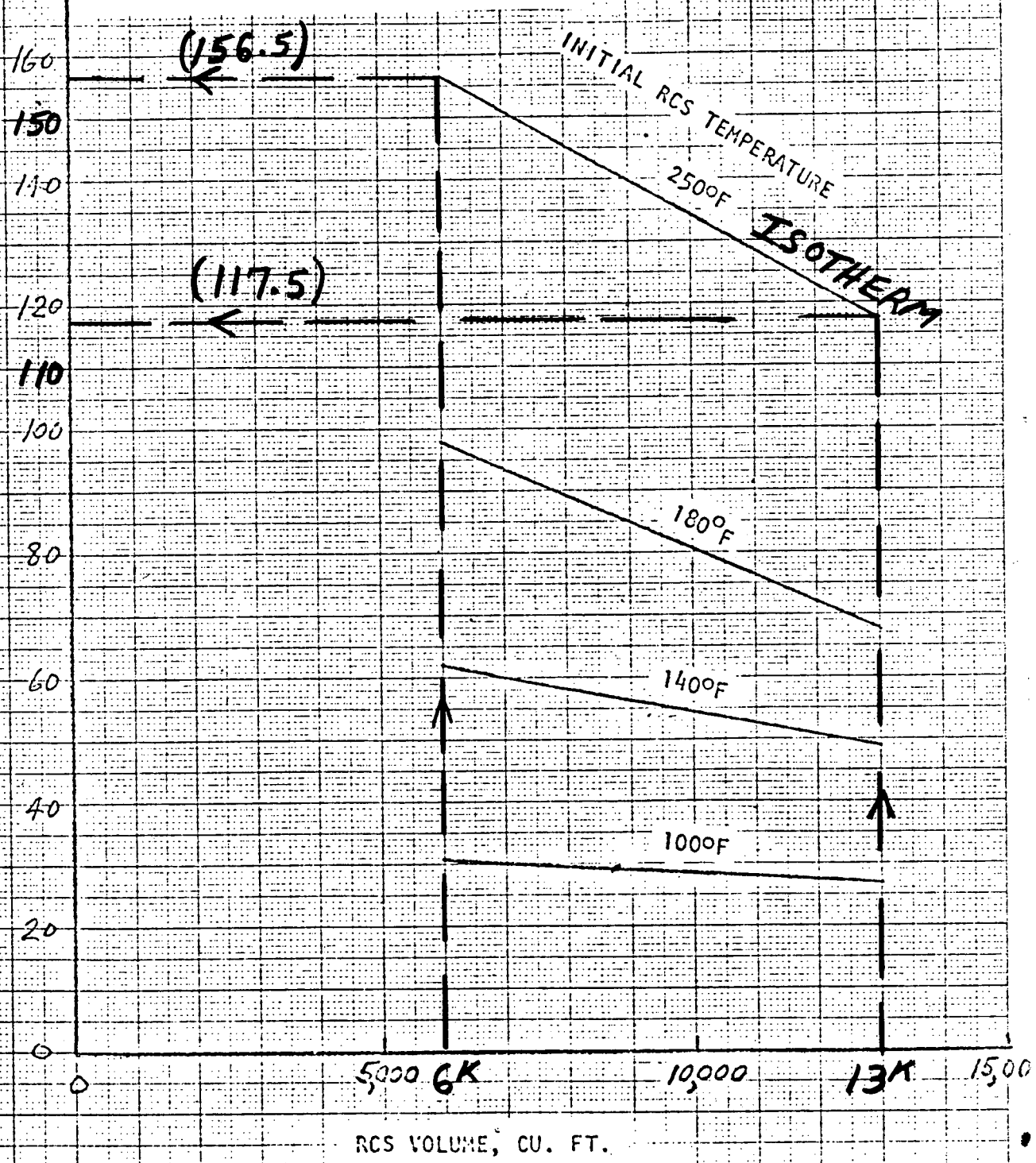
$$P_{PORV\ setpoint} = 1318 - 93$$

$$P_{PORV\ setpoint} = 1225 \text{ PSIG}$$

FIGURE 6

EFFECT OF RCS VOLUME ON PRESSURE OVERSHOOT

