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

DEC 5 1979

Docket No. 50-289

MEMORANDUM FOR: Steve Scott, Acting Chief
Distribution Services Branch, ADM

FROM: Richard H. Vollmer, Director
Three Mile Island Support

SUBJECT: THREE MILE ISLAND UNIT 1 BOARD NOTIFICATION 10 CFR 50.54 REQUEST REGARDING DESIGN ADEQUACY OF BABCOCK AND WILCOX NSSS

Please forward the enclosed material to the Three Mile Island Unit 1 Board Notification Service Listings. The material is an October 25, 1979, letter signed by H. R. Denton and an evaluation of the letter.

Richard H. Vollmer, Director
Three Mile Island Support

Enclosure:
As stated

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Three Mile Island-1 (Docket No. 50-289)

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EVALUATION OF THE ENCLOSED MATERIAL

It is the staff's judgment that the enclosed material is relevant to the scope of the proceedings before the ASLB in the matter of the restart of Three Mile Island Nuclear Station, Unit 1.

The enclosed material (October 25, 1979 letter from H. R. Denton to the Tennessee Valley Authority) requests information concerning the sensitivity to feedwater transients of the Bellefonte Nuclear Plant now under construction which has a Babcock and Wilcox nuclear steam supply system.

Similar letters have been sent to all utilities holding construction permits for plants with a Babcock and Wilcox nuclear steam supply system. The request for information is directly related to the low water inventory and the presence of a liquid-vapor interface in the B&W once-through steam generator which closely couples the primary system to the steam generator conditions with a consequently high sensitivity to feedwater flow rate perturbations.

The additional information being requested from holders of construction permits will allow a determination by the staff whether it is necessary to halt all or portions of the construction of the affected plants pending the results of an ongoing staff study.

The staff has a study now underway, using Crystal River, Unit 3 an operating facility with a B&W nuclear steam supply system, as a reference facility. The objective of this study is to identify accident sequences leading to core damage having high frequency compared to the Reactor Safety Study (WASH-1400). The study uses event tree and fault tree techniques.

This assessment to be completed in about 6 months will focus on the risk implications of the sensitivity of the B&W design and on the potential interactions arising from the integrated control system.

Primary System Perturbations Induced by Once Through Steam ...

I. Introduction

B&W plants employ a once through steam generator (OTSG) design, rather than U-tube steam generators which are used in other pressurized water reactors. Each steam generator has approximately 15,000 vertical straight tubes, with the primary coolant entering the top at 603-608°F and exiting the bottom at about 555°F. Primary coolant flows down inside the steam generator tubes, while the secondary coolant flows up from the bottom on the shell side of the OTSG. The secondary coolant turns to steam about half way up, with the remaining length of the steam generator being used to superheat the steam.

The secondary-side heat transfer coefficient, in the steam space of the OTSG, is much less than that in the bottom liquid section. This results in a heat transfer rate from the primary system which is quite sensitive to the liquid level in the steam generators. If a feedwater increase event occurs, the liquid-vapor interface rises, increasing the overall heat transfer. This decreases the outlet temperature below 555°F and initiates an overcooling event, which can lead to primary system depressurization. By contrast, if a feedwater decrease event occurs, the overall heat transfer decreases, the outlet primary temperature increases, and a pressurization transient ensues.

In either of these cases, the response of the primary system pressure and pressurizer level to a change in main feedwater flow rate (or temperature) is comparatively rapid. These rapid primary system pressure changes due to changes in feedwater conditions is known herein as system "sensitivity" and is

unique to the B&W OTSG design.

Following the incident at Three Mile Island, various actions were taken to increase the reliability of the auxiliary feedwater systems and improve plant transient response. System modifications to increase the reliability of the AFW may have resulted in more frequent AFW initiation. However, use of AFW results in introduction of cold (100°F vs. 400°F) feedwater into the more sensitive upper section of the steam generators. This may act to enhance system sensitivity.

Further system modifications provide control-grade reactor trips based on secondary system malfunctions, such as turbine or feedwater pump trip. While these reactor trips do serve to reduce undercooling feedwater transients by reducing reactor power promptly following LOMFW, they may amplify subsequent overcooling.

