UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION
Harold R. Denton, Director

In the Matter of	)	Docket No. 50-289
	)	
GPU NUCLEAR CORPORATION	)	
	)	(10 CFR 2.206)
(Three Mile Island Nuclear Station, Unit 1)	)	

INTERIM DIRECTOR'S DECISION UNDER 10 CFR 2.206

I. INTRODUCTION

By Petition dated January 20, 1984 (Petition) and filed before the Commission on January 23, 1984, Ellyn R. Weiss and Robert D. Pollard, on behalf of the Union of Concerned Scientists (petitioner) requested that the Commission continue the suspension of the Three Mile Island Nuclear Station, Unit 1 (TMI-1) operating license "unless and until the plant's Emergency Feedwater (EFW) System complies with NRC rules applicable to systems important to safety (including safety-grade, safety-related, and engineered safety feature systems)." In support of its request, petitioner alleges five basic deficiencies with the EFW system for which petitioner seeks resolution prior to resuming power operation at TMI-1: (1) failure of the EFW system to be environmentally qualified; (2) failure of the EFW system to be seismically qualified; (3) the inability of the EFW system to withstand a single component failure; (4) the inaccuracy of the EFW flow instruments; and (5) the inadequacy of the Main Steam Line Rupture Detection System (MSLRDS). Petitioner recognized that one or more of the identified deficiencies, when viewed individually, would not necessarily pose an "intolerable risk." However, petitioner contended that "[i]n the aggregate. . . [the deficiencies] thoroughly compromise the reliability of one of the most important safety systems in the plant and destroy the fundamental principle of defense-in-depth espoused by the NRC."[1]

The Petition was referred to the staff on February 3, 1984 for treatment as a request for action pursuant to section 2.206 of the Commission's regulations. The licensee responded to the Petition pursuant to the staff's request under 10 CFR 50.54(f) on February 24, 1984, and amended its response on March 26, 1984. The Commission recently instructed the staff to complete its review of the petition with respect to those issues raised by the petitioners for which sufficient information was available to make a determination. Accordingly, the staff expedited its review of four of the issues raised by the petitioners. For the reasons stated herein, the staff does not intend to take the action requested by the petitioner with respect to those issues at this time. However, the staff has not yet reached a decision as to the issues raised by the petitioner concerning environmental qualification of the EFW system, and the aggregate effect of the five deficiencies cited by the petitioner on the reliability

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1 The Petition also implies that there may be deficiencies in emergency procedures and operator training related to the EFW system, but it does so only in passing and provides no specific information for staff consideration. However, by virtue of the restart proceeding and the associated certification activities which specifically required EFW-related procedure revisions and operator training, review activities of NUREG-0737 Action Item I.C.1 (Emergency Operating Procedures), and the verification that specific procedural changes related to seismic events had been implemented, see section III.A. infra, the staff has performed extensive reviews of the TMI-1 emergency procedure and operator training programs. Based on those reviews, the staff concludes that the Petition provides no basis to question the adequacy of those programs.

of the EFW system. The staff reserves judgment on whether its analysis of the outstanding issues may impact this interim decision. A final Director's Decision will be issued upon completion of the staff's review.

II. THE RESTART PROCEEDING

The adequacy of TMI-1 EFW system has been extensively litigated as a principal design issue in the TMI-1 restart proceeding. Although testimony was offered as to numerous aspects of the EFW system, the licensing and appeal boards adjudicating the matter restricted their findings, for the most part, to those elements of the EFW system called into question by the accident at the Three Mile Island Nuclear Station, Unit 2, namely small-break loss of coolant accidents and feedwater transients. See Metropolitan Edison Co. (Three Mile Island Nuclear Station, Unit 1), ALAB-724, 17 NRC 559, 559-60 (1983). See also Metropolitan Edison Co. (Three Mile Island Nuclear Station, Unit 1), CLI-83-5, 17 NRC 331, 331-32 (1983). To the extent that the issues raised by the petitioner were litigated in the restart proceeding, the staff would not initiate new enforcement proceedings to consider the same issues. See Rockford League of Women Voters v. NRC, 679 F. 2d 1218, 1222 (7th Cir. 1982); Pacific Gas and Electric Co. (Diablo Canyon Nuclear Power Plant, Units 1 and 2), CLI-81-6, 13 NRC 443, 446 (1981). In this regard, petitioner raises an issue which was fully explored in the restart proceeding, the accuracy of the emergency feedwater flow instrumentation. Staff testimony on the accuracy requirements for this system was that each flow instrument should have an accuracy of "on the order of ± 10%".[2] Licensee testimony was that the accuracy would be "better than or equal to 5%."[3] The issue was not pursued any further before the Atomic Safety and Licensing Board. See Metropolitan Edison Co. (Three Mile Island Nuclear Station, Unit 1), LBP-81-59, 14 NRC 1211, 1362 (1981). However, by letter dated May 24, 1983, the licensee advised the staff the system design could not be successfully implemented. By letter dated August 25, 1983, the licensee advised the staff of additional system difficulties and proposed an alternate design. The staff reviewed and subsequently approved the licensee's proposed design.[4] Upon installation of the alternate design, licensee later advised the staff, by letter dated November 23, 1983, that oscillations had been observed at low flow conditions which exceeded the accuracy criteria established by the staff. The licensee has now taken the position that the present instrumentation is adequate. The petitioner, a party to the restart proceeding, contests this view, and has responded to the licensee's November 23, 1983 letter

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2 See NUREG-0680, TMI-1 Restart (June 1980).

