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

APR 18 1983

MEMORANDUM FOR: [redacted], Director
Division of Licensing

FROM: Roger J. Mattson, Director
Division of System Integration

Hugh L. Thompson, Jr., Director
Division of Human Factors Safety

SUBJECT: GENERIC FOLLOWUP EVALUATION TO BOARD NOTIFICATION BN-83-21 FOR B&W PLANTS

Reference:

1. Memorandum, Mattson and Thompson to Eisenhut, "Board Notification", dated February 18, 1983.

2. Memorandum, Mattson and Thompson to Eisenhut, "Followup Evaluation to Board Notification BN-83-21 for TMI-1", dated March 11, 1983.

Our memorandum of February 18, 1983 requested that you notify licensing boards associated with reactors designed by Babcock and Wilcox of new information involving auxiliary feedwater effectiveness. Our memorandum of March 11, 1983 provided our evaluation of this matter for TMI-1 which concluded that the information does not adversely affect our present conclusions regarding the ability of TMI-1 to achieve and maintain decay heat removal by natural circulation through the steam generators under transient and accident conditions.

We have now completed a generic evaluation for all B&W designed plants and have reached the same conclusion as for TMI-1. The generic evaluation is enclosed. We request that the generic evaluation be provided to the remaining licensing boards.

for Roger J. Mattson, Director
Division of Systems Integration

Hugh L. Thompson, Jr., Director
Division of Human Factors Safety

Enclosure:
As stated

cc: see next page

~~84~~ 8305110221

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ENCLOSURE

Background

On February 18, 1983, the staff issued Board Notification BN 83-21. This board notification identified information which had recently come to the attention of the staff that was potentially significant with respect to achieving and maintaining natural circulation in B&W-designed reactors. The genesis of the concerns was the staff review of the GPU-B&W lawsuit trial transcript. In this transcript, testimony by two individuals raised questions on two technical areas concerning natural circulation.

The Issues

The details of the technical issues identified were discussed in BN 83-21 and are repeated here.

During the trial, testimony by Dr. R. Lahey of Rensselaer Polytechnic Institute (RPI) and Dr. G. Wallis of Dartmouth College identified two concerns. These are (1) the adequacy of emergency operating procedures to assure that a sufficient condensing surface would be established in the steam generators under all design basis conditions for which decay heat removal by the steam generators was required and (2) the ability to establish an effective condensing surface at the elevation of the auxiliary feedwater sparger ring in light of new data which shows limited penetration into the tube bundle of feedwater entering the steam generator from the emergency feedwater sparger ring.

The first concern was raised by Dr. Lahey. It deals with procedures and relates to whether or not the operators have sufficient instructions and training to assure that they will raise the secondary level of the steam generator to 95 percent of the operating level under all conditions necessary to assure natural circulation. Following the TMI-2 accident, it was learned that the then current procedures instructed operators to raise the secondary level to 50 percent of the operating range. Under certain circumstances, it was possible to postulate that natural circulation would not be reestablished with the secondary level at 50 percent. Subsequently, it was determined that raising the level to 95 percent of the operating range would assure natural circulation if the RCS was saturated. However, because of overcooling considerations, it is not desirable to raise the level to 95 percent for all cases of loss of forced circulation. Thus, specific plant circumstances dictate the appropriate steam generator level and the manner to achieve this level. The operating procedures and training to describe the correct actions are, therefore, important to the issue.

A discussion of this issue was presented in NUREG-0565 ("Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock and Wilcox Designed 177-FA Operating Plants," dated January 1980) and was provided in Attachment 2 to BN-83-21. A copy of the relevant sections of Dr. Lahey's testimony was provided as Attachment 3 to BN-83-21. Dr. Wallis' testimony was provided to the Appeal Board and parties to the reopened TMI-1 restart proceding on March 8, 1983.

The second concern was raised by Dr. Wallis. It involves recent test data from the Alliance Research Center which shows that auxiliary feedwater entering from the sparger ring does not penetrate uniformly into the steam generator tube bundle but only contacts a small percentage of the tubes. This has the effect of lowering the elevation of the effective condensing surface in the steam generator. Previous analysis models assume good penetration of auxiliary feedwater spray into the tube bundle but recent B&W models account for the new data.

The issues raised by Dr. Lahey and Dr. Wallis are of concern only for B&W design plants with a lowered loop configuration. This arrangement is shown in Figure 1 and is applicable to ANO Unit 1, Oconee Units 1, 2 and 3; Crystal River Unit 3, TMI Units 1 and 2; Rancho Seco and Midland Units 1 and 2. Other plants designed by B&W have a raised loop configuration (see figure 2). This arrangement is utilized for Davis Besse Unit 1; Bellefonte Units 1 and 2 and WNP Unit 1. The Bellefonte and WNP Plants are further identified as 205 fuel assembly plants whereas all other B&W plants have 177 fuel assemblies.

For plants with the raised loop configuration the entire secondary level would provide a condensing surface above the core and operator action to raise the steam generator level from 50% on the operating range to 95% would not be required. The effectiveness of the condensing surface at the elevation of where auxiliary feedwater enters the steam generator would not be significant for plants the raised loop design since any auxiliary feedwater which did not provide heat transfer by boiling above the secondary level would act to increase the depth and therefore the effectiveness of the secondary level as a condensing surface. The Bellefonte and WNP plants have never taken credit for auxiliary feedwater effectiveness above the secondary level since AFW enters the bottom of the steam generators for these plants. The concerns raised by Dr. Lahey and Dr. Wallis are applicable only to plants with the lowered loop design and these plants are addressed genericly in the following NRC staff evaluation.

