October 26, 1979

Docket No. 50-289

Mr. R. C. Arnold,
Senior Vice President
Metropolitan Edison Company
260 Cherry Hill Road
Parsippany, New Jersey 07054

Re: TMI-1 Restart Program - Request for Additional Information

Dear Mr. Arnold:

During our meeting in Middletown on October 17, 1979, we discussed a number of questions which the staff had thus far developed in reviewing the TMI-1 "Restart Report." These and others resulting from our further review of that report through Amendment 2 are included in the enclosed request for additional information.

Since you have had most of the enclosed items available to you since October 17, and because of the schedular demands of the Commission's Order of August 9, 1979, we will require complete and adequate responses to the enclosed items by November 7, 1979, at the latest. Earlier partial submittals may expedite our review.

As indicated above, your submittal scheduled for the week of October 22 has not yet been received and reviewed. We presume much of the missing information identified in our letter of October 2, 1979, will be answered by that submittal, and that responses to some of the enclosed items will be included. However, additional questions may be expected from our review of this material.

Sincerely,

[redacted] signed by:
[redacted] H. Vollmer

Richard H. Vollmer, Director
Three Mile Island Support

Enclosure:
Request for Additional Information

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DATE	10-24-79	10/24/79				

R. C. Arnold

October 26, 1979

Mr. J. G. Herbein,
Vice President Nuclear Operations
Metropolitan Edison Company
P.O. Box 480
Middletown, Pennsylvania 17057

Mr. E. G. Wallace, Licensing Manager
Metropolitan Edison Company
260 Cherry Hill Road
Parsippany, New Jersey 07054

Mr. G. P. Miller,
Manager Support Services and Logistics
Metropolitan Edison Company
260 Cherry Hill Road
Parsippany, New Jersey 07054

Mr. J. L. Seelinger,
Superintendent, Unit 1
Metropolitan Edison Company
260 Cherry Hill Road
Parsippany, New Jersey 07054

Mr. J. J. Colitz,
Manager Plant Engineering
Metropolitan Edison Company
260 Cherry Hill Road
Parsippany, New Jersey 07054

Mr. I. R. Finfrock
Jersey Central Power and Light Company
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Mr. R. W. Conrad
Pennsylvania Electric Company
1007 Broad Street
Johnstown, Pennsylvania 15907

J. B. Lieberman, Esq.
Metropolitan Edison Company
260 Cherry Hill Road
Parsippany, New Jersey 07054

Ms. Mary V. Southard,
Chairperson, Citizens for a Safe
Environment
c/o Metropolitan Edison Company
260 Cherry Hill Road
Parsippany, New Jersey 07054

Dr. Walter H. Jordon
881 W. Outer Drive
Oak Ridge, Tennessee 37830

Dr. Linda W. Little
5000 Hermitage Drive
Raleigh, North Carolina 27612

George F. Trowbridge, Esq.
Shaw, Pittman, Potts & Trowbridge
1800 M Street, N.W.
Washington, D.C. 20006

Karin W. Carter, Esq.
505 Executive House
P.O. Box 2357
Harrisburg, Pennsylvania 17120

Honorable Mark Cohen
512 E-3 Main Capital Building
Harrisburg, Pennsylvania 17120

Ellyn Weiss, Esq.
Sheldon, Harmon, Roisman & Weiss
1725 I Street, N.W., Suite 506
Washington, D.C. 20006

Mr. Steven C. Sholly
304 S. Market Street
Mechanicsburg, Pennsylvania 17055

Mr. Thomas Gerusky
Bureau of Radiation Protection
Department of Environmental Resources
P.O. Box 2063
Harrisburg, Pennsylvania 17120

Mr. Marvin I. Lewis
6504 Bradford Terrace
Philadelphia, Pennsylvania 19149

Ms. Jane Lee
R.D. 3, Box 3521
Etters, Pennsylvania 17319

1323 100

Walter W. Cohen, Consumer Advocate
Department of Justice
Strawberry Square, 14th Floor
Harrisburg, Pennsylvania 17127

OFFICE ▶						
SURNAME ▶						
DATE ▶						

Robert L. Knupp, Esq.
Assistant Solicitor
Knupp and Andrews
P.O. Box P
407 N. Front Street
Harrisburg, Pennsylvania 17108