3 See Recommended Requirements for Restart of Three Mile Island Nuclear Station, Amendment 22.

4 See letter from J. F. Stolz (NRC) to H. D. Hukill (GPUN) (September 22, 1983).

by filing a response with the Commission.[5] The licensee responded by filing a reply with the Commission, which was responded to by the petitioner.[6] By Board Notification 84-088 dated April 24, 1984, the staff advised the Commission, restart proceeding boards and parties, including petitioner, that it considered the existing TMI-1 EFW flow instruments to be acceptable.[7] The recent filings have placed the issue of EFW flow instrumentation accuracy before the Commission.[8] To the extent that a full consideration of EFW flow

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5 See Union of Concerned Scientists Response to GPU Letter of December 6, 1983, Regarding Emergency Feedwater Flow Instumentation (December 9, 1983).

6 See Licensee's Reply to UCS Response to GPU Letter of December 23, 1983 (December 23, 1983) and Petitioner Rebuttal to Licensee's Reply Regarding EFW Flow Instrumentation (January 6, 1984).

7 The basis for the staff's conclusion is that the accuracy of the flow indications available to the operator at low flows is taken into account by the plant operating procedures and is acceptable, even though the flow indication accuracy at low flows may exceed the criteria established by the staff.

8 It should be noted that, by order dated January 27, 1984, the Commission took review of five specific design issues addressed by the Appeal Board in Metropolitan Edison Co. (Three Mile Island Nuclear Station, Unit 1), ALAB-729, 17 NRC 814 (1983), including the Appeal Board's treatment of the Licensing Board's quantitative analysis of the reliability of the EFW system. The staff, licensee, and petitioner have each filed briefs addressing those issues.

The Commission's January 27, 1984 order also took review of whether the issue concerning environmental qualification of electrical equipment had been removed from the restart proceeding by the Commission's generic rulemaking on the subject and offered an opportunity for the parties to comment on the adequacy of the licensee's proposed solution to the MSLRDS problem. The staff, in its March 19, 1984 filing, argued that the environmental qualification issue was removed from the proceeding, that the proposed MSLRDS solution is adequate with respect to the EFW system concerns of the restart proceeding, and further, that the concerns regarding the potential failure of the non-safety grade MSLRDS to isolate main feedwater leading to the possibility of containment overpressurization are not within the scope of the restart proceeding and should properly be addressed during review of this Petition. The UCS filing, dated March 19, 1984, argued that all aspects of both issues should properly be addressed in the restart proceeding.

instrumentation accuracy is necessary to evaluate petitioner's concern that the aggregate effect of the EFW deficiencies it raises compromise the reliability of the EFW system, the staff will consider EFW flow instrumentation when a final decision on the petition is issued.

III. CONSIDERATION OF THE ISSUES

A. Seismic Qualification of the Emergency Feedwater System

The Petition alleges that operation of TMI-1 would pose an undue risk to public health and safety because the EFW system is not seismically qualified.[9] The fundamental contentions in this regard can be characterized as: (1) contrary to NRC regulations, the TMI-1 EFW system is not seismically qualified and the licensee does not intend to make it so prior to operating the plant, and (2) the staff's safety evaluation on the seismic capability of the TMI-1 EFW system does not provide an adequate basis for such operation.

When TMI-1 was licensed, the EFW system was not classified as an engineered safety feature system and accordingly was not required to be

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9 Seismic qualification of the TMI-1 EFW system was not addressed in the restart proceeding because such matters are unrelated to the March, 1979 accident at TMI-2 and the concerns which led to the restart proceeding. See Metropolitan Edison Co. (Three Mile Island Nuclear Station, Unit 1), CLI-83-5, 17 NRC 331 (1983).

seismically qualified.[10] In February 1981, the staff issued Generic Letter 81-14 to all operating pressurized water reactors. This generic letter stated the intent to increase the seismic resistance, where necessary, in a timely, systematic manner to ultimately provide reasonable assurance that auxiliary and emergency feedwater systems would be able to function after the occurrence of earthquakes up to and including the safe shutdown earthquake (SSE). In this regard, TMI-1 was treated in a manner consistent with other operating reactors in that the matter was considered resolved when (a) all seismic improvements had been identified and scheduled for implementation in a timely manner, and (b) continued plant operation during the interim period had been justified on an acceptable basis. The licensee has committed to seismic upgrade modifications during the first refueling outage following restart (i.e., prior to Cycle 6 operation) and has provided compensatory measures for Cycle 5 operation. The staff has concluded that there is reasonable assurance that, should restart be authorized, the TMI-1 EFW system would be able to perform its safety function after the occurrence of an SSE and that the system does comply with Commission regulations.

