UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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|---|---|---|
| In the Matter of | ) | |
| | ) | |
| CONSOLIDATED EDISON COMPANY | ) | Docket No. 50-247 |
| OF NEW YORK, INC. | ) | |
| (Indian Point Station, | ) | |
| Unit No. 2) | ) | |

AMENDMENT NO. 1 TO
APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

On November 3, 1978, Consolidated Edison Company of New York, Inc. ("Consolidated Edison"), as holder of Facility Operating License No. DPR-26, filed with the U. S. Nuclear Regulatory Commission (NRC) an "Application for Amendment to Operating License", sworn to by Mr. William J. Cahill, Jr. on November 3, 1978. That Application requested changes to the Indian Point Unit No. 2 Technical Specifications to establish limiting conditions for operation (LCOs) and surveillance requirements for the recently installed Overpressure Protection System (OPS). That Application was submitted in response to the letter dated August 28, 1978 from Mr. A. Schwencer (NRC) to Mr. William J. Cahill, Jr. (Consolidated Edison).

Consolidated Edison hereby amends its November 3, 1978 Application to include the results of our pressurizer bubble analyses and to incorporate new reactor coolant system heatup and cooldown curves and a new OPS setpoint limit curve applicable through five (5) effective full power years (EFPY) of reactor operation. Attachment A of this Amendment contains a complete package of proposed Technical Specification page revisons. These new proposed pages incorporate those changes requested by our November 3, 1978 Application as modified and supplemented by this Amendment.

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A Safety Evaluation of the proposed changes is set forth in Attachment B of this Amendment. This evaluation demonstrates that the proposed changes do not represent a significant hazards consideration and will not cause any change in the types or an increase in the amounts of effluents or any change in the authorized power level of the facility.

CONSOLIDATED EDISON COMPANY
OF NEW YORK, INC.

By: William J Cahill Jr
William J. Cahill, Jr.
Vice President

Subscribed and sworn to before
me this 5th day of January, 1979.

Angela Roberti
Notary Public
ANGELA ROBERTI
Notary Public, State of New York
No. 41-8593813
Qualified in Queens County
Commission Expires March 30, 1980

ATTACHMENT A

Amendment No. 1 to
Application for Amendment
To Operating License

Technical Specification
Page Revisions

Consolidated Edison Company of New York, Inc.

Indian Point Unit No. 2

Docket No. 50-247

January, 1979

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LIST OF FIGURES

## 3 LIMITING CONDITIONS FOR OPERATION

### 3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

Specification

A. OPERATIONAL COMPONENTS

1. Coolant Pumps

a. At least one reactor coolant pump or one residual heat removal pump in the Residual Heat Removal System when connected to the Reactor Coolant System shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

b. When the reactor is critical and above 2% rated power, except for natural circulation tests, at least two reactor coolant pumps shall be in operation.

c. Reactor power shall not be increased above 60% of rated power with only three pumps in operation unless the overtemperature $\Delta T$ trip setpoint for three loop operation has been set in accordance with specification 2.3.1.B-4.

d. Reactor operation with one of the four loops out of service will be permitted for up to 24 hours. If the fourth loop can not be returned to service within 24 hours, the reactor will be put in a hot shutdown condition using normal procedures.

e. A reactor coolant pump may be started if:

(1) another reactor coolant pump is running, or

(2) the secondary side temperature of each steam generator ($T_{SG}$) has been verified to be no more than 50°F greater than the reactor coolant temperature ($T_{RC}$), or

(3) a nitrogen bubble has been established in the pressurizer with a *minimum* gas volume ($V_{N_2}$) as follows:

$$V_{N_2} = 3.39\ (T_{SG} - T_{RC}) - 214$$

or

$125\ ft^3$, whichever is greater.

(For $T_{RC} > 180°F$, use $T_{RC} = 180°F$)

2. Steam Generator

Two steam generators shall be capable of performing their heat transfer function whenever the reactor is critical and the average coolant temperature is above 350°F.

3. Safety Valves

a. At least one pressurizer code safety valve shall be operable whenever the reactor head is on the vessel except for hydrostatically testing the RCS in accordance with the applicable ASME Section XI Boiler and Pressure Vessel Code.

b. All pressurizer code safety valves shall be operable whenever the reactor is critical.

c. The pressurizer code safety valve lift setting shall be set at 2485 psig with ±1% allowance for error.

4. Overpressure Protection System (OPS)

a. The OPS shall be "armed" and "operable" whenever the reactor coolant system (RCS) temperature is below 290°F and the RCS is not depressurized and vented with an equivalent opening of ≥ 2.00 square inches. The OPS pressurizer power operated relief valves (PORVs) shall have lift settings within the limits of the PORV setpoint curve specified in Figure 3.1-3 whenever the OPS is required to be operable.

b. The requirements of 3.1.A.4.a may be modified to permit one PORV and/or its series MOV to be inoperable for a maximum of seven (7) consecutive days. If the single PORV and/or its series MOV is not restored to meet the requirements of 3.1.A.4.a within this seven (7) day period, or if both PORVs and/or their series MOVs are in-

operable when required to be operable by 3.1.A.4.a, then, utilizing normal operating procedures, either:

(1) the RCS must be depressurized and vented with an equivalent opening of $\geq$ 2.00 square inches, or

(2) the RCS must be heated and maintained above 338°F.

c. If the requirements of 3.1.A.4.b cannot be satisfied, then the plant may be brought to the cold shutdown condition only for the following limited conditions:

(1) the repairs cannot be accomplished within the seven (7) day time period requirement of 3.1.A.4.b or cannot be performed under hot conditions, or

(2) another action statement requires cooldown, or

(3) protection and safety of plant personnel or equipment requires cooldown.

d. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2.f within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

Basis

When the boron concentration of the Reactor Coolant System is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant.

Heat transfer analyses show that reactor heat equivalent to 10% of rated power can be removed with natural circulation only[1]; hence, the specified

upper limit of 2% rated power without operating pumps provides a substantial safety factor.

Three loop operation is allowed over a 24 hour period to permit corrective action to return the fourth loop to service and limit the number of unnecessary shutdown cycles. During these periods of three loop operation, the reactor coolant system parameters will be maintained within the limits described for three loop operation in Section 2.1 and 3.1 of the Technical Specifications.

Each of the pressurizer code safety valves is designed to relieve 408,000 lbs. per hr. of saturated steam at the valve set point. Below approximately 350°F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure.[2]

If no residual heat were removed by the Residual Heat Removal System the amount of steam which could be generated at safety valve relief pressure would be less than half the capacity of a single valve. One valve therefore provides adequate protection for over-pressurization.

The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from complete loss of load[3] without a direct reactor trip or any other control.

Two steam generators capable of performing their heat transfer function will provide sufficient heat removal capability to remove core decay heat after a reactor shutdown.

The OPS is designed to relieve the RCS pressure for certain unlikely incidents to prevent peak RCS pressure from exceeding the 10 CFR 50, Appendix G, limits.

When the OPS is "armed" it means that MOVs 535 and 536 are in the open position. This OPS "arming" can be accomplished either automatically by the OPS when the RCS temperature is $<290°F$ or manually by the control room operator.

The likelihood of occurrence of the Heat Input initiating event (i.e., SG/RC temperature differential) for overpressure of the RCS when the RCS is below 290°F is significantly reduced if another RC pump is running, or the steam

generator secondary side temperature is no more than 50°F greater than the reactor coolant temperature, or the appropriate nitrogen bubble exists in the pressurizer. For assumed SG/RC temperature differentials, the equation specified in specification 3.1.A.1.e.(3) provides conservative bounding minimum pressurizer nitrogen bubble volumes necessary to assure that 10 CFR 50, Appendix G, limits are not exceeded for postulated heat input initiated events. That equation was developed from the results of parametric studies of the heat input initiated transient assuming an initial reactor coolant temperature of 180°F and varying SG/RC temperature differentials up to 170°F (i.e., the differential associated with a SG temperature of 350°F - the highest temperature at which changeover from secondary side decay heat removal to the RHR system could take place). An initial reactor coolant temperature of 180°F was assumed for the parametric studies since the initial RCP is normally started below that temperature and since for a given SG/RC temperature differential the higher the initial reactor coolant temperature the greater the transient pressure overshoot.

Therefore, based on the parametric studies performed, the equation for the minimum nitrogen volume requires that for SG/RC temperature differentials less than 100°F, the minimum bubble size for the 100°F differential (i.e., 125 $ft^3$) is conservatively used. Likewise, for initial reactor coolant temperatures greater than 180°F, the temperature is conservatively assumed to be 180°F (thus increasing the assumed temperature differential to compensate for increased initial reactor coolant temperature).

The determination of $T_{RC}$ may be made from the Control Room instrumentation. The determination of $T_{SG}$ may be made in the following ways:

(a) Actual measurement of the secondary side steam generator water temperature, or

(b) Conservatively assuming that the secondary side water temperature is at the reactor coolant temperature at which the last RCP was stopped during cooldown, or

(c) Assuming that the secondary side water temperature is at the saturation temperature corresponding to the secondary side steam pressure indicated on the Control Room instrumentation.

The preventive measures for the Mass Input initiating event (i.e., Safety Injection pump flow) as well as the Heat Input initiating event have been fully described in the Reference 4) and 5) submittals to the NRC. (Also refer to specification 3.3.A, Safety Injection and Residual Heat Removal Systems).

The OPS need not be operable when the RCS temperature is <290°F if the RCS is depressurized and vented with an equivalent opening of at least 2.00 square inches. This opening is adequate to relieve the worst case analyzed incidents and the PORVs need not, therefore, be operable.

An RCS temperature of 290°F is the minimum temperature at which an RCS hydrostatic test can be performed (i.e., 300°F less 10°F for possible instrument errors - see Figure 4.3-1). Therefore, the OPS arming temperature of <290°F permits the performance of an RCS hydrostatic test without activating the OPS. Upon OPS inoperability, the RCS may be heated above 338°F. This RCS temperature is that value for which the RCS heatup and cooldown curves (Figures 3.1-1 and 3.1-2) permit pressurization to the setting of the pressurizer code safety valves. Accordingly, with an inoperable OPS and an RCS temperature >338°F, the pressurizer code safety valves will preclude exceeding the 10 CFR 50, Appendix G, limits.

<u>Reference</u>

1) FSAR Section 14.1.6

2) FSAR Section 9.3.1

3) FSAR Section 14.1.1

4) Attachment 1 to the letter dated October 25, 1976, from Mr. William J. Cahill, Jr. (Con Edison) to Mr. Robert W. Reid (NRC).

5) Attachments 1 and 2 to the letter dated February 28, 1977 from Mr. William J. Cahill, Jr. (Con Edison) to Mr. Robert W. Reid (NRC).

B. HEATUP AND COOLDOWN

Specifications

1. The reactor coolant temperature and pressure and system heatup and cooldown rates averaged over one hour (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 for the service period up to 5 effective full-power years. The heatup or cooldown rate shall not exceed 100°F/hr.

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b. Figure 3.1-1 and Figure 3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. The limit lines shown in Figure 3.1-1 and Figure 3.1-2 shall be recalculated periodically using methods discussed in WCAP-7924A and results of surveillance specimen testing as covered in WCAP-7323.(7) The order of specimen removal may be modified based on the results of testing of previously removed specimens.

3. The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

4. The pressurizer heatup and cooldown rates averaged over one hour shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

5. Reactor Coolant System integrity tests shall be performed in accordance with Section 4.3 of the Technical Specifications.

