DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATIONS

REMOVE PAGES	INSERT PAGES
2.2-1	2.2-1
2.3-3	2.3-3
2.3-5	2.3-5
2.3-7	2.3-7

2.2 SAFETY LIMITS - REACTOR COOLANT SYSTEM PRESSURE

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.

Specification

2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.

2.2.2 The setpoint of the pressurizer code safety valves shall be in accordance with ASME, Boiler and Pressurizer Vessel Code, Section III, Article 9, Summer 1967.

Bases

The reactor coolant system [1] serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME code, Section III, is 110% of design pressure. [2] The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under USAS Section B31.7 is 110% of design pressure. Thus, the safety limit of 2750 psig (110% of the 2500 psig design pressure) has been established. The settings, the reactor high pressure trip (2355 psig) and the pressurizer safety valves (2500 psig) [4] have been established to assure never reaching the reactor coolant system pressure safety limit. The initial hydrostatic test was conducted at 3125 psig (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the Reactor Coolant pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2450 psig.

REFERENCES

(1) FSAR, Section 5

(2) FSAR, Section 5.2.3.10.1

(3) FSAR, Section 5.2.2.3, Table 5.4-7

(4) FSAR, Section 5.4.6, Table 5.1-1

Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the reactor coolant system (RCS) high pressure setpoint is reached before the nuclear overpower trip setpoint. The high RCS pressure trip setpoint (2355 psig) ensures that the pressure remains below the safety limit (2750 psig) for any design transient.[2] The low pressure (1800 psig) and variable low pressure (11.14 T_{out} - 4706) trip setpoints shown in Figure 2.3-1 ensure that the minimum DNBR is greater than or equal to the minimum allowable DNBR for those accidents that result in a reduction in pressure.[3,4] The limits shown in Figure 2.3-1 bound the pressure-temperature curves calculated for 4, 3, and 2 pump operation.

Accounting for calibration and instrumentation errors, the safety analyses used a variable low RCS pressure trip setpoint of (11.14 T_{out} - 4756).

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setpoint (618°F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperatures. Accounting for calibration and instrumentation errors, the safety analyses used a trip setpoint of 620°F.

Reactor Building Pressure

The high reactor building pressure trip setpoint (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

Shutdown Bypass

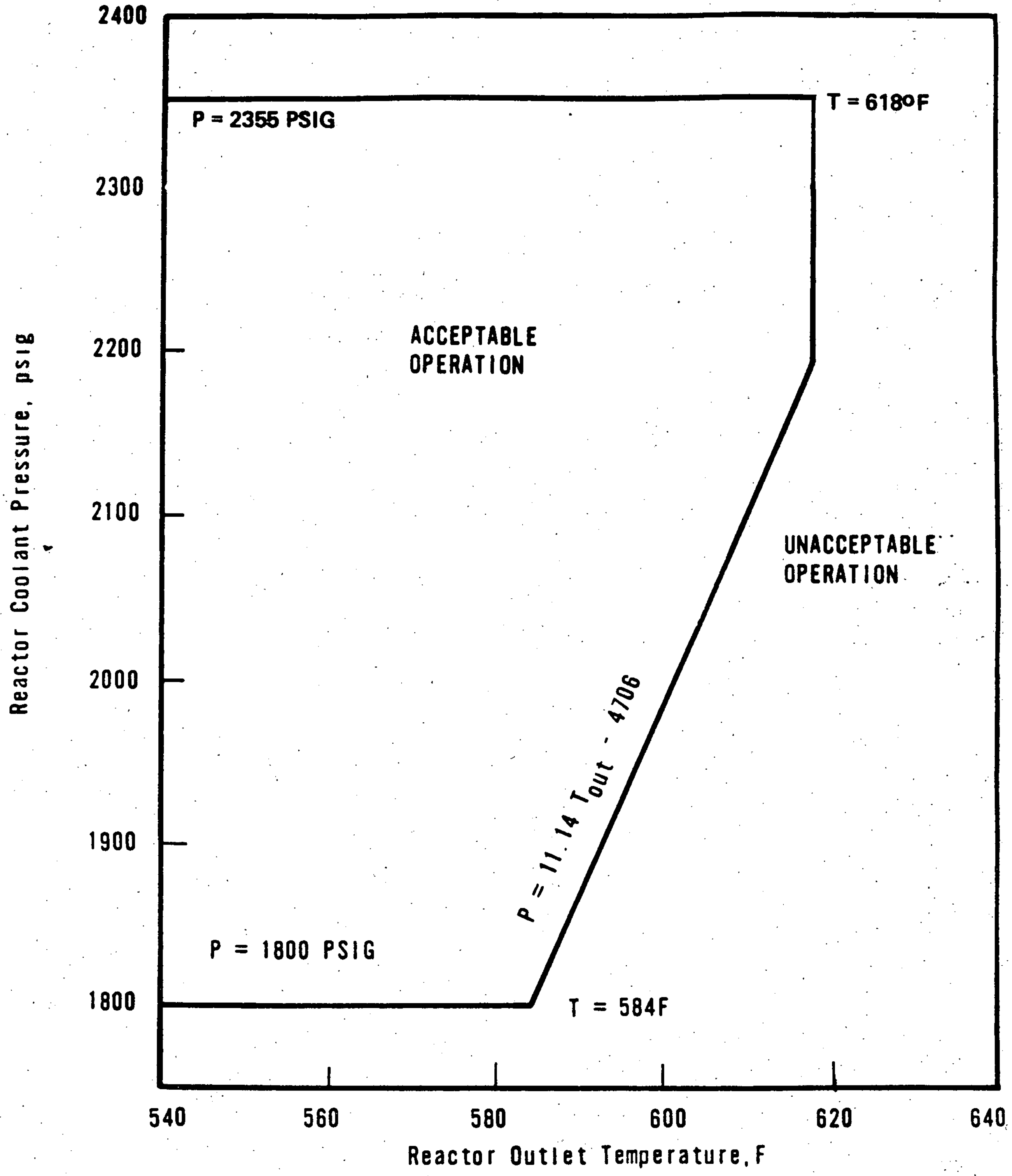
In order to startup the reactor and to be able to perform control rod drive tests and zero power physics tests (see Technical Specification 3.1.9), there is provision for bypassing certain segments of the reactor protective system (RPS). The RPS segments which can be bypassed are given in Table 2.3-1. Two conditions are imposed when the RPS is bypassed:

1. By administrative control the Nuclear overpower trip setpoint is reduced to a value of $\leq$ 5.0% of rated power.

2. The high reactor coolant system pressure trip setpoint is automatically lowered to 1720 psig.

The high RCS pressure trip setpoint is reduced to prevent normal operation with part of the RPS bypassed. The reactor must be tripped before the bypass is initiated since the high pressure trip setpoint is lower than the normal low pressure trip setpoint (1800 psig).

The overpower trip setpoint of $\leq$ 5.0% prevents any significant reactor power from being produced when performing physics tests. If no reactor coolant pumps are operating, sufficient natural circulation would be available to remove 5.0% of rated power.[5]



PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINTS
UNITS 1, 2, AND 3
OCONEE NUCLEAR STATION

Figure 2.3-1

TABLE 2.3-1

Reactor Protective System Trip Setting Limits

	RPS Trip		RPS Trip Setpoint	Shutdown Bypass
1.	Nuclear Overpower		105.5% Rated Power	5.0% Rated Power$^{(1)}$
2.	Flux/Flow/Imbalance		1.07	Bypassed
3.	Pump Monitors	a.	> 0% Rated Power loss of two pumps in one reactor coolant loop	Bypassed
		b.	> 55% Rated Power loss of two pumps	
		c.	> 0% Rated Power loss of one or two pumps during two pump operation	
4.	High Reactor Coolant System Pressure		2355 psig	1720$^{(2)}$
5.	Low Reactor Coolant System Pressure		1800 psig	Bypassed
6.	Variable Low Reactor Coolant System Pressure		P (psig) = (11.14 T_{out} - 4706)$^{(3)}$	Bypassed
7.	High Reactor Coolant Temperature		618°F	618°F
8.	High Reactor Building Pressure		4 psig	4 psig

(1) Administratively controlled reduction set only during reactor shutdown.

(2) Automatically set when other segments of the RPS are bypassed.

(3) T_{out} is in degrees Fahrenheit (°F).

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 2

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Duke Power Company (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the Commission's regulations in 10CFR50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

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

The proposed change will result in improved operational safety through a reduction in challenges to safety systems, fewer equipment transient cycles, and reduced complexity of transients experienced, with a negligible increase in the PORV opening probability. The total PORV openings per Reactor year is neglibly changed by this proposed amendment since operator actions and to a lesser degree, instrumentation and control faults dominate the total PORV opening frequency. As discussed in the NRC Staff's Safety Evaluation (NRC letter dated April 22, 1986), the probability of a SBLOCA caused by a stuck-open PORV is within the WASH-1400 range of 10^{-2} to 10^{-4} per reactor-year. In raising this setpoint the consequences of a stuck-open PORV is not altered in any fashion. Therefore, this change will not increase the probability or consequences of an accident.

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

It has been determined that a new or different kind of accident will not be possible due to this change. The design high pressure trip setpoint is 2355 psig, therefore, the original FSAR analyses remain applicable for this setpoint. This change in setpoint does not create the possibility of a new or different kind of accident.

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

The change will result in improved operational safety through a reduction in challenges to safety systems, fewer equipment transient cycles, and reduced complexity of transients experienced. The change would provide more margin to the high pressure reactor trip setpoint and would allow some minor plant upsets to either avoid reactor trip or provide the operator sufficient time to perform an action which would not result in a reactor trip. In addition, the staff had concluded within the safety evaluation provided by NRC letter dated April 22, 1986 that this setpoint change meets the NRC requirements of NUREG-0737 items II.K.3.2 and II.K.3.7 regarding PORV openings and PORV caused SBLOCA. Therefore, there will not be a reduction in the margin of safety.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards consideration. Example (vi)

relates to a change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan.

In this case, the change proposed in this request is similar to Example (vi) in that raising the RCS high pressure trip setpoint will result in a negligible increase in the probability of a previously-analyzed accident and may in some way reduce a safety margin, but the results of this change are clearly within all acceptable criteria.

By letter dated April 22, 1986. NRC Staff review of B&W topical report BAW-1890 on the high pressure reactor trip setpoint concluded that it is acceptable to increase the high pressure reactor trip setpoint for B&W plants from 2300 psig to 2355 psig while the PORV setpoint remains at 2450 psig. Additionally, the Staff concluded that this setpoint change meets the NRC requirements of NUREG-0737, Items II.K.3.2 and II.K.3.7 regarding PORV openings and PORV caused SBLOCA. Similarly, the requirements on this matter embodied in IE Bulletin 79-05B are also met.

The overall requirement for the maximum allowable Small Break LOCA probability at the PORV is that this occurs less than .001 times per reactor year. The specific requirement for the high pressure trip transient is that no more than 5% of the high pressure trips are allowed to open the PORV. Previous analyses calculated 2.2 E-4 PORV opening per high pressure event when the PORV was 150 psi above the trip setpoint. The current analyses calculated 10E-5 PORV opening per high pressure event when the PORV was 95 psi above the trip setpoint. Based on the last 5 years of operating experience and data, the current analysis is considered the more realistic of the two. However, both analyses are clearly within all acceptable criteria for NRC Small Break LOCA requirements.

Duke has concluded, based on the above, the subject B&W topical report, and Staff comments in the April 22, 1986 letter, that there is a No Significant Hazards Consideration involved in this amendment request.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 3

JUSTIFICATION FOR RAISING SETPOINT FOR REACTOR TRIP ON HIGH PRESSURE

BAW-1890
SEPTEMBER 1985

BAW-1890
77-1159095-00
SEPTEMBER 1985

THE B&W OWNERS GROUP

Transient Assessment Program

JUSTIFICATION FOR RAISING SETPOINT FOR REACTOR TRIP ON HIGH PRESSURE

Babcock & Wilcox
a McDermott company

BAW-1890
77-1159095-00
SEPTEMBER 1985

JUSTIFICATION FOR RAISING SETPOINT
FOR
REACTOR TRIP ON HIGH PRESSURE

Babcock & Wilcox
Nuclear Power Division
P. O. Box 1260
Lynchburg, Virginia 24505

Babcock & Wilcox
a McDermott company

-1890
77-1159095-00

SEPTEMBER 1985

JUSTIFICATION FOR RAISING SETPOINT
FOR
REACTOR TRIP ON HIGH PRESSURE

BY

J. W. PEGRAM

C. W. TALLY

J. A. WEIMER

CONTENTS

List of Tables

List of Tables (Cont'd)

List of Figures

Executive Summary

This report documents the results of an engineering evaluation performed to justify increasing the setpoint for reactor trip on high pressure from its current value of 2300 psig to its pre-1979 value of 2355 psig. The potential benefit of the change is a reduction in the frequency of reactor trips.