The staff issued a safety evaluation on the seismic capability of the TMI-1 EFW system on August 12, 1983. In light of the arguments set

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10 The staff position that auxiliary/emergency feedwater systems be seismically qualified first became effective for new plants in 1972. <u>See</u> Regulatory Guide 1.29. The requirement was not backfit to include plants for which certain licensing milestones had been reached, which was the case for TMI-1. Thus, TMI-1 and a number of other operating reactors do not, and are not required to have seismically qualified auxiliary/emergency feedwater systems.

forth in the Petition, the staff has reconsidered its position on this matter and its safety evaluation. In so doing, the staff has reaffirmed the conclusion that, at restart, there is reasonable assurance that the TMI-1 EFW system would be able to perform its safety function following the occurrence of an SSE.[11]

B. Single Failure Capability of the Emergency Feedwater System

The Petition asserts that until the long-term system upgrades are complete, the TMI-1 EFW system is vulnerable to single failures which would, for certain accidents, prevent it from providing cooling water for decay heat removal. In this regard, the petitioner is correct in stating that, should restart be authorized, the TMI-1 EFW system will have a single flow control valve in each of the feedwater headers to the two steam generators. The petitioner argues that for those events requiring isolation of one steam generator, such as a main steam line break, steam generator tube rupture

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11 The Petition provided no information that was not considered during the 1983 staff review of this matter, with one exception. The exception deals with postulated interaction from failures of non-seismic portions of other systems, namely, the vent stacks (discharge paths) for the safety relief valves (MS-V-22A, B) and the atmospheric dump valves (MS-V-4A,B). After review of this question, the staff concludes that there is reasonable assurance that local manual actions will not be precluded by a steam environment during the interim period of Cycle 5 operation. Further details concerning the staff's most recent review of this issue are found in the Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Interim Director's Decision Under 10 CFR 2.206 (Seismic Capability of Emergency Feedwater), Three Mile Island Nuclear Station, Unit No. 1., dated April 27, 1984.

(under certain circumstances), or a feedwater line break, failure of the flow control valve to open in the feedwater header to the intact steam generator could result in an inability to deliver emergency feedwater flow for decay heat removal through the steam generator. Further, the Petition points out that a single failure in the Integrated Control System (ICS), which currently controls the EFW flow control valves, could also result in an inability to deliver EFW flow by preventing the flow control valves from opening.

The staff has been aware of these system deficiencies for some time, and the issue has been fully explored during the restart proceeding. The staff considers the system to be acceptable, provided that certain short-term modifications are completed prior to restart.[12] Among these modifications is a change in failure mode for the flow control valves. These valves will fail so as to permit full EFW flow on either loss of instrument air or loss of control power.[13] Further, a separate remote manual control station independent of the ICS has been provided in the control room. This modification will permit the operator to remotely open

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12 See NUREG-0680, TMI-1 Restart (June 1980) and Supplement 3 to NUREG-0680 (April 1981).

13 The restart proceeding record shows that the flow control valves fail to the mid position on loss of control signal. However, by filing dated March 26, 1984, counsel for licensee indicated that the existing flow control valve converters would be replaced with environmentally and seismically qualified converters by June 1984, and that with these new converters the flow control valves would fail to the open position on loss of control power.

the EFW flow control valves should they fail closed due to an ICS malfunction. The flow control valves could also be manually opened locally by means of a hand wheel.

An additional single failure vulnerability hypothesized by the Petition is that "each EFW flow path contains only a single block (isolation) valve. Failure of this valve would prevent isolation of EFW flow to the steam generator with the broken main steam line or ruptured tube." See Petition at 20. The petitioner's statement as to the existence of a "single block (isolation) valve" in each EFW flow path is inaccurate.[14] Nevertheless, for those events requiring isolation of a steam generator (main steam or feedwater line break, or steam generator tube rupture), a cavitating venturi has been installed in each EFW supply line to limit EFW flow to the ruptured steam generator and ensure sufficient flow to the intact steam generator. Because of this modification, the main steam line rupture detection system (MSLRDS) signals to the EFW flow control valves have been deleted to prevent inadvertent EFW isolations caused by failures in the MSLRDS. See section III.C. infra. Since it may be desirable to eventually isolate EFW to a ruptured steam generator, the operator would close the appropriate EFW flow control valve. If this valve failed to close, EFW flow to the

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14 The staff bases this view on its review of the present EFW system design drawings, the restart proceeding record and a physical inspection of the system by the resident inspector. The only valves in the steam generator flow path which can be readily identifed are the flow control valves and check valves. There are however, motor operated sectionalizing block valves in the discharge cross-tie header between the EFW pumps. These valves do not serve as steam generator isolation valves since the motor driven EFW pumps discharge downstream of the valves.

ruptured steam generator could be stopped by closing the appropriate EFW pump discharge cross-tie sectionalizing valve and tripping the respective EFW pump.

C. Main Steam Line Rupture Detection System

One purpose of the main steam line rupture detection system (MSLRDS) is to prevent containment pressure from exceeding its design pressure in the event of a main steam line rupture inside containment. The system does this by isolating feedwater flow to a given steam generator when a relatively low pressure is detected in that steam generator. A concern raised in the restart proceeding was that spurious actuation of the non-safety grade MSLRDS could inadvertently isolate all feedwater flow to both steam generators. Resolution of this concern is being pursued within the restart proceeding.[15] The petitioner suggests that because the MSLRDS is not safety grade, there can be no assurance that the containment will not be overpressurized following a main steam line rupture inside containment. Therefore, argues petitioner, "operation of TMI-1 would pose an undue risk to public health and safety."