John E. Minnich, Chairman
Dauphin Co. Board of Commissioners
Dauphin County Courthouse
Front and Market Streets
Harrisburg, Pennsylvania 17101

Robert Q. Pollard
Chesapeak Energy Alliance
609 Montpelier Street
Baltimore, Maryland 21218

Chauncey Kepford
Judith H. Johnsrud
Environmental Coalition on Nuclear Power
433 Orlando Avenue
State College, Pennsylvania 16801

Ms. Frieda Berryhill, Chairlady
Coalition for Nuclear Power Plant Postponement
2610 Grendon Drive
Wilmington, Delaware 19808

Holly S. Keck
Anti-Nuclear Group Representing York
245 W. Philadelphia Street
York, Pennsylvania 17404

John Levin, Esq.
Pennsylvania Public Utilities Commission
P.O. Box 3265
Harrisburg, Pennsylvania 17120

Jordon D. Cunningham, Esq.
Fox, Farr and Cunningham
2320 N. Second Street
Harrisburg, Pennsylvania 17110

Ms. Kathy McCaughin
Three Mile Island Alert, Inc.
23 South 21st Street
Harrisburg, Pennsylvania 17104

Ms. Marjorie M. Aamodt
R.D. #5
Coatesville, Pennsylvania 19320

Ms. Karen Sheldon
Sheldon, Harmon, Roisman & Weiss
1725 I Street, N.W., Suite 506
Washington, D.C. 20006

Earl B. Hoffman
Dauphin County Commissioner
Dauphin County Courthouse
Front and Market Streets
Harrisburg, Pennsylvania 17101

*Ivan W. Smith, Esq.
Atomic Safety & Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

*Atomic Safety and Licensing Appeal Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

*Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

*Docketing and Service Section
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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SURNAME ▶						
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REQUEST FOR ADDITIONAL INFORMATION

1. Restart Submittal Section 2.1.1.7.3 (Aux. Feedwater) describes the main feed pump differential pressure sensing equipment as control grade. Provide safety grade automatic initiation for emergency feedwater.

2. Provide a description, supplemented with sufficient electrical drawings, of how the manual EFW initiation and control to be added to the design can function in the presence of failures in the automatic initiation and control portion of the design and vice versa.

3. Provide the results of the detailed loading study on the diesel generators that confirm the acceptability of adding the AFW pumps. Provide your schedule and a description of the actual testing planned.

4. Restart Submittal Section 2.1.1.7.6 f (EFW) states that only one flow indicator is to be provided for each steam generator. This is unacceptable. Describe redundant (diverse) means that can be used for this purpose in the exisitng design or modify your design to meet the single failure criteria. For example, use of safety grade OTSG level indication as backup indication would be acceptable.

5. Provide details of the provisions for startup and periodic functional testings the new EFW initiating circuits.

6. In the Restart Submittal, Section 2.1.1.7.3 you indicate that the Controltron flow-sensing devices to be installed for emergency feedwater flow indication are safety related. Describe what is meant by "safety related." Indicate what qualification testing these instrumentations have received, and whether or not they are testable. We will require that these devices be safety grade for TMI-1 (see Question #4).

7. In the Restart Submittal, Section 2.1.1.7.7 you indicate that a functional test of the new manual control valve station and the emergency feedwater flow instrumentation will be performed at cold shutdown condictions. In order to properly assure manual emergency feedwater flow control, it is our position that a confirmatory test be performed during startup at low power to show adequate operator control under real dynamic conditions. Provide a test plan for such a test including acceptance criteria, or alternatively provide test data from tests already conducted or actual system response to transients which demonsteate satisfacotry manual EFW control. Differences between tests and the actual system control following transients would have to be justified.

8. Describe and justify the method used to determine the minimum required emergency feedwater flow capacity (sizing criteria). Verify that the minimum emergency feedwater is consistent with your safety analyses for all anticipated accident and transient conditions assuming a single failure of any system component.

9. You indicate that the failure mode on loss of all air pressure for the emergency feedwater control valves will be changed to fully open in order to assure emergency feedwater flow when required. Verify that this change will not result in possible

overfilling of the steam generator with resulting adverse effects to the secondary side piping system. How long would it take to secure valves? Why is failing full-open best? Why not fail to some modulated position to minimize over-cooling? Assuming the control valves fail open, how much time is available for the operator to shutoff/or throttle flow? Identify the acceptance criteria used to establish this time period. Provide your calculations.