The conclusion of this study is that the high pressure trip setpoint can be raised to 2355 psig with negligible impact on the frequency of opening the PORV during anticipated overpressurization transients. The sum of the transient induced pressure overshoots* and the total effective instrument string errors would result in the PORV (set at 2450 psig) opening one time out of 100,000 high pressure trip transients.** This is a sufficiently low probability to assure that a normal high pressure trip would virtually never open the PORV and, consequently, will satisfy the NRC small break LOCA (SBLOCA) criteria for the PORV. Also, since 2355 psig is the design high pressure trip setpoint, the original FSAR analyses remain applicable for this setpoint.

Of the 47 high pressure trip transients reviewed in detail for this study, twelve were judged to have high potential for avoidance if more margin to reactor trip had been available. In addition, if the high pressure trip setpoint were raised to 2355 psig, B&W plants would be capable of surviving turbine trips

- - - - -

*Overshoot is the difference between the maximum indicated RC pressure and the actual trip pressure.

**In five years (1980 through 1984) all B&W NSS plants experienced a total of ˜65 high pressure trip transients. This is less than 2 trips per plant per year.

without reactor trips at power levels well above the current 20% arming threshold for the anticipatory reactor trip (ART) on turbine trip. Raising the ART arming threshold to a level consistent with the runback capability at the 2355 psig trip setpoint would have avoided twelve of the reactor trips on turbine trip experienced during 1980-84. Based on 1980-84 experience, the results of this study suggest that the potential benefit of raising the high pressure trip setpoint to 2355 psig is a 10% reduction in the average reactor trip rate for B&W plants.

The NRC requirements regarding the PORV are: (1) probability of SBLOCA due to stuck open PORV must be less than .001 per reactor year and (2) less than 5% of the high pressure trips are allowed to open the PORV. Both requirements are met with the high pressure trip setpoint at 2355 psig.

The methods used for this analysis are similar to previous analyses on this subject (see References 2 and 3). A major difference is that this study predicts overshoot based on actual plant high pressure trip transients in which the PORV did not open. The previous analyses calculated overshoots (for a closed PORV) based on plant data where the PORV opened prior to reactor trip. Compensating for the open PORV added more uncertainty to these past calculations. The data used for the current analysis provided a more realistic overshoot and probability distribution of the overshoot. The comparable results of these two studies are 2.2×10^{-4} PORV opening per high pressure event (old) verses 1.0×10^{-5} PORV opening per high pressure event (new).

1. BACKGROUND

The B&W NSSS, with its Once Through Steam Generators (OTSG) and Integrated Control System (ICS), was designed with the capability to adjust to minor plant upsets and certain anticipated events without a reactor trip. The system was designed to initiate a plant runback, upon detection of an upset or equipment malfunction, to a power level consistent with the limiting condition. The effectiveness of this runback capability is illustrated by Table 1-1. Table 1-1 was extracted from reference (1) and shows 47 successful runback actions for one B&W station for 5.5 years of reactor operation prior to 1979.

Following the TMI-2 accident, several changes were made to B&W plants for the purpose of reducing challenges to the Pilot Operated Relief Valve (PORV). These changes have been effective in reducing challenges to the PORV but have also resulted in a substantial reduction in runback capability for B&W plants. The changes which have had the greatest impact on runback capability are:

- Raising the PORV opening setpoint from 2255 psig to 2450 psig
- Lowering the setpoint for reactor trip from 2355 psig to 2300 psig
- Implementation of an Anticipatory Reactor Trip (ART) on turbine trip for reactor power levels of 20% and above.

A consequence of this loss of runback capability is that events such as the loss of 1 of 2 main feedwater pumps or minor feedwater upsets now nearly always result in a reactor trip.

The B&WOG is committed to reducing reactor trips to improve plant availability by keeping plants on line and to improve safety by

reducing challenges to safety systems. As part of a generic trip reduction program, the B&WOG has identified the restoration of runback capability as a prime candidate for achievement of trip reduction. Accordingly, the B&WOG has undertaken various studies to:

- Identify the current constraints on runback capability
- Examine the feasibility of removing or relaxing these constraints
- Identify modifications which would enhance the probability of successful runbacks.

Included in this effort was a study to examine the potential benefits and feasibility of restoring the setpoint for reactor trip on high pressure to its original value of 2355 psig while keeping the PORV setpoint at 2450 psig. The results of this study are reported herein.

Table 1-1. Successful Runback Actions (Pre 1979)

Event	No.	Initial power range	Final power range %
Turbine trip/load rejection	14	15-100	2-22
Feed pump trips (1 of 2)	14	100	55-75
Rod drops	4	100	50-60
10% step load increases	4	17-90	27-100
10% step load decreases	11	37-100	15-90
Total	47		

2. POTENTIAL BENEFITS

2.1. Impact of Post-TMI Changes

In an effort to identify the potential benefits of restoring the high pressure trip setpoint to its original value, a detailed comparison of pre- and post-TMI trip frequency was made. A primary objective of this comparison was to determine the impact of the changes on high pressure trip frequency.

Figure 2-1 displays the average high pressure trip frequency for B&W plants for the pre-1979 and post-1979 periods. The pre-1979 data for each plant includes trips occurring from the date of commercial operation through the end of 1978. The post-1979 data includes trips from 1980 through 1984. The trips which occurred in 1979 were not included in the comparison as the setpoint changes were implemented at various times during the year. Figure 2-1 shows that high pressure trip frequency increased for some plants and decreased for others. The average for B&W plants as a group shows no change pre-1979 versus post-1979.

This simple comparison of pre- and post-1979 high trip rates does not, however, accurately reflect the impact of the changes. The reason for this is the effect of the addition of the Anticipatory Reactor Trips which occurred in 1979. These trips, turbine trip (power > 20%) and loss of both feedwater pumps, pre-empt high pressure trips in the post-1979 period. These trips, in the post-1979 period, are in fact anticipatory high pressure trips. If the ART were not in place, a turbine trip or loss of both feed pumps event would result in a high pressure trip with the current setpoints. Thus, a more reasonable comparison, for assessing the impact of post-TMI changes on trip frequency, is to compare pre-

1979 high pressure trips with post-1979 high pressure plus ARTS trips.

When this comparison is made, a different picture emerges, as shown in Figure 2-2. This figure shows that, when trips due to the same causes (turbine trip, loss of both MFWPs, other) are compared, the average trip frequency due to these causes has doubled for B&W plants as a group since the implementation of post-TMI setpoint changes. Figure 2-2 also shows that some plants have been affected more than others by the changes. The reason for this is that individual B&W plants differed in pre-1979 runback capability. Those plants which had greater runback capability show the larger increases in trip frequency due to the loss of this capability.

2.2. Potential For Trip Reduction

As noted previously, two key factors contributing to the reduction of runback capability were the raising of the PORV opening setpoint and the lowering of the high pressure trip setpoint. It is recognized that, prior to TMI-2, the PORV played a significant role in keeping primary pressure below the trip setpoint during runbacks. For this reason, restoration of runback capability to pre-TMI levels cannot be expected with the PORV set to open at pressures above the high pressure trip setpoint. However, a partial restoration could be achieved if the high pressure setpoint were restored to its original 2355 psig value. The basis for this statement is as follows:

A. Increasing the high pressure setpoint to 2355 psig would increase the operating margin to high pressure trip by 55 psi. For slow moving transients resulting in primary pressure increase rates of 1 to 2 psi/second, the increased margin would provide an additional 30 seconds to one minute before reaching the trip setpoint. This would provide additional time for an operator to determine the cause of an upset and take manual actions to avoid a trip. Examples of such actions are switching to an alternate input signal

upon discovery of a failed input, initiation of pressurizer spray, or placing appropriate portions of the ICS in manual.

It is not possible from a review of past transients to establish with certainty that a given trip could have been avoided by operator action if more margin were available. A qualitative indication of trip reduction potential was developed, however, by categorizing previous trips according to their potential for avoidance by operator action. For example, a fast moving transient such as that initiated by a load rejection (CR-3, 7/5/82) was judged as having poor potential for avoidance. A slow moving transient which might be caused by a failed ICS input (CR-3, 7/29/83) was judged as having good potential for avoidance. The 47 high pressure trips for which sufficient data was available were categorized as described above. This review identified 12 trips as having good potential for avoidance through operator action if increased margin were available. These trips are identified in Table 2-1.

B. In a separate study, B&W demonstrated by analysis that if the high pressure setpoint were raised to 2355 psig, B&W plants could survive a turbine trip from 40% power without a reactor trip. Thus, if the high pressure setpoint were raised to 2355 psig and the power level for arming the Anticipatory Reactor Trip on turbine trip were raised to a value above 40%, turbine trips occurring at powers below 40% would no longer result in reactor trips. Figure 2-3 indicates the potential reduction. Assuming successful runbacks for all turbine trips at initial power levels ≤ 40%, 12 of the 1980-1984 trips could have been avoided. These trips are also listed in Table 2-1.

C. Had the 24 trips listed in Table 2-1 been avoided, the average trip frequency for B&W plants for the 1980 through 1984 period would have been 4.6 trips/year/plant instead of 5.3 trips/year/plant. This represents a 13% reduction in

average trip frequency. Assuming a future mix of transients similar to that of the 1980-1984 period, it is reasonable to project a reduction in average trip frequency of approximately 10%.

Table 2-1. Reactor Trips Having Good Potential for Avoidance with Increased Setpoint

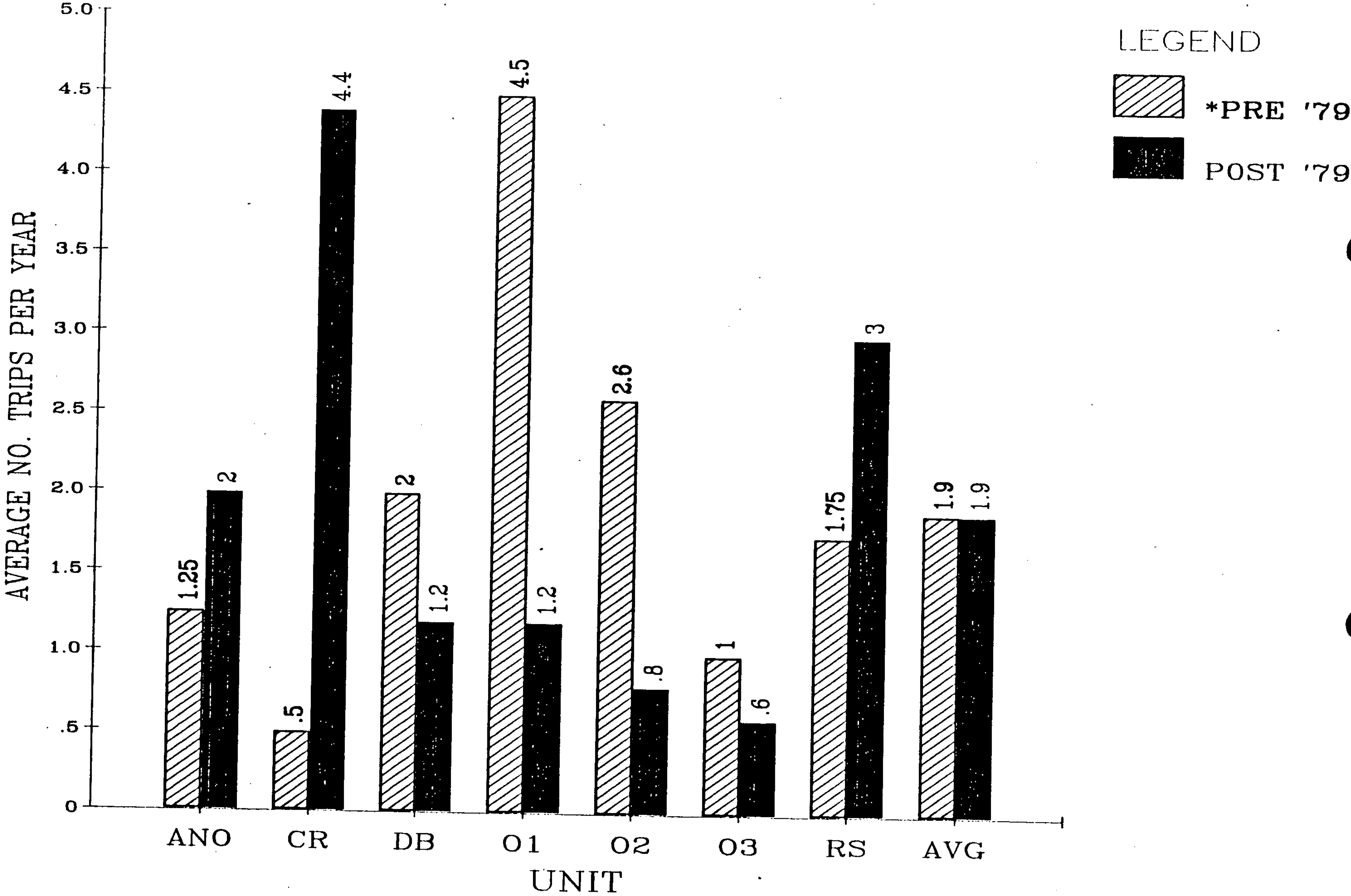
High Pressure Trips

Date	Plant	Initial Power	Comment
3/19/80	CR-3	53	Maneuvering/FWP Trip
12/5/80	O-2	14	Manual Underfeed
9/5/81	O-3	30	FW Valve Sticking
6/23/81	RS	50	Manual FWP Operation, Flow Oscillations
6/17/81	RS	80	MFWP Trip
8/8/82	CR-3	94	Loss of RC Flow Signal
7/15/82	CR-3	100	Loss of RC Flow Signal
11/4/82	O-2	100	MFW Pump Trip
5/21/82	O-1	90	Runback on Rod Drop
3/24/82	O-1	51	Turbine & FW A Control in Manual
7/256/83	CR-3	9	MFW Flow Reduction
12/3/83	RS	65	MFW Upset, Valves & Pumps in Manual

