DnB 016

May 30, 1986

Docket No. 50-302

Mr. Walter S. Wilgus
Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing
& Fuel Management
P. O. Box 14042; M.A.C. H-3
St. Petersburg, Florida 33733

Dear Mr. Wilgus:

SUBJECT: ACCEPTANCE OF BABCOCK & WILCOX TOPICAL REPORTS BAW-1890 AND BAW-1893 FOR REFERENCE IN LICENSE APPLICATIONS

By letters dated April 22 and April 25, 1986, the staff informed Babcock & Wilcox Company of the acceptability of the following Topical Reports for referencing in license applications:

1. BAW-1890, "Justification for Raising Setpoint for Reactor Trip on High Pressure", and

2. BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip".

Copies of the staff letters are enclosed for your information.

Original signed by

Harley Silver, Project Manager
PWR Project Directorate #6
Division of PWR Licensing-B

Enclosure: As Stated

cc w/enclosure: See Next Page

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Mr. W. S. Wilgus
Florida Power Corporation

Crystal River Unit No. 3 Nuclear
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 22, 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

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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 overpressure 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.

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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 NNI 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 overshoot. 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.

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

April 25, 1986

Mr. J. H. Taylor, Manager, Lic nsing
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-1893, "BASIS FOR RAISING ARMING THRESHOLD FOR ANTICIPATORY REACTOR TRIP ON TURBINE TRIP"

The Nuclear Regulatory Commission (NRC) staff has completed its review of the Babcock & Wilcox Licensing Topical Report BAW-1893 entitled, "Basis For Raising Arming Threshold For Anticipatory Reactor Trip On Turbine Trip," that was prepared for the B&W Owners Group. The report discusses the effect of the power threshold for the anticipatory reactor trip (ART) on turbine trips and power runbacks in B&W reactors. The report describes the impact of the turbine trip ART power level threshold on reactor trip frequency, the plant transient data, the analysis methodology, and the results that were obtained.

The staff finds 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.

The staff does not intend to repeat its 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. The staff's 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 months 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.

Should the staff's criteria or regulations change such that its conclusions as to the acceptability of the report are invalidated, B&W and/or the applicants

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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:
Safety Evaluation

cc: C. Rossi
G. Lainas

ENCLOSURE

SAFETY EVALUATION OF TOPICAL REPORT BAW-1893,
"BASIS FOR RAISING ARMING THRESHOLD FOR ANTICIPATORY
TRIP ON TURBINE TRIP"

I. INTRODUCTION

This Babcock & Wilcox (B&W) report was submitted on behalf of the B&W Owners Group to justify increasing the anticipatory reactor trip (ART) setpoint on turbine trip from its current value of 20% power to 45% power. The current value of the 20% power ART setpoint on turbine trip was based on changes required by the staff (Ref. 1) subsequent to the TMI accident to reduce challenges 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) lowering the high pressure reactor trip setpoint from 2355 psig to 2300 psig. These modifications have met the NRC requirements that (1) the PORV will open less than 5% of the time for all anticipated overpressure transients (Ref. 2, Item II.K.3.7) and (2) the probability of a small-break LOCA (SBLOCA), caused by a stuck-open PORV, will be less than 0.001 per reactor-year (Ref. 2, Item II.K.3.2) which is based on the WASH-1400 (Ref. 3) median 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 and any reduction in reactor trip frequency will contribute to overall plant safety as well as plant availability.

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The report states that a number of turbine trips would not have resulted in a reactor trip if more margin had been available in the ART power level setpoint. The report further states that the present analysis demonstrates that the NRC requirements would be met with the ART power level setpoint at 45% power rather than at 20% power. In fact, the report states that these requirements on PORV openings would be met regardless of whether or not ART is implemented. Moreover, if the high pressure reactor trip setpoint is increased from 2300 psig to 2355 psig, an additional reduction in reactor trip frequency would be posssible. The total reduction in reactor trip frequency is estimated to be about 10%. The B&W report (Ref. 4) on raising the high pressure reactor trip setpoint has been evaluated by the staff (Ref. 5). This staff safety evaluation report concluded that it was acceptable to raise this high pressure reactor trip setpoint from 2300 psig to 2355 psig. This increased high pressure reactor trip setpoint is assumed in the analyses performed in support of raising the turbine trip ART power level threshold to 45%.

The report discusses the post-TMI turbine trip/reactor trip data base and the impact on the reactor trip frequency. A discussion is presented of the analysis methodology. The results of the present study are used to justify the turbine trip ART proposed threshold power level of 45%.

