NRC 3/31/80

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY ET AL.	)	Docket No. 50-289
	)	
	)	
(Three Mile Island, Unit 1)	)	

NRC STAFF RESPONSE TO UNION OF CONCERNED SCIENTISTS
FIRST AND SECOND SET OF INTERROGATORIES

Pursuant to 10 CFR §2.720 and 10 CFR §2.744, the NRC Staff has responded to all but 10 of the interrogatories posed by the Union of Concerned Scientists (UCS) in its first and second set of interrogatories to NRC Staff.[1/] Each interrogatory is restated and a response provided. Affidavits identifying the individuals who prepared responses and verifying them which are not included today will be forwarded at a later date.

Respectfully submitted,

Lucinda Low Swartz

Lucinda Low Swartz
Counsel for NRC Staff

Dated at Bethesda, Maryland
this 31st day of March, 1980.

1/ The Licensing Board stated that the Staff would not be required to answer the fifth "preliminary question" for each UCS contention which asks the Staff to identify all sections and page numbers of the SER or FSAR which contain subject matter pertaining to each contention. (Special Prehearing Conference, February 21, 1980. Tr. at 4). Accordingly, the Staff has not provided answers to Interrogatories 5, 18, 30, 37, 44, 54, 65, 79, 87, 96, 105, 112, 132, 154, and 167.

Answers not provided today will be filed on or before April 10, 1980.

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UCS Interrogatory 1

Explain the present Staff position on UCS Contention 1.

"The accident at Three Mile Island Unit 2 demonstrated that reliance on natural circulation to remove decay heat is inadequate. During the accident, it was necessary to operate at least one reactor coolant pump to provide forced cooling of the fuel. However, neither the short nor long term measures would provide a reliable method for forced cooling of the reactor in the event of a small loss-of-coolant accident ("LOCA"). This is a threat to health and safety and a violation of both General Design Criterion ("GDC") 34 and 35 of 10 CFR Part 50, Appendix A."

Response

The staff believes that the TMI-I reactor core can be adequately cooled by natural circulation following a small break LOCA or system transient. For break sizes greater than approximately 0.01 sq. ft., sufficient energy is discharged through the break so that the system can be depressurized without reliance on the reactor coolant (RC) pumps or natural circulation. For break sizes smaller than approximately 0.01 sq. ft. or system transients that result in tripping of the reactor coolant pumps, natural circulation will remove the decay heat. The staff position regarding natural circulation is discussed further in response to UCS Interrogatory 6. Therefore, the primary system can be depressurized such that core cooling can be established by the residual heat removal system.

If it is postulated that natural circulation can not be established, the core can be cooled in the feed and bleed mode. In this cooling mode, water is injected into the primary system (cold legs) by the high pressure injection system, and the core decay heat is removed through the pressurizer pilot operated

relief valve (PORV) and safety valves. The cutoff head of the high presssure injection pumps is above the pressure settings of the safety valves (2500 psig) so that injection can continue even though the system is at high pressure. This cooling mode can be maintained until the primary system can be depressurized using the steam generators.

UCS Interrogatory 2

Does the current position differ from the position of the Staff in any prior cases? If so, identify the case(s), explain the prior position, and explain the basis for the change in position.

Response

The position stated in Interrogatory 1 does not differ from the staff position on previous cases.

UCS Interrogatory 3

Identify any members of the Staff who dissent from the present Staff position on UCS Contention 1. Explain the reasons for which any Staff members dissented from the present Staff position on UCS Contention 1.

Response

No members of the staff are known to have a dissenting position on UCS Contention 1.

UCS Interrogatory 4

Identify the specific sections and page numbers of the SER and/or FSAR for TMI, Unit 1, which are relied upon in formulating the Staff position on UCS Contention 1.

Response

The SER or FSAR for TMI, Unit 1, were not relied upon for this position.

UCS Interrogatory 14

Explain the present staff position on UCS Contention 2.

"Using the existing equipment at TMI-1, there are only 3 ways of providing forced cooling of the reactor: 1) the reactor coolant pumps; 2) the residual heat removal system; and 3) the emergency core cooling system in a "bleed and feed" mode. None of these methods meets the NRC's regulations applicable to systems important to safety and is sufficiently reliable to protect public health and safety:

a) The reactor coolant pumps do not have an on-site power supply (GDC 17), their controls do not meet IEEE 279 (10 CFR 50.55a(h)) and they are not seismically and environmentally qualified (GDC 2 and 4).

b) The residual heat removal system is incapable of being utilized at the design pressure of the primary system.

c) The emergency core cooling system cannot be operated in the bleed and feed mode for the necessary period of time because of inadequate capacity and radiation shielding for the storage of the radioactive water bled from the primary coolant system."

Response

a) Analyses have shown that the reactor coolant pumps are not needed to assure adequate core heat removal for any normal or accident situation. Therefore, there are currently no specific plans to designate them as components important to safety.

b) A high pressure residual heat removal system is currently not required. Decay heat can be removed at the design pressure of the primary system using natural circulation through the steam generators with feed supply from the emergency feedwater system. The recent improvement and increased reliability of this system are described in our January 11, 1980 Status Report and the TMI-1 Restart Report.

c) Should the "feed and bleed" mode be ultimately necessary, the quantity of the borated water supply in the Borated Water Storage Tank, which provides the water source to the high pressure injection pumps, does not limit the operation in this mode. Water "bled" from the reactor coolant system through either the postulated break or pressurizer safety or relief valves would collect in the reactor building recirculation sump. Suction of the emergency core cooling system (ECCS) low pressure injection pumps would be aligned to take suction on this sump and discharge to the suction of the high pressure injection pumps so that a continuous "feed" can be delivered to the reactor coolant system. Storage of the radioactive water bled from the primary system would be in the reactor building recirculation sump and movement of this water would take place in the ECCS which would be operating in the recirculation mode. Integrity and shielding of systems, including the ECCS, following an accident are part of the Lessons Learned requirements

in NUREG-0578 (2.1.6) which are being implemented on TMI-1. These areas are still under review as described in our January 11, 1980 Status Report.

UCS Interrogatory 15

Does the current position differ from the position of the Staff in any prior cases? If so, identify the case(s), explain the prior position, and explain the basis for the change in position.

Response

A review of all prior "cases" was not conducted; however, no differences in position are known.

UCS Interrogatory 16

Identify any members of the Staff who dissent from the present Staff position on UCS Contention 2.

Response

No members of the Staff are known to have a dissenting position on UCS Contention 2.

UCS Interrogatory 17

Identify the specific sections and page numbers of the SER and/or FSAR for TMI, Unit 1, which are relied upon in formulating the Staff position on UCS Contention 2.

Response

The TMI-1 safety analysis is in Section 14 of the FSAR and the staff evaluation of these analyses in Section 15 of the staff safety evaluation. These documents alone were not used to formulate the staff position. Design requirements for the reactor coolant pumps, the residual or decay heat removal systems, and the recirculation capability of the ECCS are derived from the General Design Criteria in Appendix A to 10 CFR Part 50. The analyses in the TMI-1 FSAR, the analyses for other PWRs, and additional analyses performed since the TMI-2 accident form the basis of our position regarding this contention.

26. Explain the present Staff position on UCS Contention 3.

Response

The present staff position on UCS Contention No. 3 is that the pressurizer heaters and their controls are not presently characterized as "safety-grade" in the context of having to meet all applicable safety-grade criteria; but that the safety classification of these components will be evaluated and the classification will be upgraded if this is deemed necessary in order to preclude exceeding the acceptance criteria for any design basis event. This Staff position and its rationale are defined in detail in the Staff response to Interrogatories Nos. 31 and 32.

27. Does the current position differ from the position of the Staff in any prior cases? If so, identify the case(s), explain the prior position, and explain the basis for the change in position.

Response

The present staff position differs from the position on prior cases only with respect to the present requirement that "pressurizer heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability." This is discussed in detail in the staff response to Interrogatory 31. However, the difference is moot since this new requirement is being backfit to all operating PWR's and all PWR's in the licensing process.

28. Identify any members of the Staff who dissent from the present Staff position on UCS Contention 3. Explain the reasons for which any Staff members dissented from the present Staff position on UCS Contention 3.

Response -

No members of the NRC Staff are known to dissent from the position stated in Interrogatory 26.

29. Identify the specific sections and page numbers of the SER and/or FSAR for TMI, Unit 1, which are relied upon in formulating the Staff position on UCS Contention N.

Response

All of the documents cited in the staff response to Interrogatory No. 31 have been used in formulating the staff position on UCS Contention No. 3. This includes TMI-1 Safety Evaluation Report, July 11, 1973, Sections 1.5.2, 3.8.2, 5.0 and 18.0. The TMI Unit 1 FSAR was not used directly for this formulation.

33. Explain the present Staff position on UCS Contention 4.

Response

The Staff position on Contention 4 is stated in NUREG-0578, Item 2.1.1. The progress on this subject is documented in the Staff's Status Report dated January 11, 1980, Section C8-2.1.1.

34. Does the current position differ from the position of the Staff in any prior cases? If so, identify the case(s), explain the prior position, and explain the basis for the change in position.

Response

The Staff position on Contention 4 does not differ from the position of the Staff in any prior cases.

35. Identify any members of the Staff who dissent from the present Staff position on UCS Contention 4. Explain the reasons for which any Staff members dissented from the present Staff position on UCS Contention 4.

Response

No members of the Staff are known to have a dissenting position on Contention 4.

36. Identify the specific sections and page numbers of the SER and/or FSAR for TMI, Unit 1, which are relied upon in formulating the Staff position on UCS Contention 4.

Response

The SER or FSAR for TMI-1 were not relied on in formulating the Staff position on Contention 4.

38. Explain why the addition of the pressurizer heater to the on-site emergency power supplies will not degrade the capacity, capability, and realibility of the on-site emergency power source in violation of GDC 17.

Response

A. The reasons why the addition of the pressurizer heater to the on-site emergency power supplies will not degrade the capacity, capability, and reliability of the on-site emergency power source in violation of GDC 17 are explained in NUREG-0578, Item 2.1.1.

B. NUREG-0578 TMI-1 Restart Report

C. There were no other documents considered by the Staff which pertain to the subject matter questioned.

D. No further research is being done or is planned.

E. The experts have not yet been identified.

39. Explain why installation of an independent and redundant on-site emergency power supply for the pressurizer heater would not provide greater reliability of power supply to pressurizer heaters.