## Basis

### Fracture Toughness Properties

All components in the Reactor Coolant System are designed to withstand the effects of the cyclic loads due to reactor system temperature and pressure changes.[1] These cyclic loads are introduced by normal unit load transients, reactor trips, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes are shown in Table 4.1-8 of the FSAR. During unit startup and shutdown, the rates of temperature and pressure changes are limited. The maximum plant heatup and cooldown rate of 100°F per hour is consistent with the design number of cycles and satisfies stress limits for cyclic operation.[2]

The reactor vessel plate opposite the core has been purchased to a specified Charpy V-notch test result of 30 ft-lb or greater at a nil-ductility transition temperature (NDTT) of 40°F or less. The material has been tested to verify conformity to specified requirements and a NDTT value of 20°F has been determined. In addition, this plate has been 100 percent volumetrically inspected by ultrasonic test using both longitudinal and shear wave methods. The remaining material in the reactor vessel, and other Reactor Coolant System components, meet the appropriate design code requirements and specific component function.[3]

As a result of fast neutron irradiation in the region of the core, there will be an increase in the Reference Nil-Ductility Transition Temperature ($RT_{NDT}$), with nuclear operation. The techniques used to measure and predict the integrated fast neutron (E >1 Mev) fluxes at the sample location are described in Appendix 4A of the FSAR. The calculation method used to obtain the maximum neutron (E >1 Mev) exposure of the reactor vessel is identical to that described for the irradiation samples.

Since the neutron spectra at the samples and vessel inside radius are identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of reactor vessel for some later stage in plant life. The maximum exposure of the vessel will be obtained from the measured sample exposure by appropriate application of the calculated azimuthal neutron flux variation.

An approximation of the maximum integrated fast neutron (E >1 Mev) exposure is given by Figure 2-4 of WCAP 7924A[4]. Exposure of the Indian Point Unit No. 2 vessel will be less than that indicated by this figure.

The actual shift in $RT_{NDT}$ will be established periodically during plant operation by testing vessel material samples which are irradiated cumulatively by securing them near the inside wall of the vessel in the core area. These samples are evaluated according to ASTM E185.[6] To compensate for any increase in the $RT_{NDT}$ caused by irradiation, the limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown, in accordance with the requirements of the ASME Boiler & Pressure Vessel Code, 1974 Edition, Section III, Appendix G, and the calculation methods described in WCAP-7924A[4].

The first reactor vessel material surveillance capsule was removed during the 1976 refueling outage. This capsule has been tested by Southwest Research Institute (SWRI) and the results have been evaluated and reported.[8] Based on the SWRI evaluation, heatup and cooldown curves (Figures 3.1-1 and 3.1-2) were developed for up to five (5) effective full power years (EFPYs) of reactor operation.

The maximum shift in $RT_{NDT}$ after 5 EFPYs of operation is projected to be 110°F at the 1/4T and 50°F at the 3/4T vessel wall locations, per Plate B2002-3 the controlling plate. The initial value of $RT_{NDT}$ for the IP2 reactor vessel was 60°F based on Plates B2002-1 and B2002-3 as shown in Table 3.1-1. The heatup and cooldown curves for 5 EFPYs have been computed on the basis of the $RT_{NDT}$ of Plate B2002-3 because it is anticipated that the $RT_{NDT}$ of the reactor vessel beltline material will be highest for Plate B2002-3 at least through that time period.

Heatup and Cooldown Curves

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Non Mandatory Appendix G in Section III 1974 Edition of the ASME Boiler and Pressure Vessel Code and discussed in detail in WCAP-7924.[4]

The approach specifies that the allowable total stress intensity factor ($K_I$) at any time during heatup or cooldown cannot be greater than that shown on the

$K_{IR}$ curve[5] for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by pressure gradients. Thus, the governing equation for the heatup-cooldown analysis is:

$$2\,K_{Im} + K_{It} \leq K_{IR} \tag{1}$$

where:

$K_{Im}$ is the stress intensity factor caused by membrane (pressure) stress

$K_{It}$ is the stress intensity factor caused by the thermal gradients

$K_{IR}$ is provided by the code as a function of temperature relative to the $RT_{NDT}$ of the material.

During the heatup analysis, Equation (1) is evaluated for two distinct situations.

First, allowable pressure-temperature relationships are developed for steady state (i.e., zero rate of change of temperature) conditions assuming the presence of the code reference 1/4 T deep flaw at the ID of the pressure vessel. Due to the fact that, during heatup, the thermal gradients in the vessel wall tend to produce compressive stresses at the 1/4 T location, the tensile stresses induced by int rnal pressure are somewhat alleviated. Thus, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the 1/4 T location is treated as the governing factor.

The second portion of the heatup analysis concerns the calculation of pressure temperature limitations for the case in which the 3/4 T location becomes the controlling factor. Unlike the situation at the 1/4 T location, at the 3/4 T position (i.e., the tip of the 1/4 T deep O.D. flaw) the thermal gradients established during heatup produce stresses which are tensile in nature; and, thus, tend to reinforce the pressure stresses present. These thermal stresses are, of course, dependent on both the rate of heatup and the time (or water temperature) along the heatup ramp. Furthermore, since the thermal stresses at 3/4 T are tensile and increase with increasing heatup rate, a

lower bound curve similar to that described in the preceding paragraph cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point by point comparison of the steady state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve becomes mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the O.D. to the I.D. location; and the pressure limit must, at all times, be based on the most conservative case.

The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always at 1/4 T. The thermal gradients induced during cooldown tend to produce tensile stresses at the 1/4 T location and compressive stresses at the 3/4 T position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure temperature relations are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel I.D. This condition is, of course, not true for the steady-state situation. It

follows that the $\Delta T$ induced during cooldown results in a calculated higher allowable $K_{IR}$ for finite cooldown rates than for steady state under certain conditions.

Because operation control is on coolant temperature, and cooldown rate may vary during the cooldown transient, the limit curves shown in Figure 3.1-2 represent a composite curve consisting of the more conservative values calculated for steady state and the specific cooling rate shown.

Details of these calculation are provided in WCAP-7924A[4].

Pressurizer Limits

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition and associated Code Addenda through the Summer 1966 Addendum.

References

(1) Indian Point Unit No. 2 FSAR, Section 4.1.5

(2) ASME Boiler & Pressure Vessel Code, Section III, Summer 1965, N-415.

(3) Indian Point Unit No. 3 FSAR, Section 4.2.5

(4) WCAP-7924A, "Basis for Heatup and Cooldown Limit Curves", W.S. Hazelton, S.L. Anderson, S.E. Yanichko, April 1975.

(5) ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition, Appendix G.

(6) ASTM E185-70, Surveillance Tests on Structural Materials in Nuclear Reactors.

(7) WCAP-7323, "Consolidated Edison Company, Indian Point Unit No. 2 Reactor Vessel Radiation Surveillance Program", S.E. Yanichko, May 1969.

(8) Final Report - SWRI Project 02-4531 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T", E.B. Norris, June 30, 1977.

TABLE 3.1-1

Indian Point Unit No. 2

Reactor Vessel Core Region Material

| Plate | Copper Content(1) | Lowest Temperature 50 ft. lb. Charpy (Longitudinal)(2) | Lowest Temperature 50 ft. lb. Charpy (Transverse)(3) | Assumed $RT_{NDT}$(4) |
|---|---|---|---|---|
| B 2002-1 | 0.25 | 60°F | 120°F | 60°F |
| B 2002-2 | 0.14 | 62°F | 112°F | 52°F |
| B 2002-3 | 0.14 | 75°F | 120°F | 60°F |
| HAZ | -- | -45°F | 5°F | -55°F |
| Weld Material | -- | -10°F | 15°F | -45°F |

(1) Reference: Letter No. IPP-75-50, Westinghouse to Con Edison Dated May 16, 1975

(2) Reference: WCAP-7323, "Con Edison Indian Point Unit No. 2 Reactor Vessel Radiation Surveillance Program", Dated May 1969.

(3) Estimated from Longitudinal Data for 77 ft. lb/54 Mil Lateral Expansion (In All Cases, Expansion Data Exceed Requirements).

(4) Reference: ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition Appendix G, $RT_{NDT} = T_{cv} - 60°F$