Although the TMI-1 MSLRDS is not safety grade, it is redundant and primarily located outside containment where it would not be exposed to the

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15 See NRC Staff Brief Concerning the Commission's Review of Specific Design Issues in ALAB-729, (March 19, 1984).

harsh environment created by a main steam line rupture inside containment.[16] By letter dated February 16, 1984, the licensee informed the staff that the MSLRDS pressure switches located inside containment would be environmentally qualified through replacement with qualified equipment by June 1984. All MSLRDS components located inside containment will then be environmentally qualified. Therefore, in the event of a main steam line rupture inside containment, the MSLRDS would be expected to remain functional and isolate main feedwater flow to the affected steam generator, even after a postulated single active failure. For a main steam line break occurring outside containment, the environmental qualification of the MSLRDS is not a concern since the containment would not be affected.

The MSLRDS prevents containment pressure from exceeding its design pressure in the event of a main steam line rupture inside containment. The MSLRDS is not relied on in any direct manner for preventing exposure of the public to any undue risk to health and safety. The two barriers that prevent exposure of the public to the effects of a main steam line rupture are the reactor primary pressure boundary and the containment boundary. These two barriers would remain intact after a postulated main steam line rupture, with or without the MSLRDS isolating the main feedwater flow to the affected steam generator. Based on the staff's review experience with similar plants, if the MSLRDS failed to function, the reactor pressure boundary would be unaffected; and although the containment design pressure may be slightly exceeded, containment integrity would be maintained.

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16 The postulated main steam line break event at TMI-1 was evaluated in conjunction with the staff's review of IE Bulletion 80-04, "Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition."

For these reasons, it is the staff's view that the MSLRDS, as designed, and as upgraded with qualified pressure switches inside containment, will isolate feedwater flow to the affected steam generator, even after sustaining a single active failure, and containment integrity would remain intact after a postulated main steam line rupture inside containment.[17]

IV. Conclusion

Based on the foregoing discussion of the Petition, I find no adequate reason to take the requested action regarding the Three Mile Island Nuclear Station, Unit 1, operating license at this time. A final decision with respect to petitioner's request will be issued in the near future upon completion of the staff's review of the remaining issues. A copy of this decision will be filed with the Office of the Secretary for the Commission's review.

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 27th day of April 1984.

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17 Nevertheless, licensee has committed to upgrade the MSLRDS to safety grade status prior to startup from the Cycle 6 refueling outage (next refueling). See letter from H. D. Hukill (GPUN) to J. F. Stolz (NRC) (August 23, 1983).

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UNITED STATES NUCLEAR REGULATORY COMMISSION

[DOCKET NO. 50-289]

GENERAL PUBLIC UTILITIES NUCLEAR CORPORATION

(THREE MILE ISLAND NUCLEAR STATION, UNIT 1)

Issuance of Interim Director's Decision Under 10 CFR 2.206

Notice is hereby given that the Director, Office of Nuclear Reactor Regulation, has issued an interim decision concerning a petition dated January 20, 1984, submitted by the Union of Concerned Scientists. The petition requests that the Commission continue the suspension of the operating license unless and until certain modifications are made to the TMI-1 emergency feedwater system.

The Director, Office of Nuclear Reactor Regulation, has determined to tentatively deny the petitioner's request at this time with respect to four of the issues raised in the petition.

The reasons for this decision are explained in an "Interim Director's Decision under 10 CFR 2.206" (DD-84- 12) which is available for public inspection in the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and in the local Public Document Room for the TMI facility located in the Government Publications Section of the State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126. A copy of this decision will be filed with the Secretary for the Commission's review.

Dated at Bethesda, Maryland, this 27th day of April 1984.

FOR THE NUCLEAR REGULATORY COMMISSION

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Harold R. Denton, Director
Office of Nuclear Reactor Regulation

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING DIRECTOR'S INTERIM DECISION

UNDER 10 CFR 2.206 (SEISMIC CAPABILITY OF EMERGENCY FEEDWATER)

METROPOLITAN EDISION COMPANY
JERSEY CENTRAL POWER AND LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY
GPU NUCLEAR CORPORATION

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

INTRODUCTION

On January 20, 1984, the Union of Concerned Scientists (UCS) filed a petition pursuant to 10 CFR 2.206 requesting that the NRC suspend the operating license for Three Mile Island Unit 1 (TMI-1) until the plant's emergency feedwater (EFW) system "complies with the NRC rules applicable to systems important to safety (including safety-grade, safety-related, and engineered safety feature systems)." One of the issues raised by the petition is the seismic capability of the EFW system. That is the subject of this evaluation. The remaining issues raised by the petition are either addressed in the Director's Interim Decision Under 10 CFR 2.206, or remain under review at this time.

In our review, we have considered the petition, the licensee's response to the petition dated February 24, 1984, as amended by submittal dated March 26, 1984, and our earlier evaluation of this matter forwarded to GPU Nuclear under letter dated August 12, 1983.