Staff Evaluation

1. EFW Spray Effectiveness

On February 23, 1983, the B&W Owners Regulatory Response Group (RRG) met with the staff to present information on the above two technical issues. Subsequent to this meeting, the owners group submitted a technical report, "Evaluation of SBLOCA Operating Procedures and Effectiveness of Emergency Feedwater Spray for B&W designed Operating NSSS," (Reference 1) which documented the information presented at the February 23, 1983 meeting. The staff has reviewed this report and our evaluation follows:

A. Effectiveness of Emergency Feedwater Spray

In the Once-Through-Steam-Generator (OTSG) design for the lowered loop plants, emergency feedwater enters the steam generator through seven nozzles located circumferentially around the OTSG shell and at an elevation just above the top tube support plate. This is shown in Figures 3 and 4, taken from the B&W report. Also shown on Figure 3 is the operating range for feedwater level (item 24). Analysis models used by B&W and the staff have assumed that emergency feedwater injected at the sparger elevation would be uniformly distributed within the tube bundle and provide effective heat transfer to all tubes within the bundle. Data obtained from testing performed by B&W at its Alliance Research Center (ARC) shows however, that the emergency feedwater spray does not effectively penetrate the tube bundle providing uniform wetting and uniform heat transfer throughout the bundle. Rather, the emergency feedwater only contacts those steam generator tubes in the immediate vicinity of the injection nozzle. The emergency feedwater would then pool on the tube support plate and spread out, draining down the flow holes where the steam generator tubes penetrate the tube support plate. As can be seen from Figure 3, at least 6 tube support plates exist between the injection location and the top of the operating range. In Figure 5, B&W shows the emergency feedwater axial wetting profile measured in Oconee 1. This shows that as the emergency feedwater drains down the tube bundle, the tube support plates tend to spread the flow towards the center of the tube bundle. The wetting profile could be envisioned as an "inverted cone." The impact of this incomplete wetting of the tubes is that the heat transfer rates attributed to spray cooling above the secondary side pool level can be reduced from what they were originally, based on the 100 percent wetting assumption.

In the B&W report, data and data correlations are presented which allow the percent of tube area wetted above the secondary side pool level to be calculated as a function of EFW flow. In the following section, the reliance on EFW spray effectiveness will be discussed.

Subsequent to the TMI-2 accident, the staff investigated the possible causes for natural circulation not being established in TMI-2 once the reactor coolant pumps were shut off. In reference 2, it was postulated that natural circulation did not commence following RCP trip because the secondary side steam generator level was not high enough to establish a condensing surface which would allow natural circulation flow over the pump entrance. In references 3 and 4, the staff further explained and documented this concern and concluded that in order to assure natural circulation, (in particular boiler-condenser), the secondary side steam generator level must be raised to above the elevation of the reactor coolant pump. B&W proposed raising the steam generator secondary level to 95 percent of the operating range in reference 5. By raising the level to 95 percent of the operating range, a condensing surface that is above the elevation of the pump and is sufficient to remove all decay heat in assured. By establishing a condensing surface above the pump elevation, condensation of primary steam and the buildup of a condensate level on the primary side of the tubes sufficient to promote flow over the pump inlet is also assured. In addition, because the liquid levels in the core, downcomer, and steam generator are equalized due to the vessel vent valves, the 95 percent secondary level assured that a condensing surface will be established before core uncovery could occur. In other words, to assure that natural circulation (boiler-condenser) would commence and reestablish decay heat removal before core uncovery could occur, no credit for EFW spray effectiveness needs to be taken. The EFW spray could be postulated not to provide any heat transfer, as long as it contributes to the secondary pool inventory. Thus, while the effect of the reduced heat transfer due to the reduced penetration of EFW spray during a SBLOCA would be to change the degree of initial overcooling and thus the initial system pressure response, the overall conclusions regarding core cooling would not change.

Inherent in this conclusion however, is the assumption that the pool level on the secondary side of the steam generator is raised to 95 percent on the operating range in a timely manner following a SBLOCA. Presently, the EFW is automatically controlled to establish the level at 50 percent of the operating range. Operator action is required to raise the level from the 50 to 95 percent level. To estimate the time available for the operator to initate actions to raise the EFW level to 95 percent, analysis by B&W in reference 5 shows that for the largest break size for which steam generator heat removal would be required,

(.01 sq ft.), boiler condenser heat transfer was calculated to commence after 25 minutes. Moreover, at the time boiler-condenser commenced, B&W analyses indicated 105,600 lb. would still remain above the core. If this amount of coolant were assumed to exit the primary system via the break (.01 sq ft.) as saturated liquid, it would still require at least an additional 35 minutes before core uncovery. This is a total of at least 60 minutes available to establish a condensing surface for the limiting break. If the level must be raised from the 50 percent to the 95 percent level and it takes approximately 1 minute to raise the level one foot*, then we estimated it would take approximately 12 minutes to raise the level to 95 percent. Therefore, there is estimated to be in excess of 13 minutes available for the operator to recognize the event and initiate filling of the secondary side of the steam generator with EFW to achieve the 95 percent level for the most limiting small break to establish boiler-condenser in the time period assumed in the B&W analysis. A still longer time would be available before core uncovery could occur.

*This estimate is used by B&W and has been confirmed by the staff.

Figure 6, which is reproduced from Figure 3-2 of the B&W submittal, provides the results of mass and energy balances which demonstrate that even with reduced EFW penetration, the EFW spray is still sufficient to remove decay heat.

The solid curve, labeled EFW spray, represents the point at which the overall heat transfer rates from EFW spray, combined with the primary to secondary temperature difference at the indicated RC pressure, can remove all of the decay heat at the indicated time. This curve is basically the energy balance requirement. The dashed lines represent the points at which the HPI flow can match the core boiloff at the prevailing pressure. The significance of this curve is that <u>prior to any core uncovery</u>, the break must uncover, and discharge steam. The source of the steam is the core boiloff; therefore the dashed line represents the point at which HPI flow will match break flow (the mass balance). The intersection of the two curves is the point at which both all decay heat can be removed by EFW spray <u>and</u> HPI can fully make up all mass loss through the break. These occur for all times beyond about 1000 seconds for the 100% HPI case and beyond 3000 seconds for the 70% HPI case.* It is therefore only necessary to show that core

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*The 70% HPI case refers to 1 HPI train with a 30% reduction assumed to result from spillage of HPI out the break.

uncovery does not occur for any small breaks less than .02 square ft (i.e., those breaks which require the steam generators for heat removal). As previously shown, core uncovery cannot occur prior to at least 60 minutes, or 3600 seconds, which is in considerable excess of seconds as indicated above.

2. Emergency Procedure and Operator Training Adequacy

The staff's board notification BN-83-21, dated February 18, 1983, stated the importance of operating procedures and operator training in assuring that a sufficient condensing surface is established in the steam generators under all design basis conditions. The concern raised by Dr. Lahey is whether or not operators have sufficient instructions and training to assure that the secondary level of the steam generators is increased to 95 percent of the operating range in a timely manner for all conditions necessary to assure natural circulation.