Response

A. The addition of an independent dedicated emergency power supply for the pressurizer heaters would not provide greater reliability of power supply than the approved TMI-1 modification. Assuming the additional onsite emergency power supply would be Class IE as are the existing onsite emergency power supplies, the individual unreliabilities of the three power supplies would be essentially equal. From a deterministic analysis, it can be readily seen that the approved design meets the single failure criterion with respect to power supply and the proposal by Intervenors does not.

B. TMI-1 Restart Report

C. There were no other documents considered by the Staff which pertain to the subject matter questioned.

D. No further research is being done or is planned.

E. The experts have not yet been identified.

48. Explain how the motive and control components of the PORV's and their associated block valves and the vital instruments shall be supplied by the on-site emergency power source when offsite power is not available without degrading the capacity, capability and reliability of emergency power in violation of GDC 17.

Response

A. The PORV, PORV Block Valve and pressurizer level instrumentation are all powered from emergency busses as part of the original approved TMI-1 design. No plant modifications were required to meet the associated short term Lessons Learned recommendations (Item 2.1.1). Therefore, the capacity, capability, and reliability have not been degraded and the plant remains in conformance with GDC 17.

B. TMI Restart Report.

C. There were no other documents considered by the Staff which pertain to the subject matter questioned.

D. No further research is being done or is planned.

E. The experts have not yet been identified.

49. How have the devices through which motive and control power components for the PORVs and their associated block valves are connected to emergency buses been qualified in accordance with safety-grade requirements?

Response

A. As discussed in Question 48 above, this is part of the original TMI-1 design. The devices in question are part of the emergency onsite power system. For details of the environmental qualification program for the original TMI-1 design, refer to the TMI-1 FSAR.

B. TMI-1 FSAR. TMI-1 Restart Report.

C. No other documents were considered by the Staff which pertain to the subject matter questioned.

D. No further research is being done or is planned.

E. The experts have not yet been identified.

UCS Interrogatory 40

Explain the present Staff position on UCS Contention 5.

"Proper operation of power operated relief valves, associated block valves and the instruments and controls for these valves is essential to mitigate the consequences of accidents. In addition, their failure can cause or aggravate a LOCA. Therefore, these valves must be classified as components important to safety and required to meet all safety-grade design criteria."

Response

The staff position on this contention is discussed in the response to UCS Interrogatories 45, 46, and 47.

UCS Interrogatory 41

Does the current position differ from the position of the Staff in any prior cases? If so, identify the case(s), explain the prior position, and explain the basis for the change in position.

Response

A review of all prior cases was not conducted; however, no differences in position are known.

UCS Interrogatory 42

Identify any members of the Staff who dissent from the present Staff position on UCS Contention 5. Explain the reasons for which any Staff members dissented from the present Staff position on UCS Contention 5.

Response

No members of the staff are known to have dissenting views on this position. However, as discussed in Interrogatory 152, a staff member does have a dissenting opinion concerning the related issue of control system response and malfunction during accidents.

UCS Interrogatory 43

Identify the specific sections and page numbers of the SER and or FSAR for TMI Unit 1, which are relied upon in formulating the Staff position on UCS Contention 5.

Response

The TMI-1 SER and FSAR were not relied upon in formulating the staff position on UCS Contention 5.

Interrogatory 45

Does the Staff agree that proper operation of PORV's, associated block valves and the instruments and controls for these valves is essential to mitigate the consequences of accidents? Explain your response fully.

Response

Proper operation of the PORVs, associated block valves, and the instrument and controls is not required to mitigate the consequences of any current design basis accident.

In view of the TMI-2 event, which involved a stuck-open PORV, the consequences of several incidents involving a stuck-open PORV were analyzed by B&W and reviewed in NUREG-0565. The analyses considered three cases: (1) a loss of main feedwater flow accident with offsite power available and one HPI train operational; (2) loss of offsite power and one HPI train operational; and (3) loss of offsite power, one HPI train operational and no auxiliary feedwater flow available. For cases 1 and 2, in which AFW was available, there was no core uncovery. For case 3, actuation of one additional HPI train or one AFW pump before 40 minutes would prevent core uncovery. These analyses were based on use of the ANS decay heat curve values multiplied by a conservatism factor of 1.2. If the conservatism factor had not been used, core uncovery would not be predicted to occur with only one operational HPI train and no auxiliary feedwater.

Credit was not taken for closure of the pressurizer block valve in any of these cases. Hence, the pressurizer block valve was not essential to mitigate the consequences of the stuck-open PORV.

50. Explain the present Staff position on UCS Contention 6.

Response -

The Staff position has always been that the reactor coolant system safety and relief valves, be designed, fabricated, and tested to quality standards which are in keeping with the required safety functions to be performed. The safety and safety relief valves at TMI-1 were constructed to the standards accepted by the Commission at the time of construction, and as such did meet GDC 1, 14, 15, and 30. These standards should, however, be continually evaluated to determine their adequacy, and if required be modified to insure safe plant operation. This position is in keeping with GDC 1 and has not changed.

As part of continued evaluation and upgrading of standards, and based on experience of TMI lessons learned, the following questions have been raised by the staff with respect to PORV's and safety valves.

-- Can the PORV's and safety valves, and their respective supports and piping, sustain the loads imposed during accident conditions.

-- Should PORV's, block valves, and control circuitry be upgraded to safety grade classification.

These questions will be resolved by testing and as specified by NUREG-0578.

51. Does the current position differ from the position of the Staff in any prior cases? If so, identify the case(s), explain the prior position, and explain the basis for the change in position.

Response

There has been no change in the Staff position as described above.

52. Identify any members of the Staff who dissent from the present Staff position on UCS Contention 6. Explain the reasons for which any Staff members dissented from the present Staff position on UCS Contention 6.

Response

There are no known Staff members who dissent from the above stated position.

53. Identify the specific sections and page numbers of the SER and/or FSAR for TMI, Unit 1, which are relied upon in formulating the Staff position on UCS Contention 6.

Response

As specified by FSAR Section 4.1.3.4, the Reactor Coolant safety valves were designed and tested in accordance with Article 9, Section III of the ASME Code. The safety valve capacity was based on the design basis accident of a bank rod withdrawal from low initial power as described in the FSAR Section 4.3.7.

UCS Interrogatory 56

Did the Staff fully apply the analysis of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2, to determine the expected valve operating conditions? If not, provide the justification for failing to do so.

Response

Regulatory Guide 1.70, Revision 2, was issued in 1975, well after the issuance of the operating license for TMI-1 and, hence, was not used in the preparation of the SAR or the staff evaluation of the SAR.

It should be noted that the intent of the reference to Regulatory Guide 1.70 in the staff position of Section 2.1.2 in NUREG-0578 was that the maximum pressure and temperature limits of the testing program be based on the transients and accidents analyzed in the FSAR. The staff did not intend to have new analyses performed to obtain these testing limits which were part of a short-term staff requirement. Hence, in reference 1, which provided clarification of the Lessons Learned Short-Term Requirements, the restriction to Revision 2 of Regulatory Guide 1.70 was eliminated.

Reference 1. Letter of 10/30/79 from H. Denton to Licensees entitled "Discussion of Lessons Learned Short-Term Requirements."

57. Explain how the licensee chose the single failures applied to these analyses so as to maximize the dynamic forces on the safety and relief valves.

Response

The Staff has no information concerning any single failures used to maximize the dynamic forces on the safety and relief valves.

The maximum dynamic force on the safety and relief valves would result from a full capacity lift for the design base accident of a control rod assembly bank withdrawal from low initial power as described in the FSAR section 4.3.7. The safety valves were tested prior to installation as requied by the ASME Code Section III article 9 for both set pressure and capacity. Also they were set-pressure tested as installed during pre-operational testing. The relief valves have been treated as control devices and as such were not tested to Code requirements.

NUREG-0578 has required applicants and licensees to qualify safety and relief valves for design basis accident and transient conditions. (See response #55.)

UCS Interrogatory 61

Explain the present Staff position on UCS Contention 7.

"NRC regulations require instrumentation to monitor variables as appropriate to ensure adequate safety (GDC 13) and that the instrumentation shall directly measure the desired variable. IEEE 279, 4.8, as incorporated in 10 CFR 50.55a(h), states that:

> To the extent feasible and practical protection system inputs shall be derived from signals which are direct measures of the desired variables.

"TMI-1 has no capability to directly measure the water level in the fuel assemblies. The absence of such instrumentation delayed recognition of a low water level condition in the reactor for a long period of time. Nothing proposed by the staff would require a direct measure of water level or provide an equivalent level of protection. The absence of such instrumentation poses a threat to public health and safety."

Response

The present staff position on UCS Contention 7 is described in Section 2.1.3b of the "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations," NUREG-0578, July 1979, and in the clarification letter on this subject to all operating reactors on October 30, 1979. As indicated in those reports, it is the staff's position that core cooling (i.e., control of core temperature) is the important variable to be monitored to assure adequate safety. The staff's requirement can be summarized as a requirement for an unambiguous and easy-to-interpret indication of inadequate core cooling. The unambiguous measurement of water level in the reactor vessel is a reliable but

indirect indication of core cooling. A reliable measurement of water level in the reactor vessel would, therefore, be an acceptable means of fulfilling the staff's requirement. However, other means of accomplishing this function would also be considered.

UCS Interrogatory 62

Does the current position differ from the position of the staff in any prior cases? If so, identify the case(s), explain prior position, and explain the basis for the change in position.

Response

The current staff position described in the response to Interrogatory 61 was implemented by a September 13, 1979 letter to the licensees of all operating reactors. This position will also be implemented on all future applications.

Prior to September 13, 1979, the staff did not require monitoring of core cooling (or of reactor vessel water level as an indication of core cooling) on any cases. The basis for the change in position is presented in the "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations," NUREG-0578, July 1979, Section 2.1.3b.

UCS Interrogatory 63

Identify any members of the Staff who dissent from the present Staff position on UCS Contention 7. Explain the reasons for which any Staff members dissented from the present Staff position on UCS Contention 7.

Response

No members of the staff are known to have a dissenting position on UCS Contention 7.