EVALUATION

The fundamental contention of the petition regarding EFW seismic qualification is that, contrary to NRC regulations which require engineered safety feature (ESF) systems to be designed to withstand the effects of earthquakes, the TMI-1 EFW system is not seismically qualified and the licensee does not intend to make it so prior to operating the plant. To support this contention, the petition presents information that deals with: compliance with NRC requirements, the NRC contractor's report, independent evaluation-qualification of valves, loss of water sources, the effects of flooding, and Cycle 5 operation. Each of these subject areas is discussed below.

Compliance with NRC Requirements

The petition contends that the TMI-1 EFW system does not satisfy NRC regulations regarding seismic qualification. In this regard, the principal

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design criteria for plant systems are established during the construction permit (CP) application review. The TMI-1 CP was applied for in 1967 and issued in 1968. At that time, the EFW system was not classified as an ESF system and thus was not required to be seismically qualified.

The operating license (OL) was subsequently applied for in 1970, and the staff's Safety Evaluation was issued in 1973. During that period, Regulatory Guide 1.29 (Safety Guide 29) was issued, but backfit implementation was not included for plants already holding a CP or OL. Thus, the TMI-1 EFW system was not required to be seismically qualified at the time it received its operating license.

Notwithstanding the requirements discussed above, the staff has always intended that there be reasonable assurance that the plant be able to shutdown safely following a seismic event. We recognize that various systems in the plant would be available to remove decay heat. In a generic letter to all PWR licensees dated October 21, 1980, the NRC focused on steam generators and the auxiliary (emergency) feedwater system as the first choice method for accomplishing safe shutdown.

In February 1981, the NRC issued Generic Letter 81-14 to operating PWRs, which announced the intent to increase the seismic resistance, where necessary, in a timely, systematic manner, to ultimately provide reasonable assurance that the auxiliary/emergency feedwater system will function after the occurrence of earthquakes up to and including the Safe Shutdown Earthquake (SSE). Consistent with the staff handling of other backfit type improvements, we have not considered the plant to be unsafe or that plant operations need be curtailed since there is no imminent safety threat. It has been our intent to allow credit for alternate decay heat removal systems for an interim period where necessary while modifications to the auxiliary feedwater system are developed and implemented. In this EFW system seismic review, we have treated TMI-1 the same as other operating reactors.* We have considered the matter to have been resolved when all seismic improvements have been identified and scheduled for implementation in a timely manner, and continued plant operation during the interim has been justified on an acceptable basis.

<u>NRC Contractor's Report</u>

The petition raises questions about the NRC contractor's Technical Evaluation Report (TER) dated October 29, 1982, and a list of "many vital components in the TMI-1 EFW system which are not seismically qualified." This list was a preliminary list developed by an NRC contractor and did not represent the staff's final conclusion. Several items on the list were placed there for information only and are not even part of the EFW system. An example is the control system for the atmospheric relief valves (MS-V4A, B). Other items on

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* <u>See</u> CLI-81-3, 13 NRC 291 (1981) in which the Commission directed that TMI-1 was to be grouped with operating reactors, as opposed to reactors with pending applications for operating licenses, unless the TMI-1 restart record dictated to the contrary. The seismic capability of the EFW system is not within the scope of the restart proceeding.

the list are not vital to EFW system performance. Some items on the list might fail in a manner that could adversely affect EFW performance if an SSE-level earthquake were to occur; these items became the subjects of upgrade actions. The TER was revised significantly in July 1983 as a result of NRC staff review and discussions with the licensee. Therefore the list and the October 1982 TER are not final information. The NRC staff's final report on this subject was issued on August 12, 1983 and included a copy of the contractor's revised TER as an enclosure.

Independent Evaluation-Qualification of Valves

The petition claims that the contractor made no independent evaluation of the licensee's claims of seismically qualified components. The petition cites the use of static analysis alone for the valves whereas the NRC Standard Review Plan requires a testing program.

In the responses to Generic Letter 81-14, the licensee stated that the EFW system was seismically qualified, with the exception of certain identified components. In this context, "qualified" is used here to mean that the equipment had been designed, constructed, and maintained to withstand an SSE, utilizing methods and acceptance criteria consistent with that applicable to other safety-related systems in the plant. The licensee has appropriate QA records and documentation available at the site for inspection by the NRC. The NRC contractor was not asked to make an independent evaluation of equipment in this category.

The NRC contractor's review was focused on the adequacy of the components identified by the licensee as exceptions. The objective of the review of each component was to obtain a best estimate regarding whether or not the component would be able to function following an SSE. This estimate was not based upon strict compliance with any specific set of regulatory requirements (such as those listed in Regulatory Guides or in the Standard Review Plan) that a new plant would have to satisfy, but rather was based on sound engineering judgment.

In the judgment of the contractor, the valves are most likely able to withstand an SSE and to perform the necessary functions. Moreover, the licensee has stated that the analysis performed was not just a static analysis but was in fact a dynamic analysis. Further, in our technical experience, if the seismic analysis has shown the valves to be adequate, it is not likely that a testing program would indicate seismic failure. Our conclusion remains that there is reasonable assurance that the valves are able to withstand an SSE and remain functional.