10. The NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island Unit 2 recently identified additional requirements for auxiliary feedwater systems. In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident. Our specific concerns include system reliability (other than auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training. The designs and procedures of your facility should be evaluated against the following requirements to determine the degree of conformance. Provide the results of this evaluation and an associated schedule and commitment for implementation of required changes or actions described in Appendix A.

a. We require that you provide redundant level indications and low level alarms in the control room for the EFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure conditions from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity EFW pump is operating.

b. We require that you perform a 72-hour endurance test on all EFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

c. Question 8 previously submitted to you requests information on the basis for your emergency feedwater system flow requirements. Refer to Enclosure 1 for further details when responding to this question.

d. We require that plants which require local manual realignment of valves to conduct periodic tests on one EFW system train <u>and</u> which have only one remaining EFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the EFW system train from the test mode to its operational alignment.

e. We require that emergency procedures for transferring to alternate sources of EFW supply be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

- The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the EFW system pumps against self-damage before water flow is initiated; and,

- The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

f. We require that you propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary EFW system water source to the steam generators. The flow test should be conducted with EFW system valves in their normal alignment.

g. We require that the EFWS should possess the capability to automatically terminate auxiliary feedwater flow to a depressurized steam generator, and to automatically provide feedwater to the intact steam generator.

h. We require that licensees having plants with unprotected normal EFW system water supplies should evaluate the design of their EFW systems to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available to the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by emans such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure or upgrading the normal source of water to meet seismic Category I and tornado protection requirements.

i. Verify that at least one EFW system pump including its associated suxiliaries such as the turbine driven pump lube oil cooling system, and its associated flow path and essential instrumentation will automactically initiate EFW system flow and is capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

j. We require that you evaluate the consequences of a postulated break in the steam line to the turbine-driven EFW pump to determine the need to qualify the EFW system valves, valve actuators, and instrumentation for the environmental conditions resulting from such a high energy line break in order to maintain operability of the motor-driven EFW pumps and their associated flow trains.

k. We require that you provide automatic emergency feedwater initiation on low level indication in either steam generator in order to assure adequate steam generator inventory. The steam generator level sensing instrumentation shall be safety grade as required for all automatic emergency feedwater actuation signals.

11. The Restart Submittal Section 2.1.1.3.1.2 (Pressurizer Heaters) states that separation of Class 1E and non Class 1E circuits will be in accordance with Regulatory Guide 1.75 wherever practicable. Provide the actual details of your design and augment your discussion with the details of how your new design maintains separation and isolation between the redundant Class 1E portion of the design.

12. Provide sufficient electrical one line diagrams to facilitate our review of power assignments for ALL new and/or realigned equipment. Provide a detailed listing of all the above equipment which gives power source, power requirements and demonstrates the ability of each power source identified to provide the additional power requirements without degradation.

13. The Restart Submittal (Section 2.1.1.2) states that each pressurizer safety valve and the power operated relief valve will be provided with elbow tap differential pressure transmitters to measure downstream discharge flow through each valve. Provide the calculations, assumptions and descriptions of tests run by B&W which demonstrate that a satisfactory signal will be generated when any of the valves are open and conversely indicate positively that the valves have reclosed.

    Provide a complete description of the accelerometers, method of mounting on the PORV, and the test desctiption, including relief valve used, conditions, etc., which demonstrate that this device is suitable for the function being performed.

    We require that the indication of PORV and safety valve position be seismically qualified (rather than just mounted) and that the indication be environmentally qualified for the appropriate environment (any transient or accident which would cause the relief or safety valve to lift.)

    Describe the tests that will be performed after installation to verify satisfactory operation of the new position indicators.

14. Your response (Section 2.1.1.3.1.2) indicates that Babock & Wilcox has recommended at least 126 kw of pressurizer be assured power within 2 hours after a loss of offsite power. Provide or reference the calculations/tests which serve as the basis for this recommendation. Include all assumptions, conservatisms and capacity margin available. Address the situation for which the pressurizer may empty on a loss of offsite power, and therefore require more heater capacity to recover subcooling.

