



Washington Public Power Supply System
A JOINT OPERATING AGENCY

P. O. Box 988 3000 GEO. WASHINGTON WAY RICHLAND, WASHINGTON 99352 PHONE (509) 375-5000

Docket Nos: 50-460
50-513

June 2, 1980
G01-80-171

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

THIS DOCUMENT CONTAINS
POOR QUALITY PAGES

Dear Mr. Denton:

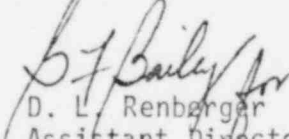
Subject: WPPSS Nuclear Projects Nos. 1 & 4
Response to NRC Supplemental 10 CFR 50.54 Request

Reference: Letter, H. R. Denton, NRC to N. O. Strand, WPPSS,
"Supplemental 10 CFR 50.54 Requests Regarding B&W
System Sensitivity for Washington Public Power
Supply System Nuclear Projects 1 & 4 (WNP-1/4),"
March 25, 1980.

In the reference letter the NRC requested that WPPSS provide additional
information regarding the changes and studies proposed in our December 3,
1979 response to the initial 10 CFR 50.54 letter.

Our response to this supplemental request is attached.

Very truly yours,


D. L. Renbarger
Assistant Director - Technology

DLR:AGH:oe

Attachment

cc: A. Bournia, NRC
T. Novak, NRC
N. S. Reynolds, D&L
C. R. Bryant, BPA
Eng. Files-1/4 (290)
G. C. Schieck, B&W
H. Kwan, UE&C-PA
A. J. Friedman, UE&C-PA

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STATE OF WASHINGTON)
) ss
COUNTY OF BENTON)

G. F. BAILEY, Being first duly sworn, deposes and says: That he is the Manager, Technical Division, for the WASHINGTON PUBLIC POWER SUPPLY SYSTEM, the applicant herein; that he is authorized to submit the foregoing on behalf of said applicant; that he has read the foregoing and knows the contents thereof; and believes the same to be true to the best of his knowledge.

DATED June 2, 1980

G. F. BAILEY
G. F. Bailey

On this day personally appeared before me G. F. BAILEY to me known to be the individual who executed the foregoing instrument and acknowledged that he signed the same as his free act and deed for the uses and purposes therein mentioned.

GIVEN under my hand and seal this 2nd day of June, 1980

Richard A. Riddle
Notary Public in and for the State
of Washington
Residing at Richland

ENCLOSURE

RESPONSE TO NPC REQUEST FOR ADDITIONAL INFORMATION
REGARDING WNP-1/4 SYSTEM SENSITIVITY

F.1 Question

Your discussion in Appendix F of the pre-TMI changes for WNP-1/4 states that newer control systems hardware (non-nuclear instrumentation (NNI)/integrated control system (ICS)) using dual auctioneer power supplies for logic modules rather than individual power supplies are being used.

- a. For this modification, provide the logic and/or your failure mode and effects analysis that shows how systems will respond to failure in the power supply and input parameters. Also provide your design criteria for the NNI and ICS with respect to these types of failures.
- b. Operating events at several plants with B&W NSSS designs (including Rancho Seco in March 1978; Oconee Power Station, Unit 3 on November 10, 1979; and the Crystal River Station on February 26, 1980) have occurred which resulted in loss of power to the ICS and/or NNI system. The loss of power resulted in control system malfunctions, feedwater perturbations, and significant loss of or confused information to the Operator. NUREG-0600 also discusses LER 78-021-03L on Three Mile Island, Unit 2 whereby the RCS depressurized and safety injection occurred on loss of a vital bus due to inverter failure. Discuss the extent to which these events would have been mitigated or precluded by the changes incorporated into the WNP-1/4 design. Include a response to action items 1 to 3 required of near-term licensees in Bulletin 79-27 and items 2, 4, 5 and 6 of Enclosure 3 of letter dated March 6, 1980 to all operating B&W Reactor Licensees pertaining to the Crystal River event.

Response

F.1a

The ICS and NNI are each supplied by a single independent 120 VAC source. Each 120 VAC source is input to redundant 24 VDC supplies. The 24 VDC supplies are auctioneered within each subsystem. Either of the redundant 24 VDC supplies is capable of supplying all cabinet modules and external instrumentation utilizing 24 VDC power. Within each system (NNI and ICS) the 120 VAC input source is used for the required 120 VAC remote mounted instrumentation and control. The NNI and ICS designs meet the following requirements with respect to power supply failures:

1. Failure of an NNI power supply shall not cause the PORV to open nor shall it prevent the PORV isolation valve from functioning.

2. Failure of NNI/ICS power supplies shall not prevent safety or protection systems from operating or prevent manual override of safety or protection system.
3. Failure of an NNI power supply shall not cause the spray valve to open nor shall it prevent the spray block valve from functioning.
4. Loss of an NNI or ICS power supply shall not cause the pressurizer heaters to fail on, or remain on, when the pressurizer level is low.
5. Upon loss of the NNI and/or ICS power supplies, the remaining plant instrumentation and controls shall be sufficient to place and maintain the plant in a safe hot shutdown condition.
6. Following loss of an NNI or ICS power supply, the capability of maintaining and/or restoring steam pressure in at least one steam generator shall be available.

F.1b

The extent to which the events described above would be mitigated or precluded by the WNP-1/4 design is presented below. The format of this discussion is in response to items 2, 4, 5 and 6 of the NRC March 6, 1980 letter.

2. The Crystal River event was caused by the loss of the +24VDC "X" bus which affected important shutdown indicators. The WNP-1/4 design is susceptible to the same kind of failure in the NNI system; however, it has an Essential Control and Instrumentation (ECI) system completely independent of the NNI. The ECI system consists of two duplicate cabinet assemblies (ECI-"X" and ECI-"Y") having redundant indications and control capable of maintaining the plant in a safe hot shutdown condition. The two ECI cabinet assemblies are powered from separate vital sources entirely separate from NNI system.
4. The WNP-1/4 instrumentation and control design is different from CR-3 in that the ICS and NNI for WNP-1/4 is a single system. Upon failure of ICS/NNI power the operator should consider all information invalid and rely upon the ECI for reliable indication.

Instrumentation provided to bring the plant to cold shutdown will be discussed in FSAR Subsection 7.4.1.4.

5. The ICS/NNI and ECI systems are separated such that a test of the various input power systems is feasible. Specifically, a loss of NNI/ICS power does not affect the ECI system and a loss of power to either ECI-X or ECI-Y does not affect the other redundant set of instrumentation indications and controls.

