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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of

METROPOLITAN EDISON COMPANY

(Three Mile Island Nuclear Station, Unit No. 1)

Docket No. 50-289
(Restart)

SUPPLEMENT TO PETITION TO INTERVENE CONTAINING FINAL CONTENTIONS AND BASES SET FORTH WITH SPECIFICITY, STEVEN C. SHOLLY, PETITIONER

Pursuant to the Board's Memorandum and Order dated 21 September 1979 and the requirements of 10 CFR 2.714, Steven C. Sholly, Petitioner, files this supplement to the petition to intervene in the above-captioned proceeding. Said supplement contains Petitioner's contentions in final form together with bases for each contention set forth with reasonable specificity.

Due to the unavailability of the Licensee's completed response to the NRC Staff recommended requirements for restart of Unit 1 and due to the uncertain nature of certain NRC rules and regulations which are subject to revision, some of the Petitioner's contentions as set forth herein are not as specific as would be possible if the aforesaid information were available before the due date for final contentions in this proceeding. Petitioner requests leniency from the Board in this respect as these contentions were prepared without the aid of legal counsel. Petitioner will make

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every effort as information becomes available to provide more specificity with regards to contentions and their bases.

Pe tioner points out to the Board that a number of very important and significant investigations into the Unit 2 accident (3/28/79) are not yet completed, and that these investigations potentially have great bearing on this proceeding because the Licensee is the same in both cases and there is a high degree of similarity between the design and construction of Units 1 and 2 at Three Mile Island. Pending the availability of reports of said investigations and the passage of a reasonable amount of time for the Petitioner to reviewsaid reports, Petitioner will seek permission from the Board to amend the petition to intervene pursuant to 10 CFR 2.714 (a)(3), should any of these reports contain information which provides a bssis for a contention regarding the protection of public health and safety in this proceeding.

Definitions

Throughout these final contentions the following definitions shall apply:

a. "Licensee" refers to Metropolitan Edison Company;

b. "Unit 1" refers to the Three Mile Island Nuclear Station, Unit No. 1, while "Unit 2" refers to Unit No. 2 of that same facility;

c. "the 3/28/79 accident" refers to the loss of feedwater transient/loss of coolant accident which began at Unit 2 on March 28, 1979;

d. "NEPA" refers to the National Environmental Policy Act;

e. "GDC" refers to the specified General Design Criteria contained in Appendix A of 10 CFR 50;

f. "NRC" or "the Commission" refers to the U.S. Nuclear Regulatory Commission;

g. "NUREG" refers to the standard code used by the NRC to number certain publications, as for example, NUREG-0600.

Final Contentions

Contention # 1

It is contended that the Unit 1 containment isolation system does not meet the following requirements:

a. Conformance with the Standard Review Plan Section 6.2.4, "Containment Isolation System";

b. Compliance with GDC 16, Containment Design;

c. Compliance with GDC 50, Containment Design Basis;

d. Compliance with GDC 54, Piping Systems Penetrating Containment.

It is further contended that as a result of the design and construction of the Unit 1 containment and the containment isolation system, Unit 1 is rendered incapable of compliance with 10 CFR 20.105, 10 CFR 20.106, and Appendix I of 10 CFR 50, and that, therefore, there exists reasonable doubt that Unit 1 can be operated without endangering the health and safety of the public. Inasmuch as the Commission has the authority pursuant to 10 CFR 50.109 to require backfitting when such backfitting is required to provide substantial, additional protection of public health and safety, it is contended that compliance with SRP Section

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6.2.4, GDC 16, GDC 50, GDC 54, 10 CFR 20.105, 10 CFR 20.106, and Appendix I of 10 CFR 50 is required to protect the public health and safety, and that therefore backfitting of the Unit 1 containment and containment isolation system is a necessary precondition to permission to restart.

Basis for Contention # 1

The General Design Criteria of 10 CFR 50 establish design, fabrication, construction, testing, and performance requirements for components, structures, and systems in a nuclear power plant which exist to provide reasonable assurance that the plant can operate without undue risk to the health and safety of the public. Failure of a particular plant to meet any of the General Design Criteria is a very serious matter requiring corrective action to provide reasonable assurance that the plant can be operated without endangering the health and safety of the public.

The 3/28/79 accident at Unit 2 revealed a number of shortcomings of the containment and containment isolation system at Unit 2 and, because of similarities in design and construction, also at Unit 1. The failure of the containment to isolate on diverse signals, including high radiation in the containment and initiation of high-pressure injection from the emergency core cooling system, led to the release of millions of curies of radioactive noble gases into the environment, resulting in offsite doses accordi[illegible] to NUREG-0600 at pages II-3-16 through 18 and pages II-3-91 an[illegible] which were in excess of radiation protection standards contained in 20 CFR 20.105 and 20 CFR 20.106. These exposures which are cited in NUREG-0600 are also apparently in violation of the "as low as is reasonably achievable" requirements of 10 CFR 50, Appendix I. The radiation releases cited herein were the direct result of the failure of the containment to isolate in a timely fashion following initiation of high-pressure injection. A large quantity of highly radioactive water was pumped from the containment sump to the auxiliary building from which the radiation was subsequently released. This effectively resulted in the defeat of the containment as a safety system which is responsible for establishing a leak-tight barrier against the release of radioactivity to the environment in an uncontrolled manner.

The fact that such a large quantity of radiation was released into the environment as a direct result of the failure of the containment to isolate in a timely manner is a violation of GDC 16 which requires that the containment establish the leak-tight barrier to the uncontrolled release of radioactivity into the environment which is referred to above. GDC 16 further requires that the containment design conditions not be exceeded for as long as postulated accident conditions require. The same conditions which exist at Unit 2 with respect to GDC 16 also exist at Unit 1.

GDC 50 requires that the containment structure, including penetrations, be so designed so that the containment structure can accomodate the calculated pressure and temperature conditions resulting from any loss-of-coolant accident without exceeding the design leakage rate. Design to accomodate the pressure and temperature conditions must be with "sufficient margin." This margin, according to GDC 50, shall reflect, among other items, experience available for defining accident phenomena and containment response. Considerable experience in this area is now available as a result of the 3/28/79 accident at Unit 2 and must, under the provisions of GDC 50, be taken into account in evaluating the adequacy of design of containments. The radiation which was released to the environment during the Unit 2 accident is evidence that the design leakage rate was in fact exceeded, and that therefore the containment does not meet GDC 50 under loss-of-coolant accident (LOCA) conditions. By virtue of design and construction similarities with Unit 2, Unit 1 also suffers from this non-compliance with GDC 50.