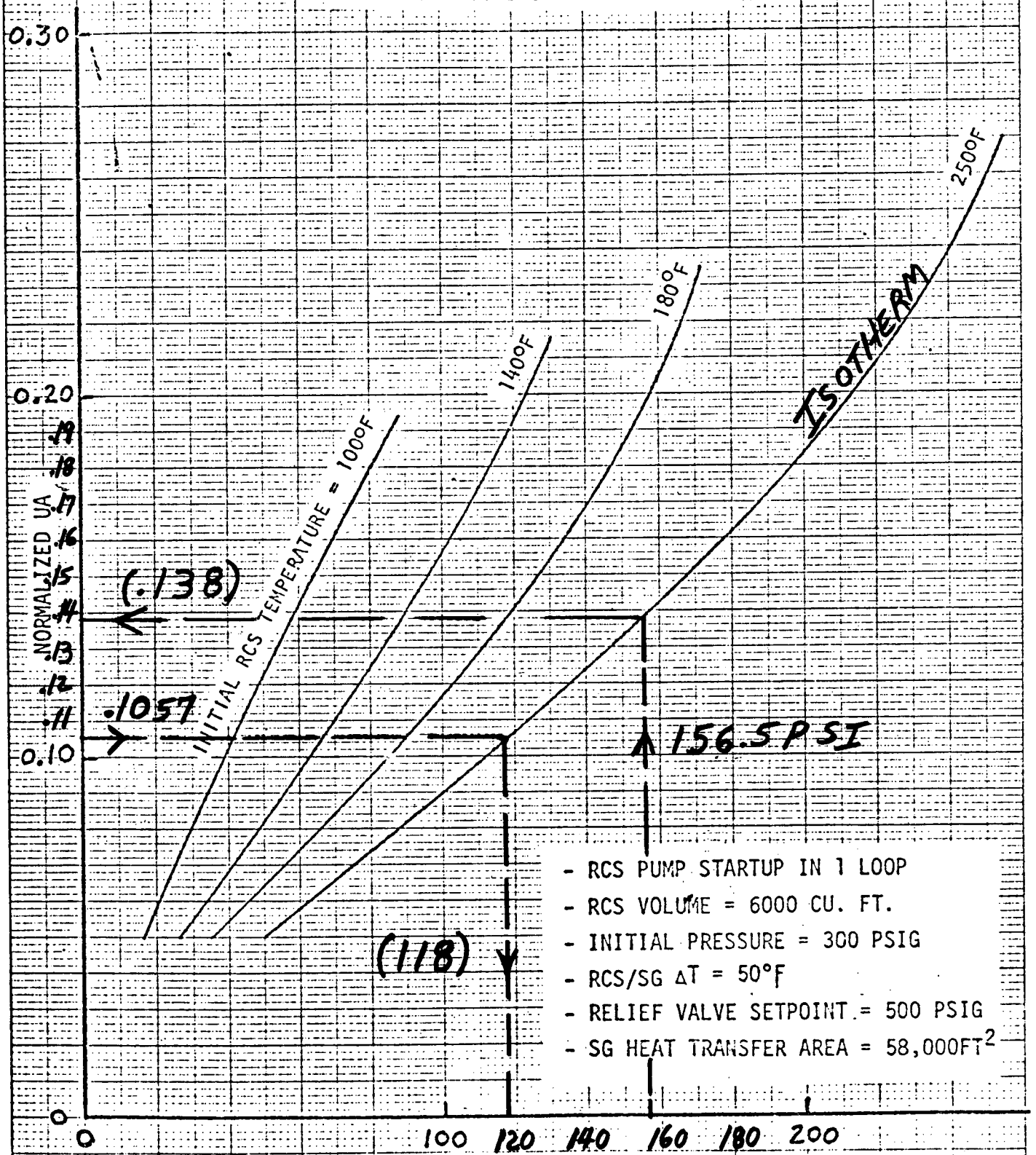
- RCS PUMP STARTUP IN 1 LOOP
- INITIAL PRESSURE = 300 PSIG
- RCS/SG ΔT = 50°F
- RELIEF VALVE SETPOINT = 500 PSIG



