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Systems to be leak tested will include but may not be limited to:

1. Makeup and Purification (including RCS letdown)

2. Decay Heat Removal System

3. Waste Gas System

Consideration will be given to the type of potential leak paths identified by NRC dated October 17, 1979 concerning the North Anna event.

A summary description of the leakage reduction program will be provided by January 1, 1980 and the program shall be implemented before restart.

2.1.1.9 Automatic Closure of the Pressurizer PORV Block Valve.

2.1.1.9.1 System Description

A modification will be installed to automatically close the Pressurizer PORV Block Valve (RC-V2) on low Reactor Coolant System pressure. Its purpose is to prevent an excessive loss of reactor coolant inventory if the PORV (RC-RV2) fails to close after opening on a high pressure excursion. RC-V2 will close automatically if the Reactor Coolant System pressure is below 1600 psig and the PORV is open. An open PORV will be detected by flow in the discharge line from the valve. The automatic closure signal will be bypassed when the PORV mode selector switch is in the NDT protection position.

2.1.1.9.2 Design Bases

The automatic closure of RC-V2 is designed to prevent excessive loss of reactor coolant inventory if the PORV fails to close after opening on a high pressure signal. The modification will be designed so as not to degrade the existing protective functions of the PORV. It will also not prevent the operator from manually relieving through the PORV when the plant procedures call for him to do so. Means will be provided to retain the low temperature NDT protective function when the plant is shutdown. The design will preclude automatic cycling of the block valve.

2.1.1.9.3 System Design

RC-V2 will be automatically closed if the Reactor Coolant System pressure goes below 1600 psig and the PORV is open. The low pressure signal will be derived from the ES Actuation System.

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The "PORV open" signal will be derived from the PORV Flow Detector System. (See Section 2.1.1.2) the block valve will go fully closed when a close signal is generated. There will be no automatic open signal to the valve. An alarm will be actuated when RC-V2 leaves its fully open position. The operator will be able to open the valve manually after the close signal has reset.

The control circuits shall be supplied from on-site power of the same power train as the AC supply to RC-V2. The automatic closure signal shall be bypassed when the PORV mode selector switch is in the NDT Protection position.

2.1.1.9.4. Design Evaluation

The design will provide a reliable means of preventing excessive reactor coolant inventory loss due to a malfunction of the pressurizer PORV. No operator intervention is required. It will not degrade any of the protective or operational functions now provided by the PORV. The control circuit will be supplied from on-site power sources. Any new components required will be specified to be suitable for the environment in which they will be located. No new components or wiring will be required inside containment.

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APPENDIX 7A

THREE MILE ISLAND NUCLEAR STATION

UNIT 1

RADIATION PROTECTION PLAN

REVISION 1

Revision 1
January 16, 1980

THREE MILE ISLAND NUCLEAR STATION

UNIT 1

RADIATION PROTECTION PLAN

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Changes to this document require approval by these positions.

Revision 1
January 16, 1980

THREE MILE ISLAND NUCLEAR STATION

UNIT 1

RADIATION PROTECTION PLAN

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Article 1 - Foundation for the TMI-1 Radiological Controls Program

This document, the Three Mile Island Unit 1 Radiation Protection Plan, sets forth the philosophies, basic policies and objectives of Metropolitan Edison Company and General Public Utilities Corporation concerning their TMI-1 Radiological Controls Program. The objective of the radiological controls program is to control radiation hazards to avoid accidental radiation exposures, to maintain exposures within the regulatory requirements, and also to maintain exposures to workers and the general population as low as is reasonably achievable. These philosophies, policies, and objectives are based on and stem from the regulations of the Nuclear Regulatory Commission (NRC) as contained in Title 10 of the Code of Federal Regulations, Parts 19, 20, 50, and 71, and appropriate Regulatory Guides, specifically 8.8 Rev. 3 (1978), 8.10 Rev. 1-R (1975), 8.13 Rev. 1 (1975), and 8.15 (1976). The TMI-1 Radiation Protection Plan is based on these references, therefore they are not repeated throughout the remainder of this document.

Specific details as to how the TMI-1 Radiation Protection Plan is implemented shall be promulgated in the TMI-1 Radiological Controls Procedures Manual (RCPM) and shall include those applicable procedures addressed in Reg. Guide 1.33 Rev. 2(1978), App. A, paragraph 7, and paragraph 8(aa), (bb); further references to the TMI-1 RCPM are not repeated throughout this document. The TMI-1 RCPM will consist of revisions to procedures which existed in the previous HPP 1600 and 1700 series, applicable Administrative procedures, and additional procedures deemed necessary. This TMI-1 Radiation Protection Plan is the first part of the TMI-1 RCPM. Requirements governing release of radioactive liquids and gases to the environment and the disposal of

solid radioactive waste are not addressed in this TMI-1 Radiation Protection Plan, but are addressed in the Environmental Technical Specifications.

Verbatim compliance with the TMI-1 RCPM is mandatory. In the event a procedure cannot be followed exactly, work under that procedure shall be stopped and shall not commence again until the procedure has been corrected.

This TMI-1 Radiation Protection Plan and the new TMI-1 RCPM are being written primarily to increase the effectiveness of the Radiological Controls Program at TMI Unit 1. Thus, the procedures have direct applicability only to TMI Unit 1. Procedures shall provide adequate guidance and specify appropriate methods or techniques to insure that the performance of each activity is in accordance with sound radiological control principles, and is in compliance with applicable regulatory provisions. The RCPM shall be prepared, reviewed, approved, and controlled as described in the RCPM Administrative procedures.

The TMI-1 Radiological Controls Program is to be fully integrated into each and every phase of operations at TMI Unit 1. The TMI-1 Radiological Controls Program when carried out as specified will assure that the operation of Unit 1 will be performed with personnel who work at the site incurring radiation exposure as low as can reasonably be achieved.

In order to meet this objective, the program must be carried out by each person involved in the TMI-1 activities. There is no group or person involved in the TMI-1 operations who does not have some degree of responsibility for the Radiological Controls Program. Failure of any person to recognize this responsibility or to comply with issued procedures will not be tolerated. A radiologically safe operation will be achieved if each individual carries out his or her responsibility.

The performance of each manager and supervisor must demonstrate support for the commitment by top management of General Public Utilities Corporation and Metropolitan Edison Company to a strong, effective radiological controls program.

Article 2 - Responsibilities of Workers

Although personnel specially trained in radiological controls normally oversee radioactive work, each individual involved in this work must constantly remain aware of the potential radiological problems. Each individual is responsible for maintaining his or her exposure as low as reasonably achievable. Each individual's actions directly affect his exposure, contamination, and overall radiological problems associated with the work. The following rules shall be followed by individuals to minimize radiological problems:

1. Obey promptly "stop-work" and "evacuate" orders of radiological control personnel.
2. Obey posted, oral, and written radiological control instructions and procedures, including instructions on Radiation Work Permits.
3. Wear TLD and self reading dosimeter where required by signs or by radiological control personnel. Report loss or unexpected exposure and offscale dosimeter to Radiological Control Department.
4. Keep track of personal radiation exposure status and avoid exceeding exposure limits.
5. Remain in as low a radiation area as practicable to accomplish work.
6. Do not loiter in radiation areas.
7. Do not smoke, eat, or chew in contaminated areas.
8. Wear anticontamination clothing and respiratory protection properly and wherever required by signs or radiological control personnel.
9. Remove anticontamination clothing and respiratory protection properly to minimize spread of contamination.

10. Frisk or be frisked for contamination when leaving a contaminated area or a radiological control point. Notify Radiological Controls personnel if contamination is found.
11. For a known or possible radioactive spill, minimize its spread and notify radiological control personnel promptly.
12. Do not unnecessarily touch a contaminated surface or allow clothing, tools, or other equipment to do so.
13. Place contaminated tools, equipment and solid waste on disposable surfaces (for example, sheet plastic) when not in use and inside plastic bags when work is finished.
14. Limit the amount of material that has to be decontaminated or disposed of as radioactive waste.
15. Notify Radiological Controls personnel of faulty or alarming radiation protection equipment.
16. Report the presence of open wounds to radiological control and medical personnel prior to work in areas where radioactive contamination exists and immediately if a wound occurs while in such an area.
17. Notify Radiological Controls personnel upon returning to the site after medical administration of radiopharmaceuticals.
18. Assure a mentally alert and physically sound condition for performing assigned work.
19. Ensure that your activities do not create radiological problems for others and be alert for the possibilities that the activities of others may change the radiological conditions to which you are exposed.

Article 3 - Audits, Reviews and Reports on the TMI-1 Radiological Controls Program

As indicated in Article 2, each individual is responsible for maintaining his or her radiation exposure as low as reasonable achievable while completing the scope of work they are required to perform. Each will be required to comply with the applicable procedures of the TMI-1 RCPM and the specific radiological controls prescribed for work in which they are engaged.

In order to ensure that these requirements are being met and to assist all site personnel in understanding and complying with these requirements, the following audit and review procedures shall be used:

1. Radiological control technicians shall monitor and aid the performance of each individual insofar as radiological work practices are concerned.
2. The Radiological Engineering shall review on a regular basis the performance of the radiological control technicians. This review includes shift coverage on those jobs which are considered likely to have a high potential for radiological difficulties.
3. Radiological assessments shall be conducted throughout the Radiological Controls Program on a continuous basis. This assessment function shall report directly to the highest level of management in the TMI-1 organization and shall be outside the Radiological Controls Department. A written report of the findings of this assessment shall be prepared and issued every month.

4. Quality Assurance audits shall be conducted of the TMI-1 Radiological Controls Program by technically qualified persons from outside the Radiological Controls Department. These audits will be conducted in accordance with procedures as outlined in the TMI-1 Quality Assurance Plan. The Quality Assurance Methods, Operations and Audit group will schedule these audits and will provide personnel from their own department and/or outside contractors as appropriate to conduct the audits. These audits shall cover the applicable portions of the Quality Assurance Plan, the TMI-1 Radiation Protection Plan and all procedures in the TMI-1 RCPM on at least an annual basis.
5. The Plant Operations Review Committee shall review and comment on the TMI-1 Radiation Protection Plan and any changes thereto. They also have the responsibility to review those procedures requested by the Manager-Radiological Controls.
6. Periodically, the services of an outside consultant will be retained to provide evaluation and guidance on ways to improve the TMI-1 Radiological Controls Program.
7. In addition to these reviews and audits, a system shall be employed to identify radiological control deficiencies. A radiological control deficiency is defined as either a violation of an established procedure or a practice which could and should be improved. Such deficiencies are recorded in a Radiological Deficiency Report. This system shall be specified in the TMI-1 RCPM embodying the following concepts. A Radiological Deficiency Report may be initiated by any individual who

observes a deviation from good radiological practices. These reports shall be evaluated by Radiological Engineering for desirable or necessary corrective action. The purpose of this system is to identify all deficiencies, regardless of how small or inconsequential, correction of which will result in an inproved Radiological Controls Program. Radiological Engineering shall prepare a monthly report summarizing the Radiological Deficiency Report findings and corrective action taken.

8. The Nuclear Regulatory Commission (NRC) also inspects and reviews the TMI-1 Radiological Controls Program. The TMI-1 Radiation Protection Plan and any changes thereto shall be submitted to the NRC for comment.

9. In the event all the preceding measures fail to prevent a radiological incident, and investigation shall be conducted to determine the causes of the incident and to determine the corrective actions and improvements needed.

Article 4 - Radiological Controls Training

1. Periodic radiological control training shall be given to ensure each person understands the radiological conditions to which he is exposed, understands his responsibility to minimize his own exposure to radiation, and understand his own responsibilities for complying with radiological control procedures. Personnel occupationally exposed to radiation shall receive instruction on the effects of radiation and the risks associated with radiation exposure.
2. General radiological indoctrination shall be given to those not directly involved with radiation so that they understand not to enter areas requiring TLDs and not to cross radiation barriers. The indoctrination shall include explanation of the radiological environment in which they work.
3. Radiological control training shall be given to personnel requiring access to a restricted area. These personnel shall be required to pass a written examination, and they shall requalify by written examination at least annually.
4. In addition to the training and written examinations of paragraph 3, those who require access to areas controlled by Radiation Work Permits shall receive more extensive training and shall be required to pass a radiological examination on their practical abilities, including use of dosimetry, frisking, anticontamination clothing, respirators, and response to unusual situations. Retraining, and both written and practical examinations shall be conducted at least annually. In addition,

spot checks shall be made that they retain the required knowledge during the period between examinations. Special briefings and extra training including use of mockups where applicable, shall be conducted for work involving higher than usual exposures to radiation and radioactivity.

5. Radiological control technicians and their foremen shall receive theoretical and practical training and training for unusual situations. Training shall also be given for changes to procedures, equipment and programs. They shall pass both written and oral examinations, in which the passing grade for foreman shall be higher than the passing grade for technicians. Periodic practical drills and oral drills shall be required for each technician and foreman. Annual requalification shall be required including both written and oral examinations. Radiological control technician assistants shall perform specific functions under direction of a qualified technician or foreman and only after being qualified for the specific function.

Article 5 - Control of External Exposure

Control of radiation exposure is based on the assumption that any exposure no matter how small involves some risk; however, exposure within the accepted limits represents a risk small compared with normal hazards of life. Therefore the policy of Metropolitan Edison Company and General Public Utilities Corporation is to maintain exposures to individuals and total man-rems as low as is reasonably achievable (ALARA). Line management from all departments as well as each individual worker shall take an active role in radiation exposure reduction.

To aid in exposure reduction, administrative radiation exposure control levels shall be established. Radiation man-rem exposure goals shall be established for each major job and for each year. Work involving radiation exposure shall be preplanned. Major exposure jobs shall require that radiological controls be incorporated in the design, that written procedures by prepared, and that pre-job briefing and rehearsals be conducted prior to commencing work. A Radiation Work Permit will be required for any work or entry to restricted areas that would involve or create any of the following: (a) high radiation area, (b) airborne radioactivity area, and (c) contaminated area, or (d) those radiation areas specified in applicable procedures.

Restricted areas used to control personnel access to radiation and radioactive materials shall be defined, access controlled, and posted in accordance with 10 CFR 20.203 with the following modifications:

1. Each High Radiation Area shall be barricaded and conspicuously posted as a High Radiation Area, and personnel desiring entrance shall obtain a Radiation Work Permit (RWP). Any individual entering a High Radiation Area sha.. (a) use a dose

rate monitoring device or (b) use a radiation dose rate integrating device which alarms at a preset dose level, or (c) assure that a radiological control technician provides periodic radiation surveillance with a dose rate monitoring instrument.

2. Any area accessible to personnel where a major portion of the body could receive in any one hour a dose in excess of one thousand mrem, shall be locked to prevent unauthorized entry. The keys to these locked barricades shall be maintained under the administrative control of the Radiological Controls Foreman on duty in accordance with the RCPM.

Radiological Controls personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties providing they are following radiological control procedures for entry into High Radiation Areas.

To evaluate radiological conditions, radiation surveys shall be conducted for air activity, removable surface contamination and external radiation at regular intervals. Surveys are performed in order to (a) monitor the suitability of control measures, (b) evaluate the needs for additional controls, (c) evaluate trends for ALARA purposes, and (d) evaluate radiological conditions in areas routinely entered without radiation work permit coverage. Surveys in unrestricted areas are provided to insure the effective control of radioactive material. Unusual conditions detected in the performance of either a routine or special survey shall immediately be brought to the attention of Radiological Controls Management. Portable radiation survey instruments will be

calibrated semiannually, except for dose rate measuring instruments, which will be calibrated quarterly, to assure a consistent, reliable and predictable response to radiation levels. Records of surveys shall be maintained on file. An Administrative program will be used to verify the calibration of personnel and field monitoring instruments.

Article 6 - Control of Internal Exposure

The policy of Metropolitan Edison Company and General Public Utilities Corporation is not to have any significant internal exposure to personnel from radioactivity associated with Three Mile Island Unit 1. For personnel exposed to radioactivity during their work, this means that no one should receive from internal radioactivity more than one tenth of their permitted annual radiation exposure.

Controls in other parts of this TMI-1 Radiation Protection Plan to minimize internal radioactivity, such as control of surface contamination and control of wounds, are not repeated in this article. The following controls are to minimize internal exposure from airborne radioactivity:

1. Engineering controls and controls on personnel access shall be applied to the maximum extent practicable so that radioactive work does not increase the amounts of airborne radioactivity inhaled. When no other controls are practicable, respirators shall be used. Those who may need to use respirators shall be medically qualified, trained, tested for respirator efficiency, and requalified in this respirator program at least annually.
2. Airborne radioactivity shall be measured regularly in areas where personnel may be exposed. Continuous monitoring representative of air the person is breathing shall be performed to supplement periodic measurements during work which has the potential to cause a worker to receive measurable internal radioactivity.

Internal radioactivity shall be measured at least annually in each person who works in an area requiring a radiation work permit; this

includes each person who wears respiratory protection. Internal radioactivity shall be measured promptly in each person who receives radioactive contamination on his skin, and in each person who is suspected of inhaling sufficient radioactivity to cause measurable internal radioactivity. Each measurement of internal radioactivity above a level near background shall be reviewed to determine the cause and to assist in minimizing internal exposures.

Article 7 - Control of Radioactive Contamination

Radioactive surface contamination shall be controlled in order to minimize possible inhalation or ingestion of radioactivity and to minimize build up of radioactivity in the environment. Measures to contain radioactivity and to minimize the number and extent of areas contaminated shall be taken in order to minimize personnel radiation exposure, to simplify subsequent personnel and area or facility decontamination, and to minimize the need to rely on anticontamination clothing.

The surface contamination limits for beta-gamma activity shall be 1000 dpm/100cm^2 for loose contamination and 0.1 mR/hr for total contamination. The preferable measurement technique is with a pancake frisker. 1000 dpm is equivalent to 100 cpm on the pancake frisker. For alpha activity the surface contamination limit is 100 dpm/100cm^2.

Emphasis in planning, training and working shall be placed on minimizing the numbers of occurrences and amounts of radioactivity involved in occurrences of radioactive surface contamination of a person's skin or of areas not controlled for radioactive surface contamination. Each such occurrence shall be reviewed in detail to determine how to correct deficiencies and improve control of radioactivity.

Article 8 - Control of Radioactive Materials

In addition to the definition of 10CFR20, any material having a dose rate measured with a beta gamma survey meter at 1 inch exceeding 0.1 mR/hr or with surface contamination in excess of the limits specified in Article 7 shall be handled as radioactive. A radioactive material control system shall be established to ensure radioactive material is not lost or misplaced in a location where personnel could unknowingly be exposed to radiation and to prevent the uncontrolled spread of radioactivity to areas where the public might be affected. This system shall include the following requirements:

1. The number of areas in which radioactive materials are stored shall be minimized.
2. Any new radioactive material storage area shall be approved before use by the Manager-Radiological Controls.
3. The numbers of radioactive items and the amount of radioactivity in storage shall be minimized.
4. Radioactive items shall be identified as radioactive before removing them from a restricted area.
5. Radioactive materials removed from the Protected Security Area or removed from a restricted area outside the Protected Security Area shall be controlled in accordance with an accountability procedure which ensures the materials are not lost or improperly handled during transfer or subject to unauthorized removal. This accountability procedure shall require inventory of radioactive materials which remain outside such areas.
6. Each incoming or outgoing shipment of radioactive material Shall be handled in strict compliance with detailed written procedures.

Each case in which radioactive material is lost or unaccounted for shall be reviewed in detail to determine the potential radiation exposure personnel might unknowingly receive, to correct deficiencies, and to improve control of radioactive materials.

Article 9 - Organization for Radiological Controls

A radiological control program cannot be strong and effective if left solely to the Radiological Controls Department. Each worker and supervisor has responsibility for radiological control; consequently, the organization for the entire Three Mile Island Unit 1 represents the organization for radiological control.

However, the Manager-Radiological Controls is responsible for ensuring that a high quality radiological controls program is established and maintained. It is the responsibility of the Radiological Control Department to evaluate radiological conditions and recommend precautionary measures. To assist the Manager-Radiological Controls, a Radiological Controls Department is organized as shown in Figure 1.

At times when demands upon the Radiological Control Department are sufficiently heavy to require a temporary increase in staff, qualified contractor personnel will be used. These personnel will be fully integrated into the department under the direction of the Manager-Radiological Controls.

Qualifications for the key radiological managers in NRC Regulatory Guide 1.8, Rev. 1-R, (1975) will be met as far as practicable. Where the corbination of strong manager and experience in radiological controls cannot practicably be obtained in the same person, credit for extensive nuclear power plant operations, maintenance, or engineering will be given.

One portion of the TMI-1 radiological controls program is the ALARA program for personnel radiation exposures to be as low as reasonably achievable. To accomplish this each engineer involved with TMI-1 has to have radiological engineering as part of his assignment. Thus, most radiological engineering functions are performed in engineering groups

rather than in the Radiological Controls Department. The overall coordination of the TMI-1 ALARA program, however, is assigned to Radiological Engineering in the Radiological Controls Department.

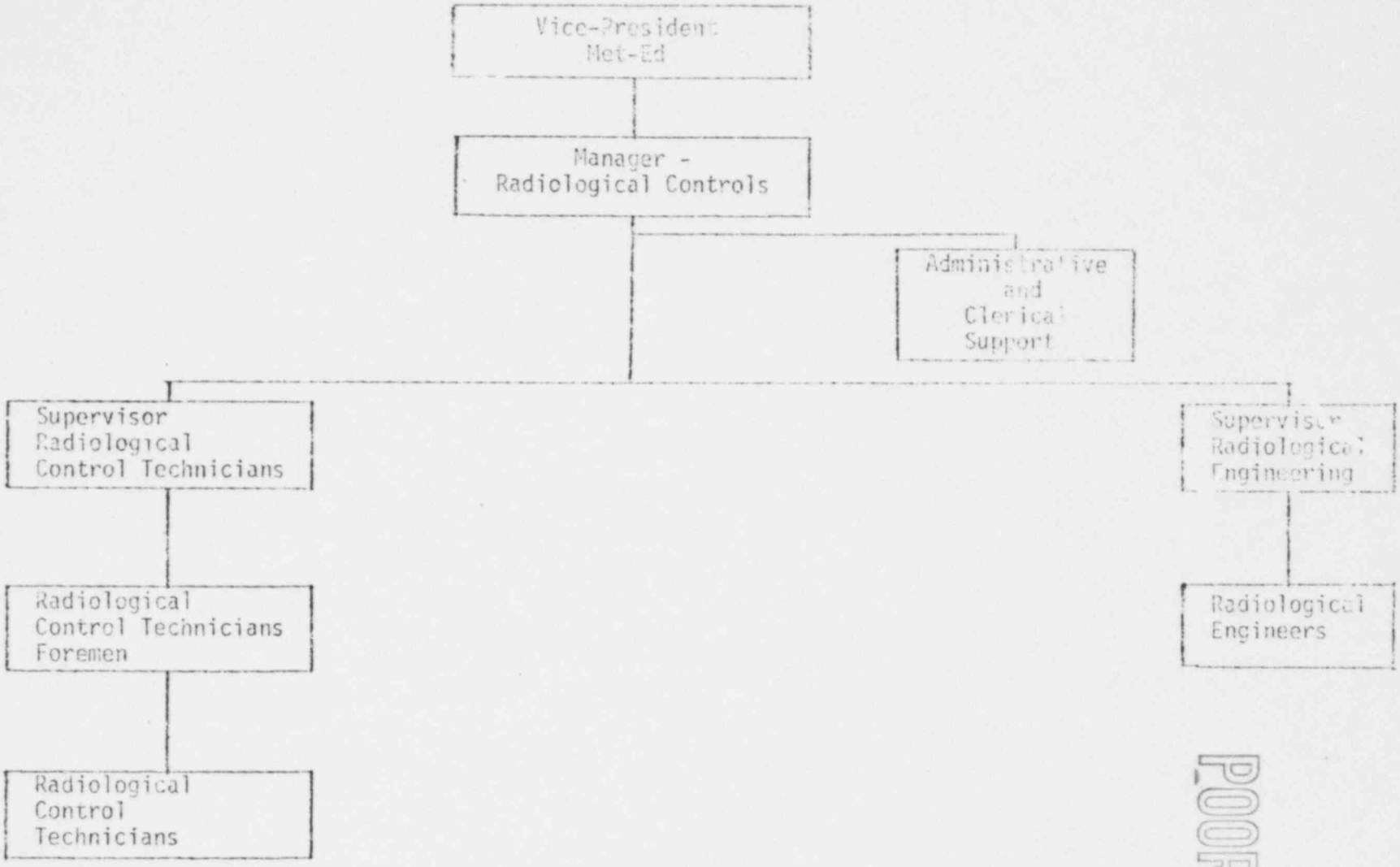

Figure 1 TMI-I Radiological Controls Department Organization

Revised 1
December 15, 1980

APPENDIX 7B

REPORT ON THE STATUS OF THE CORRECTIVE ACTION

IN RESPONSE TO AUDIT BY NUS CORPORATION

Report on the Status of Recommendations of NUS Audit Report of March 20, 1979

The following is the status as of December 21, 1979 on each of the recommendations of the NUS Audit Report of March 20, 1979. Each recommendation is identified and reference is made to the specific paragraph and page number from the NUS Audit Report.

1. "... the pool of Health Physics/Chemistry Technicians be divided into separate groups for the two disciplines." (Ref: Section 2.1, page 2-2)

Status: Met-Ed intends to propose a split in job classification during the spring of 1980. The change would result in separate Radiation Protection Technicians and Chemistry Technicians. On January 1, 1980, Met-Ed has temporarily split the group by specifically assigning individuals to each of the disciplines until the contract negotiations are finalized. Additionally, since July of 1979, Technicians have been rotated between the disciplines on a twelve week cycle, resulting in an increase in consistency in job performance and awareness of specific changes or anticipated operations within each discipline. The twelve week rotation results in each individual beginning his cycle during one of his training weeks, thereby allowing a refresher program to insure awareness of current procedures, maintenance and operations functions and plant conditions.

The recommendation has been incorporated to the extent possible and will, dependent on contract negotiations, be fully implemented.

2. "Designate one of the three Chemistry Foremen as a Chemistry Supervisor." (Ref. Section 2.1, Page 2-3)

Status: Reorganization since the Unit-2 incident has changed the reporting requirements of the chemistry supervisors to others, outside the Health Physics organization. A separate chemistry organization has been developed for each Unit. This has resulted in the Health Physics Department being able to focus strictly on radiological concerns.

The recommendation has been incorporated in the reorganization.

3. "Re-arrange the Radwaste function so that the two foremen do not report directly to... (Supervisor Radiation Protection and Chemistry)".

Status: The Radwaste function is now the responsibility of the Operations Department. A Radwaste Supervisor has been appointed who reports directly to the Supervisor of Operations, Unit-1.

The recommendation has been incorporated in the reorganization.

4. "A crew of personnel, such as Utility Workers, should be permanently assigned to Health Physics for the specific purpose of tool, equipment and respirator decontamination." (Ref: Section 2.2, page 2-4)

Status: A crew of Utility Workers will be assigned permanently to assist in radwaste and Health Physics functions. These individuals are not as yet in place, but will be available as the open Utility Worker positions are

filled through hiring of new employees. Their function will not be limited to decontamination of tools and equipment. At present, a specific group of contractor personnel have been assigned to decontaminate tools and equipment A second group is responsible to decontaminate respirators in accordance with a respirator cleaning procedure.

The long-term solution to this recommendation has not been resolved.

5. "There is immediate necessity for two additional full-time clerical personnel in the Radiation Protection/Chemistry Department." (Ref: Section 2.3, page 2-6)

Status: The reorganization of the Unit-1 Radiological Controls Department since the Unit-2 incident has resulted in the addition of an Administrator and two clerical personnel in addition to the one existing clerical position.

The recommendation has been incorporated into the reorganization.

6. "One individual should be assigned full-time responsibility for the radiation dosimetry program." (Ref: Section 2.4, page 2-6)

Status: A full-time dosimetrist has been hired and in-place since July of 1979. He has developed a staff whose full-time function is the issuing, collection, processing of dosimeters and dosimetry record control.

Action has been implemented in accordance with the recommendation.

7. "Adjustments to the Technician staffing should include the provision to re-implement the training shift on a continuing basis." (Ref: Section 3.1, page 3-2)
"Consideration should be given to including the Health Physics Foremen in selected phases of the continuing re-training program." (Ref: Section 3.1, page 3-2)

Status: A revised training program is in the final stages of development and will be implemented in January of 1980. The program text is complete and is designed to instruct both Technicians and Foremen. The training will be incorporated into the six week shift rotation schedule resulting in one week of training each six weeks for each technician and foreman. The contents of the program will include academic training in Health Physics principles, practical training including a review of site procedures, "hands on" laboratory and field sessions, and unusual events. Additional Technicians are being added such that the training program can be conducted without direct impact on daily activities. Instructors in Health Physics are being added to the Training Department to insure that the burden of preparing and presenting the Health Physics program will fall on those supervisors having daily in-plant responsibilities.

Action will be completed in accordance with the recommendation.