GDC 54 provides that piping systems, including the containment sump piping, which penetrate the containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities reflecting the importance to safety of isolating these piping systems. The containment sump piping, by virtue of its function in conveying water from the containment sump pit to the auxiliary building, should be provided with the leak detection, isolation, and containment capabilities discussed in GDC 54. This is because in the event of an LOCA, radioactively contaminated water would be expected to collect in the containment sump as it is vented from the primary reactor cooling system. It is clear from the results of the 3/28/79 accident at Unit 2 that the containment sump piping is not properly provided as per GDC 54 with necessary leak detection, isolation, and containment capabilities, because it was through this pipe system that the radioactive water from the containment was sent to the auxiliary building from which the radioactivity subsequently escaped into the environment. By virtue of the design and construction similarities of Unit 2 to Unit 1, the same noncompliance with GDC 54 exists with Unit 1 as exists with Unit 2.

The radiation exposure and doses resultant from the releases from Unit 2 during the course of the 3/28/79 accident were specifically described in NUREG [illegible]00. At page II-3-18, NUREG-0600 cites releases from Unit 2 of noble gases which were at least eleven times the concentrations permited under the limits in 10 CFR 20.106.

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At pages II-3-91 and II-3-92, NUREG-0600 cites exposure and dose rates on Kohr Island for the period from 0700 hours on 3/28/79 to 1200 hours on 3/29/79 of 30 mrem/hr and 14 mrem/hr. These exposures are in excess of the 2 mrem/hr limit contained in 10 CFR 20.105. The exposures are also in excess of the Appendix I, 10 CFR 50, as low as is reasonably achievable limits wh'ch limits the annual air dose due to gamma radiation at any location near ground level which could be occupied by individuals at or beyond the boundary of the site to ten millirads. This limit was easily exceeded by the 30 mR/hr and 14 mR/hr readings taken on Kohr Island as cited in NUREG-0600 at pages II-3-91 and II-3-92.

Since the Commission has authority pursuant to 10 CFR 50.109 to require backfitting, and since violations of General Design Criteria directly relates to the ability of the plant to operate without endangering the health and safety of the public, the Petitioner seeks to require that backfitting of the Unit 1 containment and containment isolation system be made a precondition to restart.

Contention # 2

It is contended that the reactor coolant pressure boundary for Unit 1, as designed and constructed, does not meet the requirements of GDC 14, GDC 15, and GDC 30. It is further contended that because of noncompliance with GDC 14, GDC 15, and GDC 30, the operation of Unit 1 poses an

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undue risk to the health and safety of the public. Inasmuch as the Commission is empowered pursuant to 10 CFR 50.109 to require backfitting of Unit 1 to provide compliance with GDC 14, GDC 15, and GDC 30, and because compliance with the General Design Criteria is required to provide assurance that Unit 1 can operate without endangering the public health and safety, it is contended that the reactor coolant pressure boundary of Unit 1 must be backfitted to provide compliance with GDC 14, GDC 15, and GDC 30 prior to restart.

Basis for Contention # 2

The General Design Criteria of 10 CFR 50, Appendix A, establish design, fabrication, construction, testing, and performance requirements for components, structures, and systems in nuclear power plants which exist to provide reasonable assurance that the plant can be operated without undue risk to the public health and safety. Failure of a plant to meet any of the General Design Criteria is a very serious matter requiring corrective action to provide the required reasonable assurance that the plant can be operated without endangering the public health and safety.

The 3/28/79 accident at Unit 2 revealed a number of shortcomings with regard to the reactor coolant pressure boundary of Unit 2, and, because of similar design and construction, also of Unit 1's reactor coolant pressure boundary. GDC 14 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage.

The reactor coolant pressure boundaries for Units 1 and 2 contain power-operated relief valves (PORV's) in the pressurizer system. The failure of the pressurizer system PORV to reseat properly was, according to NUREG-0578 at page A-6 and A-7, a significant contributor to the sequence of events during the 3/28/79 accident and resulted in a direct violation of reactor coolant pressure boundary integrity. Further, NUREG-0578 at page A-7 lists a number of instances, including the failure of the pressurizer system PORV to reseat, in which relief and safety valves malfunctioned. NUREG-0578 comments at page A-7, "It is not clear whether these past instances of improper operation resulted from inadequate qualification of the valve or from a basic unreliability of the valve design." In either case, use of these valves in the reactor coolant pressure boundary at Unit 1 is a violation of GDC 14 regarding design and testing of the reactor coolant pressure boundary. Inasmuch as a short-term recommendation of the Lessons Learned Task Force was to have Licensees conduct testing to qualify valves, including PORV's, in the reactor coolant pressure boundary, it is obvious that such valves have not yet been appropriately tested under anticipated operating conditions, including design basis transients and accidents. This is a direct violation of the requirements of GDC 14.

GDC 15 requires that the reactor coolant pressure boundary systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. During the first eighteen minutes of the 3/28/79 accident at Unit 2, the design pressure for the system was exceeded in that the reactor coolant system pressure dropped below the safety injection actuation set-point of 1640 psig to 1100 psig (see NUREG-0600 at page I-2-15). Since the design and construction of Units 1 and 2 are very similar, Unit 1 suffers from the same problems under the same conditions; therefore, GDC 15 is violated because the design condition of the system relating to pressure is in fact exceeded during an anticipated operational occurrence (a loss of feedwater transient with reactor trip).

GDC 30 requires that components which are part of the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards practical. It has already been noted above that NUREG-0578 at pages A-7 and A-8 recommends a Licensee program to qualify valves, including PORV's which are in the Unit 1 reactor coolant pressure boundary, under expected operating conditions including design basis transients and accidents. This obviously means that such testing has not been performed for the PORV's in Unit 1's reactor coolant pressure boundary. This is a direct violation of GDC 30. Inasmuch as violations of the General Design Criteria relate directly to the ability of the plant (Unit 1) to operate without endangering the public health and safety, and since the Commission has authority in this situation pursuant to 10 CFR 50.109 to require backfitting, Petitioner seeks to require backfitting of the reactor coolant pressure boundary and/or appropriate qualification testing of components of that system as a precondition for restart.

Contention # 3

It is contended that as a result of Licensee's Operating Procedures, the emergency core cooling system can be defeated by operator actions during the course of a transient and/or accident at Unit 1, such defeat consisting of either throttling back the high-pressure injection pumps or tripping these pumps. It is further contended that under the conditions of a loss-of-feedwater transient/loss of coolant accident at Unit 1, defeat of the emergency core cooling system high-pressure injection system by pump throttling and/or pump trip results in significant cladding metal-water reaction, causing the production of amounts of hydrogen gas in excess of the amounts required by NRC regulations to be considered in the design and accident analysis of nuclear power plants. It is contended further that such production of hydrogen gas results in the high risk of breach of containment integrity due to the explosive combustion of the hydrogen gas in the containment. Inasmuch as the emergency core cooling system is an engineered safety feature which is relied upon to protect the public health and safety, and because proper operation of the emergency core cooling system is required to provide reasonable assurance that Unit 1 can be operated without endangering the public health and safety, it is contended that the emergency core cooling system operating procedures must be modified in order to ensure compliance with the GDC 35 requirement of negligible clad metal-water reaction following a loss-of-coolant accident (LOCA). It is further contended that the emergency core cooling system operating procedures must be appropriately modified prior to restart in order to provide for protection of the public health and safety.