461510
K+E 10 X 10 TO THE CENTIMETER 18 X 25 CM.
KEUFFEL & ESSER CO. MADE IN U.S.A.

FIGURE 7

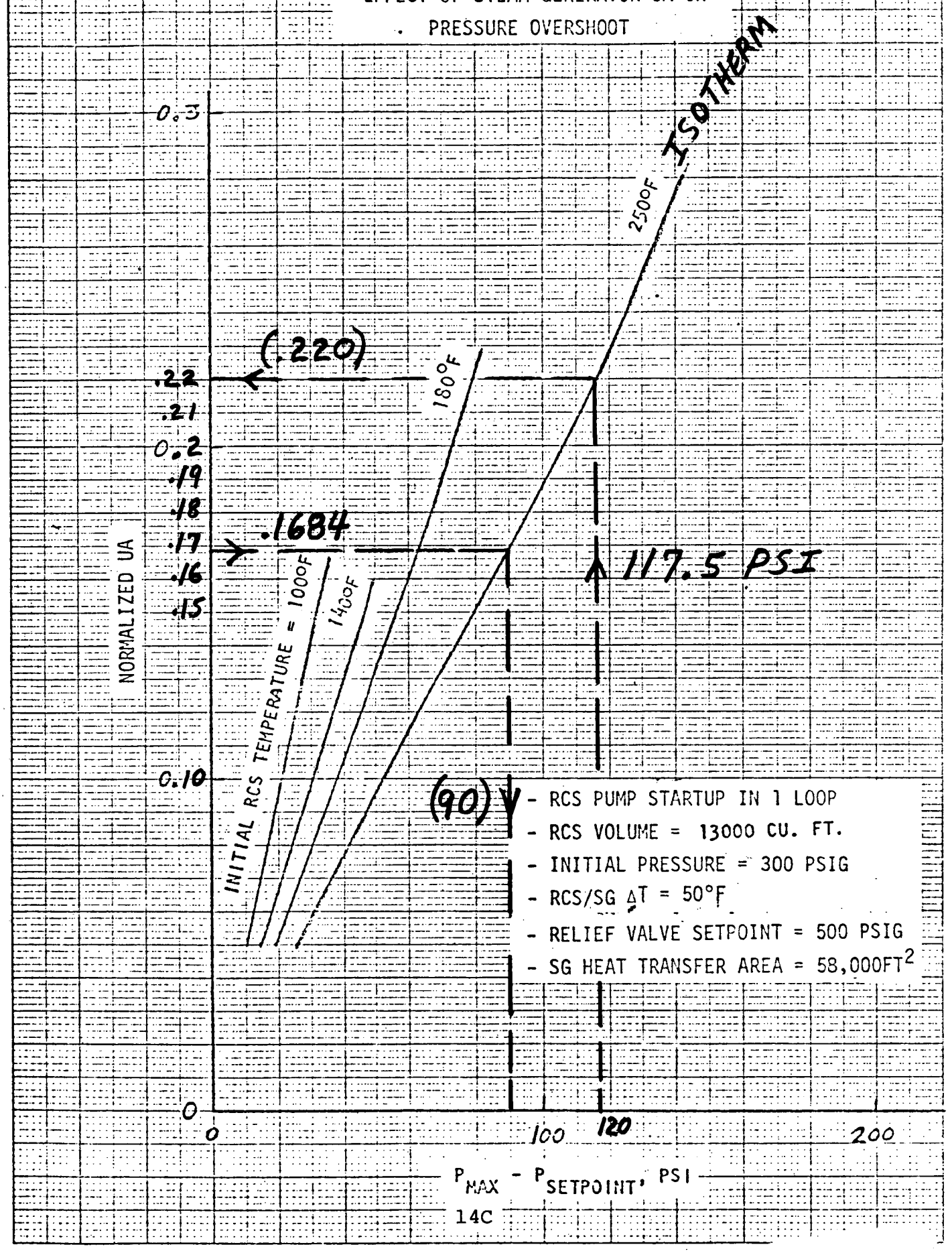
EFFECT OF STEAM GENERATOR UA ON PRESSURE OVERSHOOT



46 1320
K+E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

FIGURE 8

EFFECT OF STEAM GENERATOR UA ON PRESSURE OVERSHOOT



46 1320
K+E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

ATTACHMENT 3

Responses to the NRC Inquiries Concerning the Reactor Coolant System Overpressure Protection System for Indian Point Unit Nos. 2 and 3.

July, 1977

Abstract

The responses to the NRC inquiries are categorized into the following groups:

Group A: Consists of Responses A-2, A-3, A-4 and A-5 and corresponds to the NRC questions which were transmitted in the letter from Mr. Robert R. Reid (NRC) to Mr. William J. Cahill, Jr. (Con Edison), dated February 15, 1977.

Group B: Consists of Responses B-1 through B-7 and corresponds to the seven (7) items which were discussed at the NRC - Utility Group meeting of May 25, 1977.

Group C: Consists of Responses C-1 through C-4 and corresponds to the four (4) items which the NRC transmitted to Mr. R. Newton, Utility Group Chairman, for response by the members of the Utility Group.

INDIAN POINT UNIT NOS. 2 and 3

Question A-2 (Corresponds to Question 2 of NRC letter of 2/15/77)

Provide the failure modes and effects analysis of the overpressure protection system for startup, shutdown, and testing operations which defines the limiting combination of initiating event and additional single failure or operator error subsequent to initiation of the pressure transient.

Response A-2