UCS Interrogatory 64

Identify the specific sections and page numbers of the SER and/or FSAR for TMI-1, Unit 1, which are relied upon in formulating the Staff position on UCS Contention 7.

Response

The TMI-1 SER and FSAR were not used in formulating the staff position on UCS Contention 7.

Interrogatory 69

Explain how present procedures and instrumentation permit prompt recognition of low reactor coolant level and inadequate core cooling.

Response

The emergency procedures have been revised to include operator guidance to permit prompt recognition of inadequate core cooling. According to procedure 1202-39, "Inadequate Core Cooling," the following conditions are symptoms of inadequate cooling:

1. Superheated conditions in the RCS as indicated by any of the following instrumentation:
   a. Saturation meter.
   b. Reactor Coolant Outlet Temperature (Th) wide or narrow range indication and RC pressure narrow or wide range indication.
   c. In-core thermocouple temperature indication and RC narrow or wide range pressure indication.

2. Saturated conditions in the RCS with increasing RC pressure and temperature as indicated by any of the following instrumentation:
   a. Saturation meter.
   b. Reactor Coolant Outlet Temperature (Th) wide or narrow range indication and RC narrow or wide range pressure indication.
   c. In-core thermocouple temperature indication and RC narrow or wide range pressure indication.

NOTE: The following symptoms do not by themselves indicate inadequate core cooling but do provide possible indications of inadequate core cooling. They may also indicate that conditions are developing which could lead to inadequate core cooling.

3. Significant decrease in RCS Total Flow as recorded on console "cc" (control room teminology to indicate console location) with no change in the number or RC pumps in operation.

4. Decreasing RC pump motor current as indicated by the meter located by the motor control switches.

5. Increasing count rate as indicated by the Source Range Nuclear Instrumentation.

Interrogatory 70

What short term modifications of existing procedures and/or instruments have been made at TMI-1 for monitoring water level in the fuel assemblies?

Response

No such short term modifications have been made at TMI-1 for directly monitoring water level in the fuel assemblies. Met-Ed has noted its agreement with the B&W long term resolution of this item, which does not require the direct monitoring of water level in the fuel assemblies. The staff has not yet received the details of the B&W proposal to resolve this concern. Met-Ed intends to monitor the adequacy of core cooling as identified in responses to questions 69 and 71.

Interrogatory 71

How do any such new procedures and/or instruments differ from those in place prior to the accident at TMI-2?

Response

The licensee has changed its emergency procedures to include reference to RCS conditions which could indicate inadequate core cooling. In addition to the procedural guidance now provided in Ep 1202-39 (see response to question 69), the revised procedures require operators to monitor RCS conditions and maintain a 50°F margin to saturation. This guidance was not provided in the procedures prior to the accident at TMI-2.

The licensee has also improved the instrumentation available to the operator to identify the existence or possible onset of inadequate core cooling. Such improvements include the installation of a saturation meter, extension of the T_H indicating range, and connection of the in-core thermocouples to the computer.

Interrogatory 72

Describe the training program to inform reactor operators of new procedures. Document the number of hours of instruction on these new procedures.

Response

A review of the major procedures (new and revised) was conducted as part of the Operator Accelerated Retraining Program (OARP). This review consisted of a primary instructor (with a backup instructor monitoring lesson content) discussing the procedures with the licensed personnel and noting the significance of procedural steps. Eighteen hours were allotted for this review. In addition, procedure changes made after the formal training session, will be reviewed by licensed personnel via the revision review book.

Interrogatory 73

Explain how any modifications proposed by the staff will provide more direct measurement of reactor core cooling level and inadequate core cooling. What is the implementation schedule for any proposed modification?

Response

The staff has required the licensee to provide a description of any additional instrumentation or controls proposed to provide an unambiguous indication of inadequate core cooling. Water level indication is considered by the staff to provide a more direct measurement of reactor vessel level and the licensee has been required to consider this indication in his evaluation of additional instrumentation. Met-Ed has not yet provided us with a description of any additional measures to detect inadequate core cooling. Babcock and Wilcox was scheduled to provide such information to the licensee by February 1, 1980.

UCS Interrogatory No. 75

Explain the present Staff position on UCS Contention 8.

"10 CFR 50.46 requires analysis of ECCS performance for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the entire spectrum of postulated loss-of-coolant accidents is covered. For the spectrum of LOCA's, specific parameters are not to be exceeded. At TMI, certain of these were exceeded. For example, the peak cladding temperature exceeded 2200° fahrenheit (50.56(b)(1), and more than 1% of the cladding reacted with water or steam to produce hydrogen (50.46(b)(3). The measures proposed by the Staff address primarily the very specific case of stuck-open power operated relief valve. However, any other small LOCA to the same consequences. Additional analyses to show that there is adequate protection for the entire spectrum of small break locations have not been performed. Therefore, there is no basis for finding compliance with 10 CFR 50.46 and GDC 35. None of the corrective actions to date have fully addressed the demonstrated inadequacy of protection against small LOCA's."

Discussion

The spectrum of break analyses applicable to TMI-1 is discussed in response to UCS Interrogatory No. 80. These analyses were performed for a range of break sizes and locations, and show that the requirements of 10 CFR 50.46 are satisfied.

Our response indicates that revised calculations were to be submitted by the licensee to verify conformance with 10 CFR 50.46 as a result of an issue raised in April 1978. Our review of the documenation submitted confirms that these

revised calculations were submitted, reviewed and approved by the staff. The licensee was notified [1,2] that TMI-1 would be in conformance with 10 CFR 50.46 upon completion of the installation and testing of a modification of the ECCS high pressure injection piping. Metropolitan Edison has committed to install and test this modification prior to restart. The modification and testing are discussed in the latest report and are now under review by the NRC staff.

Since the TMI-2 accident, a number of actions have been required by the NRC to assure that the probability and consequences of small break LOCAs are mitigated. These items are discussed in various NRC documents such as the August 9, 1979 Order and Notice of Hearing, and the reports of the Bulletins and Orders Task Force (NUREG-0565 and 0623).

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1. Letter from R. W. Reid (NRC) to J. G. Herbein (Met-Ed), dated December 8, 1978.
2. Letter from R. W. Reid (NRC) to J. G. Herbein (Met-Ed), dated September 26, 1978

UCS Interrogatory No. 76

Does the current position differ from the position of the Staff in any prior cases? If so, identify the case(s), explain the prior position, and explain the basis for the change in position.

Discussion

The position stated in Interrogatory No. 75 does not differ from the staff position on previous cases.

UCS Interrogatory No. 77

Identify any members of the Staff who dissent from the present Staff position on UCS Contention 8. Explain the reasons for which any Staff members dissented from the present Staff position on UCS Contention 8.

Discussion

No members of the staff are known to have a dissenting position on UCS Contention 8.

UCS Interrogatory No. 78

Identify the specific sections and page numbers of the SER and/or FSAR for TMI, Unit 1, which are relied upon in formulating the Staff position on UCS Contention 8.

Discussion

The SER or FSAR for TMI, Unit 1 were not relied upon for this position.

CONTENTION 9:

The accident at TMI-2 was substantially aggravated by the fact that the plant was operated with a safety system inoperable, to wit: two auxiliary feedwater system valves were closed which should have been open. The principal reason why this condition existed was that TMI does not have an adequate system to inform the operator that a safety system has been deliberately disabled. To adequately protect the health and safety of the public, a system meeting the Regulatory Position of Reg. Guide 1.47 or providing equivalent protection is required.

Interrogatory 83:

Explain the present staff position on UCS Contention 9.

Response:

The Staff agrees with the first sentence of Contention 9. With respect to the remainder of the Contention, the adequacy of the existing status monitoring system at TMI-1 and the matter of any necessary design and related procedural changes are presently being reassessed by the Staff. See, also, the response to Interrogatory 91 for further detail.

Interrogatory 84:

Does the current Staff position differ from the position of the Staff in any prior cases? If so, identify the case(s), explain the prior position, and explain the basis for the change in position.

Response:

As stated in the response to Interrogatories 83 and 91, the Staff is reassessing the need to backfit Regulatory Guide 1.47. Thus, pending development of the final Staff position with respect to backfitting Reg. Guide 1.47 at TMI-1, no direct response can be given to this interrogatory.

Interrogatory 85:

Identify any members of the Staff who dissent from the present Staff position on UCS Contention 9. Explain the reasons for which any Staff members dissented from the present Staff position on UCS Contention 9.

Response:

The response to Interrogatory 84 applies.

Interrogatory 86:

Identify the specific sections and page numbers of the SER and/or FSAR for TMI, Unit 1, which are relied upon in formulating the Staff position on UCS Contention 9.

Response:

The response to Interrogatory 84 applies.

Interrogatory 91:

Is it the position of the Staff that TMI-1 can be operated with adequate protection for the public health and safety without a determination by the Staff that TMI-1 is now, finally, in full compliance with Regulatory Guide 1.47? Explain your answer fully.

CONTENTION 10:

The design of the safety systems at TMI is such that the operator can prevent the completion of a safety function which is initiated automatically; to wit: the operator can (and did) shut off the emergency core cooling system prematurely. This violates §4.16 of IEEE 279 as incorporated in 10 CFR 50.55 (a)(h) which states:

> The protection system shall be so designed that, once initiated, a protection system action shall go to completion.

The design must be modified so that no operator action can prevent the completion of a safety function once initiated.

Interrogatory 92:

Explain the present Staff position on UCS Contention 10.

Response:

The Staff agrees with the first sentence of Contention 10.

The Staff does not agree with the second sentence for reasons based principally upon the distinction between "safety function" and "protection system action." (See the responses to Interrogatories 97 and 98).

With respect to the last sentence of Contention 10, the Staff does not generally require the designs of engineered safety features (ESF) systems to be such that the operator cannot interrupt the safety function at any time subsequent to initiation. One reason is that the safety advantages of an ESF safety function that cannot be prevented by the operator from going to completion must be weighed against the potentially adverse effects on safety that could, under certain circumstances, result from continued operation of the system. For example, it may be necessary to shut off a damaged ESF pump prior to completion

of the safety function in order to prevent the loss of its physical integrity from aggravating the event.

Interrogatory 93:

Does the current position differ from the position of the Staff in any prior cases? If so, identify the case(s), explain the prior position, and explain the basis for the change in position.

Response:

No.