8. "Personnel who are responsible for initiating RWP's should be clearly appraised of the necessity to properly describe their intended scope of work. All personnel must also understand that no changes in work scope which may invalidate the adequacy of RWP protective specifications shall be made without prior Health Physics notification." (Ref: Section 4.2, page 4-2)

Status: Training programs have placed emphasis on the need for complete description of work scope on RWP's such that the Radiological Controls personnel may provide adequate protective specifications. The technicians have been instructed to insure that they fully understand the work scope prior to approving the RWP's. The initiator is contacted when insufficient information is provided. To assure that the radiological communications via the RWP is effective, verbation compliance with the procedures is required and a Health Physics Audit Program, including audits from a Radiological Controls Inspector, the Radiological Engineering Group and contracted Health Physics Group, continuously monitors the work and the compliance with good Health Physics Practices.

The recommendation has been incorporated in the Training Program, Audit Program and Health Physics Procedures.

9. "An immediate goal at TMI must be to insure that the Health Physics Technicians and Foremen are adequately trained. Once a reasonable degree of competence achieved, the decisions of these individuals must be given full support." (Ref: Section 3.2, page 3-3)

Status: The training program is discussed in item #7. To insure that decisions related to radiological control practices are based on complete information and in accordance and consistent with Health Physics Procedures and not reversed by other department supervisors, Radiological Control Technician Foremen have been assigned to shifts. The result has been a more direct involvement in off-shift operations and maintenance and a more complete awareness of all factors involved in the radiation protection aspects thereby allowing for greater consistency in applying radiation protection practices.

The recommendation has been incorporated by means of improved training programs and shift Radiation Protection Technician Foremen.

10. "A concerated effort should be made by all parties to improve the day-to-day communications among all the members of the Health Physics organization." (Ref: Section 4.3, page 4-3)

Status: Communications is recognized as the major factor in an effective radiological control program. The reorganization has resulted in a specific radiological control group without additional responsibilities such as chemistry and radioactive waste. This change has resulted in more direct communications within Health Physics. Additionally, a formalized shift turnover followed by a shift briefing has been implemented with shift briefings being held to insure proper awareness of plant operations. Shift turnovers include a communication with operations Shift Foreman.

The Unit planning meetings, (i.e. Plan of the Day Meetings, Outage Planning Meetings are attended by the Manager-Radiation Protection or his designee, followed by a department staff meeting with the intent of improving overall communications. Appropriate information is presented to technicians during shift briefings by foremen.

This recommendation has been incorporated into daily operations.

"A method should be implemented to ensure that all technicians are aware of each TCN and procedure revision." (Ref: Section 4.4, page 4-3)

Status: A document review sheet system has been implemented which requires that technicians review, each procedure change and sign the review sheet.

The recommendation has been incorporated by means of the document review sheet.

11. "Consideration should be given to making a complete re-evaluation of the TLD calibration method." (Ref: Section 5.1, page 5-2)

Status: Cnanges to the TLD calibration system are in place which reflect the use of a SR-90 Calibration as well as periodic quality assurance checks similar to those in effect before the Unit-2 incident. The creation of a dosimetry group as described in item #6 of this document has resulted in consistency of calibrations.

The recommendation has been incorporated into existing procedures and the TLD calibration program is considered to be adequate.

12. "...a full-time individual with a job classification such as Technical Analyst should be assigned to perform surveillance of the overall radiation dosimetry program." (Ref: Section 5.2, page 5-3)

    "To the extent which is achievable, all TLD's should be processed and read by the same technician, ..." (Ref: Section 5.3, page 5-3)

Status: A full-time dosimetrist has been hired and in-place since July of 1979. He has developed a staff whose full-time function is the issuing, collection, processing of dosimeters and dosimetry record control.

Action has been implemented in accordance with the recommendation.

13. "The continuous air monitoring system should be supplemented with procurement and use of portable CAMs." (Ref: Section 6.1, page 6-2)

    "A total of five or six portable CAMs should be provided at TMI. These CAMs should be of the moving filter type and should have the capability to monitor both particulate and radioiodine activity." (Ref: Section 6.1, page 6-2)

Status: Approximately twelve portable air monitors (Eberline AMS-3) are available and in use at TMI-1. These monitors are of the fixed filter type which monitor for particulate activity. Portable monitors for radioiodine activity have been evaluated. Due to the inherent problems with field monitoring for radioiodine, portable monitors have not been purchased. However sampling and lab analysis have been upgraded and procedures are in effect which define a complete air monitoring program for both particulate radioiodine, and gaseous activity. The upgraded procedures impose strict requirements for knowing the air activity prior to and during work functions.

The recommendation has been incorporated through the purchasing of new equipment and upgrading of the air sampling procedural requirements.

14. "The schedules for obtaining and analyzing air samples should be re-evaluated." (Ref: Section 6.2, page 6-2)

Status: The air sampling program has been re-evaluated and procedures upgraded. Presently, requirements exist for air sample results to be available within twelve (12) hours prior to beginning work in any area with potential for airborne activity. Continuous sampling is required for work in areas with air concentrations greater than 10% of MPC.

The recommendation has been completed and steps taken to upgrade procedures.

15. "...the schedules for performing radiation and contamination surveys should be re-evaluated." (Ref: Section 6.3, page 6-3)

Status: Procedures which establish the survey requirements prior to work commencing in any area have been reviewed and upgraded. The result has been an increase in the number of areas surveyed for specific RWP's as well as increased frequencies of routine surveys for the areas routinely entered for maintenance or operations.

The recommendation has been completed and procedures upgraded as required.

16. "Surveying and releasing of tools and equipment from the decon area should be performed by Health Physics Technicians. Procedures should provide guidance, techniques, criteria and limits for surveying and releasing these items." (Ref: Section 7.0, page 7-2)

Status: All tools and equipment which have been decontaminated are surveyed by Health Physics Technicians or other individuals specifically trained in that function. Procedures for control of contaminated material are in place. These procedures include monitoring requirements and limits for material leaving the controlled area as well as leaving the security "Protected Area."

The recommendation has been incorporated into procedures.

8.0 SAFETY ANALYSIS

8.1 INTRODUCTION

Design changes affecting the acceptance criteria for the TMI-1 FSAR safety analyses arise from several sources. First is the TMI-1 "Order and Notice of Hearing" (Reference 19) which contains NRC staff recommendations that certain changes be made to the plant. This order encompasses recommendations made in NRC bulletins 79-05 A, B and C and the TMI-2 Lessons Learned Task Force NUREG-0578 (Reference 20). Most of the changes listed below are being made in response to this order. Prior to the TMI-2 accident, B&W 177 FA plants received orders requiring modifications to the high pressure injection system to accommodate certain small break LOCA's. These changes are being evaluated as well. A third source of changes has originated from plant upgrades that Metropolitan Edison believes would improve plant performance. Some of these modifications were being evaluated prior to the TMI-2 accident on March 28, 1979.

8.2 AREAS OF INVESTIGATION

The plant modifications which are being investigated are summarized below. They are grouped according to tneir origin.

8.2.1 Modifications Resulting from the August 9, 1979 Order

1. The reactor protection high pressure trip setpoint has been changed to 2300 psig from 2390 psig. This lower trip setpoint in conjunction with the higher power operated relief valve (PORV) setpoint of 2450 psig results in a lower likelihood of PORV operation.

2. A complete loss of feedwater flow will initiate a reactor trip.

3. A turbine trip will initiate a reactor trip.

4. The emergency feedwater system will be modified before restart to allow:

   a. safety grade automatic unit ration of the steam and motor driven EFW pumps upon loss of all 4 reactor coolant pumps or a loss of both main feedwater pumps.

   b. loading of EFW pumps on the diesel generators and deletion of the blackout start interlock.

   c. alternate manual control for the EFW control valves.

   d. negative feed to steam differential pressure.

   e. loss of both main FW pumps.

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5. A long-term modification will provide safety grade actuation of the EFW pumps on the low steam generator level. This is a long-term item since further engineering is required. Plant safety therefore will be discussed with and without this feature.

8.2.2 Modification as Result of Order of May, 1978

Modifications to the high pressure injection system. The HPI injection lines have been cross connected to assure acceptable results from a break in a high pressure injection line. Cavitating venturis have been added to provide the proper flow split in the event of an HPI line break.

8.2.3 Modification Originating from within Met-Ed

1. Post accident instrument and valve operator availability will be improved by the addition of heat shrink tubing.

2. The switchover of the ECCS system suction supply from the borated water storage tank (BWST) will be accomplished automatically rather than by operator action.

3. The reactor building spray system will be modified to delete sodium thiosulfate. Sodium hydroxide will be retained. This change will provide equal drawdown of the BWST and NaOH tanks for a large spectrum of single failures.

4. Letdown Flow will be automatically isolated after a reactor trip.

5. Cavitating venturi's are being added to the emergency Feedwater system to prevent pump runout and to limit maximum flow to each OTSG.

8.2.4 I&E Bulletin 79-05C

Met-Ed has committed to install an automatic reactor coolant pump trip to be initiated on a SFAS coincident with an indication of a large (in excess of 10-20%) void fraction. (See section 2.1.2.5)

8.3 EFFECT OF CHANGES ON SAFETY ANALYSIS

Following are summaries of the accidents listed in Table 8-1. Table 8-1 indicates where FSAR analyses took credit for non-safety grade equipment, or where mitigation is dependent on a specific operating/emergency procedure or design margin. These conclusions will continue to be revised to account for plant design changes.

The event description and mitigating equipment are for the plant design before modification. The modifications discussed in the previous sections were considered in the review of each accident. If a modification affected that analysis, then a note as to its safety significance was made under the "conclusions" section.

8.3.1 Rod Withdrawal from Startup (FSAR Section 14.1.2.2)

1. Description

Uncontrolled reactivity excursion starting from a subcritical condition of 1% $\Delta k/k$ at hot standby.

2. Acceptance Criteria

i. Limit power to design overpower (112%)

ii. RCS pressure not to exceed code allowable of 2750 psig.

3. Mitigation

i. RPS trip on high pressure for fast power rises.

ii. Pressurizer code safety valves lift and peak pressure is limited to 2515 psia.

iii. Doppler coefficient provides a negative reactivity addition.

4. Conclusion

The FSAR analysis still bounds the modified TMI-1 plant design. The RCS high pressure trip is lower and safety margins are increased. Since no credit was taken for operation of the PORV, raising the valve setpoint does not change the analysis results. As discussed in Ref. 2, the PORV would lift for the worst case rod withdrawal accident which was analyzed in the FSAR. Nevertheless, the probability of occurrence has been decreased so that safety margins have been improved and lifting of the PORV is not likely for a broad spectrum of rod withdrawal accidents.

8.3.2 Rod Withdrawal at Power (FSAR Section 14.1.2.3)

1. Description

Accidental withdrawal of a control rod group at normal rated power, without ICS control and a 1% shutdown margin.

2. Acceptance Criteria

i. Limit power to design overpower of 112%.

ii. RCS pressure not to exceed code allowable (2750 psig).

3. Mitigation

i. RPS trips on high pressure for slow transients and high neutron flux for fast transients.

ii. Doppler and moderator coefficients provide negative reactivity addition.

4. Conclusions

The FSAR analysis bounds the modified TMI-1 plant design. Lowering of the reactor trip setpoint increases safety margins for this event. Credit was not taken for PORV operation. As discussed in Reference 2, some low worth rod withdrawals can result in PORV actuation. Nevertheless, the probability of such an occurrence has been greatly decreased by the changes in the PORV and high pressure trip setpoints.

8.3.3 Moderator Dilution Accident (FSAR Section 14.1.2.4)

1. Description

Diluted makeup water is inadvertently added to the reactor coolant system at a rate of 500 gpm beginning at normal power. RCS boron concentration is at its highest initial value. The result is a reactivity insertion, increased power, pressure and temperature. The addition of one makeup tank volume of unborated water changes the shutdown margin by .8% $\Delta k/k$.

2. Acceptance Criteria

i. Reactor power will be limited to less than the design overpower (112%).

ii. Reactor coolant system pressure will be limited to less than code allowable 2750 psig.

iii. The minimum shutdown margin will be at least 1% $\Delta k/k$.

3. Mitigation

i. High pressure or high temperature trip.

ii. Termination of deborated water to makeup tank on reactor trip.

iii. Termination of makeup flow on high pressurizer level.

4. Conclusion

The FSAR analysis bounds the modified TMI-1 plant design. Lowering of the high pressure trip setpoint increases the safety margins for this accident. Operation of the PORV was not assumed in the original analysis, and peak pressure is 2435 psia. Therefore, the PORV setpoint will not be reached during this transient.

Reactor power is limited to 107.3%, and the final shutdown margin is greater than 1% $\Delta k/k$ even with the most reactive rod stuck out of the core all of the acceptance criteria for this accident are met.

8.3.4 Cold Water Addition (FSAR Section 14.1.2.5)

1. Description

Startup of one or more idle reactor coolant pumps can cuase excess heat removal from the primary coolant system. This cooldown can cause positive reactivity insertions, which result in a power rise. The worst case event is the startup of two reactor coolant pumps from 50% power. A tripped rod worth of 1% $\Delta k/_k$ is used in the analysis.

2. Acceptance Criteria

i. Limit overpower to less than the maximum design overpower (112%).

3. Mitigation

i. RPS trip on high pressure for slow power increases or power/flow mismatch for rapid power increases.

ii. RC pump/power monitor limits initial conditions under which event can occur.

4. Conclusion

Lowering of reactor trip setpoint increases safety margins for this event. The FSAR analysis was performed without taking credit for PORV. Peak pressure did not exceed 2400 psia, hence the PORV will not lift during this event.

The FSAR analysis bounds the modified TMI-1 plant design.

8.3.5 Loss of Coolant Flow (FSAR Section 14.1.2.6)

1. Description

Fuel rods experience a limiting DNB transient when all four reactor coolant pumps trip on loss of offsite power or when one pump experiences a locked rotor resulting in an instantaneous loss of flow. The loss of flow analysis is performed from 114% normal power, nominal reactor coolant pump flow, a +2 F core inlet temperature error and a -65 psi error in pressure. Reactor trip delay is assumed to be 620 ms. and a 1% $\Delta k/_k$ subcritical margin is assumed at hot standby. The event is analyzed past the time that the minimum DNBR occurs.

The locked rotor accident is performed from an initial power level of 102% power, with a rampdown in flow from 100% to 75% in 100 ms. Temperature and pressure were the same as for the loss of flow accident. Reactor trip delay is assumed to be 650 ms.

2. Acceptance Criteria

i. DNB is greater than 1.3 for a loss of coolant flow.

ii. DNBR is greater than 1.0 for a locked rotor accident.

ACCIDENTS/TRANSIENTS CONSIDERED FOR RE-ANALYSIS

TABLE 8-1

	Affected by Plant Changes	Depend on Operator Action	Non-Safety Equipment Used	Don't Consider Failure of Non-Safety Equipment	Analyses Need to be more Realistic
Startup Accident	X			X	
Dilution Accident	X		X	X	
Cold Water	X			X	
Loss of Coolant Flow				X	X
Dropped Rod			X		
Loss of AC	X	X	X		
Loss of Elec. Load	X		X	X	
Steam Line Failure			X	X	X
Steam Generator Tube Failure			X	X	
Fuel Handling Accident			X		
Rod Withdrawal at Power	X			X	
Rod Ejection Accident	X				
Small Break LOCA	X		X	X	X
EVENTS NOT ANALYZED IN FSAR					
EFW Inadvertent Initiation	X		X	X	X
Loss of Feedwater	X		X		
Feed Line Break	X				
HPI Line Break	X	X	X		
Loss of Offsite Power	X			X	X

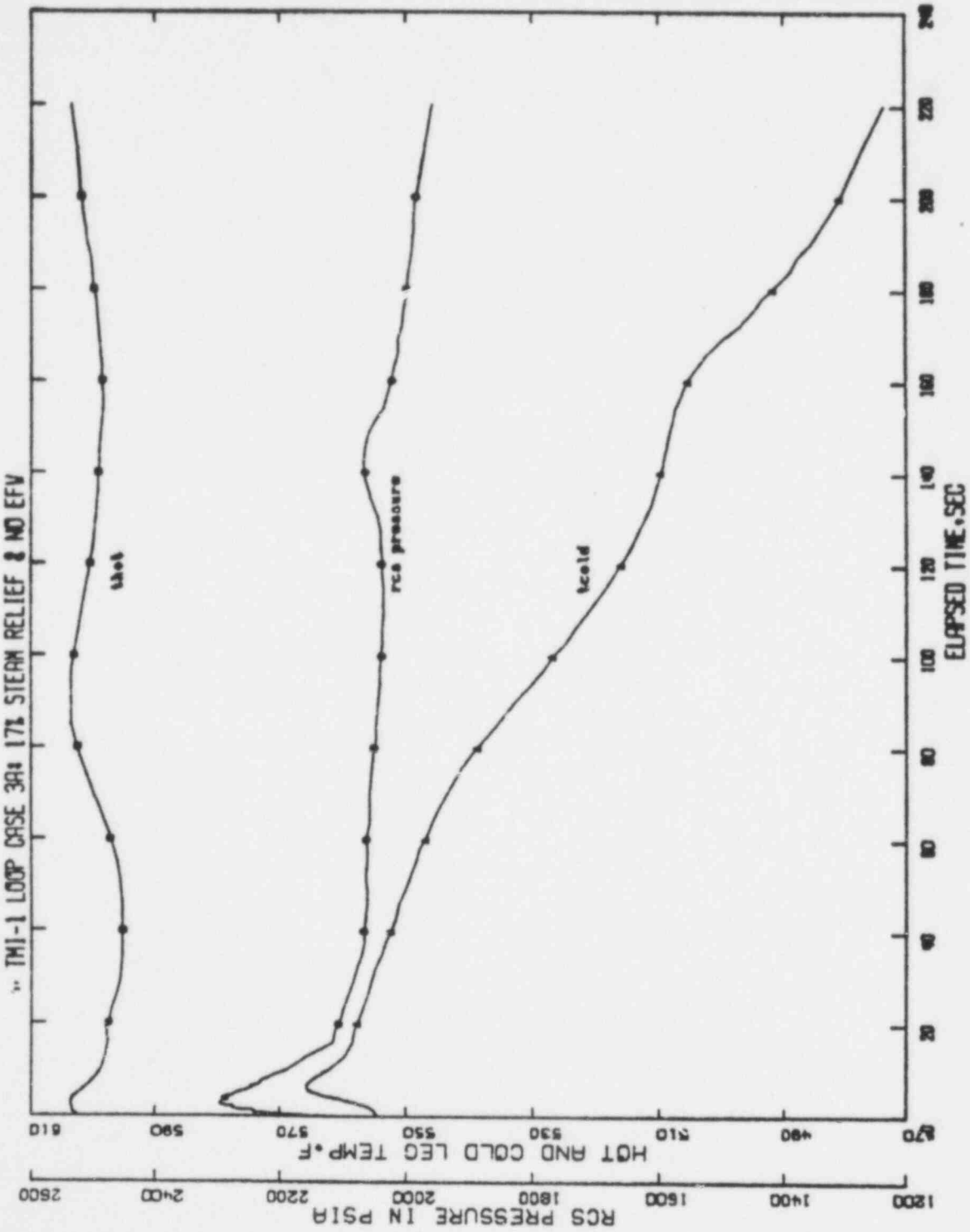
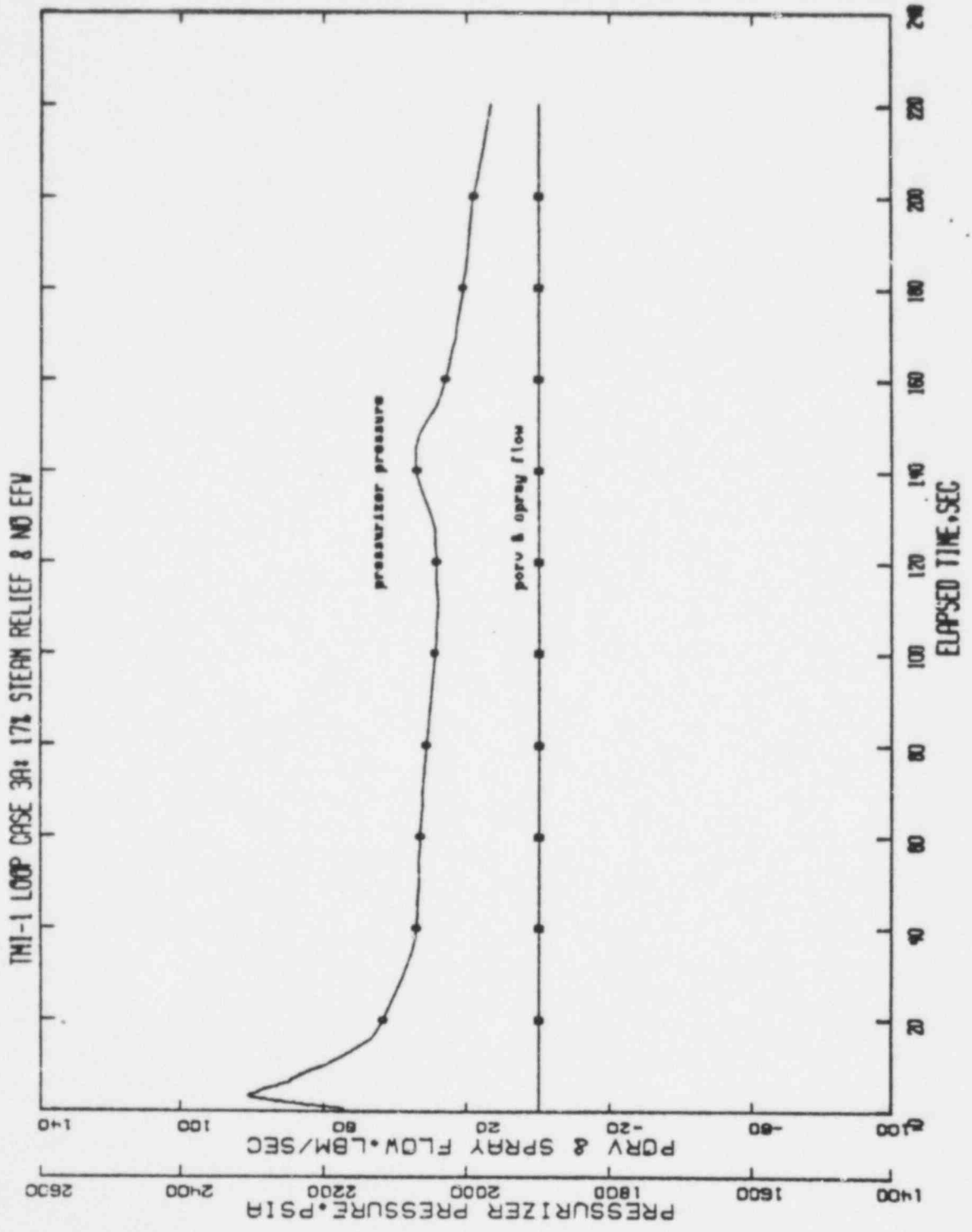


Figure 8A-2a
Sheet 1

Figure 8A-2a
Sheet 2



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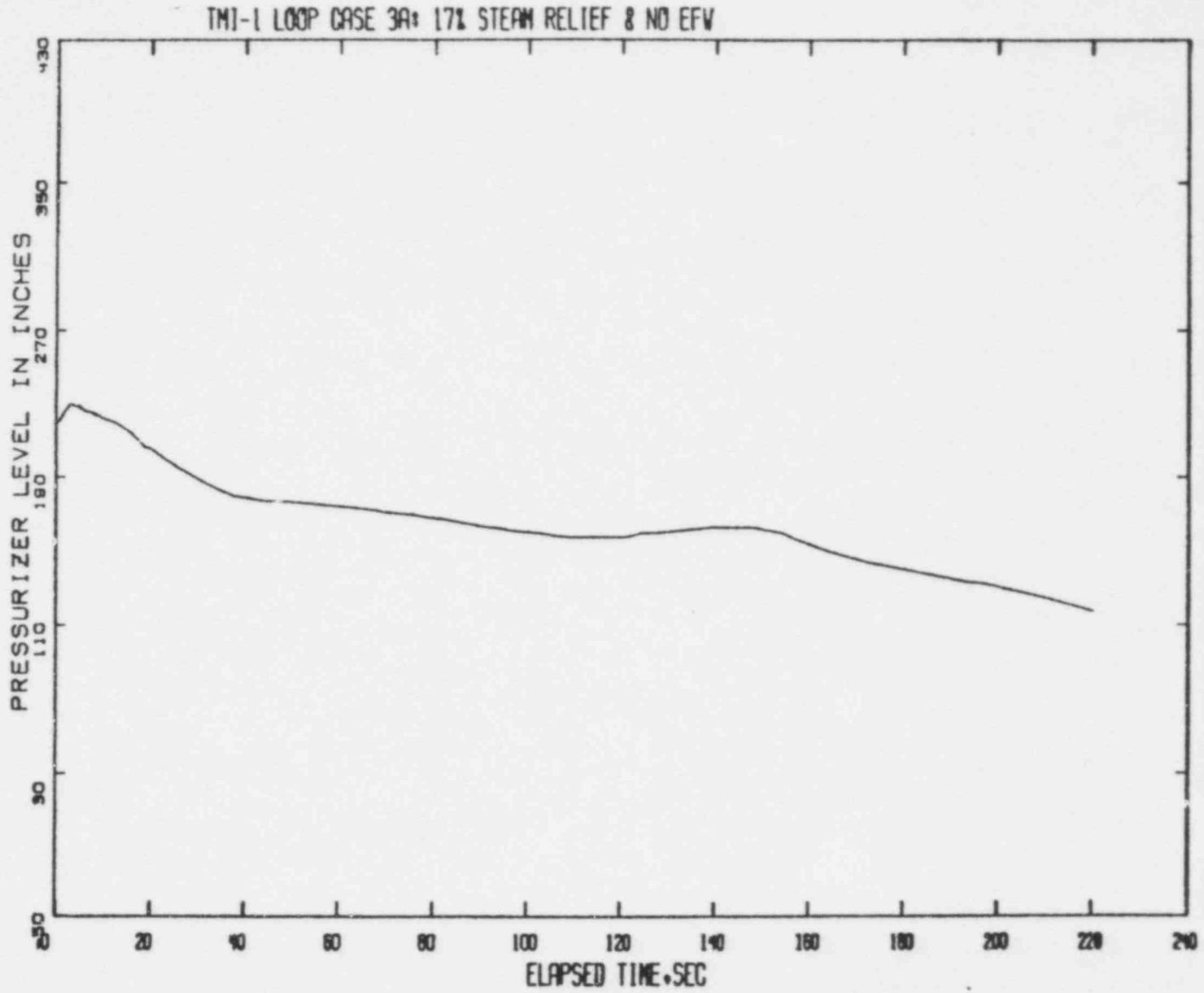

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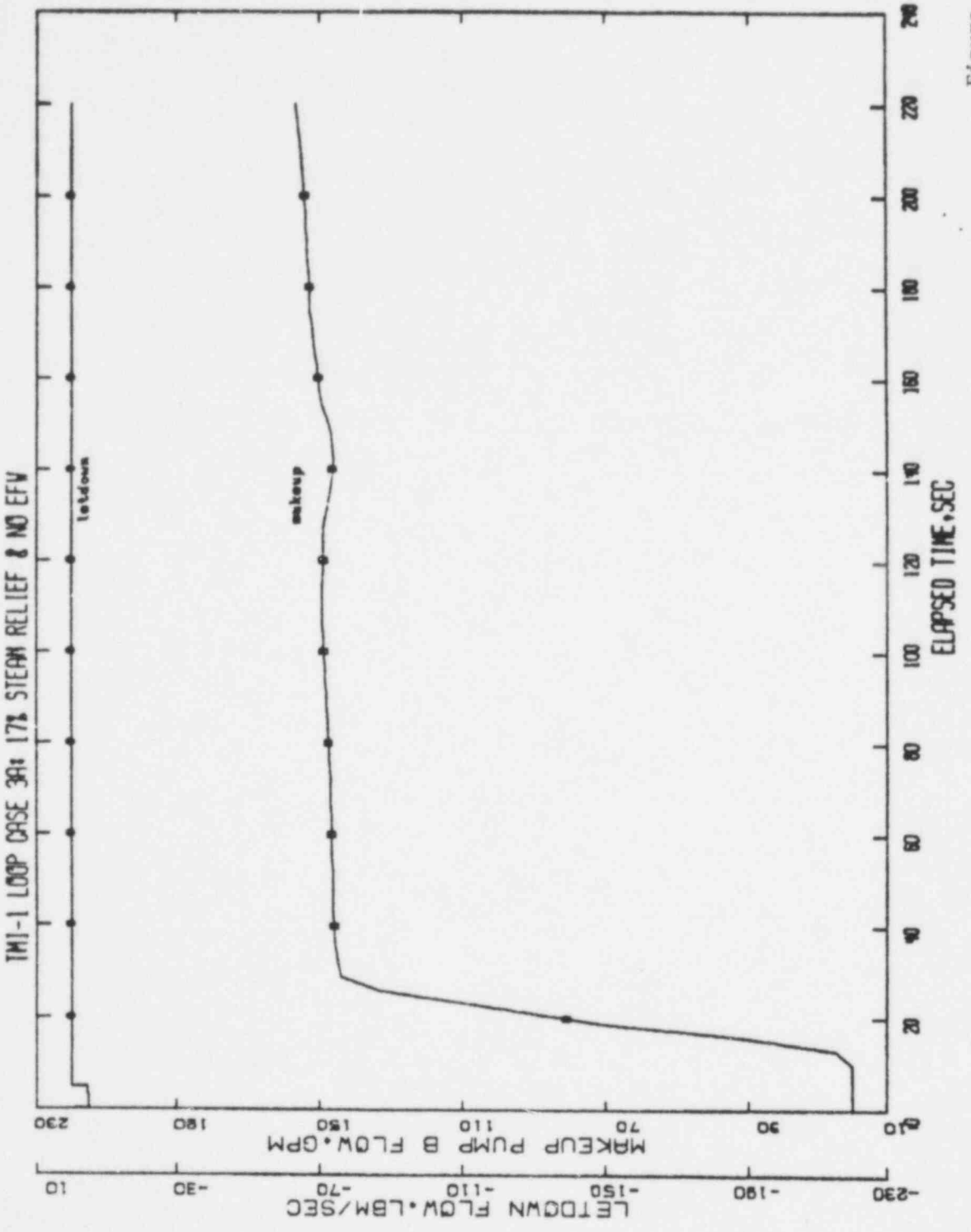