Anticipatory Reactor Trips on Turbine Trip

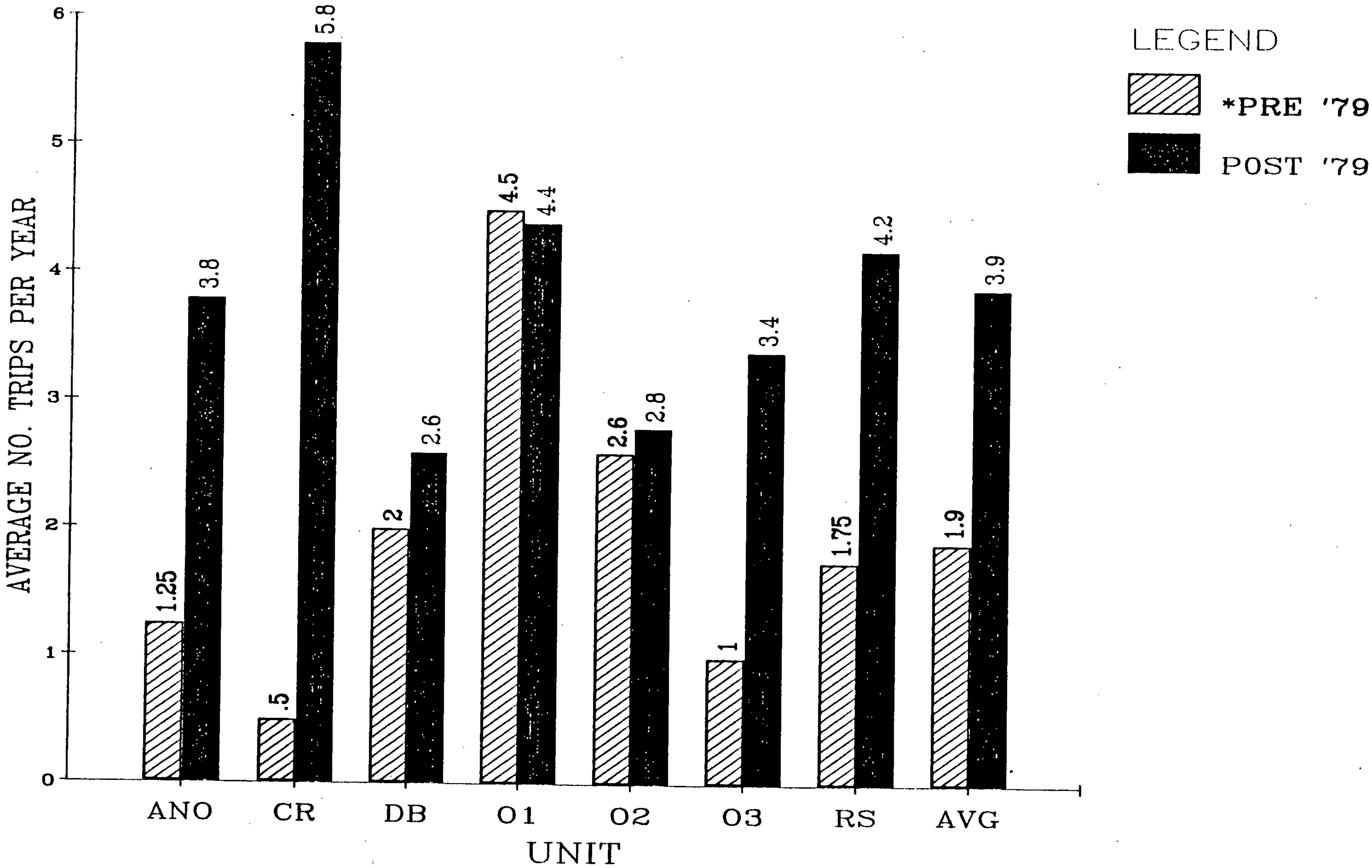
Date	Plant	Initial Power
11/6/80	DB-1	18
3/14/81	O-3	4
1/1/82	O-1	24
1/1/82	O-1	20
1/2/82	O-1	28
5/19/82	O-2	20
7/9/82	CR-3	11
7/9/82	CR-3	18
6/1/83	O-1	15
8/6/83	RS	20
8/10/83	RS	40
6/7/84	O-3	19

Figure 2-1. Average Number of High Press. Trips Per Year



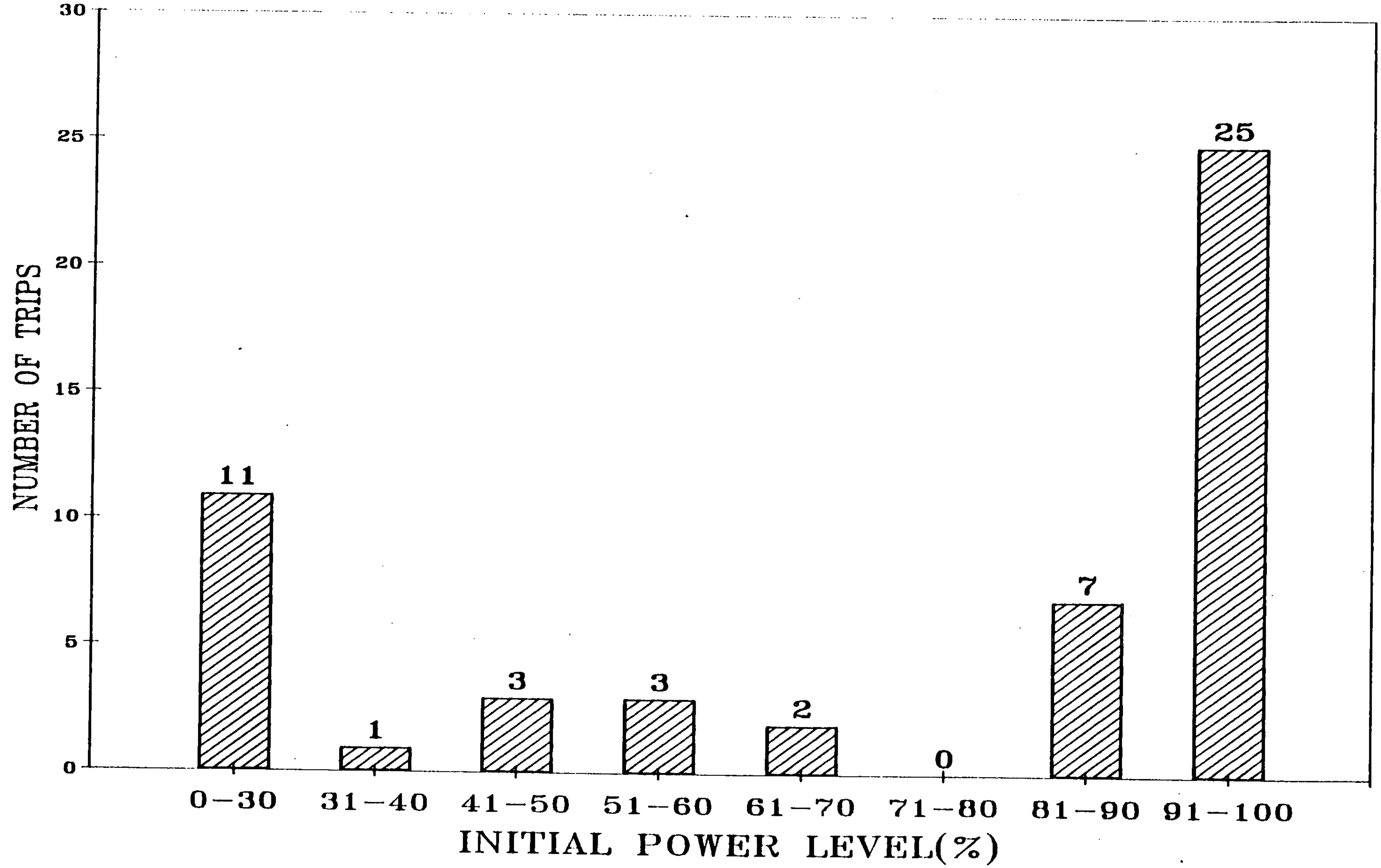
* PRE '79 INCLUDES YEARS OF COMM. OPERATION '73 THRU '78

Figure 2-2. Average Number of High Pressure and ARTS Trips Per Year



* PRE '79 INCLUDES YEARS OF COMM. OPERATION '73 THRU '78

Figure 2-3. Anticipatory Reactor Trips on Turbine Trip for B&W Plants (1980-1984)



3. IMPACT ON PORV OPENING FREQUENCY

The high pressure trip setpoint was lowered to 2300 psig and the PORV opening setpoint raised to 2450 psig to reduce the probability of opening the PORV for several categories of events. These are:

A. Overpressure transients

B. Transients with Delayed Auxiliary Feedwater

C. Operator Action Under ATOG Guidelines

D. Instrumentation and control faults

E. Overcooling transients that initiate HPI

Table 3-1 lists the PORV opening frequencies for these events as estimated in previous studies (References 2&3) and the current study. Two important points are illustrated by this table:

1. Only the PORV opening frequency for overpressure transients is affected by the high pressure trip setpoint. Opening frequencies for the other events are independent of the high pressure trip setpoint.

2. The PORV opening frequency due to overpressure transients is a relatively small contributor to total PORV opening frequency, the total PORV opening frequency is dominated by the frequency of opening due to operator actions under ATOG.

The current study concludes that the PORV opening frequency for overpressure transients is 1.86×10^{-5} openings per reactor year with a high pressure trip setpoint of 2355 psig and a PORV opening setpoint of 2450 psig. With the current 2300 psig high pressure trip setpoint, the frequency would be less than 1.86×10^{-5}. The total PORV opening frequency for all other causes (Reference 3) is 8.06×10^{-2}. If we assume the PORV opening frequency for overpressure transients to be zero at the 2300

psig trip setpoint, the following comparison can be made:

Total PORV
Opening Frequency
at 2300 psig setpoint $= 8.06 \times 10^{-2} + 0 = 8.06 \times 10^{-2}$

Total PORV
Opening Frequency
at 2355 psig setpoint $= 8.06 \times 10^{-2} + 1.06 \times 10^{-5} \quad 8.06 \times 10^{-2}$

Thus, raising the high pressure trip setpoint from the current value of 2300 psig to the original value of 2355 psig will have a negligible impact on the frequency of PORV openings.

Section 5.0 of this report discusses differences between the current study results and the results of the previous study.

Table 3-1. PORV Opening Frequency

	Opening Per Reactor Year		
	B&W '81 (Ref. 2)	TER (Ref. 3)	Current Study
Overpressure transients	$3.9x0^{-5}$	$2.2x10^{-3}$	$2x10^{-5}$
Transients with delayed aux. FW	$7.6x10^{-4}$	$7.6x10^{-4}$	No change
Operator actions under ATOG	$1.58x10^{00-2}$	$7.7x10^{-2}$	No change
I&C faults	$1.7x10^{-3}$	$1.7x10^{-3}$	No change
Overcooling transients that initiate HPI	$8.4x10^{-4}$	$8.4x10^{-4}$	No change
Total	$1.9x10^{-2}$	$8.28x10^{-2}$	$8.06x10^{-2}$

4. ANALYSIS

4.1. Overview

4.1.1. Objectives

The purpose of this analysis was to determine if the high pressure trip setpoint could be raised to a value above the present 2300 psig setpoint without exceeding NRC guidelines for PORV opening frequency. The maximum allowable set point was based on the criterion that the probability of opening the PORV will be sufficiently low so as not to exceed the maximum allowable small break LOCA limit of 10^{-3} events per reactor year. (This SBLOCA limit for the PORV was set by the NRC per Reference 4).

4.1.2. Technical Approach

The main thrust of this analysis was to determine the maximum reactor coolant (RC) pressure relative to opening the PORV during over pressure transients. High pressure transients referred to in this document do not include cases with excessive HPI or total loss of main and auxiliary feedwater. These transients could open the PORV regardless of the high pressure trip setpoint although the PORV opening would occur well after reactor shutdown.

This analysis included a statistical evaluation of instrumentation errors (associated with the trip set point and PORV), and the amount of indicated RC pressure increase above the trip set point (overshoot). The transmitter and instrument errors were based on actual plant set point and calibration data and instrument design specifications. The maximum indicated system pressure (overshoot) was evaluated based on (1) plant

data from Transient Assessment Program reports (Tap Reports), and (2) information derived from "Digital Power Train" computer code analyses.