Interrogatory 94:

Identify any members of the Staff who dissent from the present position on UCS Contention 10. Explain the reasons for which any Staff member dissented from the present Staff position on UCS Contention 10.

Response:

None have been identified.

Interrogatory 95:

Identify the specific sections and page numbers of the SER and/or FSAR for TMI, Unit 1, which are relied upon in formulating the Staff position on UCS Contention 10.

Response:

None

Response:

The Staff is presently reassessing the matter of backfitting Regulatory Guide 1.47 at operating plants. The Office of Nuclear Reactor Regulation will study the need for all licensees and applicants not presently committed to the requirements of Reg. Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems" to monitor and verify operations, test and maintenance activities by means of an automatic status monitoring system such as that described in Regulatory Guide 1.47. This study is to be performed following a review of procedures and other nonautomatic actions to verify these activities.

The Staff position with respect to backfitting Reg. Guide 1.47 at TMI-1 will be developed subsequent to completion of the aforementioned study.

CONTENTION 12

Question 113. With respect to TMI-1 list the "structures, systems, and components important to safety" within the containment and auxiliary buildings to which GDC 4 presently applies.

Answer The structures, systems and components important to safety are listed in the TMI-1 FSAR in Section 1.4.1.

Question 114. What were and are the maximum environmental parameters which each such GDC 4 structure, system, and component is qualified to withstand?

Answer The environmental parameters to which structures, systems and components were qualified to withstand correspond to conditions that exceed those which are calculated to occur during either a loss of coolant accident or a steam line break. The TMI-1 FSAR does not provide time dependent profiles of these parameters. However, IE Bulletin 79-01B dated January 14, 1980 requested a correlation between the environmental data requirements specified in the FSAR and the environmental qualifications test data for each item of Class IE electrical equipment which is identified as important to safety. The staff review of this response will provide additional assurance that all equipment important to safety is acceptably qualified.

Question 115. To what extent did the actual accident conditions at TMI-2 exceed the past and present maximum environmental parameters for each such structure, system, and component discussed in question #113?

Answer The accident environment of the TMI-2 accident was less severe than the parameters of pressure and temperature to which systems and components were qualified. Since access to the containment has not been made at this time, an evaluation to assess the radiation exposure to systems and components

important to safety has not been made. Investigations in this area are planned and ongoing and the results will be evaluated when the data is available.

Question 116. What was and is the length of time in an accident environment for which each such structure, system, and component is qualified to remain operable?

Answer Systems and components were qualified to operate in an accident environment for a period of time consistent with the design basis for the plant. The FSAR did not address operability requirements for each specific item important to safety. However, IE Bulletin 79-01B requested the time interval that the component or equipment is required to function including the document which provides the basis for this time interval as well as time interval that was demonstrated by the equipment qualification program.

Question 117. To what extent did the actual accident conditions at TMI-2 exceed the past and present time periods for which each such structure, system, and component is qualified to remain operable under accident conditions?

Answer As noted in the response to Question 116 above, the time interval for which equipment was qualified will be identified in the response to IE Bulletin 79-01B. However, as noted in the response to Question 115 above, radiation is the only parameter for which component and equipment may have been exposed at levels exceeding that for which the equipment was qualified. As noted, investigations are ongoing which will address the radiation exposure to systems and components important to safety.

Question 118. Describe in detail the method used to qualify each such safety structure, system and component as meeting GDC 4. Provide the relevant documentation.

Answer The FSAR did not describe in detail the method used to qualify each system and component important to safety. A summary of the qualification program for each

item will be provided in response to IE Bulletin 79-01B. A copy of this information will be available when it is submitted in response to the Bulletin.

Question 119. Is each such structure, system and component qualified according to the criteria of IEEE-323-1974, as modified by Regulatory Guide 1.89?

Answer The staff's evaluation of the operating license application for TMI-1 was issued on July 11, 1973 and did not address IEEE-323-1974 since the later was issued in 1974. Therefore, IEEE-323-1974 and Regulatory Guide 1.89 were not used as a basis to judge the adequacy of the environmental qualification of systems and components important to safety. The extent to which the environmental qualification of systems and components important to safety may satisfy the requirements of IEEE-323-1974 and Regulatory Guide 1.89 will be addressed in the staff's review of the response to IE Bulletin 79-01B.

Question 120. For each system, structure or component not qualified according to the criteria of IEEE-323-1974 as modified by Regulatory Guide 1.89, provide the criteria by which it was qualified.

Answer As noted in the response to Question 119 above, IEEE-323-1974 was issued following the environmental qualification of systems and components important to safety and the staff's review thereof. The criteria used for the qualification of structure, systems, and components important to safety are given in Sections 1.4.1, 1.4.23 and 7.1.1.7 of the TMI-1 FSAR.

Question 121. List all equipment within the containment and auxiliary buildings previously deemed to be qualified which failed, either wholly or partially, during or after the accident at TMI-2. Describe the way in which each such piece of equipment failed and the reason(s) for the failure.

Answer As noted in the staff's response to the UCS Petition on Indian Point Units 1, 2, and 3: " the staff knows of no Class IE instrumentation at TMI that has failed to provide the required accuracy during or after the TMI event."

Question 122. When was the decision made not to "backfit" Regulatory Guide 1.89, incorporating IEEE-323-1974?

Answer The proposed Regulatory Guide 1.89, "Qualification of Class IE Equipment for Nuclear Power Plants," was reviewed and recommended for approval by the Regulatory Requirements Review Committee during Meeting No. 15 on October 9, 1974 and included its implementation as "forward fit" as noted in the guide. Consideration for "backfit" of Regulatory Guide 1.89 was not subsequently given by the Regulatory Requirements Review Committee.

Question 123. Provide all contemporaneous documentation supporting the decision against backfitting Regulatory Guide 1.89, incorporating IEEE-323-1974.

Answer As noted in response to question 122 above, a decision for backfitting Regulatory Guide 1.89 was not made by the Regulatory Requirements Review Committee.

Question 124 What is the Staff's present rationale for failing to now require compliance with Regulatory Guide 1.89, incorporating IEEE-323-1974, for TMI-1?

Answer As noted in the response to questions 114 and 116 above, the licensee has been requested to conduct a thorough review of the environmental

qualification program for TMI-1 in response to IE Bulletin 79-01B. In addition this review will include consideration of equipment aging as was indicated in the staff's response to the UCS petition of September 17, 1979 on Indian Point Unit 1, 2, and 3. If any equipment is found to be deficient based upon this review, the staff will require requalification or replacement with suitably qualified equipment. Therefore this review will assure that structures, systems, and components are acceptably qualified and will address the acceptability of deviations from the guidance provided in IEEE-323-1974 and Regulatory Guide 1.89.

Question 125. Is it the position of the Staff that the methods and assumptions used during the licensing and review of TMI-1 to environmentally qualify safety-related systems, structures and components (including environmental parameters, testing and analysis methods, length of time for which equipment must remain functional, and identification of "safety-related" equipment) are adequate to fully comply with a) GDC 4 and b) Regulatory Guide 1.89?

Answer The TMI-1 FSAR does not make a specific reference to GDC 4. As noted in Section 1.4 of the FSAR, the licensee addressed the general design criteria for nuclear power plant construction permits as then listed in the Federal Register 50.34 Appendix A which were applicable to this unit. As noted in Section 7.1.1 of the FSAR, the licensee (then applicant) concluded that the reactor protection system is designed to meet all the requirements of IEEE-279 and the Commission's proposed General Design Criteria. As noted in the response to Question 119 above, Regulatory Guide 1.89 did not exist during the evaluation of TMI-1 applicantion for an operating license. Thus, the staff is not in a position to conclude the extent to which the environmental qualification of structures, systems, and components for TMI-1 fully comply with Regulatory Guide 1.89. With respect to GDC 4, the requirements of Criterion 1 - Quality Standards and Criterion 23 - Protection Against Multiple Disability for Protection System as addressed in Section 1.4.1 and 1.4.23 of the TMI-1 FSAR embrace the essential reuqirements of GDC 4. Thus, the staff concludes that the methods and assumption used during the licensing and review of TMI-1 are adequate to fully conform to GDC 4.

Question 126. If the answer to Question 125 is other than an unqualified, "yes," state in what manner those methods and assumptions were deficient in any respect.

Answer The staff does not conclude that those methods and assumptions were deficient in any respect.

Question 127. If the answer to Question 125 is other than an unqualified "yes," state which of the short and/or long term measures recommended by the Staff will correct those deficiencies. Explain your answer fully.

Answer The Staff has highlighted by the issuance of IE Circulars and Bulletins, the important lessons learned from environmental qualification deficiencies reported by individual licensees. In this regard, licensees were requested to examine installed safety-related electrical equipment and determine that proper documentation existed which provided assurance that this equipment would function under postulated accident conditions. By the issuance of IE Bulletin 79-01B, a complete and thorough review of the environmental qualification of system and components important to safety will be conducted by all licensees and will be reviewed by the staff. Thus, if any deficiencies are found, regardless of the detail and basis upon which they were previously accepted, they will be corrected. The staff concludes that this action will provide assurance as to the adequacy of environmental qualification for TMI-1.

Interrogatory 133

Question

Does the staff take the position that all credible accidents have been included within the design bases for TMI-1? Provide all relevant documentation supporting your conclusion.

Response

In the past the staff required that the applicant analyze and discuss the consequences of a number of transients and accidents to demonstrate that the plant can be operated without undue risk to the health and safety of the public. In specifying such design basis events, the staff recognized that they did not describe all of the possible events that could occur at nuclear power plants. However, they were considered to cover events which should be considered in terms of likelihood of occurrence and severity of consequences. In view of the conservative methods and acceptance criteria used in evaluating the design basis events the intent was to envelop a broad spectrum of likely events and consequences. However, the staff has not limited its consideration only to design basis events. Postulated events such as anticipated transients without scram (ATWS), loss of all A.C. power, and high energy line breaks with sequential failure of control systems are under review by the staff (see e.g., references 1 and 2).

The consequences to the public resulting from the TMI-2 accident were well below the 10 CFR Part 100 guidelines. However, as discussed in NUREG-0585, the accident did involve a sequence of events more severe than those included in current design basis events. Hence, the TMI-2 long-term Lessons Learned Task Force found a need to "supplement current design requirements" and "include

certain design features for mitigating accidents that are not provided by the set of design basis events." They are discussed in Recommendations 8, 9, and 10 of NUREG-0585.