$T_{cv}$ = Transfer Charpy Temperature at 50 ft. lb energy

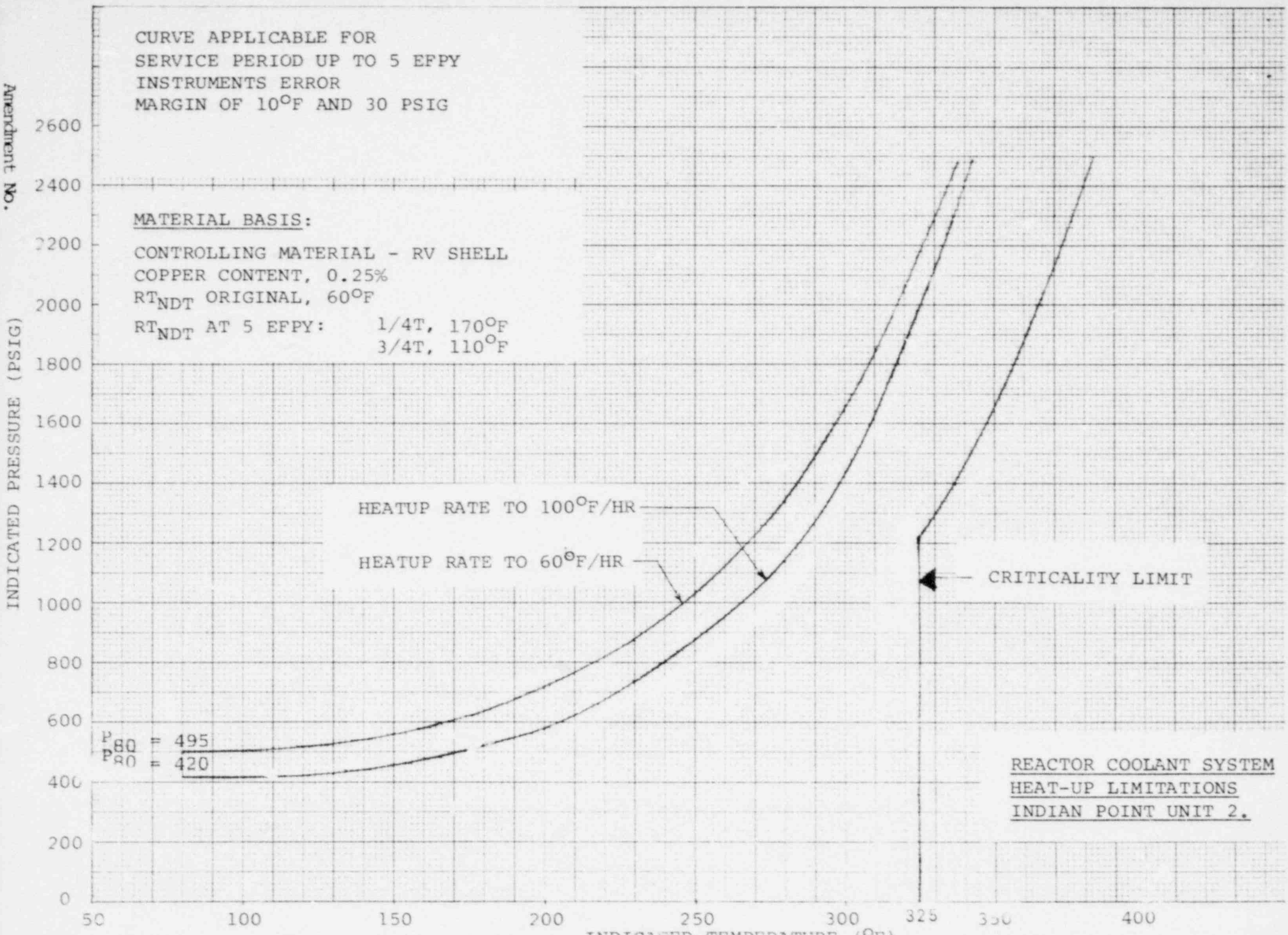


FIGURE 3.1 - 1

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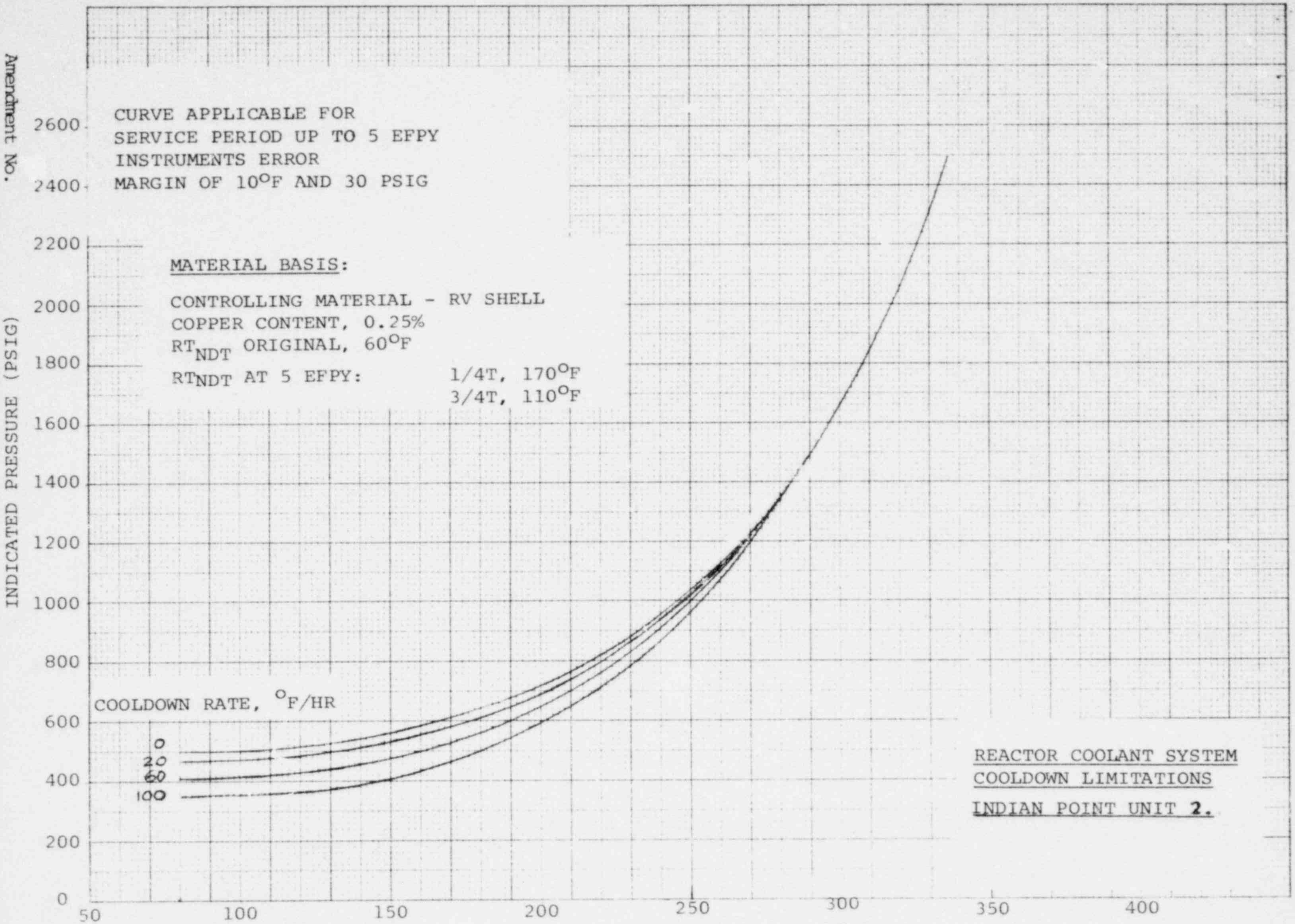


FIGURE 3.1 - 2

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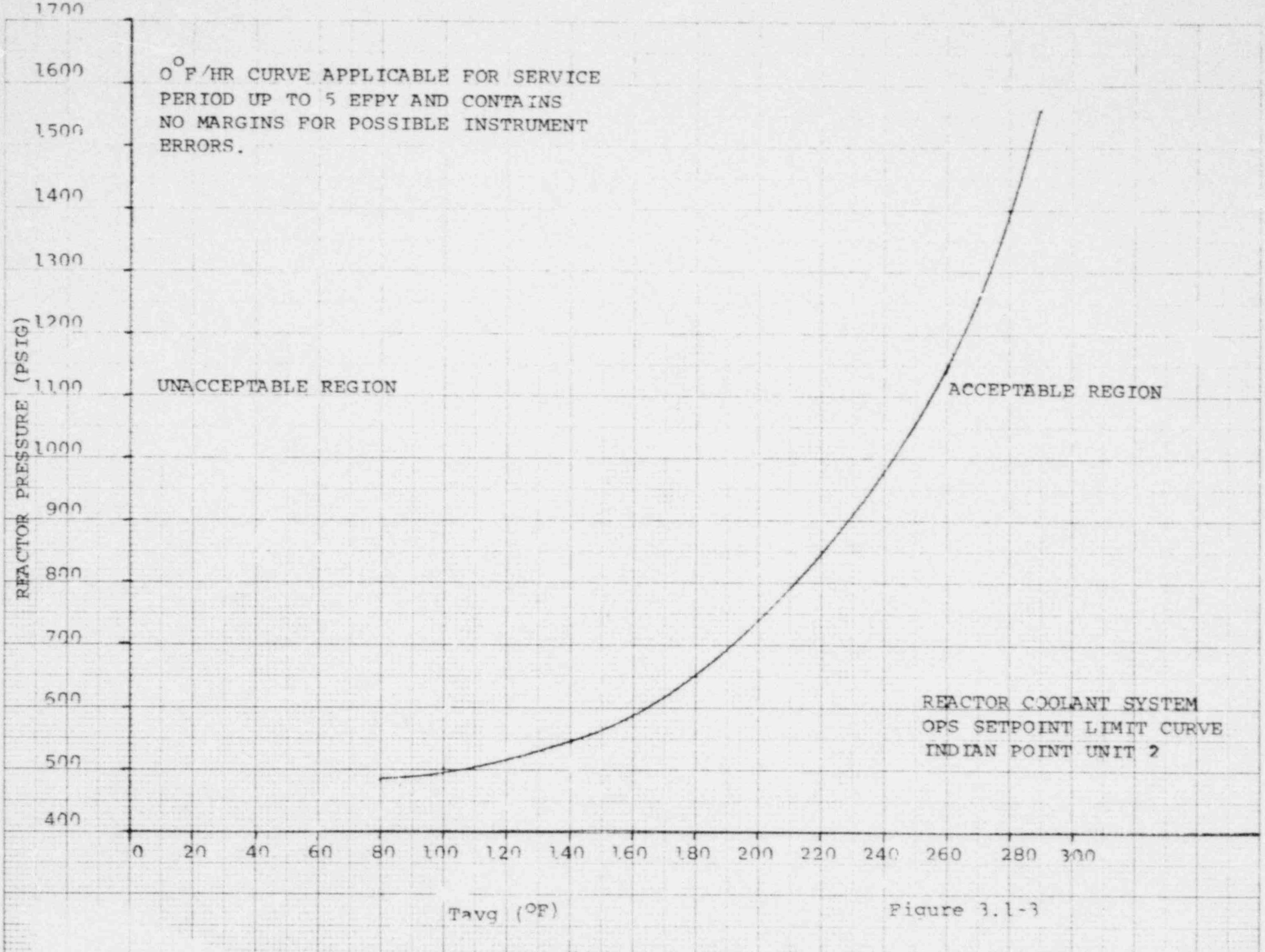


Figure 3.1-3

AMENDMENT NO.

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the Engineered Safety Features.

Objective

To define those limiting conditions for operation that are necessary: (1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment in normal operating and emergency situations, (3) to remove airborne iodine from the containment atmosphere following a Design Basis Accident, (4) to minimize containment leakage to the environment subsequent to a Design Basis Accident, (5) to minimize the potential for and consequences of Reactor Coolant System pressure transients.

Specification

The following specifications apply except during low temperature physics tests.

A. Safety Injection and Residual Heat Removal Systems

1. The reactor shall not be made critical, except for low temperature physics tests, unless the following conditions are met:

   a. The refueling water storage tank contains not less than 345,000 gallons of water with a boron concentration of at least 2000 ppm.

   b. The boron injection tank contains not less than 1000 gallons of a 11 1/2% to 13% by weight (20,000 ppm to 22,500 ppm of boron) boric acid solution at a temperature of at least 145°F. Two channels of heat tracing shall be available for the flow path. Valves 1821 and 1831 shall be open and valves 1822A and 1822B shall be closed, except during short periods of time when they can be cycled to demonstrate their operability.

   c. The four accumulators are pressurized to at least 600 psig and each contains a minimum of 800 $ft^3$ and a maximum of 815 $ft^3$ of water with a boron concentration of at least 2000 ppm. None of these four accumulators may be isolated.

   d. Three safety injection pumps together with their associated piping and valves are operable.

e. Two residual heat removal pumps and heat exchangers together with their associated piping and valves are operable.

f. Two recirculation pumps together with the associated piping and valves are operable.

g. Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the RWST are de-energized in the open position.

h. Valves 856A, C, D and E, in the discharge header of the safety injection header are in the open position. Valves 856B and F, in the discharge header of the safety injection header are in the closed position. The hot leg valves (856B and F) shall be closed with their motor operators de-energized by locking out the circuit breakers at the Motor Control Centers.

i. The four accumulator isolation valves shall be open with their motor operators de-energized by locking out the circuit breakers at the Motor Control Centers.

j. Valve 1810 on the suction line of the high-head SI pumps and valves 882 and 744, respectively on the suction and discharge line of the residual heat removal pumps, shall be blocked open by de-energizing the valve-motor operators.

k. The refueling water storage tank low level alarms are operable and set to alarm between 92,800 gallons and 99,000 gallons of water in the tank.

2. During power operation, the requirements of 3.3.A-1 may be modified to allow any one of the following components to be inoperable at any one time. If the system is not restored to meet the requirements of 3.3.A-1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.A-1 are not satisfied within an additional 48 hours the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

   a. One safety injection pump may be out of service, provided the pump is restored to operable status within 24 hours and the remaining two pumps are demonstrated to be operable.

b. One residual heat removal pump may be out of service, provided the pump is restored to operable status within 24 hours and the other residual heat removal pump is demonstrated to be operable.

c. One residual heat removal exchanger may be out of service provided that it is restored to operable status within 48 hours.

d. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided that it is restored to operable status within 24 hours and all valves in the system that provide the duplicate function are demonstrated to be operable.

e. One channel of heat tracing may be out of service for 48 hours.

f. One refueling water storage tank low level alarm may be inoperable for up to 7 days provided the other low level alarm is operable.

3. No more than one safety injection pump may be energized when the RCS temperature is <290°F, unless:

   a. the RCS is depressurized and vented with an equivalent opening of $\geq$3.00 square inches, or

   b. at least one valve in the flow path from the additional safety injection pumps to the RCS is closed and either locked (if manual) or de-energized (if motor operated).

B. Containment Cooling And Iodine Removal Systems

1. The reactor shall not be made critical unless the following conditions are met:

   a. The spray additive tank contains not less than 4000 gallons of solution with a sodium hydroxide concentration of not less than 30% by weight.

   b. The five fan cooler-charcoal filter units and the two spray pumps, with their associated valves and piping, are operable.

2. During power operation, the requirements of 3.3.B-1 may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the

The OPS has been designed to withstand the effects of the postulated worst case for Mass Input flow (i.e., Single S.I. pump) without exceeding the 10 CFR 50, Appendix G, limits. Figure 3.1-3 shows the maximum setpoint curve of the OPS PORVs which is sufficiently below the 10 CFR 50, Appendix G, limits such that PORV overshoots do not result in peak RCS pressures exceeding the 10 CFR 50, Appendix G, limits. Thus it is acceptable to energize one S.I. pump when the OPS is armed.