The generic report entitled "PRESSURE MITIGATING SYSTEMS TRANSIENT ANALYSIS RESULTS - July 1977" demonstrates that a single two-inch air operated power operated relief valve (PORV) is capable of mitigating the worst postulated pressure transients without the need for immediate operator intervention. "The Indian Point Unit Nos. 2 and 3 Plant Specific Interpolation Study - July 1977" demonstrates that the single PORV is capable of mitigating the postulated pressure transient and that the maximum Reactor Coolant System (RCS) pressures will not violate the 10 CFR50 Appendix G pressure limitation of the reactor vessel. The worst postulated initiating events are the starting of a single Safety Injection Pump for the Mass Input cases, and the starting of a single Reactor Coolant Pump with a 50°F temperature difference between the Steam Generator and the RCS for the Heat Input cases. (The RCS is in a water solid mode for the analyses of the Mass and Heat Input cases).

INDIAN POINT UNIT NOS. 2 and 3

The Overpressure Protection System (OPS) for the Indian Point Unit Nos. 2 and 3 plants is a redundant system of power operated relief valves, electric circuits and logic, designed in accordance with the NRC criteria[1] such that a single failure or operator error will not prevent the OPS from performing its intended function of relieving pressure if a postulated pressure transient occurs when these plants are in the cold shutdown mode (water solid).

Thus, the OPS for the Indian Point Unit Nos. 2 and 3 plants is capable of mitigating postulated pressure transients and avoiding a violation of the 10 CFR 50 Appendix G limits for the reactor vessel for the limiting combination of initiating event and additional single failure or operator error subsequent to the initiation of the pressure transient.

Notes: (1) Attachment 1, "Conceptual Design of the Reactor System Overpressure Protection System", to the letters of Mr. William Cahill, Jr. (Con Edison) to Mr. Robert Reid (NRC) dated February 28, 1977 for the Indian Point Unit Nos. 2 and 3 plants.