A reexamination was made of small break and loss of feedwater events for B&W plants. This resulted in a modification of operator procedures for dealing with a small break, which include prompt RCP trip and raising the water level in the steam generators to (95%) to promote natural circulation. Both these actions are taken when a prescribed low pressure set point is reached in the reactor coolant system and for anticipated transients such as loss of feedwater these actions may amplify undesirable primary system responses.

In addition to the post-TMI changes discussed above, actions were also taken to reduce the challenges to the power operated relief valve (PORV) by raising the PORV set point and lowering the high pressure reactor trip. While these actions have been successful in reducing the frequency of PORV operation, they

have resulted in an increased number of reactor trips. This occurs because the reactor will now trip for transients it previously would have ridden through by ICS and PORV operation.

The staff is concerned by the inherent responsiveness of B&W OTSG design. While some specific instances are presented in the next section of this paper, the staff concerns are also of a general nature. It is felt that good design practice and maintenance of the defense-in-depth concept, requires a stable well-behaved system. To a large part, meticulous operator attention and prompt manual action is used on these plants to compensate for the system sensitivity, rather than any inherent design features.

The staff believes that the general stability of the B&W plant control systems should be improved, and that plant response to OTSG feedwater perturbations be dampened.

II. Recent Feedwater Transients

On August 23, 1979 the staff met with the B&W licensees to discuss recent feedwater transients. One aspect which is of interest is the relationship of the operator to the functioning of the main feedwater system. In at least one instance an operator manually opened a block valve in series with a control valve (partly open but thought to be closed). This resulted in an overfeed condition. In several recent events the feed flow was reduced to the point where the reactor tripped on high pressure. Subsequent overfeed reduced pressure to below 1600 psi, where HPI was initiated, reactor coolant pumps tripped, and auxiliary feedwater flow introduced into the top of the steam generators, which increased the severity of the cooldown transient.

It appears that in many cases the main feedwater control system does not react quickly enough or is not sufficiently stable to meet feedwater requirements. Rather, the system will often oscillate from underfeed to overfeed conditions, causing a reactor trip and sometimes a high pressure injection initiation. One undesirable element of this lack of stability is that overcooling transients on the primary side proceed very much like a small break LOCA (decrease in pressurizer level and pressure). Thus, for a certain period of time the operators may not know whether they are having a LOCA or an overcooling event. Th. same type of behavior can be initiated by the normal reactor control system. This was demonstrated by a December 1978 event at Oconee, where failure of a control-grade T_{avg} recorder led to reactor trip, a feedwater transient, and ESF actuation. A partial list of recent B&W transients and their effects is contained in the Appendix to this report.

III. <u>Role of the Pressurizer Level Indicator</u>

A major area of concern arising from the B&W OTSG sensitivi , is the response of pressurizer level indication. Several B&W feedwater transients have led to loss of pressurizer level indication. Most notable was a November 1977 incident at Davis Besse where level indication was lost for several minutes. The arrival rate for this event appears to be on the order of .1-.2 per reactor year, but could be on the increase due to the potential for more reactor trips and feedwater transients resulting from post-TMI-2 system modifications. This is of concern because an overcooling event could empty the pressurizer, thereby creating the potential for forming a steam bubble in the hot leg which may interrupt natural circulation, following RCP trip on low pressure. The staff feels that the uncertainties associated with two phase natural circulation are somewhat high for an event with a recurrence interval of a few years.

Additionally, the staff believes that good design practice and adherence to the defense-in-depth concept, would require that plant operators be aware of the reactor's status during expected transients. A low-level off-scale reading on pressurizer level makes it impossible for the operators to assess system inventory and more difficult to differentiate between an accident and an excessive cooldown transient. The staff feels that the frequency with which this situation occurs is undesirable.

Some concerns also exist with regard to the operation of the pressurizer heaters when loss of level takes place. Nonsafety grade control circuitry trips the heaters off when pressurizer level is low. If these nonsafety grade cutoffs should fail, the heaters would be kept on while uncovered. This situation has the potential of overheating the pressurizer to the failure point, as happened with a test reactor at Idaho Falls.

IV. Role of ICS-MFW

The ICS appears to paly a significant role in the plant's feedwater response. The staff is currently reviewing an FMEA study on the ICS. However, review of operating experience suggests that the ICS often is a contributor to feedwater transients. In some cases the ICS appeared inadequate to provide sufficient plant control and stability. Some of the utility descriptions of feedwater transients (as summarized in the minutes of a meeting on August 23, 1979) emphasized the role of the operator in operating the MFW system. The following sequence illustrates the type of event and system response which the staff feels could potentially occur.