The sequence of the analysis is outlined below.

1. First, plant transient data was reviewed to determine the expected range of overshoots during the anticipated over pressure transients. This data was all based on the current 2300 psig trip setpoint (section 4.2).

2. Secondly, Power Train analyses were performed to determine whether the overshoot would change when the high pressure trip setpoint was increased (section 4.3).

3. Next, this overshoot data was statistically combined with instrumentation errors to predict the probabilities of the PORV opening from various high pressure trip setpoints (section 4.4).

4. Next, these results were compared to NRC limitations on the PORV and previous analyses on this subject (section 5).

4.2. Plant Data Review

4.2.1. Overview of Data

Various plant data were reviewed. The data included high pressure trip setpoints, calibration tolerances and errors associated with these set points, and TAP reports involving over-pressure transients. The trip setpoints and calibration data were used to determine the pressure at which the reactors actually tripped. This data was then factored into the TAP report evaluation to determine the overshoot.

4.2.2. Data Evaluation

4.2.2.1. High Pressure Setpoints

As previously discussed, the existing nominal high pressure trip setpoint is 2300 psig. The method used to insure that this setpoint is not exceeded is outlined below.

1. Pressure transmitters are calibrated during refueling.

2. Each of the 4 high pressure trip set point channels are initialized at a value that allows for drift and instrument string errors. (See Table 4-1)

3. RPS channels are tested periodically throughout the subsequent cycle by artificially inducing a ramping voltage to the RPS bistable and determining the trip voltage (i.e., pressure) for the channel.

The set points and error limits based on this procedure are listed in Table 4-1. The data in this table are based on conversations with B&W site personnel.

As can be seen by the data in Table 4-1, the actual trip set point varied from 2285 to 2298 and the reactors consistently trip within 2 or 3 psi of the set points. Therefore, the initial RC pressure used to calculate overshoot was assumed to be the specific trip set point of each plant. However, a uniform distribution of zero to plus five psi error was assumed for the trip set point for predictions of maximum RC transient pressure during the Monte Carlo simulation (see section 4.4).

4.2.2.2. Indicated System Pressures During Transients

The maximum indicated RC pressure during an overpressure transient would include:

1. actual overshoot that occurred due to the nature of the transient,

2. any additional instrument string error downstream of the RPS channel,

3. uncertainties due to the print out device readability,

4. uncertainties due to the frequency at which the data were recorded.

Items 2, 3, and 4 are functions of the device used to monitor pressure and are discussed in Section 4.2.2.4.

It was not apparent what part of the indicated RC pressure during a transient was due to actual overshoot and what part to any

instrument errors. Therefore, the indicated maximum RC pressure minus the trip set point was all assumed due to overshoot. This is a conservative assumption because any instrument error that may be included in this overshoot is redundantly covered in Section 4.4 (statistical evaluation of data).

4.2.2.3. Overshoot

Overshoot is the maximum RCS pressure in excess of the actual trip pressure. It is primarily a function of the relative rates of heat transfer in the steam generator compared to heat transfer in the core region - up through the first few seconds after the trip. The relative rate of heat transfer determines the rate of RC pressurizations. The delay time between the initiation of the trip signal until the core thermal power is adequately decreasing dictates how long the heat up continues. This time delay is usually one to four seconds after the trip. The maximum pressure that occurs during this time is dependent on the pressurization rate prior to the trip. This pressurization rate varies from 2 to 3 psi/second for a mild feedwater upset to 40 psi/second for some load rejection transients.

4.2.2.4. Instrumentation Accuracy

The accuracy of the indicated RC pressure is primarily dependent on whether it is a narrow range (NR) or wide range (WR) signal. The NR signal is usually more accurate than WR. However, determining a relatively accurate peak RC pressure during a transient may be more dependent on the recording device. This is because the readability of the device or the frequency of the measurement may introduce more uncertainty than the signal itself. For example, the frequency of the plant computer monitoring (every 15 seconds) could completely miss the peak RC pressure during a load rejection since the peak pressure occurs in about 5 seconds into this event. Similarly, even though the analog signal to an indicator or recorder may be very accurate, the scale of the readout device may make it difficult to read the peak pressure within 10 psi. Because of these types of problems,

the peak pressure from each high pressure trip transient was evaluated based on the particular data available. Narrow range signals were used where available. If more than one channel or device was available, the data was compared. Extrapolation or curve fitting was applied when the appropriate data were available.

Using these methods for plant data evaluation, the following general observations were noted:

1. Peak pressure from extrapolated digital output often agreed well with analog strip chart data.

2. NR pressures from the A and B loops were usually within a few psi of each other.

3. Wide range pressure changes during a transient were about the same as narrow range pressure changes.

These results led to the conclusion that, in general, the overshoot predictions were relatively accurate and that the degree of overshoot was primarily due to the transient itself and not due to any large instrument errors.

4.2.2.5. Monitoring Device

The plant data evaluation was based on five different monitoring devices. These included:

1. Reactimeter
2. Plant Computer
3. Special Transient Monitoring Equipment (i.e. Test Transient Monitor (TTM) at Oconee or the post trip review at CR-3)
4. Control Room Alarms Data
5. Strip Charts

The following is a brief description of the data from each of these devices:

Reactimeter

The reactimeter is a digital device that is typically set to track plant parameters at frequencies down to 3 seconds.* It takes a "snap shot" of the data at each time step. The reactimeter is usually connected to NR channels.

Plant Computer

The Plant Computer is functionally similar to the reactimeter except for the sampling frequency. The data is averaged and printed at intervals of 15 to 30 seconds. The usefulness of this data for this study depended on the plant, the circumstance, and the timing of the data.

Test Transient Monitor

The TTM at Oconee is similar to the plant computer except it monitors data at one second intervals. However, the output format is usually in a graphical form (in the TAP Reports) with data points often plotted every 6 to 12 seconds. The frequency of plotting is sometimes too coarse to accurately monitor peak RC pressures during many transients. However, some of the transients reviewed had listed or plotted RC pressure data on a one, two or three second interval. These cases provided accurate resolution of the time delays to peak pressure during the transient.

Alarms Data

Control room alarms data is a snapshot taken at the exact time a limit is reached or in some cases it prints an "on demand" listing of specific data groups. The usefulness of this data for maximum RCS pressure predictions also depends on the plant, the timing, and the circumstances.

- - - - -

*The reactimeter is capable of one second data monitoring but usually only set at this fast frequency for test situations.

Strip Chart Data

Strip Chart Data is an analog representation of the assigned plant parameters. Even though it is a real time analog signal, its time scaling is so coarse that it does not give a good profile of the faster transients. Strip chart data is very good, however, for showing maximum and minimum values.

Each plant has its own combination of devices and channels for monitoring transient data. As pointed out in the above discussion, each recording device has some disadvantage for determining peak RC pressure. However, a combination of data from two or more devices can give added confidence in predicting peak RC transient pressures.

4.2.2.6. Operating Plant Data

Table 4-2 is a summary of the high pressure trips in the past few years for which the best data was available. This table is based on TAP report data. The maximum RC pressures were determined using the methods previously discussed.

The conclusions drawn from a review of the data in this table are:

1. The three primary categories of high pressure trip events are (a) total or partial loss of feedwater, (b) feedwater/power mismatches during runbacks, and (c) load rejections/MSIV closings. Many of these events were initiated by an erroneous ICS input signal. Also, the RCS pressurization rate of these transients varied from 2 psi/sec to 40 psi/sec. This corresponds to a maximum transient time* of about 2.0 minutes down to a minimum time of 5 seconds.
2. The maximum transient pressure overshoots usually occurred during the loss of load transients (see transients 1/23/85, 4/11/85, 6/20/82, 1/19/82, 2/5/80).

- - - - -

*The transient time is from the initiation of the event to the time of maximum RC pressure.

3. For the 47 overpressure transients listed, the overshoot distribution is as follows:

Overshoot Range (psi)	Number of Events
0 - 10	33
11 - 20	7
21 - 30	1
31 - 40	3
41 - 50	2
51 - 60	1

This data was used to construct the overshoot distribution for the Monte Carlo analysis.

4.3. POWER TRAIN Analysis

The five years of plant daa discussed in Section 4.2 provide a sound basis for characterizing the overshoot distribution with a 2300 psig trip setpoint and no PORV opening. The objective of the POWER TRAIN analysis was to determine whether or not this overshoot distribution would change with a 2355 psig setpoint.

4.3.1. Analysis Method

The approach to this task was to:

- o Select a representative transient for evaluation
- o Develop a 177 FA plant model for evaluation
- o Compare the model's predictions for a setpoint of 2300 psig against predictions for a 2355 psig setpoint

The plant data from the 47 trip events indicates that a turbine trip event from full power with no anticipatory trip causes the largest RCS pressure overshoot. This transient represents a severe primary to secondary power mismatch during those first few seconds of the event when the steam generator nucleate boiling region is suppressed. Therefore, it is a logical choice for evaluation.

The second step of the evaluation was to compare the model's results against plant data for a setpoint of 2300 psig. For all cases that were run, the POWER TRAIN model overpredicted the peak RCS pressure by approximately 20 to 60 psi. A typical comparison

of POWER TRAIN results with plant data is shown in Figure 4.3-1. Although duplication of the exact RCS pressure profile is not essential, the goodness of agreement between the model results and the plant data add confidence to the results.

The third part of the evaluation was to simulate the turbine trip event with a high pressure trip setpoint of 2355 psig and compare the overshoot to the overshoot generated at 2300 psig. The difference between the overshoots for the two setpoints is the result of primary interest.

4.3.2. Results

POWER TRAIN predicts an overshoot of 50 psi for the turbine trip (and no ARTS trip) with the high pressure trip setpoint of 2300 psig. This compares with an observed 40 psi overshoot for similar plant transients. POWER TRAIN predicts an overshoot of 30 psi for the same event but with the trip setpoint at 2355 psig. A comparative plot of these results is depicted in Figure 4.3-2.

The conclusion from this comparison is that the pressure overshoot is a weak function of the high pressure trip setpoint in the 2300 to 2355 psig setpoint range.

The results suggest that the overshoot actually decreases as the setpoint is raised. For purposes of this evaluation, however, the overshoot distribution was assumed not to change. This conclusion can be explained in terms of the physical phenomena affecting the heat transfer rate in the steam generators, as discussed in the following paragraph.

4.3.3. Overshoot at 2355 psig vs 2300 psig

The RCS pressure overshoot is a function of the steam generator heat transfer rate. Since the heat transfer is predominantly in the nucleate boiling region of the steam generator (SG), the relative height of this region throughout the transient is an indication of the rate of SG heat transfer and consequently RC pressure. The nucleate boiling (NB) region is in turn dependent

on the SG pressure and steam flow rate. As the SG pressure increases during the transient, the NB level tends to decrease thereby decreasing heat transfer and increasing RC pressure. However, as the various steam reliefs open, the flow increases and the NB level increases promoting more SG heat transfer. When the RCS trips at 2355 psig, the steam line is already at or very close to the setpoint for the first bank of main steam safety valves. When the MSSVs open, the steam flow rate starts increasing sharply. This increased steam flow increases the NB level and quickly turns RC pressure around. The same transient with a 2300 psig trip setpoint requires more post trip time to reach the MSSV opening pressure resulting in a longer time to turn the RC pressure around.

4.4. Statistical Evaluation of Data

The probability of opening the PORV was estimated using Monte Carlo simulation methods. This technique is described in reference 5. The procedure and assumptions used are discussed generally in section 4.4.1. The details are covered in section 4.4.3. The conclusion of this activity is that there is very little chance that the PORV will be called upon to actuate at any high pressure reactor trip setpoint below 2375 psig for the transients covered by this study. The original plan was to start with the trip setpoint at 2300 psig and proceed to 2355 psig in 5 psi increments. It was noted, however, that there would be no PORV actions generated in the lower pressure ranges. Therefore, in doing the study, it was necessary to turn the problem around in order to find situations in which the PORV would be actuated. This was done by varying the reactor high pressure trip setpoint from 2355 to 2375 psig.

4.4.1. Method

The Monte Carlo simulation accounts for the effects of errors on the pressure signals to the RPS and the control of the PORV. Instrumentation errors are a combination of signal processing errors associated with the individual RPS channels and set points

for the decision to trip the reactor and analogous errors associated with the PORV. In addition to these errors, the study accounts for the pressure overshoot phenomenon.

The model is straightforward. It is assumed that there is a single true system pressure observed at each of the four narrow range pressure sensors. This true value is processed by each of the RPS channels and is subject to instrumentation errors in individual channels. This information is used to determine when, in terms of true pressure, the RPS trips. Of the four RPS channels, two are available for NNI signal processing. (For the Oconee plants, the signal to the NNI is selected from a fifth (inactive) RPS channel. As the errors for this channel are the same as for the active channels, the analysis results are applicable.) Of these two potential NNI signals only one is used by the NNI for processing to the PORV bistable. This study compares the output of these two channels and conservatively selects the larger indicated pressure to activate the PORV.