INDIAN POINT UNIT NOS. 2 and 3

Question A-3 (Corresponds to Question 3 of NRC letter of 2/15/77)

Discuss the basis for determining the most limiting initial conditions for analysis of the pressure transient. Items that must be considered include, but should not be limited to: RCS pressure, reactor coolant temperature, pressurizer temperature, pressurizer level and accumulator level.

Response A-3

Sections 1 and 2 of the generic report (which appears as Attachment 1 to this letter) entitled, "Pressure Mitigating Systems Transient Analysis Results - July 1977" discuss conditions for analysis of the pressure transient. Attachment 2 to this letter, the report entitled, "The Indian Point Unit Nos. 2 and 3 Plant Specific Interpolation Study - July 1977" applies these limiting conditions to the Indian Point Unit Nos. 2 and 3 Plants. Also refer to Response A-2.

INDIAN POINT UNIT NOS. 2 and 3

Question A-4 (Corresponds to Question 4 of NRC letter of 2/15/77)

Were approved systems computer codes used to model the over-pressure transients as presented in the generic report? If so, were any major modifications required? If the methods utilized have not been previously approved for system transient analyses, provide model descriptions, flow charts, program listing, and sensitivity studies to verify the performance of the codes.

Response A-4

Section 2.1 "LOFTRAN Program and Special Modeling" of the generic report entitled "Pressure Mitigating Systems Transient Analysis Results - July 1977" discusses the use of the Loftran computer code.

INDIAN POINT UNIT NOS. 2 and 3

Question A-5 (Corresponds to Question 5 of NRC letter of 2/15/77)

Provide the value of the backpressures assumed in the relief valve discharge pipes when calculating relief capacity. Discuss relief capacity sensitivity to this parameter.

Response A-5

Section 6.2 "Effect of Backpressure on Relief Valve" of the generic report entitled "Pressure Mitigating Systems Transient Analysis Results - July 1977" discusses the effect of back-pressure. Also refer to figure 2.2.1, of the above generic report, (Page 2-9) "Flow Through Valve vs. RCS Pressure", and to Section 2.2, "Reference Relief Valve Model". Backpressures of 10 psig to 100 psig were considered.

INDIAN POINT UNIT NOS. 2 and 3

Question B-1

Previous plant submittals deferred addressing acceptance criteria until generic analysis was completed. Provide this information.

Response B-1

The Con Edison conceptual design of the Overpressure Protection System (OPS) appeared in Attachment 1 to the Indian Point Unit Nos. 2 and 3 letters of February 28, 1977 from Mr. W. Cahill, Jr. of Con Edison to Mr. R. Reid of the NRC. The February 28, 1977 submittal addressed the NRC criteria for no credit for operator action (10 minutes), single failure criteria, testability, seismic design and IEEE 279 criteria.

"The Indian Point Unit Nos. 2 and 3 Plant Specific Interpolation Study - July 1977" demonstrates that a single power operated relief valve (two PORV's are provided for redundancy) is capable of mitigating the postulated pressure transients, and the peak RCS pressures resulting from these transients are below the 10 CFR 50 Appendix G limit for the reactor vessel.

INDIAN POINT UNIT NOS. 2 and 3

Question B-2

Identify plant conditions, and components/systems to be controlled by Administrative Procedures - consider components/systems assumed to be unavailable and therefore not analyzed.

Response B-2

A summary of the plant administrative procedures has been submitted to the NRC Staff in Attachment 1 to the Indian Point Unit Nos. 2 and 3 letters of October 25, 1976 from Mr. W. Cahill, Jr. of Con Edison to Mr. R. Reid of the NRC. The defeat of automatic S. I. is described in these administrative procedures(1), and it could be concluded that the S.I. system need not be considered for analyses (as a cause of a postulated RCS pressure transient). However, it is our understanding that although the NRC Staff agrees that these administrative procedures are helpful in eliminating postulated RCS pressure transients, the NRC Staff is of the opinion that there still remains a remote possibility that a plant operator (due to some unexplained reason) might start a single S.I. pump when the RCS is in a water solid mode; thus the case for the starting of a single S.I. pump was analyzed. The results of the pressure transient analyses(2) show that a single power operated relief valve is capable of mitigating the consequences of a pressure transient which is caused by the starting of a single S.I. pump. Also the plant specific study(3) shows that the peak RCS pressures for the Indian Point Unit Nos. 2 and 3 Plants are below the 10 CFR 50 Appendix G limits if such a transient should occur.