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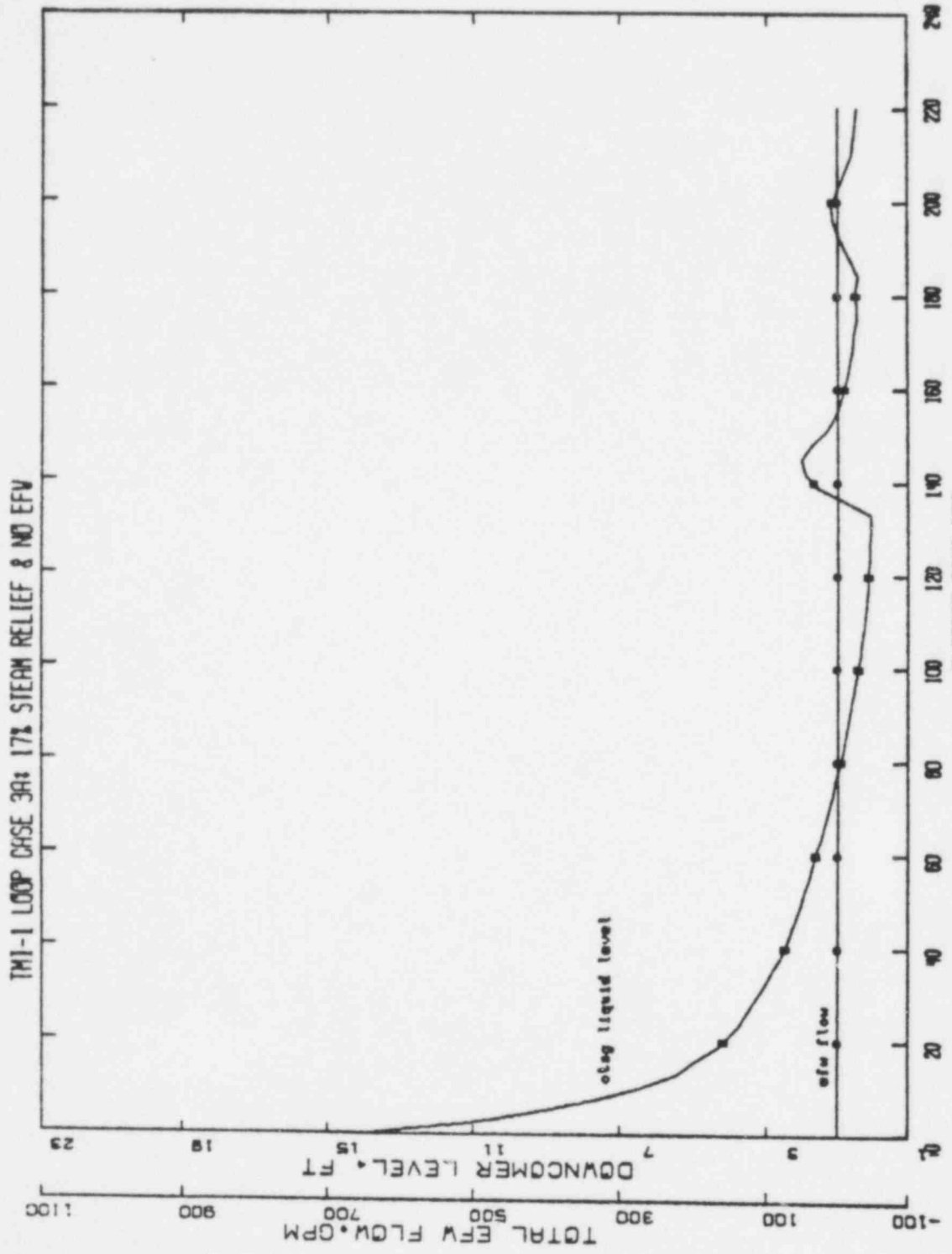

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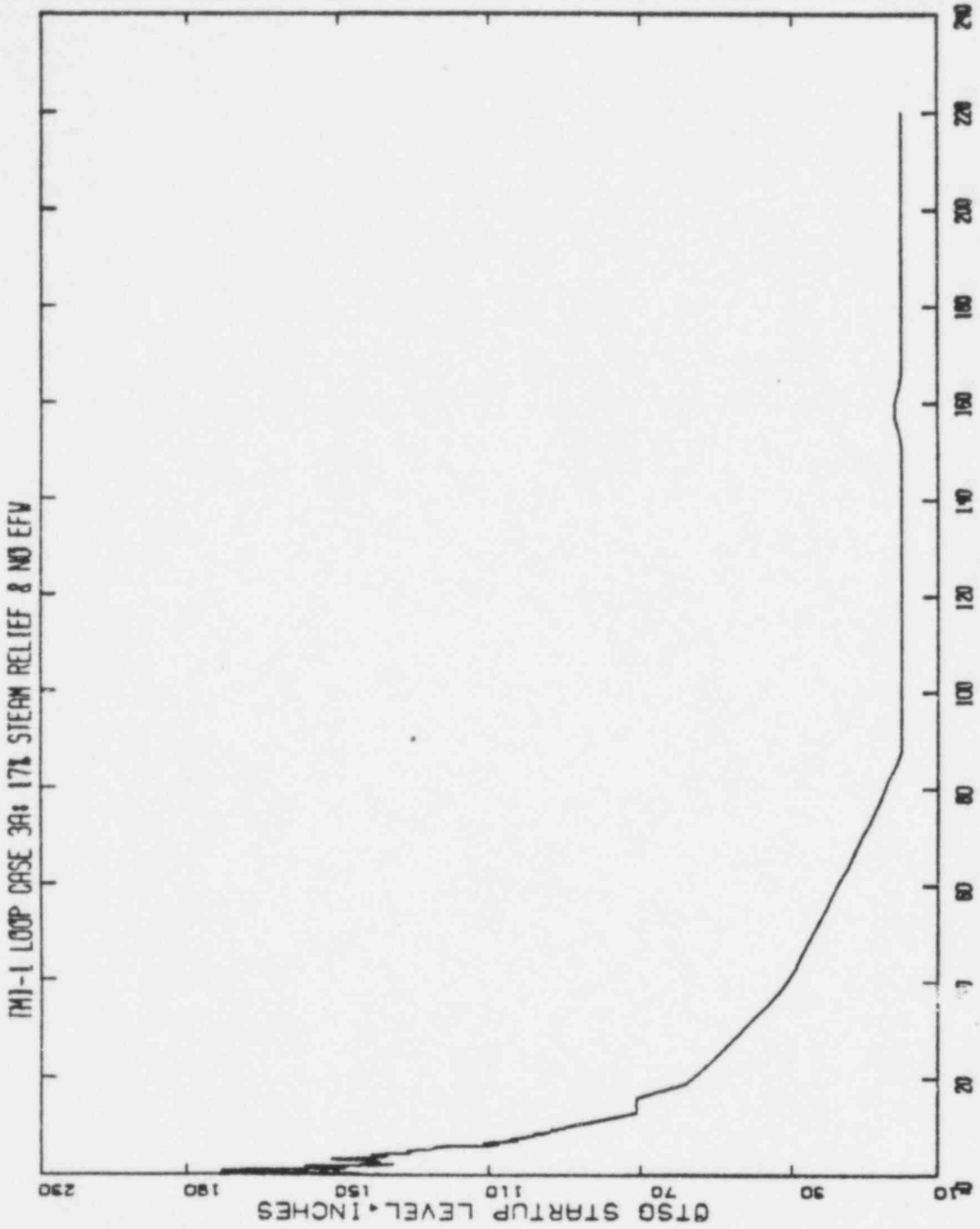

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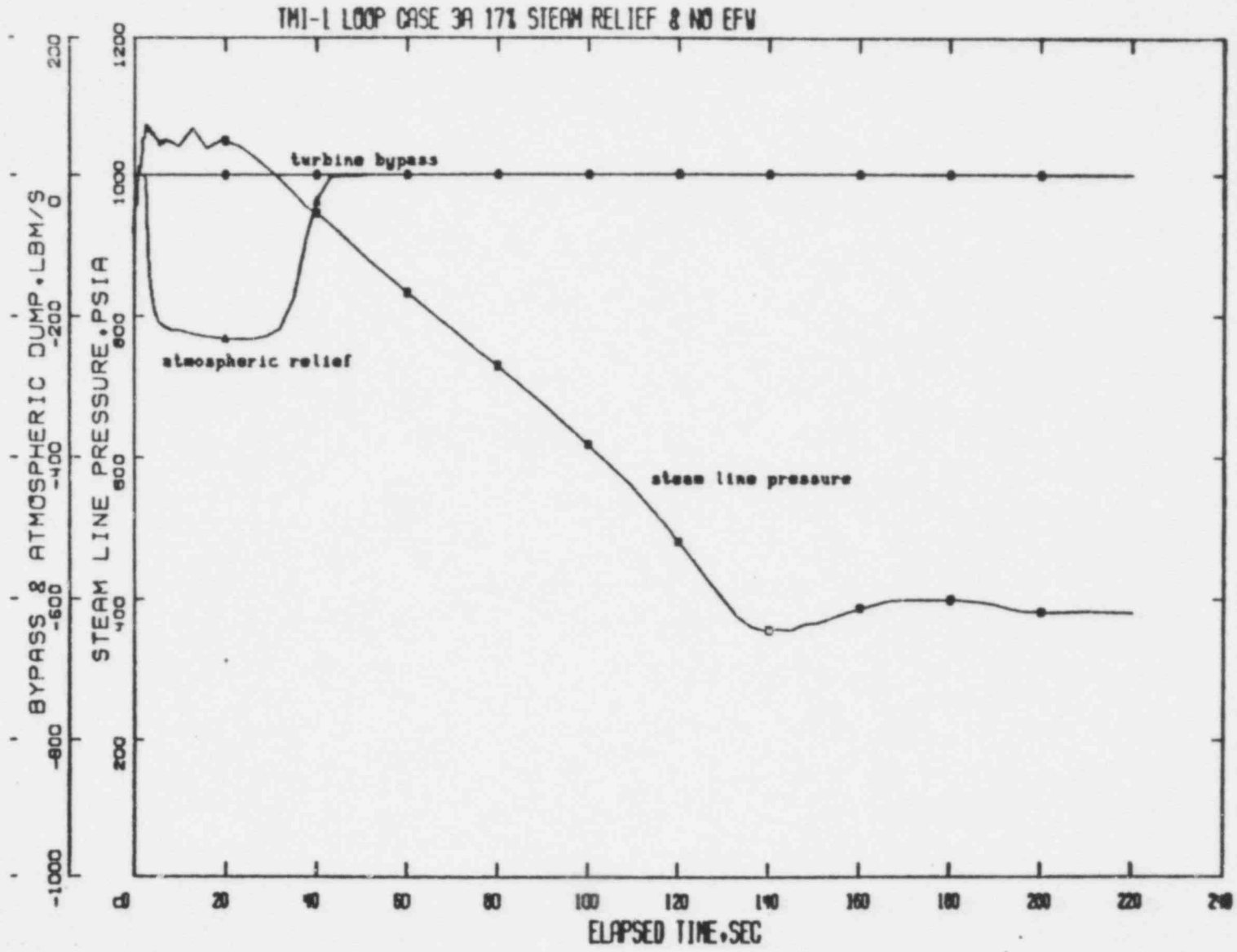

Figure 8A-2a
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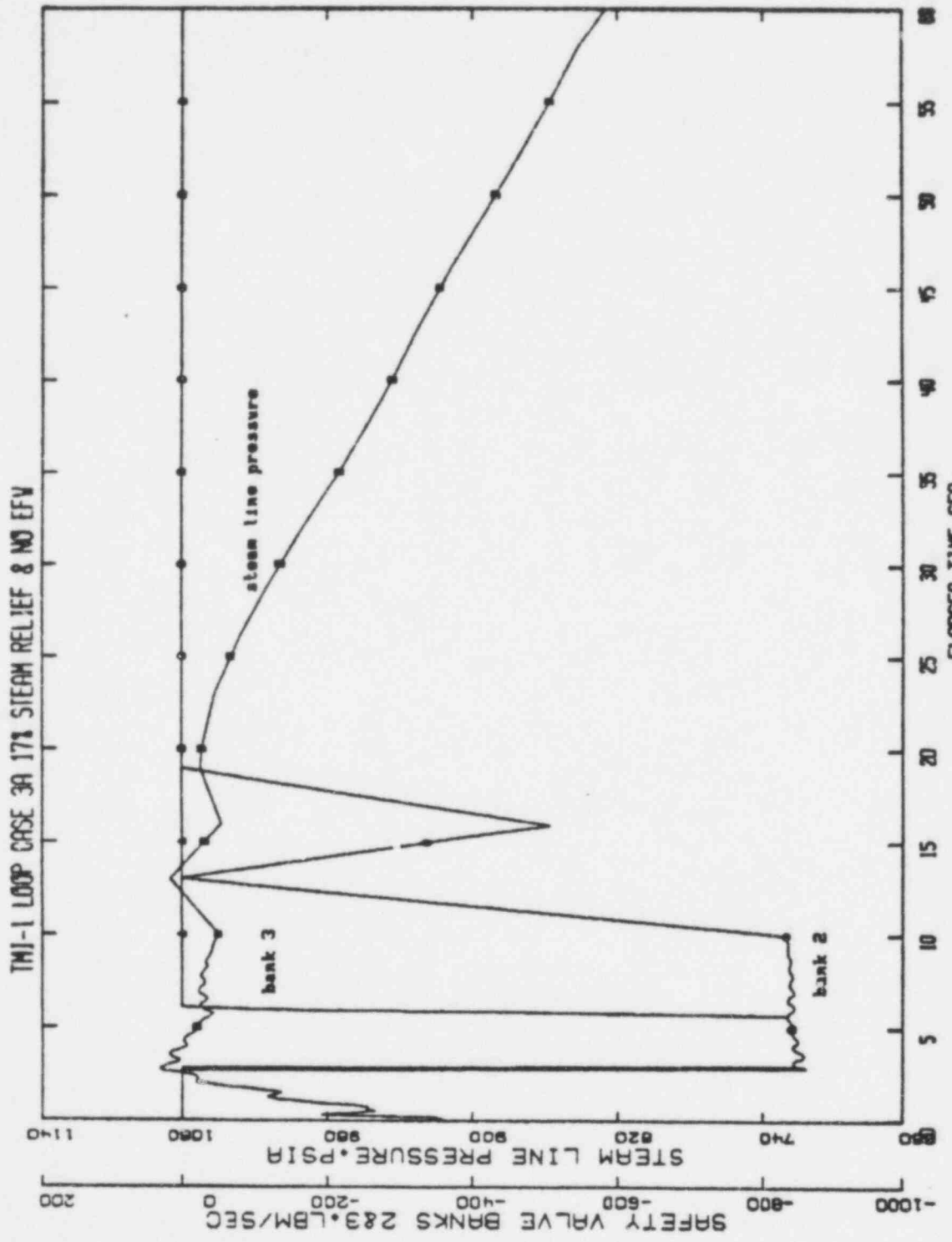

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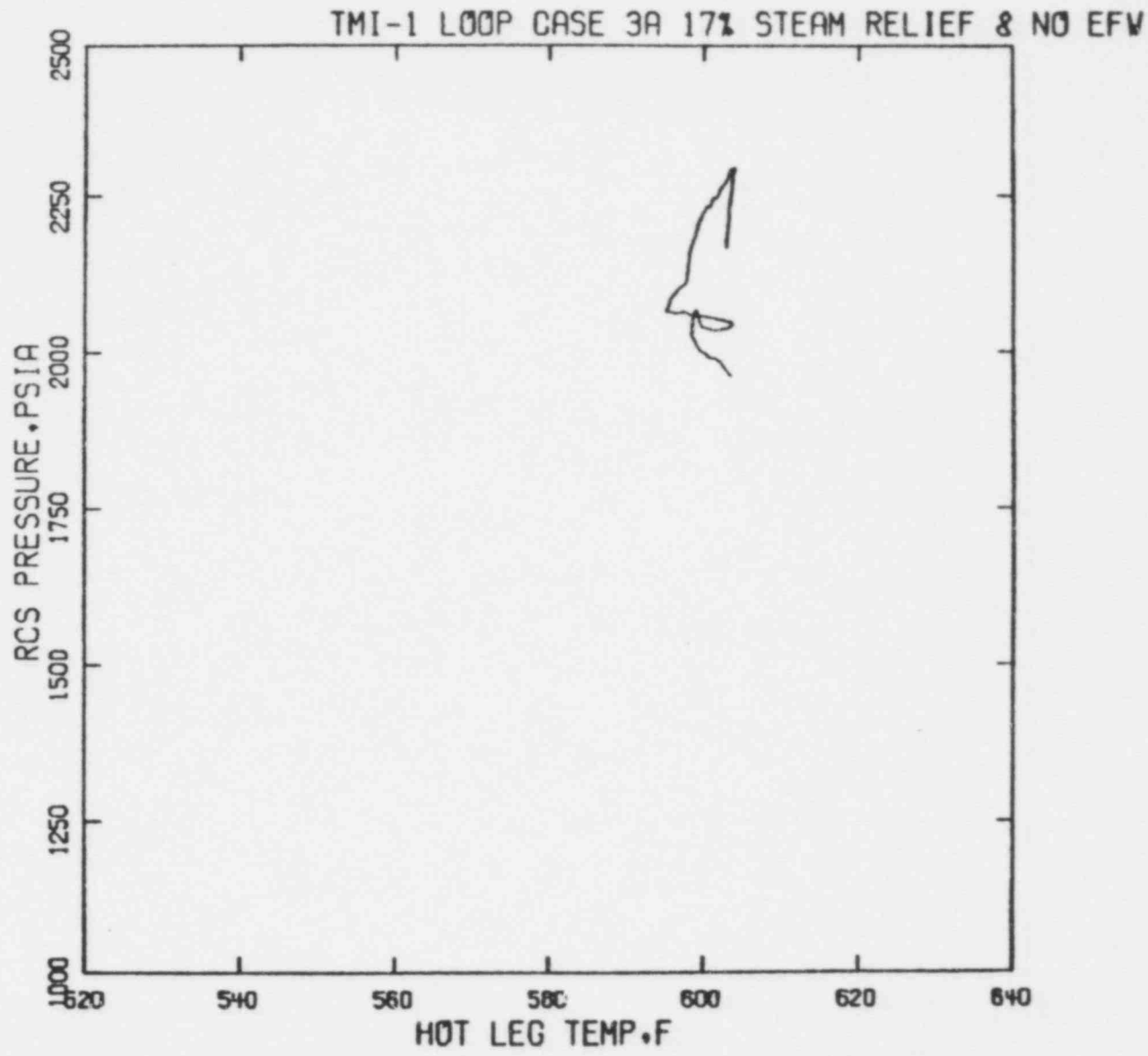

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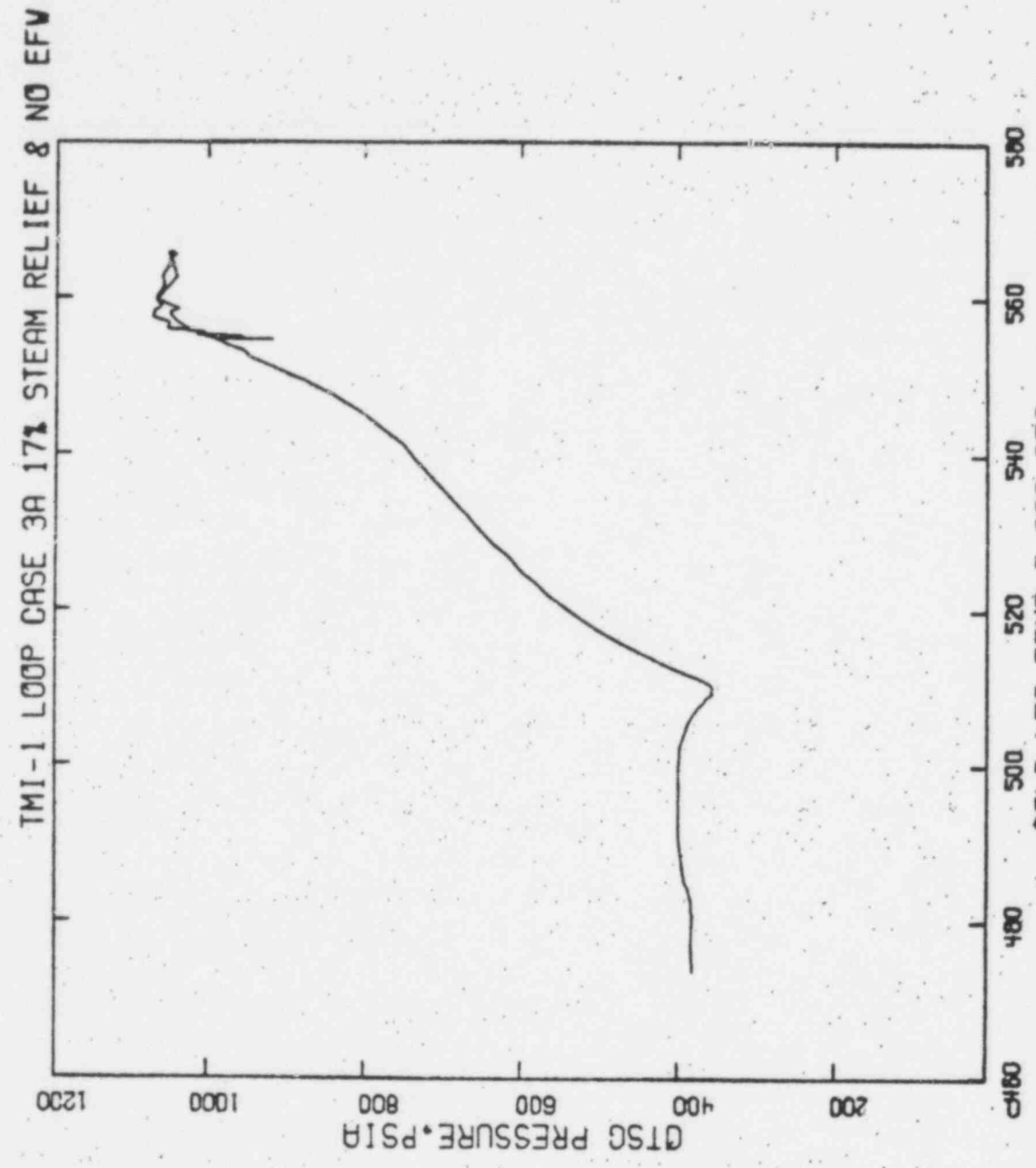

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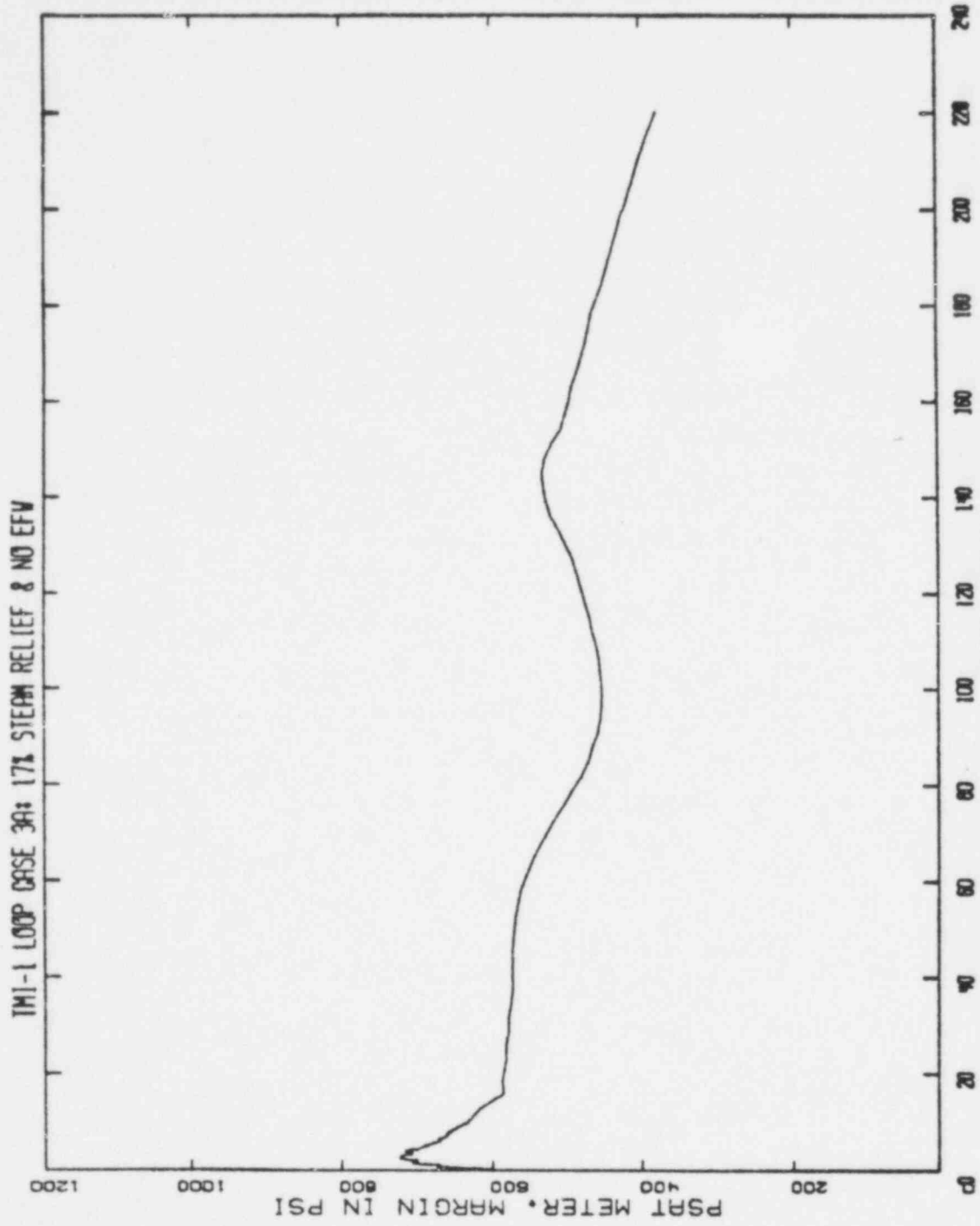

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TABLE 9.2-1

Restart Modification Additional Loads

Items	Title	Additional Load in KW	MCC or Dist. Panel	Bus
	Upgrade Decay Heat System			
	Vent Valve A	0.7	DC Panel 1E	
	Vent Valve B	0.7	DC Panel 1F	
	Vibration Monitor	0.6	MCC 1B	Bus 1S
2.1.1.1	Reactor Trip	Neg.		Vital Busses
		Each operator will be connected to its associated channel		1A 1B 1C 1D
2.1.1.6.3.1	Incore Thermocouples	No requirement		
2.1.1.5	Containment Isolation	less than 1		Later
2.1.1.2	Valve Position Indication	0.23	PNI ATB	Inverter 1E
	Computer	Less than 0.5		Inverter 1E
2.1.1.2	Power Operated Relief Valve Position Indication	Less than 1		Later
2.1.1.4	H_2 Recombiner			
	A bus heater & blowers	45	MCC 1A	1P
	B bus heater & blowers	45	MCC 1B	1S
	space heater	0.8	PNL CT-E	1S
	isolation valves	0.53	DC PNL 1A	
	isolation valves	0.53	DC PNL 1B	
	position indication	0.025	Swing DC PNL 1M (PNLS 1A and 1B)	
2.1.1.7	Emergency Feedwater	Less than 1		
		Pump A	VBA	Inverter 1A
		Pump B	VBB	Inverter B
	Changes to HPI System to Accommodate Small Break LOCA	0.05		Inverter 1E

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TABLE 9.2-1 (Cont'd.)
Restart Modification Additional Loads

Items	Title	Additional Load in KW	MCC or Dist. Panel	Bus
2.1.1.3	Pressurizer Heaters connected only on loss of offsite power and no E.S.			
	Heaters to A System	126 KW		1P
	Heaters to B System	126 KW		1S
2.1.1.6.3.3	P_{sat} Alarm	less than 1		Later
2.1.1.6	Reactor Coolant System Temp.	No Requirement		
	Reactor Building Sump Water Level	Less than 1		
	Reactor Building Cooling Fan Motors	No Requirement		
	Fire Protection Change	later		

NOTES

1. Items marked "neg." indicates less than 100 watts load and the source has not been determined.
2. Items marked "no requirement" indicates no load change or no additional load requirements.
3. Items marked "later" indicates that changes are unknown. This information will be supplied when available.