4.4.2. Results

The results of the Monte Carlo simulation approach indicate a very low probability of opening the PORV. For a trip setpoint corresponding to a true system pressure of 2355 psig, one trial in 100,000 resulted in opening the PORV. The average margin for PORV actuation relative to true pressure was 66.6 psi, with a standard deviation of 13.9 psi. Additional runs at 2365 psi and 2375 psi are included in Figure 4.4-1 to exhibit behavior of the system. The results of these two cases show that 55 PORV openings occurred at the 2365 psi set point, and 1485 occurred at the 2375 psi setpoint.

4.4.3. Details of The Monte Carlo Analysis

The direct Monte Carlo procedure used in this study will be detailed in this section. A simple drawing of the process to be modelled is shown in Figure 4.4-3.

The objective of this study was to evaluate the likelihood of opening the PORV at various RPS setpoints larger than the current nominal value of 2300 psig. A computer program was designed to perform a large number of Monte Carlo experiments for 2355, 2365, and 2375 trip setpoints.

Each individual Monte Carlo trial incorporates the error contributions of the signals to the RPS and to the PORV. The largest error sources are the RPS, the NNI, and those associated with the high pressure trip and the PORV set points. Another significant contributor is the pressure overshoot, which is treated as a physical phenomenon by a probability model and not as an uncertainty. The various error contributions used for the analysis are summarized in Table 4-3.

The RPS contributes the major portion of the instrument errors. These errors include (1) the sensor/transmitter through the bistable with its internal setpoint, (2) the NNI string to the PORV and (3) an additional conservatively assumed error applied to both the RPS and PORV setpoints.

The RPS errors were taken from reference 6. This study determined a bias component and a random component for the sensor/transmitter through the RPS high pressure setpoint under the B&W recommended method for string error treatment for operating units. The bias contribution was treated as a constant 4.83 psi and was applied in the conservative direction. The random part of the RPS error was an assumed normally (Gaussian) distributed random variable with a zero mean and a standard deviation of 6.2 psi.

The NNI buffer amplifier and the NNI bistable errors are both treated as normal random variables with means equal to zero and standard deviations of 1.4 psi and 1.3 psi respectively. This represents an assumption regarding the type of distributions for these errors.

The conservatively assumed error associated with the PORV and the RPS trip set point is treated as a uniformly distributed random contribution in the range from zero to five psi. It inflates the pressure at which the reactor will trip and decreases the pressure at which the PORV will open. These errors are included for conservative contributions due to set point variations.

The pressure overshoot is a physical phenomenon which is associated with high pressure trips. The probability model used for the pressure overshoot is an exponential distribution truncated at 10 psi and 60 psi. This distribution is based on plant data (Table 4.2) and is represented in Figure 4.4-2. While uncertainty of pressure overshoot is not directly treated, the distribution parameter is conservatively estimated as shown in Figure 4.4-2.

Any case that predicted an overshoot less than 10 psi was conservatively redone and any case that predicted overshoots greater than 60 psi were truncated to a 60 psi value. This tended to generate higher overshoot cases than the actual distribution would predict.

The program treats the single true system pressure input to each of the four RPS channels with individual channel processing errors. The result is four individual values of sensed pressure. Each of these is identified with a channel. In each channel there is a bistable with an internal setpoint. Each of the channel set points is also treated with its own random setpoint error mentioned above. Trip logic is met if two or more of the sensed pressures exceed or equal the actuation values of RPS setpoint plus the 0 to 5 psi random portion. If this trip condition is not satisfied on the first pass through, the true pressure is incremented by two psi and another iteration is made on the trip logic. This continues until the trip criterion is met.

At this point in a particular trial, the reactor has tripped. The true pressure which yielded the trip was then incremented by the overshoot. This new value (original true pressure plus overshoot) was then subjected to the RPS error treatment. As noted previously, one pressure signal can be selected from either of two RPS channels for transmittal to the NNI PORV circuit. The Monte Carlo program was set to select the highest of the two RPS channels available to the NNI. Thus, the pressure signal selected for the NNI in a given trial was the highest value of (true pressure + overshoot) + RPS errors for the two channels available. This value of indicated pressure was then treated with NNI buffer amplifier errors and sent to the PORV bistable to determine if a PORV actuation was required. The comparison at the bistable incorporated the zero to minus five psi error previously mentioned.

The margins to PORV actuation for these cases were grouped in 5 psi ranges to provide estimates of the probability of opening, or being near to opening, the PORV. This data is shown in Figure 4.4-1.

Table 4-1. Plant RPS Settings for High Pressure Trip

	Rancho Seco	Oconee (1,2 & 3)	ANO-1	CR-3	DB-1
Trip setpoint (psig)	2290	2290	2298	2296	2285
Max expected string error at bistable during weekly checks (+/-psi)	2.4 (RMS)	1.55 (RMS)	2.0 (RMS)	2.96 (Algebraic	4.0 Sum)

Table 4-2. Plant Data for High Pressure Trips

DATE	UNIT	MAX RCS PRESSURE (PSIG)	OVERSHOOT (PSI)	AVERAGE DELTA PRESSURE (PSI/SEC)	INITIAL POWER	RECORDING DEVICE	COMMENTS
2/05/80	DB-1	2336	51	37.0	100	NR REACTIMETER	TURB GOVERNOR VALVES CLOSED
2/26/80	CR-3	2320	24	----	100	STRIP CHART	LOSS OF NNI-X
5/11/80	RS	2290	0	----	11	NR TAP PLOT	NI CAL WITH ROD IN MAN.
5/30/80	RS	2300	10	10.0	95	WR REACTIMETER	MANUAL RUNBACK
6/24/80	ANO-1	2310	12	----	100	NR STRIP CHART	LOAD REJ.,MAN RUNBACK ATTEMPT
8/12/80	CR-3	2300	4	----	53	NR STRIP CHART	MFW PUMP TRIP
8/22/80	ANO-1	<2300	<2	----	90	NR ALARMS	MFW PUMP TRIP
8/29/80	CR-3	2300	4	----	72	NR,15 SEC DATA	TURB GOVERNOR VALVE PROBLEMS
10/28/80	RS	<2300	<10	----	94	NR STRIP CHART	AIR FAILURE, MFW VALVES LOCKED UP
11/08/80	DB-1	2300	15	7.0	28	NR REACTIMETER	FW OSCILLATIONS
11/12/80	DB-1	2334	49	10.6	40	NR REACTIMETER	LOSS OF ICS INPUTS FROM RPS
12/05/80	O-2	2291	1	4.2	14	NR,TTM(1SEC)	MANUAL UNDERFEED
4/08/81	ANO-1	2300	2	6.0	98	NR	FAILED NI POWER SIGNAL
4/11/81	CR-3	2305	9	10.0	100	NR TAP PLOT	LOSS OF POWER TO NNI-Y
6/17/81	RS	2300	10	2.5	80	NR TAP PLOT	MFW PUMP TRIP
6/23/81	RS	2305	15	----	50	NR TAP PLOT	FW OSCILLATIONS
6/27/81	CR-3	2305	9	----	55	NR STRIP CHART	FW OSCILLATIONS
6/30/81	CR-3	2300	4	2.5	50	NR TAP PLOT	MFW CONTROL PROBLEMS
7/08/81	ANO-1	2300	2	----	100	NR TAP PLOT	MFWP TRIP DURING MAN RUNBACK
9/01/81	ANO-1	<2300	<2	----	75	NR TAP PLOT	FAULT IN T AVE SIGNAL
9/05/81	O-3	<2300	<10	4.5	30	WR,TTM(1SEC)	FW OSCILLATION
10/16/81	DB-1	<2300	<15	----	100	NR REACTIMETER	MANUAL UNDERFEED
1/10/82	CR-3	<2300	<4	----	100	NR STRIP CHART	MFW PUMP TRIP
1/19/82	RS	2336	46	15.0	100	NR A STRIPCHART	LOSS OF TOTAL RC FLOW SIGNAL
3/24/82	O-1	<2300	<10	4.5	51	NR,TTM(1SEC)	TURB & FW A CONTROL IN MAN.
5/21/82	O-1	<2300	<10	2.0	90	NR	DURING ROD DROP RUNBACK
6/20/82	CR-3	2300	4	21.0	82	NR STRIPCHART	INADVERTENT MSIV CLOSURE
7/05/82	CR-3	2300	4	10.0	81	NR STRIPCHART	LOAD REJECTION
7/15/82	CR-3	<2300	<4	5.0	100	NR PTR	FAILED RC FLOW INPUT
8/08/82	CR-3	2300	4	----	94	NR STRIP CHART	LOSS OF RC FLOW SIGNAL
9/11/82	O-1	2301	11	----	42	NR, TTM	FW TRANSIENT
11/04/82	O-2	<2300	<10	1.5	100	NR, TTM(1SEC)	MFW PUMP TRIP
3/19/83	CR-3	<2300	<4	----	20	NR STRIP CHART	FW PUMP TRIP
5/10/83	DB-1	<2300	<15	4.0	90	WR B	ICS UPSET
7/26/83	CR-3	<2300	<4	5.0	16	NR STRIP CHART	LOSS OF FW FLOW INPUT TO ICS
	CR-3	<2300	<4	1.0	9	NR STRIP CHART	FW FLOW REDUCTION
8/22/83	CR-3	<2300	<4	2.5	65	NR STRIP CHART	TURB GOVERNOR VALVES CLOSED
8/26/83	CR-3	<2300	<4	----	75	NR A	MFW PUMP PROBLEMS
8/31/83	ANO-1	2310	12	4.5	100	NR	FW PUMP TRIP DURING RUNBACK
11/12/83	CR-3	<2300	<4	5.0	75	NR A	FW BLOCK VALVE CLOSURE
12/03/83	RS	<2300	<10	4.0	65	NR B,STRIP CHART	FW UPSET, VALVES & PUMPS IN MAN.
2/16/84	O-3	<2300	<10	----	100	WR,TTM	FAILED RC FLOW XMITTER
5/12/84	O-1	2322	32	14.0	100	NR,TTM(1SEC)	FAILED T HOT IN ICS
1/22/85	O-1	2328	38	40.0	100	WR,TTM(1SEC)	TURB INTERCEPT VALVE CLOSURE
4/11/85	O-1	2293	3	3.8	17	NR,TTM(1SEC)	FW OSCILLATIONS
	O-1	2330	40	----	100	NR STRIP CHART	TURB INTERCEPT VALVE CLOSURE
4/26/85	O-2	2300	10	18.0	75	NR,TTM(1SEC)	

Table 4-3. Summary of Major Inputs to Monte Carlo Simulation Program

RANDOM ERRORS

Source	Probability Distribution (Model)	Distribution Parameters
Reactor Protection System	Normal	Average 4.83 psi Standard Deviation 6.2175 psi
Non-Nuclear Instrumentation -Buffer Amplifier	Normal	Average 0.0 psi Standard Deviation 1.4 psi
-Bistable	Normal	Average 0.0 psi Standard Deviation 1.3 psi
Setpoints RPS	Uniform	Range 0 to + 5 psi
NNI (PORV)	Uniform	Range -5 to 0 psi

PHYSICAL PHENOMENA

Pressure Overshoot	Exponential (Truncated $\leq$ 60 psi $\geq$ 10 psi)	16.4 psi

Figure 4.3-1. POWER TRAIN Vs. Plant Data
Turbine Trip Without ARTS
2300 psig Setpoint