More than one S.I. pump may be energized when the RCS temperature is <290°F if at least one valve in the flow path from the additional pumps to the RCS is closed and either locked (if manual) or de-energized (if motor operated), or the RCS is depressurized and vented with an equivalent opening of at least 3.00 square inches. This opening is adequate to relieve the RCS pressure transient resulting from a postulated start of all three (3) S.I. pumps.

References


(1) FSAR Section 9

(2) FSAR Section 6.2

(3) FSAR Section 6.2

(4) FSAR Section 6.3

(5) FSAR Section 14.3.5

(6) FSAR Section 1.2

(7) FSAR Section 8.2

(8) FSAR Section 9.6.1

(9) FSAR Section 14.3

(10) Indian Point Unit No. 2 "Analysis of the Emergency Core Cooling System in Accordance with the Acceptance Criteria of 10CFR50.46 and Appendix K of 10CFR50", January 1977.

(11) Letter from William J. Cahill, Jr. of Consolidated Edison Company of New York, to Robert W. Reid of the Nuclear Regulatory Commission, dated July 13, 1976. Indian Point Unit No. 2 Small Break LOCA Analysis.

(12) Indian Point Unit No. 3 FSAR Sections 6.2 and 6.3 and the Safety Evaluation accompanying "Application for Amendment to Operating License" sworn to by Mr. William J. Cahill, Jr. on March 28, 1977.

TABLE 3-3 (CONTINUED)

| NO. | FUNCTIONAL UNIT | 1<br>NO. OF CHANNELS | 2<br>NO. OF CHANNELS TO TRIP | 3<br>MIN. OPERABLE CHANNELS | 4<br>MIN. DEGREE OF REDUNDANCY | 5<br>OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET |
|---|---|---|---|---|---|---|
| 3. | OVERPRESSURE PROTECTION SYSTEM (OPS) | 3 | 2 | 2 | 1 | **** |

****Refer to Specification 3.1.A.4.

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TABLE 4.1-1 (CONTINUED)

| | Channel Description | Check | Calibrate | Test | Remarks |
|---|---|---|---|---|---|
| 24. | Turbine First Stage Pressure | S | R | M | |
| 25. | Logic Channel Testing | N.A. | N.A. | M | |
| 26. | Turbine Overspeed Protection Trip Channel (Electrical) | N.A. | R | M | |
| 27. | Control Room Ventilation | N.A. | N.A. | R | Check damper operation for accident mode with isolation signal |
| 28. | Overpressure Protection System (OPS) | N.A. | R | * | |

Note: Specified intervals may be adjusted plus or minus 25% to accommodate normal test schedules.

S - Each shift
M - Monthly
Q - Quarterly
S.A. - Semi-annually
D - Daily
P - Prior to each startup if not done previous week
W - Weekly
R - Each Refueling Shutdown, but not to exceed 18 months, except for the first fuel cycle.
N.A. - Not Applicable

* Within 31 days prior to entering a condition in which OPS is required to be operable and at monthly intervals thereafter when OPS is required to be operable.

## 4.3 REACTOR COOLANT SYSTEM INTEGRITY TESTING

### Applicability

Applies to test requirements for Reactor Coolant System integrity.

### Objective

To specify tests for Reactor Coolant System integrity after the system is closed following normal opening, modification or repair.

### Specification

a) When the Reactor Coolant System is closed after it has been opened, the system will be leak tested at not less than 2335 psig at NDT requirements for temperature.

b) When Reactor Coolant System modifications or repairs have been made which involve new strength welds on components, the new welds will meet the requirements of the applicable version of ASME Section XI as specified in the Con Edison Inservice Inspection and Testing Program in effect at the time.

c) The Reactor Coolant System leak test temperature-pressure relationship shall be in accordance with the limits of Figure 4.3-1 for heat-up for the first five (5) effective full-power yrs. of operation. Figure 4.3-1 will be recalculated periodically. Allowable pressures during during cooldown for the leak test temperature shall be in accordance with Figure 3.1-2.

### Basis

For normal opening, the integrity of the system, in terms of strength, is unchanged. If the system does not leak at 2335 psig (Operating pressure +100 psi: ±100 psi is normal system pressure fluctuation), it will be leak tight during normal operation.

For repairs on components, the thorough non-destructive testing gives a very high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. In all cases, the leak test will assure leak tightness during normal operation.

The inservice leak temperatures are shown on Figure 4.3-1. The temperatures are calculated in accordance with ASME Code Section III, 1974 Edition, Appendix G. This Code requires that a safety factor of 1.5 times the stress intensity factor caused by pressure be applied to the calculation.

For the first five (5) effective full-power years, it is predicted that the highest $RT_{NDT}$ in the core region taken at the 1/4 thickness will be 170°F. The minimum inservice leak test temperature requirements for periods up to five (5) effective full-power years are shown on Figure 4.3-1.

The heatup limits specified on the heatup curve, Figure 4.3-1, must not be exceeded while the reactor coolant is being heated to the inservice leak test temperature. For cooldown from the leak test temperature, the limitations of Figure 3.1-2 must not be exceeded. Figures 4.3-1 and 3.1-2 are recalculated periodically, using methods discussed in the WCAP-7924A and results of surveillance specimen testing, as covered in WCAP-7323.

Reference

1. FSAR, Section 4.

Amendment No.

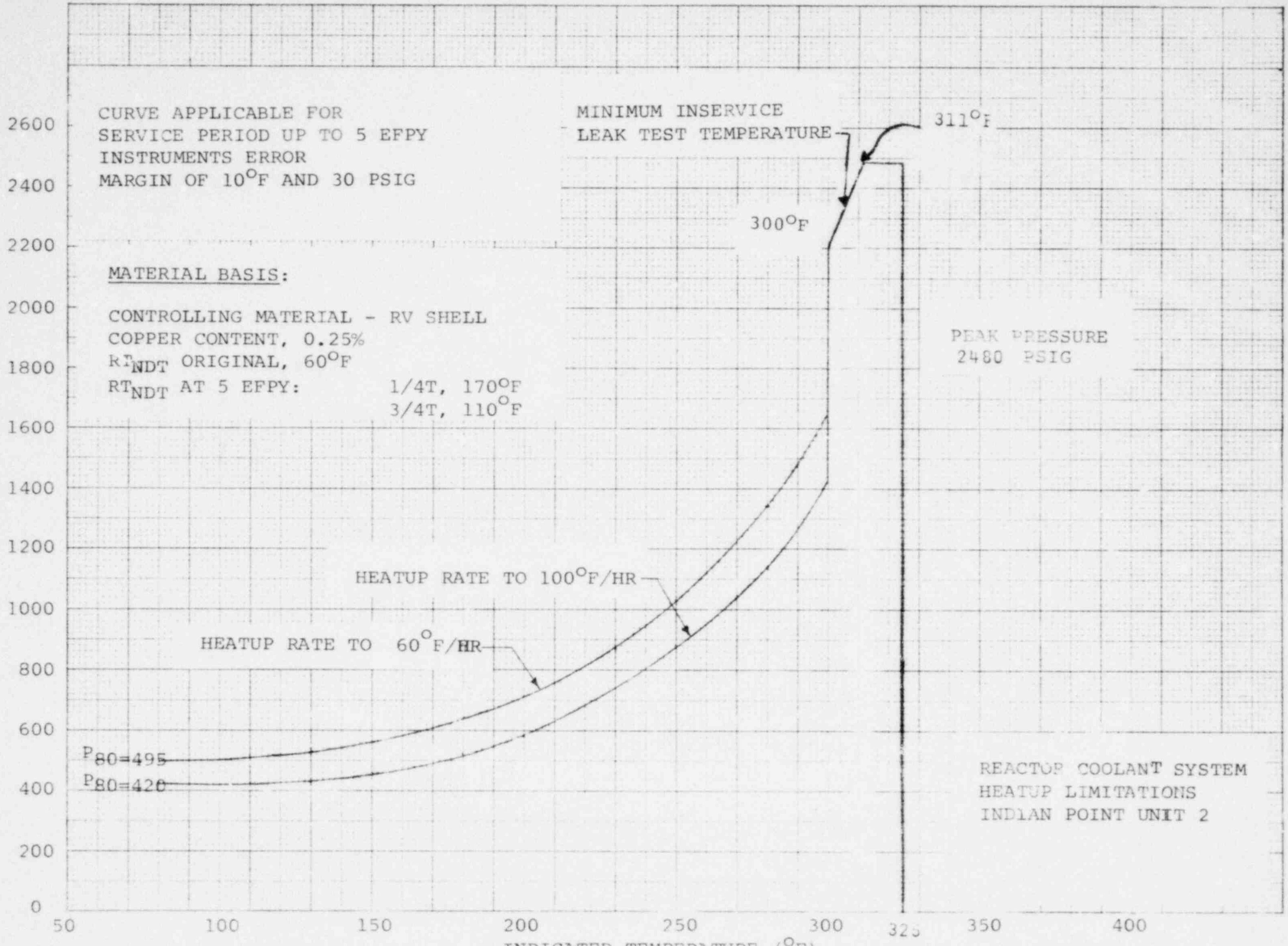



FIGURE 4.3 - 1

## 4.16 Overpressure Protection System (OPS)

### Applicability

This specification applies to the surveillance requirements for the OPS provided for prevention of RCS overpressurization.

### Objective

To verify operability of the OPS.

### Specification

A. When the OPS PORVs are being used for overpressure protection as required by specification 3.1.A.4, their associated series MOVs shall be verified to be open at least twice weekly with a maximum time between checks of 5 days.

B. When RCS venting is being used for overpressure protection as permitted by specifications 3.1.A.4 and 3.3.A.3.a, the RCS vent(s) shall be verified to be open at least daily. When the venting pathway is provided with a valve(s) which is locked, sealed, or otherwise secured in the open position, then only these valves need be verified to be open at monthly intervals.

C. Whenever the RCS temperature is below 290°F:

1. the safety injection pumps required to be inoperable by specification 3.3.A.3 shall be demonstrated inoperable at monthly intervals by verifying lock-out of the pump circuit breakers at the appropriate 480V switchgear, or

2. if specification 3.3.A.3.a is being applied to satisfy specification 3.3.A.3, the requirements of specification 4.16.B above shall be followed, or

3. if specification 3.3.A.3.b is being applied to satisfy specification 3.3.A.3, the appropriate safety injection valve(s) shall be demonstrated inoperable at monthly intervals by verifying that manual valve(s) are locked or that de-energized motor operated valve(s) have their circuit breakers locked-out at the appropriate motor control center(s).

D. The remaining OPS surveillance requirements shall be as established in Table 4.1-1.

Basis

These specifications establish the surveillance program for the RCS Overpressure Protection System (OPS) provided to reduce the potential for and mitigate the consequences of RCS pressure transients. This surveillance program is intended to verify operability of this system and will identify for corrective action any conditions which could prevent any portion of the system from performing its intended function.

The PORVs and MOVs associated with the OPS are not included in this specification since the valve cycling and operability tests for these valves are performed in accordance with the applicable testing requirements of the ASME Code Section XI and 10 CFR 50.55a.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Region I Office of Inspection and Enforcement within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Each containment integrated leak rate test shall be the subject of a summary technical report including results of the local leak rate test since the last report. The report shall include analyses and interpretations of the results which demonstrate compliance in meeting the leak rate limits specified in the Technical Specifications.

b. A report covering the X-Y xenon stability tests within three months upon completion of the tests.

c. To provide the Commission with added verifications of the safety and reliability of the pre-pressurized Zircaloy-clad nuclear fuel, a limited program of non-destructive fuel inspections will be conducted. The program shall consist of a visual inspection (e.g., underwater TV, periscope, or other) of the two lead burnup assemblies in each region during the first, second, and third refueling shutdowns. Any condition observed by this inspection which would lead to unacceptable fuel performance may be the object of an expanded surveillance effort. If another domestic plant which contains pre-pressurized fuel of a similar design reaches fuel exposures equal to or greater than at Indian Point Unit No. 2, and if a limited inspection program is or has been performed there, then the program may not have to be performed at Indian Point Unit No. 2. However, such action requires approval of the Nuclear Regulatory Commission. The results of these inspections will be reported to the Nuclear Regulatory Commission.

d. Inoperable fire protection and detection equipment (Specification 3.13).

e. Sealed source leakage in excess of limits (Specification 4.15).

f. Operation of Overpressure Protection System (Specification 3.1.A.4.d).

