June 22, 1983

ADJUDICATORY ISSUE

SECY-83-246

(Affirmation)

For: The Commission

From: Herzel H. E. Plaine
General Counsel

Subject: REVIEW OF ALAB-724 -- IN THE MATTER OF METROPOLITAN EDISON COMPANY

Facility: Three Mile Island Nuclear Station, Unit No. 1

Purpose: To advise the Commission of an Appeal Board decision which, in the General Counsel's view, [redacted]

Ex 5

Review Time Expires: July 15, 1983, as extended

Petitions for Review: None

Discussion: In ALAB-724 the Appeal Board raised two safety concerns which, in the Appeal Board's view, "require careful and prompt consideration:"

> (1) whether the power operated relief valve (PORV) should be safety-grade because of the potential for using it to mitigate the consequences of design basis steam generator tube rupture accidents; and

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(2) a corrosive problem with the PORVs that could result in the PORV not functioning when needed.

The Board found, however, that these matters were outside the scope of the proceeding and therefore that it could not pursue them.

The Appeal Board apparently issued ALAB-724, dated April 20, 1983, to notify the Commission and the parties of its concerns with these two issues and to state that its decision on design issues would be premised on the assumption that these problems would be resolved outside the adjudicatory context. The Appeal Board did not specify whether it believed the issues needed to be resolved prior to restart.[1]

We will address the concerns raised by the Appeal Board, in reverse order, to determine (1) whether the Appeal Board erred in finding that the concerns are outside the scope of the proceeding; and (2) what the Commission's options are and the implications of those options for the Commission's immediate effectiveness decision in light of the Appeal Board's actions with regard to each concern.

I. <u>Corrosive Problem With PORVs</u>

The Appeal Board's concern with corrosion in the PORVs arose from two licensee event reports (LER 82-011199X-0 and LER 83-003/01T-0) which indicated that the last two PORVs removed from TMI-1 were heavily corroded and probably would not have functioned if needed.

The scope of the restart proceeding was limited to incidents having a nexus to the TMI-2 accident. [We believe] Ex.5

[1]The Appeal Board issued its design/hardware decision (ALAB-729) on May 26, 1983, and again did not specify whether it believed these issues needed to be resolved prior to restart.

EX.5

we recommend that

II. Whether PORVs Should be Fully Safety-Grade

The Appeal Board's other concern arose from Board Notification (BN) 83-47. In BN-83-47 the staff concluded that PWRs need "a capability for rapid primary system depressurization ... in order to effectively mitigate the design basis steam generator tube rupture accident," and that "the components and systems to provide this depressurization capability should be safety-grade." As staff explained, where reactor coolant pump (RCP) flow is lost, Westinghouse and B&W plants rely

x.5

on the pressurizer PORVs to accomplish this depressurization, and the pressurizer PORVs in most plants are not safety-grade.[3] Although staff further indicated that this depressurization function could be accomplished by the auxiliary pressurizer spray system, staff stated in BN-83-47 that the spray system is not safety-grade at TMI and, regardless, "the pressurizer PORV would be the preferred means of depressurizing the RCS" (reactor coolant system).

Staff issued BN-83-47 to bring to the attention of the licensing boards this reliance on a non-safety-grade piece of equipment to mitigate a design basis accident. Staff stated in BN-83-47 that it will require "for OLs now under review" that the PORVs and their ancillary systems be fully safety-grade and environmentally qualified. The extent of future staff actions with regard to operating reactors is under consideration.[4]

The Appeal Board interpreted BN-83-47 as a shift from staff's position in the restart proceeding, accepted by the Licensing Board, that the PORV had to meet only those safety-grade criteria applicable to its role as part of the reactor coolant system pressure boundary. The Appeal Board stated that given "the NRC staff conclusion that, in the event of a steam generator tube rupture in some ... plants (including TMI-1), the best accident mitigation procedure is to depressurize the primary system rapidly by use of the ... [PORV], the staff now believes that the PORV

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[3]In steam generator tube rupture scenarios where the RCPs remain operational, this reactor coolant system depressurization function is accomplished by the non-safety-grade normal pressurizer spray as well as the turbine steam bypass system, both of which rely on offsite power.