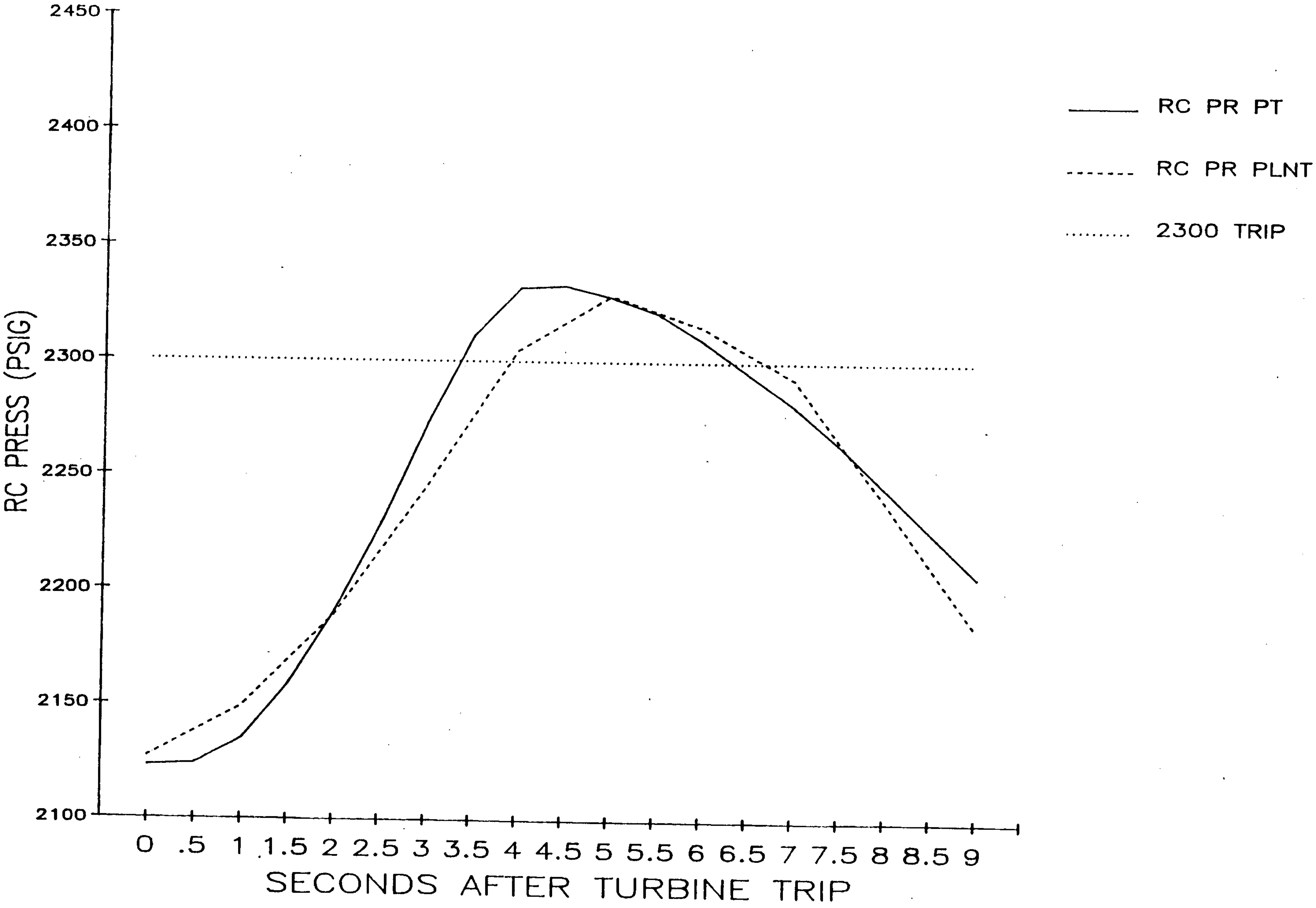


Figure 4.3-2. POWER TRAIN Results Turbine Trip Without ARTS

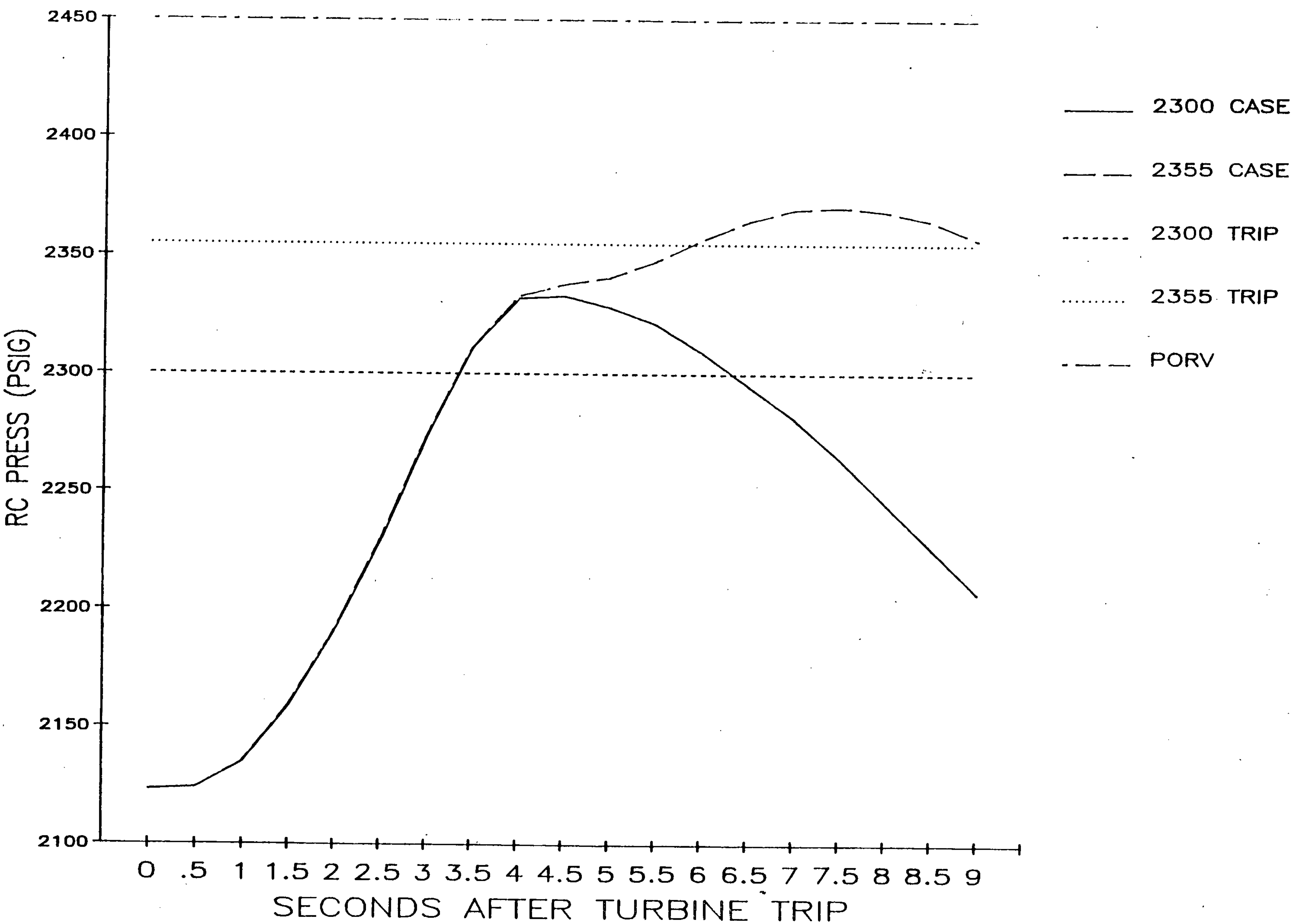


Figure 4.4-1. Monte Carlo Results

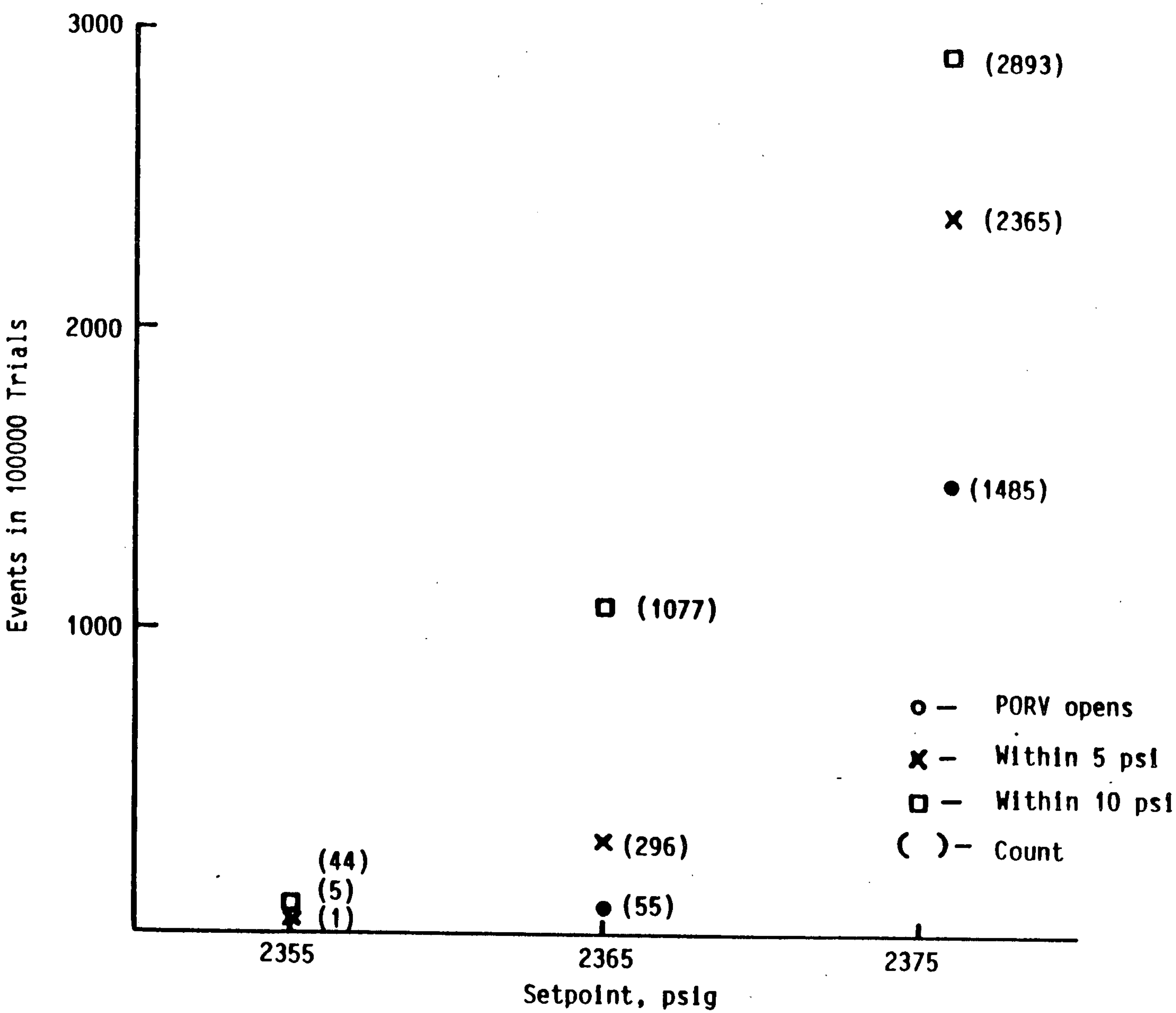
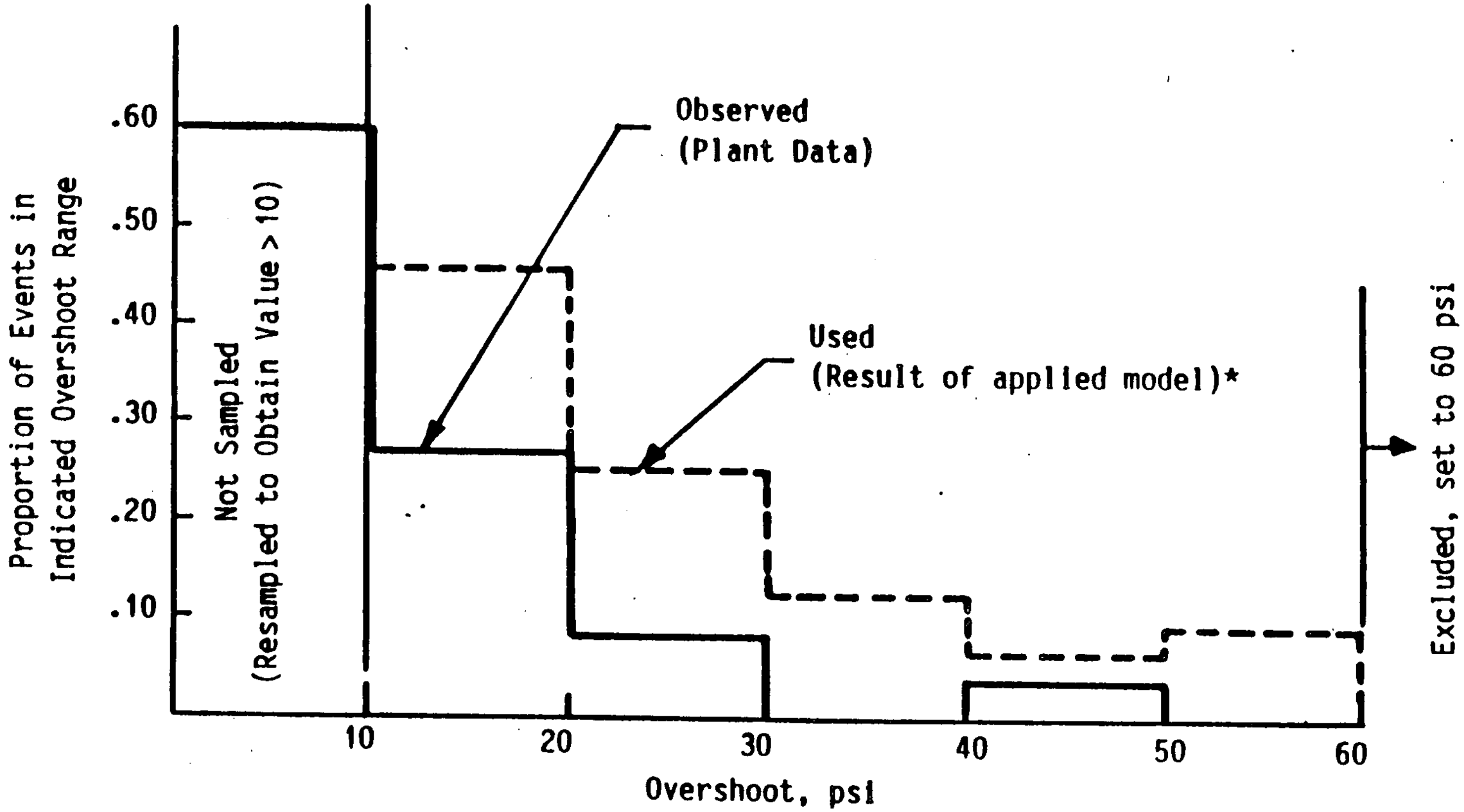
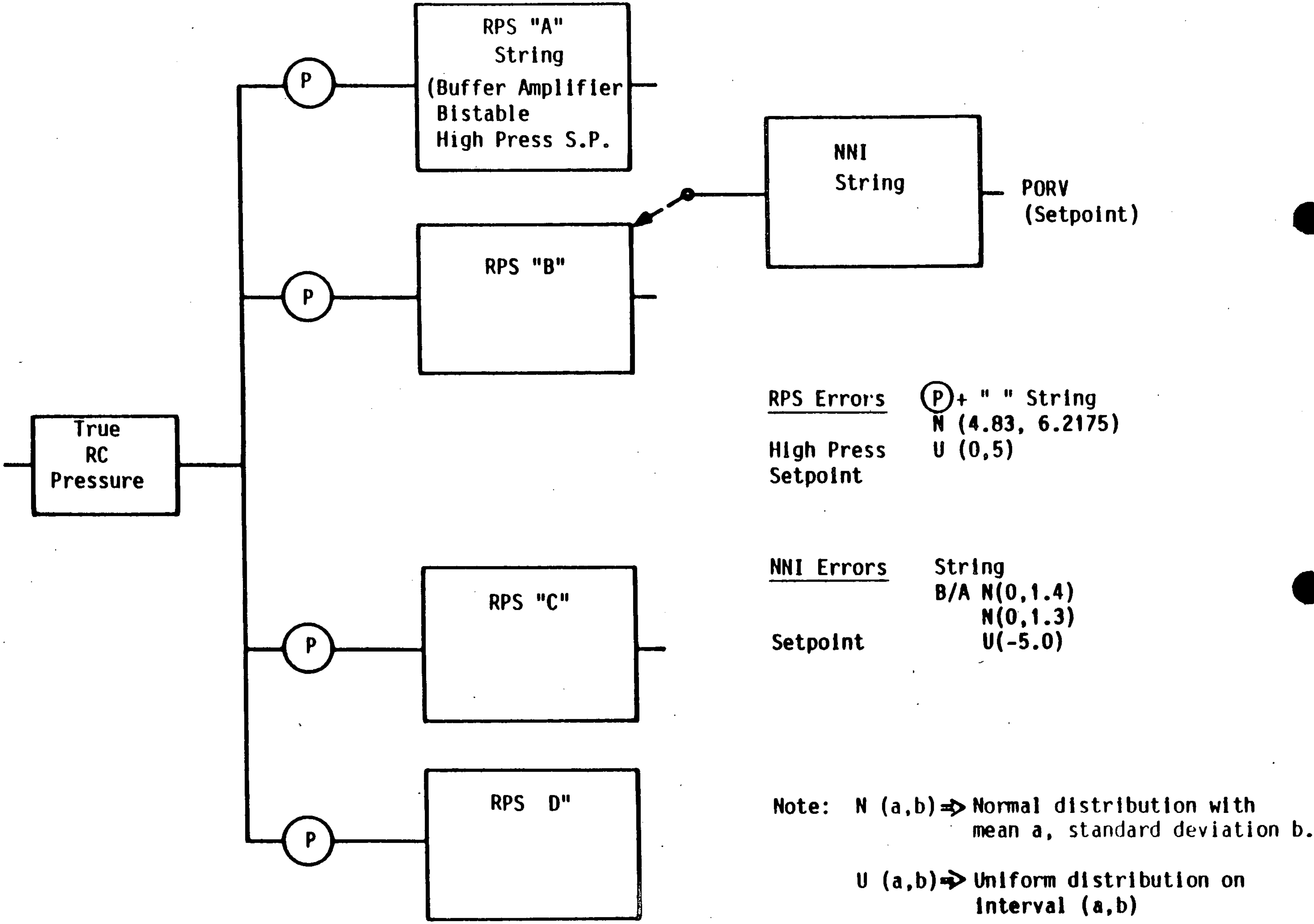