Basis for Contention # 3

The emergency core cooling system is required for Unit 1 pursuant to 10 CFR 50.46. The emergency core cooling system is required by GDC 35 to prevent significant clad metal-water reaction during an LOCA. NUREG-0578, at page A-22, makes the following statement regarding the production of hydrogen during the 3/28/79 accident at Unit 2, "This amount of hydrogen generation was well in excess of the amount required by the Commission regulations as a design basis for any type of of post-accident combustible gas control system." The large production of hydrogen gas was reduced in seriousness at the Unit 2 3/28/79 accident by combustion at approximately 9 hours and 50 minutes after the initiation of the accident, resulting in a containment pressure spike of 28 psig (see NUREG-0600, at page IA-87). This combustion must be considered to have been fortuitous because it was not initiated on purpose by the operators of the plant. Absent this "lucky" occurrence, significant quantities of hydrogen gas would have collected in the containment, thus posing the risk of breach of containment integrity due to hydrogen explosion. Considering the lack of a hydrogen recombiner system at Unit 1 in a similar situation (which could occur given the similarity in design and construction of Units 1 and 2), presence of a dangerous amount of hydrogen gas in the containment requires the venting of the containment by plant operators (see NUREG-0578, at pages A-21 through A-25). Under accident conditions where there is a large amount of radioactivity in the containment, venting the containment would release this radioactivity to the environment. Had this procedure been necessary in the Unit 2 accident, off-site radiation doses would have been far in excess of those permitted under 10 CFR Part 20.

NUREG-0600 in the Operational Sequence of Events for the 3/28/79 accident (Appendix I-A of NUREG-0600) lists several instances in which plant operators, following operational procedures, either throttled or tripped the high-pressure

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injection system pumps; such events occurred at the following times after the reactor trip at 04:00:37 on 3/28/79:

a. 3 minutes, 13 seconds, operator throttles MU-V16 valves to pump 1C;

b. 4 minutes, 38 seconds, operator fully closes MU-V16 valves and trips pump 1C; pump 1A still running in throttled condition and is the only high-pressure injection (HPI) pump running at this point;

c. 5 minutes, 15 seconds, operator throttles pump 1A to minimum;

d. 10 minutes, 24 seconds, operator stops, restarts, and stops pump 1A within four seconds;

e. 11 minutes, 43 seconds, operator restarts pump 1A;

f. 93 minutes, operator running HPI flow at 150-200 gpm/loop (below maximum);

g. 3 hours, 20 minutes, operator manually starts pump 1C; pumps 1A and 1C now running with valves wide open;

h. 3 hours, 27 minutes, operator resets ECCS;

i. 3 hours, 37 minutes, operator trips pump 1C;

j. 3 hours, 56 minutes, pump 1C and containment isolation automatically initiated by high-pressure signal;

k. 4 hours, zero minutes, ECCS and isolation defeated by operator;

l. 4 hours, 17 minutes, operator secure pumps 1A and 1C, leaving to HPI pumps running;

m. 4 hours, 19 minutes, ECCS and isolation initiated automatically, but pump 1A is in pull-to-lock position and does not start;

n. 4 hours, 19 minutes, 18 seconds, operator defeats ECCS and isolation;

o. 4 hours, 22 minutes, operator restarts pump 1B;

p. 4 hours, 27 minutes, operator starts pump 1C; HPI at 250 gpm (below maximum) until 9 hours and 4 minutes;

q. 9 hours and 4 minutes, operator trips pump 1C

r. 9 hours, 50 minutes, 28-psig pressure spike causes ECCS and isolation initiated; pumps 1B and 1C now running (1A still locked out);

s. 9 hours, 51 minutes, operator trips pump 1C.

According to NUREG-0600 at page I-2-21, failure of the operators to maintain HPI flow according to design and procedural requirements resulted in apparent serious core damage and onsite and offsite exposure to radioactive materials. Failure to follow procedure 2202-1.3 in this regard is reported by NUREG-0600 at page I-2-21 to be under consideration as a potential item of noncompliance pursuant to Technical Specification 6.8.1.a. This situation demonstrates inadequate procedures on the part of the Licensee to ensure that operators adhere to Technical Specifications. This situation applies equally to Unit 1 due to design and construction similarities and procedural similarities between Units 1 and 2.

Because this situation involves an engineered safety feature of Unit 1 which is required for the protection of the public health and safety, and because it is beyond the capabilities of predictive sciences to determine that the ECCS will not be required if procedural changes are postponed until after restart, Petitioner contends that changes are necessary prior to restart to protect the public health and safety.

Contention # 4

It is contended that the ability of the Licensee to provide radiation exposure and dose data to responsible officials having decision-making responsibilities with respect to off-site radiation releases and emergency response to such releases is significantly impaired due to the lack on the part of the Licensee of on-site environmental TLD processing facilities. It is further contended that the Licensee is not prepared to implement Health Physics Procedure 1670.6, "Offsite Radiological Monitoring," due to the lack of on-site TLD processing capabilities. It is contended further that this lack of preparedness to implement Health Physics Procedure 1670.6 does not adequately protect the public health and safety under conditions of off-site release of radioactive materials, because this limits the Licensee's environmental TLD data to five off-site stations beyond five miles from the site, conditions which can permit plumes to fall in between TLD sites, thus severely limiting the ability of the Licensee to provide accurate and timely radiation exposure and dose estimates to responsible public officials. It is further contended that protection of public health and safety under accidental release of radioactivity conditions requires that the Licensee have on-site TLD processing capability prior to restart.

Basis for Contention # 4

The basis for this contention comes from NUREG-0600 at pages II-3-95 and II-3-96 where it is revealed that the Licensee is dependent on off-site processing of TLD's by Teledyne Isotopes and Radiation Management Corporation.

It is further revealed in NUREG-0600 at page II-3-96 that, by the admission of an employee of the Licensee, the Licensee is not prepared to implement Health Physics Procedure 1670.6 due to the lack of on-site TLD processing capability. Since the winds in the area of the Unit 1 site are variable in speed and direction, the existence of a well-defined plume in the area of an existing TLD site cannot be presumed. Therefore, the inability of the Licensee to implement the referenced Health Physics procedure effectively precludes accurate and timely radiation exposure and dose data from being available for use by officials in determining the need for emergency protective actions in the event of a release of radioactive materials to the off-site environment. This is clearly not in the interests of public health and safety, and the Petitioner therefore contends that on-site TLD processing capability is necessary prior to restart.