Loss of Water Sources

The petition claims that sources of cooling water for the EFW system will not be available following an SSE as a result of the postulated failures of the Condensate Storage Tank (CST) low level alarms, failures of the isolation valves CO-V-14A, B (which isolate the CST's from non-seismic piping to the condenser hotwell), and the inability of an operator to perform local actions due to a postulated steam environment. The enclosed sketch of the EFW system shows the various numbered valves.

The purpose of the CST level alarm, in this context, is to provide information to the operator as to when it is necessary to isolate the CST's from the condenser hotwell. A failure modes analysis has been performed for the CST level alarms. Four failure modes were identified. For the most likely failure mode, the alarm would be generated prematurely, i.e., before the CST reaches the low level setpoint. In this case, the operator would isolate the CST from the non-qualified piping to the condenser hotwell before such isolation would be necessary. Two other failure modes would also generate a premature alarm. One failure mode of the four, which is considered unlikely, involves the complete crimping of the instrument sensing line, causing the level indication to be constant which is not conservative. The licensee believes the operator would be able to detect this failure by the absence of a CST level draw-down indication. Nevertheless, to be more conservative, the licensee has revised its emergency procedures (#1202-30) to require that the operator isolate the CSTs from the non-qualified piping, regardless of the level indication or alarm, as soon as the seismic event occurs. A threshold seismic alarm set at 0.01g is provided in the control room. Therefore, we conclude that the operator action of isolating the CST will not be negated by lack of information as to when such isolation is to be taken. Therefore, the lack of seismic qualification of the CST level instrumentation during Cycle 5 would not cause a significant information loss in view of the compensatory measures provided and is therefore acceptable.

Isolation of the CST from the non-qualified piping would normally be a remote manual action in that it would be performed from the main control room. However, the operator may find that his actions are not effective. Failure of the isolation valves (CO-V-14A, B) could occur due to either of two possible causes. The valve could fail to respond to the control room initiation due to loss of electric power to the motor operated valve, due to lack of seismic power cable installation through the turbine building. For this case, an operator would be dispatched to the valve location to close the valve manually. These valves are located in the corridor outside the EFW pump rooms. The failure of the valve due to loss of electric power would not affect the operator's ability to manually operate the valve and a handwheel is provided on the valve for this purpose. This local manual operator action is prescribed by plant procedures (#1202-30). The staff's review of the licensee's analysis indicates that if this procedure is completed within 20 minutes of the seismic event, the quantity of water remaining in one CST will be sufficient. In order to provide additional assurance, we visited the plant, performed a procedural walk-through and concluded that the operator could get to the area and complete the required actions in less than 15 minutes, with a typical time of about 5 minutes.

Failure of the isolation valve could also occur as a single random failure. In this case, local manual actions might not be effective. Since either of the two CSTs has sufficient inventory for the EFW safety function, the loss of one of the tanks due to a valve failure is not unacceptable so long as the CSTs are isolated from each other. Consequently, plant emergency procedures also require that, upon receiving the seismic event alarm, the CST cross-connect valves (CO-V-111A, B) be closed immediately by operator action, either remotely from the control room or locally in the event of a power loss. These valves are located in the same corridor as CO-V-14A, B. Therefore, we conclude that, [illegible] the failure of non-seismically qualified components is considered and a single failure is simultaneously considered, the CSTs would still remain capable of providing a sufficient source of water for the EFW system. Moreover, in the event both CSTs were to be lost, the plant configuration includes a fully qualified safety-related alternate source of water known as the Emergency River Water System.

The petition claims that local manual actions will be precluded because the environment in the Intermediate Building would prevent entry. The petition asserts that a severe steam environment would be generated due to the failure of the non-seismically qualified components in other systems, namely, the vent stacks (discharge paths) for the safety relief valves (MS-V-22A, B) and the atmospheric dump valves (MS-V-4A, B) which are routed through the floor of the Intermediate Building.

The seismic review of the EFW system at TMI-1 did not include the interactions due to the failure of non-seismic portions of other systems.

The licensee has stated that there is a low probability of release of steam to the Intermediate Building from these vent stacks, and there is reasonable assurance, during Cycle 5 operation, that the operator will be able to function in the Intermediate Building. The licensee states in its February 24, 1984 submittal that the potential that the safety valves MS-V22A, B would open has been reduced because the upstream pressure regulating valve MS-V6 has been limited to 65% of its stroke. The licensee also stated that, for the MS-V4A, B atmospheric dump valves, the failure mode is to the closed position upon loss of control air.

The postulated failure of these vent stacks was discussed during our visit to the plant on March 6, 1984. The licensee stated that the vent stacks were designed to the ASME B31.1 piping code. The licensee related that during actual seismic events of significant magnitude, large power generating stations designed to B31.1 suffered only very limited damage and no fluid systems were rendered inoperable. Therefore, the licensee believes that the vent stacks have considerable seismic resistance. The licensee also related to us that their general power plant experience shows that, after the steam release is terminated, the steam dissipates rapidly, and entry can be made in a matter of a few minutes.