ATTACHMENT B

Amendment No. 1 to

Application for Amendment

to Operating License

Safety Evaluation

Consolidated Edison Company of New York, Inc.

Indian Point Unit No. 2

Docket No. 50-247

January, 1979

## Safety Evaluation

On November 3, 1978, Consolidated Edison filed with the NRC an "Application for Amendment to Operating License", sworn to by Mr. William J. Cahill, Jr. on November 3, 1978. The proposed technical specifications contained therein would establish limiting conditions for operation (LCOs) and surveillance requirements for the recently installed Indian Point Unit No. 2 Overpressure Protection System (OPS). The November 3, 1978 Application was submitted in response to Mr. A. Schwencer's August 28, 1978 letter.

As described in our November 3, 1978 forwarding letter, we have been performing detailed analyses to establish a minimum pressurizer bubble size to be utilized during the starting of the first reactor coolant pump (RCP). We believe that the use of the pressurizer bubble during initi RCP startup significantly reduces the effects of reactor coolant system (RCS) pressure transients and the potential for exceeding the 10 CFR 50, Appendix G, limits. Our analyses have been completed and the results have been incorporated into the proposed revisions to technical specification section 3.1.A.1.e by this Amendment No. 1 to Application for Amendment to Operating License.

With regard to the forementioned analyses, they were performed utilizing a plant specific Consolidated Edison Evaluation Model which we have developed for evaluating the effects of various RCS pressure transients on Indian Point Unit No. 2. The computer codes developed as part of the evaluation model can evaluate both the mass input and heat input initiated RCS pressure transient events. Furthermore, we have "bench marked" the Consolidated Edison Evaluation Model against the Westinghouse generic bounding evaluation model with excellent correlation. The attached Figures M17, M19 and M26 and Tables I and II provide comparisons between the two models for certain generic mass input initiated transients while the attached Figures Q-1, H29, H32, and H33 and Table III provide model comparisons for certain generic heat input initiated transients. It is evident from a review of the above data that results obtained using the Consolidated Edison Evaluation Model are generally conservative with respect to results obtained using the Westinghouse generic model. Some of the attached figures show the Consolidated Edison Model curves slightly "out-of-phase" with respect to the Westinghouse model curves. The reason for this is that Consolidated Edison has assumed conservatively

faster pump starting times than Westinghouse in the evaluated transients. This, in effect, causes the peak pressure to be reached slightly sooner in the Consolidated Edison model evaluations, while the trends of the events and the maximum pressures attained are essentially the same for both models.

Therefore, we believe the Consolidated Edison Evaluation Model to be conservative and acceptable for use in evaluating RCS pressure transients for Indian Point Unit No. 2. Although this model is valid for evaluating all plant specific mass input and heat input initiated events, we have utilized the model at this time only in regard to establishing the relationship for the minimum pressurizer bubble for initial RCP startup. The remainder of the LCOs and surveillance requirements for the OPS, the safety injection pumps and for calibration of the PORVs are based on the generic evaluations performed with the Westinghouse generic bounding model.

Also contained in this Amendment are proposed changes to technical specification sections 3.1.B and 4.3 and figures 3.1-1 and 3.1-2 and 4.3-1 to incorporate the reactor coolant system heatup, cooldown and hydrostatic test limitations applicable through five (5) effective full power years (EFPYs) of reactor operation. The basis for these changes is the Southwest Research Institute report entitled, "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T", dated June 30, 1977. Since changes to the RCS heatup and cooldown limitations directly impact on the calibration and operability requirements of the OPS, the necessary revisions to proposed technical specification sections 3.1.A.1.e, 3.3.A.3, 4.16 and Figure 3.1-3 have also been included in this Amendment.

Finally, since a number of changes and additions have been made to the technical specification changes proposed in the November 3, 1978 Application, the specific page revisions contained in Attachment A of that Application are superseded in full by the page revisions contained in Attachment A of this Amendment.

The proposed changes have been reviewed by both the Station Nuclear Safety Committee and the Consolidated Edison Nuclear Facilities Safety Committee. Both Committees concur that the proposed changes do not represent a significant hazards consideration and will not cause any change in the types or an increase in the amounts of effluents or any change in the authorized power level of the facility.

Table I

## SUMMARY TABLE - MASS INPUT RESULTS

RCS VOLUME = 6000 CU.FT.

| INITIAL RCS PRESSURE (psig) | RELIEF VALVE | | | | | MASS INPUT MECHANISM | | | RESULTS | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | $P_{MAX} - P_{SETPOINT}$ (Psi) | |
| | Setpoint (psig) | Number of Valves | Linear (L) or Non-Linear (NL) | Max. Opening Time (sec) | Valve Opens (O)/ Closes (C) | SI Pump SU (SI) or Charging/Letdown Isolation (C/LI) | CC Pump or PD Pump | Letdown Isolation Δt, sec | RCS $P_{MAX}$ (psig) WESTINGHOUSE | WESTINGHOUSE | CON EDISON |
| 50 | 600 | 1 | L | 3.0 | 0 | SI | --- | --- | 755 | 155 | 157.4 |
| 50 | 600 | 1 | L | 3.0 | 0/C | SI | --- | --- | 755 | 155 | |
| 50 | 600 | 2 | L | 3.0 | 0 | SI | --- | --- | 720 | 120 | 118.6 |
| 50 | 600 | 2 | L | 3.0 | 0/C | SI | --- | --- | 720 | 120 | |
| 50 | 600 | 1 | NL | 3.0 | 0/C | SI | --- | --- | 741 | 141 | |
| 50 | 600 | 2 | NL | 3.0 | 0/C | SI | --- | --- | 720 | 120 | |
| 50 | 600 | 1 | L | 1.5* | 0 | SI | --- | --- | 662 | 62 | 60.6 |
| 50 | 600 | 2 | L | 1.5 | 0 | SI | --- | --- | 635 | 35 | 31.5 |
| 450 | 600 | 1 | L | 3.0 | 0 | SI | --- | --- | 751 | 151 | 152.4 |
| 450 | 600 | 1 | L | 3.0 | 0/C | SI | --- | --- | 751 | 151 | |
| 450 | 600 | 2 | L | 3.0 | 0 | SI | --- | --- | 717 | 117 | 118.6 |
| 450 | 500 | 1 | L | 3.0 | 0 | SI | --- | --- | 667 | 167 | 175.3 |
| 450 | 500 | 2 | L | 3.0 | 0 | SI | --- | --- | 626 | 126 | 130.4 |

*All 1.5 sec. cases have 0.0 sec. time delay for valve to begin opening

Table I (Cont'd)

SUMMARY TABLE - MASS INPUT RESULTS

RCS VOLUME = 6000 CU.FT.

| INITIAL RCS PRESSURE (psig) | RELIEF VALVE | | | | | MASS INPUT MECHANISM | | | RESULTS | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | $P_{MAX}-P_{SETPOINT}$ (psi) | |
| | Setpoint (psig) | Number of Valves | Linear (L) or Non-Linear (NL) | Max. Opening Time (sec) | Valve Opens (O)/ Closes (C) | SI Pump SU (SI) or Charging/Letdown Isolation (C/LI) | CC Pump or PD Pump | Letdown Isolation Δt, sec | RCS $P_{MAX}$ (psig) WESTINGHOUSE | WESTINGHOUSE | CON EDISON |
| 50 | 400 | 1 | L | 3.0 | O | SI | --- | --- | 592 | 192 | 195.4 |
| 50 | 400 | 2 | L | 3.0 | O | SI | --- | --- | 544 | 144 | 143.8 |
| 50 | 400 | 1 | NL | 3.0 | O/C | SI | --- | --- | 566 | 166 | |
| 50 | 400 | 2 | NL | 3.0 | O/C | SI | --- | --- | 543 | 143 | |
| 50 | 400 | 1 | L | 1.5 | O | SI | --- | --- | 485 | 85 | 85.8 |
| 50 | 400 | 2 | L | 1.5 | O | SI | --- | --- | 449 | 49 | 45.7 |
| 50 | 600 | 1 | L | 3.0 | O/C | C/LI | CCP | 2 | 610 | 10 | |
| 50 | 600 | 2 | L | 3.0 | O/C | C/LI | CCP | 2 | 610 | 10 | |
| 50 | 600 | 1 | L | 3.0 | O | C/LI | CCP | 10 | 610 | 10 | |
| 50 | 600 | 1 | L | 3.0 | O/C | C/LI | PDP | 2 | 605 | 5 | |
| 450 | 600 | 1 | L | 3.0 | O/C | C/LI | CCP | 2 | 610 | 10 | |
| 50 | 500 | 1 | L | 3.0 | O/C | C/LI | PDP | 2 | 505 | 5 | |
| 50 | 400 | 1 | L | 3.0 | O/C | C/LI | CCP | 2 | 405 | 5 | |
| 50 | 400 | 1 | L | 3.0 | O | C/LI | CCP | 10 | 410 | 10 | |

Table II

SUMMARY TABLE - MASS INPUT RESULTS

RCS VOLUME = 13,000 CU.FT.

| INITIAL RCS PRESSURE (psig) | RELIEF VALVE | | | | | MASS INPUT MECHANISM | | | RESULTS | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | $P_{MAX} - P_{SETPOINT}$ (psi) | |
| | Setpoint (psig) | Number of Valves | Linear (L) or Non-Linear (NL) | Max. Opening Time (sec) | Valve Opens (O)/ Closes (C) | SI Pump SU (SI) or Charging/Letdown Isolation (C/LI) | CC Pump or PD Pump | Letdown Isolation Δt, sec | RCS $P_{MAX}$ (psig) WESTINGHOUSE | WESTINGHOUSE | CON EDISON |
| 50 | 600 | 1 | L | 3.0 | 0 | SI | --- | --- | 675 | 75 | 78.1 |
| 50 | 600 | 2 | L | 3.0 | 0 | SI | --- | --- | 657 | 57 | 57.3 |
| 50 | 600 | 1 | NL | 3.0 | 0 | SI | --- | --- | 667 | 67 | |
| 50 | 600 | 2 | NL | 3.0 | 0 | SI | --- | --- | 658 | 58 | |
| 50 | 600 | 1 | L | 1.5 | 0 | SI | --- | --- | 628 | 28 | 29.2 |
| 50 | 600 | 2 | L | 1.5 | 0 | SI | --- | --- | 616 | 16 | 14.9 |
| 450 | 600 | 1 | L | 3.0 | 0 | SI | --- | --- | 573 | 73 | 78.0 |
| 450 | 600 | 2 | L | 3.0 | 0 | SI | --- | --- | 656 | 56 | 57.3 |
| 450 | 500 | 1 | L | 3.0 | 0 | SI | --- | --- | 583 | 83 | 88.1 |
| 450 | 500 | 2 | L | 3.0 | 0 | SI | -- | --- | 562 | 62 | 63.5 |
| 50 | 400 | 1 | L | 3.0 | υ | SI | --- | --- | 495 | 95 | 99.7 |
| 50 | 400 | 2 | L | 3.0 | 0 | SI | --- | --- | 470 | 70 | 70.7 |
| 50 | 400 | 1 | NL | 3.0 | 0 | SI | --- | --- | 480 | 80 | |

Table II (Cont'd)