Since the TMI-2 accident there have been a number of reviews by the staff and other organizations and a large number of actions already taken or planned with the objective of reducing the probability of occurrence of a TMI-2 accident or other similar accidents in the future. With respect to TMI-1, reference 3 summarizes over 100 corrective measures to meet this objective. These include measures to prevent or reduce the probability of occurrences of initiating events, measures to reduce the probability of occurrences of subsequent failure, and measures directed at dealing effectively with and mitigating the consequences of a core melt.

References

1. U.S. Nuclear Regulatory Commission, 1978 Annual Report.

2. Memorandum from Paul S. Check (NRC) to Darrell G. Eisenhut (NRC) re "Status Report - High Energy Line Break With Consequential Control System Failure," dated December 19, 1979.

3. U.S. Nuclear Regulatory Commission, Supplement to NRC Staff Position on Need to Consider Class 9 Events (to be filed with the Board on or before March 21, 1980).

Interrogatory 138

Question

Prior to the accident at TMI-2, did the Staff have an opinion as to the probability of the TMI-2 accident? What was the opinion?

Response

Prior to the accident at TMI-2, the Staff did not have an opinion on the probability of the TMI-2 accident. A class of accidents including many features of the TMI-2 accident was considered in the risk assessment of the Surry reactor in WASH-1400. Probabilitities for specific sequences in WASH-1400 are not expected to be applicable to plants of differing procedures or design, such as TMI.

141. Provide all draft and final analyses, memoranda, reports, recommendations or other documents produced by, relied upon or consulted by the Staff relating to the probability or consequences of accidents beyond the current design basis and/or measures designed to mitigate the consequences of such accidents.

Response

The Staff does not rely upon analyses of the probability or consequences of accidents beyond the design basis in regulatory decisionmaking. A list of documents "consulted by" includes anything anyone in the Staff may ever have read bearing upon the subject. Such a list clearly cannot be constructed. A bibliography of some of the principal publications bearing upon these issues is attached.

Bibliography of
Class 9 Accident Studies

1. Reactor Safety Study, WASH-1400, U.S.N.R.C., October 1975.

2. Three Mile Island--A Report to the Commissioners and to the Public, NRC Special Inquiry Group, January 1980.

3. Effect of Containment Venting on the Risk from LWR Meltdown Accidents, NUREG/CR-0138, Battelle Columbus Laboratories, June 1978.

4. A Value-Impact Assessment of Alternate Containment Concepts, NUREG/CR-0165, Sandia Laboratories, June 1978.

5. Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants, NUREG-0611, U.S.N.R.C., January 1980.

6. Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering Designed Operating Plants, NUREG-0635, U.S.N.R.C., January 1980.

7. Analysis of the Three Mile Island Accident and Alternative Sequences, NUREG/CR-1219, Battelle Columbus Laboratories, January 1980.

8. Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock and Wilcox Designed 177-FA Operating Plants, NUREG-0565, U.S.N.R.C., January 1980.

9. Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company, NUREG-0560, U.S.N.R.C., May 1979.

10. Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission, NUREG/CR-0400, Ad Hoc Risk Assessment Review Group, September 1978.

11. Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors, WASH-1270, Regulatory Staff of U.S.A.E.C., September 1973.

12. Anticipated Transients Without Scram for Light Water Reactors, NUREG-0460, U.S.N.R.C., April 1978.

Emergency Planning Research Within RES/PAS Since the RSS

The following reports have been published:

1. McGrath, P. E., D. M. Ericson, Jr., and I. B. Wall, "The Reactor Safety Study (WASH-1400) and Its Implications for Radiological Emergency Response Planning," International Symposium on the Handling of Radiation Accidents, 28 February 1977, Vienna, Austria, IAEA-SM-215/23.

2. Aldrich, D. C., D. M. Ericson, Jr., and J. D. Johnson, Public Protection Strategies for Potential Nuclear Reactor Accidents: Sheltering Concepts with Existing Public and Private Structures, SAND77-1725, Sandia Laboratories, Albuquerque, NM (1977).

3. Aldrich, D. C. and D. M. Ericson, Jr., Public Protection Strategies in the Event of a Nuclear Reactor Accidents: Multi-compartment Ventilation Model for Shelters, SAND77-1555, Sandia Laboratories, Albuquerque, NM (1977).

4. Aldrich, D. C., R. M. Blond, and R. B. Jones, A Model of Public Evacuation for Atmospheric Releases, SAND78-0092, Sandia Laboratories, Albuquerque, NM (1978).

5. Ericson, D. M. Jr., Accident Descriptions for Emergency Response Exercise Scenarios, SAND78-0269, Sandia Laboratories, Albuquerque, New Mexico (1978).

6. Aldrich, D. C., P. E. McGrath, and N. C. Rasmussen, Examination of Offsite Radiological Emergency Protective Measures for Nuclear Reactor Accidents Involving Core Melt, SAND78-0454, Sandia Laboratories, Albuquerque, NM (1978).

7. Cohen, A. F., and B. L. Cohen, Infiltration of Particulate Matter into Buildings, SAND79-2079, Sandia Laboratories, Albuquerque, NM (1979).

8. Aldrich, D. C., D. M. Erikson, N. C. Finley, N. R. Ortiz, J. L. Sprung, and J. M. Taylor, Emergency Response Scenarios for Transportation Accidents Involving Radioactive Materials, SAND79-2017, Sandia Laboratories, Albuquerque, NM (October 1979).

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REFERENCE LIST OF STUDIES AND
IMPROVEMENTS TO CRAC

Overview of the Reactor Safety Study Consequence Model, NUREG-0340, U.S. Nuclear Regulatory Commission, Washington, DC, 20555, June 1977.

Aldrich, D. C., P. E. McGrath, and N. C. Rasmussen, Examination of Offsite Radiological Emergency Protective Measures for Nuclear Reactor Accidents Involving Core Melt, Sandia Laboratories' Report (to be published).

Effects of Wind Shear on the Consequence Model of the Reactor Safety Study, J. L. Sprung and H. W. Church, SAND76-0619, NUREG-0175, Sandia Laboratories, Albuquerque, NM, 87185, January 1977.

Sensitivity of the Reactor Safety Study Consequence Model to Mixing Heights, J. L. Sprung and H. W. Church, SAND76-0618, NUREG-0174, Sandia Laboratories, Albuquerque, NM, 87185, January 1977.

Influence of Plume Rise on the Consequences of Radioactive Material Releases, A. J. Russo, J. R. Wayland, L. T. Ritchie, SAND76-0534, Sandia Laboratories, Albuquerque, NM, January 1977.

Effects of Rainstorm and Runoff on Consequences of Nuclear Reactor Accidents, L. T. Ritchie, W. D. Brown, J. R. Wayland, SAND75-0429, Sandia Laboratories, Albuquerque, NM, October 1976.

McGrath, P. E., D. M. Ericson, Jr., and I. B. Wall, "The Reactor Safety Study (WASH-1400) and Its Implications for Radiological Emergency Response Planning," International Symposium on the Handling of Radiation Accidents, 28 February 1977, Vienna, Austria, IAEA-SM-215/23.

Aldrich, D. C., D. M. Ericson, Jr., and J. D. Johnson, Public Protection Strategies for Potential Nuclear Reactor Accidents: Sheltering Concepts with Existing Public and Private Structures, SAND77-1725, Sandia Laboratories, Albuquerque, NM (1977).

Aldrich, D. C. and D. M. Ericson, Jr., Public Protection Strategies in the Event of a Nuclear Reactor Accident: Multi-compartment Ventilation Model for Shelters, SAND77-1555, Sandia Laboratories, Albuquerque, NM (1977).

Aldrich, D. C., R. M. Blond, and R. B. Jones, A Model of Public Evacuation for Atmospheric Releases, SAND78-0092, Sandia Laboratories, Albuquerque, NM (1978).

Ericson, D. M. Jr., Accident Descriptions for Emergency Planners, Sandia Laboratories' Report (to be published)

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145. Given that TMI-2 has been identified by the Staff as a Class 9 accidents pose the greatest threat to the public safety of all possible nuclear reactor accidents, explain how the Staff can adequately protect the public without consideration of Class 9 accidents and consequences.

Response

A response to this question is contained in "Supplement to NRC Staff Position on Need to Consider Class 9 Events" filed with the Board on March 21, 1980.

UCS Interrogatory 150

Explain the present Staff position on UCS Contention 14.

Response

As noted in Reference 1,* "Current practice in the licensing of nuclear power plants is to apply design requirements to one class of components, equipment, systems, and structures, the so-called safety-grade class, but not to another nonsafety-grade class. This system of classification is based on the premise that things can be classified either as important to safety (that is, the function is credited in the analysis of a design basis event or is specified in the regulations) or not important to safety." Reference 1 also states that "there is a general requirement that failure of nonsafety-grade equipment or structures should not initiate or aggravate an accident" and that "the term 'failure' when applied to nonsafety-grade equipment has generally been defined as 'failure to operate upon demand'". The above general requirement is not in the Standard Review Plan (SRP). For example, Reference 2, page 17, cites Section 7.7 of the SRP which gives a general guideline for reactor control systems** However, as noted in Reference 2, there are no guidelines in the SRP that generally apply to the many other nonsafety systems. It is also stated in Reference 2 that "the Task Force will reassess this approach to consider the need to expand the regulatory coverage to other systems such as the power conversion system and the auxiliary systems."

In References 1 and 2, the staff has presented detailed discussions of possible weaknesses of this "current practice." The quote from Reference 2 given in UCS

*References 1-5 can be found after the response to Interrogatory 153.

**References 3, 4, and 5 contain discussions of the staff position with respect to control systems and credit for nonsafety-grade in the steam line break accident.

Contention 14 is a portion of these discussions and deals in part with the role played by the nonsafety-grade systems in initiating the series of events involved in the TMI-2 accident and used in mitigation of the accident in ways not previously considered in the safety analysis.