11.0 TECHNICAL SPECIFICATIONS

11.1 INTRODUCTION

A considerable number of plant modifications are being accomplished in response to TMI-2 Lessons Learned (NUREG-0578), the TMI-1 Order and Notice of Hearing - August 9, 1979, IE Bulletins, and Met-Ed's review of the TMI-2 accident. The hardware modifications are described in Section 2.0 of this report. In some instances, Technical Specification changes are appropriate to account for systems and changes to systems that play a significant role in mitigating the consequences of accidents or transients. These new draft Technical Specifications are discussed in Section 11.2. Formal requests to modify the TMI-1 Technical Specifications will be forwarded to the NRC at an appropriate time following PORC and GRC review of the Technical Specifications which must be completed before final submittal.

11.2 DRAFT TECHNICAL SPECIFICATIONS

This section contains evaluations of those proposed modifications for which changes to the Technical Specifications will be requested. Draft Technical Specifications pages in the TMI-1 format are contained in Appendix 11A.

11.2.1 Reactor Trip on Loss of Feedwater or Turbine Trip

Introduction

Item B.5 of IE Bulletin 79-05B requires licenses to "Provide for NRC approval a design review and schedule for implementation of a safety grade automatic anticipatory reactor scram for loss of feedwater, turbine trip, or significant reduction in steam generator level." Item B.7 requires the submittal of Technical Specifications for design changes including those changes associated with Item B.5.

Evaluation

The reactor trips on loss of feedwater or turbine trip are designed as anticipatory reactor trips which respond to equipment situations which would produce significant primary system pressure transients. By tripping the reactor on the anticipation of a pressure transient, (1) the probability of an overpressure trip is reduced and (2) the challenge rate to the pressurizer power operated (Electromatic) relief valve and the pressurizer code safety valves is reduced. The design of the reactor trips on loss of feedwater or turbine trip incorporate a 2-out-of-4 logic, are fully testable, and meet the single failure criterion of IEEE-279. A description and evaluation of these reactor trips are contained in Section 2.1.1.1 of "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1."

Since the reactor trip on loss of feedwater and the reactor trip on turbine trip are of the same safety grade as other Reactor Protection System trip functions, the Limiting Conditions for Operation in draft Technical Specifications 3.5.1.1 and the Surveillance Requirements in draft Technical Specification 4.1.1 (Items 45 and 46 of Table 4.1-1) have been chosen to be consistant with other Reactor Protection System functions, a "check" being required each shift, a "test" each month, and a "calibration" during each refueling outage. Bypass of the loss of feedwater trip below 10% power, and turbine trip below 20% power is permitted to allow normal reactor startup oeprations.

Conclusions

In conclusion, we have determined that, with regard to the reactor trip on loss-of-feedwater and the reactor trip on turbine trip:

(1) The probability or consequences of accidents previously evaluated have not been increased. The proposed trips are anticipatory and have not been taken credit for in the accident analyses. The reactor trip on overpressure and the pressurizer Code Safety Valves remain the principal means of mitigating pressure transients.

(2) No accidents, other than those previously considered, will be introduced. The additional reactor trips have been designed so as not to introduce additional failure modes into the Reactor Protection System or other safety equipment. Moreover, by anticipating significant pressure transients, the challenge rate to the overpressure trip and pressure relieving capacity has been reduced.

(3) No safety margins have been reduced. Since the additional reactor trips scram the reactor on anticipation of significant pressure transients, the peak pressure associated with these transients can be expected to decrease which would result in an increase in safety margins as a result of postulated turbine trip or loss-of-feedwater transients.

For these reasons, we conclude that implementation of the design for additional reactor trips, and adoption of associated Technical Specifications, does not involve unreviewed safety questions with regard to the criteria of 10CFR Part 50, Section 50.59(a)(2).

11.2.2 Position Indication of PORV and Safety Valves, Setpoints

Introduction

Section 2.1.3 of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979, contains NRC recommendations on installations of valve position indications for safety and relief valves. The guidance on safety/relief valve position is to, "Provide in the control room either a reliable, direct position indication for the valves or a reliable flow indication device downstream of the valves."

Evaluation

In response to the above recommendations, a system of indirect safety and relief valve position indications has been designed. The safety/relief valve position indication system, described in Section 2.1.1.2 of "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1", consists of the following:

(1) delta-pressure taps at discharge piping elbows downstream of the safety and relief valves, and

(2) acoustic monitors (accelerometers) mounted on the pressurizer power operated relief valve.

The above sensors are in addition to the tailpipe thermocouples which are presently installed.

Technical Specifications, Limiting Conditions for Operation and Surveillance Requirements, will be proposed for the safety/relief valve position instrumentation. The Limiting Conditions for Operation, contained in draft Technical Specification 3.5.5, requires the three delta-pressure monitors and an accoustic monitor to be operable. The remedial action to be taken if one or more delta-pressure monitors or the acoustic monitor becomes inoperable is to fix the monitors prior to startup following the next cold shutdown. This requirement is based upon the following considerations:

(1) The sensors for the delta-pressure and accoustic monitors are located inside containment and would most likely require containment entry for repair or replacement.

(2) Extended periods of operation without the delta-pressure or acoustic monitors will not preclude the reactor operator detecting a leaking or stuck-open safety or relief valve. Several indications of safety or relief valve discharge flow include:

   (a) reactor coolant drain tank (pressurizer quench tank), level, temperature, and pressure,

   (b) safety and relief valve tailpipe temperature.

In addition to delta-pressure and accoustic monitor Technical Specifications, Limiting Conditions for Operation are proposed to establish "dual" setpoints for the pressurizer power-operated (Electromatic) relief valve. Draft Technical Specification 3.1.1.4a requires a setpoint of 2450 psig $\pm$ 1% when the reactor coolant system temperature is above 275°F. The increase in this setpoint (from 2255 psig to 2450 psig) conforms to the NRC guidance contained in IE Bulletin No. 79-05B. No operability or remedial action is associated with draft Technical Specification 3.1.1.4a since the Electromatic relief valve is not safety-grade and thus is not credited in the safety analyses. Draft Technical Specifica-

tion 3.1.1.4b provides the second of the "dual" setpoints as 485 psig ± 1% when the reactor coolant temperature is below 275°F. The 485 psig setpoint is associated with considerations relating to overpressurization of the reactor coolant system under "cold" conditions. Overpressurization is addressed in our March 13, 1978 submittal, Technical Specification Change Request No. 74.

An additional aspect of IE Bulletin 79-05B, item B.3, involved a decrease in the RPS high pressure trip setpoint to reduce the challenge rate to the pressurizer Electromatic relief valve. The recommendations of B&W, in an April 20, 1979 communication, indicated that the RPS high pressure trip setpoint should be reduced to 2300 psig. The decrease in the RPS high pressure trip, from 2390 to 2300 psig is incorporated in TMI-1 draft Technical Specification 2.3.1 (see Section 11.2.12).

The Surveillance Requirements associated with the delta-pressure and acoustic safety and relief valve monitors are contained in draft Technical Specification 4.1.1 (Items 47 and 48 in Table 4.1-1); these monitors are to be checked each shift and tested/calibrated each refueling period. The "check" and "test" surveillance need not be performed when T_{AVG} is below 200°F since the reactor would be shutdown and this safety function unnecessary. Surveillance that is not performed due to a reactor shutdown greater than one month should be performed prior to startup. This requirement has been presented in draft Technical Specification 4.1.1 and made applicable to all surveillance requirements in Table 4.1-1. Accessibility considerations, as noted previously are significant with regard to the delta-pressure and accoustic monitors and are the determining factor in the test/calibration interval.

A Surveillance Requirement (setpoint test) for the pressurizer Electromatic relief valve is incorporated in draft Technical Specification 4.1.2; a refueling period interval has been selected to be consistent with the pressurizer safety valve surveillance interval.

Conclusion

In conclusion, with regard to the additional requirements for the delta-pressure and acoustic monitors, and the setpoint requirements for the pressurizer Electromatic relief valve:

(1) The probability or consequences of accidents, previously evaluated have not been increased. The requirement for operability and surveillance of the safety and relief valve monitors increases the probability that misoperation of the relief or safety valves will be detected, remedial action taken, and thus reduces the consequences associated with certain small-break loss-of-coolant accidents.

(2) No accidents, other than those previously considered, will be introduced. The delta-pressure and acoustic instrumentation have no automatic functions and therefore cannot change the course of any accident or transient; sufficient confirmatory information is available in the control room to detect improper functioning of these monitors. With regard to the pressurizer Electromatic relief valve, the requirement for periodic testing of the setpoint will enhance the availability of this equipment.

(3) No safety margins have been reduced. Although the setpoint of the pressurizer Electromatic relief valve has been increased, no credit was taken for this equipment in the safety analysis. The decrease in the RPS high pressure trip setpoint will cause the reactor to trip earlier in the course of significant pressure transients and thus reduce the peak pressure during the transient.

For the reasons presented above, implementation of the design changes associated with the delta-pressure and acoustic monitors and associated Technical Specifications, including those addressing the setpoint of the pressurizer Electromatic relief valve, do not involve any unreviewed safety considerations with regard to the criteria of 10CFR Part 50, Section 50.59(a)(2).

11.2.3 Emergency Power Supply Requirements - Pressurizer Heaters

Introduction

Section B.1.b of IE Bulletin 79-05B requires licensees of operating reactors to develop procedures and train personnel to "...assure availability of adequate capacity of pressurizer heaters, for pressure control and maintain primary system pressure to satisfy the subcooling cirterion for natural circulation." Section 2.1.1 of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," goes further in that it recommends that reactors, "Provide redundant emergency power for the minimum number of pressurizer heaters required to maintain natural circulation conditions in the event of loss of offsite power. Also provide emergency power to the control and motive power systems for the power-operated relief valves and associated block valves and to the pressurizer level indication instrument vhannels."

Evaluation

Section 2.1.3 of "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1" describes design changes, and operator actions, that are required to supply 126 KW of pressurizer heaters from each of two independent engineered safeguard power sources in the event offsite power is lost. The manual transfer of power from the normal (balance of plant) to the back-up (engineered safeguards) power source involves the use of a "Kirk Key" system that assures proper transfer of power from the normal to the back-up power

source. In the event that an engineered safeguards actuation signal is received while the pressurizer heaters are powered from the diesel generators, the pressurizer heater loads are automatically shed from the diesel generators to prevent overloading of the diesels. Upon existence of an engineered safety features actuation signal (indicating a LOCA), primary system pressurization is no longer a consideration.

To assure proper load shedding (breaker operation) of the pressurizer heaters, from the diesel generators upon an engineered safety feature actuation signal, a test of the engineered safety features pressurizer heater supply breaker will be undertaken on a periodic basis. Technical Specification 4.6.1.b requires a test of the diesel generators, during each refueling shutdown, to determine proper automatic response under loss of normal AC power conditions concurrent with an engineered safety features actuation signal. Draft Technical Specification 4.6.1.b would also include a requirement to confirm proper operation of the engineered safety features pressurizer heater supply breakers upon receipt of an engineered safety features actuation signal.

With regard to the remaining requirements of NUREG-0578, Section 2.1.1, to supply emergency power capability for the PORV, the block valve, and pressurizer level instrumentation, Sections 2.1.1.3.2, 2.1.1.3.3, and 2.1.1.3.4 of the Restart Report indicate that these requirements are satisfied by existing equipment.

Conclusion

In conclusion, with regard to the provisions for transfer of pressurizer heater loads from normal to back-up power supplies:

(1) The probability or consequences of accidents, previously evaluated, have not increased. Periodic testing of the engineered safety features pressurizer heater supply breaker will assure that, in the event that the pressurizer heaters are powered by the diesels-generators when an engineered safety features actuation signal is received, the pressurizer heaters will be shed from the diesel generators supply busses.

(2) No accidents, other than those previously considered, will be introduced. The manual transfer of the pressurizer heater supply loads, in the correct manner, is assured by the Kirk Key system. Periodic testing of the engineered safety features pressurizer heater supply breaker will prevent diesel overloading in the event that the pressurizer heaters are powered by the diesels-generators when an engineered safety features actuation signal is received.

(3) No safety margins have been reduced. The availability of the pressurizer heaters on loss of offsite power provides additional assurance that the primary system subcooling margin can be maintained such that natural circulation will be enhanced.

Based upon the above, we conclude that plant modifications necessary to allow manual transfer of selected pressurizer heater loads, from normal backup power sources, and adoption of associated Technical Specifications, do not involve an unreviewed safety question with regard to the criteria of 10CFR Part 50, Section 50.59(a)(2).

11.2.4 Post-LOCA Hydrogen Recombiner System

Introduction

Section 2.1.5 of "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," NUREG-0578, July 1979, contains a Task Force minority opinion that, "...all operating reactors, which do not already have the capability, be required to provide the capability to add, within a few days after an accident, a hydrogen recombiner system for post-accident hydrogen control." Section 2.1.4 of "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1" contains a description and evaluation of design modifications that are required to:

(1) Install at TMI-1 a hydrogen recombiner that was purchased for TMI-2, and

(2) Provide structural, piping, and electrical facilities such that a second hydrogen recombiner can be installed after an accident, within the time period available before it is required to be operational.

Evaluation

Present NRC Policy, as reflected in "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors", NUREG-0103, requires Technical Specifications for installed hydrogen recombiners. The proposed Technical Specifications for the hydrogen recombiners have been adopted from the Technical Specifications for TMI-2, specifically, TMI-2 Technical Specification 3/4 6.4.2. The hydrogen recombiner Technical Specifications for TMI-2, based upon NRC's Babcock and Wilcox Standard Technical Specifications, contain:

(1) Limiting Conditions for Operation requiring one operable hydrogen recombiner during startup and power operation,

(2) Surveillance Requirements for the following inservice inspection program:

    (a) A recombiner functional test once per 92 days, and

    (b) The following surveillance every 18 months: Channel calibration of recombiner instrumentation, visual examination, heater functional test, and heater electrical circuit integrity.

In addition, the hydrogen recombiner is required to undergo surveillance prior to startup following on extended outage.

The Limiting Condition for Operation for the hydrogen recombiner is incorporated in draft TMI Technical Specification 3.6, "Reactor Building"; the Surveillance Requirements for the hydrogen recombiner are contained in TMI-1 draft Technical Specification 4.4.4, "Hydrogen Recombiner System."

Conclusion

In conclusion, with regard to the installed hydrogen recombiner and associated Technical Specifications:

(1) The probability or consequences of accidents previously evaluated have not increased. The use of hydrogen recombiners at TMI-1 would result in lower off-site doses, in the event of a LOCA, when compared with other post-accident hydrogen control techniques requiring containment venting.*

(2) No accidents, other than those previously considered, will be introduced. The design and installation features of the hydrogen recombiner are designed so as to preclude the compromising of containment integrity or other safety features.

(3) No safety margins have been reduced. The hydrogen recombiner is a post-accident system that is not operated under normal conditions and thus is not involved in consideration of any safety margin.

Based upon the above, we conclude that plant modifications needed for installation of the hydrogen recombiner(s), and associated Technical Specifications, do not involve any unreviewed safety questions with regard to the criteria of 10CFR Part 50, Section 50.59(a)(2).

11.2.5 Containment Isolation Modifications

Introduction

Section 6 of IE Bulletin 70-05A required licensees of operating B&W facilities to, "Review the containment isolation design and procedures, and prepare and implement all changes necessary to cause containment isolation of all lines whose isolation does not degrade core cooling capability upon automatic initiation of safety injection." Section 2.1.4 of "TMI-2 Lessons Learned Task Force Status

*The recombiner cooling air is vented directly to the environment. An evaluation involving failure of this cooling air system indicates that the resulting off-site doses are not significant (see Question 91, Supplement; Part 2, Restart Report)

Report and Short-Term Recommendations," NUREG-0578, July 1979, expanded the requirements of I&E Bulletin 79-05A, Section 6, as follows:

"Provide containment isolation on diverse signals in conformance with Section 6.2.4 of the Standard Review Plan, review isolation provisions for non-essential systems and revise as necessary, and modify containment isolation designs as necessary to eliminate the potential for inadvertent reopening upon reset of the isolation signal."

Evaluation

Section 2.1.1.5 of "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1" provides the details and evaluation of a redesigned containment isolation system with the following new features:

(1) Containment isolation on reactor trip,

(2) Containment isolation on 30 psig building pressure

(3) Specific line isolation on high radiation

With regard to the revised containment isolation design, this design meets the NRC's requirements in that:

(1) The system initiates automatically on safety injection (IE Bulletin 79-05A) - The reactor trip signal is utilized to obtain a diverse isolation signal. Since the RPS trips the reactor on low pressure (1800 psig)* prior to the safety injection signal (1600 psig), an RPS trip signal on low pressure will always preceed a safety injection signal. The reactor trip signal, therefore, isolates the containment more quickly than a safety injection signal.

(2) The system is diverse (NUREG-0578) - The redesigned containment isolation system provides containment isolation on the following signals:

    (a) Reactor trip

    (b) High radiation (individual line isolation)

    (c) Pipe break (individual line islolation)

    (d) The 1600 psig safety features actuation signal

    (e) The 30 psig containment signal

    (f) The 4 psig containment signal (to be eventually removed)

*Section 11.2.12 herein proposes an increase in the low pressure trip setpoint from 1800 psig to 1900 psig.

(3) Following isolation, lines should not be vulnerable to inadverten reopening (NUREG-0578). Overriding the containment isolation signal does not open the containment isolation valves, deliberate operator action is required to reopen selected individual valves.

Draft Technical Specifications, described herein, provide Limiting Conditions for Operation and Surveillance Requirements for the additional containment isolation functions. Limiting Conditions for Operation for containment isolation on the RPS Trip and the 30 psig containment pressure have been incorporated into TMI-1 draft Technical Specification 3.5.1 1 (Items 3.c and 3.d of Table 3.5-1). The minimum channel operability for containment isolation on RPS trip, and on Reactor Building 30 psig, have been chosen to be the same as the existing containment isolation functions; this would require a minimum of two channels to be operable or place the reactor in hot shutdown. With regard to Surveillance Requirements, surveillance for containment isolation on RPS trip, and on Reactor Building 30 psig, have been incorporated into TMI-1 draft Technical Specification 4.1.1 (Items 19.c and 19.d of Table 4.1-1) and chosen to be the same as the existing containment isolation system surveillance requirements the Reactor Building 4 psig signal; this requires a channel check each shift, testing each month, and calibration each refueling period.* A surveillance requirement for the manual containment isolation function has also been included (Item 19.b of Table 4.1-1) requiring a check each shift and a monthly test. Surveillance Requiements for line isolation on high radiation are presently provided for in Technical Specification 4.1.1 (Item 28, "Radiation Monitoring Systems," Table 4.1-1). The "check" and "test" surveillances are required to be performed only when containment integrity is required. This provision deletes surveillance requirements during extended outages when containment isolation may not be needed.

Conclusion

In conclusion, with regard to the revised containment isolation design and associated Technical Specifications:

(1) The probability or consequences of accidents previously evaluated have not increased. The increased diversity of the containment isolation signals increases the probability and timeliness of containment isolation.

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*This containment isolation function is not calibrated since no analog function is involved.

(2) No accidents, other than those previously considered, will be introduced. The revised containment isolation design does not in any way hamper the function of systems designed to mitigate the consequences of postulated accidents. Supurious initiation of any of the additional containment isolation signals would not isolate any components that would not also be isolated by a spurious initiation of the existing 4 psig building pressure signal.

(3) No safety margins have been reduced. The plant safety features required to mitigate the consequences of postulated transients and accidents are not impacted by the revised containment isolation design.

Based upon the above, we conclude that the modifications associated with the revised containment isolation design, and associated Technical Specifications, do not involve any unreviewed safety questions with regard to the criteria of 10CFR Part 50, Section 50.59(a)(2).

11.2.6 Instrumentation to Detect Inadequate Core Cooling

Introduction

Section 2.1.3b of Appendix A to "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," NUREG-0578, July 1979, requires that:

> "...each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition (SIC). Operator instruction as to use of this meter shall include consideration that is (SIC) not to be used exclusive of other related plant parameters."

Section 2.1.1.6 of "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1" contains a description of a saturation margin meter which is proposed for installation at TMI-1.

Evaluation

The saturation margin meter will display, in the control room, the margin between the actual primary plant temperature (T_H) and the saturation temperature (T_{sat}) for the existing plant pressure.

The T_{sat} will be computed using primary system pressure measurements and compared to the wide range T_H instrument reading. The temperature margin will be displayed in the control room. An alarm will be initiated if the margin falls below a pre-set value. Redundancy will be provided by computing T_{sat} margin independently

for each reactor coolant loop. The lower temperature for each loop will automatically be selected for the computations. Saturation pressure margin is also computed in a similar manner so that the operator has the option of displaying the saturation margins in terms of temperature or pressure. The equipment used for these computations will be safety grade and seismically qualified. In addition, the plant computer, using the same inputs, will independently compute P_{sat} and T_{sat} margin for logging, trending, and alarm.

Draft Technical Specifications Limiting Conditions for Operation and Surveillance Requirements are presented, herein for the saturation margin meter. Draft TMI-1 Technical Specification 3.5.6, "Saturation Margin Meter," requires the saturation margin meter and alarm to be operable during start-up and power operation. If the saturation margin meter is not operable, the reactor operator is to have a procedure available for calculation of saturation temperature. This remedial action is appropriate since (1) no automatic actuations of safety features are associated with the saturation meter and (2) saturation temperature is easily calculated using reactor coolant system measurements and "steam tables." Surveillance Requirements for the saturation margin meter are incorporated in TMI-1 draft Technical Specification 4.1.1 (Item 49 of Table 4.1-1). The proposed surveillance requires the saturation margin meter to be checked each shift, tested monthly, and calibrated each refueling period. The "check" surveillance is only required when T_{AVG} is above 200° such that this requirement is deleted during extended outages when the saturation margin meter is not needed. The proposed surveillance schedule is consistent with surveillance schedules for other safety grade instrumentation at TMI-1 and is sufficient to assure reliable performance from the saturation meter.

Conclusion

In conclusion, with regard to the saturation margin meter and associated Technical Specifications:

(1) The probability or consequences of accidents previously evaluated have not increased. The saturation margin meter is not required for the prevention or mitigation of accidents, or transients, previously considered.

(2) No accidents, other than those previously considered, will be introduced. No automatic actuations of safety features are associated with the saturation margin meter nor is the saturation margin meter capable of effecting any safety features.

(3) No safety margins have been reduced. The saturation margin meter is not associated with any safety margins; both low pressure and high temperature RPS trips protect the reactor's thermal margins. Based upon the above, we conclude that the installation and use of the saturation margin meter, and the associated Technical Specifications, do not involve any unreviewed safety questions with regard to the criteria of 10CFR Part 50, Section 50.59(a)(2).

11.2.7 Emergency Feedwater System Modifications

Introduction

By letter dated June 28, 1979, Met-Ed presented NRC with recommendations for modifications to TMI-1 which would be completed prior to restart of TMI-1. The June 28, 1979 letter recommended the following changes to the emergency feedwater system and associated procedures:

1. Automatic initiation of the motor driven AFW pumps upon loss of both feedwater pumps or loss of four (4) Reactor Coolant Pumps.

2. Modification of the AFW control valves such that they fail open on loss of control air.

3. Automatic block loading of the motor driven AFW pumps on the emergency diesel-generators.

4. Incorporation of AFW in the TMI-1 Technical Specifications as specified in IE Bulletin 79-05A, item 8. Verification that Technical Specification requirements of AFW capacity are in accordance with the accident analysis will be conducted.

5. Provide indication in the control room of AFW flow to each Steam Generator.

6. Provide procedures and training to assure that AFW is available and properly applied when required. Procedures will identify the need to verify proper operation when AFW is initiated.

7. To assure that AFW will be aligned in a timely manner to inject on all AFW demand events when in the surveillance test mode, procedures will be implemented and training conducted to provide an operator at the necessary location in communications with the control room during the surveillance mode to carry out alignment changes necessary upon AFW demand events.

8. Design review and modifications, as necessary, will be conducted to provide control room annunciation for all auto start conditions of the AFW system.

On August 9, 1979 the NRC issued an "Order and Notice of Hearing" which addressed modifications to the TMI-1 facility. With regard to those changes proposed for the emergency feedwater system in the June 28, 1979 letter, the August 9, 1979 Order directed that these changes should be made. A description and evaluation of changes to the emergency feedwater system involving equipment modifications (items 1,2,3,5 and 8, as described above) are contained in Section 2.1.1.7 of "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1." Draft Technical Specifications for the modified emergency feedwater system are discussed herein.

Draft TMI-1 Technical Specification 3.4 provides Limiting Conditions for Operation for the emergency feedwater system. Guidance on operability of the emergency feedwater System is contained in IE Bulletin 79-05A, Item 8, as follows:

> "Prepare and implement immediately procedures which assure that two independent steam generator auxiliary feedwater flow paths, each with 100% flow capacity, are operable at any time when heat removal from the primary system is through the steam generators. When two independent 100% capacity flow paths are not available, the capacity shall be restored within 72 hours or the plant shall be placed in a cooling mode which does not rely on steam generators for cooling within the next 12 hours.
>
> When at least one 100% capacity flow path is not available, the reactor shall be made subcritical within one hour and the facility placed in a shutdown cooling mode which does not rely on steam generators for cooling within 12 hours or at the maximum safe shutdown rate."

The guidance contained in IE Bulletin 79-05A has been incorporated in Draft TMI-1 Technical Specification 3.4.1 with the exception that the restoration time for the emergency feedwater system has been reduced from 72 hours to 48 hours as a result of subsequent requirements from the NRC. Existing Technical Specifications 3.4.3 and 3.4.6 have been rewritten to incorporate remedial action in the event that the condensate storage tanks and/or the main steam safety valves are inoperable. For both the condensate storage tank and the main steam safety valves, remedial action has been proposed that is consistent with NRC guidance as reflected in the B&W Standard Technical Specifications.

With regard to surveillence requirements, draft Technical Specification 4.9, "Emergency Feedwater System," has been modified as follows:

(1) Existing Technical Specification 4.9.1.1 which requires testing of the emergency feedwater pumps every three months, as modified, would require pump testing every 31 days and also require verification of specific pump flow values during the testing. The flow testing would be based the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Article IWP-3220, and would confirm that the emergency feedwater system can deliver at least 500 gpm to either steam generator flow path.

(2) Draft Technical Specification 4.9.1.2 would require valve lineup verification for valves in the flow paths of the emergency feedwater system, every 31 days.

(3) Draft Technical Specification 4.9.1.3 would require a test, each 18 months, of automatic pump start logic and automatic valve lineup following an emergency feedwater actuation signal. In addition, the operability of the manual control valve station would be verified.

(4) Item 10F of NRC's October 26, 1979 letter to Mr. R. C. Arnold requires a, "...Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary EFW system water source to the steam generators. The flow test should be conducted with EFW system valves in their normal alignment." This test is incorporated in draft Technical Specification 4.9.1.5 where the term "extended cold shutdown" is interpreted as "a cold shutdown of longer than 30 days' duration."

(5) Existing Technical Specification 4.1.2 (Table 4.1-2), would require a functional test of the Backup Instrument Air Supply System (backup air supply for the emergency feedwater control valves), every refueling period.

(6) Existing Technical Specification 4.1.1 (Item 50 in Table 4.1-1), as modified, would require a check each shift, monthly testing, and calibration each refueling period, for the emergency feedwater flow instrumentation. The "check" and "test" surveillances would not be required when T_{AVG} is less than 200°F since the reactor would be shutdown and this safety function not needed.

(7) Existing Technical Specification 4.5.1.1, as modified would incorporate the motor driven feedwater pumps into the list of equipment whose operation is verified during the testing of the emergency diesel generators. In this case, only operation of the interlock would be verified since the pumps do not actually start on loss of AC power (the actual start signal is on loss of main feedwater or loss of reactor coolant pumps.)

Conclusion

In conclusion, with regard to the modifictions to the emergency feedwater systems and associated Technical Specifications:

(1) The probability or consequences of accidents previously evaluated have not increased. The more conservative Limiting Condition for Operation and Surveillance Requirements for the emergency feedwater system provide increased assurance that the system will operate properly, when required.

(2) No accidents, other than those previously considered, will be introduced. The modifications to the emergency feedwater system could only effect the non-operation or spurious operation of the system; both of these conditions have been previously evaluated. The only aspect of the emergency feedwater system modification with the potential for effecting other systems involves the loading of the motor driven feedwater pumps on the emergency diesel generators. An analysis of the diesel generator loading indicates that the

maximum load, with the emergency feedwater pumps is 2817 Kw compared to the 2000 hour rating of 3000 Kw. The proper diesel generator loading sequence with the emergency feedwater pumps will be verified prior to startup and every 18 months thereafter. Other aspects of the emergency feedwater system will be tested prior to startup, and periodically thereafter.

(3) No safety margins have been reduced. The modifications to the emergency feedwater system did not involve any changes which resulted in a decrease in capacity of this system to perform its designed function.