Applicability of each planned CR-3 action to the WNP 1/4 plant is as follows:

Intermediate

1. Determine failure cause in NNI

Not applicable.

2. PORV closure on NNI failure

The circuit design is such that the PORV will close on loss of power due to either NNI internal power supply (+ 24 V DC) failure or in general, to a complete loss of power to the NNI.

3. Pressurizer spray valve operation on NNI failure

Circuit design is such that the pressurizer spray valve will not open automatically on loss of power due to either an NNI internal power supply (+ 24 V DC) failure or in general, to a complete loss of power to the NNI.

4. PORV and relief valve indication

Positive relief valve indication will be provided in response to NUREG-0578.

5. Procedural control of NNI selector switches

Not applicable since the WNP-1/4 NNI has one power source. The ECI system is available in the event of the loss of the NNI system.

6. Operator training for NNI and ICS failures

All licensed operators and licensed operator candidates will receive training on the WNP-1 simulator. This will include training for a loss of power to the NNI and/or ICS systems. The training shall be targeted toward identifying and controlling overcooling transients and overpressure transients which result from the loss of power to the instrument circuits.

7. ICS power from vital bus

The ICS and NNI are powered from an uninterruptible bus (120 VAC distribution panel) that is supplied by a non-Class 1E inverter. No design change is required.

8. Event recorder system

A surveillance procedure for WNP-1/4 will be developed for a periodic functional check of the events recorder/annunciator system (PMIS).

9. Redundant indication

ECI system provides this function independent of NNI/ICS

At Next Refueling

1. Power indication lights

This concern will be addressed during our review of IE Bulletin 79-27 discussed below.

2. Fuse Access

The non-Class 1E, inverter backed, 120V AC distribution panel which provides power to the NNI and ICS is designed with hinged doors to facilitate quick access to the fuses. No design change is proposed.

3. AFW pump start on low S.G. level

This feature is included in WNP-1/4 design of ECI system.

Long Term

Upgrade of NNI

1. Any required upgrade for CR-3 will be evaluated for applicability to WNP-1/4.

2. Remote Shutdown

Provided in WNP-1/4 design by ECI system

3. Backup AC sources

The inverters which supply power to the NNI and ICS are provided with a static transfer switch which automatically transfers to a backup voltage regulated power source on inverter failure.

Regarding Items 1 to 3 of IE Bulletin 79-27:

Items 1 and 3 (Review of 1E and non 1E power to safety and non-safety instrumentation and control systems)

Our review of the power supplies to safety related and non-safety related instrumentation and control systems has just been initiated. This review will consider the Rancho Seco, Oconee and Crystal River events and IE Bulletin No. 79-27 and IE Circular No. 79-02. We expect this review to be completed during the third quarter of 1980.

Item 2 (emergency procedures used to obtain cold shutdown during loss of 1E and non-1E power)

WNP-1/4 will develop and write emergency procedures to include the steps required to achieve a cold shutdown upon loss of each class 1E and non class 1E bus that supply power to safety and non-safety related instruments and control systems. Also, a test demonstrating the ability to obtain shutdown and cooldown using only

safety grade instrumentation will be scheduled once definite methodology and acceptance criteria are determined. Finally, see our response, item 6 of the CR-3 "Intermediate" action.

The emergency procedures shall include:

- a) Identification of alarms, indicators and symptoms to alert the operator to the loss of power to each bus.
- b) The use of alternate indication and/or control circuits which may be powered from other class 1E or non class 1E instrumentation and control buses.
- c) Methods for restoring power to the bus.

F.2 Question

We are concerned that control system response could lead to transients initiating with plant parameters more severe than those assumed for the safety analysis or significantly increase the number of challenges to the protection system during early plant life. In this regard:

- a. *Operating experience at the Crystal River plant has indicated a control system response problem when bringing the plant up to power with a pump out of service. Specify your criteria and describe WNP-1/4 design features to preclude this type of response problem.*
- b. *Describe your design criteria, features, and operational requirements for the ICS and its supporting systems to preclude control response problems when switching from manual to automatic control and vice versa.*

Response

In response to the concern that the ICS may cause NSS instabilities that significantly increase the number of challenges to the protection system, operating experience at B&W plants has demonstrated that the ICS is a reliable system that tends to mitigate NSS upsets rather than initiate them. The data tabulated below demonstrate that B&W plants have been subjected to fewer challenges to the protection system than plants of other PWR vendors.

		<u>B&W</u>	<u>CE</u>	<u>W</u>
<u>1976</u>	Number of Auto Trips	25	46	147
	Number of Plants	6	5.1	19.13
	Trips/Plant/Year	4.17	9.02	7.68
<u>1977</u>	Number of Auto Trips	30	31	147
	Number of Plants	6.85	6.67	21.6
	Trips/Plant/Year	4.38	4.65	8.06

<u>1978</u>	Number of Auto Trips	43	41	150
	Number of Plants	3	7	23.4
	Trips/Plant/Year	5.38	5.86	6.41

Three-Year Average

	Trips/Plant/Year	4.64	6.51	7.38
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This information was extracted from the NRC "Gray Book" (NUREG-0020, Operating Units Status Report) for the years indicated.

- a. The difficulties encountered at Crystal River in mid-1979 during startup in a three-pump mode were due to operator unfamiliarity with this type of startup rather than control instability of the ICS. The three-pump startup procedure was later reviewed and modified to give more explicit instructions. The operators were also given further instructions in the proper execution of a three-pump startup. The ICS is designed and fully capable of providing adequate NSS control during three-pump startup as evidenced by the successful three-pump startups at Davis-Besse-1 in April 1978 and March 1980. The WNP-1/4 ICS incorporates the same design features to allow control or restart in the three-pump operating mode.
- b. The ICS design criteria, system features and operational requirements will be described in Section 7.7 of the WNP-1/4 FSAR. The ICS is designed to facilitate "bumpless" transfer to preclude control response problems when switching various ICS hand/automatic control stations from manual to automatic or vice versa. This feature allows the operator to place control stations in manual from automatic without perturbation. "Bumpless" transfer from manual to automatic requires operator action to zero the error across the hand/automatic station by adjustment of either demand or setpoint. The "bumpless" transfer feature then corrects minor offsets in control signals by a timed release provided by control system capacitance.