EX 5

must meet all safety-grade criteria." ALAB-724 at 2-3.[5]

A. Whether concern raised by BN-83-47 is outside scope of hearing

We believe that

X.5

[5]We believe that

X.5

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6 The licensee in its April 20, 1983 response to ALAB-724

E1.5

indicated that the recently installed safety-grade pressurizer vent might fulfill the requirement for rapid primary system depressurization, and, regardless, that the TMI-1 steam generators and steam lines can be flooded to terminate the leakage.

EX.5

we

believe that

Conclusion

We recommend

X.5

we

believe that

EX.5

We believe

Herzel H. E. Plaine
General Counsel

Attachments:
1. Draft order
2. Draft Staff requirements memo
3. ALAB-724
4. BN-83-47

Commissioners' comments should be provided directly to the Office of the Secretary by c.o.b. Wednesday, July 13, 1983.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Wednesday, July 6, 1983, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

This paper is tentatively scheduled for affirmation at an Open Meeting during the Week of July 11, 1983. Please refer to the appropriate Weekly Commission Schedule, when published, for a specific date and time.

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Enclosure 1

Enclosure 2

Enclosure 3

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Release

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Gary J. Edles, Chairman
Dr. John H. Buck
Dr. Reginald L. Gotchy

In the Matter of METROPOLITAN EDISON COMPANY, <u>ET AL</u>. (Three Mile Island Nuclear Station, Unit No. 1)	)	Docket No. 50-289 (Design Issues)

<u>MEMORANDUM</u>

April 20, 1983

(ALAB-724)

We have before us appeals from a Licensing Board decision disposing of various issues regarding plant design and procedures in connection with the proposed restart of Unit 1 at the Three Mile Island Nuclear Station. LBP-81-59, 14 NRC 1211 (1981). The design and procedures issues are limited to those that have a nexus either to the specific TMI-2 accident (<u>i.e.</u>, an accident involving a loss of main feedwater or a small break loss of coolant) or to questions which that accident raised about whether TMI-1 could be operated safely. [1/] On at least one occasion on which we

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1/ See CLI-79-8, 10 NRC 141 (1979) and ALAB-705, 16 NRC ___, ___ (December 10, 1982)(slip opinion at 21-22). <u>Cf.</u> CLI-83-5, 17 NRC ___ (March 7, 1983).

sought permission to pursue safety questions not having such a nexus, the Commission, as is its prerogative, elected to pursue those questions itself. 2/

Two safety matters have recently surfaced which we believe require careful and prompt consideration. For the following reasons, we have decided that we are unable to pursue them but that they should nonetheless be brought to the Commission's attention now.

On April 12, 1983, we received "Board Notification Regarding the Need for Rapid Primary System Depressurization Capability in PWRs" (BN-83-47), dated April 4, 1983. This notification concerned a memorandum from Roger J. Mattson, Director, Division of Systems Integration, Office of Nuclear Reactor Regulation, which presented the NRC staff conclusion that, in the event of a steam generator tube rupture in some Westinghouse or Babcock & Wilcox designed plants (including TMI-1), the best accident mitigation procedure is to depressurize the primary system rapidly by use of the power-operated relief valve (PORV). Given that conclusion, the staff now believes that the PORV must meet all

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2/ CLI-82-12, 16 NRC ___ (July 16, 1982). Our authority to consider new matters is discussed in ALAB-685, 16 NRC ___, ___ n.5 (August 2, 1982) (slip opinion at 6 n.5 and accompanying text).

safety-grade criteria. This constitutes a shift from the staff's previous position on appeal. 3/.