Based upon the above, we conclude that the modifications to the emergency feedwater system, and associated Technical Specifications, do not involve any unreviewed safety questions with regard to the criteria of 10CFR Part 50, Section 50.59(a)(2).

11.2.8 Post Accident Monitoring

Introduction

Section 2.1.8 of, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," NUREG-0578, July 1979 makes the following recommendations with regard to the increased range of radiation monitors:

"Provide high range radiation monitors for noble gases in plant effluent lines and a high-range radiation monitor in the containment. Provide instrumentation for moitoring effluent releases lines capable of measuring and identifying radioiodine and particulate radioactive effluents under accident conditions." In addition, the NRC recommended that facilities "Provide instrumentation for accurately determining in-plant airborne radioiodine concentrations to minimize the need for unnecessary use of respiratory equipment. In an August 13, 1979 ACRS memorandum to the NRC, the ACRS recommended the following additional post-accident instrumentation: (1) containment pressure, (2) containment water level, and (3) on-line monitoring of hydrogen concentration in the containment.

The post-accident monitoring instruments to be installed at TMI-1 are responsive to the recommendations of the NRC and the ACRS.

Evaluation

Section 2.1.2.1 of "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1" describes the post-accident monitoring instrumentation to be installed at TMI-1. The post-accident instrumentation, in conformance with Regulatory Guide 1.97, consists of the following:

(1) Containment Pressure - the range will be - 5 psig to three times the containment design pressure;

(2) Containment Water Level - a narrow range monitor will measure containment sump level while the wide range monitor will measure from the bottom of the containment to a 10 ft. level;

(3) Containment Hydrogen Indication - continuous reading of the concentration of hydrogen in the containment, from 0 to 10%, will be available in the control room;

(4) High Range Containment Radiation Monitor - two monitors with a range to 10^7R/hr will be provided;

(5) High Range Effluent Monitors:

(a) Undiluted Containment Exhaust - 10^5 Ci/cc

(b) Diluted Containment Exhaust - 10^4 Ci/cc

(c) Auxiliary and Fuel Handling Building Exhaust 10^3 Ci/cc

(d) Condenser Off Gas - 10^2 Ci/cc

(e) High Range Effluent Radio Iodine & Particulate Sampling and Analysis - silver zeolite cartridges.

Although the above instrumentation does not actuate safety equipment, nor is it required by safety analyses, it is appropriate to provide Surveillance Requirements to assure reliable post-accident performance of the instrumentation. Surveillance Requirements for post-accident monitoring instrumentation is incorporated into TMI-1 Draft Technical Specification 4.1.1 (Table 4.1-1) as follows:

(1) Item 13 of Table 4.1-1, "High Reactor Building Pressure," is provided with a footnote to include the post-accident instrumentation in the existing containment pressure instrumentation surveillance program;

(2) Item 28 of Table 4.1-1, "Radiation Monitoring Systems," is provided with a footnote to include the post-accident instrumentation, described in item (5)(a) thru (5)(d) above, in the existing radiation monitor system surveillance program;

(3) Item 37 of Table 4.1-1 "Reactor Building Sump Level" has been changed to "Reactor Building Sump and Containment Level." A foot-note has also been added to include the post-accident intrumentation in the sump level instrument surveillance program.

(4) A new item 52, "Reactor Building Hydrogen Concentration," has been added to address Surveillance Requirements for the reactor building hydrogen concentration instrumentation. The "check" and "test" surveillance need not be performed when T_{AVG} is less than 200°F since the reactor would be shutdown and this safety function not needed.

Conclusion

With regard to TMI-1 post-accident monitoring instrumentation, and associated Technical Specifications, since the instrumentation does not actuate safety equipment, nor is it required by the safety analysis:

(1) The probability or consequences of accidents previously evaluated have not increased.

(2) No accidents of a type, not previously evaluated, will occur, and

(3) No safety margins have been reduced.

Based upon the above, we conclude that the post-accident monitoring instrumentation, and associated Technical Specifications, do not involve any unreviewed safety issues with regard to the criteria of 10CFR, Part 50, Section 50.59 (a)(2).

11.2.9 Reactor Coolant Pump Trip on Coincident ESFAS and Coolant Voiding

Introduction

The IE Bulletin Nos. 79-05C and 79-06C, July 26, 1979 states that, "Recent preliminary calculations performed by Babcock & Wilcox, Westinghouse and Combustion Engineering indicate that, for a certain spectrum of small breaks in the reactor coolant system, continued operation of the RCPs can increase the mass lost through the break and prolong or aggravate the uncovering of the reactor core.

The damage to the reactor core at TMI-2 followed tripping of the last operating RCP, when two phase fluid was being pumped through the reactor coolant system. It is our current understanding that all three of the nuclear steam system suppliers for PWRs now agree that an acceptable action under LOCA symptoms is to trip all operating RCPs immediately, before significant voiding in the reactor coolant system occurs."

With regard to reactor coolant pump trip, 1E Bulletin Nos. 79-05C and 79-06C recommends the following long-term action:

> "Propose and submit a design which will assure automatic tripping of the operating RCPs under all circumstances in which this action may be needed."

Section 2.1.2.5 of "Report in Response to NRC Staff Recommended Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1" contains a description of the reactor coolant pump trip that is proposed for TMI-1.

Evaluation

The logic for the reactor coolant pump trip receives inputs from the High Pressure Injection (HPI) signal from the ESFAS, and redundant pump current sensors from each of four reactor coolant pumps. The pump trip will occur on concurrent HPI and low current in two of the four reactor coolant pumps. Non-operating reactor coolant pumps have effectively tripped current sensors. With only two reactor coolant pumps operating, therefore, these pumps will trip on HPI. The output of the trip logic provides a trip signal to each reactor coolant pump. Limiting Conditions for Operation and Surviellance Requirements for the Reactor Coolant Pump Trip are addressed below.

Limiting Conditions for Operation for the Reactor Coolant Pump Trip are presented in TMI-1 draft Technical Specification 3.5.7. The draft Technical Specification requires that an HPI actuation channel and one pump current channel from each operable pump be available to the reactor coolant pump trip logic. In the event that the reactor coolant pump trip is inoperable, hot shutdown must be achieved within 24 hours. The draft Technical Specification only allows continued reactor operation if the pump trip is in a condition which assures reliable operation. The remedial action is specified to allow a reasonable time to restore the reactor coolant pump trip to operability or achieve an orderly shutdown. The Surveillance Requirement for the reactor coolant pump trip is contained in TMI-1 draft Technical Specification 4.1.1 (Table 4.1-1). A new item, number 51, proposes a pump trip channel check each shift, a test each month, and a calibration each refueling period. The draft Surveillance Requirement for the pump trip is consistent with the surveillance for other safety instrumentation channels. The "check" and "test" surveillances need not be performed when T_{AVG} is less than 200°F since the reactor is shut down and this safety function is not needed.

Conclusion

With regard to the reactor coolant pump trip, the logic is designed to provide high assurance that the reactor coolant pumps will be triped when required. Any single failure within the reactor coolant pump trip logic will result in only a single reactor coolant pump being tripped. The draft Limiting Condition for Operation for the reactor coolant pump trip prevents extended reactor operation if the reactor coolant pump trip is significantly degraded. The draft Surveillance Requirement for the reactor coolant pump trip provides assurance of reliable operation.

11.2.10 TMI-1/TMI-2 Separation

Introduction

Item II.4 of the NRC's August 9, 1979 "Order and Notice of Hearing," requires that, "The licensee shall demonstrate that decontamination and/or restoration operations at TMI-2 will not affect safe operations at TMI-1. The licensee shall provide separation and/or isolation of TMI 1/2 radioactive liquid transfer lines. Fuel handling areas, ventilation systems, and sampling lines. Effluent monitoring instruments shall have the capability of discriminating between effluents resulting from Unit 1 or Unit 2 operations."

Section 7.2 of "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1" describes a plan to separate TMI-1/TMI-2 interfaces that have the potential of transferring significant quantities of contamination as a result of restoration activities at TMI-2.

Evaluation

The two major pathways for potential transfer of contamination from TMI-2 to TMI-1 are the waste management interconnections and the common air space of the Fuel Handling Building. The following TMI-1/TMI-2 waste management interfaces have been identified:

(1) Unit 2 Reactor Coolant Bleed Holdup Tank - Unit 1 Reactor Coolant Waste Evaporator.

(2) Unit 1 Miscellaneous Waste Evaporator - Unit 2 Evaporator Condensate Test Tank.

(3) Unit 2 Nuetralizer Tanks, Contaminated Drain Tanks, Reactor Coolant Bleed Holdup Tanks, Auxiliary Building Sump Tanks and Miscellaneous Waste Holdup Tanks - Unit 1 Liquid Waste Disposal System.

(4) Unit 1 Evaporator Concentrate - Unit 2 Evaporator Concentrate.

(5) Unit 1 Spent Ion Exchange Resin - Unit 2 Spent Ion Exchange Resin.

Draft TMI-1 Technical Specification 4.1.2 (Table 4.1-2, Item 13) requires the isolation devices (valves, blank flanges, etc.) on the above tielines to be verified to be isolated, by visual inspection, on a monthly basis. Draft TMI-1 Technical Specification 3.19 requires that, if an isolation device is found to be open without prior NRC authorization, a "Thirty Day Written Report" must be prepared per TMI-1 draft Technical Specification 6.9.2.B(5). In addition, TMI-1 draft Technical Specification 3.19.2 requires NRC approval prior to creation of additional TMI-1/TMI-2 system interties that can transfer potentially significant quantities of contamination.

With regard to the separation of the air space in the Fuel Handling Building, the details of this modification have not been finalized. Additional evaluations and preparation of draft Technical Specifications will be undertaken, if appropriate, following finalization of the design details of the Fuel Handling Building isolation system.

Conclusion

The draft TMI-1 Technical Specifications 3.19 and 4.1.2 for the TMI-1/TMI-2 interties provide assurance that:

(1) System intenties that could potentially transfer significant quantities of contamination from TMI-1 to TMI-2 will remain closed.

(2) If permission is received from the NRC to open system interties, these interties will be used in accordance with plant procedures.

(3) No new system interties, with the potential for transferring significant quantities of contamination from TMI-2 to TMI-1, will be created without prior NRC approval.

The above controls limit releases from TMI-1 to materials under control at TMI-1 and thus to previously evaluated quantities and concentrations of contamination.

11.2.11 Low Reactor Coolant System Pressure Channel for HPI/LPI Initiation

Introduction

The Low Reactor Coolant System Pressure Channel setpoint, which is used as input to the ESFAS logic, is determined based on a generic LOCA analysis. The generic LOCA analysis for TMI, referenced as "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS," BAW-10103, has referenced the Low Reactor Coolant System Pressure setpoint as 1600 psig compared with the Technical Specification value of 1500 psig. The setpoint actually used in the BAW-10103 calculations, however, was 1350 psig.

Evaluation

The TMI-1 Technical Specification 3.5.3.1, "Engineered Safeguards Protection System Actuation Setpoints," requires the Low Reactor Coolant System Pressure HPI/LPI initiation setpoint to be $\geq$ 1500 psig. Draft TMI-1 Technical Specification 3.5.3.1 would require the Low Reactor Coolant System Pressure HPI/LPI initiation setpoint to be raised to $\geq$ 1600 psig. In the event of a LOCA, the only impact of the 100 psig increase in the minimum Low Reactor Coolant System Pressure setpoint would be to initiate actions, based on this signal, at an earlier time in the accident (e.g., in conjunction with the 4 psig High Reactor Building Pressure, both HPI and LPCI pumps would start earlier in the accident.)

Conclusion

With regard to the 100 psig increase in the minimum Low Reactor Coolant System Pressure HPI/LPI initiation setpoint:

(1) The probability or consequences of accidents previously evaluated have not increased. The potential initiation of engineered safety feature equipment at an earlier time in a LOCA is not expected to have a significant impact on peak clad temperature and other LOCA limits (any changes would be expected to be in direction of a less severe accident).

(2) No accidents of a type not previously evaluated will occur. The proposed change in the Low Reactor Coolant System Pressure Setpoint would have only a small impact on the severity of the LOCA, in the conservative direction, rather than change the nature of the accident.

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(3) No safety margins have been reduced. The applicable LOCA calculations continue to be those for which the Low Reactor Coolant System Pressure HPI/LPI initiation setpoint is 1350 psig; operationally, raising the minimum setpoint to 1600 psig would slightly increase the LOCA margins.

Based upon the above, we conclude that raising the minimum Low Reactor Coolant Pressure setpoint from 1500 psig to 1600 psig does not involve any unreviewed safety questions with regard to the criteria of 10CFR Part 50, Section 50.59(a)(2).

11.2.12 Raising the Reactor Protection System (RPS) Trip Setpoint from 1800 psig to 1900 psig

Introduction

The TMI-1 Technical Specification 2.3.1 (Table 2.3-1, Figure 2.3-1) provides a value of 1800 psig for the RPS Low Reactor Coolant Pressure trip setpoint. The B&W generic ECCS analysis, "ECCS Analysis of B&W's 177-FA Lowered Loop NSS," BAW-10103, Rev. 2, April 1976, referenced a value of 1900 psig for the Low Reactor Coolant Pressure Trip setpoint. Draft Technical Specification 2.3.1 would increase the Low Reactor Coolant System Pressure Trip Setpoint from 1800 psig to 1900 psig.

Evaluation

The principal reason for the Low Reactor Coolant System trip setpoint is to maintain thermal margins for the fuel by preventing the minimum DNB ratio from decreasing below the safety limit of 1.3; the transient analysis for TMI-1 is based on an 1800 psig Low Reactor Coolant System trip setpoint. The Low Reactor Coolant System trip setpoint is also credited in the ECCS analysis since a reactor trip is part of the assumed LOCA scenario.

By increasing the Low Reactor Coolant System Pressure setpoint from 1800 psig to 1900 psig, the reactor would trip earlier in the LOCA scenario and thus the decay heat would be slightly less when the ECCS functions. Increasing the Low Reactor Coolant System trip setpoint also has the effect of increasing the margin to DNB following a trip on low pressure; the reactor would trip earlier on low pressure and thus the final minimum DNB would be higher (more conservative) than if the reactor tripped at 1800 psig.

Conclusion

With regard to increasing the Low Reactor Coolant System trip setpoint from 1800 psig to 1900 psig:

(1) The probability or consequences of accidents previously considered have not increased. For any accident that involves a pressure decrease, the reactor will trip earlier in the transient and thus the result of the accident will be more conservative.

(2) No accident of a type not previously evaluated, will occur. The increasing of the Low Reactor Coolant System trip setpoint will not have any effect other than tripping the reactor at an earlier time in pressure reduction transients.

(3) No safety margins have been decreased. It is expected that for pressure reduction transients, DNB following the reactor trip will be higher (more conservative) and for the LOCA, the peak clad temperature and other system parameters will be more favorable.

Based upon the above, we conclude that increasing the Low Reactor Coolant System trip setpoint from 1800 psig to 1900 psig does not involve unreviewed safety questions with regard to the criteria of 10CFR Part 50, Section 50.59(a)(2).

11A. DRAFT TECHNICAL SPECIFICATIONS

The following sections contain the draft TMI-1 Technical Specifications that are referenced in Section 11.

Draft Technical

Specifications Corresponding

to Section 11.2.1

TABLE 3.5-1 Continued

INSTRUMENTS OPERATING CONDITIONS

Functional Units	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A Cannot be Met
Reactor Protection System			
8. Reactor coolant pressure			
a. High reactor coolant pressure instrument channels	2	1	Maintain hot shutdown
b. Low reactor coolant pressure instrument channels	2	1	Maintain hot shutdown
9. Power/number of pumps instrument channels	2	1	Maintain hot shutdown
10. High reactor building pressure channels	2	1	Maintain hot shutdown
Other Reactor Trips			
1. Loss of Feedwater	2 [1]	1 [1]	Maintain hot shutdown [1]
2. Turbine Trip	2 [1]	1 [2]	Maintain hot shutdown [2]

1. The main turbine trip bypass may be placed in effect when reactor power is less than 20%. The bypass will be removed when the reactor power is increased above 20%.

2. Bypass of the feedwater pump trip signal may be placed in effect when reactor power is less than 10%. The bypass will be removed when reactor power is raised above 10%.

reactor coolant temperature instrument channels, four reactor coolant flow instrument channels, four reactor coolant pressure instrument channels, four pressure-temperature instrument channels, four flux-imbalance flow instrument channels, four power-number of pumps instrument channels, and four high reactor building pressure instrument channels. The reactor trip, on loss of feedwater, may be bypassed below 10% reactor power. The reactor trip, on turbine trip, may be bypassed below 20% reactor power. The safety features actuation system must have two analog channels functioning correctly prior to startup.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column "B" (Table 3.5-1). This is in agreement with redundancy and single failure criteria of IEEE 279 as described in FSAR Section 7.

There are four reactor protection channels. Normal trip logic is two out of four. Required trip logic for the power range instrumentation channels is two out of three. Minimum trip logic on other instrumentation channels is one out of two.

The four reactor protection channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided alarm and lights to indicate when that channel is by-passed. There will be one reactor protection system bypass switch key permitted in the control room.

Each reactor protection channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypas switch is being used.

Power is normally supplied to the control rod drive mechanisms from two separate parallel 460 volt sources. Redundant trip devices are employed in each of these sources. If any one of these trip devices fails in the untripped state on-line repairs to the failed device, when practical, will be made, and the remaining trip devices will be tested. Eight hours is ample time to test the remaining trip devices and in many cases make on-line repairs.

REFERENCE

FSAR, Section 7.1

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE		REMARKS
38.	Steam Generator Water Level	W	NA	R		
39.	Turbine Overspeed Trip	NA	R	NA		
40.	Sodium Thiosulfate Tank Level Indicator	NA	NA	R		
41.	Sodium Hydroxide Tank Level Indicator	NA	NA	R		
42.	Diesel Generator Protective Relaying	NA	N	R		
43.	4 KV ES Bus Undervoltage Relays (Diesel Start)	NA	M(1)	R	(1)	Relay operation will be checked by local test pushbuttons.
44.	Reactor Coolant Pressure DH Valve Interlock Bistable	S(1)	M	R	(1)	When reactor coolant system is pressurized above 300 psig or Taves is greater than 200°F.
45.	Loss of Feedwater Trip	S(1)	M(1)	R	(1)	When reactor >10% power
46.	Turbine Trip/Reactor Trip	S(1)	M(1)	R	(1)	When reactor >20% power

S - Each Shift

D - Daily

W - Weekly

M - Monthly

T/W - Twice per week

B/M - Every 2 months

Q - Quarterly

P - Prior to each startup if not done previous week

R - Each Refueling Period

NA - Not Applicable

B/W - Every two weeks

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at the intervals of each refueling period.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies set forth are considered acceptable.

Testing

On-line testing of reactor protection channels is required once every four weeks on a rotational or perfectly staggered basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel.

The rotation schedule for the reactor protection channels is as follows:

Channels A, B, C, & D	Before Startup, when shutdown greater than 24 hours
Channel A	One Week After Startup
Channel B	Two Weeks After Startup
Channel C	Three Weeks After Startup
Channel D	Four Weeks After Startup

The reactor protection system instrumentation test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action in a channel, the instrumentation associated with the protection parameter failure will be tested in the remaining channels. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protection channels coincidence logic and control rod drive trip breakers are trip tested every four weeks. The trip test checks all logic combinations and is to be performed on a rotational basis. The logic and breakers of the four protection channels shall be trip tested prior to startup when the reactor has been shutdown for greater than 24 hours. Discovery of a failure that prevents trip action requires the testing of the instrumentation associated with the protection parameter failure in the remaining channels.

For purposes of surveillance, reactor trip on loss of feedwater and reactor trip on turbine trip are considered reactor protection system channels.

The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the equipment and systems in a safe operational status.

REFERENCE

(1) FSAR, Section 7.1.2.3.4

Draft Technical

Specifications Corresponding

to Section 11.2.2

3. LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

3.1.1 Operational Components

Applicability

Applies to the operating status of reactor coolant system components.

Objective

To specify those limiting conditions for operation of reactor coolant system components which must be met to ensure safe reactor operations.

Specification

3.1.1.1 Reactor Coolant Pumps

a. Pump combinations permissible for given power levels shall be as shown in Specification Table 2.3.1.

b. Power operation with one idle reactor coolant pump in each loop shall be restricted to 24 hours. If the reactor is not returned to an acceptable RC pump operating combination at the end of the 24-hour period, the reactor shall be in a hot shutdown condition within the next 12 hours.

c. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant.

3.1.1.2 Steam Generator

a. Both steam generators shall be operable whenever the reactor coolant average temperature is above 250°F.

3.1.1.3 Pressurizer Safety Valves

a. The reactor shall not remain critical unless both pressurizer code safety valves are operable with a lift setting of 2500 psig ±1%.

b. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III.

3.1.1.4 Pressurizer Electromatic Relief Valve

a. The setpoint for the pressurizer Electromatic relief valve shall be 2450 psig ±1% when reactor coolant system temperature is greater than 275°F.

b. The setpoint for the pressurizer Electromatic relief valve shall be reset within one hour to 485 psig ±1% when reactor coolant system temperature is less than 275°F.

Bases

The limitation on power operation with one idle RC pump in each loop has been imposed since the ECCS cooling performance has not been calculated in accordance with the Final Acceptance Criteria requirements specifically for this mode of reactor operation. A time period of 24 hours is allowed for operation with one idle RC pump in each loop to effect repairs of the idle pump(s) and to return the reactor to an acceptable combination of operating RC pumps. The 24 hours for this mode of operation is acceptable since this mode is expected to have considerable margin for the peak cladding temperature limit and since the likelihood of a LOCA within the 24-hour period is considered very remote.

A reactor coolant pump or decay heat removal pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One decay heat removal pump will circulate the equivalent of the reactor coolant system volume in one-half hour or less.

The decay heat removal system suction piping is designed for 300°F and 370 psig; thus, the system can remove decay heat when the reactor coolant system is below this temperature. (2,3)

Both steam generators must be operable before heatup of the Reactor Coolant System to insure system integrity against leakage under normal and transient conditions. Only one steam generator is required for decay heat removal purposes.

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (4) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal or feedwater line break accidents. (5) The pressurizer code safety valve lift setpoint shall be set at 2500 psig $\pm$1% allowance for error and each valve shall be capable of relieving 280,800 lb/h of saturated steam at a pressure not greater than three percent above the set pressure.

The purpose of the pressurizer electromatic relief valve is to reduce the challenge rate of the code safety valves. No credit is taken for the pressurizer relief valve in the design overpressure transient. A duel setpoint is specified; the lower setpoint is associated with considerations relating to the potential for overpressurization of the reactor coolant system under cold conditions.

References

(1) FSAR, Tables 9-10 and 4-3 through 4-7
(2) FSAR, Sections 4.2.5.1 and 9.5.2.3
(3) FSAR, Section 4.2.5.4
(4) FSAR, Sections 4.3.10.4 and 4.2.4
(5) FSAR, Section 4.3.7

3.5.5 Pressurizer Safety and Relief Valve Monitors

Applicability

Applies to the operability requirements for the delta-pressure and acoustic instrumentation that monitors the status of the pressurizer code safety valves and the pressurizer Electromatic relief valve.

Objective

To provide reactor operators with a reliable means of detecting and monitoring pressurizer safety and relief valve discharge flow.

Specification

3.5.5.1 The three (3) delta-pressure monitors, one for each of the two (2) pressurizer code safety valves and one for the pressurizer Electromatic relief valve, shall be operable during STARTUP and power OPERATION. If one or more of the delta-pressure monitors in inoperable, it shall be returned to operation prior to startup following the next cold shutdown.

3.5.5.2 An acoustic monitor for the pressurizer Electromatic relief valve shall be operable during STARTUP and power OPERATION. If the acoustic monitor is inoperable, it shall be returned to operation prior to startup following the next cold shutdown.

Bases

Discharge flow from the two (2) pressurizer code safety valves and the Electromatic relief valve is measured by differential pressure transmitters connected across elbow taps downstream of each valve. A delta-pressure indication from each pressure transmitter is available in the control room to indicate safety or relief valve line flow. An alarm is also provided in the control room to indicate that discharge from a pressurizer safety or relief valve is occuring. In addition, an acoustic monitor is provided to detect flow in the relief valve discharge line. An alarm and a flow indication is provided in the control room for the acoustic monitor.

In the event that a delta-pressure monitor or the acoustic monitor becomes inoperable, access to the containment would most likely be required; however, a reactor shutdown to allow containment access for this repair is not justifiable due to the existence of alternate means of detecting and monitoring safety or relief valve discharge flow. The following indications are available to the reactor operator to monitor safety or relief valve discharge flow:

(1) Reactor coolant drain tank level, pressure, and temperature

(2) Safety and relief valve tailpipe temperatures

Based upon the existence of these means of monitoring safety and relief valve discharge flow, continued operation until the next cold shutdown is acceptable in the event that a delta-pressure or acoustic monitor is inoperable.

4. SURVEILLANCE STANDARDS

Specified intervals may be adjusted plus or minus 25 percent to accommodate normal test schedules.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specificaion

4.1.1 The minimum frequency and type of surveillance required for reactor protection system and engineered safety feature protection system instrumentation when the reactor is critical shall be as stated in Table 4.1-1. Surveillances in Table 4.1-1, not performed due to reactor shutdown greater than one month, shall be performed prior to STARTUP.

4.1.2 Equipment and sampling test shall be performed as detailed in Tables 4.1-2 and 4.1-3.

Bases

Check

Failures such as blown instrument fuses, defective indicators, or faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers shall be checked and calibrated if necessary, every shift against a heat balance standard. The frequency of heat balance checks will assure that the difference between the out-of-core instrumentation and the heat balance remains less than 4%.

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE		REMARKS
38.	Steam Generator Water Level	W	NA	R		
39.	Turbine Overspeed Trip	NA	R	NA		
40.	Sodium Thiosulfate Tank Level Indicator	NA	NA	R		
41.	Sodium Hydroxide Tank Level Indicator	NA	NA	R		
42.	Diesel Generator Protective Relaying	NA	N	R		
43.	4 KV ES Bus Undervoltage Relays (Diesel Start)	NA	M(1)	R	(1)	Relay operation will be checked by local test pushbuttons.
44.	Reactor Coolant Pressure DH Valve Interlock Bistable	S(1)	M	R	(1)	When reactor coolant system is pressurized above 300 psig or Taves is greater than 200°F.
45.	Loss of Feedwater Trip	S(1)	M(1)	R	(1)	When reactor >10% power
46.	Turbine Trip/Reactor Trip	S(1)	M	R	(1)	When reactor >20% power
47.	Pressurizer Code Safety Valve and Electromatic Relief Valve delta P/flow	S(1)	R	R	(1)	When T_{AVG} is greater than 200°F
48.	Pressurizer Electromatic Relief Valve - acoustic/flow	S(1)	R	R	(1)	When T_{AVG} is greater than 200°F

S - Each Shift

D - Daily

W - Weekly

M - Monthly

T/W - Twice per week

B/M - Every 2 months

Q - Quarterly

P - Prior to each startup if not done previous week

R - Each Refueling Period

NA - Not Applicable

B/W - Every two weeks

TABLE 4.1-2

MINIMUM EQUIPMENT TEST FREQUENCY

Item	Test	Frequency
1. Control Rods	Rod drop times of all full length rods	Each refueling shutdown
2. Control Rod Movement	Movement of each rod	Every two weeks, when reactor is critical
3. Pressurizer Safety Valves	Setpoint*	50% each refueling period
4. Main Steam Safety Valves	Setpoint	25% each refueling period
5. Refueling System Interlocks	Functional	Start of each refueling period
6. Main Steam Isolation Valves	(See Section 4.8)	
7. Reactor Coolant System Leakage	Evaluate	Daily, when reactor coolant system temperature is greater than 525°F
8. Charcoal and high efficiency filters for Control Room, and RB Purge Filters	DOP test on HEPA filters, freon test on charcoal filter units	Each refueling period and at any time work on filters could alter their integrity
9. Spent Fuel Cooling System	Functional	Each refueling period prior to fuel handling
10. Intake Pump House Floor (Elevation 262 Ft. 6 in.)	(a) Silt Accumulation-Visual inspection of Intake Pump House Floor	Each refueling period
	(b) Silt Accumulation Measurement of Pump House Flow	Quarterly
11. Pressurizer Electromatic Relief Valve	Setpoint	Refueling period

*The setupoint of the pressurizer code safety valves shall be in accordance with ASME Boiler and Pressurizer Vessel Code, Section III, Article 9, Winter, 1968.

Draft Technical

Specification Corresonding

to Section 11.2.3

4.6 EMERGENCY POWER SYSTEM PERIODIC TESTS

Applicability

Applies to periodic testing and surveillance requirement of the emergency power system.

Objective

To verify that the emergency power system will respond promptly and properly when required.