Contention # 5

It is contended that the Licensee has not provided sufficient radiation monitoring capacity in the containment, spaces which could contain LOCA fluids, effluent discharge paths, and the plant environs, as required by GDC 64. It is further contended that existing radiation monitoring in said locations does not provide for sufficient monitoring for radioactivity which may be released due to anticipated operational occurrences and due to postulated accidents, capacity for which is required in GDC 64. It is contended that lack of compliance with GDC 64 will prevent the Licensee from making accurate estimates of radioactivity releases from Unit 1 under conditions of anticipated operational occurrences and postulated accidents, and that this lack of compliance places the public health and safety at significant risk because such information is required by public officials to provide bases for decision-making related to emergency actions which may be required to protect the public health and safety. It is further contended that until the Licensee provides sufficient numbers and distribution of radiation release monitors which are capable of yielding on-scale results for all conditions of radiation release resulting from anticipated operational occurrences and postulated accidents, permission for the Licensee to restart Unit 1 should be denied.

Basis for Contention # 5

The basis for this comes from NUREG-0578 which discusses the subject of radiation monitoring capability. Under GDC 64, the Licensee is required to provide means for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents. According to NUREG-0578 at page A-37, "A recent survey of existing gaseous effluent monitoring capabilities of operating plants shows that less than 20 percent of operating plants have monitors that would have stayed on scale under the conditions of the TMI accident. It can also be shown, however, that the potential releases from postulated accidents may be several orders of magnitude higher than was encountered at TMI. Under such circumstances, none of the effluent monitors now in service at any operating plant would remain on scale." The clear implication of these statements from NUREG-0578 is that gaseous effluent monitoring at Unit 1 is incapable of on-scale readings during at least some postulated accident conditions. Since off-scale readings permit only lower-bound estimations to be made on the amount of radiation released, the inability of Unit 1 effluent monitors to provide on-scale results under certain accident conditions introduces uncertainty into a situation during which accurate information is a necessity, as was so vividly demonstrated during the 3/28/79 accident at Unit 2. Accurate radiation release information is required for determinations of emergency actions by public officials charged with responsibilities in this respect during accident conditions. Lack of reliable and accurate information during such a critical time clearly is not in the interest of public health and safety, and where releases exceed the capability of the monitor by orders of magnitude (as with the Unit 2 vent monitor which, according to NUREG-0578 at page A-39 was off-scale at 0.5 Ci/second, compared with estimated release rate calculated from NUREG-0600 at page II-3-17 for the 33-hour period beginning 0700 hours on 3/28, which was over 45 Ci/second) the health and safety of the public can be seriously jeopardized by underestimation of radiation release rates.

Contention # 6

It is contended that Unit 1's design contains the following features which make the plant unusually sensitive to off-normal transient conditions originating in the secondary system:

a. Design of the steam generators to operate with a relatively small liquid volume in the secondary side;

b. Lack of direct initiation of reactor trip upon the occurrence of off-normal conditions in the feedwater system;

c. Reliance on an integrated control system to automatically regulate feedwater flow;

d. Actuation before reactor trip of a pilot-operated relief valve on the primary system pressurizer;

e. Low steam generator elevation relative to the reactor vessel which provides a smaller driving head for natural circulation.

It is further contended that these features result in the placing of a greater burden on successful ECCS function and proper operator decisions and actions than was anticipated in design to deal with off-normal system behavior during anticipated transient conditions. It is contended that the total effect of these factors results in the lack of reasonable assurance that Unit 1 can be operated without endangering the public health and safety, and that until the design and operating procedures at Unit 1 are modified to provide such reasonable assurance, permission for restart must be denied.

It is further contended that the short-term actions identified in the Commissions Order and Notice of Hearing dated 9 August 1979 are insufficient to provide the requisite reasonable assurance of operation without endangering the public health and safety.

Basis for Contention # 6

There are a number of sources for the basis of this contention. The five features of the Unit 1 reactor which make it unusually sensitive to off-normal transient conditions originating in the secondary system were identified as factually existing in the Commission's Order and Notice of Hearing dated 9 August 1979 at page 3. The contention of the Petitioner that these features place a greater burden on ECCS and operator performance during such transients also comes from the same Order at page 3. Inasmuch as the five identified features relate to compliance with GDC 10, 14, 15, 30, 34, and 35 and since compliance with the General Design Criteria of 10 CFR 50, Appendix A, is required to provide reasonable assurance that Unit 1 can be operated without undue risk to the public health and safety, it is inconsistent for the Commission to permit restart until all items relating to compliance with the General Design Criteria are complete. The Commission proposes to permit restart (as per Order and Notice of Hearing dated 9 August 1979 at pages 7 and 8) before the Licensee is to submit a failure mode and effects analysis of the Integrated Control System. The ICS is used to regulate feedwater flow, which relates to compliance with GDC 34. If the Commission is unsure if the ICS will function properly, it would appear to be inviting non-compliance with its own regulations by permitting restart while compliance with GDC 34 is not assured.

Similarly, the Commission proposes to permit restart before the Licensee is required to comply with Category "B" recommendations of the NUREG-0578 report. These recommendations include the following items which relate to compliance with the General Design Criteria and/or to the five features identified as contributing to the Unit's sensitivity to off-normal transients:

a. Completion of instrumentation installation for detection of inadequate core cooling (relating to GDC 13, 15, and 10);

b. Completion of installation of hydrogen gas control penetrations of the containment (relating to GDC 16, 41, 50, 54, and 56);

c. Review of basis for recombiner use (relating to GDC 41);

d. Completion of installation of high-range effluent monitor system (relating to GDC 64).

Again, the Commission appears to be inviting non-compliance by proposing to permit restart prior to completion of these items. This is inconsistent with the requisite finding of "reasonable assurance" prior to restart.

Contention # 7

It is contended that the Licensee's analysis of loss-of-coolant (LOCA) accidents is imcomplete, that the Licensee is therefore not in compliance with the requirement of 10 CFR 50, Appendix K, requiring analysis of a spectrum of postulated LOCA's, and that to the extent such analyses are incomplete, operational procedures used by the Licensee to define operator action with respect to LOCA's are defective. It is further contended that without complete analysis of LOCA's there is a lack of reasonable assurance that Unit 1 can be operated without endangering the public health and safety. It is therefore contended that until LOCA analyses are complete and operational procedures relative to operator actions in dealing with LOCA's are revised and accurate, permission for restart must be denied to protect the public health and safety.