Figure 4.4-2. Overshoot Distribution for Monte Carlo Input



* Approximated

Figure 4.4-3. Model for Monte Carlo Analysis



5. NRC REQUIREMENTS/PREVIOUS ANALYSIS COMPARISONS

Shortly after the PORV and high pressure trip set points were reset in 1979 the NRC requested an analysis showing the probability of a small break LOCA at the PORV. A response was submitted in January 1981 (reference 2). This study included a statistical analysis of all the possible methods the PORV could open, one of which was the high pressure trip transient. The NRC evaluation of this submittal was subsequently documented in a safety evaluation based on a Technical Evaluation Report (TER), Reference 3. Table 3.1 summarizes the results of these previous studies. This section compares the results of this current study with the previous work and discusses both realtive to the NRC requirements on this subject.

5.1. Guidelines

Reference 4 describes the guidelines for the maximum allowable SBLOCA porbability at the PORV. The overall requirement is that this type of SBLOCA occur less than .001 times per reactor year. The specific requirement for the high pressure trip transient is that no more than 5% of them are allowed to open the PORV.

5.2. Current Study Results

The current study developed an estimated PORV opening frequency for overpressure transients of 1×10^{-5} openings per event. For the period 1980 through 1984, the seven B&W operating plants experienced 65 high pressure trip transientss. This is an average of 65/(7x5) = 1.86 events per reactor year. Thus, the estimated PORV opening frequency for overpressure transients is:

$$1.86 \frac{\text{Events}}{\text{Reactor year}} \times 10^{-5} \frac{\text{Openings}}{\text{Event}} = 1.86 \times 10^{-5} \frac{\text{Openings}}{\text{Reactor year}}$$

As stated previously, raising the high pressure trip setpoint affects only the PORV opening frequency for overpressure transients. Thus, the previous analyses (References 2 & 3) need only be compared for the impact of the overpressure transient contribution to PORV openings. Reference 3 estimated a PORV opening frequency for overpressure transients of 2.2×10^{-3}. Since the 2.2×10^{-3} PORV opening frequency yielded acceptable results, it follows that the 1.86×10^{-5} openings per reactor year frequency will also result in a probability of a small break LOCA due to a stuck open PORV of less than 1×10^{-3}.
The results of the current study are summarized as follows:

	NRC Guidelines	Current Study Results
Probability of SBLOCA due to Stuck-open PORV	$<1 \times 10^{-3}$	2.53×10^{-4}
PORV Openings on Overpressure Transients	$<5 \times 10^{-2}$	1×10^{-5}

5.3. Comparison of Current Previous Analyses

The previous analyses addressed all aspects of PORV opening frequency and probability of a SBLOCA due to a stuck open PORV. The current analysis focused on only the PORV opening frequency for overpressure transients since only this frequency is affected by a change in high pressure trip setpoint. The PORV opening frequency for overpressure transients developed in this study is considerably lower than that estimated in reference 3 (1.86×10^{-5} vs 2.2×10^{-3}). The reasons for this difference are discussed below.

The methodology and statistical components of the previous and the current analyses are similar with the exception of overshoot

predictions. The previous analysis was based on transient plant data where the PORV was open prior to the high pressure reactor trip. If the PORV had been closed, it was estimated that the pressure overshoot would be 17.4 psi higher. Therefore, without the PORV the mean pressure overshoot was calculatd to be 26.6 psi (9.2 + 17.4 from reference 2 page 4) on a normal distribution with a 27.52 standard deviation. The normal distribution was assumed due to the wide scatter of plant data used. On the other hand, the current analysis used plant data where the PORV did not open during the transient and consequently eliminated the need to estimate the effect of the PORV. Also, the data scatter in the overshoot prediction was reduced due to the methods discussed in section 4.2. This current data allowed for a more realistic distribution (see section 4.4) with a 16.4 mean overshoot. The effect of the inlet skewed overshoot distribution is reflected in the lower probability calculation of (1.86×10^{-5}) compared to 2.2×10^{-3} openings per reactor year of the 1981 results.

5.4. Summary

The previous analysis (Reference 3) calculated 2.2×10^{-4} PORV opening per high pressure event when the PORV was set 150 psi above the trip set point. The current analysis calculated 10^{-5} PORV opening per high pressure event when the PORV was 95 psi above the trip set point. Based on the last 5 years of operating experience and data, the current analysis is considered the more realistic of the two. However, both analyses meet the NRC SBLOCA requirements.

6. REFERENCES

1. BAW-1564, "Integrated Control System Reliability Analysis", August 1979.

2. 12-1122779 Rev. 1, "Report on PORV Opening Probability and Justification for Present System ans Setpoints," Jan. 1981.

3. TER-C5506-410, "Operating Reactor PORV Reports (F-37), Generic Report - Babcock Designed Units," July, 1983.

4. NUREG 0737, "Clarification of TMI Action Plan Requirements," Nov. 1980.

5. Hahn, G. J. and Shapiro, SS, "Statistical Models in Engineering, John Wiley & Sons, NY, Ny, 1967.

6. 32-1125056-02, "Statistical Errors - Rancho Seco," by H. T. Dass, NPD.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 4

SAFETY EVALUATION OF TOPICAL REPORT BAW-1890

APRIL 22, 1986

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 22, 1986

J. H. Taylor

APR 25 1986

Mr. J. H. Taylor, Manager, Licensing
Babcock & Wilcox Company
3315 Old Forest Road
Post Office Box 1260
Lynchburg, Virginia 24505-1260

Dear Mr. Taylor:

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT BAW-1890, "JUSTIFICATION FOR RAISING SETPOINT FOR REACTOR TRIP ON HIGH PRESSURE"

The Nuclear Regulatory Commission (NRC) staff has completed its review of the Babcock & Wilcox Licensing Topical Report BAW-1890 entitled, "Justification For Raising Setpoint For Reactor Trip On High Pressure," that was prepared for the B&W Owners Group. The report discusses the effect of the high pressure reactor trip setpoint on overpressure transients in B&W reactors. The report describes the impact of the setpoint on reactor trip frequency, the plant transient data, the analysis methodology, the NRC requirements that must be met, and the results that were obtained.

We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that B&W publish an accepted version of this report within three monthes of receipt of this letter. The accepted version shall incorporate this letter and the enclosed evaluation after the title page. The accepted version shall include an -A (designating accepted) following the report identification symbol.

CONTACT:
Daniel Fieno, RSB/DPL-B
x27742

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, B&W and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Assistant Director
for Technical Support
Division of PWR Licensing-B

Enclosure:
Topical Report Evaluation

cc: C. Rossi
G. Lainas

ENCLOSURE

SAFETY EVALUATION OF TOPICAL REPORT BAW-1890,

"JUSTIFICATION FOR RAISING SETPOINT FOR REACTOR TRIP ON HIGH PRESSURE"

TOPICAL REPORT EVALUATION

I. INTRODUCTION

This Babcock & Wilcox (B&W) report was submitted on behalf of the B&W Owners Group to justify increasing the high pressure trip setpoint from its current value of 2300 psig to 2355 psig. The current value of the 2300 psig high pressure trip setpoint was based on changes required by the staff (Ref. 1), subsequent to the TMI-2 accident, to reduce challenge to and opening of the power operated relief valve (PORV). Two other changes that are pertinent to this report were required: (1) raising the PORV setpoint from 2255 psig to 2450 psig and (2) implementation of a safety-grade automatic anticipatory reactor trip for, among other things, a turbine trip for power levels of 20 percent and higher. These modifications have met the NRC requirements that (1) the PORV will open less than 5% of the time for all anticipated over-pressure transients (Ref. 2, Item II.K.3.7) and (2) the probability of a small-break LOCA (SBLOCA), caused by a stuck-open PORV, is not a significant contributor to the probability of a small-break LOCA (Ref. 2, Item II.K.3.2) based on the WASH-1400 (Ref. 3) probability of a SBLOCA (Sequence S_2). Although these TMI required modifications have met the objectives of reducing challenges to and opening of the PORV during anticipated high pressure transients, they have increased the frequency of reactor trips. Each reactor trip results in a challenge to plant safety systems. Appropriate reductions in reactor trip frequency will contribute to overall plant safety as well as plant availability.

The report states that a number of high pressure transients would not have resulted in a reactor trip if more margin had been available to the high pressure trip setpoint. The report further states that the present analysis demonstrates that the NRC requirements would be met with the high pressure trip setpoint at 2355 psig rather than at 2300 psig. Moreover, if the anticipatory reactor trip (ART) on turbine trip setpoint is raised from 20% to 45% power, an additional reduction in reactor trip frequency would occur. The total reduction in reactor trip frequency is estimated to be about 10%. The B&W report (Ref. 4) on raising the ART setpoint power is the subject of a separate staff evaluation.