The ICS is designed to minimize plant upsets by use of a "Load Tracking" mode feature. In this mode, subsystems in automatic control will follow a subsystem which has been placed in manual with only slight error. If the operator exercises reasonable care in correcting this remaining error prior to switching from manual to automatic, mode switching is not a problem.

F.3 Question

Experience at operating B&W plants have indicated that the dynamics associated with main feedwater termination and steam generator pressure control following a reactor trip can lead to overcooling of the primary system. Discuss your criteria and the adequacy of your existing and proposed design features and changes to preclude this overcooling situation.

Response

The dynamics associated with main feedwater termination and steam generator pressure control following a reactor trip do not normally lead to overcooling of the primary system. Overcooling usually occurs only after an off-normal or failure situation. The normal situations are controlled by the ICS which has the following design criteria for proper control of feedwater after a reactor trip:

1. Reduce feedwater flow to both steam generators. Accomplished by reducing flow demand which results in closing the main control and block valves.
2. Maintain the OTSG low level setpoint. Accomplished by closing the feedwater startup valves until additional feedwater is required.
3. Maintain the low water level in each steam generator. Accomplished by continuing to use the startup feedwater valves.
4. Control main feedwater temperature at a minimum of 390F. Accomplished by secondary side system design.

Steam pressure control is also necessary to prevent an overcooling event. The design criteria for the steam pressure control is the following:

1. Maintain steam pressure below the design pressure of 1250 psia or that of the low set code safety valve whichever is lower.
2. Maintain steam pressure at 1200 psia following a reactor trip to regulate the heat sink temperature high enough to control reactor coolant pressure with significant operating margin to prevent HPI actuation.

Following a reactor trip, the ICS reducing flow demand which results in closing the main feedwater control and block valves to terminate main feedwater until the water level decreases below the two foot low level setpoint. The startup feedwater valves then open to control main feedwater thereby maintaining the steam generator low level and providing for the removal of decay heat.

With the simultaneous trip of the turbine on a reactor trip, the turbine stop valves close causing the steam pressure in both steam generators to increase and turbine bypass valves to the condenser and atmospheric dump valves to open to relieve the excess steam pressure. Thereafter, the steam pressure is maintained at 1200 psia by the turbine bypass valves.

This setpoint for steam pressure has been selected so that the primary system cold leg temperatures (which are nearly equal to the secondary side steam saturation temperatures) will maintain a proper cooldown of the primary system.

Figure 1 illustrates reactor coolant temperature, pressure, feedwater flow and pressurizer level following a reactor trip with proper feedwater flow and main steam pressure control. The rapid decrease in reactor power causes reactor coolant temperature to decrease (due to large heat transfer surface area in each steam generator); the resultant reactor coolant contraction causes a decrease in reactor coolant liquid volume and pressure. The reactor coolant cold leg temperature reaches an equilibrium value nearly equal to the saturation temperature of the secondary side steam pressure (567F at 1200 psia), and the reactor coolant pressure will eventually be restored to the normal operating pressure of 2210 psia.

The above discussion describes the system normal response to a reactor trip with a simultaneous turbine trip. Overcooling does not occur. Overcooling is defined as that cooldown of the primary system which causes either pressurizer level to go off-scale low or reactor coolant pressure to decrease below the setpoint for automatically initiating the HPI system. The HPI system is initiated by an Engineered Safety Features Actuation System (ESFAS) signal when the reactor coolant system (RCS) pressure falls below 1600 psig.

In an overcooling situation, the pressure in either steam generator may have decreased significantly below the 1200 psia setpoint due to either of the following causes:

1. Improper venting of steam through the safety relief valves.
2. Overcooling due to large flow rates of low temperature auxiliary feedwater.

A decrease of 150 psi or more in steam generator pressure below the 1200 psia setpoint is sufficient to cause a decrease in the primary system temperatures and approach to the overcooling conditions defined previously. The control of steam generator pressure in returning steam pressure from below 1200 psia back to the setpoint is accomplished by the heating and repressurization via the decay heat of the primary system.

WNP-1/4 design features of turbine bypass, ICS runback, and power operated relief valve (PORV) actuation keep the reactor on-line to minimize the reactor trip frequency and the probability of subsequent overcooling. However, the following items are being considered to further enhance the effectiveness of the safety and control systems. These proposed hardware and procedural changes would be designed to preclude overcooling events caused by improper steam generator pressure or feedwater flow control following a reactor trip:

1. Upgrade the two-channel, Class 1E Auxiliary Feedwater Control System to limit the rate of primary system cooldown by limiting the rate of steam generator level increase following a reactor trip where AFW is initiated (i.e., limiting AFW flowrates).
2. Review the current Main Feedwater System design to identify changes which would significantly decrease the frequency of feedwater upsets which might cause reactor trip, thereby minimizing the probability of subsequent overcooling.
3. Install both Main and Auxiliary Feedwater Overfill Limiter to preclude feedwater overfill above a preset steam generator level, thereby minimizing overcooling due to failures in the main or auxiliary feedwater flow control system following reactor trip.
4. Add a control function to the ICS to provide for positive and rapid reduction of main feedwater flow following a reactor trip.
5. Interlock ICS operation of atmospheric dump and turbine bypass valves to preclude a single failure from opening more than 25% steam dump capacity.

Overcooling is a moderate event which is safely mitigated by the actuation of the Engineered Safety Features Actuation System. The combination of existing and proposed design features for WNP-1/4 will serve to further reduce the frequency of overcooling by proper steam generator pressure and feedwater flow control following reactor trip.

F.4 Question

Discuss the advantages and disadvantages, if any, of a control independent of the ICS to terminate main feedwater flow following a reactor trip.

Response

The routine termination of main feedwater following a reactor trip would be a drastic solution to a low probability event, overfill of the steam generators following trip. At this point, two events that are sometimes confused should be distinguished, steam generator overfill following trip and temporary overfeed. The

first of these is definitely an undesirable event and can cause RCS overcooling. It is, however, a low probability event and certainly does not routinely occur following reactor trip. The second event, temporary overfeed, occasionally occurs following reactor trip; but while a departure from ideal and expected post-trip performance, it is not serious. Temporary overfeed is the result of less than perfect control system performance but has no safety implications and does not result in overcooling.

The routine termination of main feedwater (the preferred source of water for the steam generator) following reactor trip would unnecessarily exercise the auxiliary feedwater system, complicate the control room operators' duties following a trip, and superimpose an additional transient upon the steam generators following trip. Further, this action would place the entire Nuclear Steam Supply System in a degraded condition by deliberately defeating the primary means of cooling the reactor core, main feedwater.