15. (Section 2.1.1.2) Include the use of the new pressurizer relief and safety valve indications in all appropriate procedures. For example, EP-1202-29 does not direct the operator to refer to these indications to identify a stuck open pressurizer safety valve. Submit revised procedures for our review.

16. We note that you intend (Section 10.2.8) to participate in an Industry Program to test the pressurizer relief and safety valves (NUREG-0578, 2.1.2). Provide the name of the industry group sponsoring the test program and a date when the program will be complete.

17. The requirements of paragraph 2.1.3.b of NUREG-0578 did not intend that short-term modifications to existing instruments should be considered to the exclusion of existing, unmodified instrumentation (such as reactor coolant pump current and flow measurement) to detect inadequate core cooling. The Met-Ed response considers only the modified instrumentation. Identify the existing, unmodified instruments which were considered for detection of inadequate core cooling. Describe those instruments which were selected for recognition of this condition. Perform analysis and implement procedures and training based on those unmodified instruments as well as the modified instrumentation described previously.

18. Identify those procedures which require the use of a) in-core thermocouples, b) wide range reactor outlet emperature measurement, c) reactor coolant saturation pressure margin, and d) other instrumentation identified in the response to Question #17 above.

19. The number of operational in-core thermocouples available to assess core conditions is a function of core design (which may change during future reloads) and thermocouple reliability. State the minimum number of as-designed in-core thermocouples considered acceptable in the Met-Ed proposal. State the minimum fraction of installed thermocouples which must be operational in the Met-Ed proposal. Show the analysis used to reach these conclusions.

20. [illegible] a description of the proposed subcooling meter as requested in Appendix B.

21. Rest[illegible] [illegible]ittal Section 2.1.1.5.1.3) explicitly state whether or not the non-ECCS s[illegible]t [illegible]rvices for RCP operation will be upgraded to Seismic Category I (N) [illegible]ec[illegible] rom BOTH pipe whip and jet impingement.

22. [illegible]y [illegible]entify the valves referred to in the Restart submittal Section [illegible] by penetration number and valve designation.

23. [illegible] Submittal Section 2.o.1.5.2) - Provide detailed descriptions for subparts [illegible], 3, 4, 5, 7, and 9. Provide sufficient electrical drawings to allw an independent evaluation for subparts 1, 2, 3, 4, 5, and 7. Provide documentation to support subpart 6.

24. Provide the documentation that supports your statement that the RCP's can run for one week with only seal water providing the cooling for the pump seals.

25. Restart Submittal Section 2.1.1.5.5.2 states:

> "Spurious initiation of an isolation signal will not introduce transients into the plant that are of significance. Thus, no new accidents/transients are introduced into the plant design."

Provide the detailed bases that support the above conclusions.

26. Your description of the Containment Isolation System does not specifically address if any of the isolated valves will automatically revert to their pre-isolated state (i.e., reopen). Address this aspect of your design.

27. (Restart Submittal Table 2.1-2)

a. Valves CM-V1, CM-V2, CM-V3 and CM-V4 do not receive a diverse safety grade automatic isolation signal. This is unacceptable. Modify your design accordingly.

b. Valve MU-V3 does not receive a diverse safety grade automatic isolation signal. This is unacceptable. Modify your design accordingly.

c. Valves IC-V2, IC-V3, IC-V4 and IC-V6 do not receive a diverse safety grade automatic isolation signal unless a safety grade line break isolation signal is included in the design. Sufficient details have not been provided to ascertain if this design feature is incorporated in your design. If the line break detection feature is incorporated in your design, provide the details supplemented with sufficient electrical drawings to allow an independent evaluation of whether or not the design meets IEEE Std. 279-1971. If the line break detection feature is not incorporated in your design, the design is unacceptable and must be modified accordingly.

d. Valves RB-V2A and RB-V7 are listed with three identical options and accompanying note. The first option provides automatic isolation only on hi-hi containment pressure. This option is unacceptable. Options two and three are acceptable providing that the hi-containment pressure isolation signal is retained. Modify your description by deleting the multiple choice options and providing the specific details of your design consistent with the above evaluation.

e. Valves MU-V33A, MU-V33B, MU-V33C and MU-V33D receive no automatic safety grade isolation signals. Containment hi-radiation is used to provide an alarm for subsequent remote-manual operator action. Provide the bases and rationale why diverse parameters are not included and why automatic isolation is not required. In the absence of acceptable justificaiton, we shall require single-failure-proof safety-grade automatic isolation of these valves.