SUMMARY TABLE - MASS INPUT RESULTS

RCS VOLUME = 13,000 CU.FT.

| INITIAL RCS PRESSURE (psig) | RELIEF VALVE | | | | | MASS INPUT MECHANISM | | | RESULTS | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Setpoint (psig) | Number of Valves | Linear (L) or Non-Linear (NL) | Max. Opening Time (sec) | Valve Opens (O)/ Closes (C) | SI Pump SU (SI) or Charging/Letdown Isolation (C/LI) | CC Pump or PD Pump | Letdown Isolation Δt, sec | RCS $P_{MAX}$ (psig) WESTINGHOUSE | $P_{MAX}-P_{SETPOINT}$ (psi) WESTINGHOUSE | $P_{MAX}-P_{SETPOINT}$ (psi) CON EDISON | |
| 50 | 400 | 2 | NL | 3.0 | 0 | SI | --- | --- | 469 | 69 | | |
| 50 | 400 | 1 | L | 1.5 | 0 | SI | --- | --- | 440 | 40 | 42.4 | |
| 50 | 400 | 2 | L | 1.5 | 0 | SI | --- | --- | 422 | 22 | 21.9 | |
| 50 | 600 | 1 | L | 3.0 | 0 | CL/I | CCP | 2 | 605 | 5 | | |
| 50 | 600 | 1 | L | 3.0 | 0/C | CL/I | CCP | 2 | 605 | 5 | | |
| 50 | 600 | 1 | L | 3.0 | 0 | CL/I | CCP | 10 | 605 | 5 | | |
| 50 | 400 | 1 | L | 3.0 | 0 | CL/I | CCP | 10 | 605 | 5 | | |
| 50 | 400 | 1 | L | 3.0 | 0 | CL/I | CCP | 2 | 605 | 5 | | |
| 50 | 400 | 1 | L | 3.0 | 0/C | SI | --- | --- | 495 | 95 | | |

Table III

**SUMMARY TABLE - HEAT INPUT RESULTS**

RCS VOLUME = 13,000 CU.FT.

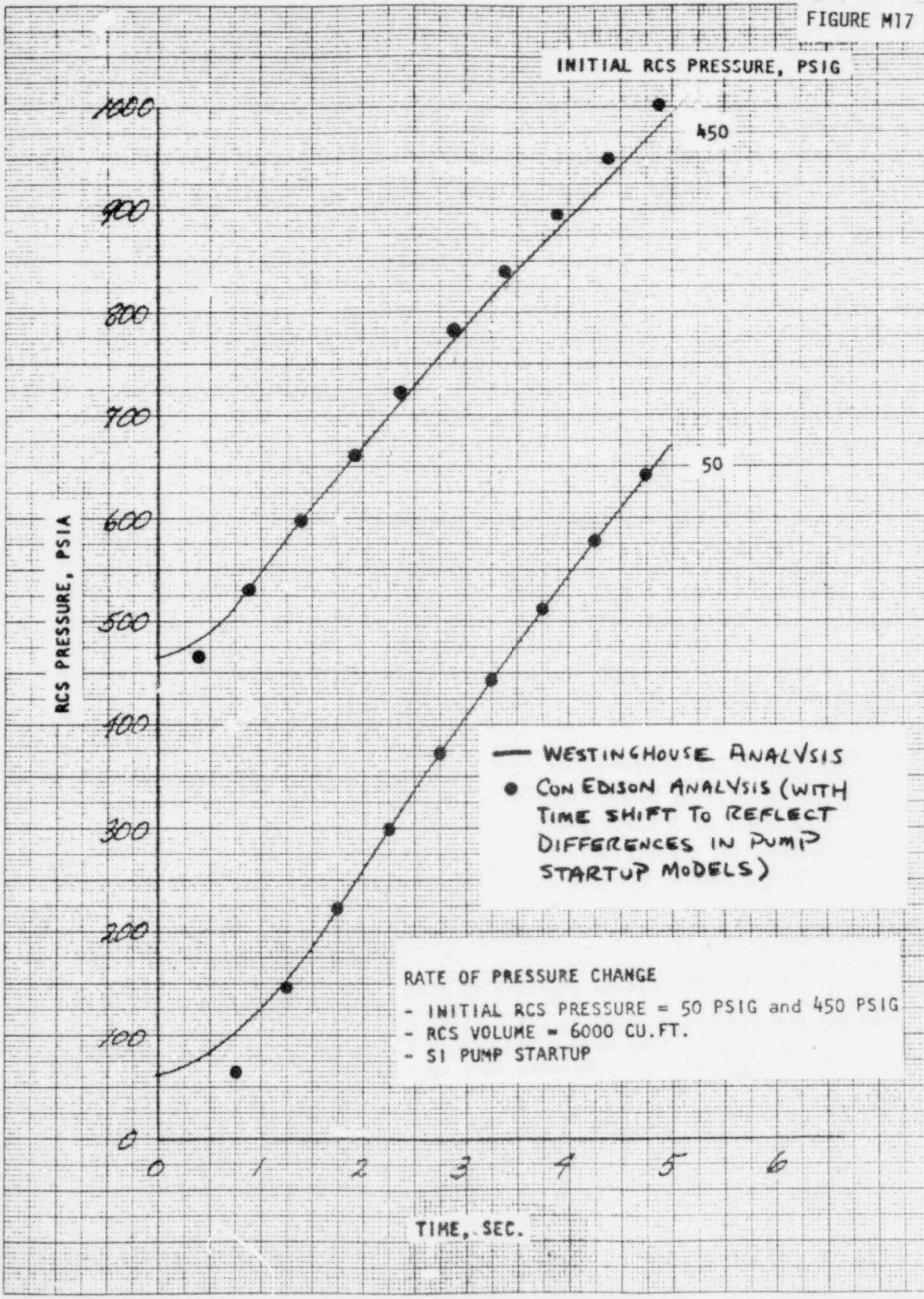
| INITIAL SYSTEM TEMPERATURES (°F) | | | REFERENCE RELIEF VALVE | | SG MODEL | RESULTS | | | |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | $P_{MAX} - P_{SETPOINT}$ (PSi) | | |
| ΔT | RCS | SG | Setpoint (psig) | Number | Conservative (C) or Less Conservative (LC) | RCS $P_{MAX}$ (psig) WESTINGHOUSE | WESTINGHOUSE | CON EDISON | |
| 50 | 100 | 150 | 500 | 1 | C | 527 | 27 | 30.5 | |
| 50 | 140 | 190 | 500 | 1 | C | 550 | 50 | 50.8 | |
| 50 | 180 | 230 | 500 | 1 | C | 569 | 69 | 73.3 | |
| 100 | 100 | 200 | 600 | 1 | C | 710 | 110 | 101.9 | |
| 100 | 100 | 200 | 600 | 2 | C | 680 | 80 | 67.1 | |
| 100 | 100 | 200 | 600 | 1 | LC | 650 | 50 | | |
| 100 | 140 | 240 | 600 | 1 | C | 775 | 175 | 180.7 | |
| 100 | 140 | 240 | 600 | 2 | C | 725 | 125 | 103.0 | |
| 100 | 180 | 280 | 600 | 1 | C | 908 | 308 | 910.5 | |
| 100 | 180 | 280 | 600 | 2 | C | 765 | 165 | 143.3 | |
| 100 | 180 | 280 | 600 | 1 | LC | 725 | 125 | | |
| 100 | 100 | 200 | 500 | 1 | C | 608 | 108 | 93.8 | |
| 100 | 100 | 200 | 500 | 2 | C | 575 | 75 | 61.0 | |

Table III (Cont'd)

## SUMMARY TABLE - HEAT INPUT RESULTS

RCS VOLUME = 13,000 CU.FT.

| INITIAL SYSTEM TEMPERATURES (°F) | | | REFERENCE RELIEF VALVE | | SG MODEL | RESULTS | | | |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | $P_{MAX} - P_{SETPOINT}$ (psi) | | |
| ΔT | RCS | SG | Setpoint (psig) | Number | Conservative (C) or Less Conservative (LC) | RCS $P_{MAX}$ (psig) WESTINGHOUSE | WESTINGHOUSE | CON EDISON | |
| 100 | 140 | 240 | 500 | 1 | C | 667 | 167 | 205.6 | |
| 100 | 140 | 240 | 500 | 2 | C | 615 | 115 | 92.9 | |
| 100 | 180 | 280 | 500 | 1 | C | 855 | 355 | 462.5 | |
| 100 | 180 | 280 | 500 | 2 | C | 650 | 150 | 129.7 | |
| 100 | 100 | 200 | 400 | 1 | C | 495 | 95 | 78.6 | |
| 100 | 100 | 200 | 400 | 2 | C | 465 | 65 | 50.2 | |
| 100 | 100 | 200 | 400 | 1 | LC | 435 | 35 | | |
| 100 | 140 | 240 | 400 | 1 | C | 577 | 177 | 243.8 | |
| 100 | 140 | 240 | 400 | 2 | C | 490 | 90 | 77.0 | |
| 100 | 180 | 280 | 400 | 1 | C | 793 | 393 | 518.5 | |
| 100 | 180 | 280 | 400 | 2 | C | 500 | 100 | 107.5 | |
| 100 | 180 | 280 | 400 | 1 | LC | 505 | 105 | | |

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

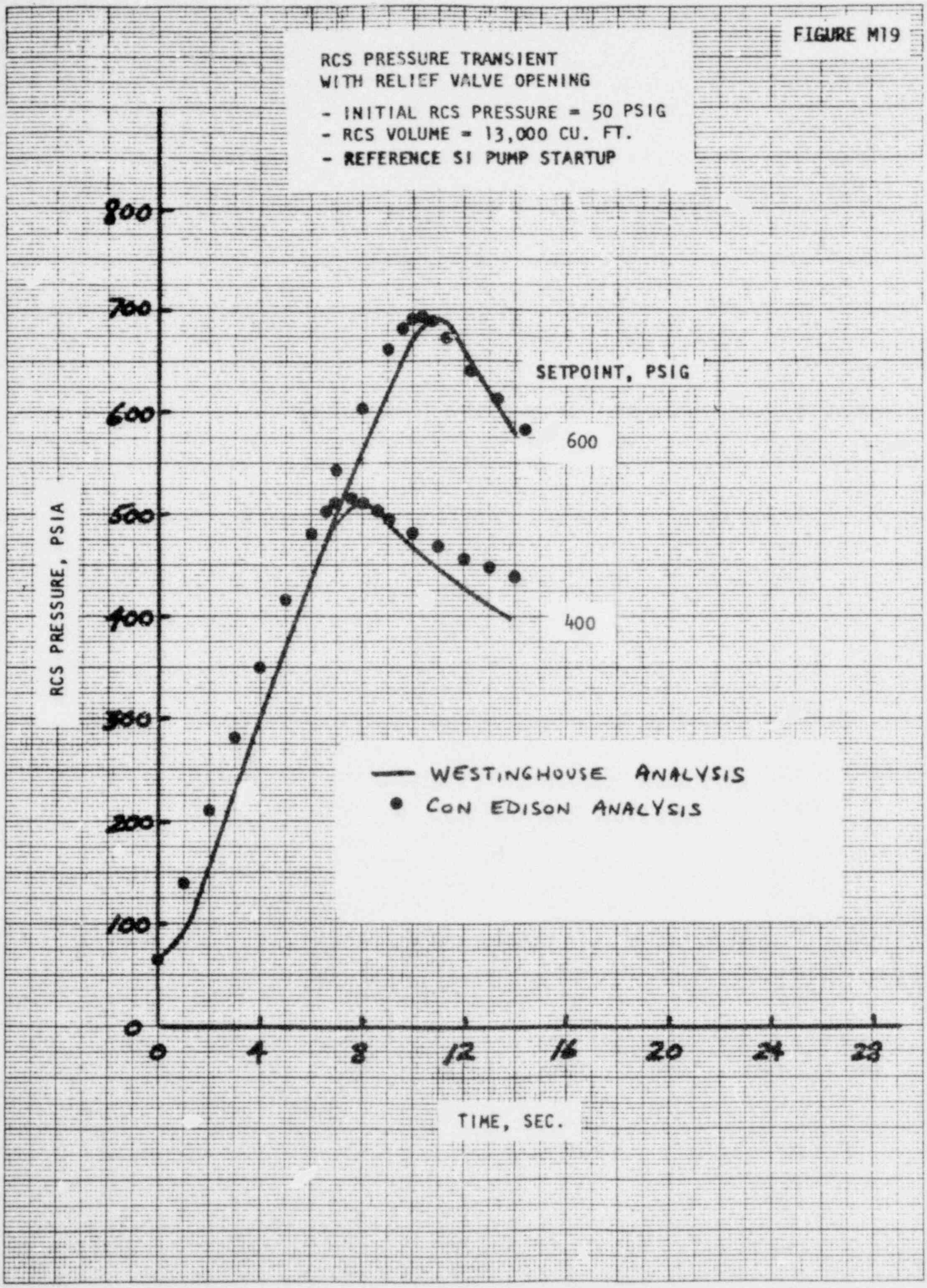
FIGURE M19
RCS PRESSURE TRANSIENT
WITH RELIEF VALVE OPENING
- INITIAL RCS PRESSURE = 50 PSIG
- RCS VOLUME = 13,000 CU. FT.
- REFERENCE SI PUMP STARTUP
RCS PRESSURE, PSIA
0
100
200
300
400
500
600
700
800
SETPOINT, PSIG
600
400
WESTINGHOUSE ANALYSIS
CON EDISON ANALYSIS
0
4
8
12
16
20
24
28
TIME, SEC.