In the discussion of recommendations with respect to improvements in plant design, the Lessons Learned Task Force stated (Ref. 1, page 3-3):

> "The Task Force concludes that comprehensive studies of the interaction of non-safety-grade components, equipment, systems and structures with safety systems and the effects of these interactions during normal operation, transients, and accidents need to be made by all licensees and license applicants (see Recommendation 9). This would constitute a significant alteration of the current unresolved safety issue concerning systems interaction. The Office of Standards Development has previously been requested to develop a Regulatory Guide that would specify generic requirements for some safety-related systems that do not presently fall within the safety-grade classification. This effort would have to be closely coordinated with the study by licensees that we are now recommending. In the interim, the effects of the abnormal conditions that accompany transients and accidents on the operation and failure of non-safety-grade items should be reviewed by all licensees to determine if there are any probable adverse interactions. The extent of simultaneous interactions considered in this review should reflect the number of non-safety systems simultaneously exposed to conditions for which they were not designed. Equipment identified as the cause of unacceptable interactions should be appropriately modified to reduce the probability of

that interaction, or the safety system that is adversely affected should be modified to cope with the interaction. In either event, operating procedures and operator training must be expanded to include consideration of the possible permutations and combinations of non-safety-grade system interactions with safety systems."

It is expected that the results of studies performed in accordance with Recommendations 8 and 9 of Reference 1 will be reflected in changes to the Standard Review Plan dealing with the staff position on nonsafety-grade systems.

In summary, as quoted in UCS Contention 14, the NRC staff has identified a need to examine whether further requirements may be necessary to improve the current capability for use of nonsafety systems during transient or accident situations. This need has been translated into the program described in Action Plan IIC of NUREG-0660. The present position of the Staff is that satisfactory completion of the short-term actions and reasonable progress in the long-term actions identified in the Order will provide reasonable assurance that the facility can be operated without endangering the health and safety of the public.

Interrogatory 151

Does the current position differ from the position of the Staff in any prior cases? If so, identify the case(s), explain the prior position, and explain the basis for the change in position.

Response

The current position differs from the previous position with respect to the staff actions already taken and planned as discussed in the response to Interrogatory 150 and the reference cited in that response. The basis for the changes in position are discussed in detail in the references.

UCS Interrogatory 152

Identify any members of the Staff who dissent from the present Staff position on UCS Contention 14. Explain the reasons for which any Staff members dissented from the present Staff position on UCS Contention 14.

Response

Mr. Demetrois Basdekas raised technical issues involving nonsafety-grade equipment which are discussed in detail in References 3, 4, and 5. No other members of the staff are known who dissent from the present staff position on UCS Contention 14.

UCS Interrogatory 153

Identify the specific sections and page numbers of the SER and/or FSAR for TMI, Unit 1, which are relied upon in formulating the Staff position on UCS Contention 14.

Response

The SER and FSAR for TMI, Unit 1 were not relied on in formulating the staff position on UCS Contention 14.

References

1. NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report," October 1979.

2. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979.

3. Memo, H. Denton, NRC, to Commissioner J. F. Ahearne , NRC, "Safety Implications of Control Systems and Plant Dynamics," October 22, 1979.

4. NUREG-0153, "Staff Discussion of Twelve Additional Technical Issues Raised by Responses to November 3, 1976 Memorandum From Director, NRR to NRR Staff," December 1976.

5. NUREG-0138, "Staff Discussion of 15 Technical Issues Listed in Attachment to November 3, 1976 Memo from Director, NRR to NRR Staff," November 1976.

163. Explain the present staff position on UCS Contention 16.

Response

The NRC Policy Statement (44FR61123) with regard to the Environmental Protection Agency and the Nuclear Regulatory Commission task force report on guidance for use in emergency response planning is the current staff position on emergency response planning for nuclear power reactors accidents.

The major recommendation of the report is that two Emergency Planning Zones (EPZs) should be established around light water nuclear power plants. The EPZ for airborne exposure has a radius of about 10 miles; the EPZ for contaminated food has a radius of about 50 miles. Predetermined protective action plans are needed for the EPZs. The exact size and shape of each EPZ will be decided by emergency planning officials after they consider the specific conditions at each site. These distances are considered large enough to provide a response base which would support activity outside the planning zone should this ever be needed.

The Commission is directing its staff to incorporate the planning basis guidance into existing documents used in the evaluation of state and local emergency response plans to the extent practicable. The NRC has recently published an Advance Notice of Proposed Rulemaking concerning additional regulations on emergency plans, 44FR41484, Tuesday, July 17, 1979. Additional guidance will be provided following this rulemaking. This additional guidance can be expected to consider how local conditions such as demography, land use, and meteorology can influence the size an shape of the EPZs and to address other issues, such as evacuation planning.

Specific implementation dates for full implementation of the task force, recommendations and any others that are developed will be established as part of the ongoing rulemaking effort.

164. Does the current position differ from the position of the staff in any prior cases? If so, identify the case(s), explain the prior position, and explain the basis for the change in position.

Response

The current staff position has developed as a result of the accident at TMI. The current position is generic to all operating nuclear power plants, including TMI.

165. Identify any members of the Staff who dissent from the present Staff position on UCS Contention 16. Explain the reasons for which any Staff members dissented from the present Staff position on UCS Contention 16.

Response

There are no known members of the Staff who dissent from the present Staff position on UCS Contention 16.

166. Identify the specific sections and page numbers of the SER and or FSAR for TMI, Unit 1, which were relied upon in formulating the Staff position on UCS Contention 16.

Response

The Staff's position on emergency preparedness for the restart of TMI Unit 1 is based on the Restart Report Section 4.

The Staff's position on the generic issue of emergency planning is based on the NRC Policy Statement.

UCS Interrogatory 192

With respect to IE Bulletin 79-05A, Item #2 (p. C2-2), what was the staff's basis for requiring review of only those transients similar to the Davis-Besse event which occured at TMI-1 rather than a review of all similar transients at all B&W facilities?

Response

Item #2 of IE Bulletin 79-05A required TMI-1 to review any transients at their facility that were similar to both the Davis-Besse event and the TMI-2 event (Enclosure 1 of IE Bulletin 79-05A). The purpose of this particular bulletin item was for the utilities to determine if any significant deviations from expected performance occurred for transients of this type at their facility. It was the staff's opinion, at the time of the issuance of this bulletin, that these two events provided adequate examples for TMI-1 to reevaluate any similar events that may have occurred at their facility.

UCS Interrogatory 193

Is it the staff's position that a design (as distinguished from procedures or training) which permits operators to override automatic actions of engineered safety features before the safety function goes to completion meets the commission's regulations. For example (this is only an example), does the staff take the position that a design which permits operators to prevent a core cooling system from going to completion meets the regulations?

Response

Yes, it is the staff's position that a design which permits operators to override automatic actions of engineered safety features before the safety function goes to completion does not violate the regulations. The protection systems are designed so that once they are initiated, the safety function shall go to completion if no operator action is taken. The operator does have the capability of stopping any ECCS equipment and this flexibility is designed into these systems to reduce the potential for such occurrences as damage to system due to component or support system failure.

UCS Interrogatory 195

The staff notes the "extreme seriousness and consequences" of the simultaneous blocking of all auxiliary feedwater trains. (p. C2-8) Describe the "extreme seriousness and consequences" referred to.

Response

The "extreme seriousness and consequences" referred to applies to the blocking of a system whose function is assumed in the plant safety analysis and the resulting violation of plant Technical Specifications which require availability of the system function.

The consequences of a loss of all feedwater have been evaluated more fully since the issuance of IE Bulletin 79-05A, which is referred to above. Based on the small break analyses performed by B&W, a loss of all feedwater with (1) an isolated PORV, but safety valves opening and closing as designed or (2) a stuck-open PORV does not result in uncovering the reactor core, provided either EFW or HPI (two pumps) is initiated within 20 minutes. See response to Interrogatory 191 and NUREG-0565.

209. With respect to both groups of persons identified in the two previous answers, provide a statement of their educational background, training, and qualifications.

Response

Professional qualifications statements for R. Tedesco, R. Vollmer, and B. Grimes are attached. A statement for F. Pagano will be forwarded as soon as possible.

ROBERT L. TEDESCO

PROFESSIONAL QUALIFICATIONS

DIVISION OF OPERATING REACTORS

OFFICE OF NUCLEAR REACTOR REGULATION

In 1979, I was appointed to be the Acting Deputy Director for Operating Reactor in the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. Previously, I was the Assistant Director for Reactor Safety and was with the TMI-2 Lessons Learned Task Group.

I accepted an appointment with the technical staff of the AEC Regulatory organization in 1964. I have had primary responsibility for safety reviews of various reactor plants. In 1972, I was appointed Assistant Director for Containment Safety in the Division of Technical Review and in 1975 became the Assistant Director for Plant Systems in the Division of Systems Safety. In these positions, I was responsible for the review of the safety aspects of nuclear power plants for systems in support of the reactor.

My formal education was obtained at the University of Connecticut where I received the B.A. degree in Mathematics and related Sciences in 1951. Subsequently, I attended Trinity College and received the M.S.

degree in Physics with additional course work in Electrical Engineering in 1959. I have also taken engineering courses at Rensselaer Polytechnic Institute, the University of Connecticut, and the University of Hartford. Additional post-graduate courses in Nuclear Engineering were taken at the Catholic University of America. These courses were of a graduate level and included reactor thermal analysis, core design, reactor systems evaluation, and reactor safety studies.

From 1957 to 1960, I was employed at Combustion Engineering, Naval Reactors Division, located in Windsor, Connecticut. My responsibilities included various studies on reactor and plant dynamics and the safety evaluation of the SIC prototype submarine propulsion reactor. These studies included transient thermal analyses of the reactor plant during various startup and confirmatory test programs of the SIC submarine base plant. From 1960 to 1964, I was employed at the CANEL office of Pratt and Whitney Aircraft Company which was located in Middletown, Connecticut. I participated in the preparation of the operational program for a proposed high temperature liquid metal cooled reactor for potential aircraft and space applications. My responsibilities included the preparation of the safety evaluation for the proposed reactor plant and the development of experimental techniques for determining reactor stability during the confirmatory test programs.

In 1971, I participated with members of the Regulatory Task Force for the Emergency Core Cooling System (ECCS) initial reappraisal effort for boiling water reactors. I am also involved with foreign programs on reactor safety principally in the area of containment testing.