F.5 Question

Specify the extent to which control limitations such as valve and pump speed responses affect main feedwater response during startup from the manual to the automatic mode.

Response

The control limitations of valve and pump speed responses will not strongly affect the main feedwater flow response during startup from the manual to the automatic mode where the operating procedures properly reflect these limitations. During startup with the ICS feedwater controls in automatic (which is the preferred control mode), the pumps and valves are normally capable of following the gradual increase in reactor power.

Startup is usually performed with the feedwater valves in automatic on level or flow control with the pump controls in auto maintaining a set pressure drop across the control valves. If any of the valves, pumps or feedwater demand hand/auto stations is in manual control and the operator desires to transfer to auto, certain precautions must be taken to ensure that "bumpless" transfer occurs. The operator accomplishes this by ensuring that the controls are adjusted to produce zero error (flow or pump speed error) prior to the transfer. The control error signal may be manually adjusted to a zero value by the operator.

The ability to effectively control feedwater flow rates during startup is affected by: 1) valve controllability over the flow range, 2) sequencing of valves during startup, 3) valve leakage, and 4) control of feed pump speed and recirculation flow. These factors may act to produce a less than ideal flow response but this response generally will be manageable by the control system or by utilizing specific operating modes which minimize the effects of placing a sub-loop in manual, e.g., pump speed control.

The design of the feedwater flow controls is such that some degree of degraded valve and pump response will not produce an unacceptable overall flow response. In the event that perturbations in feedwater flow rates influence the primary system to a significant extent, adjustments may be required to be made to the valve or pump speed controllers or to the ICS control modules.

The WNP-1/4 plants are designed to allow a minimum feed pump speed over the low power range such that the proper control valve pressure drop can be maintained. Also the feed pump recirculation flows will be set to maintain the required minimum pump flow rates at approximately 35% of design flow by a direct-acting modulating recirculation valve. These conditions will minimize the limitations of pump speed and flow control such that these should not be a source of concern for flow controllability over the power operating range.

F.6 Question

State the design objectives of the improved auxiliary feedwater control system. Also indicate whether it will:

- a. *Initiate for all loss of MFW events, either total or partial and at what lower limit;*
- b. *Initiate on loss of offsite power;*
- c. *Preclude overcooling or undercooling of the primary system even with a single failure in the system (e.g., failures in input, power, valves);*
- d. *Interact in any adverse fashion with the Feed-Only-Good-Generator interlock.*

Response

The design objectives of the Auxiliary Feedwater Control System are as follows:

1. Provide redundant and independent initiation and control circuits for each AFW train such that the capability to initiate and control at least one AFW train when required is maintained even when degraded by a single random failure. Redundancy and independence will be provided from the sensors through the actuated devices.
2. The redundant portions of the AFW Control System will be powered by separate Class 1E vital, battery-backed busses such that the objective of Item 1 above can be accomplished with the loss of a single vital bus or with the loss of all AC power except that derived from inverters.

3. The AFW system and its controls will be designed such that AFW flow will be initiated within 40 seconds of sensing the conditions listed below. This time limit includes the time required for diesel startup and generator loading.

The Auxiliary Feedwater Control System will:

- a. Initiate on total loss of MFW or ESFAS actuation. Low steam generator level initiation provides protection against partial loss of feedwater and backup to the total loss of MFW signal.
- b. Indirectly initiate on loss of offsite power by initiating on loss of all RC pumps.
- c. Reduce overcooling or undercooling of the primary system by controlling the rates of steam generator level increase and providing two level setpoints, one for forced reactor coolant circulation and one for natural circulation. This will ensure that adequate cooling is provided even with maximum decay heat levels while minimizing the potential for overcooling by excessive AFW flow.

The level rate feature of the control system is not intended to be designed to single failure requirements; a single failure could result in full AFW flow and overcooling. The safety function of the AFW system is to provide decay heat removal. Designing level rate control to single failure requirements would result in a degradation of the ability to meet the safety function of the AFW system.

- d. Not interact in any adverse fashion with the FOGG system because the AFW Control System signals will be overridden by FOGG signals.

Dynamic response of the auxiliary feedwater control system will be demonstrated by operational testing during the plant startup test program.

F.7 Question

For your intended revision to the AFW initiation logic, identify the signals (e.g., generator level, no feedwater flow, loss of pump suction pressure, SIAS, and loss of steam flow to pumps) that will be used to initiate AFW and justify their use.

Response

No revision to the AFW initiation logic was proposed in our initial response. The parameters sensed for initiation of AFW in the existing design and their purposes, are as follows:

- 1) Loss of MFW: The AFW system provides a backup source of feedwater sufficient to remove decay heat and pump heat should the primary source (MFW) be lost. The means of sensing a loss of MFW flow has not yet been selected.

- 2) Low steam generator level: Low level in either steam generator is indicative of insufficient feedwater flow and provides a backup for initiation on loss of MFW.
- 3) ESFAS actuation: ESFAS actuation on low RC pressure, low steam pressure, or high Containment pressure will result in main steam and main feed isolation on both steam generators. Therefore, AFW is required to remove decay heat. FOGG logic will prevent AFW flow to a ruptured steam generator.

F.8 Question

In addition to the improved FOGG logic to be provided as part of your revised AFW evaluations, identify those events and combinations of events which have been and will be evaluated to assure that no confused or inadvertent inputs (such as from a previously unrecognized event or event combination) can lead to a malfunction or undesirable operation of the FOGG system. Also describe any studies and tests performed to assure proper integration and interaction of the FOGG interlock with other systems.

Response

The events and combination of events which have been evaluated to assure that no confused or inadvertent inputs can lead to a malfunction or undesirable operation of the FOGG system will be defined in Chapter 15 of the FSAR. Any required changes resulting from this evaluation will be incorporated in the design.

F.9 Question

You state that you are considering changes to improve the algorithm used for AFW flow control to limit primary cooldown rate following AFW actuation. Describe how these changes would provide the capability to distinguish in a positive manner between transients and accidents. Also describe how two-phase level during swell from depressurization affects level detection and how this will be treated.

Response

The auxiliary feedwater flow control system changes to limit primary cooldown rate following AFW actuation will not affect the ability to distinguish between transients and accidents. The AFW system and associated controls perform the appropriate function regardless of whether the event is a "transient or an accident". AFW is supplied to the appropriate steam generator(s) at the appropriate rate and to the appropriate level setpoint for the existing Reactor Coolant System and steam generator conditions.