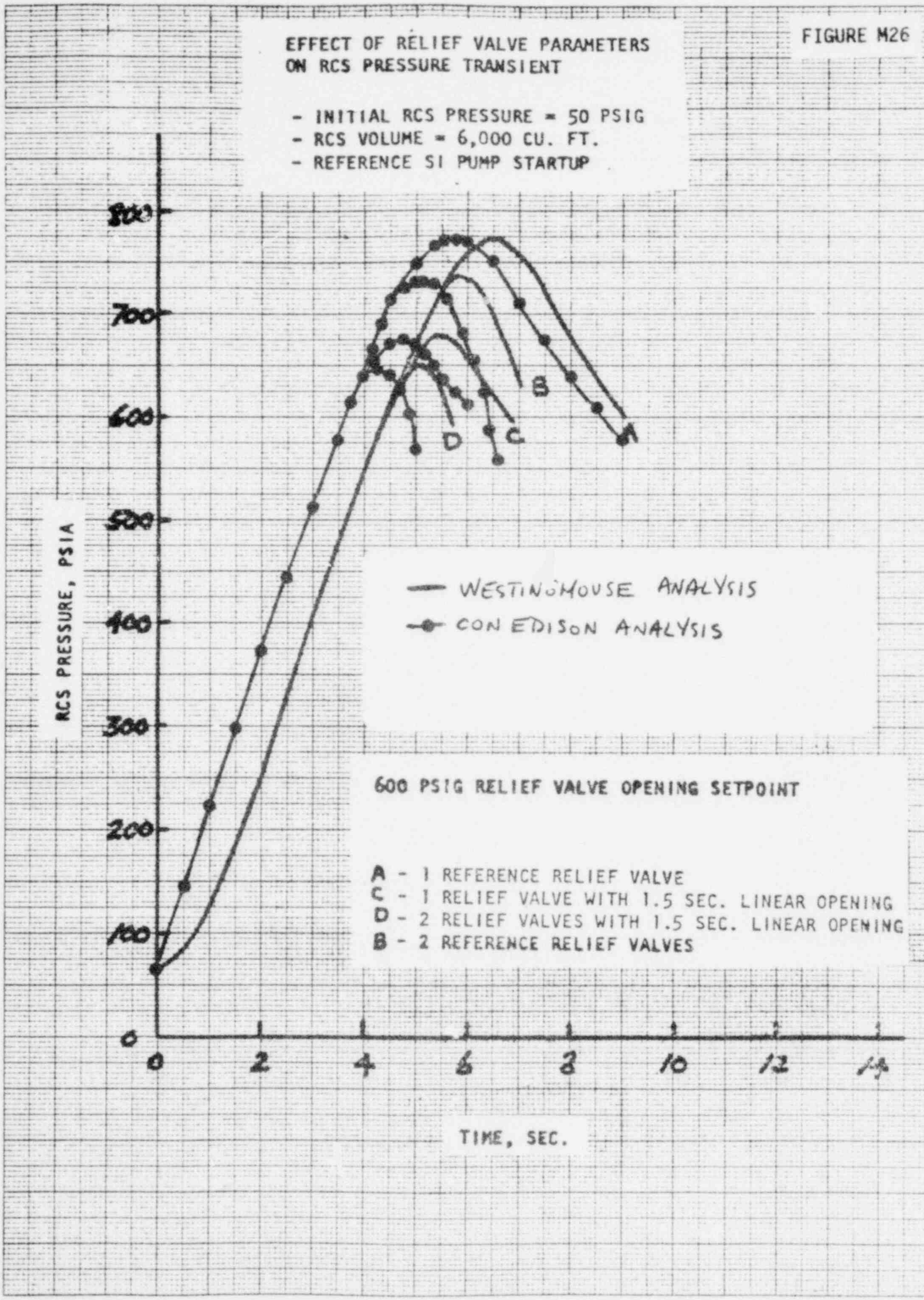
FIGURE M26
EFFECT OF RELIEF VALVE PARAMETERS
ON RCS PRESSURE TRANSIENT
- INITIAL RCS PRESSURE = 50 PSIG
- RCS VOLUME = 6,000 CU. FT.
- REFERENCE SI PUMP STARTUP
RCS PRESSURE, PSIA
800
700
600
500
400
300
200
100
0
0
2
4
6
8
10
12
14
TIME, SEC.
A
B
C
D
WESTINGHOUSE ANALYSIS
CON EDISON ANALYSIS
600 PSIG RELIEF VALVE OPENING SETPOINT
A - 1 REFERENCE RELIEF VALVE
C - 1 RELIEF VALVE WITH 1.5 SEC. LINEAR OPENING
D - 2 RELIEF VALVES WITH 1.5 SEC. LINEAR OPENING
B - 2 REFERENCE RELIEF VALVES

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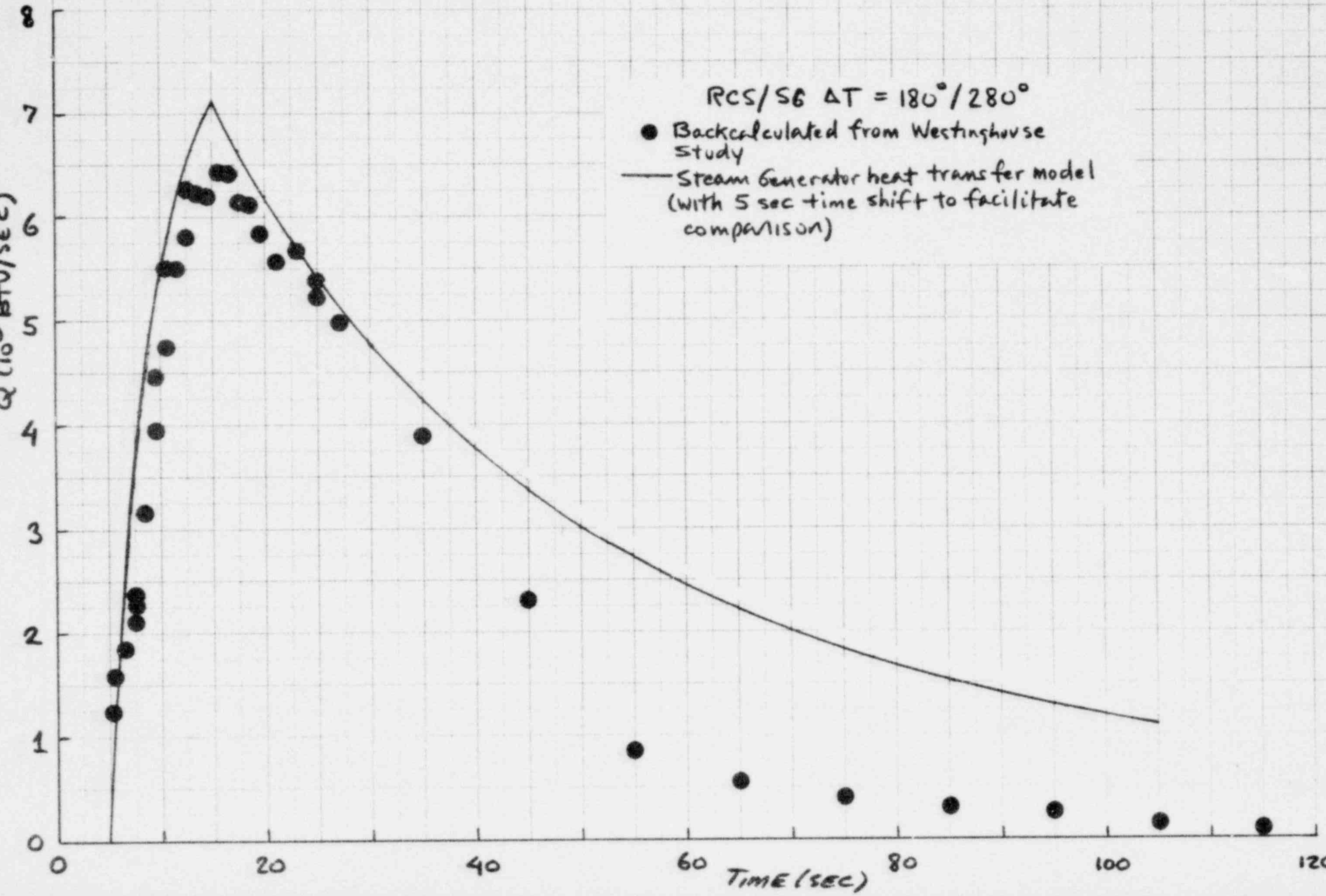
FIGURE Q-1