RICHARD H. VOLLMER

PROFESSIONAL QUALIFICATIONS

My name is Richard H. Vollmer. I am Assistant Director for Site Analysis in the Division of Site Safety and Environmental Analysis, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission. I was formerly Chief of the Quality Assurance Branch in the Office of Nuclear Reactor Regulation. In the latter position, I was responsible for the review and evaluation of the quality assurance programs proposed for reactor design, construction, and operation;of the qualification of plant staffing for operation; and of the program for initial tests and operations.

I received a B. S. Degree in physics from the University of Notre Dame in 1952. I was then employed by the DuPont Company at the Argonne National Laboratory and at the Commission's Savannah River Plant and participated in experiments and analytical studies dealing with production reactor design, operation, and safety. In 1958 I joined Allis-Chalmers and directed the design, construction, and operation of the prototype reactor experiments for the Pathfinder boiling water reactor and other technical and design activities. In this capacity, I managed a multi-disciplined organization,part of which had responsibility for an analytical and design activity and part of which was responsible for procurement of equipment, installation and construction, and testing and operation.

In 1962 I was employed by Atomics International and directed the reactor test analysis and safety work associated with the SNAP (Systems for Nuclear

Application in Space) prototype reactor testing conducted at full power and temperature. In this capacity, I managed a multi-disciplined analysis unit. Later at Atomics International, I was the project manager for the physics and the safety studies associated with an advanced heavy water reactor concept.

I joined the Commission in 1968 as a Lead Reactor Engineer in the Division of Reactor Licensing and became Technical Coordinator for Reactor Operations in 1970. In that position, I was responsible as a technical specialist in many facets of reactor operation. In 1972, I was designated Chief of the Quality Assurance Branch and have been involved in standards work and formulation of Commission regulations and guides on QA matters, and have been responsible for the development of the Standard Review Plans in the areas of conduct of operations, initial test programs, and quality assurance.

BRIAN K. GRIMES

PROFESSIONAL QUALIFICATIONS

OFFICE OF NUCLEAR REACTOR REGULATION

I am employed as Assistant Director for Engineering and Projects in the Division of Operating Reactors, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D. C. I am currently on special assignment as Director, Emergency Preparedness Task Group in the Office of Nuclear Reactor Regulation. I am also the NRC Cochairman on the joint NRC/Federal Emergency Management Agency (FEMA) Steering Committee for Emergency Preparedness. Responsibilities under my current assignments include directing the activities of personnel in the review of emergency plans for operating power reactors, operating licenses and construction permits and coordinating NRC and FEMA efforts in the review of emergency preparedness at and around nuclear power plant sites.

I attended the University of Washington, Seattle, Washington, and received a BS degree in Chemical Engineering in 1962 and a MS degree in Nuclear Engineering in 1964. While completing my graduate work, I was employed as a research assistant at the University of Washington Engineering Experiment Station; my duties involved performing analytical and experimental work on the University of Washington research reactor.

In 1963, I accepted employment with the Division of Reactor Licensing, USAEC. My first assignment involved attendance at the International Institute for Nuclear Science and Engineering at Argonne National Laboratory for four months. Upon completion of this course, I was assigned as a Nuclear Engineer in the

Division of Reactor Licensing. My initial duties included primary responsibility for the continuing review of the nuclear safety aspects of various research reactors. I subsequently participated in the safety evaluation of a number of construction permit applications for both pressurized and boiling water power reactors.

Later, as a Reactor Project Engineer in the Division of Reactor Licensing, I had primary responsibility for the safety review of the construction permit application for the Commonwealth Edison Company's Quad-Cities Units 1 and 2, for the Duke Power Company's Oconee Nuclear Station Units 1, 2 and 3, for the Metropolitan Edison Company's Three Mile Island Nuclear Station Unit 1, and for the Indiana & Michigan Electric Company's Donald C. Cook Nuclear Plant Units 1 and 2. I was assigned to the position of Technical Coordinator for Reactor Projects in October, 1968. Prior to March, 1970, I served as Technical Coordinator for both pressurized and boiling water reactors. After March, 1970, as Technical Coordinator for Boiling Water Reactors, my responsibilities included coordinating the technical aspects of all safety reviews in the Boiling Water Reactor group, providing liaison with the pressurized water reactor group and serving as administrative assistant to the Assistant Director for Boiling Water Reactors.

I was assigned to the position of Chief of the Radiological Safety Branch, Division of Reactor Licensing in July, 1971, in which position I was responsible for the review of systems necessary for the control and treatment of radioactivity under normal and accident conditions. In January, 1972, the functions of this branch were divided and I was appointed Chief of the Accident Analysis

Branch. My responsibilities as Chief of the Accident Analysis Branch included reviewing calculational models, procedures and methods developed by members of the Branch for both conservative assessment and a realistic assessment of the consequences of a spectrum of accidents for all nuclear power plants and reviewing analyses of all nuclear power reactor sites performed by members of the Branch with regard to site related hazards and compliance with the guidelines of 10 CFR Part 100. In January, 1976, I was assigned to the position of Chief of the Environmental Evaluation Branch in the newly formed Division of Operating Reactors. In this position my responsibilities included supervising the review of radiological and non-radiological impacts of operating nuclear power plants from both a safety and environmental standpoint. Branch review areas included accident analyses, site-related hazards, effluent treatment systems, off-site radiological effects, and thermal and chemical effluents.

On April 1, 1978 I was appointed Assistant Director for Engineering and Projects in the Division of Operating Reactors. In this position my responsibilities included managing the activities of the Engineering Branch, the Environmental Evaluation Branch, Operating Reactors Project Branch No. 3, Operating Reactors Project Branch No. 4 and the Standard Technical Specification Group. On June 25, 1979, I was assigned Acting Assistant Director for Systems Engineering in the Division of Operating Reactors, and managed the Plant Systems Branch and the Reactor Safety Branch. On October 25, 1979, I was designated Director of the Emergency Preparedness Task Group reporting to the Director of the Office of Nuclear Reactor Regulation.

I am a member of the American Nuclear Society.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, et al.	)	Docket No. 50-289
	)	
(Three Mile Island, Unit 1)	)	

AFFIDAVIT OF ROBERT G. FITZPATRICK

I, Robert G. Fitzpatrick, being duly sworn, do depose and state:

1. I am a Senior member of the Power Systems Branch in the Division of Systems Safety, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for the electrical aspects of the safety review of assigned nuclear power plants, including Three Mile Island, Unit 1 Restart Program.
2. The answers to UCS Interrogatories 33, 34, 35, 36, 38, 39, 48 and 49 were prepared by me. I certify that the answers given are true and accurate to the best of my knowledge.

Robert G. Fitzpatrick

Subscribed and sworn to

before me this 22 day of

_____.

_____

Notary Public

My Commission expires: _____

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, ET AL.	)	Docket No. 50-289
	)	
(Three Mile Island, Unit 1)	)	

AFFIDAVIT OF HARLEY SILVER

I, Harley Silver, being duly sworn, do depose and state:

1. I am a Senior Project Manager in the Division of Project Management, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for managing the safety review of assigned nuclear power plants, including Three Mile Island, Unit 1 Restart Program.

2. The answers to UCS' Interrogatory 13 was prepared by me. I certify that the answers given are true and accurate to the best of my knowledge.

Harley Silver

Subscribed and sworn to
before me this 14th day of

Elizabeth Ann Tipton
Notary Public

My Commission expires: July 1, 1982

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the matter of	)	
	)	
METROPOLITAN EDISON COMPANY, et al.	)	Docket No. 50-289
	)	
(Three Mile Island, Unit 1	)	

AFFIDAVIT OF EDGAR G. HEMMINGER

I, Edgar G. Hemminger, being duly sworn, do depose and state:

1. I am a Mechanical Engineer in the Division of Systems Safety, Mechanical Engineering Branch, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for review and evaluation of structural integrity, operability, and functional capability of safety related mechanical and electrical equipment.

2. The answers to UCS' Interrogatoreis 24, 55, 57, 59, and 60 were partially prepared by me. I certify that the answers given are true and accurate to the best of my knowledge.

Edgar G. Hemminger

Subscribed and sworn to
before me this 12 day of
_____

Notary Pubiic

My Commission expires: 12-6-82

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY & LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, et al.	)	Docket No. 50-289
	)	
(Three Mile Island, Unit 1)	)	

AFFIDAVIT OF FRANK C. CHERNY

1. I am a Mechanical Engineer in the Division of Systems Safety, Office of Nuclear Reactor Regulation. I am responsible for managing the safety review of the mechanical engineering aspects of assigned nuclear power plants, including Three Mile Island, Unit 1 Restart Program.

2. The answers to UCS Interrogatories 24, 55, 57, 59, and 60 were partially prepared by me. I certify that the answers given are true and accurate to the best of my knowledge.

Frank C. Cherny

Subscribed and sworn
to before me this 22 day of
May, 1980.

Notary Public

My Commission expires: 4-30-1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, et al.	)	Docket No. 50-289
	)	
(Three Mile Island, Unit 1)	)	

AFFIDAVIT OF BYRON L. SIEGEL

I, Byron L. Siegel, being duly sworn, do depose and state:

1. I am a senior reactor engineer in the Division of Systems Safety, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for reviewing accident analyses and ECC and RHR systems of assigned nuclear plants.
2. The answer to Union of Concerned Scientists Interrogatory 66 was prepared by me. I certify that the answer given is true and accurate to the best of my knowledge.

Byron L. Siegel

Byron L. Siegel

Subscribed and swron to
before me this 24 day of
Dec., 1982

Notary Public

My Commission expires: July 1, 1985

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, et al.	)	Docket No. 50-289
	)	
(Three Mile Island, Unit 1)	)	

AFFIDAVIT OF DONALD F. SULLIVAN

I, Donald F. Sullivan, being duly sworn, do depose and state:

1. I am a Senior Nuclear Engineer in the Office of Standards Development of the United States Nuclear Regulatory Commission. I am responsible for preparing assigned standards, codes, and guides in the area of nuclear power plant instrumentation and electric systems.

2. The answers to UCS' Interrogatories 67, 88, 89, 90, 97, and 98 were prepared by me. I certify that the answers given are true and accurate to the best of my knowledge.

Donald F. Sullivan

Subscribed and sworn to
before me this 13 day of
December, 1982.