"Two-phase level" exists during all periods of steam production above 15% power and is normal for the OTSG. Steam generator depressurization can affect level detection by changing the average mixture density. While this phenomenon is considered to be minor, of short duration, and have little effect on core cooling, it and other error mechanisms are under evaluation and will be accounted for if necessary by analytical, setpoint, or procedural means.

F.10 Question

The modifications, recommendations, and studies you present to reduce sensitivity are in the direction of additional automation of the plants. While this approach leaves the operator free to verify system performance and should improve the control of transients, we are concerned that potential system interaction effects might result. Therefore, a complete and integrated review of the primary and secondary system should be performed to assure that no significant adverse interactions result from the modifications that are ultimately made. Describe your plans and schedules with regard to performing such a comprehensive integrated evaluation of these changes, based upon conservative and realistic analyses and simulator comparisons as appropriate.

Response

Improvements in the WNP-1/4 design and anticipated changes identified in attachment f to our letter of December 3, 1979 are intended to minimize, mitigate, or eliminate certain undesirable interactions. A comprehensive evaluation of these changes will be performed. It will include interaction analysis and operability analysis. A program similar to the Abnormal Transient Operating Guideline (ATOG) program will also be considered.

The interaction analysis will be performed for each change generally as follows:

- a. review of interfaces
- b. development of "event trees" identifying potential interactions
- c. analysis of interaction effects using appropriate simulation techniques and models

The operability analysis will include all normal transients and upset conditions for which the reactor is not expected to trip. Its objective is to assure adequate margins between operating conditions and operational and trip limits.

An ATOG program including the WNP-1/4 design would provide a realistic examination of major upsets to allow for development of appropriate operating guidelines.

A detailed schedule has not been developed. However, when finally developed, the schedule will show completion of a comprehensive evaluation along the general lines described above of all potential modifications prior to fuel loading.

F.11 Question

Provide the following analyses:

- a. *Overcooling event initiated by steam pressure regulator or throttle valve malfunction resulting in increased steam flow.*
- b. *Overcooling event initiated by feedwater system malfunctions that result in decreased feedwater temperature.*

For these analyses, assume no beneficial operator action before 10 minutes. Also, only qualified safety systems should be assumed for mitigation. Identify which safety and non-safety grade systems are considered to operate during this transient and specify the part each of these systems take in the transients. Identify the signals acting upon these systems during the transients.

The analyses should be performed for a period of at least 10 minutes after transient initiation. If existing analyses which are presented for a shorter duration are utilized for this response, then confirm that during the time not shown out to 10 minutes:

- (1) No operator action is required or assumed.*
- (2) No changes in operating systems are required.*
- (3) No significant changes result out to 10 minutes, such that extrapolation from the results presented is considered valid.*

Response

- (a) The steam pressure regulator malfunction event has been analyzed and is presented in Reference 1.
- (b) The overcooling event initiated by feedwater system malfunctions that result in decreased feedwater temperature will be provided in the WNP-1/4 FSAR, Subsection 15.1.1. The overcooling effect is less severe than the steam generator overfill and steam pressure regulator malfunction events; therefore, it is not included as part of the response to 50.54 (f). The WNP-1/4 FSAR analysis is carried out for 120 seconds. If this analysis were continued for a full 10 minutes, operator action would not be necessary since the plant parameters would trend from their 120 seconds value as expected, to a self-regulating steady state condition.

F.12 Question

You have stated during related meetings with NRC and with ACRS subcommittee that the analyses presented in your current 50.54 (f) response were not necessarily selected to represent the worst case. Provide your recommendations as to what criteria, assumptions, and experience should be recognized in defining the worst case for design purposes.

Response

A full spectrum of overcooling events from those considered moderate frequency to design basis has been provided in Reference 1 which was submitted subsequent to your March 25, 1980 supplemental request. The results varied from no voiding in the RCS to large steam voids being formed. However, adequate core cooling was maintained in all cases analyzed.

F.13 Question

To prevent automatic tripping of the reactor coolant pumps due to ESFAS initiated by overcooling events, you state that the WNP-1/4 pump trip logic will include coincidence circuitry sensing pump motor current. This input is intended to actuate on degraded pump current indicative of significant RCS void formation characteristic of a LOCA; but for overcooling events, the extent of void formation should not reach a point where degraded pump current will trip the pumps and undesirable pump trip will thus be avoided. Describe the significant elements of the development program for this circuitry, including that phase directed to the distinction of a valid motor current signal. What criteria will distinguish a valid signal? How will the system be verified in an actual nuclear power plant or under realistic conditions? Provide your current schedule for this program.

Response

WPPSS is pursuing the development of an automatic reactor coolant (RC) pump trip design generically through participation in the Babcock & Wilcox (B&W) Owners Group. The goal of this effort is a design which will trip the RC pumps for all events identified by B&W analyses as being required to assure compliance with 10 CFR 50, Appendix K criteria, while limiting to the extent practicable, pump trip for non loss-of-coolant accident (non-LOCA) events. In the WPPSS reply to your 10 CFR 50.54 (f) request, it was stated that the WNP-1/4 automatic pump trip circuitry would incorporate a coincidence circuitry sensing RC pump motor current to minimize unnecessary pump trips.

Subsequent to this response, difficulties were encountered in implementing this design concept, especially in the analysis of the correlation between the total RC system void, the localized void at the RC pump suction, and the corresponding RC pump motor

current. As a result, B&W is reviewing the feasibility of RC pump motor current providing an acceptable coincidence signal while also investigating alternative concepts for providing this feature. Response to your detailed questions concerning program development and design criteria will be provided upon better definition of the design concept to be employed.

F.14 Question

After the PORV closed during the transient at Crystal River Unit 3 on February 26, 1980, the reactor coolant system pressure increased from approximately 1400 psi to 2400 psi in less than 3 minutes. The last 600 psi (from 1800 to 2400 psi) of this increase occurred in less than 1 minute. This caused lifting of the code safety valves. Operating guidelines for B&W supplied plants typically recommend termination of high pressure injection when hot and cold leg temperatures are at least 50°F below the saturation temperature of the existing reactor coolant system pressure and the action is necessary to prevent the indicated pressurizer level from going off scale.