Specification

The following tests and surveillance shall be performed as stated:

4.6.1 Diesel Generators

a. Manually-initiated start of the diesel generator, followed by manual synchronization with other power sources and assumption of load by the diesel generator up to the nameplate rating (3000 kw). This test will be conducted every month on each diesel generator. Normal plant operation will not be affected.

b. Automatic start of each diesel generator and restoration to operation of particular vital equipment, initiated by an actual loss of normal a-c station service power supply together with a simulated Engineered Safeguards Actuation Signal. Following input of the Engineered Safeguards Actuation Signal, it shall be verified than the circuit breakers, supplying power to the manually transferred loads for pressurizer heater Groups 8 and 9, have been tripped. This test will be conducted during reactor shutdown for refueling to assure that the diesel-generator will start assuming load in ten seconds and assume the load of all safeguards equipment listed in 4.5.1.1b within 60 seconds after the initial starting signal.

c. Each diesel generator shall be given an inspection at least annually in accordance with the manufacturer's recommendations for this class of stand-by service.

4.6.2 Station Batteries

a. The voltage, specific gravity, and liquid level of each cell will be measured and recorded monthly.

b. The voltage and specific gravity of a pilot cell will be measured and recorded weekly.

c. Each time data are recorded, new data shall be compared with old to detect signs of abuse or deterioration.

d. The battery will be subjected to a load test at a frequency not to exceed refueling periods. The battery voltage as a function of time will be monitored to establish that the battery performs as expected during this load test.

Bases

The tests specified are designed to demonstrate that one diesel-generator will provide power for operation of safeguards equipment. They also assure that the emergency generator control system and the control systems for the safeguards equipment will function automatically in the event of a loss of normal a-c station service power or upon the receipt of an an engineered safeguards Actuation Signal. The automatic tripping of manually transferred loads, on an Engineered Safeguards Actuation Signal, protects the diesel generators from a potential over-load condition. The testing frequency specified is intended to identify and permit correction of any mechanical or electrical deficiency before it can result in a system failure. The fuel oil supply, starting circuits, and controls are continuously monitored and any faults are alarmed and indicated. An abnormal condition is these systems would be signaled without having to place the diesel generators on test.

Percipitous failure of the station battery is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails.

REFERENCE

(1) FSAR, Section 8.2

Draft Technical

Specification Corresponding

to Section 11.2.4

3.6 REACTOR BUILDING

Applicability

Applies to the containment integrity of the reactor building.

Objective

To assure containment integrity.

Specification

3.6.1 Containment integrity, as defined in Section 1.7, shall be maintained whenever all three of the following conditions exist:

a. Reactor coolant pressure is 300 psig or greater.

b. Reactor coolant temperature is 200 °F or greater.

c. Nuclear fuel is in the core.

3.6.2 Containment integrity shall be maintained when both the reactor coolant system is open to the containment atmosphere and a shutdown margin exists that is less than that for a refueling shutdown.

3.6.3 Positive reactivity insertions which would result in a reduction in shutdown margin to less than 1% k/k shall not be made by control rod motion or boron dilution unless containment integrity is being maintained.

3.6.4 The reactor shall not be critical when the reactor building internal pressure exceeds 2.0 psig or 1.0 psi vacuum.

3.6.5 Prior to criticality following refueling shutdown, a check shall be made to confirm that all manual containment isolation valves which should be closed are closed and are conspicuously marked.

3.6.6 If, while the reactor is critical, a reactor building isolation valve is determined to be inoperable in a position other than the required position, the other reactor building isolation valve in the line shall be tested to insure operability. If the inoperable valve is not restored within 48 hours, the operable valve will be closed or the reactor shall be brought to the cold shutdown condition within an additional 24 hours.

3.6.7 One hydrogen recombiner shall be operable during STARTUP and POWER OPERATION. With the hydrogen recombiner inoperable, restore the recombiner to operable status or bring the reactor to hot standby within seven (7) days.

Bases

The Reactor Coolant System conditions of cold shutdown assure that no steam will be formed and hence no pressure will build up in the containment if the Reactor Coolant System ruptures.

The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

A condition requiring integrity of containment exists whenever the reactor coolant system is open to the atmosphere and there is insufficient soluble poison in the reactor coolant to maintain the core one percent subcritical in the event all control rods are withdrawn.

The reactor building is designed for an internal pressure of 55 psig, and an external pressure 2.5 psi greater than the internal pressure.

The operability of the hydrogen recombiner ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. The recombiner unit is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiodlytic decomposition of water and 3) corrosion of metals within containment. The recombiner is designed in accordance with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971, the acceptance criteria of S.R.P. 6.2.5., and NUREG-0578, July 1979. In addition to the installed hydrogen recombiner a second hydrogen recombiner can be installed after an accident within the time period available before it is needed; all piping, electrical, and structural provisions for the second recombiner are available.

The hydrogen mixing is provided by the reactor building ventilation system to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

REFERENCES

FSAR Section 5.2.2.4.3

4.4.4 Hydrogen Recombiner System

Applicability

Applies to the testing of the hydrogen recombiner and associated controls.

Objective

To verify that the hydrogen recombiner and associated controls are operable.

4.4.4.1 Specification

a. At least once per 92 days, during STARTUP or POWER OPERATION, perform a hydrogen recombiner system functional test to demonstrate that the minimum heater sheath temperature increases to $\geq$ 700°F within 90 minutes and is maintained $\geq$ 700°F for at least 2 hours. This test shall also be performed prior to STARTUP following a reactor outage greater than 90 days.

b. At least once per 18 months, perform the following surveillance:

1. A channel calibration of all recombiner instrumentation and control circuits.

2. Verify through a visual examination that there is no evidence of abnormal conditions within the recombiners (i.e., loose wiring or structural connections, deposits of foreign materials, etc.)

3. Verify during a recombiner system functional test that the heater sheath temperature increases to $\geq$ 1200°F within 5 hours and is maintained $\geq$ 1200°F for at least 4 hours.

4. Verify the integrity of the heater electrical circuits by performing a continuity and resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be $\geq$ 10,000 ohms.

Bases

The surveillance program described above provides high assurance that the hydrogen recombiner system will be available to perform its post-LOCA function of reducing the containment hydrogen concentration to below 4.1 volume percent.

Draft Technical

Specifications Corresponding

to Section 11.2.5

TABLE 3.5-1 Continued

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A cannot be met (a)
Engineered Safeguards			
3. Reactor Building Isolation and Reactor Building Cooling System			
a. Reactor Building 4 psig Instrument Channel	2	1	Hot Shutdown
b. Manual Pushbutton	2	1	Hot Shutdown
c. RPS Trip	2	1	Hot Shutdown
d. Reactor Building 30 psig	2	1	Hot Shutdown

(a) If minimum conditions are not met within 24 hours, the unit shall then be placed in a cold shutdown condition.

(b) Also initiates Low Pressure injection.

4. Reactor Building Spray System			
a. Reactor Building 30 psig Instrument Channel	2 (b)	1	Hot Shutdown
b. Spray Pump Manual Switches (c)	2	1	Hot Shutdown

(a) If minimum conditions are not met within 24 hours, the unit shall then be placed in a cold shutdown condition.

(b) Two out of three switches in each actuation channel operable.

(c) Spray valves opened by manual pushbutton listed in item 3 above.

TABLE 4.1-1 Continued

INSTRUMENTS OPERATING CONDITIONS

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
19.	Reactor Building Emergency Cooling and Isolation System Channels				
	a. Reactor Building 4 psig Channels	S(1)	M(1)	R (1)	When CONTAINMENT INTEGRITY is required
	b. Manual Pushbutton	S(1)	M(1)	NA (1)	When CONTAINMENT INTEGRITY is required
	c. RPS Trip	S(1)	M(1)	NA (1)	When CONTAINMENT INTEGRITY is required
	d. Reactor Building 30 psig	S(1)	M(1)	R (1)	When CONTAINMENT INTEGRITY is required
20.	Reactor Building Spray System Logic Channel	NA	Q	NA	
21.	Reactor Building Spray System Analog Channels				
	a. Reactor Building 30 psig Channels	NA	M	R	
22.	Pressurizer Temperature Channels	S	NA	R	
23.	Control Rod Absolute Position	S(1)	NA	R (1)	Check with Relative Position Indicator
24.	Control Rod Relative Position	S(1)	NA	R (1)	Check with Absolute Position Indicator
25.	Core Flooding Tanks				
	a. Pressure Channels	S(1)	NA	R (1)	When Reactor Coolant system pressure is greater than 700 psig
	b. Level Channels	S(1)	NA	R	
26.	Pressurizer Level Channels	S	NA	R	
27.	Makeup Tank Level Channels	D(1)	NA	R (1)	When Makeup and Purification System is in operation

Draft Technical

Specification Corresponding

to Section 11.2.6

3.5.6 Saturation Margin Meter

Applicability

Applies to the Saturation Margin Meter and associated alarm.

Objective

Provide the reactor operator with a reliable indication of the margin that exists between the water properties of the reactor coolant system and saturation conditions.

Specification

3.5.6.1 The Saturation Margin Meter shall be operable during startup and power operation. If the Saturation Margin Meter is inoperable, a procedure for calculation of saturation pressure margin and saturation temperature margin shall be made readily accessible to the reactor operator and the saturation meter returned to service at the earliest practicable time.

Bases

The Saturation Margin Meter provides a quick and reliable means for determination of saturation temperature and saturation pressure margins. The hand calculation of saturation pressure and saturation temperature margins can be easily and quickly performed since it only requires knowledge of recirculation loop temperatures and system pressure, and the use of steam tables; accordingly, hand calculation provides a suitable backup for the Saturation Margin Indicator.

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE		REMARKS
38.	Steam Generator Water Level	W	NA	R		
39.	Turbine Overspeed Trip	NA	R	NA		
40.	Sodium Thiosulfate Tank Level Indicator	NA	NA	R		
41.	Sodium Hydroxide Tank Level Indicator	NA	NA	R		
42.	Diesel Generator Protective Relaying	NA	NA	R		
43.	4 KV ES Bus Undervoltage Relays (Diesel Start)	NA	M(1)	R	(1)	Relay opreation will be checked by local test pushbuttons.
44.	Reactor Coolant Pressure DH Valve Interlock Bistable	S(1)	M	R	(1)	When reactor coolant system is pressurized above 300 psig or Taves is greater than 200°F.
45.	Loss of Feedwater Trip	S(1)	M(1)	R	(1)	When reactor > 10% power.
46.	Turbine Trip/Reactor Trip	S(1)	M(1)	R	(1)	When reactor > 20% power.
47.	Pressurizer Code Safety Valve - delta P/flow	S(1)	R	R	(1)	When T_{AVG} is greater than 200°F.
48	Pressurizer Electromatic Relief Valve - acoustic/flow	S(1)	R	R	(1)	When T_{AVG} is greater than 200°F.
49.	Saturation Margin Meter	S(1)	R	R	(1)	When T_{AVG} is greater than 200°F.

S - Each Shift

D - Daily

W - Weekly

M - Monthly

T/W - Twice per week

B/M - Every 2 months

Q - Quarterly

P - Prior to each startup if not done previous week

R - Each Refueling Period

NA - Not Applicable

B/W - Every two weeks

Draft Technical

Specification Corresponding

to Section 11.2.7

3.4 DECAY HEAT REMOVAL - TURBINE CYCLE

Applicability

Applies to the operating status of equipment that functions to remove decay heat, utilizing the secondary side of the steam generators.

Objective

To define the conditions necessary to assure immediate availability of the auxiliary feedwater system and main steam safety valves.

Specification

3.4.1 With the reactor coolant system temperature greater than 250°F, three independent steam generator emergency feedwater pumps and and associated flow paths shall be OPERABLE with:

a. Two emergency feedwater pumps, each capable of being powered from an OPERABLE emergency bus, and

b. One emergency feedwater pump capable of being powered from an OPERABLE steam supply system.

With one emergency feedwater pump or flow/path inoperable, restore the inoperable pump or flow path to OPERABLE status within 48 hours or be in COLD SHUTDOWN within the next 12 hours. With more than one emergency feedwater pumps or flow path inoperable, restore the inoperable emergency feedwater pumps or flow paths to operable status or be subcritical within 1 hour, in at least HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 6 hours.

c. Four of six turbine bypass valves are OPERABLE.

3.4.2 The condensate storage tanks (CSTS) shall be OPERABLE with a minimum of 150,000 gallons of condensate available in each CST. With a CST inoperable, restore the CST to operability within 4 hours or be in at least HOT STANDBY within the next 6 hours, at least HOT SHUTDOWN within the next 6 hours, and COLD SHUTDOWN within the next 24 hours.

3.4.3 With the reactor coolant system temperature greater than 250°F, all eighteen (18) main steam safety valves shall be operable or, if any are not operable, the maximum overpower trip setpoint (see Table 2.3-1) shall be reset as follows:

Maximum Number of Safety Valves Disabled on Any Steam Generator	Maximum Overpower Trip Setpoint (% of Rated Power)
1	92.4
2	79.4
3	66.3

With more than 3 main steam safety valves inoperable, restore at least fifteen (15) main steam safety valves to operable status within 4 hours or be in at least hot standby within the next 6 hours and in cold shutdown within the following 30 hours.

Bases

A reactor shutdown following power operation requires removal of core decay heat. Normal decay heat removal is by the steam generators with the steam dump to the condenser when system temperature is above 250°F and by the decay heat removal system below 250°F. Core decay heat can be continuously dissipated up to 15 percent of full power via the steam bypass to thee condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the steam generators is provided by the main feedwater system.

The main steam safety valves will be able to relieve to atmosphere the total steam flow if necessary. If main steam safety valves are inoperable, the power level must be reduced, as stated in Technical Specification 3.4.3, such that the remaining safety valves can accommodate the decay heat.

In the unlikely event of complete loss of off-site electrical power to the station, decay heat removal is by either the steam-driven emergency feedwater pump, or two half-sized motor-driven pumps. Steam discharge is to the atmosphere via the main steam safety valves and controlled atmospheric relief valves, and in the case of the turbine driven pump, from the turbine exhaust.(1)

Both motor-driven pumps are required initially to remove decay heat with one eventually sufficing. The minimum amount of water in the condensate storage tanks, contained in Technical Specification 3.4.2, will allow cooldown to 250°F with steam being discharged to the atmosphere. After cooling to 250°F, the decay heat removal system is used to achieve further cooling.

An unlimited emergency feedwater supply is available from the river via either of the two motor-driven reactor building emergency cooling water pumps for an indefinite period of time.

The requirements of Technical Specification 3.4.1 assure that before the reactor is heated to above 250°F, adequate auxiliary feedwater capacity is available. One turbine driven pump full capacity (920 gpm) and the two half-capacity motor-driven pumps (460 gpm, each) are specified. However, only one half-capacity motor-driven pump is necessary to supply auxiliary feedwater flow to the steam generators in the onset of a small break loss-of-coolant accident (Reference 2).

The requirements of Technical Specification 3.4.1 assure that at least 920 gpm is available at all times to both steam generators giving redundant capacity except for a limited time of 72 hours to allow for component maintenance. Further degradation of the emergency feedwater system requires the reactor to be subcritical within 1 hour.

The feedwater line break accident performed for TMI-2 (Reference 3) shows satisfaction of core thermal power limits and reactor coolant system pressure limits assuming full auxiliary feedwater flow within 40 seconds. The Technical Specification 3.4.1 provides assurance that this flow will be available with automatic initiation following loss of both main feedwater pumps.

REFERENCES

(1) FSAR, Section 10.2.1.3

(2) "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant," Volume I and II, Babcock and Wilcox, May 7, 1979.

(3) Three Mile Island Nuclear Station - Unit 2, Final Safety Analysis Report, USNRC Docket No. 50-320.

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE		REMARKS
38.	Steam Generator Water Level	W	NA	R		
39.	Turbine Overspeed Trip	NA	R	NA		
40.	Sodium Thiosulfate Tank Level Indicator	NA	NA	R		
41.	Sodium Hydroxide Tank Level Indicator	NA	NA	R		
42.	Diesel Generator Protective Relaying	NA	N	R		
43.	4 KV ES Bus Undervoltage Relays (Diesel Start)	NA	M(1)	R	(1)	Relay operation will be checked by local test pushbuttons.
44.	Reactor Coolant Pressure DH Valve Interlock Bistable	S(1)	M	R	(1)	When reactor coolant system is pressurized above 300 psig or Taves is greater than 200°F.
45.	Loss of Feedwater Trip	S(1)	M(1)	R	(1)	When reactor > 10% power.
46.	Turbine Trip/Reactor Trip	S(1)	M(1)	R	(1)	When reactor > 20% power.
47.	Pressurizer Code Safety Valve and Electromatic Relief Valve delta P/flow	S(1)	R	R	(1)	When T_{AVG} is greater than 200°F.
48.	Pressurizer Electromatic Relief Valve - acoustic/flow	S(1)	R	R	(1)	When T_{AVG} is greater than 200°F.
49.	Saturation Margin Meter	S(1)	M(1)	R	(1)	When T_{AVG} is greater than 200°F.
50.	Emergency Feedwater Flow Instrumentation	NA	M(1)	R	(1)	Emergency Feedwater is not normally in operation.

S - Each Shift
D - Daily
W - Weekly
M - Monthly
T/W - Twice per week
B/M - Every 2 months
Q - Quarterly
P - Prior to each startup if not done previous week
R - Each Refueling Period
NA - Not Applicable
B/W - Every two weeks

TABLE 4.1-2

MINIMUM EQUIPMENT TEST FREQUENCY

Item	Test	Frequency
1. Control Rods	Rod drop times of all full length rods	Each refueling shutdown
2. Control Rod Movement	Movement of each rod	Every two weeks, when reactor is critical
3. Pressurizer Safety Valves	Setpoint*	50% each refueling period
4. Main Steam Safety Valves	Setpoint	25% each refueling period
5. Refueling System Interlocks	Functional	Start of each refueling period
6. Main Steam Isolation Valves	(See Section 4.8)	
7. Reactor Coolant System Leakage	Evaluate	Daily, when reactor coolant system temperature is greater than 525°F
8. Charcoal and high efficiency filters for Control Room, and RB Purge Filters	DOP test on HEPA filters, freon test on charcoal filter units	Each refueling period and at any time work on filters could alter their integrity
9. Spent Fuel Cooling System	Functional	Each refueling period prior to fuel handling
10. Intake Pump House Floor (Elevation 262 Ft. 6 in.)	(a) Silt Accumulation-Visual inspection of Intake Pump House Floor	Each refueling period
	(b) Silt Accumulation Measurement of Pump House Flow	Quarterly
11. Pressurizer electromatic relief valve	Setpoint	Each refueling period
12. Back-up instrument air supply system	Functional	Each refueling period

*The setupoint of the pressurizer code safety valves shall be in accordance with ASME Boiler and Pressurizer Vessel Code, Section III, Article 9, Winter, 1968.

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4.5 EMERGENCY LOADING SEQUENCE AND POWER TRANSFER, EMERGENCY CORE COOLING SYSTEM AND REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

4.5.1 EMERGENCY LOADING SEQUENCE

Applicability

Applies to periodic testing requirements for safety actuation systems.

Objective

To verify that the Emergency loading sequence and automatic power transfer is operable.

Specifications

4.5.1.1 Sequence and Power Transfer Test

a. During each refueling interval, a test shall be conducted to demonstrate that the emergency loading sequence and power transfer is operable.

b. The test will be considered satisfactory if the following pumps and fans have been successfully started and the following valves have completed their travel on preferred power and transferred to the emergency power as evidenced by the control board component operating lights, and either the station computer or pressure/flow indication or, in the case of the Motor Driven Emergency Feedwater Pumps, the pump Interlock.

- M. U. Pump
- D. H. Pump and D. H. Injection Valves and D. H. Supply Valves
- R. B. Cooling Pump
- R. B. Ventilators
- D. H. Closed Cycle Cooling Pump
- N. S. Closed Cycle Cooling Pump
- D. H. River Cooling Pump
- N. S. River Cooling Pump
- D. H. and N. S. Pump Area Cooling Fan
- Screen House Area Cooling Fan
- Spray Pump. (Initiated in coincidence with a 2 out of 3 R. B. 30 psi Pressure Test Signal.)
- Motor Driven Emergency Feedwater Pump Interlock.

4.5.1.2 Sequence Test

a. At intervals not to exceed 3 months, a test shall be conducted to demonstrate that the emergency loading sequence is operable, this test shall be performed on either preferred power or emergency power.

b. The test will be considered satisfactory if the pumps and fans listed in 4.5.1b have been successfully started and the valves listed in 4.5.1.1b have completed their travel as evidenced by the control board component operating lights, and either the station computer or pressure/flow indication.

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Bases

The Emergency loading sequence and automatic power transfer controls the operation of the pumps associated with the emergency core cooling system and Reactor Building cooling system. A successful test of the emergency loading sequence and automatic power transfer is a prerequisite to any system test of the emergency core cooling system or reactor building cooling system.

References

(1) FSAR Section 7

(2) FSAR Section 1.4

4.9 EMERGENCY FEEDWATER SYSTEM PERIODIC TESTING

Applicability

Applies to the perodic testing of the turbine driven and two motor-driven emergency feedwater pumps, assocaited actuation signals, and valves.

Objective

To verify that the auxiliary feedwater system is capable of performing its design function.

Specification

4.9.1 TEST

4.9.1.1 Whenever the Reactor Coolant System temperature is greater than 250°F, the emergency feedwater pumps shall be tested in the recirculation mode in accordance with the requirements and acceptance criteria of ASME Section XI Article IWP-3000. The test frequency shall be at least every 31 days ±7 days of plant operation at Reactor Coolant Temperature above 250°F.

4.9.1.2 At least once per 31 days each valve (manual, power operated, or automatic) shall be verified to be in its correct position. This applies to valves CO V10 A and B, EF V1 A and B, EF V2 A and B, EF V10 A and B, EF V16 A and B, and MSV 2A and B.

4.9.1.3 At least once per 18 months, during shutdown, verify that:

(a) each emergency feedwater pump starting logic actuates upon receipt of an auxiliary feedwater actuation signal, and

(b) valves in the emergency feedwater flow paths actuate to their correct position on an emergency feedwater actuation signal and that the manual control valve station functions properly.

4.9.1.4 On a quarterly basis, the valves which are a part of the emergency feed system discharge (EFV-30A and 30B) will be checked for proper operation by cycling the valve over its full stroke.

4.9.1.5 Prior to start-up, following a cold shutdown of longer than 30 days' duration, conduct a test to demonstrate that the motor driven emergency feed pumps can pump water from the CST to the steam generators.

4.9.2 ACCEPTANCE CRITERIA

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly.

Bases

The 31 day testing frequency will be sufficient to verify that the turbine driven and two motor-driven emergency feedwater pumps are operable and that the associated valves are in the correct alignment. ASME Section XI Article IWP-3000 specifies requirements and acceptance standards for the testing of nuclear safety related pumps. Compliance with the normal

acceptance criteria of IWP-3000 assures that the emergency feedwater pumps are operating as expected. The test frequency of 31 days (nominal) has been demonstrated by the B&W Emergency Feedwater Reliability Study to assure an appropriate level of reliability. If testing under Article IWP-3000 indicates that the flow and/or pump head for a particular pump is not within the normal acceptance standard, Article IWP-3000 requires that an evaluation of the pump performance shall be completed within 96 hours or the pump declared inoperable. For the case of the emergency feedwater system, the system shall be considered operable if under the worst case single pump failure, a minimum of 500 gpm of emergency feedwater can be delivered when steam generator pressure is 1050 psig and one steam generation is isolated. A flow of 500 gpm, at 1050 psig head, ensures that sufficient emergency feedwater, demonstrated to be acceptable for plant cooling requirements under transient and accident conditions, can be delivered to either steam generator flow path. The 18 month surveillance requirements ensure that the overall emergency feedwater system functional capability is maintained comparable to the original design standards.

Draft Technical

Specifications Corresponding

to Section 11.2.8

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
10.	Flux-Reactor Coolant Flow Comparator	S	M	R	
11.	Reactor Coolant Pressure Temperature Comparator	S	M	R	
12.	Pump Flux Comparator	S	M	R	
13.	High Reactor Building Pressure Channel (1)	S	M	R	(1) Includes post-accident monitoring instrumentation (a) -5 psig to three-times design pressure
14.	High Pressure Injection Logic Channel	NA	Q	NA	
15.	High Pressure Injection Analog Channels				
	a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or $T_{av.}$ is greater than 200°F
16.	Low Pressure Injection Logic Channel	NA	Q	NA	
17.	Low Pressure Injection Analog Channels				
	a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or Taves is greater than 200°F
18.	Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	Q	NA	

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
28.	Radiation Monitoring Systesm (1)	W(2)	M	Q(3)	(1) Includes post accident monitoring instrumentation (a) High Range-Containment (b) Undiluted Containment Exhaust (c) Auxiliary and Fuel Handling Building Exhaust (d) Condenser Off Gas (2) Using the installed check source when background is less than twice the expected increase in cpm which would result from the check source alone. Background readings greater than this value are sufficient in themselves to show that the monitor is functioning. (3) Except area gamma radiation monitors RM-G6, RM-G7, and RM-G8 which are located in high radiation areas of the Reactor Building. These monitors will be calibrated quarterly or at the next scheduled reactor shutdown following the quarter in which calibration would normally be due, if a shutdown during the quarter does not occur.
29.	High and Low Pressure Injection Systems: Flow Channels	NA	NA	R	

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE		REMARKS
30.	Borated Water Storage Tank Level Indicator	W	NA	R		
31.	Boric Acid Mix Tank					
	a. Level Channel	NA	NA	R		
	b. Temperature Channel	M	NA	R		
32.	Reclaimed Boric Acid Storage Tank					
	a. Level Channel	NA	NA	R		
	b. Temperature Channel	M	NA	R		
33.	Containment Temperature	NA	NA	R		
34.	Incore Neutron Detectors	M(1)	NA	NA	(1)	Check functioning; including functioning of computer readout or recorder readout when reactor power is greater than 15%
35.	Emergency Plant Radiation Instruments	M(1)	NA	R	(1)	Battery check
36.	Strong Motion Accelerometer	Q(1)	NA	Q	(1)	Battery check
37.	Reactor Building Sump and Containment Level (1)	NA	NA	R	(1)	Includes post-accident monitoring Instrumentation (a) Narrow Range (sump) (b) Wide Range (Containment)

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE		REMARKS
51.	Reactor Coolant Pump Trip Trip	S(1)	M(1)	R	(1)	When T_{AVG} is greater than 200°F.
52.	Reactor Building Hydrogen Concentration	S(1)	M(1)	R	(1)	When T_{AVG} is greater than 200°F.

Draft Technical

Specification Corresponding

to Section 11.2.9

3.5.7 Reactor Coolant Pump Trip

Applicability

Applies to the operational status of the reactor coolant pump trip.

Objective

To specify requirements for operability of the reactor coolant pump trip.

Specification

3.5.6.1 The reactor coolant pump trip, with a minimum of a high pressure injection actuation channel input and one pump current channel input per OPERABLE reactor coolant pump, shall be OPERABLE during reactor STARTUP and POWER OPERATION. With the reactor coolant pump trip inoperable, place the reactor in hot shut-down within 24 hours.

Bases

Analysis has shown that, for a certain range of small primary breaks, inacceptable clad temperatures may result if the reactor coolant pumps are not tripped at a time when the Reactor Coolant System void fraction has achieved a high level. To prevent these detrimental consequences, the reactor coolant pump trip will promptly trip the reactor coolant pumps when system conditions indicate that a small break in this range may be in progress. The system actuates when High Pressure Injection has been initiated and the reactor coolant system void fraction has reached a nominal value, as sensed by the reactor coolant pump current monitors, which indicates that a high void fraction may develop.

Table 4.1-1 (Continued)

	Channel Description	Check	Test	Calibrate		Remarks
51.	Reactor Coolant Pump Trip	S(1)	M(1)	R	(1)	When T_{AVG} is greater than 200°F.
52.	Reactor Building Hydrogen Concentration	S(1)	M(1)	R	(1)	When T_{AVG} is greater than 200°F.

Draft Technical

Specification Corresponding

to Section 11.2.10

3.19 Separation of TMI-1 and TMI-2

Applicability

Applies to interconnections between TMI-1 and TMI-2 which have the potential for transferring significant quantities of contamination between units.

Objective

To control the transfer of radioactivity from TMI-1 to TMI-2 via system interties.

Specification

3.19.1 The isolation devices for the system interties described in Table 3.19-1 shall remain isolated unless written approval has been received from the NRC. If approval for use of interties is received, use shall proceed under preestablished plant procedures. If a system intertie, listed in Table 3.19-1 is found defeated without prior NRC approval, a "Thirty Day Written Report" shall be prepared and submitted as required by Technical Specification 6.9.2.B.(5).

3.19.2 No additional TMI-1/TMI-2 interties, with the potential of transferring significant quantities of radioactivity, shall be created without prior NRC approval.

Bases

Interties exist between TMI-1 and TMI-2 that have the potential for transferring contamination to TMI-1 as a result of restoration activities at TMI-2. These interties should remain isolated unless approval for their use is received from the NRC.