Subsequent to board notification of this issue, the staff met with members of the B&W Owner's Regulatory Response Group on February 23, 1983, to discuss design features, the emergency operating procedures and operator training. All owners of B&W design operating reactors were represented. The staff was informed by representatives of each plant that instructions necessary to bring the steam generator levels to 95 percent of the operating range were included in procedures and that the totality of training and procedures were sufficient to ensure that necessary operations would be performed. Moreover, the staff requested and obtained emergency operating procedures from Oconee, Arkansas Nuclear One, Rancho Seco and Crystal River to ascertain what instructions are provided relative to maintaining steam generators levels. The Davis-Besse plant procedures were not requested since the raised loop plants do not have the problem. The review included procedures for responding to loss-of-coolant and natural circulation. The procedure organization and formats were different among the plants and represented the individual owners' procedural philosophy. However, our review concluded that the procedures provide specific instructions to increase steam generator levels to 95 percent of the operating range when the reactor coolant pumps are tripped following a loss-of-coolant. It should be noted that Arkansas Power & Light Company's event-based emergency procedures have been replaced with emergency operating procedures based on the B&W Owners' Group Abnormal Transient Operating Guidelines. These new procedures do not require the determination that a loss-of-coolant be identified to enter the procedure. Instead the new emergency procedure is entered upon receipt of a scram. Plant symptoms direct the operator to the appropriate procedural steps for coping with the threat to loss of a safety function. The small break loss-of-coolant symptoms direct the operator to actions for loss of subcooling which requires as the first action reactor coolant pump trip. The operator is instructed to raise steam generator levels to 95 percent on the operating range as the second action.

In addition, the composite of each licensee's procedures contained instructions and guidance to assure the steam generator levels will be raised to 95 percent on the operating range for conditions of no forced cooling flow and existence of indications of superheat in the RCS.

Discussions with licensee representatives present at the February 18, 1983 meeting indicated that operators have been trained in the use of these procedures. The staff did not review the training programs in detail; however, licensee representatives stated that operator training on use of emergency operating procedures included instructions to remain in the appropriate emergency procedure in use unless the procedure specifically instructs the user to exit it or unless there is another valid reason to exit it. This gives further confidence that the operator will not exit a procedure prior to establishing 95 percent level in the steam generators, if required.

Based on the staff review of procedures and discussions with representatives of B&W designed operating plants regarding operator training, the staff concludes that there is reasonable assurance that the operator will increase steam generator level to 95 percent of the operating range under the conditions for which it is necessary to establish natural circulation.

Conclusions

The staff has evaluated information provided by the B&W owners group regarding the effect of reduced EFW penetration on decay heat removal capability for the reactors with lowered loops. Based on this evaluation, we have concluded that for the design basis SBLOCA scenarios, EFW spray cooling need not be relied upon to assure adequate decay heat removal, and decay heat removal solely by primary steam condensation in the region of the secondary side pool, which is at an elevation of 95 percent of the operating range, is adequate. This conclusion assumes correct operator action within approximately 10 to 15 minutes to initate raising the steam generator level to the 95 percent level. For scenarios not normally considered in the design basis, including delayed EFW, credit for EFW spray cooling is relied upon to assure core cooling. Analyses by B&W show that after accounting for the reduced EFW penetration into the steam generator tube bundle, the EFW spray cooling will still provide effective decay heat removal.

Based on staff review of the procedures for loss-of-coolant and natural circulation and discussions with representatives of the licensees for B&W designed plants with lowered loops regarding operating training, the staff concludes that there is reasonable assurance that the operators will increase steam generator levels to 95 percent of the operating range under the conditions for which it is necessary to establish natural circulation.

REFERENCES

1. "Evaluation of SBLOCA Operating Procedures and Effectiveness of Emergency Feedwater Spray for B&W-Designed Operating NSSS," B&W DOC. ID. 77-1141270-00 dated February, 1983

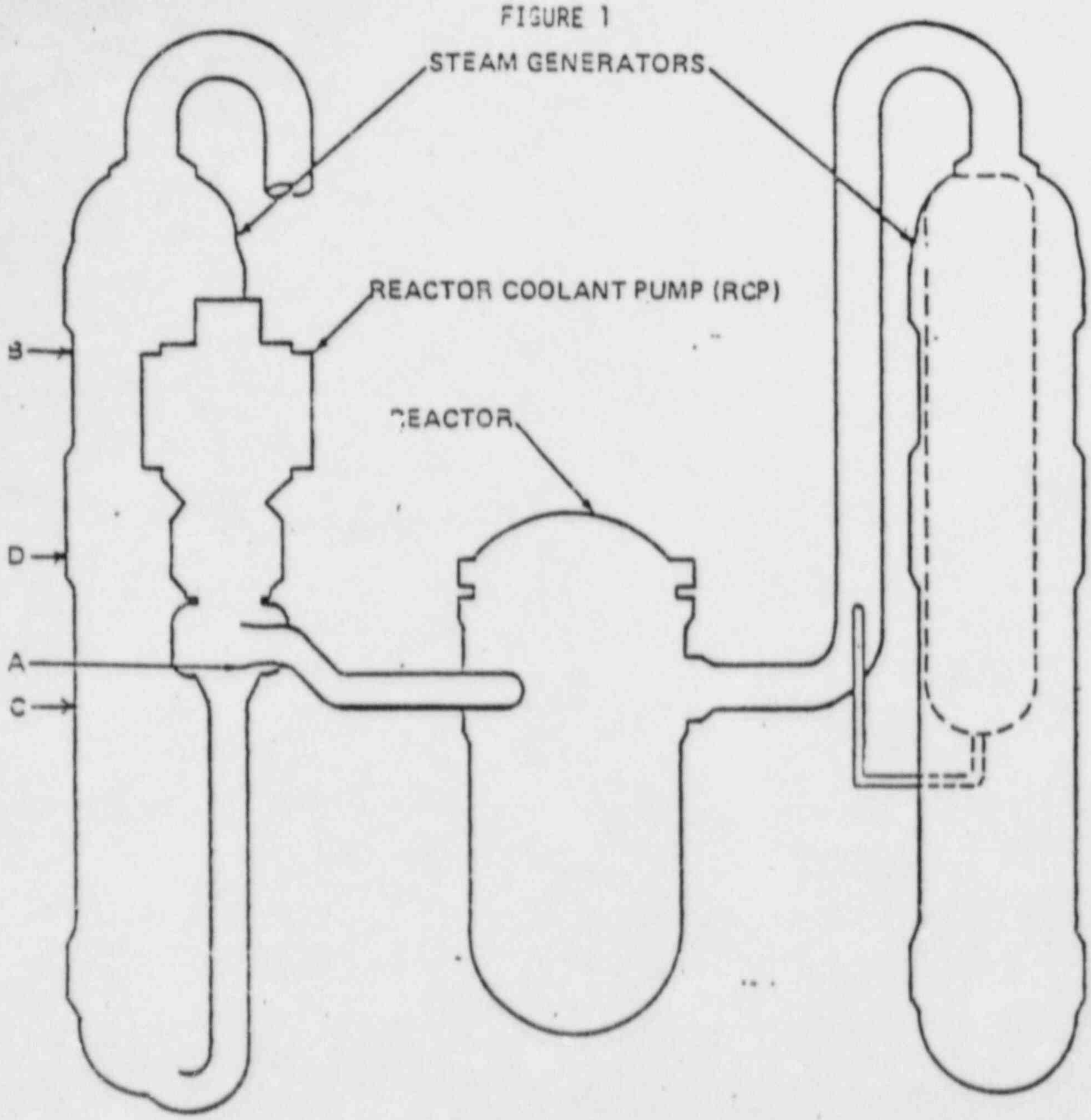
2. Memorandum, B. W. Sheron to Z. R. Rosztoczy "Pool Boiling -Condensation Natural Circulaticn In TMI-2," dated July 23, 1979.