Since receiving BN-83-47, we have noted two licensee event reports from H.D. Hukill, Vice-President, GPU Nuclear Corporation, to R.C. Haynes, Regional Administrator for Region I. These publicly available reports, dated October 28, 1982 (LER 82-011/99X-0) and March 7, 1983 (LER 83-003/01T-0), indicate that the last two PORV valves removed from TMI-1 (the first in the summer of 1981 and its replacement in February of 1983) were found to be heavily corroded and probably would not have functioned if they had been needed. According to the reports, the corrosion appeared to be due to sulphur; elemental sulphur was found in the replacement valve.

As noted above, our appellate review is necessarily limited to the requirements for the use of the PORV in incidents that have a nexus to the TMI-2 accident. In this connection, the Licensing Board found that the PORV need meet only those safety-grade design criteria applicable to its role as part of the reactor coolant system pressure boundary. 14 NRC at 1282. Although the intervenor Union of Concerned Scientists has raised in general terms the

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3/ See NRC Staff's Brief in Response to the Exceptions of Others to the Licensing Board's Partial Initial Decision on Plant Design and Procedures, Separation and Emergency Planning Issues (May 20, 1982) at 23-24.

argument that the PORV must meet all safety-grade criteria, we do not believe that we may evaluate the issues raised in BN-83-47 because they stem from matters outside the scope of this proceeding. 4/

The Commission, however, is examining a number of safety issues as part of its immediate effectiveness review, including issues outside the hearing record such as those associated with the steam generators. It should be reasonably close to making a decision as to whether or not TMI-1 should be allowed to resume operations. 5/ In our view, the issues raised in BN-83-47 and the licensee event reports have a direct nexus to the steam generator issues

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4/ Quite apart from that controlling consideration, it is worthy of note that this proceeding has already been the subject of lengthy adjudicatory hearings and ongoing appeals. We heard oral argument on September 1, 1982. On December 29, following the receipt of several Board notifications and extensive comments by the parties, we were compelled to reopen the record in order to clarify various inconsistencies in the parties' positions and the testimony regarding certain methods of decay heat removal. We held four days of evidentiary hearings between March 7 and 17, 1983, supplemental briefs were filed on April 12, and our decision is nearing completion. No party has asked us to reopen the record to examine matters raised in BN-83-47, and we now anticipate that we will be able to issue our decision no later than May 31, 1983.

5/ Under present Commission practice, a licensing board decision authorizing the commencement (or, in this case, resumption) of operations is reviewed by the Commission to determine whether it shall become effective pending administrative appellate review. See generally 10 CFR §2.764(b).

now under review by the Commission and should be explored before restart. We are also concerned that the corrosion problem noted with respect to the PORVs may also affect the safety relief valves, particularly now that the loop seals have been eliminated.

In sum, we believe it desirable to alert the parties and the Commission as promptly as possible to the matters which we have noted above. In addition, we think it useful to announce our intention to premise any decision we may reach with regard to design issues on the assumption that the problems involving use of the PORV during steam generator tube break accidents and the corrosive contamination present will be resolved outside the adjudicatory context.

FOR THE APPEAL BOARD

C. Jean Shoemaker

C. Jean Shoemaker
Secretary to the
Appeal Board

Enclosure 4

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 4, 1983

Release

'83 APR 11 P1:52

Docket Nos: 50-522, 529, 530
50-289, 361, 362
50-437, 389, 370
50-275/323, 483

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

FROM: Darrell G. Eisenhut, Director
Division of Licensing

SUBJECT: BOARD NOTIFICATION REGARDING THE NEED FOR RAPID PRIMARY SYSTEM DEPRESSURIZATION CAPABILITY IN PWR's (BN-83-47)

In accordance with the procedures for Board Notification, the enclosed information is being transmitted directly to the Commission.

The Division of Systems Integration (DSI) of NRR has recently concluded that a capability for rapid primary system depressurization is needed for PWR's in order to effectively mitigate the design basis steam generator tube rupture accident. Furthermore, DSI has concluded that the components and systems to provide this depressurization capability should be safety-grade.