1. Reactor at 100% power.
2. Reactor trip, from arbitrary cause (does not matter).
3. Plant stabilizes in hot shutdown, for a few minutes, heat rejection to condenser (and/or secondary dump valves).

4. Overfeed transient [illegible]

pressurizer level shrinks, pressure reaches 160 psi, RS actuates; RCP tripped; AFW on. (Possible RCP seal failure).

5. Operator manually controls AFW (possibly MFW instead or in addition, if MFW not isolated such that OTSG level comes up to 95% of operating range. This massive addition of cold water may lead to emptying of pressurizer and interruption of natural circulation (or, the hot leg may flash due to depressurization and interrupt natural circulation even if pressurizer does not empty).

6. HPI delivers cold water; no heat transfer in OTSG; vapor from core leads to system repressurization; steam may condense or PORV may lift.

7. No pump restart criteria available, circulation may not be reestablished.

It appears that an upgraded safety quality ICS, which is designed to balance power to OTSG level in a better fashion, could reduce the sensitivity, illustrated in the above sequence.

V. Role of ECCS and Auxiliary Feedwater

It is known that some feedwater transients result in overcooling to the extent that the HPI actuation setpoint is reached. Traditionally, the operator isolates letdown and turns on an extra makeup pump following trip so as to avert this actuation. If this manual action is not performed quickly enough, or if the cooldown transient is too severe, the HPI set point will be reached and the pumps automatically started. Following procedures, the operator would then trip all ma coolant pumps and utilize recovery procedures based on the plant symptoms. If the incident was actually a feedwater event and not a small LOCA, he would then presumably go to the loss of forced circulation procedures. When pressure has recovered such that the coolant system has become 50°F subcooled, the operator can secure HPI. One problem is the difficulty in differentiating between a small

break LOCA and an excessive feedwater transient. The operator would be forced to assume a small LOCA until proven otherwise. However, following the small break procedures and introducing cold auxiliary feedwater, may increase the severity of an overcooling event. Initiation of AFW and delivery to the OSTG, especially if accompanied by filling to the high level required by new procedures (95%) will continue the cooldown and depressurization. Thus, the AFW system acts to increase the responsiveness of the reactor to feedwater transients where excessive cooldown is occurring.

VI. Conclusions

The staff believes that the current B&W plants are overly responsive to feedwater transients because of the OTSG design, pressurizer sizing and PORV and high pressure trip set point. Some of the sensitivity also arises from inadequacies in the ICS to deal with expected plant perturbations.

Regardless of the reasons, B&W plants are currently experiencing a number of feedwater transients which the staff feels are undesirable. The staff believes that modifications should be considered to reduce the plant sensitivity to these events and thereby improve the defense-in-depth which will enhance the safety of the plant.

APPENDIX

FEEDWATER TRANSIENT SUMMARY

FACILITY	TRANSIENT DATE	DESCRIPTION
CR-3	8/16/79 (0259	Reactor Trip on High Pressure - 4 to 3 RCP, A-S/G underfed -72%
	8/16/79 (1125)	Reactor Trip on High Pressure - 3 RCP - A-S/G underfed - 45% Pwr
	8/17/79 (0706)	Reactor Trip on High Pressure - 3 RCP - A-S/G underfed - 48%Pwr
	8/17/79 (1825)	Reactor Trip on High Pressure - 3 RCP - A S/G underfed - 26% Pw
	8/02/79 (0202)	Reactor Trip on Low-Low Level in both S/G - 10% Pwr.
ANO-1	8/13/79 (1749)	Turbine Trip - Antic. Trip did not work - Rx Trip on Hi Press -
Oconee-1	6/11/79 (0333)	Reactor Trip on Anti. Trip (LOFW) - 99% Pwr.
	6/11/79 (0752)	Reactor Manually Tripped when FWPT "1B" Tripped
Oconee-2	5/07/79 (0346)	Reactor Trip on High Pressure - feedwater oscillations - 18% P
	6/03/79 (2046)	Reactor Trip on High Pressure - feedwater oscillations - 30% P
Rancho Seco	7/12/79 (1714)	Reactor Trip on Antic. Trip (LOFW) - 100% Pwr.
Davis-Besse	NONE	

IREP - INITIAL PLANT STUDY

We have attempted to develop a general framework for the conduct of a limited risk assessment of a B&W reactor aimed at identifying any unique risk-impacting sequences relative to the Reactor Safety Study. An absolute determination of risk is not intended. We have selected Crystal River 3, a plant owned and operated by Florida Power Corporation for analysis. The architect-engineer for this Babcock and Wilcox reactor was Gilbert Associates. It began commercial operation in March 1977.