3. "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock and Wilcox Designed 177-FA Operating Plants" NUREG-0565 dated January, 1980.

4. B. W. Sheron, "Generic Assessment of Delayed Reactor Coolant Pump Trip during Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors" NUREG-0623 dated November, 1979 (Appendix A).

5. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) "Evaluation of Transient BEhavior and Small Reactor Coolant System Breaks in the 177-FA Plant," Volumes I and II, dated May 7, 1979.

FIGURE 1

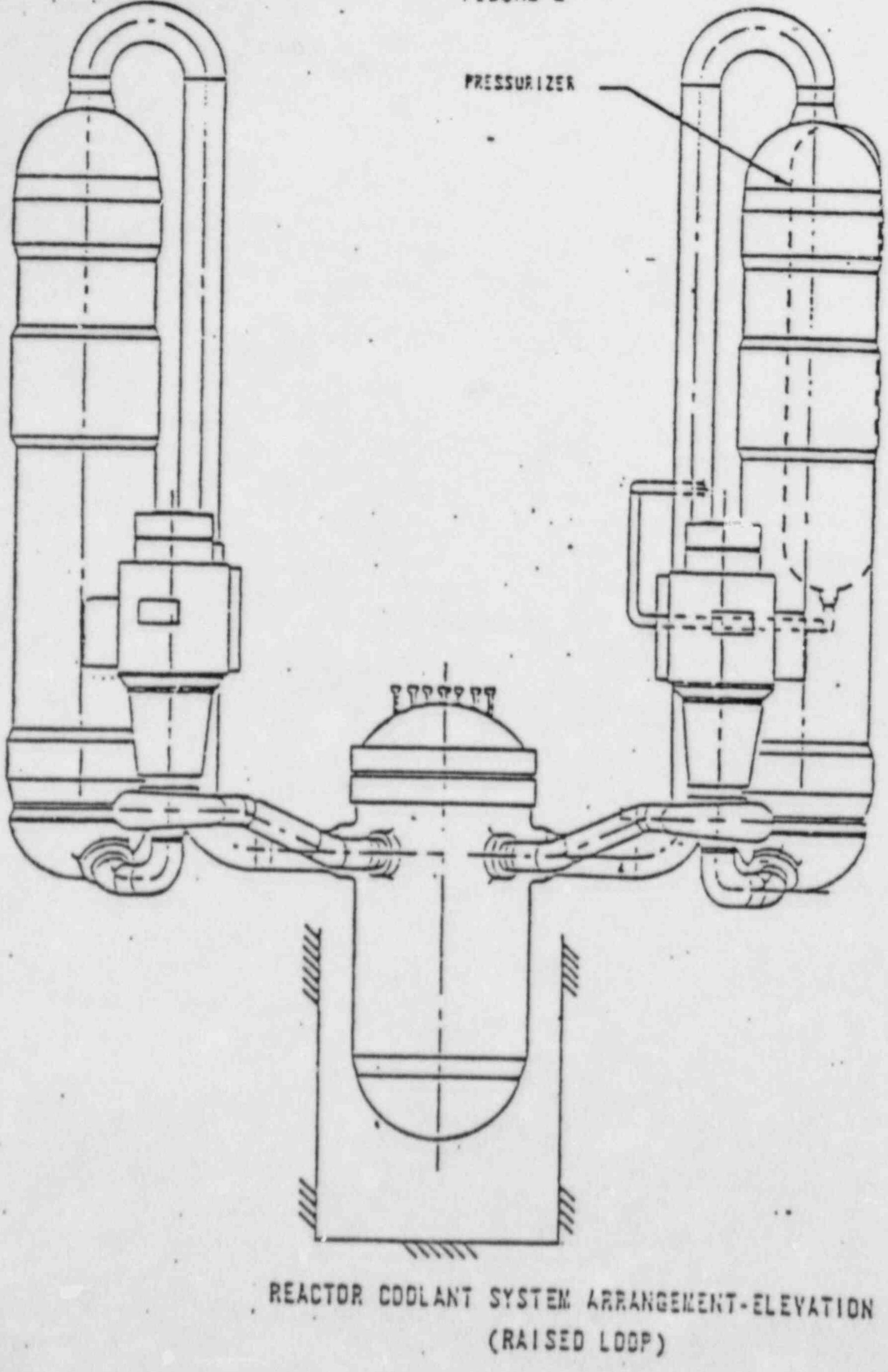


ELEVATION A – BOTTOM OF RCP DISCHARGE NOZZLE
ELEVATION B – APPROXIMATE ELEVATION OF AFW SPARGER
ELEVATION C – 50% OF OPERATING RANGE
ELEVATION D – 95% OF OPERATING RANGE