28. The TMI-2 accident highlighted the fact that relatively high levels of radioactivity can be released inside containment without an associated significant pressure rise. In addition, a significant amount of reactor coolant was released through the PORV before the 4 psig setpoint was reached. It appears that the 4 psig setpoint may be too high a level to properly initiate containment isolation. Evaluate the merits of lowering the containment hi-pressure setpoint to something like 2.0 or 2.5 psig, and report your findings.

29. You state (Section 10.2) that item 1 of Bulletin 05A, which requires you to review a chronology of the TMI-2 accident so that an understanding of the events will ensure against such an occurrence at Unit 1, is "not applicable." Correct this statement and describe the adequacy of your review.

30. Your response (Section 8.2) to the requirements of item 3 in Bulletin 05B states that you are still investigating the changes to the PORV and high pressure reactor trip set points. Provide your commitment to make these changes or alternatives such that you respond to this bulletin item.

31. Your response (Section 2.1.1.1) to Bulletin 05B, item 5, only addresses control-grade reactor trips. Provide for our review a design/schedule for implementation of a safety-grade automatic anticipatory reactor scram prior to restart for loss of feedwater, turbine trip, and/or low steam generator level.

32. Your response (Section 3.1.1) to Bulletin 05A, items 3 and 4, and Bulletin 05B, item 1, indicates that procedures have been and are still being revised. Provide the necessary procedures for our review and/or schedule for their completion.

33. Your response (Section 3.1.1) to Bulletin 05C indicates that you are still evaluating this bulletin and will revise procedures and that supporting analyses will be submitted later. Provide a schedule for submitting this information. Provide a safety-grade pump trip design description.

34. Provide detailed scenarios of the two reactor trip/turbine trip events (#11, #12) discussed in section 10.3.1. Include (1) RCS temperature and corresponding pressure vs. time; (2) effects of liquid relief on PORV in Reactor Trip #11; and (3) corrective action for mainsteam safety valves failure to reseat (redesign new valves, etc.?). Were these events reported as LER's? If so, provide copies.

35. The small break LOCA analyses assumptions state that the TMI-1 conditions are more conservative than the generic analysis assumption. In this regard, justify an ESFAS trip for the plant at 1500 psig rather than 1600 psig as assumed in the generic analysis. (The lower TMI-1 pressure setting would result in later HPI initiation.) (p.8-16) Should you recommend changing the ESFAS trip to 1600 psig, confirm that the accident and transient analyses for TMI-1 will still show acceptable consequences.

36. It is stated that a small break LOCA analysis performed at 2535 MWt with an HPI split of 64%/36% is acceptable since the generic analysis is performed at 2772 MWt and a 70%/30% split. Justify the statement. (p.8-17).

    a. Provide a schedule for the HPI line break analysis which demonstrates that 250 gpm core makeup is adequate. (p.8-17)

    b. Provide drawings depicting the location of the venturis as well as the new cross connects.

    c. Describe the testing which will be done to confirm the adequacy of this design change and address the impact of this change on the cross connect modification

37. Your submittal states that cavitating venturis will be added to the high pressure injection lines to eliminate the need for operator action on an HPI line break.

38. Submit modified Technical Specifications to support the procedural and design modifications required by the Order and Bulletins 05A, 05B and 05C.

39. NUREG-0578, Section 2.1.3.b, requires the development of procedures to be used by the operator to recognize inadequate core cooling.

a. Describe the guidance provided the operator to recognize inadequate core cooling. Include existing and future instruments to be used and the expected instrument response under conditions of inadequate core cooling.

b. Describe the training provided licensed operators with regard to recognizing inadequate core cooling.

40. Your submittal indicates that the Lessons Learned requirements on Shift Supervisor Responsibilities (NUREG-0578, Section 2.2.1.a) will be provided at a later date. Provide the schedule for completion of this item.

41. Paragraph 5.4.5 of the Met-Ed/GPU TMI-1 Restart submittal indicates that the Shift Foreman should hold a Reactor Operator License, while Figure 5.2.1 indicates that a Senior Operator License is required. Since a Senior Operator License is necessary to direct the activities of Reactor Operators, revise paragraph 5.4.5 to require Shift Foremen to hold a Senior Operator License.