INDIAN POINT UNIT NOS. 2 and 3

The case for exempting the S.I. accumulators from analyses was stated in the Indian Point Unit Nos. 2 and 3 letters of October 25, 1976 from Mr. W. Cahill, Jr. of Con Edison to Mr. R. Reid of the NRC. The third paragraph of page 3 of these letters stated:

"The preliminary evaluations performed indicate that the use of the pressurizer relief system for protection of a solid reactor coolant system from an inadvertently opened safety injection accumulator, which is charged to its design pressure, is not practical. It is our opinion that administrative controls similar to the controls employed for assuring that the accumulator valves are open during normal operation, which satisfy the single failure criteria, will provide adequate protection against over-pressurization during solid system operation. We are revising our procedures to incorporate such controls and are continuing to investigate the feasibility of automatic systems to supplement these controls."

Notes:

(1) The Con Edison responses to NRC questions concerning defeat of automatic S.I. appear in Responses 9a through 9e of Attachment 2 to the Indian Point Unit Nos. 2 and 3 letters of February 28, 1977 from Mr. W. Cahill, Jr. of Con Edison to Mr. R. Reid of the NRC.

(2) "Pressure Mitigating Systems Transient Analysis Results - July 1977"

(3) "Indian Point Unit Nos. 2 and 3 Plant Specific Interpolation Study - July 1977"

INDIAN POINT UNIT NOS. 2 and 3

Question B-3

Include electrical schematics and logic diagrams.

Response B-3

The logic diagrams have previously been submitted to the Commission; they appear on Page 6 of Attachment 1, "Conceptual Design of the Reactor Coolant System Overpressure Protection System" to the letters from Mr. William Cahill, Jr. (Con Edison) to Mr. Robert Reid (NRC) dated February 28, 1977 for the Indian Point Unit Nos. 2 and 3 plants.

INDIAN POINT UNIT NOS. 2 and 3

Question B-4

Dual setpoint "enabling" alarm must be included, a computer-generated alarm is not acceptable; alarm must be hard wired to the PORV isolation valve to assure it is opened when required.

Response B-4

The Indian Point Unit Nos. 2 and 3 plants have an automatic arming feature which will open the motor operated isolation valves which are located upstream of the power operated relief valves. This arming feature includes an alarm in the Central Control Room which will inform the plant operator that the Overpressure Protection System is armed (i.e., the motor operated isolation valves are opened).

INDIAN POINT UNIT NOS. 2 and 3

Question B-5

Provide additional details on modeling of PORV's - consider effects of flashing on assumed relief capacity.

Response B-5

Section 2.2, "Reference Relief Valve Model" of the generic report entitled "PRESSURE MITIGATING SYSTEMS TRANSIENT ANALYSIS RESULTS - July 1977" provides the details on modeling of the PORV's.

Section 6.5, "Relief Valve Capacity Change With Flashing" of the generic report entitled "PRESSURE MITIGATING SYSTEMS TRANSIENT ANALYSIS RESULTS - July 1977" discusses the effects of flashing on the relief valve capacity.