During our walkdown of the EFW system, we noted that the atmospheric dump valves (MS-V4A,B), the safety valves (MS-V22A, B) and the associated vent stacks are located in the same compartment as the turbine-driven EFW pump in the Intermediate Building. This compartment is in direct connection (via an open doorway) with the corridor where valves CO-V-14 A, B; CO-V-111 A, B; and the river water system valves are located. It appears therefore that access to the corridor for manual actions could be impeded for a period of time if the vent stacks failed and a steam environment were generated.

The probability and severity of the postulated steam environment would depend upon: the probability that the valves (MS-V4A, B; MS-V22A, B) opened, the probability that the associated vent stack were to fail due to the seismic event, the probability that the valve(s) could not be reclosed or isolated to terminate the steam release, and the time necessary for the released steam to dissipate. Although the atmospheric dump valves may not be necessary for safe shutdown, they are controlled by the Integrated Control System (ICS). Failures of this system, or its inputs, could cause a spurious signal which would open the atmospheric dump valves. We believe that, if such a situation were to arise, the licensee would be able to reclose these dump valves independent of the ICS (via the "manual loader") from the control room and thereby quickly terminate any steam release.

In summary, the likelihood of a significant seismic event at the TMI-1 site during Cycle 5 operation is small, the likelihood that the steam safety valves (MS-V22's) will lift is small, the likelihood that the stacks for these safety valves will fail is small, the likelihood that the atmospheric dump valves will open is small and if they should, any steam release could be terminated quickly, and any steam released to the compartment via the dump valves would dissipate quickly. In view of these considerations, we conclude that there is reasonable assurance, for the interim period of Cycle 5 operation, that local manual actions can effectively compensate for postulated seismically induced failures of the EFW system.

Effects of Flooding

The petition claims that there was no evaluation of the effects of flooding due to a failure of the non-seismic portions of the EFW system. The petition further claims that such flooding would preclude local manual actions and could cause EFW equipment failures due to spray or submersion.

There are two areas of non-seismically qualified piping related to the EFW system: first, a portion of the recirculation lines downsteam of both the flow restricting orifices and the isolation valves and outward toward CST "B"; and second, at the interface with the condensate system, the feed lines from the CSTs downsteam of isolation valves CO-V-14A,B and outward to the condenser hotwell.

As part of the EFW seismic review, a public meeting was held in Bethesda, Maryland, on January 7, 1983. At this meeting, the staff raised the specific question of whether or not the licensee had considered the spray and flooding effects of a failure of the EFW recirculation piping. The licensee stated that such an evaluation had been conducted with the conclusion that such a failure does not lead to the loss of vital EFW equipment. During our March 6, 1984 plant visit and system walkdown, we confirmed the reasonableness of this conclusion. The licensee provided a discussion of this matter in his followup submittal of February 4, 1983.

At the January 7. 1983 meeting, the licensee was asked to address the flooding effects due to a failure of the CST feed lines to the hotwell. The February 4, 1983 followup submittal discusses this evaluation and concludes that the spill from such a failure would occur in the turbine building, not in the intermediate building, thereby having no spray or flooding impact on the EFW system or operator access.

These matters are well documented both in the licensee's February 4, 1983 submittal and the NRC summary of the January 7, 1983 meeting (dated January 16, 1983). As a result of our review of this subject, we find no technical merit to the petition claim regarding flooding by EFW system failures. We conclude that spray or flooding due to the EFW failures would not cause loss of the EFW safety function.

Cycle 5 Operation

The petition alleges that the staff attempted to justify allowing plant operation during Cycle 5 (which precedes the seismic upgrades) with an EFW system that both the licensee and the NRC contractor state will not be qualified until the start of Cycle 6.

This allegation seems to arise from confusion with regard to the usage of the term "qualified." In one sense, a system may not be considered "qualified" until it has full documentation to demonstrate strict compliance with all the standards required for a new plant about to be licensed. This may be the usage of the petitioner. It is not necessary, for backfit purposes, that operating reactors have all systems qualified to this degree.

In a second sense, a system can be considered "qualified when all components vital to the functioning of the system have been shown to be equivalent to qualification without full documentation, if engineering judgment indicates that each such component is expected to withstand an SSE. In this case, the system may be said to have an SSE-level of seismic capability. This is the sense that the NRC contractor and the licensee have used when describing the status of the EFW at the completion of the next refueling outage.

In a third sense, the system can be considered "qualified" when compensatory measures are adequate to overcome system weaknesses due to the possibility that certain components might not withstand the stresses associated with an earthquake as severe as the SSE. Because the compensatory measures often involve manual operator actions, the staff has allowed this degree of qualification only for temporary periods of plant operation.

It should be noted that in all three senses, it must be established that there is reasonable assurance that the safety function of the system can be accomplished following the postulated occurrence of the SSE.

In making a determination regarding the acceptability of seismic capability of the EFW system for Cycle 5 operation, the staff found that most of the components were qualified to the extent of having full documentation, a few components were equivalently qualified and were expected to withstand the SSE, and that one or two components might not withstand the SSE. The staff concluded that the compensatory measures being provided were sufficient to accommodate any such failures and that there was indeed reasonable assurance that the safety function of the EFW system could be provided following an SSE. On this basis, the staff determined that the EFW was acceptable for Cycle 5 of plant operation.