REACTOR COOLANT SYSTEM ARRANGEMENT

(LOWERED LOOP)

FIGURE 2



REACTOR COOLANT SYSTEM ARRANGEMENT-ELEVATION
(RAISED LOOP)

Figure 3 NUCLEAR ONCE-THROUGH STEAM GENERATOR (OTSG)

(From Reference 1)

1. PRIMARY INLET NOZZLE
2. PRIMARY OUTLET NOZZLE (2)
3. FEEDWATER HEADER
4. FEEDWATER SPRAY NOZZLES (32)
5. FEEDWATER HEATING CHAMBER
6. "BLEED" STEAM PORT
7. SATURATED FEEDWATER
8. PORTS
9. GENERATING TUBES (15,500)
10. DEPARTURE FROM NUCLEATE BOILING
11. 100% QUALITY
12. SUPERHEATED STEAM
13. STEAM ANNULUS
14. STEAM OUTLET NOZZLES (2)
15. LOWER SHELL
16. UPPER SHELL
17. LOWER TUBE SHEET
18. UPPER TUBE SHEET
19. ADJUSTABLE ORIFICE
20. EMERGENCY FEEDWATER INLET
21. TUBE SUPPORT PLATES (15)
22. CYLINDRICAL BAFFLE
23. STARTUP RANGE LEVEL
24. OPERATING RANGE LEVEL
25. WIDE RANGE LEVEL

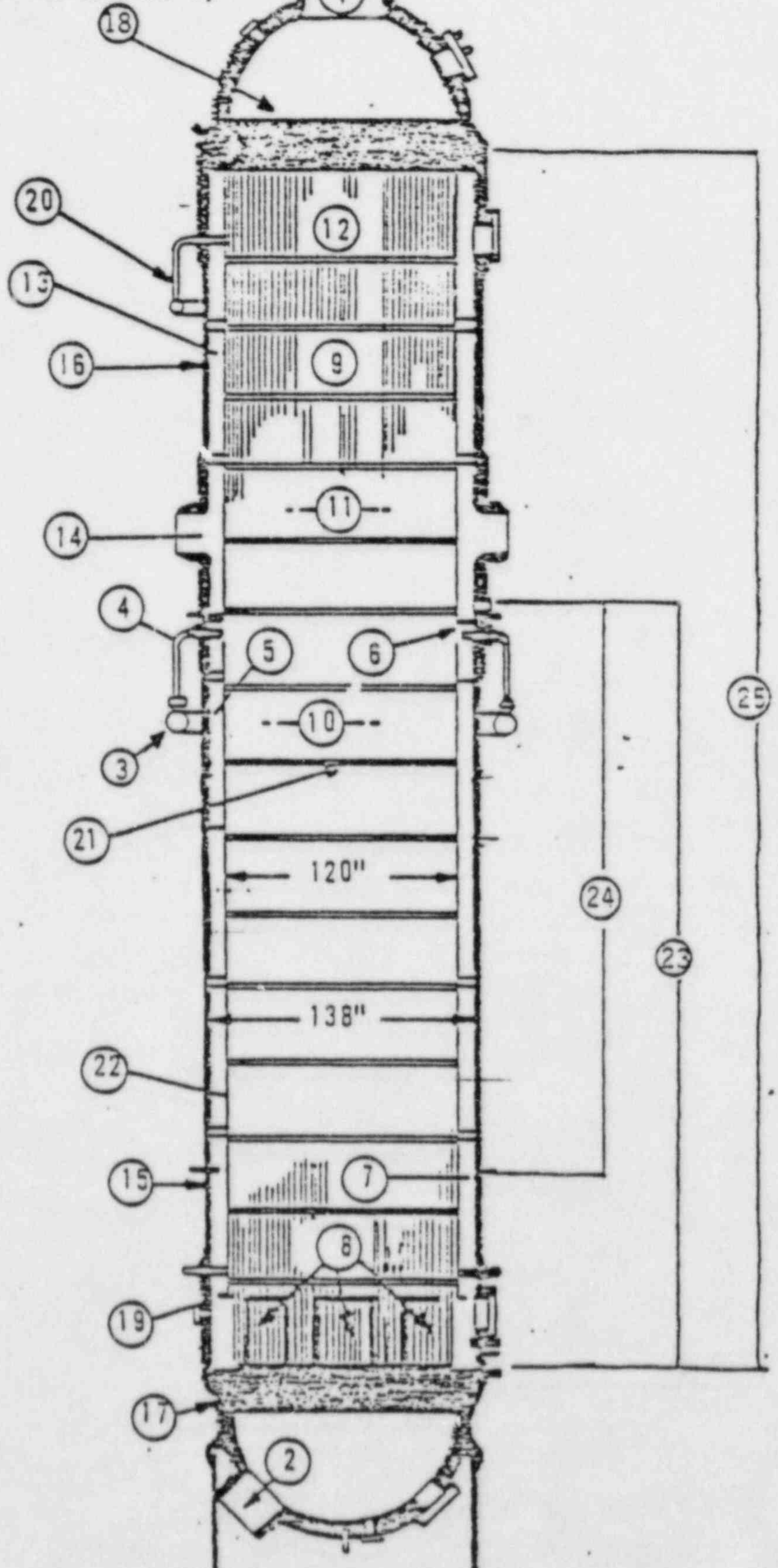


FIGURE 4 TYPICAL EMERGENCY FEEDWATER ARRANGEMENT

(From Reference 1)