INDIAN POINT UNIT NOS. 2 and 3

Question B-6

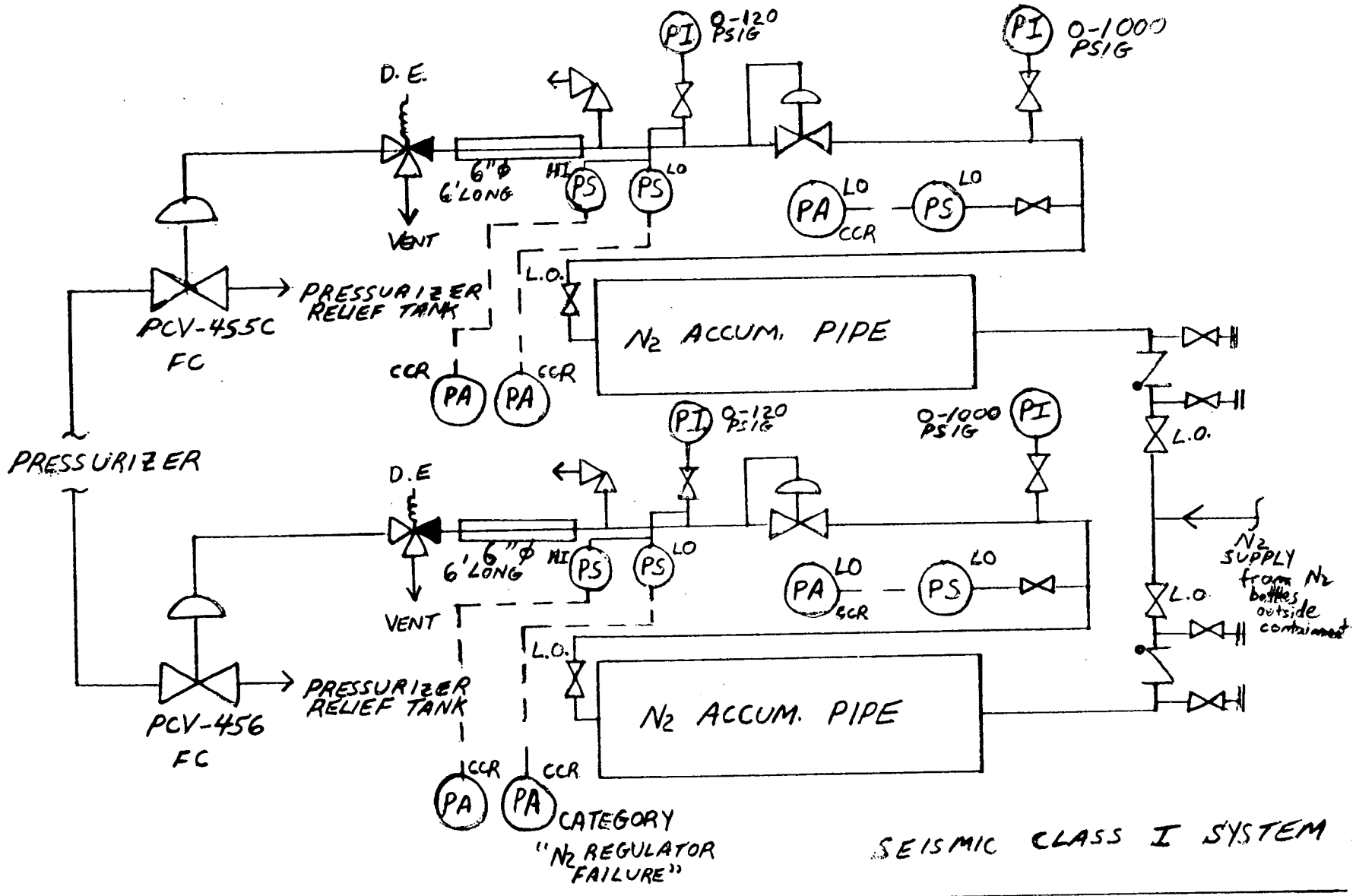
Explain how the criteria will be satisfied regarding loss of air/ loss of offsite power. Must be addressed in plant specific submittal.

Response B-6

The OPS for the Indian Point Unit Nos. 2 and 3 Plants will be provided with a new nitrogen system, including redundant nitrogen accumulators. In addition to normal operation, each of the nitrogen accumulators will be sized to permit 10 minutes of cyclic operation of the PORV's in case the main nitrogen supply is lost; this conforms to the NRC criteria of "No credit can be taken for operator action until 10 minutes after the operator is aware that a pressure transient is in progress" and a "single failure" to the nitrogen system. (The new nitrogen system will replace the existing instrument air system supply to the PORV's, and loss of instrument air will have no effect on the operation of the PORV's.) Sketch B-6 is a flow diagram of the nitrogen system, and accumulators, for the operation of the PORV's. The nitrogen used in this system is supplied from 2000 PSI bottles located outside containment.