Basis for Contention # 7

The basis for this contention comes from a statement in the Commission's Order and Notice of Hearing dated 9 August 1979 at page 5, which states as a short-term action that the Licensee shall, "Complete analyses for potential small breaks and implement operating instructions to define operator action." This clearly implies that the LOCA analysis for Unit 1 are imcomplete, which appears to violate 10 CFR 50, Appendix K requirements. It also leaves open the possibility that a thus-far unanalyzed LOCA could occur for which there are no operating procedures. This is clearly not in the interests of public health and safety. There are many operational procedures used by the Licensee which could define operator actions with respect to LOCA's which could be defective as a result of incomplete LOCA analysis. It is difficult to specify these procedures without benefit of discovery to obtain copies of the Licensee's operating procedures, but a preliminary review of the "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station, Unit 1" gives the following as a list of probable defective operational procedures:

a. Group 2 Procedures 1303-11.8 (High and Low Pressure Injection), 1303-11.9 (R.B.Emergency Cooling System), 1303-8.1 (Reactor Coolant System Test), 1302-5.10 (R.B. 4 PSIG Channels), 1302-5.11 (R.B. 30 PSIG Channels), 1303-5.4 (Emergency Feedwater Pumps), 1303-1.2 (R.C. Flow Surveillance), 1203-28 (Post Accident H_2 Purge), 1104-55 (Reactor Bldg. Atmosphere Cleanup System), 1101-1 (Plant Limits and Precautions), 1101-2 (Plant Setpoints), 1102-14 (Reactor Building Purging and Venting), 1104-1 (Core Flood System), and 1104-5 (Reactor Building Spray System);

b. Group 1 Procedures AB 1203-15 (Loss of Reactor Coolant Makeup), EP 1202-37 (Cooldown from Outside Control Room), EP 1202-26A (Loss of Feedwater to OTSG), OP 1102-4 (Power Operation), EP 1202-12 (Excessive Radiation Levels), EP 1202-35 (Loss of Decay Heat Removal), EP 1202-14 (Loss of Flow), EP 1202-6 (Loss of RC Coolant), SP 1303-5.4 (EFW Pumps), AP 1004 (Emergency Plan and Procedures), and 1106-3 (Feedwater System).

This list is not meant to be definitive but is points out procedures which might require revision upon completion of LOCA analysis in order to adequately protect public health and safety. Failure of the Commission to require revision of appropriate procedures with respect to LOCA's risks tragic consequences, and therefore Petitioner contends that the procedures must be revised prior to restart.

Contention # 8

It is contended that the Licensee's emergency plan for Unit 1 is defective because it does not provide sufficiently for the protection of public health and safety under all conditions of operation, including anticipated operational occurrences and postulated accidents. It is further contended that the proposal by the Commission to extend the Licensee's emergency action capabilities out to a distance of ten miles from the site does not adequately address the protection of public health and safety under accident conditions, especially when highly variable factors such as changing weather conditions, strikes affecting transportation facilities, the presence of significant numbers of extra persons in the area during certain time of year, and the nature of the geography and transportation routes in the surrounding area are considered. It is contended that the Licensee's emergency plan does not contain sufficient detail to permit the finding of reasonable assurance that <u>appropriate</u> <u>measures</u> <u>can</u> <u>and</u> <u>will</u> be taken in the event of an emergency to protect the public health and safety and prevent damage to property, as required by Appendix E of 10 CFR Part 50. It is further contended that the Licensee's emergency plan does not provide for means for determining the magnitude of the release of radioactive materials, including sufficient criteria for determining the need for notification of local, state and federal agencies, and criteria for determining when protective measures should be considered within and outside the site boundary to protect public health and safety and prevent damage to property, which are also required under Appendix E of 10 CFR 50. It is also contended that the emergency procedures of the Licensee do not assure prompt notification of the public and local officials of the need for public evacuation or other protective measures which may be necessary in the event of an emergency at Unit 1, assurance of which is required under Appendix E of 10 CFR 50. It is contended that until the Licensee's emergency plans are suitably revised and sufficiently detailed to assure that appropriate measures can and will be taken in the event of an emergency to protect the public health and safety and prevent damage to property, permission for restart must be denied.

Basis for Contention # 8

There are several aspects to this contention dealing with the adequacy of emergency planning by the Licensee. The Commission proposed that the Licensee be required to extend the capability to take appropriate emergency actions for the population around the site to a distance of ten miles (Order and Notice of Hearing, 9 August 1979, at page 8). The Commission placed this requirement in the context of permitting a finding of reasonable assurance of safety in the long-term operation of Unit 1. Inasmuch as there is no assurance that the emergency plan would not be needed in the short-term, and since the Commission obviously finds the Licensee's plan lacking in certain respects (see Order and Notice of Hearing, 9 August 1979, at page 6 and 8), for the Commission to permit restart in the absence of workable and sufficient emergency plan would be an act of irresponsibility and would appear to violate the Commission's own regulations (Appendix E of 10 CFR 50). It is not acceptable for the Licensee to be permitted to restart Unit 1 without a workable and sufficient emergency plan. Further, the Petitioner finds numerous additional problems with the Licensee's emergency plan in addition to those addressed in the Commission's Order. Appendix E of 10 CFR 50 requires demonstration of reasonable assurance that the Licensee's plans provide that the Licensee not only <u>can</u> but in fact <u>will</u> take appropriate emergency actions to protect public health and safety and prevent property damage in the event of an emergency at Unit 1. The performance of the Licensee during the 3/28/79 accident at Unit 2 certainly throws considerable doubt on whether the Licensee <u>will</u> in fact act by taking the necessary emergency steps required. In the light of the Licensee's actions during the Unit 2 accident, the Petitioner believes that it should be imcumbent upon the Licensee to demonstrate beyond reasonable doubt that it not only has the capability to take emergency actions when they are needed, but that it also will in fact take those actions in a timely manner. In the Unit 2 incident, the incident began at 0400 hours. By 0600, the Licensee clearly new the reactor trip was not normal (NUREG-0600, page IA-38 and 39). Both alarm printer and utility printer were out of service from 0513 to at least 0648--no alarm data was printed during this time period. At 0656, a site emergency was declared. The

Pennsylvania Emergency Management Agency was not notified for six additional minutes (until 0702). By this time the auxiliary building had already been evacuated and the containment dome monitor was reading 600 R/hr. A general emergency was not called until 0724. The fact that the lead state emergency agency was not contacted about the incident until three hours after its start is evidence that the Licensee was not at that time prepared to promptly notify local, state, and federal agencies in time of emergency. A minimal step would have been to put the state on alert when the first radiation monitors began registering alarm set-points or when it was realized that the trip had not been normal, yet the Licensee did nothing until there had been a formal declaration of site emergency.

NUREG-0600 at page II-2-7 evaluates the emergency conditions relative to the Licensee's existing plan at the time of the 3/28/79 accident at Unit 2. The investigators from the Office of Inspection and Enforcement found that a site emergency should have been called at 0430 hours on 3/28 rather than at 0656 because a condition of site emergency involving loss of primary coolant pressure coincident with either high reactor building pressure or high reactor building sump level was met by 0430. Upon a projection of a 40 R/hr dose rate for Goldsboro, a general emergency should have been called at 0710; instead, the calculated dose was discounted even though there was not actual data available for evaluating the correctness of the calculated radiation exposure rate. The Petitioner, as a result of these factors and a well-documented lack of forthrightness in releasing information in the early days of the 3/28/79 accident on the part of the Licensee, believes that there is ample evidence to suggest that the Licensee will not act to take appropriate emergency measures on a timely basis, and that therefore the Licensee should be required to demonstrate that this attitude will not continue in the future should Unit 1 be permitted to restart.