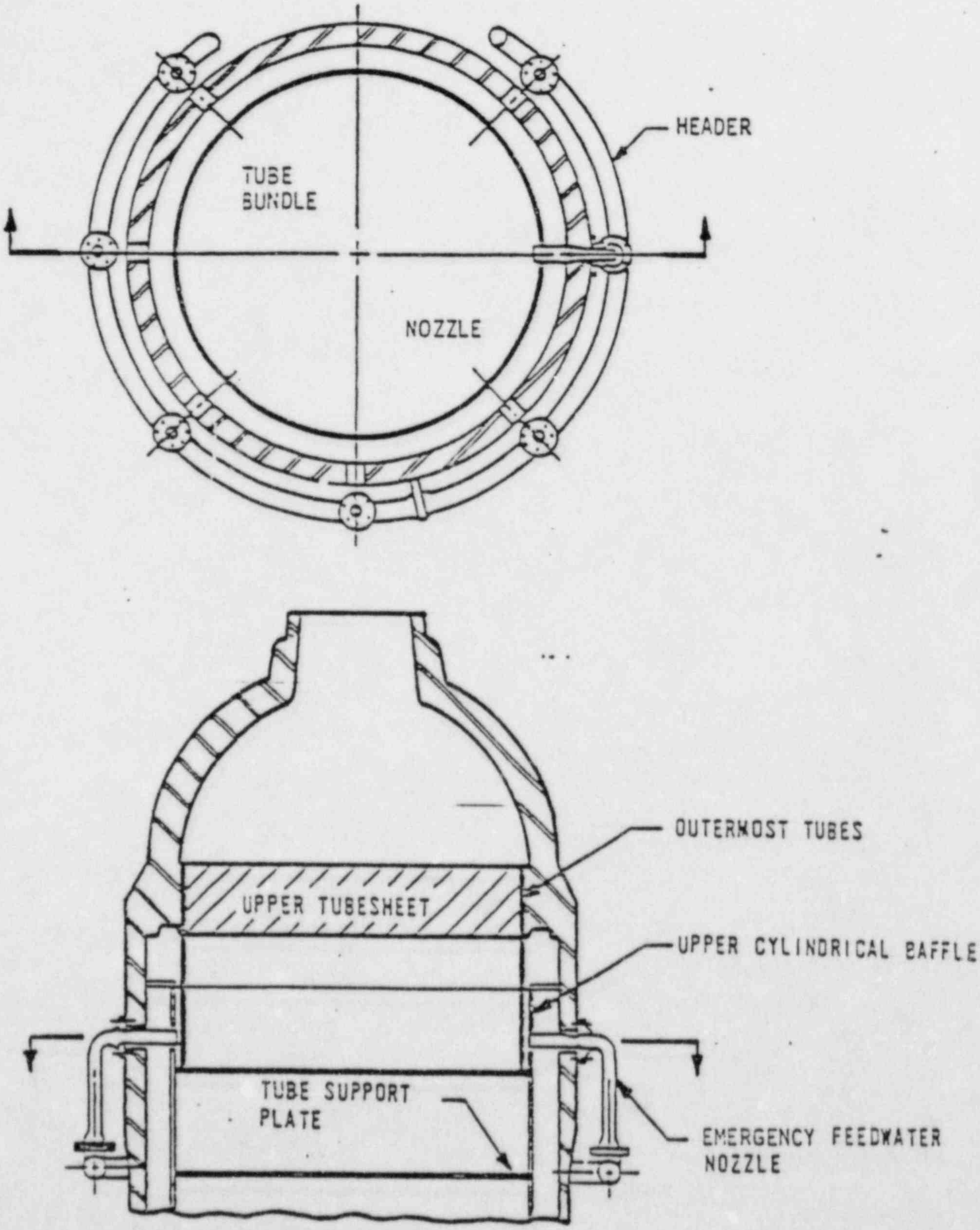


Figure 5 EFW AXIAL WETTING PROFILE
(OCONEE 1 - 1B OTSG)
(From Reference 1)