The staff evaluation of this licensing topical report follows.

II. EVALUATION

A. Impact of Previous and Proposed Turbine Trip ART Power Threshold

The report discusses the response of B&W plants to turbine trips. Prior to the TMI accident, a turbine trip caused a reactor power runback. For some plants successful runbacks were demonstrated for power levels as high as 100%. However, these runbacks were dependent, to some degree, on the PORV opening. Since the TMI accident, the turbine trip ART, among other changes, was instituted to reduce challenges to the PORV. This turbine trip ART now results in a reactor trip whenever the turbine trips and the reactor power level is 20% or higher. Although the NRC requirements on PORV challenges are met by the various post-TMI changes, an undesired side-effect of increased frequency of reactor trip and consequent challenges to the plant safety system has occurred. The data presented in the report show that 52 turbine trips occurred in the period from January 1, 1980 to January 1, 1985. Twelve of these trips occurred between power levels of 20% to 40%. Raising the turbine trip ART power level threshold has the potential for reducing the reactor trip frequency without affecting PORV opening frequency. Based on our review, we concur with the applicant that the analysis of plant data on reactor trips caused by turbine trips demonstrates that reactor trip frequency increased as a result of TMI modifications.

B&W evaluated the potential for reactor trip reduction for (1) increasing the high pressure reactor trip setpoint by 55 psi back to the original FSAR value of 2355 psig (Ref. 4) and (2) increasing the power level threshold for the turbine trip ART from 20% to 45%. The first change would provide more margin to the 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 some additional low power turbine trips. We find that the analysis of potential reactor trip frequency reduction is reasonable and demonstrates from the data in the report and Reference 4 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 posssible.

B. Results of Analysis of Turbine Trip ART

The POWERTRAIN (Ref. 6) program was used by B&W to evaluate the factors which are important in power runback on turbine trips without a reactor trip. These factors lead to the determination of the highest initial power level or threshold for the turbine trip ART. Factors evaluated included (1) the total bypass steam flow, (2) the moderator temperature coefficient, (3) the initial power level, (4) the power runback rate, and (5) the pressurizer spray flow rate. The cases evaluated were turbine trips with runbacks modeled with a reactor closely resembling Rancho Seco. A successful runback case was defined by B&W to have the following desirable performance characteristics: (1) no reactor trip on high reactor system pressure, (2) no auxiliary feedwater actuation on low steam generator level, (3) no steam generator overfill affecting steam quality, and (4) no loss of subcooled margin as affected by reactor system pressure and temperature. Since the modeling, assumptions, and criteria used in the analysis considers the principal factors in a turbine trip with runback, the staff concludes that the methodology used is, therefore, acceptable. In addition, since the POWERTRAIN program has been reviewed and approved by the staff (approval letter dated November 28, 1983) the staff concludes that its use is, therefore, acceptable.

From the POWERTRAIN analyses it was determined that the total steam bypass flow was one of the most important factors in determining whether or not a reactor power runback on turbine trip was successful. The total steam bypass flow included turbine bypass flow, atmospheric vent flow and flow through at least one bank of Main Steam Safety Valves (MSSV). At least one bank of MSSVs will open at the high pressure reactor trip setpoint of 2355 psig (Ref. 5). In the analyses, if the core power decreases because of control rod insertions and negative moderator temperature coefficient, to the total steam bypass flow before the high pressure reactor trip setpoint is reached, sufficient primary to secondary heat transfer exists to stop the reactor system pressure from increasing. These results presented show that the larger the total steam bypass flow the higher the power threshold that can be tolerated by the turbine trip ART.

The reactor coolant temperature and pressure increases during the early stages of a turbine trip. The moderator (and Doppler) reactivity coefficient are negative throughout a reactor cycle. These negative coefficients, therefore, help to reduce the reactor power and thus help the reactor power runback process caused by control rod insertion. POWERTRAIN results were obtained for near beginning-of-cycle (BOC) and end-of-cycle (EOC) cases which demonstrates this effect. Therefore, for the same total steam bypass flow and control rod insertion rate, successful reactor power runbacks are more probable the more negative the moderator temperature coefficient becomes.