42. Paragraph 5.4.8 of the Met-Ed/GPU TMI-1 Restart submittal outlines the educational background requirements for a Shift Technical Engineer (STE).

*a. Provide the details of the training the STE will receive to assure a thorough knowledge of normal reactor operations, anticipated transients, and effects of multiple equipment failures and operational errors.

b. Describe the normal duties of the STE and his proximity to the control room.

c. Describe the responsibility and authority of the STE during off normal situations.

43. Provide the schedule for submittal of information required by NUREG-0578, Section 2.2.1.c Shift and Relief Turnover Procedures.

44. Provide the schedule for submittal of information required by NUREG-0578, Section 2.2.2 (a, b, c) Control Room Access, Onsite Technical Support Center, and Onsite Operational Support Center.

*Information Request at 10/17 Meeting

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45. (Order Item 1(d)) Your response to this item indicates that procedures have been or are still being revised. Provide the procedures developed to define operator action during small break LOCA's.

46. (Order Item 1(e))* All licensed operators at B&W facilities have received a special exam on the TMI-2 accident; including transient effects, operator response, and related procedure/design changes. Provide a similar exam for the TMI-1 licensed operators with a minimum passing grade of 90%. Also provide the details for retraining those individuals who may score less than the minimum.

47. (Order Item 1(e))* Section 6.6.D.1 of the Operator Accelerated Retraining Program (OARP) states that an audit exam will be given to all licensed operators. Provide the passing criteria for this examination and the retraining/reauditing for those who may not achieve the passing score.

48. (Order Item 1(e))* Outline how licensed operators will be provided with information related to procedure/design changes that are implemented after the completion of the related training module.

49. (Order Item 1(e))* Describe how the specific program objectives of the OARP (Section 6.2) will be factored into the future training and requalification of operators.

50. (Order Item 1(e)) To what extent will outside organizations be used to provide training in the OARP? This includes organizations that may be providing training material, instructors, audit exams, etc.

51. Bulletin 05A Item 3 What guidance is provided operators to enhance core cooling in the event that voids in the primary system are large enough to compromise core cooling capability, especially natural circulation capability?

52. Bulletin 05A Item 4 How does the OARP emphasize the use of various plant parameter indications in evaluating plant conditions?

53. Bulletin 05A Item 5 Your response to these item is incomplete. Outline how you intend to review all safety-related valve positions and positioning requirements to assure that valves are positioned in a manner to ensure proper operation of engineered safety features.

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*Information requested at 10/17/79 meeting

54. Bulletin 05A Item 5 Commit to perform an independent valve alignment verification when returning the emergency feedwater system to operability after maintenance or surveillance testing.

55. Bulletin 05B Item 1 Your procedure 1102-16, Natural Circulation, includes anticipatory filling of the OTSG prior to securing the reactor coolant pumps. Submit the analysis performed to provide guidance as to the expected system response.

56. Bulletin 05B Item 2 State the procedural guidance and training provided operating personnel with respect to the override/ termination of engineered safety features.

57. Bulletin 05B Item 6 Outline the guidance provided operating personnel for prompt notification of the NRC.

58. Bulletin 05B Item 7 Identify the procedural controls that have been implemented to assure proper positioning of the emergency feedwater system manual valves and manually operated motor-driven valves.

59. Bulletin 05B Item 9 Identify the procedures and guidance provided operators with respect to resetting containment isolation.

60. Bulletin 05B Item 10 Your response indicates that shift relief procedures will be used to notify reactor operating personnel when safety-related systems are removed from or returned to service. Regulatory Guide 1.47 addresses visual indications for bypassed and inoperable status of equipment. Outline the design features or procedural steps that provide visual indication at TMI-1.

61. Bulletin 05B Item 11 Your submittal indicates that the response to this item will be provided at a later date. Provide the schedule for completion of this item.

62. Bulletin 05C Item 1 Your response did not include Section B of this item. State your intention to provide two licensed operators in the control room at all times during operation.

63. Your submittal states that you will provide an automated switchover to recirculation on the emergency core cooling system. Provide a detailed description and associated drawings related to this modification.