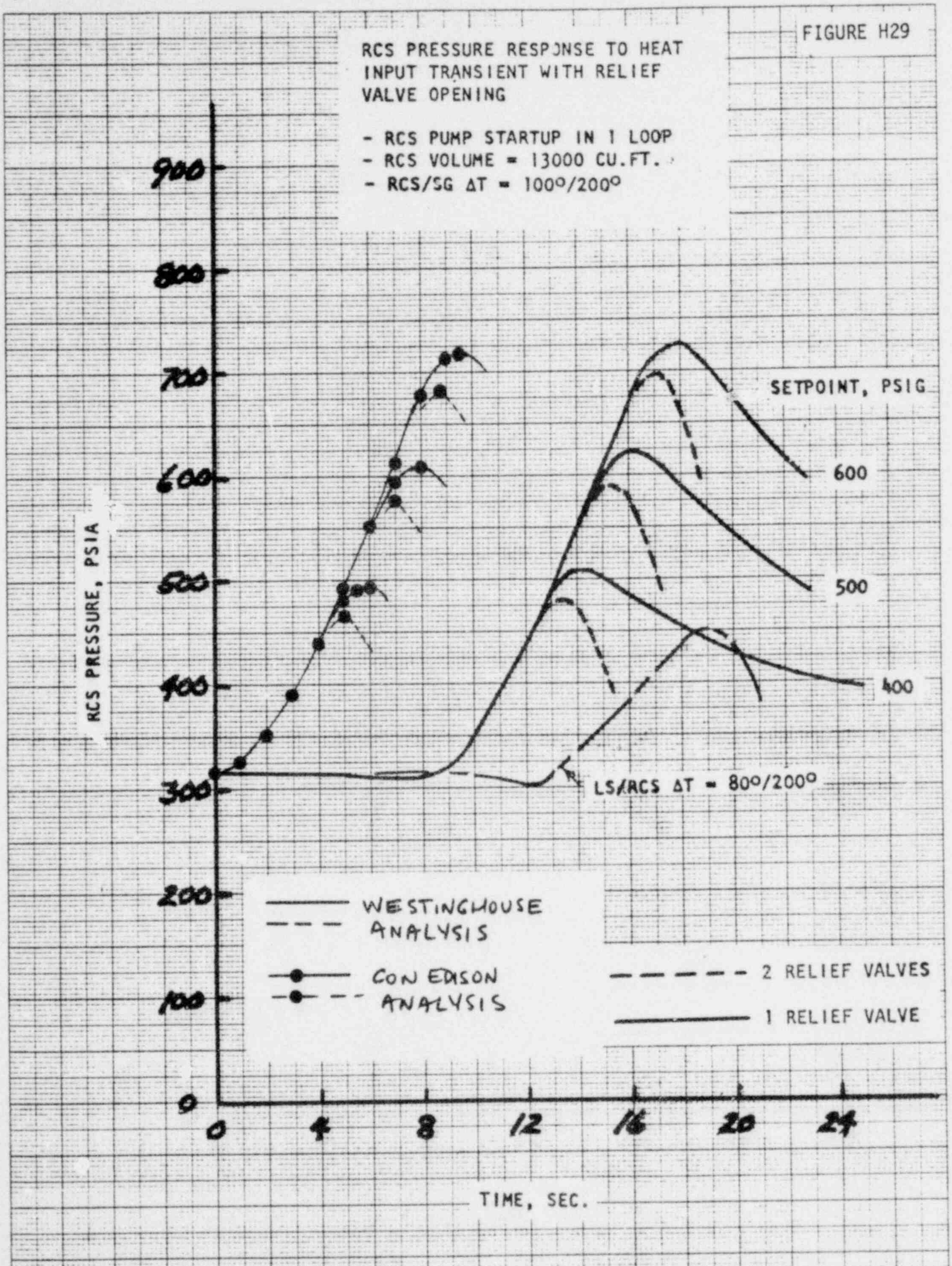


FIGURE H29
RCS PRESSURE RESPONSE TO HEAT INPUT TRANSIENT WITH RELIEF VALVE OPENING
- RCS PUMP STARTUP IN 1 LOOP
- RCS VOLUME = 13000 CU.FT.
- RCS/SG ΔT = 100°/200°
SETPOINT, PSIG
600
500
400
LS/RCS ΔT = 80°/200°
RCS PRESSURE, PSIA
900
800
700
600
500
400
300
200
100
0
0
4
8
12
16
20
24
TIME, SEC.
WESTINGHOUSE ANALYSIS
CON EDISON ANALYSIS
2 RELIEF VALVES
1 RELIEF VALVE

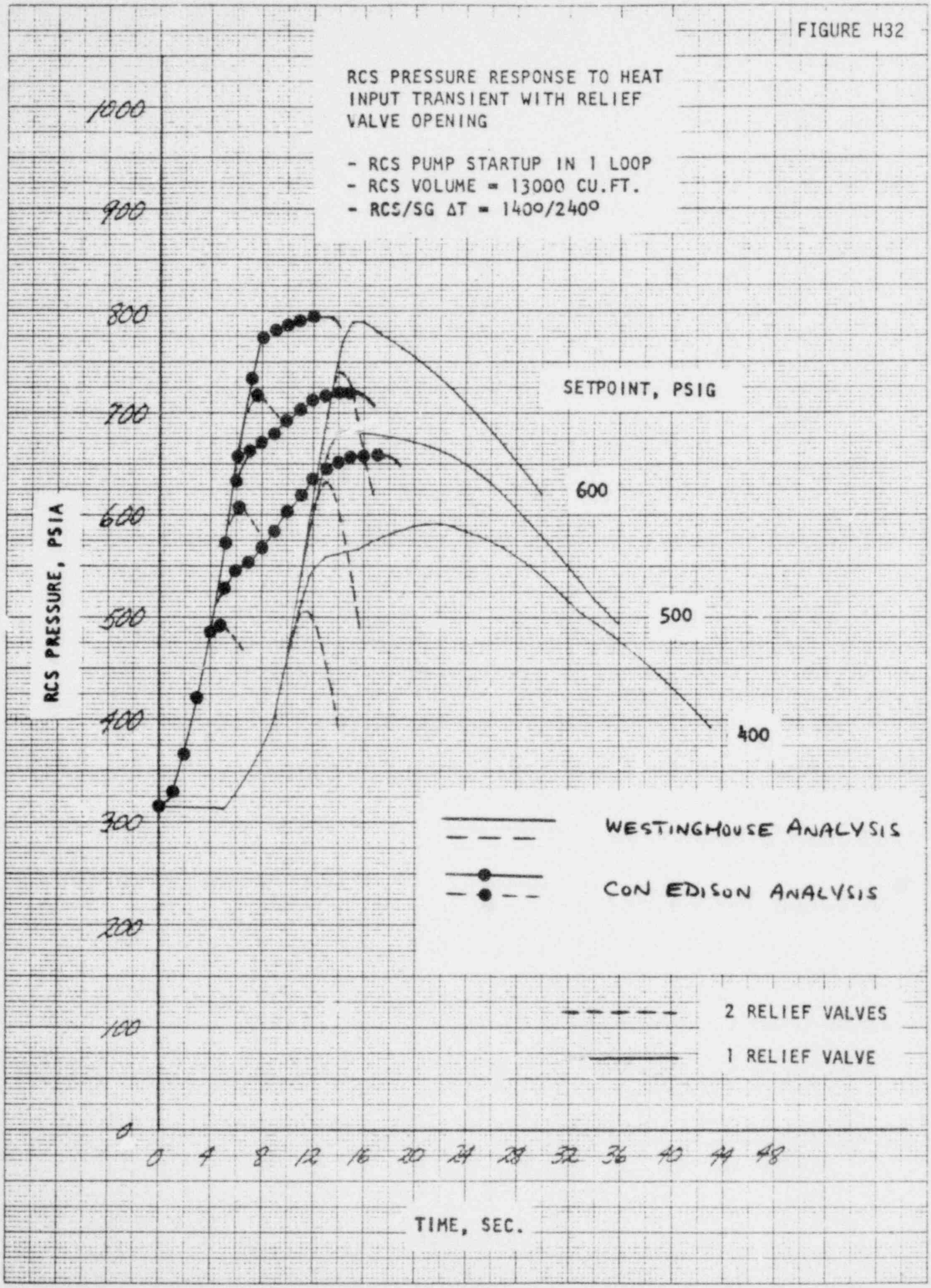
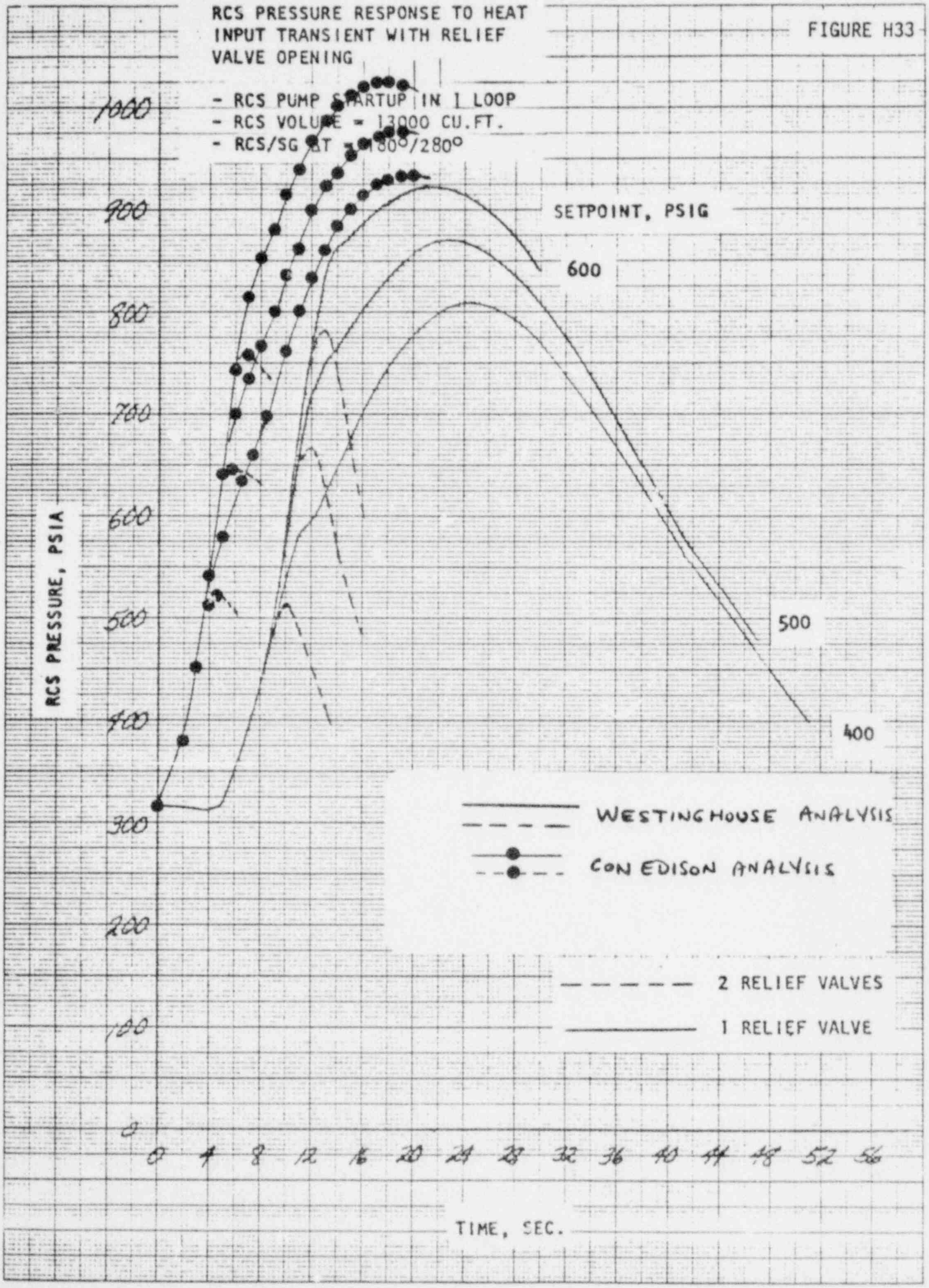


FIGURE H32
RCS PRESSURE RESPONSE TO HEAT INPUT TRANSIENT WITH RELIEF VALVE OPENING
- RCS PUMP STARTUP IN 1 LOOP
- RCS VOLUME = 13000 CU.FT.
- RCS/SG ΔT = 140°/240°
SETPOINT, PSIG
600
500
400
RCS PRESSURE, PSIA
1000
900
800
700
600
500
400
300
200
100
0
0 4 8 12 16 20 24 28 32 36 40 44 48
TIME, SEC.
WESTINGHOUSE ANALYSIS
CON EDISON ANALYSIS
2 RELIEF VALVES
1 RELIEF VALVE



RCS PRESSURE RESPONSE TO HEAT INPUT TRANSIENT WITH RELIEF VALVE OPENING
FIGURE H33
- RCS PUMP STARTUP IN 1 LOOP
- RCS VOLUME = 13000 CU.FT.
- RCS/SG ΔT = 180°/280°
SETPOINT, PSIG
600
500
400
WESTINGHOUSE ANALYSIS
CON EDISON ANALYSIS
2 RELIEF VALVES
1 RELIEF VALVE
RCS PRESSURE, PSIA
1000
900
800
700
600
500
400
300
200
100
0
0
4
8
12
16
20
24
28
32
36
40
44
48
52
56
TIME, SEC.