The OPS for the Indian Point Unit Nos. 2 and 3 plants will be provided with three (3) independent D.C. electrical supplies which go to inverters and supply three (3) 118 volt A.C. instrument busses (1 bus per instrument channel). Additionally the two (2) PORV's and their actuation logic are supplied from 2

separate D.C. busses. Thus, the criteria concerning single failure of the electrical supply or loss of offsite power is satisfied.



SEISMIC CLASS I SYSTEM

SKETCH B-6
INDIAN POINT UNITS 2 and 3
NITROGEN SYSTEM FOR
OPERATION OF PORV'S.

INDIAN POINT UNIT NOS. 2 and 3

Question B-7

Final design must be capable of satisfying Appendix G limit over 40 year plant life. If plant can't meet Appendix G, you may choose to propose meeting a lesser Appendix G fluence level, but with the understanding that these would be considered as interim design changes and a final design must be proposed and changes implemented within next refueling outage.

Response B-7

Refer to Figure 1 of the report entitled, "Indian Point Unit Nos. 2 and 3 Plant Specific Interpolation Study - July 1977". Figure 1 shows the 10 CFR 50 Appendix G limits based on 3 and 2 Effective Full Power Years (EFPY) of plant operation, for Indian Point Unit No. 2 and Indian Point Unit No. 3 respectively, and is considered as an interim Appendix G fluence level.

The OPS design for Indian Point Unit Nos. 2 and 3 is considered a final design, and will assure that the 10 CFR 50 Appendix G limits will not be exceeded for the life of the plant. The PORV setpoint curve will be changed when the Appendix G curve is changed, so that the PORV setpoint curve corresponds to the Appendix G curve contained in the plant Technical Specification at that particular time.

INDIAN POINT UNIT NOS. 2 and 3

Question C-1

Reference Attachment 4 to June 8, 1977 letter to Utility Group Participants. Item 5 will be addressed in the generic report. The second part of item 4 requests an alarm on the PORV isolation valves. What the NRC is willing to accept for this is a non-safety grade status light which shows the system to be operable and includes the open position of the PORV isolation valves.

Response C-1

Item 5 refers to Question B-5 which has been answered in Response B-5.

Item 4 refers to Question B-4 which has been answered in Response B-4.

Question C-2

Previous NRC questions concerning the model used in the evaluation will be addressed in the generic report.

Response C-2

The previous NRC questions concerning the model used in the evaluation have been answered in Responses A-2, A-3, A-4 and A-5. These questions were dealt with in the generic report entitled "PRESSURE MITIGATING SYSTEMS TRANSIENT ANALYSIS RESULTS - July 1977".

INDIAN POINT UNIT NOS. 2 and 3

Question C-3

The method for handling valve cycling with loss of instrument air must be addressed for each plant based on the method used to supply gas to the valve.

Response C-3

Refer to Response B-6, and Sketch B-6, for the discussion of the nitrogen system for the operation of the PORV's. (The new nitrogen system will replace the existing instrument air system supply to the PORV's, and loss of instrument air will have no effect on the operation of the PORV's.)

Question C-4

Surveillance requirements must be described for plant conditions associated with the mitigating system, eg., if a bottled air supply is used in case of loss of instrument air, how often is the pressure of the bottle checked. Items dealing with administrative controls may require surveillance. This is separate from the testing requirement.

Response C-4

High - low pressure alarms, located in the Central Control Room, will be provided in order to meet the surveillance requirement for the nitrogen system, which will operate the PORV's. Also refer to Response B-6, and Sketch B-6, for the discussion of the nitrogen system, and accumulators for the operation of the PORV's.