In view of this characteristic of rapid depressurization (Sic), what operator action, and basis thereof, is proposed to reduce the potential for lifting of the WNP-1/4 code safety valves?

Response

The uncontrolled addition of HPI can result in repressurization of the RCS. To control the rate and magnitude of the RCS pressure increase, the operator's principal actions are:

1. To throttle or stop HPI once control criteria are satisfied.
2. To stabilize the reactor coolant temperature.
3. To manually open the PORV (i.e., if automatic controls are not operative or the PORV block valve is closed) if high RCS pressure occurs.

The first two actions above are essentially the first line of defense to controlling RCS repressurization. HPI control, in practice, can be viewed in two parts:

1. HPI may be reduced (i.e., stop all but one HPI pump or throttle flow using the HPI injection valves) anytime the reactor coolant subcooled margin is established
2. HPI may be stopped any time the reactor coolant subcooled margin is established and pressurizer level is "on-scale" and increasing. Normal makeup should be restarted

Operator action to limit RCS repressurization (control HPI) can thus be initiated as soon as the reactor coolant subcooling margin is established. An uncontrolled addition of HPI to restore pressurizer level as well is not necessary; pressurizer level can be restored in a more controlled fashion.

RCS temperature control is identified because a heat-up of the reactor coolant can cause an insurge into the pressurizer (in addition to HPI) and enhance RCS repressurization. To achieve RCS temperature control, the operator must verify proper operation of secondary inventory and pressure control systems. Following severe overcooling events, a manual reduction in steam pressure (limits the temperature to which the reactor coolant will reheat) may also be necessary to stabilize pressurizer level after HPI is stopped.

Use of the PORV is the second line of defense in controlling RCS repressurization. The PORV has sufficient capacity to prevent lifting the safety valves when the HPI system is at maximum capacity and the reactor coolant is subcooled. Operating procedures will include instructions to verify automatic PORV operation or to manually open the PORV or PORV block valve if high RCS pressures occur.

The causes and corrective actions for RCS repressurization will be extensively covered in training programs and the operator will acquire practical experience on the WNP-1/4 simulator when treating abnormal transients which require HPI. In general, the operator will not be faced with as rapid system changes that were deliberately induced during the Crystal River event (i.e., the operators initiated HPI cooling using all three HPI pumps and intentionally did not control repressurization because the validity of primary and secondary system signals could not be immediately determined). More time for operator action will typically be available. With training in determining the cause of RCS repressurization, the importance of timely action and the practical application of corrective action, the operator should be highly effective in controlling RCS repressurization without lifting the pressurizer safety valves.

F.15 Question

It is our understanding that the B&W 205 plants operate with a considerably smaller water inventory in the steam generators than the B&W 177 plants. Explain what effect this has on the sensitivity of the 205 plants to both undercooling and overcooling events. Include the impact of MFW response items and reliabilities in your evaluation.

Response

The steam generator water inventory, at rated power, is less per Mwt in the B&W 205 plant than in the B&W 177 plant. In general, this effect tends to limit overcooling transients and make overheating transients more severe. However, the relationship is not

necessarily one-to-one as the system response is dependent on many factors other than steam generator inventory. Therefore, the impact of main feedwater response times and reliabilities are dependent on the total system interaction and not just the steam generator type. System behavior has been evaluated as part of the FSAR accident analyses which demonstrate acceptable response of the system. The WNP-1/4 FSAR will contain this information in Chapter 15.