Table 3.19-1

TMI-1/TMI-2 Interties

(1) Unit 2 Reactor Coolant Bleed Holdup Tank - Unit 1 Reactor Coolant Waste Evaporator

(2) Unit 1 Miscellaneous Waste Evaporator - Unit 2 Evaporator Condensate Test Tanks

(3) Unit 2 Neutralizer Tanks, Contaminated Drain Tanks, Reactor Coolant Bleed Holdup Tanks, Auxiliary Building Sump Tanks and Miscellaneous Waste Hold-up Tanks - Unit 1 Liquid Waste Disposal System

(4) Unit 1 Evaporator Concentrate -
Unit 2 Evaporator Concentrate

(5) Unit 1 Spent Ion Exchange Resin -
Unit 2 Spent Ion Exchange Resin

Table 4.1-2 (Continued)

Item	Test	Frequency
13. Isolation devices on Unit 1/Unit 2 tielines:	Visual Inspection	Monthly
(a) Unit 2 Reactor Coolant Bleed Holdup Tank - Unit 1 Reactor Coolant Waste Evaporator		
(b) Unit 1 Miscellaneous Waste Evaporator - Unit 2 Evaporator Condensate Test Tanks		
(c) Unit 2 Neutralizer Tanks, Containment Drain Tanks, Reactor Coolant Bleed Holdup Tanks, Auxiliary Building Sump Tanks and Miscellaneous Waste Holdup Tanks - Unit 1 Liquid Waste Disposal System		
(d) Unit 1 Evaporator Concentrate - Unit 2 Evaporator Concentrate		
(e) Unit 1 Spent Ion Exchange Resin - Unit 2 Spent Ion Exchange Resin		

6.9.2 REPORTING REQUIREMENTS (cont'd)

Note: This item is intended to provide for reporting of potentially generic problems

B. Thirty Day Written Reports. [1/]The reportable occurrences discussed below shall be the subject of written reports to the Director of the appropriate Regional Office within thirty days of occurrence of the event. The written report shall include narrative material to provide complete explanation of the cause of the event, circumstances surrounding the event, any corrective action, and component failure data.

(1) Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

(2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

Note: Routine surveillance testing, instrument calibration, or preventative maintenance which require system configurations as described in items 6.9.2.B(1) and 6.9.2.B(2) need not be reported except where test results themselves reveal a degraded mode as described above.

(3) Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

(4) Abnormal degradation of systems other than those specified in item 6.9.2.A(3) above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

(5) Observed defeat of an isolation device, which separates a tieline between Units 1 and 2, without prior NRC approval.

Draft Technical

Specification Corresponding

to Section 11.2.11

3.5.3 ENGINEERED SAFEGUARDS PROTECTION SYSTEM ACTUATION SETPOINTS

Applicability

This specification applies to the engineered safeguards protection system actuation setpoints.

Objective

To provide for automatic initiation of the engineered safeguards protection system in the event of a breach of Reactor Coolant System integrity.

Specification

3.5.3.1 The engineered safeguards protection system actuation setpoints and permissible bypasses shall be as follows:

Initiating Signal	Function	Setpoint
High Reactor Building Pressure (1)	Reactor Building Spray	$\leq$ 30 psig
	High-Pressure Injection	$\leq$ 4 psig
	Low-Pressure Injection	$\leq$ 4 psig
	Start Reactor Building Cooling & Reactor Building Isolation	$\leq$ 4 psig
Low Reactor Coolant System Pressure	High Pressure Injection	$\geq$1600(2) and $\geq$500(3) psig
	Low Pressure Injection	$\geq$1600(2) and $\geq$500(3) psig

(1) May be bypassed for reactor building leak rate test.

(2) May be bypassed below 1750 psig and is automatically resinstated above 1750 psig.

(3) May be bypassed below 900 psig and is automatically resinstated above 900 psig.

Bases

High Reactor Building Pressure

The basis for the 30 psig and 4 psig setpoints for the high pressure signal is to establish a setting which would be reached in adequate time in the event of a LOCA, cover a spectrum of break sizes and yet be far enough above normal operation maximum internal pressure to prevent spurious initiation.

Low Reactor Coolant System Pressure

The basis for the 1600 and 500 psig low reactor coolant pressure setpoint for high and low pressure injection initiation is to establish a value which is high enough such that protection is provided for the entire spectrum of break sizes and is far enough below normal operating pressure to prevent spurious initiation. Bypass of HPI below 1750 psig, and LPI below 900 psig, prevents ECCS actuation during normal system cooldown.

Draft Technical

Specifications Corresponding

to Section 11.2.12

c. Reactor coolant system pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip set point is reached before the nuclear overpower trip set point. The trip setting limit for high reactor coolant system pressure has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient. (6)Due to calibration and instrument errors, the safety analysis assumed a 45 psi pressure error in the high reactor coolant system pressure trip setting.

The high pressure trip setpoint was subsequently lowered from 2390 psig to 2300 psig. The lowering of the high pressure trip setpoint and raising of the setpoint for the pressurizer Electromatic Relief Valve, from 2255 psig to 2450 psig, has the effect of reducing the challenge rate to the pressurizer Electromatic Relief Valve while maintaining ASME Code Safety Valve capability. (7)

The low pressure (1900 psig) and variable low pressure (11.75 T_{OUT} - 5103) trip setpoint were established to maintain the DNB ratio greater than or equal to 1.3 fore those design accidents that result in a pressure reduction (3,4). The B&W generic ECCS analysis however, assumed a low pressure trip of 1900 psig and is therefore the basis of low pressure reactor trip. Figure 2.3-1 shows the high pressure, low pressure, and variable low pressure trips.

d. Coolant outlet temperature

The high reactor coolant outlet temperature trip setting limit (619 F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperature in the operating range.

The calibrated range of the temperature channels of the RPS is 520 to 620° F. The trip setpoint of the channel is 619°F. Under the worst case environment, power supply perturbations, and drift, the accuracy of the trip string is 11F. This accuracy was arrived at by summing the worst case accuracies of each module. This is a conservative method of error analysis since the normal procedure is to use the root mean square method.

Therefore, it is assured that a trip will occur at a value no higher than 620°F even under worst case conditions. The safety analysis used a high temperature trip set point of 620°F.

The calibrated range of the channel is that portion of the span of indication which has been qualified with regard to drift, linearity, repeatability, etc. This does not imply that the equipment is restricted to operation within the calibrated range. Additional testing has demonstrated that in fact, the temperature channel is fully operational approximately 10% above the calibrated range.

Since it has been established that the channel will trip at a value of RC outlet temperature no higher than 620°F even in the worst case, and since the channel is fully operational approximately 10% above the calibrated range and exhibits no hysteresis or foldover characteristics, it is concluded that the instrument design is acceptable.

e. Reactor building pressure

The high reactor building pressure trip setting limit (4 sig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

f. Shutdown bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. By administrative control the nuclear overpower trip set point must be reduced to a value $\leq$ 5.0 percent of rated power during reactor shutdown.

2. A high reactor coolant system pressure trip set point of 1720 psig is automatically imposed.

The purpose of the 1720 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The overpower trip set point of $\leq$ 5.0 percent prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation (5) would be available to remove 5.0 percent of rated power if none of the reactor coolant pumps were operating.

Refereces

(1) FSAR, Section 14.1.2.3

(2) FSAR, Section 14.1.2.2

(3) FSAR, Section 14.1.2.7

(4) FSAR, Section 14.1.2.8

(5) FSAR, Section 14.1.2.6

(6) Technical Specification Change Request No. 31, January 16, 1976, and Technical Specification Change Request No. 84, June 23, 1978.

(7) "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant," Volumes I & II, Babcock and Wilcox, May 7, 1979.

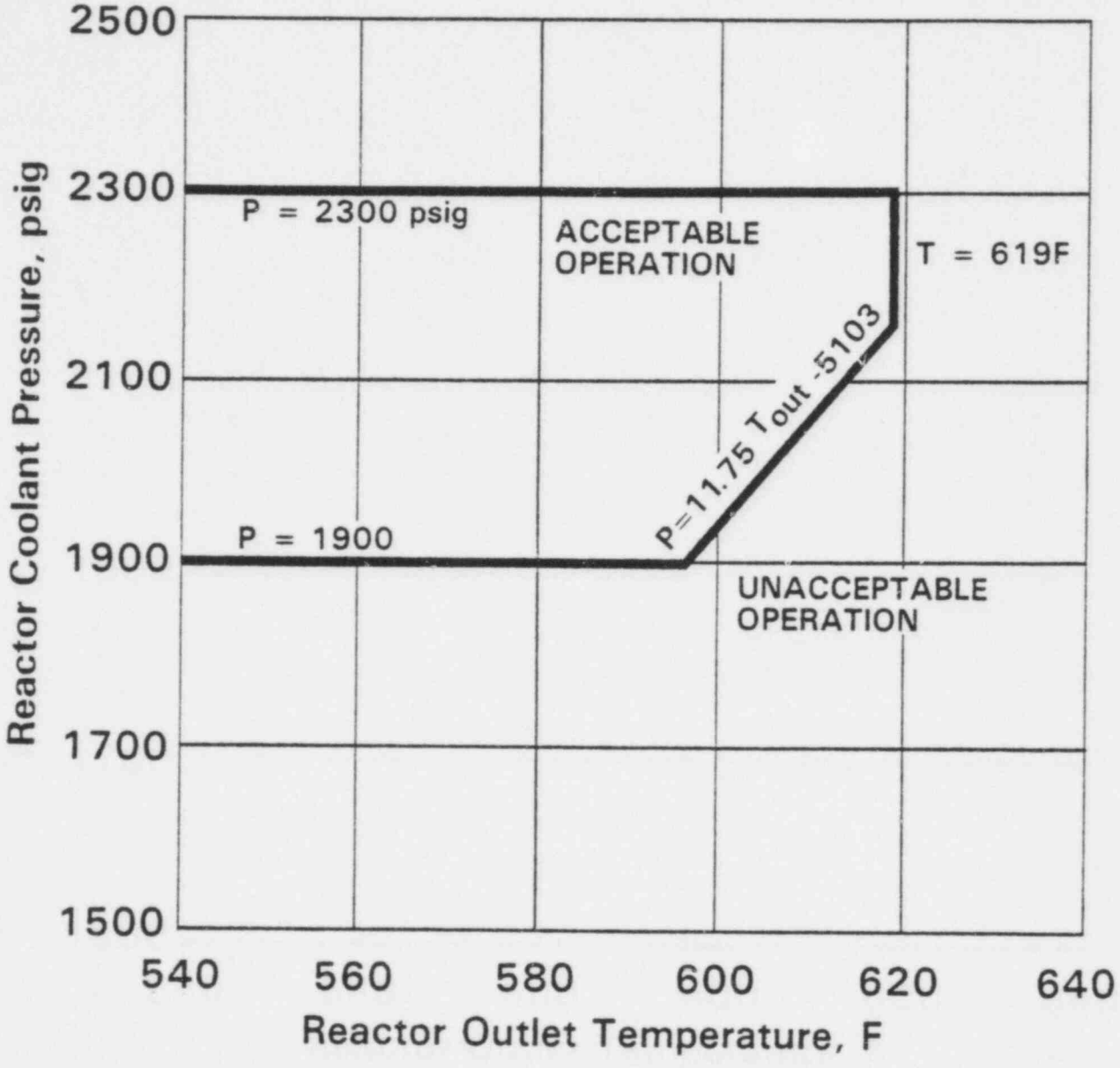
(8) "ECCS Analysis of B&W's 177-FA Lowered Loop NSS," BAW-10103, Rev. 2, Babcock and Wilcox, April 1976.

TABLE 2.3-

REACTOR PROTECTION SYSTEM TRIP SETTING LIMITS

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power - 75%)	One Reactor Coolant Pump Operating in Each Loop (Nominal Operating Power- 49%)	Shutdown Bypass
1. Nuclear power, Max. % of rated power	105.5	105.5	105.5	5.0(3)
2. Nuclear power based on flow (2) and imbalance max. of rated power	1.08 times flow minus reduction due to imbalance(s)	1.08 times flow minus reduction due to imbalance (s)	1.08 times flow minus reduction due to imbalance(s)	Bypassed
3. Nuclear power based (5) on pump monitors, max. % of rated power	NA	NA	91%	Bypassed
4. High reactor coolant system presure, psig, max.	2300	2300	2300	1720(4)
5. Low reactor coolant system pressure, psig min.	1900	1900	1900	Bypassed
6. Variable low reactor coolant system pressure psig, min.	(11.75 Tout-5103)(1)	(11.75 Tout-5103)(1)	11.75 Tout-5103(1)	Bypassed
7. Reactor coolant temp. F., Max.	619	619	619	619
8. High Reactor Building pressure, psig, max.	4	4	4	4

(1) Tout is in degrees Fahrenheit (F)
(2) Reactor coolant system flow, %
(3) Administratively controlled reduction set only during reactor shutdown
(4) Automatically set when other segments of the RFS (as specified) are bypassed
(5) The pump monitors also produce a trip on: (a) loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps during two-pump operation
(6) Trip settings limits are setting limits on the setpoint side of the protection system bistable connectors.



TMI-1
Protection System Maximum Allowable Set Points

Figure 2.3-1

QUESTION

38. With regard to a recent event at Oconee Unit 3 in which certain indications in the control room became unavailable, discuss the vulnerability of TMI-1 to a similar malfunction. Also, consider modifications which would reduce the potential for this type of event.

RESPONSE

The electrical distribution system to the ICS at TMI-1 has been reviewed and it has been concluded that the ICS is vulnerable to an event of the type that took place recently at Oconee-3. This would occur upon loss of inverter 1A and failure of the automatic transfer switch. A modification is being made which will ensure that the operator can take prompt action to minimize the consequences of such an event. Manual transfer contactors, operable from the control room will be provided to transfer the ICS supply bus (Distribution Panel ATA) from the inverter bus (Distribution Panel VBA) to the Regulated AC Supply (Distribution Panel TRA). The contactors will be interlocked to prevent paralleling of the supplies. An alarm will be provided to indicate that voltage to Distribution Panel ATA has been lost. This modification will be installed before restart.

The following longer term actions are being pursued:

1. Investigation of the feasibility of modifying the ICS to allow some of ICS power feeds to be supplied from a separate source.

2. Further review of the electrical distribution system to determine whether the reliability of the ICS/NNI power supplies can be improved.

SUPPLEMENT 2

THREE MILE ISLAND NUCLEAR STATION

UNIT 1

OPERATIONAL QUALITY ASSURANCE PLAN

TMI / GPU

Generation Group

OPERATIONAL

QUALITY ASSURANCE PLAN

FOR

THREE MILE ISLAND

NUCLEAR STATION

STATEMENT OF POLICY AND AUTHORITY

It is the policy of the Metropolitan Edison Company to operate the Three Mile Island Nuclear Station so as to ensure the safety and health of the public and the personnel on site.

It is also the policy of the Metropolitan Edison Company to comply with the requirements of the Code of Federal Regulations, the NRC Operating Licenses and the applicable codes, guides and standards with respect to operation, inservice inspection, refueling, maintenance, procurement, repair and modification of the Station.

The Senior Vice President of Metropolitan Edison Company has the overall responsibility for establishing the policies, goals and objectives of the Quality Assurance Program. He utilizes the support and services of the GPU Service Corporation in implementing the requirements of the Quality Assurance Program. The Senior Vice President of Metropolitan Edison Company is also the Vice President -Generation of the GPU Service Corporation. This dual role gives him the authority to manage and control the activities of the TMI Generation Group. The TMI Generation Group was formed to strengthen the management of and to provide greatly increased technical resources to the Three Mile Island Nuclear Station. The Senior Vice President Metropolitan Edison/Vice President GPU Service Corporation is the head of the TMI Generation Group. In this position, he reports directly to the President of GPU Service Corporation (who is also President of General Public Utilities Corp.) and President of Metropolitan Edison.

This reporting structure provides a direct line of authority from the Chief Operating Officer of these three companies to the activities at the Three Mile Island Nuclear Station. A primary objective of the group is to ensure safe operations by means which include strict adherence to NRC Regulations, Technical Specifications and Plant Procedures.

The Director-Reliability Engineering who reports directly to the Senior VP Met-Ed/VP GPUSC provides, by way of the Quality Assurance Department, the staff necessary to develop and maintain the Quality Assurance Program consistent with the applicable Federal and State requirements and to verify the implementation of the Program.

The Manager-Quality Assurance, who reports directly to the Director-Reliability Engineering, has the overall authority and organizational freedom to identify quality assurance or management control problems and provide recommended solutions. This authority and responsibility includes stop work authority in activities associated with operations, maintenance, repair, modification, refueling and manufacturing at or for the Three Mile Island Nuclear Station. With regard to the stoppage of work including the recommendation that an operating nuclear unit be shut down, the Manager-Quality Assurance has direct access to the Senior VP Met-Ed/VP GPUSC and shall use this path when differences of opinion within the Generation Group regarding quality arise.

The effectiveness of any Quality Assurance Program is dependent upon the individuals who implement the program. Accordingly, all personnel of the General Public Utility System and their contractors must comply with the applicable requirements of this Quality Assurance Program. All members of management must give full support to maintaining an effective quality program as defined in this Plan.

The Quality Assurance Program, as described in this Plan, is approved for implementation at Three Mile Island Nuclear Generation Station.

Date	President, Met-Ed (Acting)
Date	President, GPU Service Corp.

TABLE OF CONTENTS

OPERATIONAL QA PLAN

Introduction

This Quality Assurance Plan is formatted in such a manner to provide all users with a functionally workable document. It is structured to describe how the Quality Assurance Program is to be functionally implemented with due regard to the safety and health of the public and the personnel onsite. The plan contains a description of the organizations responsible for the implementation of the Quality Assurance Program (Section 1) and an overall description of the Program (Section 2). The remaining sections are structured in a functional manner.
The requirements for administrative control which are generic and apply to all subsequent sections are as follows: control of documents and records are contained in Section 3.0; control of design is contained in Section 4.0; control of materials and services through the control of procurement activities is contained in Section 5.0. Sections 6.0 and 7.0 contain the program requirements for those direct and supportive activities, associated with the operation and safety of the plant; construction and/or modifications associated with corrective maintenance, plant improvement, and/or repair; and the processing and transportation of radioactive wastes. Specific requirements such as those including control of measurement and test equipment, inspection, special processes, test control, and status of inspections, tests and operations are included therein.

Sections 8.0 and 9.0 again apply to all functions covered by the scope of this Quality Assurance Program. Section 8.0 addresses the subjects of identification and disposition of nonconformances associated with all aspects of the program. In addition, this section contains the management controls provided for evaluating, collectively all nonconformances and determining what corrective actions should be taken to preclude their recurrence. Section 9.0 contains the requirements and administrative controls applicable to safety reviews and audits. Appendices A, B, and C contain additional Quality Program requirements associated with the functional areas discussed in the plan.

1.0 Organization

1.1 Policy

The responsibility for the safe and economical operation of the TMI Nuclear Station rests with the TMI Generation Group. As identified in the Statement of Policy and Authority, the TMI Generation Group utilizes the support and services of the GPUSC in implementing the Operational Quality Assurance Program. The organizations having responsibility for the operation, maintenance, refueling, inservice inspection, modifications and repair of TMI Units 1 and 2 include the TMI Generation Group, under the direction of the Senior Vice President of Met-Ed/Vice President GPUSC, and Materials Management under the direction of the Vice President, GPUSC Materials Management.

It is the policy of the Metropolitan Edison Company (Met-Ed) to meet the quality assurance requirements of Nuclear Regulatory Commission as presented in 10 CFR 50, Appendix B, and other applicable regulatory guides, codes, and standards pertinent to the operation of the Three Mile Island Nuclear Station. The Program, which is described in the following sections, shall be implemented throughout the operation phase in documented approved policies, procedures, instructions which comply with this Plan and the design specified in the license application.

1.2 Organization

The structure of the organizations responsible for the operation, maintenance, modification, repair, inservice inspection and refueling of the TMI Nuclear Station is illustrated in Figure 1. The overall organization chart is provided to illustrate the interfaces between the various departments and to identify those normally located on-site and off-site.

1.2.1 President

In the Statement of Policy and Authority, the Presidents of Met-Ed and GPUSC have identified that the safe operation of TMI is the responsibility of all persons performing activities which affect quality and that management will

give full support to the proper and complete implementation of the OQA Program. Lines of authority and responsibilities have been established for maintaining and implementing the Program; for providing independent verification of the activity; and for appraising management of the effectiveness of the Program.

The responsibilities for establishment, implementation and measurement of the effectiveness of the Program are assigned to the TMI Generation Group with support from Materials Management. In providing support to the TMI Generation Group, Materials Management will comply with the requirements of this Operational Quality Assurance Plan as defined further in Subsection 1.2.7.

1.2.2 Senior Vice President Met-Ed/Vice President GPUSC (Figure 1)

The TMI Generation Group, under the direction of a Senior Vice President Met-Ed/Vice President GPUSC, is responsible for establishing the policies, goals and objectives of the OQA Program and for providing the on-site and off-site staffs necessary to implement the Program and provide the verification necessary to assure the effectiveness of the Program. The responsibilities are carried out through four (4) Directors and one (1) Manager.

The Senior Vice President Met-Ed/Vice President GPUSC has the responsibility for directing and assuring that the management controls and the quality assurance program necessary for the safe operation of TMI are established and effectively executed. To this end, they include providing the management personnel, the staff support, and the appropriate investment of time and financial resources to enable the designated individuals to properly execute their responsibilities.

The Senior Vice President Met-Ed/Vice President GPUSC is also responsible for assessing the effectiveness of the program and for assuring that decisions affecting nuclear safety are made at the proper level of responsibility and with the necessary technical advice and review. This responsibility shall be met, as a minimum, by:

a. Assuring that an independent managment review of the effectiveness of the OQA Program is conducted annually. The results of this review snall be documented in a report.

b. Receiving and reviewing summaries of reports prepared by the organizations performing independent audits and safety review. These organizations include the Quality Assurance Department, the Nuclear Safety Evaluation Department, the Generation Review Committee (GRC), and the Plant Operations Review Committee (PORC).

To the extent necessary to assure the health and safety of the public and the employees and contractors working at TMI, the Senior Vice President Met-Ed/VIce President GPUSC shall have the authority, the organizational freedom and the responsibility to order the shutdown of one or both of the operating units.

1.2.3 Director - TMI Unit 1

The Director - TMI Unit 1 is responsible for the overall safety of the TMI Nuclear Station, for ensuring that the applicable procedures for the management control and quality assurance program activities are implemented in the conduct of operations, preventative and corrective maintenance, replacment, modification, refueling, engineering support, in-service inspection, radiation protection and control of radioactive wastes, training, plant security, and for ensuring that those activities are performed in accordance with the provisions and limitations set forth in the licenses and permits of the jurisdictional agencies of Federal, State, and local governments. He is also responsible for ensuring that the operations organization is adequately staffed and that the personnel are adequately trained and qualified to perform their assigned tasks. Additionally, the Vice President-Nuclear Operations is responsible for activities such as support services and logistics and for the planning and scheduling of

plant operations such as start-up and test, refueling and planned outages, and productivity of the generating station.

The Director - TMI Unit 1 gives his fullest support to the quality assurance requirements set forth in this Operational Quality Assurance Plan, assuring compliance to the fullest degree by his staff.

1.2.3.1 Manager-Plant Engineering

The Manager-Plant Engineering is responsible for maintaining technical liaison and coordination between operating shift personnel and the technical support engineering staff. This is accomplished by providing on-shift engineers to the Operations staff for direct technical coverage of the plant reactor performance and associated safety systems in order to improve the safety of unit operations and maintenance performance. In addition, the Manager-Plant Engineering is responsible for in-plant engineering support in the nuclear, mechanical, instrumentation and control, and electrical engineering areas. The Manager-Plant Engineering is also responsible for plant chemistry, fire protection and engineering input for procurement of items and services important to safety.

1.2.3.2 Manager-TMI Unit 1

The Manager - TMI Unit 1 is responsible for the day-to-day operation of Unit 1. He will have a Shift Foreman directing the operations of each shift through the Control Room Operators and Auxiliary Operators; a maintenance force under the direction of a Supervisor-Maintenance, covering the areas of electrical, mechanical and instrument control maintenance and surveillance, will also report to the Manager. Additionally, the Manager - TMI Unit 1 is responsible for the coordination of start-up and test evaluations.

1.2.3.3 Manager-Radiological Controls

The Manager-Radiological Controls is responsible for the personnel, procedures and administrative controls of the radiation protection programs. He provides the administrative

and technical guidance applicable to operations in the areas of radiation protection, radioactive waste, respiratory protection, health physics engineering including ALARA programs, and dosimetry control.

The Manager-Radiological Controls is responsible for providing and maintaining up-to-date procedures controlling the activities of the department, providing training of all Unit personnel in the basic rules of radiation protection, providing adequate staffs of trained personnel to perform the duties of radiation protection, implementing the "as low as reasonably achievable" policy and making it a formal part of the Radiation Protection Program, and assisting in the development, training and implementation of the Station Emergency Plan.

1.2.3.4 Manager-Training

The Manager-Training is responsible for overall administrative control and coordination of all training activities. The purpose of the training program is to develop and maintain an organization fully qualified to ensure the continued safe and efficient operation and maintenance of the Units. In this position, the Manager-Training is responsible for the training of personnel requiring an NRC License and the training of personnel not requiring an NRC License. The former includes an accelerated operator retraining program and the continuous program of training and examination of operators necessary to maintain their NRC Operator's License; the latter includes management and technical training in the areas of Radiation Protection, Industrial Safety, Plant Security, Quality Assurance, Fire Protection, Maintenance and specialized areas as may be identified by management.

The Manager-Training is responsible for the coordination and scheduling of the training, assuring that the training is given by individuals or organizations qualified in the specific subjects, and maintaining records of the training provided and the attendance.

1.2.3.5 Manager-Administration and Services

The Manager-Administration and Services is responsible for coordination of facility functions such as office management, facilities, personnel, station security, and the Station Document Center. Relative to the activities applicable to this Quality Assurance Plan, these responsibilities include establishing, supervising and operating the Station Document Center; providing and maintaining up-to-date procedures for controlling the distribution of documents, and the collection, indexing and storage of records; providing the staff necessary to fulfill these responsibilities include establishing, supervising and operating the Station Document Center; providing and maintaining up-to-date procedures for controlling the distribution of documents, and the collection, indexing and storage of records; providing the staff necessary to fulfill these responsibilities, and ensuring that staff is adequately trained and qualified to perform their assigned tasks.

1.2.4 Director-Technical Functions

The Director-Technical Functions reports directly to the Senior Vice President Met-Ed with responsibility for the detailed development, direction and overall coordination of all engineering activities. He is responsible to assure compliance and implementation of the Quality Assurance Program requirements applicable to technical support activities. Technical support includes various disciplines such as mechanical, civil, electrical and instrumentation, nuclear, and plant operations. He is responsibile to develop and control the Quality Classification List (QCL). Additionally, he is responsible for nuclear fuel management, process computer, control and safety analysis, and plant operational analysis.

The Director-Technical Functions and his staff give full support to the TMI Operational Assurance Plan set forth herein, thereby assuring that all work performed under their cognizance will conform to and support the requirements as applicable to their activities.

1.2.4.1 Manager-Engineering and Design

The Manager-Engineering and Design is responsible for providing technical support for the operations of the TMI Nuclear Station. He is responsible to assure compliance with the implementation of the Quality Assurance Program requirements applicable to engineering and design activities. He will assure the maintenance of technical capability in the various disciplines, such as general mechanical, civil, electrical and instrumentation, and engineering mechanics. The department will review, and where appropriate, approve the work of Architect/Engineering Organizations, and will perform basic engineering and design for modifications. The department has capabilities in the following areas:

a. Engineering Mechanics

b. Mechanical Systems

c. Mechanical Components

d. Electrical Power & Instrumentation

e. Design & Drafting

Additionally, he is responsible for the identification and classification of materials and activities important to safety, and the development and control of the Quality Classification List.

1.2.4.2 Manager-Systems Engineering

The Manager-Systems Engineering is responsible for technical support in the areas of nuclear fuel management, process computer, control and safety analysis, and plant operational analysis. His department provides capabilities in the following functional areas:

a. Nuclear Analysis and Fuels

b. Process Computers

c. Control and Safety Analysis

d. Plant Analysis

He is responsible to assure compliance with and implementation of the Quality Assurance Program requirements applicable to Systems Engineering activities.

1.2.4.3 Project Engineering Manager

The Project Engineering Manager is responsible for the coordination, staffing and directing of engineering projects that are assigned by the Director-Technical Functions. These activities will vary, depending on the scope and purpose of the assigned project. These responsibilites generally include providing the technical support necessary for a study, an evaluation or a modification and include the coordination of the Departments within Technical Functions with those of Plant Operations.

1.2.5 Director-Environment, Health and Safety

The Director-Health and Safety reports directly to the Senior Vice President Met-Ed and is responsible for the development, direction and overall coordination of the environmental, regulatory, water resources, offsite radiological health and safety efforts at TMI in compliance with the TMI Quality Assurance Program. His responsibilities include:

a. Developing and implementing Environment, Health and Safety Group procedures covering items such as safety evaluations, and others as required to fulfill the responsibilities of this Plan.

b. Concurring with important to safety Design Criteria Documents from the standpoint of having addressed all applicable regulatory requirements and licensing commitment.

c. Exercising project control of amendment requests to the Safety and Environmental Technical Specifications and the FSAR in accordance with 10 CFR.

d. Developing biological monitoring programs and special studies in the Environmental Technical Specifications

to quantify the impact of Station operation on the environment.

e. Performing an environmental evaluation of proposed modifications, including publishing of environmental reports.

f. Maintaining liaison between the TMI Generation Group and NRC's Project Management regarding licensing and environmental issues which are applicable to operating facilities.