The Licensee is required by various Technical Specifications, NRC regulations, and Operating Procedures to make numerous radiation surveys in the event of an accidental release of radiation. Despite the fact that a site emergency was declared at 0654 and a general emergency was declared at 0724 on 3/28/79 during the Unit 2 accident, the first environmental radiation survey was not performed until 0748 (see NUREG-0600 at page II-3-77). In addition, during the period from 0700 on 3/28 and 1600 on 3/29, when 6.6 million curies of noble gases were released from the plant, there were two time periods of more than five hours and two hours during which stable plume conditions were known to have existed and during which the Licensee failed to conduct radiation surveys. A third survey made in Goldsboro revealed radiation levels of 20 mR/hr gamma and 30 mR/hr beta/gamma, in excess of permitted levels. It is clear that the Licensee does not provide the means for determining the magnitude of a radiation release in the emergency plan, means for which are required under Appendix E of 10 CFR 50.

Emergency actions under the Licensee's plans are not divided into what can be done under differing weather conditions. Heavy snowfalls, fog, heavy rain, and temperature inversions are all meteorological phenomena which can and do occur near the site of Unit 1, and all of these can drastically impact on the success of emergency actions and the time frame within which such emergency actions must be executed. During certain times of the year (for instance, during Farm Show week in Harrisburg), there are many more persons in the area than would normally be the case; this is not addressed sufficiently in the Licensee's emergency plans. There are drastic differences in terrain and the availability of transportation routes out of the area depending upon which compass direction a radiation plume may be headed; this can impact significantly on decisions regarding emergency actions and the time frame within which such actions must be taken, yet this is not addressed in the Licensee's emergency plans.

Licensee's emergency plans are seriously defective, requiring far more in the way of revision than the largely cosmetic revisions suggested as sufficient by the Commission. Such revisions must be accomplished prior to restart to protect the public health and safety.

Contention # 9

It is contended that the Licensee's environmental radiation monitoring program contains an insufficient number of monitoring sites and an inadequate distribution of monitoring sites within twenty miles of the Unit 1 site to provide sufficient protection of the public health and safety. It is further contended that there is in the Licensee's environmental radiation monitoring program an unwarranted reliance on the use of thermoluminescent dosimeters (TLD's) for providing information used to calculate radiation exposure data. and that this unwarranted reliance on TLD's seriously underestimates radiation doses to the public. It is also contended that the Licensee does not possess adequate portable radiation monitors to provide additional information in the event of an offsite radiation release, and that the Licensee does not exercise adequate administrative control over the maintenance of these units, nor the training of personnel in their use. It is contended that the radiation monitoring program of the Licensee must be greatly upgraded prior to restart to ensure adequate protection of the public health and safety .

Basis for Contention # 9

It is clear from correspondence between the Licensee and the Commission and from NUREG-0600 (at pages II-3-92 and II-3-95) that there are only five TLD locations outside a five-mile radius from the Unit 1 site and only 15 TLD locations within the five-mile radius at off-site locations. Plumes from off-site radiation releases could easily fall in between such a limited unmber of radiation monitoring sites. In fact, during the first 68 hours of the 3/28/79 accident at Unit 2, when most of the radioactivity was released, the wind blew in a given sector for several hours at a time only 30% of that time period; during the major part of the 68-hours period, the plume was either not well defined or tended to meander, causing actual exposure rates at a given point to fluctuate considerably (see NUREG-0600 at page II-3-95). Even though the Licensee's own procedure (Health Physics Procedure 1670.6) required the deployment of additional TLD's to assist in compiling accurate dose estimates, the Licensee failed to accomplish this because the Licensee was not able to implement the procedure (see NUREG-0600 at pages II-3-95 and 96).

There are other examples of inadequate management administrative control over radiation monitoring involved with the Unit 2 accident. NUREG-0600 at page II-3-97 lists two time periods totalling 12.5 hours (time periods when significant amounts of radiation were being released) when inadequate assessments of radiological conditions were made. In addition, NUREG-0600 goes on to say that the original radiation survey records were discarded, leaving only Emergency Control Station records from results which were radioed in. The ECS log does not contain complete records of the specific instruments used in the surveys, the mode in which the instruments were used, the orientation of the instrument with respect to the source being measured, and the duration of the actual measurement.

In addition, NUREG-0600 lists in page II-F-4 as a potential item of noncompliance the fact that eight environmental air samplers used in the Licensee's radiation monitoring program had not been calibrated for a period beginning in 1974 up to March 1979, in violation of the Environmental Technical Specification 5.7 and Surveillance Procedure 1302-5.24.

In addition, actions of the plant personnel as evaluated by NUREG-0600 at pages II-3-68 and 69 revealed that the employees of the Licensee, including chemistry and radiation protection staffs, did not have adequate comprehension of:

a. The methods of evaluating internal and skin doses of radiation;
b. The limitations of portable survey instruments, including range, energy response, and the significance of open and closed window readings;
c. The plant systems and components;
d. Concepts and techniques used to minimize radiation doses;
e. Regulations, license conditions, and plant procedures;
f. The emergency organization.

It is clear from a reading of NUREG-0600 that the Licensee did not exercise proper managment administrative controls over the radiation monitoring program, and therfore the Petitinner seeks to require proof of such proper controls prior to restart.

Contention # 10

It is contended that until a method for decontaminating and restoring Unit 2 has received NRC approval and the environmental impact of that method and the impact of that method on the waste handling and storage capacity on Unit 1 have been evaluated, to proceed with restart would place an unnecessary and unreasonable risk on the public health and safety. It is further contended that until the Licensee can provide reasonable assurance that it can safely operate Unit 1 while decontaminating and restoring Unit 2, restart must be postponed.

Basis for Contention # 10

This contention is based on GDC 5 which prohibits the sharing of structures, systems, and components important to safety among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. The petitioner desires the Licensee to make a showing of insignificant impairment with respect to Units 1 and 2, considering accidents at either unit. Assuming that the Licensee plans to decontaminate Unit 2, Petitioner feels that the Licensee must be able to show that an accident at Unit 1 will not so affect activities at Unit 2 as to pose a significant risk to public health and safety.

Restart of Unit 1 without knowing how the activities at Unit 2 will impact on Unit 1 places an unnecessary risk on the public health and safety because if Unit 1 waste handling and storage capacity is used to assist in the Unit 2 activities and an accident occurs at Unit 1 which requires the use of Unit 1's waste handling and storage capabilities, it is unclear what would occur. It is possible that an uncontrolled release of liquid wastes would result in such a situation, and that could easily result in significant risk to public health and safety, especially via the drinking water pathway. It is therefore necessary that the impact of Unit 2 activities be known in advance to the extent possible prior to the restart of Unit 1.