The project, as presented in Figure 1, will require the following tasks:

1. A survey of the LER files as now established in ORNL and AO reports, as well as the Sandia and Fluor-Zion systems interactions studies to identify interactions and common mode failures which have occurred in similar plants. This survey should parallel construction of system logic models and event trees since it will ensure that actual experience is incorporated into the assessments performed.

2. Event trees for loss-of-coolant accidents and transient conditions. Specific attention will be given to more frequent LOCAs and these will include a feedwater transient tree which incorporates experience at B&W plants and will explore the post-TMI modifications. Emphasis will be given toward understanding the human coupling interaction between systems at the event tree sequence level.

3. Fault trees for the key systems identified in the event trees. They will be constructed to the component level and will include control, actuation, and electric power considerations. Human errors will be included as well as the ability of the operator to cope in the time span available. Our preliminary opinion is that simplified fault trees will be required for the following systems: auxiliary feedwater and secondary steam relief, high pressure emergency core cooling in the injection and recirculation modes, low pressure emergency core cooling in both injection and recirculation modes, containment spray and containment heat removal systems and a limited study of loss of AC power, considering the 480 and 4160 busses and the emergency diesel generators, with limited analysis of high voltage switchyard faults. Separate fault trees will probably be required for ECCS and AFWS initiation logic and the system trees must include the contribution from auxiliary systems such as instrument air, ventilation, component cooling, etc., and control-induced failures. Truncation of the fault trees will be permitted provided a written basis is provided. This basis will present the rationale why no coupling of cutsets or event sequences is expected from further development of the tree.

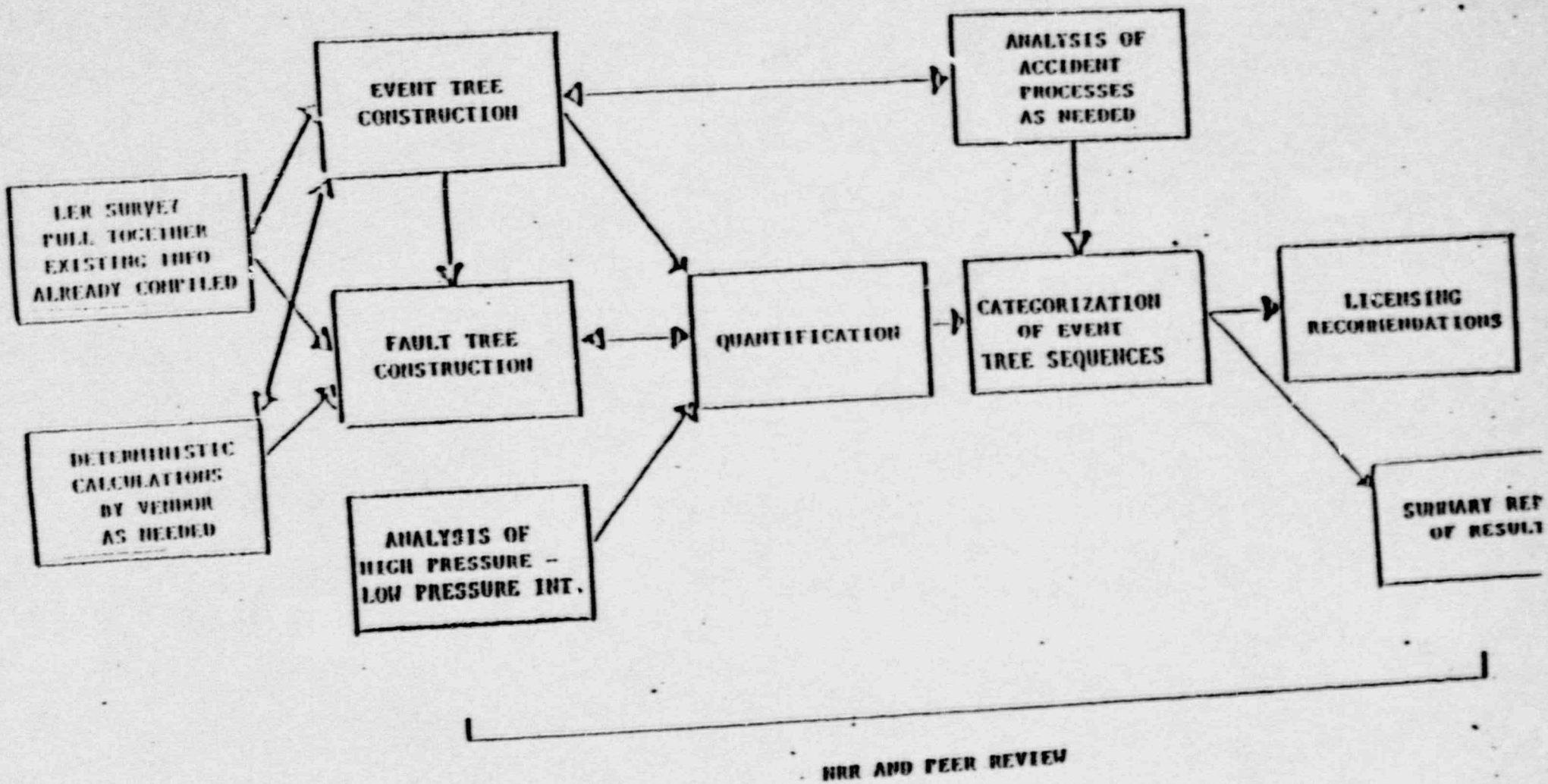
4. An investigation of the adequacy of high pressure-low pressure interfaces.