The enclosed memorandum states that CE plants have a safety-grade auxiliary pressurizer spray to provide such capability but Westinghouse and B&W plants (except Midland) do not have a safety-grade capability. Therefore, this position affects prior staff positions in the safety evaluations for these facilities. The staff actions on this issue regarding operating reactors are under consideration.

Darrell G. Eisenhut, Director
Division of Licensing

Enclosure:
As Stated

cc: SECY
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ASLB and Parties to the Proceedings for:

Byron Units 1 & 2 (Smith, Callihan, Cole) (50-454/455)
Callaway, Units 1 & 2 (Gleason, Bright, Kline) (50-483/486)
Midland Units 1 & 2 (Bechhoefer, Cowan, Harbour) (50-329/330)
Waterford Unit 3 (Wolfe, Foreman, Jordan) (50-382)
San Onofre Nuclear Power Units 2 & 3 (Kelley, Hand, Johnson) (50-361/362)
Rancho Seco (Cole, Shon)
Indian Point (Gleason, Paris, Shon) (50-247/286)
Comanche Peak Units 1 & 2 (Miller, Jordan, McCollom) (50-445/446)
Palo Verde 1,2,3 (Lazo, Callihan, Cole)(50-528, 529, 530)

ASLAB and Parties to the Proceedings for:

Callaway, Units 1 & 2 (Rosenthal, Edles, Gotchy) (50-483/486)
Waterford Unit 3 (Eilperin, Johnson, Kohl) (50-382)
Diablo Canyon 1 & 2 (Moore, Johnson, Buck) (50-275/323)
Palo Verde 1,2,3 (Rosenthal, Eilperin, Wilber) (50-528, 529, 530)
San Onofre 2 & 3 (Eilperin, Gotchy, Johnson) (50-361/362)
Rancho Seco (Rosenthal, Buck, Kohl) (50-312)
Point Beach 1 & 2 (Moore, Gotchy, Johnson) (50-266/301)
TMI-1 (Edles, Buck, Gotchy, Kohl) (50-289)

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

April 4, 1983

ORB#4 DISTRIBUTION FOR BOARD NOTIFICATION NO. 83-47

RE: need for rapid primary sys. depressurization capability in PWRs

Docket File
NRC PDR
L PDR
ORB#4 Rdg
ORB#4 Memo File
ORB#4 Board Notification File
HDenton
ECase
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Judge Reginald L. Gotchy, ASLAP, EW-529
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Release

MAR 2 7 1983

MEMORANDUM FOR: Darrell Eisenhut, Director, Division of Licensing

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

SUBJECT: BOARD NOTIFICATION REGARDING PORVs

The purpose of this memorandum is to request you notify all licensing boards associated with PWRs designed by Westinghouse and Babcock and Wilcox of recent staff findings regarding reliance on PORVs for steam generator tube rupture mitigation.

Prior to the Ginna Steam Generator Tube Rupture (SGTR) in January of 1982, the thermal-hydraulic performance of SGTR events was not explicitly analyzed and presented in licensing submittals; neither was it reviewed.

Instead, the SGTR event was reviewed only for the radiological consequences and very general, unverified assumptions were made regarding the system performance. These performance assumptions were not usually questioned since the radiological assessment imposed a number of conservative assumptions. For example, a conservative iodine spiking value was assumed concurrent with the worst case meteorological conditions. Also, offsite power was assumed unavailable thus necessitating the release of steam directly to the environment.

Following the TMI-2 accident, the staff began to examine transient and accident events in more detail, particularly with respect to required operator actions and equipment availability and performance. Moreover, as a result of a reactor coolant pump seal water leak at H. B. Robinson Unit 2 on November 30, 1981, in which recovery was aggravated by malfunctioning pressurizer relief and block valves, the staff initiated an evaluation of the role of PORVs in accident management and mitigation. Finally, we specifically reviewed the role of PORVs in SGTR management and mitigation following the Ginna SGTR event of January, 1982.