Notary Public

My Commission expires: July 1, 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
METROPOLITAN EDISON COMPANY, et al.	)	Docket No. 50-289
(Three Mile Island, Unit 1)	)	

AFFIDAVIT OF DOUGLAS V. PICKETT

I, Douglas V. Pickett, being duly sworn, do depose and state:

1. I am a Containment Systems Engineer in the Division of Systems Safety, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for reviewing the containment related systems of assigned nuclear power plants, including Three Mile Island, Unit 1 Restart Program. My professional qualifications statement is attached.

2. The answers to the Union of Concerned Scientists' Interrogatories 101, 102, 103, 104, 106, 107 and 205(a) were prepared by me. I certify that the answers given are true and accurate to the best of my knowledge.

Douglas V. Pickett

Subscribed and sworn to
before me this ___ day of
_____.

_____
Notary Public
My Commission expires: _____

PROFESSIONAL QUALIFICATIONS

Douglas V. Pickett

I am a Containment Systems Engineer in the Containment Systems Branch of the Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission. In this position, which I have held since October 1975, I am responsible for the review and technical evaluation of containment related aspects for PWR applications for both construction permits and operating licenses. Among the plants for which I have or have had this responsibility are Davis Besse Nuclear Power Station, Unit 1; Arkansas Nuclear One, Unit 2; Erie Nuclear Plant, Units 1 and 2; Virgil C. Summer Nuclear Station; Shearon Harris Nuclear Power Plant, Units 1, 2, 3 and 4; Midland Plant, Units 1 and 2; RESAR 414; and Haven Nuclear Plant, Units 1 and 2.

From June 1973 to October 1975, I was employed as an engineer in the Power Division of Stone and Webster Engineering Corporation, Boston, Massachusetts. My responsibilities included subcompartment analysis to establish design criteria for both PWR and HTGR containments, preparing answers to AEC requests for additional information concerning containment spray systems and sump design and writing the appropriate SAR sections. In addition, I wrote technical specifications for field equipment purchases.

During the summer of 1972 I was employed as a technician for Nuclear Fuel Services in Rockville, Maryland. Here, I assisted in the development of computer codes which predicted the fuel depletion rates for both pressurized and boiling water reactors.

My academic training includes a Bachelor of Science in Nuclear Enginnering from the University of Virginia in 1973 followed by a Master of Mechanical Engineering from the Catholic University of America in 1978.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, et al.	)	Docket No. 50-289
	)	
(Three Mile Island, Unit 1)	)	

AFFIDAVIT OF CHARLES C. GRAVES

I, Charles C. Graves, being duly sworn, do depose and state:

1. I am a Principal Reactor Engineer in the Division of Systems Safety, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission.

2. The answer to UCS Interrogatory 133 was prepared by me. I certify that the answer given is true and accurate to the best of my knowledge.

Charles C. Graves
Charles C. Graves

Subscribed and sworn to before me
this 17 day of March 1980.

Notary Public

My Commission expires: July 1 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of)
METROPOLITAN EDISON COMPANY, et al.) Docket No. 50-289
(Three Mile Island, Unit 1))

AFFIDAVIT OF JACK ROE

I, Jack Roe, being duly sworn, do depose and state:

1. I am a Emergency Planning Analyst, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for reviewing the emergency planning of assigned nuclear power plants, including Three Mile Island, Unit 1 Restart Program.

2. The answers to Union of Concerned Scientists Interrogatories 163-166 and 168-189 were prepared by me. I certify that the answers given are true and accurate to the best of my knowledge.

Jack W Roe
Jack W. Roe

Subscribed and sworn to
before me this 24 day of
June, 1982.

Notary Public

My Commission expires: July 1, 1985

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, et al.	)	Docket No. 50-289
	)	
(Three Mile Island, Unit 1)	)	

AFFIDAVIT OF BYRON L. SIEGEL

I, Byron L. Siegel, being duly sworn, do depose and state:

1. I am a senior reactor engineer in the Division of Systems Safety, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for reviewing accident analyses and ECC and RHR systems of assigned nuclear plants.

2. The answers to Union of Concerned Scientists Interrogatories 192 and 193 were prepared by me. I certify that the answers given are true and accurate to the best of my knowledge.

Byron L. Siegel

Byron L. Siegel

Subscribed and sworn to
before me this 24 day of
March 1982

Notary Public

My Commission expires: July 1, 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, et al.	)	Docket No. 50-289
	)	
(Three Mile Island, Unit 1)	)	

AFFIDAVIT OF PAUL E. NORIAN

I, Paul E. Norian, do depose and say under oath as follows:

1. I am Section Leader of the Systems Analysis Section, Analysis Branch, Division of Systems Safety, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for supervising the review of reactor vendor transient and LOCA analysis methods, the improvement of NRC analysis methods used in related accident analyses, and the performance of staff audit calculations for transients and accidents.
2. The answer to UCS Interrogatory 191 was partially prepared by me. I certify that the answer given is true and accurate to the best of my knowledge.

Paul E. Norian
Paul E. Norian

Subscribed and sworn to before
me this 7 day of March 1980.

Notary Public

My Commission expires: 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY et.al.	)	Docket No. 50-289
	)	
(Three Mile Island, Unit 1)	)	

AFFADAVIT OF PAUL E. NORIAN

Paul E. Norian deposes and says under oath as follows:

1. I am Section Leader at the Systems Analysis Section, Analysis Branch, Division of Systems Safety. I am responsible for supervising the review of reactor vendor transient and LOCA analysis methods, the improvement of NRC analysis methods used in related accident analyses, and the performance of staff audit calculations for transients and accidents. My professional qualifications statement is attached.

2. The answers to the Union of Concerned Scientists' Interrogatories 75-78 were prepared by me. I hereby certify that the answers given by me are true and accurate to the best of my knowledge.

Paul E. Norian

Subscribed and sworn to before
me this 28 day of March, 1980

Notary Public

My Commission Expires: July 1, 1982

PAUL E. NORIAN

PROFESSIONAL QUALIFICATIONS

I am Section Leader of the Systems Analysis Section, Analysis Branch, Division of Systems Safety. I have held this position since 1975 and am responsible for supervising the review of reactor vendor transient and LOCA analysis methods, the improvement of NRC analysis methods used in related accident analyses, and the performance of staff audit calculations for transients and LOCAs. From June through December 1979, I was assigned to the Bulletins and Orders Task Force as a member of the Analysis Group. I served as Alternate Group Leader and coordinated the reviews of small break loss-of-coolant accidents (LOCA) and transient analyses submitted by the vendor owner's groups since the Three Mile Island accident.

I graduated from Lehigh University in June 1955 with a Bachelor of Science Degree in Engineering Physics. I also attended Drexel Institute of Technology, Catholic University of America, and the University of Maryland where I have taken various graduate courses in mathematics, physics, and electrical engineering.

In July 1955, I began work as a physicist with the duPont Company at the Savannah River Plant in Aiken, South Carolina. From that time until March 1962, I worked in the Works Technical Department on operational physics problems associated with the heavy water production reactors at Savannah River. This work included such assignments as the development of monitoring systems, performance of physics calculations required in reactor operation and in the development of new fuel elements, the review of operating procedures, and the analysis of various operating problems. In March 1962, I was transferred to the duPont Company's Chestnut Run Laboratories in Wilmington, Delaware, and worked for its Film Department on the development of industrial applications for plastic films.

In December 1963, I accepted a position with the Division of Reactor Licensing of the U.S. Atomic Energy Commission, and was project leader in the construction permit review of Consolidated Edison's Indian Point No. 2 reactor and Wisconsin-Michigan's Point Beach No. 1 reactor. I was assigned as a nuclear engineer in the Systems Performance Branch of the Division of Reactor Standards in March 1967. My responsibilities included analyzing and evaluating the performance of engineered safety systems and performing computer calculations for the evaluation of containment response and loss-of-coolant accidents. In March 1971, I participated in the Regulatory Task Force reappraisal of emergency core cooling systems for light water reactors. My main responsibility for the task force was the review of computer codes and input assumptions for LOCA analyses. In May 1973, I was assigned to the Core Performance Branch in the Directorate of Licensing. I served as Section Leader in the Thermal Hydraulics Section and supervised the review of portions of reactor vendor model changes to conform with the new requirements for LOCA models specified in Appendix K to 10 CFR Part 50.

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, et.al.	)	Docket No. 50-289
	)	
(Three Mile Island, Unit 1)	)	

AFFIDAVIT OF GARY M. HOLAHAN

I, Gary M. Holahan, being duly sworn, do depose and state:

1. I am a Senior Nuclear Engineer in the Division of Systems Safety, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for performing the safety review of assigned nuclear power plants, including Three Mile Island, Unit 1 Restart Program.

2. The answers to USC' Interrogatories 61, 62, 63 and 64 were prepared by me. I certify that the answers given are true and accurate to the best of my knowledge.

Gary M. Holahan

Gary M. Holahan

Subscribed and sworn to
before me this 31 day
of March 1980.

Notary Public

My Commission expires: July 1, 1982

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the matter of	)	
	)	
METROPOLITAN EDISON COMPANY, et al.	)	Docket No. 50-289
	)	
(Three Mile Island, Unit 1	)	

AFFIDAVIT OF EDGAR G. HEMMINGER

I, Edgar G. Hemminger, being duly sworn, do depose and state:

1. I am a Mechanical Engineer in the Division of Systems Safety, Mechanical Engineering Branch, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for review and evaluation of structural integrity, operability, and functional capability of safety related mechanical and electrical equipment.

2. The answers to UCS' Interrogatories 50, 51, 52 and 53 were prepared by me. I certify that the answers given are true and accurate to be best of my knowledge.

Edgar G. Hemminger

Subscribed and sworn to
before me this 31st day of
March, 1982

Notary Public

My Commission expires: July 1, 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, et al.	)	Docket No. 50-289
	)	
(Three Mile Island, Unit 1)	)	

AFFIDAVIT OF THOMAS G. DUNNING

I, Thomas G. Dunning, being duly sworn, do depose and state:

1. I am a Section Leader in the Instrumentation and Control Systems Branch in the Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for reviewing and evaluating safety analysis reports and topical reports in the area of nuclear power plant instrumentation and control systems.

2. The answers to UCS' Interrogatories 113 through 127 were prepared by me. I certify that the answers given are true and accurate to the best of my knowledge.

Thomas G. Dunning

Subscribed and sworn to
before me this 3rd day of
March 1980.

Notary Public

My Commission Expires: July 1, 1982