The initial power level is a factor in determining a successful power runback along with the total steam bypass flow and moderator temperature coefficient. POWERTRAIN results established, as expected, that successful reactor power runbacks from higher initial reactor power would require higher total steam bypass flow. POWERTRAIN results were also obtained for two other factors. These were the Integrated Control

System (ICS) runback rate on control rod insertion and the pressurizer spray flow rates. The ICS runback rate was changed from 20% per minute to 50% per minute but this did not change the overall control rod insertion rate during the important early stages of a turbine trip transient where the moderator temperature coefficient is also important. Therefore, the indicated ICS runback rate had negligible influence on the reactor power runback on a turbine trip event. Similarly, the pressurizer spray rate was found to have very little effect in turning around the reactor coolant pressure in the time period of interest.

The conclusions of this POWERTRAIN analysis were that, for a given control rod reactivity insertion rate and high pressure reactor trip setpoint, the most important factors, in determining whether or not a reactor power runback, on turbine trip is successful, are the initial power level and the total steam bypass flow. The study also concluded that the negative moderator temperature coefficient helped the reactor power runback especially at EOC when it is more negative than say, for example, near BOC. The report concluded that other factors had negligible impact on reactor power runbacks. Since the results in the report were obtained with the approved POWERTRAIN program and since the principal effects were evaluated, the staff concludes that the POWERTRAIN results are, therefore, acceptable.

The report states that the results are applicable to all the B&W 177 fuel assembly (FA) plants. Based on the review of the analyses presented, we concur on the applicability of these results to the 177 FA B&W plants. The report concludes that, for the total steam bypass flow credited in the analysis, the reactor trip on turbine trip power level threshold could be increased from 20% to 45% with a high pressure reactor trip setpoint of 2355 psig. Based on the review of the plant data presented in the report and the POWERTRAIN results, the staff concludes that the B&W assessment regarding the raising of the turbine trip ART power level threshold to 45% is, therefore, acceptable.

C. Effect of Turbine Trip ART Proposed Power Level Threshold On PORV Openings and NRC Requirements

Although the results presented in the report are applicable to all B&W 177 FA plants, differences in a number of plant parameters may not lead to successful reactor power runbacks on turbine trips with a turbine trip ART power level threshold of 45% and a high pressure reactor trip setpoint of 2355 psig. An unsuccessful power runback will lead to a high pressure trip. Therefore, it is essential to evaluate the effect of these potential additional high pressure trips on the frequency of PORV openings and to determine whether or not NRC requirements on PORV openings are met.

The report assumes that 30% of the reactor power runbacks will be unsuccessful. Assuming the same turbine trip frequency at power levels equal to or below 45% as occurred in the post-TMI period, the report finds the following:

$$\frac{12 \text{ (turbine trips)} * .30 \text{ (reactor trip/turbine trip)}}{5 \text{(years)} * 7 \text{ (reactors)}}$$

$$= 0.10 \ \frac{\text{high pressure trips}}{\text{reactor year}}$$

Then the high pressure trip frequency would increase from 1.86 per reactor-year (Ref. 4) to (1.86 + .10) or 1.96 per reactor-year. The number of PORV openings from high pressure trip events would now be:

$$\frac{1.96 \text{ events}}{\text{year}} * 1.0 \times 10^{-5} \ \frac{\text{PORV opens}}{\text{event}} = 1.96 \times 10^{-5} \ \frac{\text{PORV openings}}{\text{year}}$$

The total number of PORV openings per reactor-year for all events, as given in Reference 4, is 8.06×10^{-2} and is negligibly affected by this change. The results of Reference 4 on PORV openings and the probability of a SBLOCA (Sequence S_2) remain applicable. Therefore, the staff concludes that the requirements of Item II.K.3.2 and Item II.K.3.7 of NUREG-0737 (Ref. 2) are met even if a number of reactor power runbacks are unsuccessful at the proposed turbine trip ART power threshold of 45%.

III. CONCLUSION

The staff has reviewed the Babcock & Wilcox licensing topical report on the turbine trip ART power level threshold and concludes that it is acceptable to increase the turbine trip ART power level threshold for B&W plants from 20% to 45%. The staff concludes that this power level threshold 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 while benefitting plants by potentially reducing the reactor trip frequency. 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.

Due to the modeling, assumptions made, and data used, the results presented in the report, as is the case for any analysis, may contain uncertainties. Therefore, as plant experience is accumulated with the proposed turbine trip ART power threshold, the staff should be kept informed of any significant deviations from the 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 Plant," WASH-1400, 1975.

4. "Justification For Raising Setpoint For Reactor Trips on High Pressure," BAW-1890, September 1985.

5. Letter from D. M. Crutchfield (NRC) to J. H. Taylor (B&W) on "Acceptance For Referencing of Licensing Topical Report BAW-1890, 'Justification For Raising Setpoint For Reactor Trip on High Pressure,'" April 1986.

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