RCS RESPONSE TO REACTOR TRIP

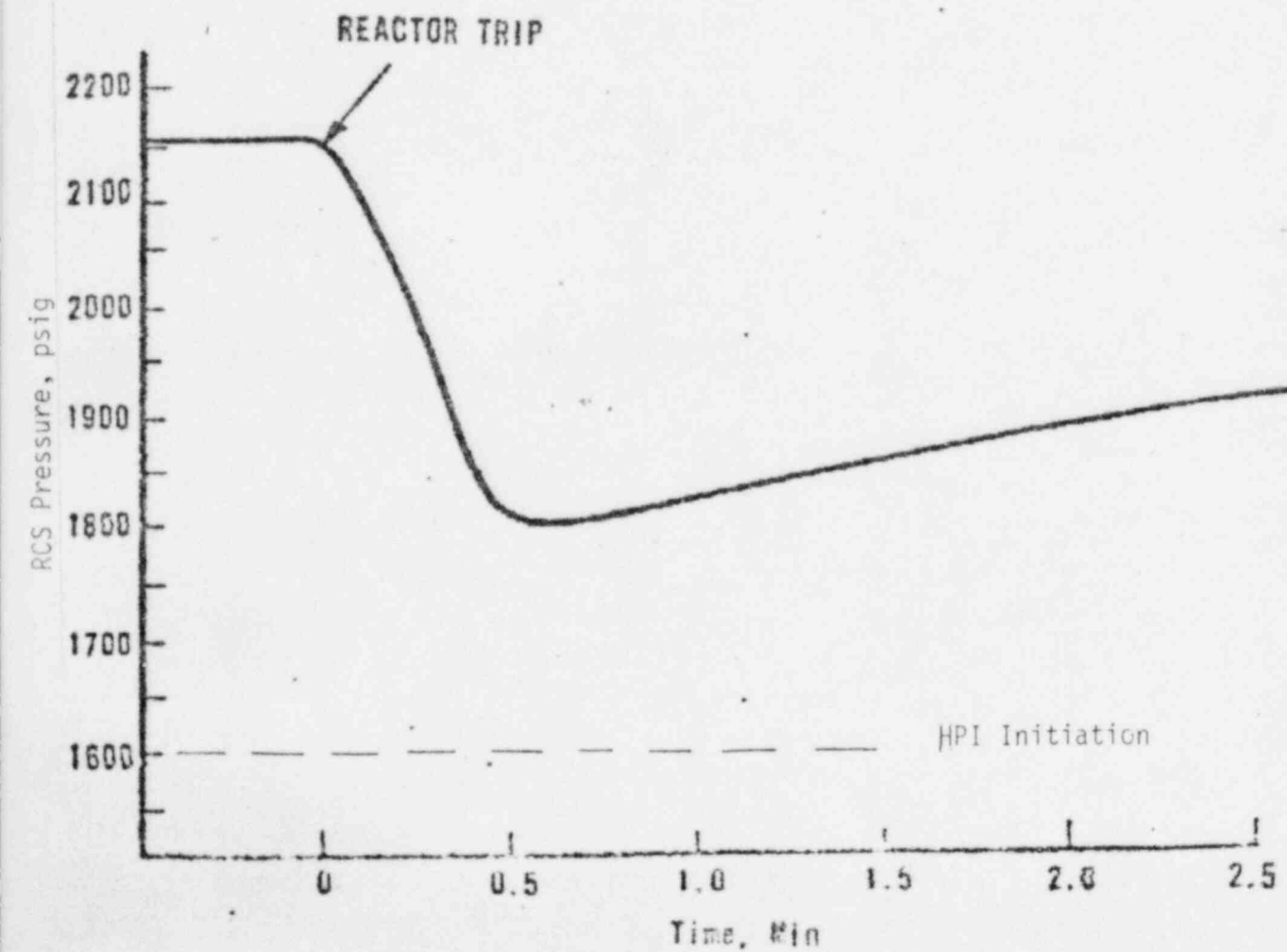
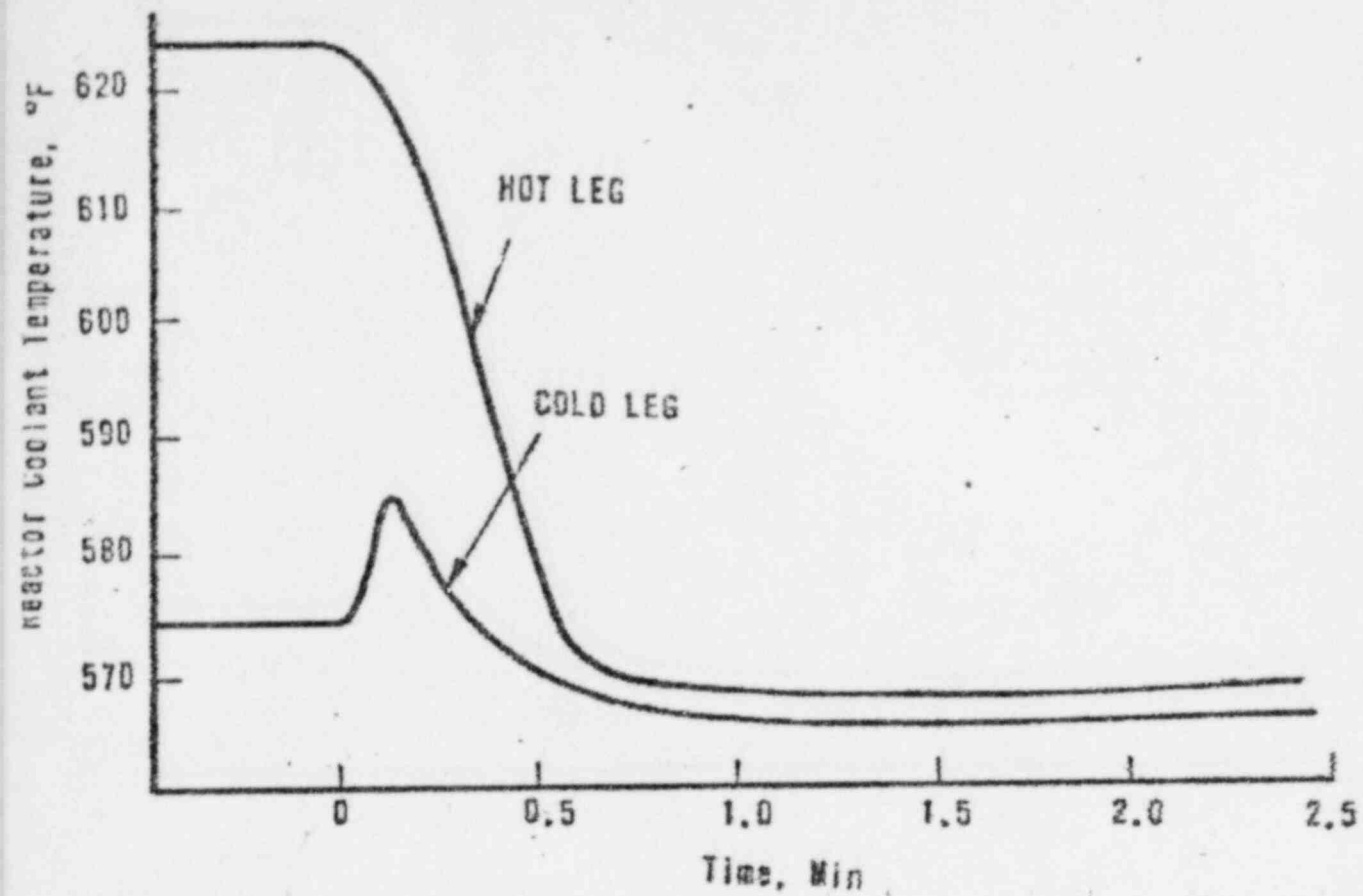
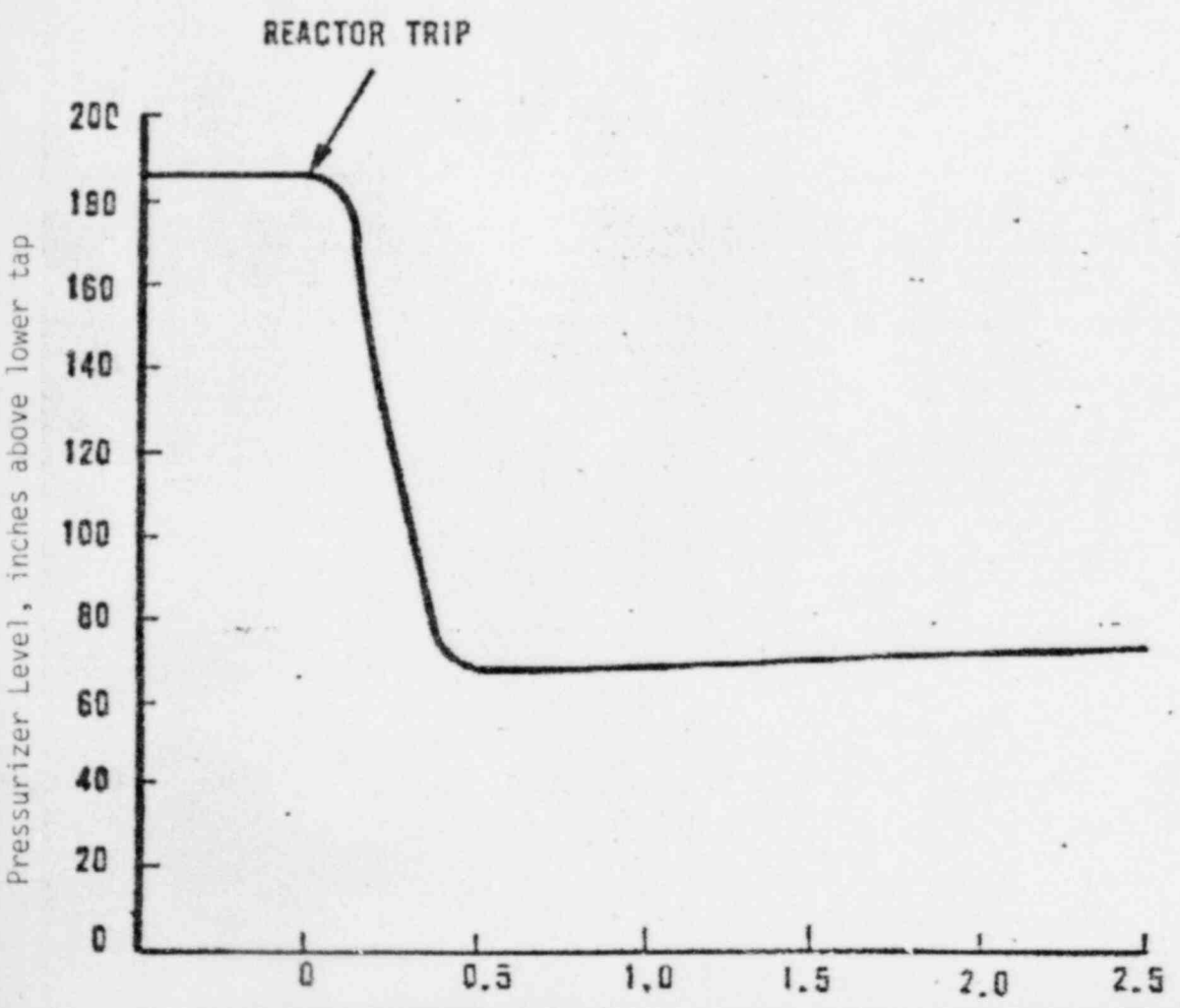
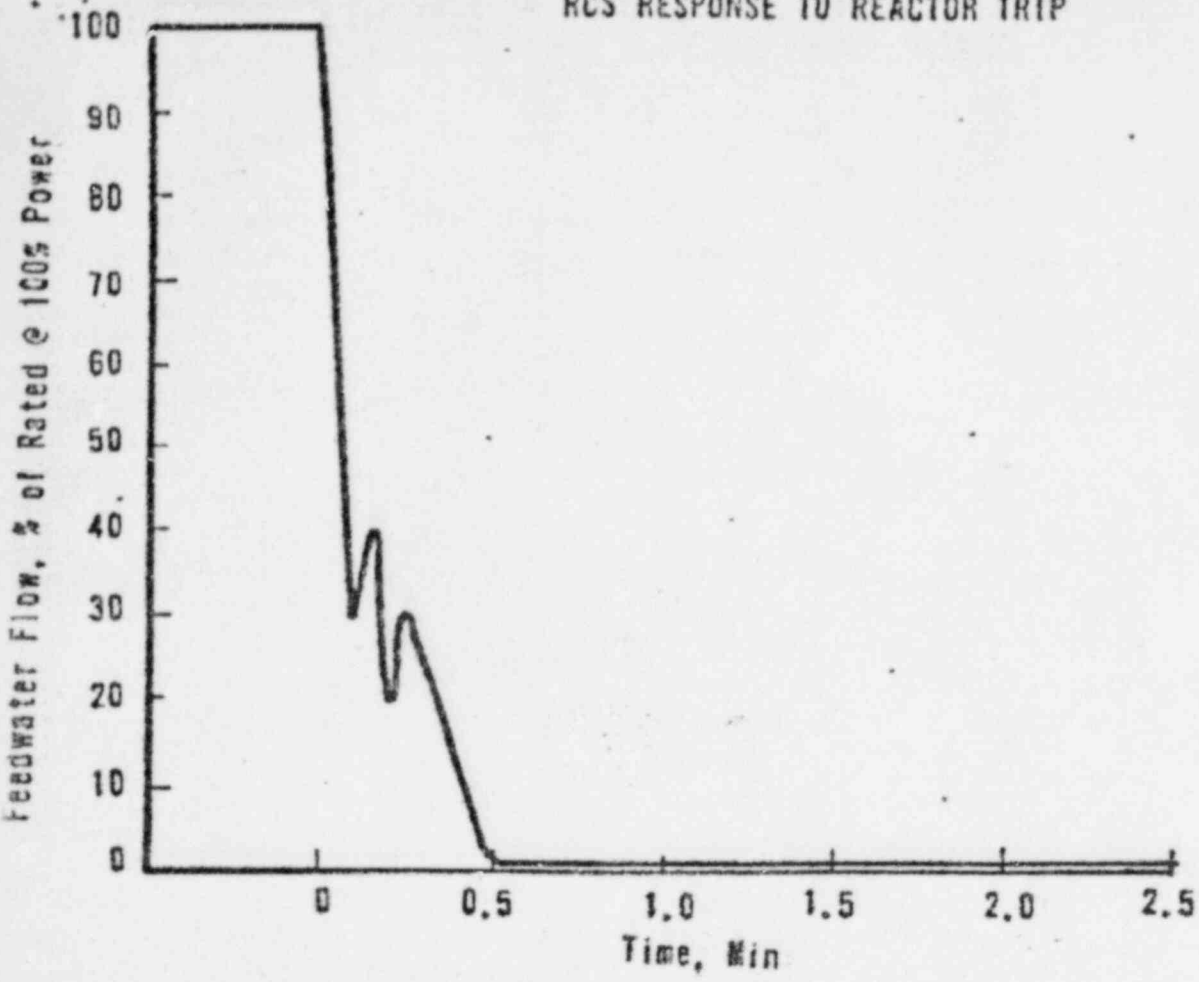


FIGURE 1 (Cont'd)
RCS RESPONSE TO REACTOR TRIP



Reference 1

Letter, D. L. Renberger, WPPSS to H. R. Denton, NRC, "Response to NRC 10 CFR 50.54 Letter of October 25, 1979, Revision 1." May 5, 1980.