5. Analysis of the physical phenomena associated with dominant sequences to obtain estimates of the magnitude of releases from the containment. This will aid in categorizing releases into appropriate release categories.

To conduct a program of this magnitude in a short time period, delays associated with acquiring and transferring information must be minimized. Optimally, the event tree and fault tree analysts should share a common location during the initial portion of the project. As the fault trees progress below the top logic, however, the analysts should be located at or near the site with immediate access to as-built drawings and procedures as well as a representative of the plant operations staff. This will permit verification of engineering and procedural details and will minimize information transfer and print reproduction. Access should also be arranged between the fault tree analysts at the site, the remaining team in Bethesda, the architect-engineer, and the vendor.

In addition to basic plant data, deterministic calculations may be required to understand the behavior of the plant under off-normal conditions. This may also involve real-time simulation at an appropriate simulator to the extent possible. The arrangements with the vendor should cover this possibility and it may be desirable to have confirmatory calculations made by one of the NRC contractors on a selected basis

FIGURE 1


LER SURVEY
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BY VENDOR
AS NEEDED
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CONSTRUCTION
FAULT TREE
CONSTRUCTION
ANALYSIS OF
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LOW PRESSURE INT.
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ANALYSIS OF
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PROCESSES
AS NEEDED
CATEGORIZATION
OF EVENT
TREE SEQUENCES
LICENSING
RECOMMENDATIONS
SUMMARY REP
OF RESULT
NRR AND PEER REVIEW

ENCLOSURE 3

PRELIMINARY IDENTIFICATION OF SYSTEMS AND COMPONENTS THAT MAY BE IMPACTED BY DESIGN CHANGES

HPI System

EFW System

DHR System

CFT System

RCS Pressure Control System

Makeup/Letdown System

SG Pressure Control System

Steam Generator

Pressurizer

Quench Tank

Control Room Layout

RCS Piping

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

October 25, 1979

Docket No.: 50-438
and 50-439

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500 Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

SUBJECT: 10 CFR 50.54 REQUEST REGARDING THE DESIGN ADEQUACY OF BABCOCK & WILCOX NUCLEAR STEAM SUPPLY SYSTEMS UTILIZING ONCE THROUGH STEAM GENERATORS (BELLEFONTE NUCLEAR PLANT)

Several hardware and procedural changes have been made to operating B&W plants to reduce the likelihood of recurrence of a TMI-type accident. These changes have been in the area of auxiliary feedwater systems, integrated control system, reactor protection system, small-break loss-of-coolant accident analysis and operator training and procedures. However, at this time, we are beginning to look more deeply into additional design features of B&W plants to consider if any further system modifications are necessary.