The report discusses the post-TMI high pressure reactor trip data base and the impact on the reactor trip frequency. A discussion is provided of the analysis methodology. The results of the present study are compared to previous results and are demonstrated to meet NRC requirements.

The staff evaluation of this licensing topical report follows.

8604250298XA

II. EVALUATION

A. Impact of Previous and Proposed Post-TMI Changes

B&W compared the average high pressure trip frequency for its plants in the pre-1979 and post-1979 periods. B&W found that the average trip frequency for its plants remained about the same although individual plant data varied. However, B&W notes that ARTs in the post-1979 period are, in effect, anticipatory high pressure trips and should be included in the post-1979 data base. When these trips are included in the data base, the post-1979 high pressure reactor trip frequency is about double the pre-1979 frequency. None of the plant data presented in the report reached the PORV pressure setpoint thereby demonstrating the efficacy of the post-TMI modifications to the PORV and high pressure setpoints and the ART in preventing the PORV from opening. This analysis of plant data on high pressure trip frequency is acceptable and demonstrates the increased reactor trip frequency caused by the TMI modifications.

B&W evaluated the potential for reactor trip frequency reduction for (1) an increase in the high pressure reactor trip setpoint by 55 psi back to the original FSAR value of 2355 psig and (2) an increase in the power level threshold for the turbine trip ART from 20% to 45% (Ref. 4). The first change would provide more margin to the high pressure reactor trip setpoint and would allow some minor plant upsets to either avoid reactor trip or provide the operator sufficient time to perform an action which would not result in a reactor trip. The second change in conjunction with an increased high pressure reactor trip setpoint, would not require a reactor trip for additional low power turbine trips. This second change will be the subject of a separate staff evaluation. The analysis of potential reactor trip frequency reduction demonstrates, from the data, that a number of high pressure and anticipatory reactor trips could be avoided. That is, a potential 10% reduction in reactor trip frequency may be possible.

B. Staff Reviews of NUREG-0737 Requirements on the PORV

Raising the high pressure reactor trip setpoint may reduce the frequency of reactor trips but NRC imposed post-TMI requirements on the PORV must still be met. The report contains a new analysis which is the main subject of this review, to demonstrate that these requirements are met. A report (Ref. 5) had previously been provided by B&W in response to Item II.K.3.2 of Reference 2 that demonstrated that a stuck-open PORV, with a high pressure trip setpoint of 2300 psig and a PORV setpoint of 2450 psig, would not be a significant contributor to a SBLOCA (Sequency S_2). This report was reviewed by a staff consultant, Franklin Research Center (A Division of the Franklin Institute), who submitted an evaluation (Ref. 6) which concluded that the B&W licensees met the requirements of Item II.K.3.2. The staff issued its own safety evaluation report (Ref. 7) concluding that: "We have

determined that the requirements of NUREG-0737, Item II.K.3.2 are met with the existing PORV, SV, and high pressure reactor trip setpoints..." This staff safety evaluation report trip implies, in addition, that the requirement of NUREG-0737, Item II.K.3.7, with regard to the frequency of PORV opening per high pressure transient, is met.

C. Method of Analysis of Effect of Proposed High Pressure Reactor Trip Setpoint on PORV Openings

The report presents analyses to demonstrate that the proposed high pressure reactor trip setpoint will meet the NRC requirements on PORV openings during high pressure transients. Those transients with excessive HPI or total loss of main and auxiliary feedwater are not considered since they could result in the PORV opening regardless of the high pressure reactor trip setpoint. The report reviews the actual high pressure reactor trip setpoints and the allowance made for instrument drifts and uncertainties. This error was assumed to vary from 0 to +5 psi in the Monte Carlo simulation to be discussed later. The determination of the error to be applied to the analysis of PORV openings is, therefore, acceptable since the error increases the high pressure reactor trip setpoint (i.e., less reactor pressure overshoot would be required to open the PORV).

The amount of pressure overshoot (i.e., the maximum reactor pressure minus the high pressure reactor trip setpoint) that occurs during a high pressure transient is a function of the heat transfer rates between the primary and secondary systems. The maximum reactor pressure is dependent on the pressurization rate prior to reactor trip and the time after trip when the reactor power is decreasing sufficiently. Some 47 plant transients were examined to determine the actual pressure overshoots that occurred. Although instrument string errors downstream of the Reactor Protection System (RPS), uncertainties due to print out device readability, and uncertainties due to data recording frequencies are included in the data, the indicated maximum pressure minus the high pressure reactor trip setpoint was conservatively assumed to be entirely due to pressure overshoot. The various errors will, however, be included in the Monte Carlo simulation to be discussed later. These errors are, therefore, counted twice in the analysis. The 47 transients indicated that the three most important categories of high pressure trip events are: (1) total or partial loss of feedwater, (2) feedwater/power mismatches during turbine runbacks, and (3) load rejections/MSIV closures. The pressurization rates for these transients varied from about 2 to 40 psi/sec with a corrsponding time to maximum reactor pressure varying from about 2 minutes to about 5 seconds. Our review of the information and data presented indicates that the pressure overshoot distribution that was obtained is acceptable since (1) a sufficient number and range of applicable transients were evaluated, (2) a conservative determination of the overpressure was made, and (3) the capabilities of the recording devices were taken into account.

Since the overshoot distribution was obtained from transients with a 2300 psig high pressure reactor trip setpoint, analyses were performed with the POWERTRAIN (Ref.8) program to determine if the distribution would be valid at the 2355 psig setpoint. POWERTRAIN has been reviewed and approved by the staff (approval letter dated November 28, 1983). A turbine trip from full power with no anticipatory reactor trip was selected for study since it would cause the largest pressure overshoot. The results indicated that POWERTRAIN was in agreement with plant data obtained at the 2300 psig high pressure reactor trip results. Analyses at the higher setpoint of 2355 psig indicated that pressure overshoot is a weak function of the high pressure reactor trip setpoint. In fact, the overshoot actually decreases as the setpoint is raised because of the complex behavior of the nucleate boiling region in the steam generators. The over-pressure distribution from plant high pressure reactor trips at the 2300 psig setpoint is conservative and is, therefore, acceptable when used at the higher setpoint in the Monte Carlo simulation to be discussed below.

The report describes the Monte Carlo analysis used to stochastically simulate the response of the four channels of the RPS and the control instrumentation for the PORV on the receipt of a pressure signal. The major sources of uncertainty included in the simulation are the uncertainties in the RPS and the NN1 signal processing and the uncertainties in the high pressure trip and PORV setpoints. The NNI channel provides the signal for opening the PORV. The high pressure trip uncertainty is taken to be a uniform distribution from 0 to +5 psi while the PORV setpoint uncertainty is taken to be a uniform distribution from 0 to -5 psi. The pressure overshoot results obtained from the plant high pressure reactor trip data is treated as a physical phenomenon having an exponential distribution. This distribution is truncated between 10 psi and 60 psi. Cases in the Monte Carlo analyses that gave overshoots less than 10 psi were set to 10 psi and cases that gave overshoots greater than 60 psi were set to 60 psi. This resulted in a conservative representation of the distribution derived from the 47 plant transients, as the pressure overshoot in these transients was always less than 60 psi.

A successful Monte Carlo simulation resulted when 2 out of 4 RPS channels trip on the assumed high pressure trip setpoint. The pressure, chosen as the highest value from the 2 of 4 channels that caused the trip, is next incremented with the pressure overshoot chosen from the exponential distribution. This pressure is then processed by the Monte Carlo program using the NNI channel to determine if the PORV setpoint has been reached. This Monte Carlo process is repeated until a sufficient number of high pressure trip events have been accumulated to provide adequate statistics for the specified high pressure trip setpoint. Based on our review, we conclude that the treatment of the uncertainties used and their distribution, the treatment of the pressure overshoot distribution, and the Monte Carlo simulation process itself are conventional and appropriate and are, therefore, acceptable.

D. Comparison of Results for PORV Opening with NRC Requirements

The Monte Carlo simulation indicated that there would be one PORV opening per 100,000 high pressure trips at the proposed high pressure reactor trip setpoint of 2355 psig. This frequency of 0.00001 is much less than the NRC requirement of less than 0.05 PORV openings per overpressure transient events that required a reactor trip. Therefore, Item II.K.3.7 of NUREG-0737 remains satisfied.

The report states that there were 65 high pressure trips from 1980 through 1984 for the 7 operating B&W reactors. This yields an average of 65/35 or 1.86 events per reactor year. Thus, the probability of a PORV opening per reactor year is given by:

$$1.86 \ \frac{\text{events}}{\text{reactor-year}} * 1.0 \times 10^{-5} \ \frac{\text{PORV openings}}{\text{event}} = 1.86 \times 10^{-5} \ \frac{\text{PORV openings}}{\text{reactor-year}}$$

The PORV opening frequency from all other causes is 8.06×10^{-2} (Ref. 6). Therefore, the total PORV opening frequency at the proposed setpoint of 2355 psig is :

$$8.06 \times 10^{-2} + 1.86 \times 10^{-5} = 8.06 \times 10^{-2} \ \frac{\text{total PORV openings}}{\text{reactor year}}$$

The total PORV openings per reactor year is negligibly changed over the values presented in References 5 and 6 since operator actions under ATOG (abnormal transient operating guidelines) and, to a lesser degree, instrumentation and control faults dominate the total PORV opening frequency. Using the Reference 7 value of 2×10^{-2} failures per demand for the PORV failure probability gives:

$$\frac{\text{PORV failures}}{\text{reactor-year}} = 8.06 \times 10^{-2} \ \frac{\text{PORV openings}}{\text{reactor-year}} * 2 \times 10^{-2} \ \frac{\text{failures}}{\text{demand}}$$

$$= 1.6 \times 10^{-3}$$

Since the probability of a SBLOCA (Sequence S_2) caused by a stuck-open PORV is within the WASH-1400 (Ref. 3) range of 10^{-2} to 10^{-4} per reactor-year, the requirements of Item II.K.3.2 of NUREG-0737 remains satisfied. This is as expected since the PORV opening frequency due to over pressure reactor events that cause a high pressure trip is neglibibly affected by the proposed high pressure reactor trip setpoint of 2355 psig.

E. Comparison of Present Analysis to Previous Analysis

The report states that the main difference between the present analysis and the previous analysis was in the treatment of the pressure over-shoot. The analysis methodology and other statistical components are similar. In the previous analysis the overpressure had to be based on plant data where the PORV opened. This led to a large uncertainty in

the actual pressure overshoot determination. This was reflected in the use of a normal distribution with a large standard deviation (27.5 psi) to accommodate the wide scatter in the data. The present analysis uses plant data for transients for which the PORV did not open. It is believed that the present analysis has a more realistic assessment of the actual overpressure that occurs for the high pressure transients considered in this report. The staff concurs with this assessment of the differences between the present and previous analyses.

III. CONCLUSION

The staff has reviewed the Babcock & Wilcox licensing topical report on the high pressure reactor trip setpoint and concludes that it is acceptable to increase the high pressure reactor trip setpoint for B&W plants from 2300 psig to 2355 psig while the PORV setpoint remains at 2450 psig. The staff concludes that this setpoint change meets the NRC requirements of NUREG-0737, Items II.K.3.2 and II.K.3.7 regarding PORV openings and PORV caused SBLOCA. Similarly, the requirements on this matter embodied in IE Bulletin 79-05B are also met.

Accordingly, the staff concludes that the licensing topical report may be referenced in licensing submittals by the B&W Owners Group members.

Since this report, of necessity, must use analyses based on a statistical approach, uncertainties are inherent in the results obtained. Additional uncertainty in the results are caused by the modeling, the assumptions made, and the data that are used. Therefore, as plant experience is accumulated with the proposed high pressure reactor trip setpoint, the staff should be kept informed of any significant deviation from the assumptions and results presented in the report.

IV. REFERENCES

1. "Nuclear Incident at Three Mile Island - Supplement," IE Bulletin 79-05B, April 21, 1979.

2. "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

3. "Reactor Safety Study - An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants," WASH-1400, 1975.

4. "Basis For Raising Arming Threshold For Anticipatory Reactor Trip on Turbine Trip," BAW-1893, October 1985.

5. "Report on PORV Opening Probability and Justification for Present Systems and Setpoints," 12-1122779 Rev. 1, Babcock & Wilcox report, January 1981.

6. "Operating Reactor PORV Reports (F-37), Generic Report - Babcock & Wilcox Designed Units," Franklin Research Center, July 20, 1983.

7. NRC Memorandum from F. H. Rowsome to G. C. Lainas dated August 24, 1983; entitled "Safety Evaluation of the B&W Licensees' Responses to TMI Action Item II.K.3.2."

8. "POWERTRAIN: Hybrid Computer Simulation of a Babcock & Wilcox Nuclear Power Plant," N.S. Yee and J. A. Weimer, BAW-10149, Rev. 1, November 1981.