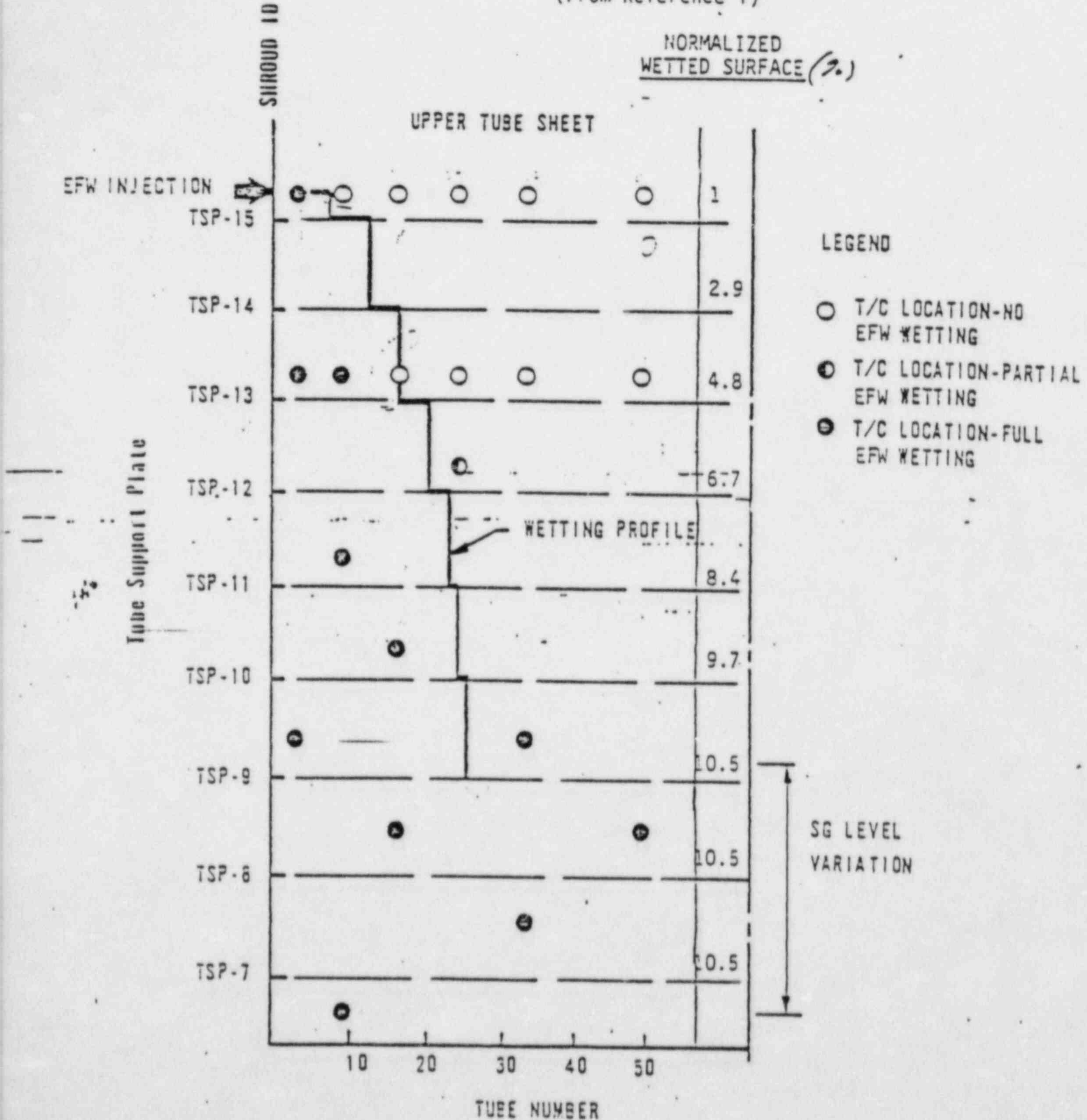
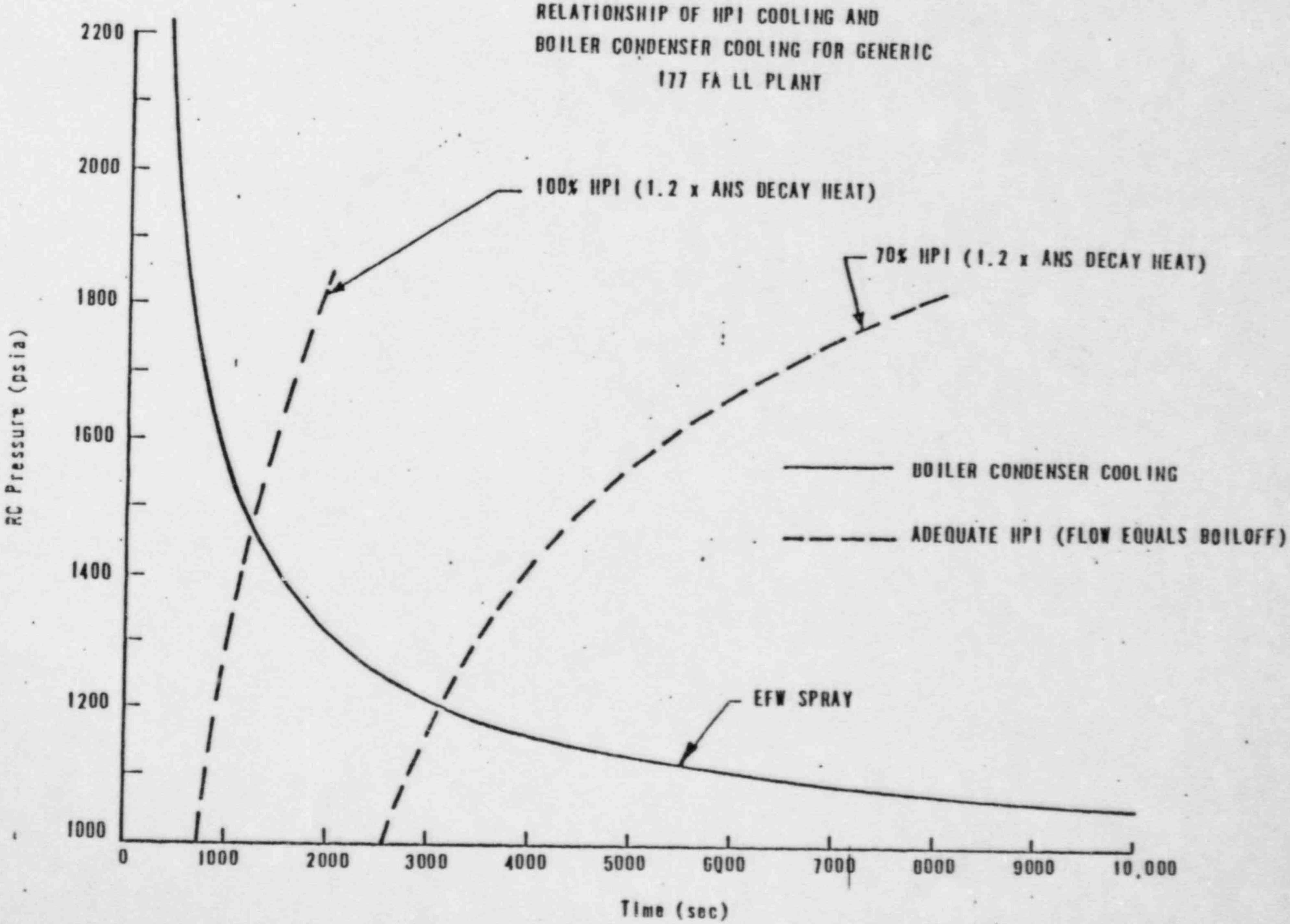


Figure 6
RELATIONSHIP OF HPI COOLING AND
BOILER CONDENSER COOLING FOR GENERIC
177 FA LL PLANT



Fil -Bd not Midland

MAY 2 4 1983

MEMORANDUM FOR: D. G. Eisenhut, Director, Division of Licensing, NRR

FROM: R. F. Warnick, Director, Office of Special Cases

SUBJECT: NOTIFICATION OF LICENSING BOARD

The Atomic Safety and Licensing Board (ASLB) was notified, during direct te ..mony by the Region III Staff on April 27, 1983, of new information which the Region III Staff perceived as being material and relevant to the Midland OM/OL proceeding. This new information concerned:

1. The licensee identified and reported two cracks in the Service Water Pump Structure (SWPS) that had reached the alert limit. These cracks were being investigated/evaluated by the licensee.

2. During shallow probing near the SWPS, the licensee drilled through a safety-related electrical duct bank.

3. The results of the load test on Pier 11-West were inconclusive in that the full load (applied at the top of the pier) was not reaching bearing stratum (bottom of the pier). Related problems encountered during the load test included a much greater skin friction than anticipated and the failure of the Carlson meters (load cells) to function properly.