The use of once-through-steam-generators (OTSG) in B&W plants has an operational advantage in that it provides a small degree of steam superheat, as contrasted with the conventional saturated U-tube steam generator. In addition, it provides for less water inventory thus making a steam line break less severe. However, the relatively low water inventory with the presence of a liquid-vapor heat transfer interface in the active heat transfer zone closely couples the primary system to the steam generator conditions with a consequently high sensitivity to feedwater-flow rate perturbations. Enclosure 1 to this letter addresses system problems and staff concerns in this area. At present, we are investigating whether B&W plants are overlysensitive to feedwater transients, due to the OTSG concept, as coupled with the pressurizer sizing, ICS design, and PORV/reactor trip set points.

As part of the post TMI-2 effort, detailed analyses have been made of undercooling transients for B&W plants. However, due to the sensitivity of the OTSG design, B&W plants have also been experiencing a number of relatively severe overcooling events.

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For your information, NRC is initiating a research task to quantitatively assess B&W system designs, including the integrated control system, aimed at identifying obvious accident sequences leading to core damage having a high frequency as compared to the Reactor Safety Study, see Enclosure 2. (A complete determination of risk will not be attempted). The objective of this assessment is to identify high-risk accident sequences (including TMI implications) utilizing event tree and simplified fault tree analyses. Included will be estimation of release categories, approximate quantification of expected frequency of selected event sequences and sensitivity studies for reliability of operator response. The study will focus on the risk implications of the sensitivity of the B&W design and on the potential interactions arising from the integrated control system. We estimate this study to be completed in about six months. We will use the Crystal River, Unit 3 plant as the referenced facility to be analyzed.

We have been holding generic discussions with Babcock and Wilcox Company concerning this matter. However, system sensitivity to feedwater transients involves balance-of-plant equipment and systems as well as the nuclear steam supply system, and such plant-specific characteristics must be considered.

We are also considering whether it is necessary to halt portions of the construction of B&W plants, pending the outcome of the reliability assessment. As a preliminary consideration, we have identified those systems and components that may be impacted by possible design changes as a result of this study. Enclosure 3 is a preliminary listing of such systems and components.

Under the authority of Section 182 of the Atomic Energy Act of 1954, as amended, and Section 50.54(f) of 10 CFR Part 50, additional information is requested to allow us to determine whether it is necessary to halt all or portions of the construction of your plant pending the results of our study. We request you provide:

a) Identify the most severe overcooling events (considering both anticipated transients and accidents) which could occur at your facility. These should be the events which causes the greatest inventory shrinkage. Under the guidelines that no operator action occurs before 10 minutes, and only safety systems can be used to mitigate the event, each licensee should show that the core remains adequately cooled.

b) Identify whether action of the ECCS or RPS (or operator action) is necessary to protect the core following the most severe overcooling transient identified. If these systems are required, you should show that its design criterion for the number of actuation cycles is adequate, considering arrival rates for excessive cooling transients.

c) Provide a schedule of completion of installation of the identified systems and components.

d) Identify the feasibility of halting installation of these systems and components as compared to the feasibility of completing installation and then effecting significant changes in these systems and components.

e) Comment on the OTSG sensitivity to feedwater transients.

f) Provide recommendations on hardware and procedural changes related to the need for and methods for damping primary system sensitivity to perturbations in the OTSG. Include details on any design adequacy studies you have done or have in progress.

We are sending similar letters to all utilities holding construction permits for plants with B&W nuclear steam supply systems.

We request your reply by December 3, 1979. We believe that a meeting with you and the other utilities together with the staff and the Babcock and Wilcox Company to discuss this matter would be beneficial to all parties. At that time, we will provide further details on the Crystal River Study. We are scheduling such a meeting for November 6, 1979 at 10:00 a.m. in Room P-422 at our offices in Bethesda, 7920 Norfolk Avenue, Bethesda, Maryland.

Please call Dr. Anthony Bournia at (301) 492-7200 if you have any questions concerning this letter.

Sincerely,

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: See next page

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