Contention # 11

It is contended that the production of hydrogen in the reactor core from clad metal-water reactions following an LOCA poses an unacceptably high risk of catastrophic failure of the reactor pressure vessel and the reactor containment, with the subsequent release of a substantial portion of the core inventory into the environment. It is further contended that until a safe and reliable means for eliminating hydrogen gas from the containment is installed at Unit 1, and is provided with suitable redundancy as required by GDC 41, restart of Unit 1 poses a risk to public health and safety and must be denied.

Basis for Contention # 11

There was, according to NUREG-0578, the production of amounts of hydrogen in the 3/28/79 accident at Unit 2 which are above the amount required to be considered in design bases under Commission regulations. A 28-psig pressure spike in the containment at Unit 2 during the accident caused by hydrogen combustion. According to NUREG-0578 at page A-21, had there not been provisions for recombiners at Unit 2, it would have been conceivable that containment atmosphere venting would have been required to control hydrogen gas concentrations. Such a venting, in view of the grave amounts of radiation in the containment, would have resulted in off-site doses far in excess of anything permitted under NRC regulations. It is clear that in the case of Unit 1, which currently has not hydrogen recombiner capability, venting of containment atmosphere following an LOCA of similar magnitude to the accident at Unit 2 would also result in greatly excessive radiation doses at off-site locations. Due to the large population at risk and the large amount of radioactivity involved in such an instance, venting is clearly not in the interests of public health and safety. In view of the fact, according to NUREG-0578 at page A-22, that hydrogen recombiner systems are not very costly, and pursuant to backfitting authority of the Commission under 10 CFR 50.109, a requirement for installation of a hydrogen gas control system at Unit 1 prior to restart would be a prudent measure which is well in line with established policy regarding defense-in-depth and protection of public health and safety.

Collection of high concentrations of hydrogen gas in the containment at Unit 1 following a postulate LOCA under current conditions would present the choice between venting a large amount of radiation to the environment or risking the chance of catastrophic pressure vessel/containment failure in the event of a massive hydrogen explosion. Providing hydrogen control systems at Unit 1 will eliminate this choice.

Contention # 12

It is contended that regardless of the substance of the Final Order in this proceeding, the decision of the Board and the Commission will, in view of the extraordinary circumstances of this proceeding, constitute a major federal action which could significantly affect the quality of the human environment, and that as such, the decision of the Board and the Commission in this case falls under the requirements of Section 102 of NEPA. It is further contended that the environmental impact statement required under Section 102 of NEPA must be issued before the NRC's decision can take effect in this case. It is further contended that as a result of the similarities in design and construction of Units 1 and 2, the results of the 3/28/79 accident at Unit 2 have rendered invalid major sections of the Final Environmental Impact Statement on Three Mile Island Unit 1 (NUREG-0552), and that such sections as have been rendered invalid must be addressed in the new environmental impact statement which is required to be prepared. It is also contended that the psychological impact of Unit 1 restart must be evaluated in the NEPA statement required under Section 102 for this proceeding.

Basis for Contention # 12

Petitioner has thus far been unable to find any other instance of a hearing conducted for the sole purpose of determining whether a Licensee should be permitted to restart a nuclear power plant following the suspension of the Licensee's operating license. In view of the unprecedented nature of these proceedings, and in view of the unprecedented accident at Unit 2, a unit with similar design and construction and run by the same Licensee, Petitioner finds that an environmental impact statement is clearly called for under Section 102 of NEPA which requires such a statement for any "major Federal action."

The accident at Unit 2 has substantially rendered invalid major portions of the FES for Units 1 and 2, NUREG-0552. The sections rendered invalid are as follows:

a. Section III, D, 2, which covers radioactive wastes released during operation and the solid wastes expected to generated during operations;
b. Section V, which covers the environmental impact of plant operations, especially part D, covering the radiological impact of routine operations;
c. Section VI, which covers the environmental impact of postulated accidents, especially as it discusses Class 9 accidents (Unit 2 accident on 3/28/79 was classified as Class 9 by the Commission), and LOCA's;
d. Section XI, which covers the cost-benefit ratio, which should be recalculated to include the costs associated with the Unit 2 accident and cleanup and the costs of modifying Unit 1 for restart.

Evaluation of psychological impacts under NEPA is required, inasmuch as Paragraph B of Section 102 of NEPA requires all federal agencies to develop methods and procedures to insure the evaluation in decisionmaking of "presently unquantified environmental amenities and values," along with economic and technical issues. The lack of environmentally-caused stress is certainly an environmental amenity and is therefore subject to these provisions.

NEPA does not list the specific impacts to be considered in the environmental impact review process but rather aims to include a wide variety of concerns which are not limited to simply economic and technical concerns. Indeed, Paragraph A of Section 102 of NEPA mandates the use of a "systematic, interdisciplinary approach which will insure the integrated use of the natural and social sciences" in decisionmaking which may have an impact on the environment. The reference to social sciences could reasonably include the evaluation of psychological stress caused by the proposed restart. Petitioner contends therefore that psychological stress associated with Unit 1 restart is subject to NEPA review, even if only in qualitative form, and that such stress must be balanced against any potential benefits of the proposed action.

Contention # 13

It is contended that the Unit 1 computer system does not meet the requirements for instrumentation and control specified in GDC 13, and is inadequate to insure proper operation of the Unit 1 reactor under all conditions of normal operation, including anticipated operational occurrences and postulated accident conditions. It is further contended that the lack of real-time printout capability during accident conditions and the lack of sufficient redundancy in the computer

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system place the public health and safety at significant risk during accident conditions, especially if computer function is lost and no backup unit is available. It is contended that until the Unit 1 computer system is upgraded to meet the standards of GDC 13 and until suitable redundancy is provided within the computer system to assure real-time printout capability at all times, permission for restart must be denied on the basis of risk to public health and safety due to inadequate availability of operational information to Unit 1 operators.

Basis for Contention # 13

The requirements of GDC 13 for adequate instrumentation to provide for monitoring variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure safety, are certainly applicable to the Unit 1 computer system. The computer system is a crucial tool used by the plant operators to maintain control over the reactor and is used also to provide rapid availability of information for the operators. The records of the computer are also used in the analysis of transients and accidents by investigating teams within and outside the Licensee's organization and the NRC.

Computer generated records are typed on the Alarm Printer, the Utility Typer, and the Log Typer, and are inked on Analog Trend Recorders. For the period covered by the 3/28/79 Unit 2 accident, several information gaps exist, partially because of equipment failures and partially because the records in question were either lost, discarded, or purposely destroyed. Alarm Typer and Utility Typer out put were lost due to equipment problems and due to probable operator action to actuate the alarm suppress function at 0648 hours on the 28th to print out current data and wipe out historical information. Further problems with the Alarm Typer and Utility Typer were recorded during the period 1848 to 1910 hours when the typers jammed with paper. Investigators from the Office Of Inspection and Enforcement concluded that the information lost was not critical to understanding the accident, but the Petitioner feels that this conclusion is at best speculative due to the unprecedented nature of the accident. Utility Typer output from 0000 to 0324 and from 2008 to 2012 was lost (3/28/79) and were not found by investigators. In addition, the strip chart from Analog Trend Recorder # 2 was never found for the day of the 28th of March. Petitioner finds this lack of records appalling and believes it conclusively demonstrates the inadequacy of the Unit 2 computer system, and, due to similarities, also the Unit 1 computer system. Had redundancy been available, such computer generated records would not have been lost.