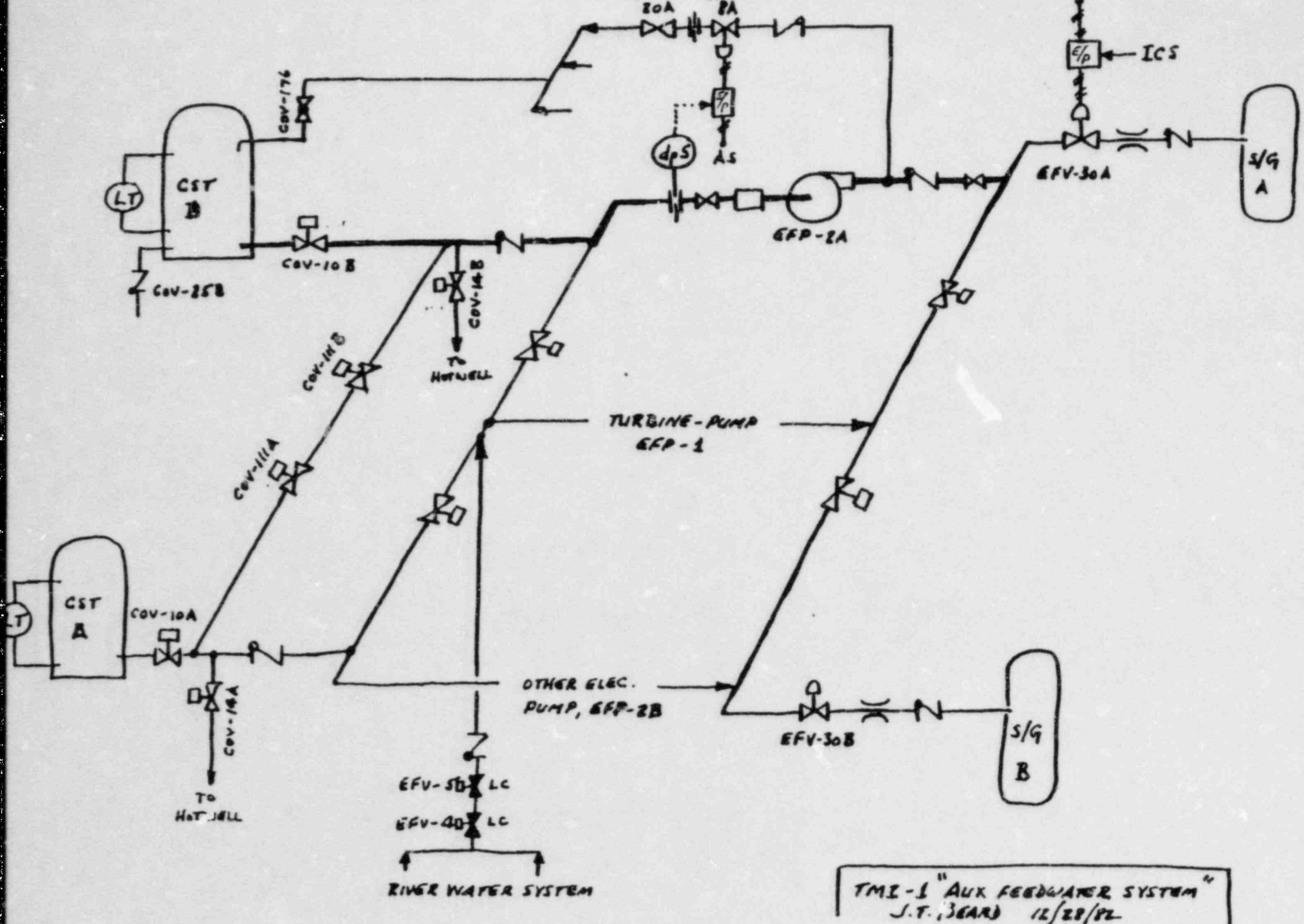
SUMMARY AND CONCLUSIONS

The UCS 2.206 petition claims that the seismic capability of the TMI-1 EFW system does not satisfy the NRC's regulations and that the licensee does not intend to make it satisfy the requirements prior to operating the plant.

At TMI-1, there is reasonable assurance that a seismic event would not incapacitate the EFW system, and therefore the EFW system does satisfy NRC regulations. The NRC contractor did in fact conduct an independent evaluation and found that seismic capability of EFW valves is acceptable. The probability and consequences of a loss of the level instrumentation for the CST are acceptable, and postulated failures do not cause loss of all water sources for the EFW system. The effects of flooding have been evaluated and are acceptable.

Based on the above evaluation, we reiterate the conclusion in our August 12, 1983 Safety Evaluation that, in view of the system modifications planned for the refueling outage prior to Cycle 6 operation (first refueling) and the interim compensatory measures being provided for Cycle 5 operation, there is reasonable assurance that the Emergency Feedwater System at Three Mile Island, Unit 1, would be able to withstand a Safe Shutdown Earthquake and perform its safety function.

Dated: April 27, 1984



TMI-1 "AUX FEEDWATER SYSTEM"
J.T. BEARD 12/28/82