64. Your submittal states that you intend to modify the reactor building spray system. Provide the detailed description and associated drawings related to this modification.

65. The Radiation Protection and Chemistry Supervisor or some other individual knowledgeable in health physics practices shall be present at PORC meetings whenever health physics related procedures or policies are reviewed. Revise Section 3.1 of the Restart Submittal to incorporate this practice in the PORC Quorum description.

66. The ANSI Standard (N18.1-1971) referenced in Section 5.4.10 of the Restart Submittal has been substantially revised. The current applicable standard is ANSI/ANS-3.1-1978. Provide assurance that the minimum qualifications of the Supervisor-Radiation Protection and Chemistry comply with those stipulated in ANSI/ANS-3.1.

67. The licensee has not described the proposed radiation protection program plan. We will require a detailed description, analogous to recent FSAR submittals, taking into account lessons learned and other additional considerations reflecting problem areas raised by the accident. Areas to be covered are characterized in Appendix C.

APPENDIX A

Basis for Auxiliary Feedwater System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of onsite and offsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above

b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

- Maximum RCS pressure (PORV or safety valve actuation)
- Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
- RCS cooling rate limit to avoid excessive coolant shrinkage
- Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

b. Time delay from initiating event to reactor trip.

c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

d. Minimum steam generator water level when initiating event occurs.

e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

h. RC flow condition - continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature

j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

Information on the Subcooling Meter

Plant Name: _____

Vendor: _____

Reference for Information: _____

Display

Information Displayed (T-Tsat, Tsat, Press, etc) _____

Display Type (analog, Digital, CRT) _____

Continuous or on Demand _____

Single or Redundant Display _____

Location of Display _____

Alarms (include setpoints) _____

Overall uncertainty (°F, PSI) _____

Range of Display _____

Qualifications (seismic, environmental, IEEE279) _____

Calculator

Type (process computer, dedicated digital or analog calc.) _____

If process computer is used specify availability. (% of time) _____

Single or redundant calculators _____

Selection Logic (highest T, lowest press) _____

Qualifications (seismic, environmental, IEEE279) _____

Calculational Technique (Steam Tables, functional fit, ranges) _____

Input

Temperature (RTD's or T/C's) _____

Temperature (number of sensors and locations) _____

Range of temperature sensors _____

Uncertainty*of temperature sensors (°F at 1 σ) _____

Qualifications (seismic, environmental, IEEE279) _____

Pressure (specify instrument used) _____

Pressure (number of sensors and locations) _____

Range of Pressure sensors _____

Uncertainty*of pressure sensors (PSI at 1 σ) _____

Qualifications (seismic, environmental, IEEE279) _____

Backup Capability

Availability of Temp & Press _____

Availability of Steam Tables etc. _____

Training of operators _____

Procedures _____

*Uncertainty assessment must address conditions of forced flow and natural circulation.

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TENTATIVE OUTLINE OF AREAS TO BE COVERED

1. Policy -

   Implementation
   Capability
   QA

2. Management -

   Responsibilities
   Communications
   Organization - reporting chain
   Staffing - clerical support and appropriate specialists
   Qualifications of Staff - knowledge of systems, procedures
   Training - frequency, scope
   Procedures

3. Communications -

   Status of plant activities
   H.P. problem identification
   Changes in procedure
   Maintenance planning
   Changes in RWP work scope

4. Personnel Dosimetry -

   Badges, pocket dosimeters
   Calculations
   Bioassay
   Whole Body counting
   Review and evaluation

5. Radiation Control -

   Access control - RWP's
   Monitoring - Portable - Fixed
   Control procedures - Adm. - ALARA

6. Contamination Control -

   R/A Mat'l control - Posting, etc.
   Monitoring Program
   Surface - Air
   Clothing
   Control Procedures/Boundary

7. Internal Dose Control -

   Resp. Protection
   Air Monitoring
   Equipment
   Analysis - qual/quant.

8. Design/Engineering reviews for Radiation Protection

9. Training -

   Plant - Management
   H.P. Tech - Records

10. Instrumentation -

    Capability to monitor for Iodines
    Selection, maintenance, calibration and numbers

11. Internal QA Audits

12. Special Problems -

    Beta Dosimetry
    Interface with Unit 2

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R. C. Arnold

October 26, 1979

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