Based on the three efforts, we have concluded that it is necessary for PWRs to have a rapid primary system depressurization capability in order to limit the primary to secondary leakage to the values predicted in the SGTR licensing analyses. In SGTR scenarios where the RCPs remain operational, this RCS depressurization function is accomplished by the non-safety grade normal pressurizer spray as well as the turbine steam bypass system, both of which rely on offsite power. However, in SGTR scenarios where the RCP flow is lost (i.e., loss of offsite power for compliance with GDC-17), depressurization must be accomplished by other means. Westinghouse and B&W plants rely on the pressurizer PORVs which are, in most plants, also non-safety grade. In plants designed by Combustion Engineering, this depressurization function would be accomplished by the safety grade auxiliary pressurizer spray system. Although Westinghouse and some B&W plants have an auxiliary pressurizer spray system, it is not safety grade. Moreover, for Westinghouse plants, we understand it is not recommended as the preferred means for RCS depressurization following a SGTR scenario in which the main pressurizer spray is not available.

It is this reliance on a non-safety grade piece of equipment to mitigate a design basis accident which we wish to bring to the attention of the licensing boards. It is our understanding that auxiliary pressurizer spray is a safety grade system for Midland, but not for TMI nor for most Westinghouse and B&W PWRs.

We also wish to bring to the attention of the boards an additional concern that has recently come to light regarding the SGTR for Babcock and Wilcox plants. Because of the once-through steam-generator (OTSG) design, it may not be possible to stop the primary-to-secondary leakage in a SGTR while maintaining the RCS in a subcooled state. The increased tendency for the OTSG leakage to continue throughout the event is a result of the tubes being directly exposed to the OTSG steam space. The increased leakage results in an increased tendency to overfill the OTSG and to add to the radiological consequences.* Generally, the emergency procedures instruct the operator to discharge steam to the atmosphere or, if available, to the condenser to control level in the damaged OTSG as necessary. In at least one plant, however, if the water supply for safety injection pumps is approaching a minimum level or if the offsite radiological consequences are becoming excessive, the OTSG is allowed to completely fill, thus terminating the leakage. The number of B&W plants that permit filling of the OTSG is not known at present. We do not believe the potential for prolonged leakage and the associated offsite radiological consequences have been factored into FSAR SGTR accident analyses, with the exception of Midland. The Midland license applicant has recently submitted a revised SGTR analysis taking into account prolonged leakage.

In conclusion, the staff believes a rapid depressurization technique is necessary for effective mitigation of the design basis SGTR accidents, and this function is accomplished by non-safety grade equipment in Westinghouse and B&W plants. If offsite power is lost, or if the RCPs are not operating, the pressurizer PORV would be the preferred means of depressurizing the RCS. However, for most operating plants and some near term OL plants, the PORV's are not fully safety grade, since

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*This facet of the OTSG design is described in the recent INPO evaluation of the Oconee 2 tube leak, INPO 82-030.

MAR 2 7 1983

the reliability, environmental qualification and operability for effective mitigation may not have been established according to our usual interpretation of the GDCs for other safety related equipment. For OLs now under review (excluding CE plants), the staff will require that the PORVs as well as ancillary mechanical and electrical systems be fully safety grade and environmentally qualified.

Further, as a result of the B&W steam generator design, sustained primary-to-secondary leakage may occur if the steam generator cannot be intentionally filled; however it is not known which B&W plants can - intentionally fill their OTSGs due to system valving, dead weight loads and water hammer considerations. This may add to the offsite radiological consequences beyond those considered in the FSAR SGTR safety analyses. As a result, we are requesting B&W applicants to address this potential for prolonged leakage and to modify their FSAR SGTR analyses if necessary. For operating plants, the staff is currently prioritizing a proposed staff action to evaluate the potential for and conseournces of SG overfill during SGTRs. The extent of future staff and operating reactor licensee actions to address this problem will be determined based on this prioritization.

Roger J. Mattson

Roger J. Mattson, Director
Division of Systems Integration

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