Printers were running behind real time early in the accident. According to NUREG-0600 at page I-4-79, by 0646 hours on the 28th of March, the Utility Typer was running one hour and 33 minutes behind real time; by 1315 hours, the

Alarm Typer was running two hours and 39 minutes behind (had the printer memory not been wiped out at 0648 hours, the lag would have been over four hours because the lag is cumulative). Lack of real-time data presented problems to operators in analyzing what was happending during the course of the Unit 2 accident. Such conditions would also cause problems with the Unit 1 operation. Lack of available information could easily have caused operator error, and thus risked the public health and safety.

Contention # 14

It is contended that the Licensee has negligently violated NRC regulations and technical specifications and that such violations place the safety of the public and the protection of the public health in question. It is contended further that the performance of the Licensee during the Unit 2 accident in terms of violations of regulations and technical specifications, and in terms of timely execution of safety-related functions, is directly applicable to Unit 1 since such violations call into question the management and administrative capabilities of the Licensee. It is further contended that until the Licensee can conclusively demonstrate that it possesses the necessary managerial and administrative capabilities required to operate Unit 1 in compliance with all applicable rules and regulations while, at the same time, properly and safely decontaminate and restore Unit 2, permission for restart of Unit 1 must be denied.

Basis for Contention # 14

The past record of violations of NRC rules and regulations and technical specifications by the Licensee demonstrates a lack of managerial and administrative control. The proposed items of noncompliance due to the Unit 2 accident, contained in Appendix IB and Appendix IIF of NUREG-0600, also call such capabilities into question.

In addition, the NUREG-0600 report also points out numerous examples of actions on the part of the Licensee which, while not constituting violations, collectively demonstrate a lack of managerial and administrative control, especially with respect to plant operating procedures.

NUREG-0600 is replete with examples of Licensee staff actions which demonstrate inadequate knowledge of operating procedures and basic radiation safety concepts. Loss of crucial records and lack of maintenance of other records which are required by NRC regulations to be maintained is further evidence for lack of effective managerial and administrative controls.

Specific examples (these are examples, not a definitive listing) of items which demonstrate lack of sufficient managerial and administrative capabilities on the part of the Licensee include the following, all taken from NUREG-0600 at the indicated page(s):

a. Failure of staff to comprehend methods used to evaluate radiation doses and the concepts and techniques used to minimize radiation dose (II-3-68 and 69);
b. Inadequate supply of lapel air samplers, extremity monitors, wide-range beta-gamma survey meters, high-range pocket dosimeters, and alarming personnel dosimeters (II-3-68);
c. Loss of computer printer and analog trend recorder records (I-4-45 through 47 and I-4-78 through 81);
d. Potential item of noncompliance, in which eight environmental air samplers had not been calibrated in nearly five years (II-F-4);
e. Potential item of noncompliance, improper change to Health Physics Procedure 1670.7 made without required review and approval (II-F-3);
f. Potential item of noncompliance, personnel assigned to emergency teams did not receive proper training during 1978 (II-F-2 and 3);
g. Failure to declare site emergency upon detection of condition for such declaration (II-F-5 and 6);
h. Failure to maintain records of 500 surveys of off-site areas for radiation levels (II-F-6 and 7);
i. Failure to conduct radiation survey in well-established plume for over five hours (II-3-84);
j. Failure to make every reasonable effort to maintain occupational exposures as low as was reasonably achievable (II-3-70 and 71);
k. Failure to keep three independent steam generator emergency feedwater pumps and flowpaths operable during Power Operations due to improper surveillance tests procedure (IB-2).

Contention # 15

It is contended that the design of the Unit 1 Control Room, instrumentation, and controls is such that operators cannot maintain system variables and systems within prescribed operating ranges during feedwater transients and LOCA's. It is further contended that this violates the provisions of GDC 13 regarding instrumentation and controls. It is contended that in view of the numerous operating difficulties encountered with Unit 2, and the similarities in design and construction between

Units 1 and 2, a thorough human factors enginering review of Unit 1's Control Room is called for in order to provide assurance that the operator-instrumentation interface is such that the operators can exercise adequate control over the reactor and prevent off-site consequences from anticipated operational occurrences and postulated accidents. It is further contended that in order to assure maximum protection for the public health and safety, the human factors engineering reivew and any necessary changes recommended as a result of this review must be completed prior to restart.

Basis for Contention # 15

It is clear from a close reading of NUREG-0600 that there are numerous examples of control room design inadequacies which impacted on the ability of the operators to control the sequence of events during the Unit 2 3/28/79 accident. A few examples of this are:

a. At 18 minutes into the event, the fuel handling exhaust monitors showed a ramp increase in iodine readings, but the location of the instrument on the lower part of the vertical back panel prevented the operator standing at the front panel from viewing the trend (NUREG-0600, page IA-24);
b. Lack of positive indication of valve closure led to permitting PORV on Unit 2 pressurizer to remain open, thus causing loss of reactor coolant (NUREG-0578, pages A-9 and 10);
c. Lack of instrumentation for detecting inadequate core cooling which provided positive indication of such lack led indirectly to core damage during LOCA (NUREG-0578, page A-11 and 12).

In fact, NUREG-0578 at page 7 states, "A widely accepted lesson learned from the TMI-2 accident is that the man-machine interface in some reactor control rooms needs significant improvement."

A study performed by the Electric Power Research Institute (EPRI Report # NP309) detailed human factors engineering problems with nuclear power plant control rooms. From publicly available pictures of Unit 1's control room and Unit 2's control room, it is obvious that many of the concerns discussed in the EPRI Report are present at Three Mile Island Unit 1 and that these concerns should be addressed in a human factors engineering review of the Unit 1 control room. Inasmuch as the ability of the operators to control the reactor and safety systems has a great deal to do with the protection of public health and safety, with specific reference to the requirements of GDC 13 and the general philosophy of the General Design Criteria, these reviews and necessary modifications as a result of the reviews should be accomplished prior to restart.

Respectfully submitted,

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Dated: 22 October 1979

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

CERTIFICATE OF SERVICE

I hereby certify that single copies of "SUPPLEMENT TO PETITION TO INTERVENE CONTAINING FINAL CONTENTIONS AND BASES SET FORTH WITH SPECIFICITY, STEVEN C. SHOLLY, PETITIONER" have been served upon the following, either by deposit in the United States mail, postage prepaid, on the 22nd of October 1979, or by hand at the meeting of intervenor groups in Lancaster, Pennsylvania, on the 21st of October 1979:

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