If you have any questions or desire further information regarding this matter, please call me.

"Original signed by R. F. Warnick"

R. F. Warnick, Director
Office of Special Cases

cc: See attached distribution
list

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OFFICE	RIII RNL	RFW	JJH	RFW				
SURNAME	Landsman/ls	Gardner	Harrison	Warnick				
DATE	5/20/83							

NRC FORM 318 (10-80) NRCM 0240

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The Honorable Jerry Harbour, ASLB
The Honorable Frederick P. Cowan, ASLB
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(2) File BN

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

MAY 13 1983

MEMORANDUM FOR: D. G. Eisenhut, Director, Division of Licensing, NRR

FROM: R. F. Warnick, Director, Office of Special Cases

SUBJECT: RECOMMENDATION FOR NOTIFICATION OF LICENSING BOARD

In accordance with present NRC procedures regarding Board Notifications, the following information is being provided as constituting new information relevant and material to the Midland OM/OL proceedings. This information deals with the licensee's May 9, 1983 decision to stop Remedial Soils work due to violations of applied Hold Tag requirements.

The pertinent facts that relate to the stop work are as follows:

1. On May 6, 1983, MPQAD issued a nonconformance report (NCR) to document drift set deficiencies identified on previous Remedial Soils installations. As a result of the NCR, Hold Tags were applied.

2. On May 7, 1983, MPQAD issued an NCR to document drift set deficiencies identified during installation of pier KC-2 (East). As a result of the NCR, Hold Tags were applied.

3. On May 9, 1983, the licensee determined that work had continued on pier KC-2 (East) despite the presence of the Hold Tags. An additional NCR was issued to document the Hold Tag violations. At noon on May 9, 1983, the Field Soils Organization (FSO) stopped Remedial Soils work activities due to the Hold Tag violations. Although a formal Stop Work Order was not issued, 53 workers were sent home.

4. At 8:00 a.m. on May 10, 1983, the licensee resumed Remedial Soils work activities. The resumption of work was allowed after a resolution of differences between MPQAD and FSO pertaining to the significance of NCR's and Hold Tags. The NRC was informed of the Remedial Soils stop work by Stone and Webster (S&W) personnel during their meeting with the Midland Resident Inspectors to discuss the monthly S&W report of Remedial Soils work activities.

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If you have any questions or desire further information regarding this matter, please call me.

R. F. Warnick, Director
Office of Special Cases

cc: A. B. Davis
J. J. Harrison
R. N. Gardner
R. B. Landsman
R. J. Cook

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SURNAME	Gardner/ls	Harrison	Warnick				
DATE	5/12/83	5/12/83	5/12/83				

~~File~~
(2) File
Midland BN
& Zimmer

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

May 3, 1983

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Docket Nos. 50-528, 529, 530, 361, 289
362, 437, 275, 323, 483,
389 and 370

MEMORANDUM FOR: Chairman Palladino
Commissioner Gilinsky
Commissioner Ahearne
Commissioner Roberts
Commissioner Asselstine

FROM: Darrell G. Eisenhut, Director
Division of Licensing

SUBJECT: DIFFERING PROFESSIONAL OPINION REGARDING SYSTEMS INTERACTION AND SAFETY CLASSIFICATION (BN 83-57)

In accordance with present NRC procedures regarding Board Notifications, the enclosed is provided for your information. This information is applicable to all nuclear power plants.

This information relates to Board Notification 83-17 and 83-44, which were issued on February 18, 1983 and April 4, 1983, respectively. These dealt with the staff position regarding unresolved safety issue A-17. The enclosed differing professional opinion deals with certain aspects of existing policy and practice in the areas of systems interaction and safety classification.

The staff will keep you informed regarding the resolution of this differing professional opinion.

Darrell Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: OPE
OGC
EDO
SECY

Contact:
M. Williams, NRR
xt. 28285

MAY 12 1983

~~8305040323~~

cc: The Atomic Safety and Licensing Boards for:

Byron (Miller, Callihan, Cole)
Callaway (Gleason, Bright, Kline)
Comanche Peak 1 & 2 (Bloch, Jordan, McCollom)
Midland 1 & 2 (Bechhoefer, Cowan, Harbour)
Palo Verde 2 & 3 (Lazo, Callihan, Cole)
Perry 1 & 2 (Bloch, Bright, Kline)
San Onofre 2 & 3 (Kelley, Hand, Johnson)
Seabrook 1 & 2 (Hoyt, Harbour, Luebke)
Shoreham (Brenner, Carpenter, Morris)
Waterford 3 (Wolf, Foreman, Jordan)
Zimmer (Frye, Hooper, Livingston)
Indian Point (Gleason, Paris, Shon)

The Atomic Safety and Licensing Appeal Boards for:

Callaway 1 & 2 (Rosenthal, Edles, Gotchy)
Diablo Canyon 1 & 2 (Moore, Johnson, Buck)
Fermi 2 (Eilperin, Gotchy, Moore)
San Onofre 2 & 3 (Eilperin, Gotchy, Johnson)
Waterford 3 (Eilperin, Johnson, Kohl)
Zimmer (Rosenthal, Eilperin, Wilber)
TMI-1 (Edles, Buck, Gotchy, Kohl)
Rancho Seco (Rosenthal, Buck, Kohl)

DISTRIBUTION LIST FOR BOARD NOTIFICATION

Byron Units 1&2, Docket Nos. 50-454/455
Callaway Unit 1, Docket No. STN 50-483
Comanche Peak Units 1&2, Docket Nos. 50-445/446
Diablo Canyon Units 1&2, Docket Nos. 50-275/323
Fermi Unit 2, Docket No. 50-341
Indian Point Units 2&3, Docket Nos. 50-247/286
Midland Units 1&2, Docket Nos. 50-329/330
Palo Verde Units 2&3, Docket Nos. 50-529/530
Perry Units 1&2, Docket Nos. 50-440/441
Rancho Seco, Docket No. 50-312
San Onofre Units 2&3, Docket Nos. 50-361/362
Seabrook Units 1&2, Docket Nos. 50-443/444
Shoreham Unit 1, Docket No. 50-322
TMI Unit 1, Docket No. 50-289
Waterford Unit 3, Docket No. 50-382
Zimmer Unit 1, Docket No. 50-358

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