1.2.6 Director-Reliability Engineering

The Director-Reliability Engineering has the overall authority and direct responsibility for all Quality Assurance Department activities as defined in this plan. These activities include, but are not limited to:

a. Development, distribution, and maintenance of the Quality Assurance Plan,

b. Assessing program implementation and evaluating its effectiveness,

c. Identification of quality problems,

d. Initiation and recommendations of corrective actions for quality related problems.

Additionally, he is responsible for providing Quality Assurance and Quality Control support services such as laboratory analysis, safety review (Nuclear Safety Evaluation), audits and reliabiliaty information systems. The Director-Reliability Engineering utilitizes the following management staff members in carrying out his responsibilities:

Manager-Quality Assurance

Manager-Nuclear Safety Evaluation

Manager-System Laboratory

1.2.6.1 Manager-Quality Assurance (Figure 2)

The Manager-Quality Assurance Department (QAD) has the functional authority, independence and

responsibility to verify the effective implementation of the administrative controls and compliance to the Quality Assurance Program during the operational phase of TMI Nuclear Station. The Manager of QAD reports directly to the Director-Reliability Engineering. Additionally, he has direct unencumbered access to the Senior Vice President of Met-Ed, the Vice President-Materials Management, and the Director-TMI Unit 1 with regard to quality activities.

This reporting relationship has been established to provide the quality assurance organization with sufficient independence from the influence of costs and schedules to be able to effectively assure conformance of operational Quality Assurance Program requirements. Figure 2 identifies the Quality Assurance Department organizational elements which function under the Quality Assurance Program. The Manager-QAD has no duties or responsibilities unrelated to Quality Assurance that would prevent his full attention to Quality Assurance matters, and he has authority:

a. To evaluate the manner in which all activities, both onsite and offsite are conducted, with respect to quality, by means of review, survey, audit, surveillance, monitoring, and inspection.

b. To perform evaluations on a planned and periodic basis to verify that the Quality Assurance Program is being effectively implemented.

c. To identify quality problems, and to initiate, recommend or provide solutions through designated channels to verify implementation of resolutions.

d. To stop work or further processing, delivery, or installation of nonconforming material, to stop work on nonconforming activities, to initiate unit shutdown recommendations and to obtain unit shutdown with appropriate upper management concurrence as described in applicable Quality Assurance procedures.

The specific responsibilities of the Manager-QAD, include the following:

a. Provide for the review and acceptance of the Quality Assurance Program of contractors providing services affecting quality and of vendors supplying materials, parts, or components covered by the scope of this Quality Assurance Program.

b. Provide for review and acceptance of procedures prepared by other TMI organizations when these procedures control or exercise an effect upon items and activities important to safety.

c. Provide direction and management of the QAD.

d. Provide a working interface and communication with the TMI Generation organizations, consultants, contractors, vendors, and others with respect to QA matters. Additionally, in conjunction with the licensing organization, he shall provide a working interface and communications with the NRC with respect to QA matters.

e. Provide, as applicable, planned and periodic audits, monitoring, surveillance, and inspections of organizations, contractors, and vendors performing work functions important to safety.

f. Establish and assure the continuous implementation of an indoctrination and training program for QA and QC personnel and assure that a quality assurance indoctrination is provided to appropriate personnel outside the Quality Assurance organization.

g. Issue periodic reports to the Director-Reliability Engineering, and the Director-TMI Unit 1 on the status of quality activities, and bring to thier attention immediately any significant quality-related problem or deficiency.

h. Provide for quality assurance review and acceptance of design and engineering documents, as delineated in the detailed procedures.

i. Provide for quality assurance review and acceptance of procurement documents generated for the acquisition of materials and services within the scope of the program.

j. Provide for and maintain quality assurance records generated by QAD until such time as they are turned over to document control for storage.

The Manager-Quality Assurance shall have, as a minimum, a baccalaureate degree in Engineering or Science, with at least five years of experience in nuclear power plant operations and supporting activities.

1.2.6.2(a) Quality Assurance Design and Procurement Manager

The Quality Assurance Design and Procurement Manager is responsible for establishing quality programs and inspection requirements in support of design and procurement activities in compliance with the TMI Quality Assurance Program. These activities include, but are not limited to:

a. Review and approve contractor and vendor quality programs for those supplying services or items important to safety.

b. Reporting quality trends to his supervisor and to the cognizant purchasing or contract manager.

c. Review and accept design control procedures prepared by other TMI organizations when these procedures control or exercise an effect upon systems, components, or activities important to safety.

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1.2.6.3(b) Quality Assurance Manufacturing Assurance Manager

The QA Manufacturing Assurance Manager is responsible to perform the necessary postaward quality-related activities in compliance with the TMI Quality Assurance Program required to assure that vendor products are designed, manufactured, tested and/or inspected in accordance with the procurement specifications. These activities include post-award surveys and surveillances, and source inspections.

He is responsible for the coordination with the QA Modifications/Operations Section to assure that Manufacturing Assurance discrepancies are available to the receiving inspectors and the cognizant purchasing or contracts manager. Additionally, he is responsible for providing the Design/Procurement Assurance Section with the results of Manufacturing Assurance activities and recommendations relative to the acceptability of a vendor.

1.2.6.4(c) Quality Assurance Modifications/Operations Manager

The Quality Assurance Modifications/Operations Manager is responsible for monitoring the implementation and effectiveness of the Quality Assurance Program onsite. These activities include the establishiment of adequate site monitoring and inspection programs necessary to verify conformance to Quality Assurance Program requirements. In addition, he is responsible to review site procedures from a QA/QC standpoint and to provide nondestructive examination support for TMI. He reports directly to the Manger of Quality Assurance and, he periodically reports on the implementation and effectiveness of the Operational Quality Assurance Program to the Director - TMI Unit 1. He has the authority and organizational freedom to identify quality assurance problems, provide or recommend solutions, and verify implementation of solutions. He has the authority to stop work on all important to safety activities associated with the onsite TMI Nuclear Station Operational QA program. He is responsible to notify appropriate TMI Station management and

the Manager of Quality Assurance immediately of any condition that warrants operational shutdown of a nuclear unit as defined in appropriate QAD procedures. The Quality Assurance Modifications/Operations Manager is assisted in carrying out his responsibilities by an Operations Quality Assurance Supervisor, a Quality Control Manager and their associated staffs located onsite.

1.2.6.5(d) Quality Assurance Methods/Operations/Audits Manager

The Quality Assurance Methods/Operations/Audits Manager is responsible for maintaining the Quality Assurance Plan and all those procedures applicable to the activities of the QAD. He is responsible, therefore, for coordinating the activities associated with the requirements of the Quality Assurance Plan, including interpretations. In addition, he is responsible for coordinating Quality Assurance indoctrination and training to employees and contractors conducting independent evaluations and assessments of the Program's implementations by performance of internal audits.

The Quality Assurance Methods/Operations/Audits Manager maintains a full-time staff of quality assurance engineers and qualified quality auditors at both the corporate and site offices. The audit activities and the results of the audits are provided to the audited organization and to the Safety Review Groups who provide the management assessments of the significance of the audit findings and the effectiveness of the Quality Assurance Program.

1.2.6.6(e) Materials Technology Manager

The Materials Technology Manager directs and supervises the offsite engineering organizations which have the responsibility for activities in the establishment of requirements for welding, inservice inspection, materials, and materials evaluations. Materials Technology provides NDE and ISI program flow analysis and reporting, technical requirements for repair and repair program and related

corrective action recommendations to Engineering. Additionally, he has a staff function to support manufacturing and the evaluation of system materials technology problems. He is directly responsible for the implementation and compliance with the Quality Assurance Program requirements applicable to his areas of responsibility.

The specific disciplines included in the Materials Technology sections are:

a. Nondestructive Examination

b. Inservice Inspection

c. Materials Engineering

d. Welding Engineering

e. Metallurgical Analysis

1.2.6.7.1 Minimum Qualifications of Quality Assurance Personnel

The qualification requirements and experience levels for key Quality Assurance personnel are such as to assure competence commensurate with the responsibilities of the position. Quality managers and supervisory personnel are required to have a degree in Engineering (or equivalent) and experience in a position having responsibility for the performance of quality activities. The degree requirement may be waived for personnel with exceptional qualifications and a minimum of seven (7) years related experience.

1.2.6.8 Manager-System Laboratory

The Manager-System Laboratory is responsible for the administration and operation of the Environmental and Operational Chemistry Analyses Section of the System Laboratory in compliance with the TMI Quality Assurance Program. This section provides the centralized laboratory analyses services for TMI. The specific responsibilities of the Manager-System Laboratory include the following:

a. Perform analysis of water and wastewater samples submitted by the generating station,

b. Prepare calibration standards for the labs at the generating stations,

c. Monitor the analysis capabilities of the labs at the generating stations through audits and independent analysis of samples,

d. Assist station personnel in unusual operations such as chemical cleaning,

e. Provide consultation on equipment startup and performance, as requested,

f. Perform chemical analysis of fuels, lubricants, insulating fluids and ion exchange resins,

g. Ensure that the laboratory is adequately staffed and that the laboratory personnel are adequately trained and qualified to perform their assigned tasks,

h. Ensure that the System Laboratory meets the applicable requirements of the Quality Assurance Program,

i. Develop and implement laboratory procedures covering the control of the laboratory activities and the records documenting the results of the analysis.

j. Provide support to Materials Technology regarding material specimen preparation and testing.

1.2.6.9 Manager-Nuclear Safety Evaluation

The Manager-Nuclear Safety Evaluation is responsible for the development, direction and supervision of the Nuclear Safety Evaluation Department in compliance with the TMI Quality Assurance Program. The function of this group is to review the broad range of activities, practices and conditions which may have an adverse effect on quality, to assess the safety significance of the conditions and to

make recommendations to the appropriate levels of management for corrective action to preclude repetition. The activities of the Nuclear Safety Evaluation Department will include the receipt and review of all documents and reports identifying conditions adverse to quality (Audit Reports, Nonconformance Reports, Surveillance/Inspection Reports, Reportable Occurrences, NRC Inspections, etc.); analyzing the conditions reported, both individually and collectively; identifying the safety significance of the conditions and reporting, at intervals not to exceed six (6) months, the results of the evaluations to the Senior Vice President of Met-Ed and the Director-TMI Unit 1. Additionally, the Nuclear Safety Evaluation Department working with Systems Engineering will evaluate the operational experience of other nuclear power stations to improve plant operational status and derive benefit from other stations experience.

1.2.7 Vice President-Materials Management

The Vice President-Materials Management, who reports to the President-GPUSC, via the Executive VP GPUSC, is responsible for assuring that the technical and quality requirements, as established by the Generation Group, are incorporated into procurement documents. He is directly responsible for the compliance and implementation of the TMI Quality Assurance Program with regard to procurement activities. His responsibilities will be administered through the following:

Director-Materials Management Systems

Manager-Contracts, TMI Site

1.2.7.1 Director-Materials Management Systems

The Director-Materials Management Systems is responsible for the development and management of a procurement system in compliance with the TMI Quality Assurance Program including:

a. the establishment and implementation of a procurement control process,

b. the incorporation of quality assurance program requirements, as identified by engineering documents, into procurement documents,

c. the coordination of quality assurance activities in the procurement process.

He reports administratively to the GPUSC Vice President - Materials Management, and functionally to the Senior Vice President - Met-Ed for the priorities accorded to the TMI Generation Group's requirements. His responsibilities will be adminisered through the following:

Manager - Field Warehousing, TMI

Manager - Field Procurement, TMI

1.2.7.1.1 Manager-Field Warehousing, TMI

The Manager-Field Warehousing, TMI is responsible for maintaining an inventory, initiating requisitions for inventory reorder, receiving both direct turnover and inventory items, maintaining adequate storage space and facilities, and issuance of material from storage.

1.2.7.1.2 Manager-Field Procurement, TMI

The Manager-Field Procurement, TMI is responsible for all TMI purchasing and expediting activities including the following:

a. the implementation of an approved procurement control process,

b. the receipt, review, recording and tracking of purchase requisitions,

c. the incorporation of engineering requirements into purchase orders,

d. compliance and implementation of the TMI Quality Assurance Program with regard to his areas of responsibility.

e. the preparation and document control of all purchase orders including those for contr

1.2.7.2 Manager-Contracts, TMI Site

The Manager-Contracts, TMI Site is responsible for all site-related contracts with respect to the bidding, bid evaluation, award and administration of construction, maintenance, water processing, decontamination, and waste removal contracts, and those for various technical and other consulting services. He is additionally responsible for the evaluation, validation, dismissal or negotiation, where warranted, of vendor extras, delays and other claims and proposals. He is directly responsible for the compliance and implementation of the TMI Quality Assurance Program with regard to his areas of responsibility.

1.2.8 Manager-Management Services

The Manager-Management Services, who reports directly to the Senior Vice President Met-Ed, is responsible for the administrative control of the Generation Group in compliance with the TMI Quality Assurance Program. In this capacity, he provides support activities relating to the Quality Assurance Program. The specific responsibilities of the Manager-Management Services include the following:

a. Delegating the preparation of the Generation Group Administrative Procedures, coordinating their review, approving procedures and ensuring their implementation in accordance with Appendix B of this Plan.

b. Establishing, supervising and operating the TMI Generation Group Document Centers in accordance with Generation Group procedures.

c. Establishing an effective interface between the Corporate and Station Document Centers.

d. Supervising and directing the activities for maintenance and control of the off-site TMI Generation Group records.

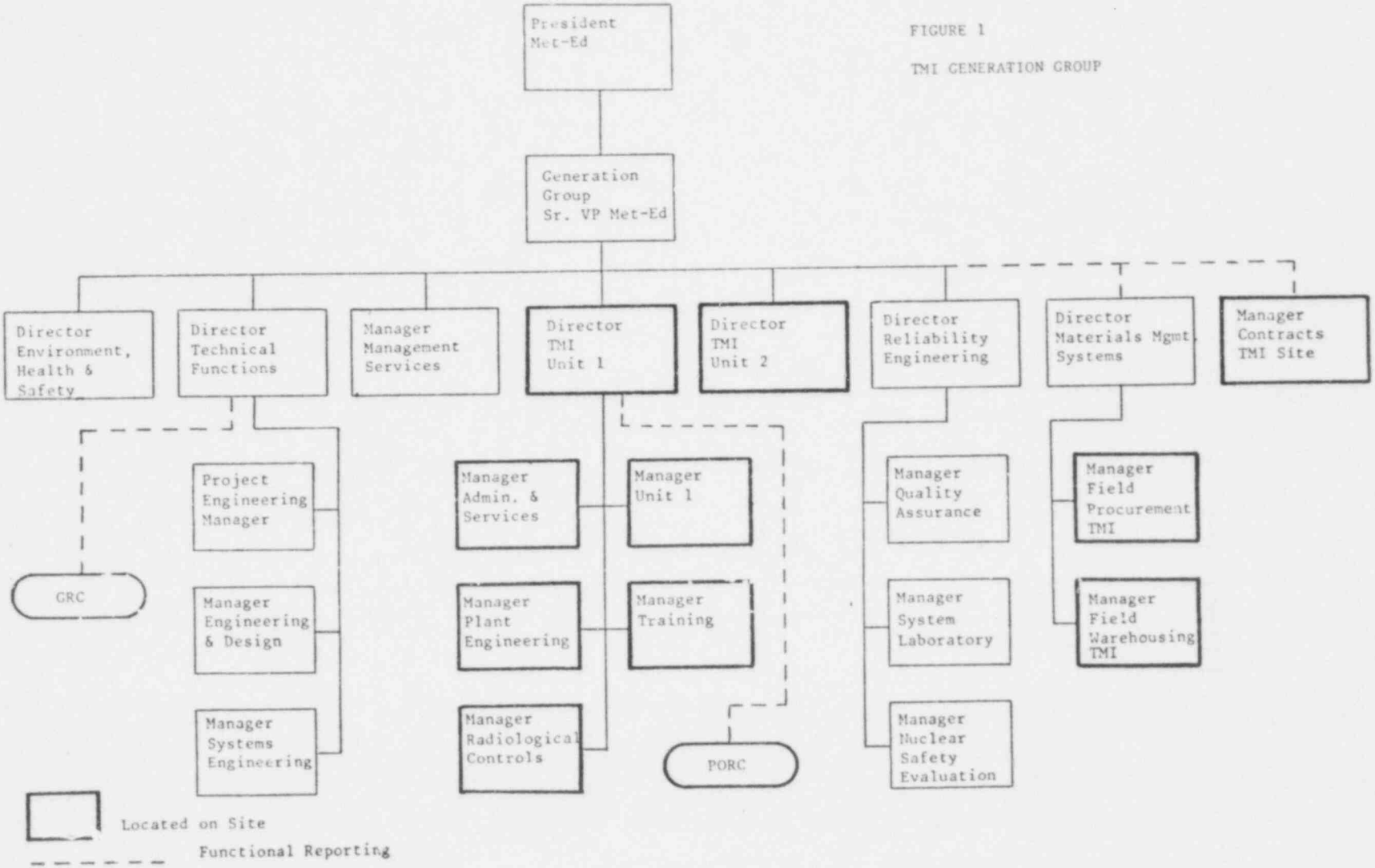
e. Coordinating personnel management development and professional training.

The Manager-Management Services accomplishes these responsibilities via the direct supervision of the activities of the Supervisor-Accounting and Budgets, and Supervisor-Corporate Records Management.

The Manager-Management Services and his staff give full support to the TMI Quality Assurance Program set forth herein, thereby assuring that all work performed under their activity will conform to and support the requirements as applicable to their activity.

FIGURE 1

TMI GENERATION GROUP



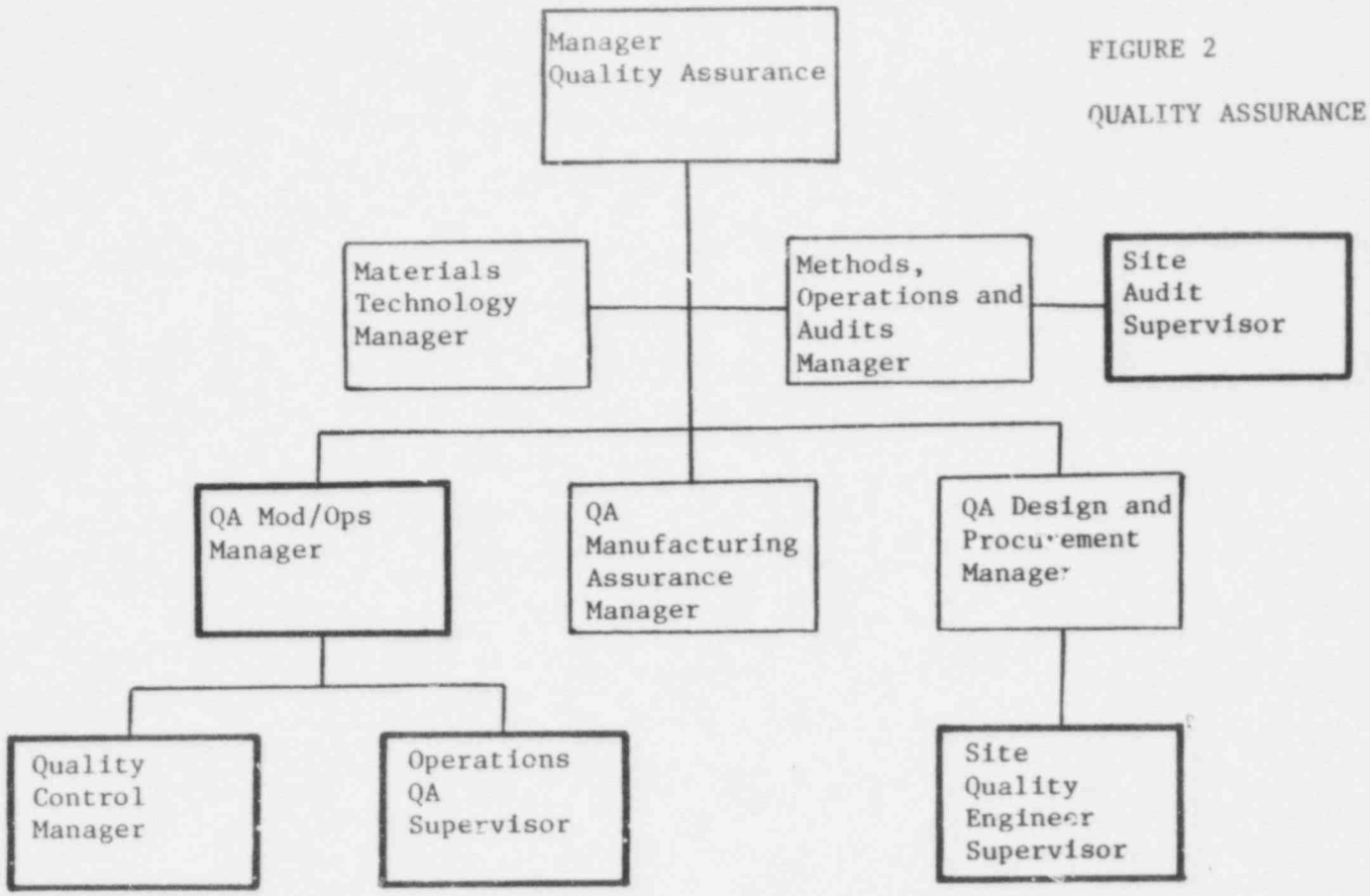


FIGURE 2

QUALITY ASSURANCE DEPARTMENT

Located on Site

2.0 Quality Assurance Program

2.1 Policy

2.1.1 General

The TMI Operational Quality Assurance Program has been established to provide overall quality assurance of operations activities within the scope of the program. Adherence to the requirements of the TMI Operational Quality Assurance Program is mandatory for all TMI organizations and for all contractors or vendors providing items or services covered under the scope.

2.1.2 Scope

The scope of the TMI Operational Quality Assurance Program includes all items and activities considered to be "important to safety." This term is intended to be broader than "safety-related" and encompasses structures, systems, and components (including nuclear fuel and radwaste) which have been designated as Safety-Related, Safety Class, IEEE Class IE, Seismic Category 1 Fire Protection. The scope of the Program will include all items required by the following:

a. Title 10, Code of Federal Regulations, Part 50, Appendix A "General Design Criteria for Nuclear Power Plants"

b. Title 10, Code of Federal Regulations, Part 50, Appendix B "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"

c. Title 10, Code of Federal Regulations, Part 71, Appendix E "Quality Assurance for Shipping Packages for Radioactive Material"

d. United States Nuclear Regulatory Commission Regulatory Guide 1.33 "Quality Assurance Program Requirements (Operation)"

e. United States Nuclear Regulatory Commission Regulatory Guide 1.143 "Design

Guidance for Radioactive Waste Management Systems, Structures and Components Installed in Light Water Cooled Nuclear Power Plants"

f. United States Nuclear Regulatory Commission Regulatory Guide 1.120 "Fire Protection Guidelines for Nuclear Power Plants"

The Program also includes certain non-safety related items when designated by engineering.

Activities which are important to safety shall include, but not be limited to, those activities covered by Appendix A of Regulatory Guide 1.33 and ANSI N18.7.

In addition, the requirements of other Regulatory Guides applicable to operations, maintenance, modification, repair, and refueling of a nuclear power plant were considered. Met-Ed position with regard to these Regulatory Guides is included as Appendix C. The determination that an item or activity is, or is not, important to safety is a design decision governed by approved engineering procedures. Items and activities determined to be important to safety are defined as those items on the Quality Classification List (QCL) and those activities covered by procedures which have been designated during the review cycle as "important to safety." The QCL may also be utilized to record information regarding the level of Quality Assurance Program deemed appropriate for a particular item or service.

This information is not required, but may be utilized to facilitate procurement and implementation. For new design efforts, such as plant modifications and new construction, the classification determination is recorded on design criteria documents. New items will be included in the next appropriate revisions to QCL.

2.1.3 Quality Assurance Plan

This Quality Assurance Plan is the primary document which provides a description of the program. The program is authorized by the

President of Met-Ed and GPUSC to assure that the appropriate levels of management, as designated herein, are directed to implement the program. The Plan is controlled to assure that only the latest approved revision is implemented. The Plan is implemented by approved detailed procedures and instructions.

The purpose of this Plan is to establish the principles which, when implemented, will provide that level of quality assurance which is appropriate to each activity affecting quality. It is recognized that the degree of management control or quality assurance to be applied varies with different systems and activities, and the degree of applicability of any specific item in this Program will differ from system to system and activity to activity.

The degree to which the requirements of this Plan and its implementing procedures are applied will be based upon the following:

a. The importance of a malfunction or failure of the item to safety;

b. The design and fabrication complexity or uniqueness of the item;

c. The need for special controls and surveillance or monitoring of processes, equipment and operational activities.

d. The degree to which functional compliance can be demonstrated by inspection or test; and

e. The quality history and degree of standarization of the item or activity.

The TMI Generation Group is committed to a comprehensive Quality Assurance Program consisting of a three level approach to assure satisfactory and complete implementation of the program commensurate with its requirements for safety and performance. The Program's foremost considerations are the protection of the general public's health and safety. The three level approach is defined below:

Level 1 - activities at this level include independent inspections, checks and tests.

This level of activity may be performed by the Operations Department by surveillance tests, calibration of instruments, radiation surveys, analyses of samples, etc., the Quality Control Section by receipt inspection or inspections of modification or corrective maintenance activities, or by contractors as part of their scope of work. In all cases, the activity is performed by individuals knowledgeable of the activity being performed and qualified to perform the work. Checklists or data sheets are also used for documenting the results of the activity and for providing a permanent plant record of the performance of the activity.

In all cases where the first level activities involve inspection for purposes of acceptance and/or verification of modifications to safety systems, the activity will be performed by personnel who are independent of those performing the work.

Level II - the activities at this level are primarily those of surveillance or monitoring and are performed as deemed necessary by the QA Modifications/Operations, QA Design and Procurement or QA Manufacturing Assurance Sections. The level of surveillance/monitoring applied is consistent with the importance of the item to safety. For activities, whereby Quality Control is performing first level inspection, no second level activity will be required.

At this level procedures and instructions are established and surveillance records will be completed and maintained. Such surveillance/monitoring normally includes observation of quality control tests and inspections, observation of significant operations, review of records, verifications of test reports, and direct inspection on a spot check basis. The organizations performing this activity have the levels of authority, the lines of internal and external communication for management direction, and the properly trained personnel for implementation of these activities.

Level III - the purpose of this level of activity is to assure through a comprehensive program of review and auditing that the first and second levels of the program are properly functioning. The purpose of this level is also to establish that all other organizations including Operations, Maintenance, Engineering, Materials Management, etc. are properly satisfying all the requirements of the Operational Quality Assurance Program.

At this level procedures and instructions are established including the use of comprehensive checklists for documentation of the audit or third level activity in accordance with requirements of ANSI N45.2.12. Qualified audit personnel are included that satisfy the requirements of ANSI N45.2.23. Additional technical experts from areas with administrative reporting outside the function that is being audited will be included as the Audit Team Leader deems necessary. The organization performing this activity has sufficient authority and lines of internal and external communications for obtaining the necessary management direction.

Appendix A is included to provide a comparison of the sections of the Plan with the requirements of 10CFR50, Appendix B, 10CFR71, Appendix E, ANSI N18.7, and ANSI N45.2.

2.1.4 Quality Assurance Program Review

The TMI Quality Assurance Program effectiveness and implementation is periodically evaluated by independent review groups reporting to TMI Generation Group management. These groups provide safety review reports and operational methods. These groups each have technical expertise necessary to support their areas of concern. The independent review committees and operational review groups include the Generation Review Committee, the Plant Operations Review Committee and the Nuclear Safety Evaluation Department. In addition, the Quality Assurance Department conducts activities which provide management with additional information pertaining to effectiveness and implementation.

2.1.5 Training

The TMI Quality Assurance Program includes requirements for formal training programs for personnel performing or verifying activities important to safety.

2.2 Requirements

2.2.1 Quality Assurance Plan

The Operational Quality Assurance Plan and any significant revisions shall be approved by the following:

Sr. VP Met-Ed/VP GPUSC

VP - Materials Management

Director - TMI Unit 1

Director - Reliability Engineering

Manager - Quality Assurance

The Plan includes a Statement of Policy which is signed by the Presidents - GPUSC and Metropolitan Edison. The Statement of Policy provides authorization and evidence of management commitment to the Quality Assurance Program.

Plan revisions which represent significant changes or personnel reassignments of a substantive nature shall be submitted to the Nuclear Regulatory Commission for approval. The Manager-Quality Assurance is responsible for notifying the NRC of all changes to the Plan within 30 days of the change and for obtaining the required approvals prior to issuance.

Plan revisions not considered by the Manager - Quality Assurance to be significant can be issued with approval of the Manager - Quality Assurance and the Director - TMI Unit 1.

Copies of Quality Assurance Plan may be distributed as "Controlled" or "Uncontrolled" copies in accordance with the requirements established in Section 3.

2.2.2 Classification

The TMI Operational Quality Assurance Program applies to all items on the QCL and activities designated as "important to safety." The QCL will be periodically updated to include new plant modifications or construction, or any changes in classification. The list will be treated as a controlled document.

The QCL will normally list systems and components, but not parts. For procurement of spare or replacement parts, classification will be on a case by case basis. The determinations will not necessarily be added to the QCL. An approved engineering evaluation shall be documented and maintained as a quality assurance record. This does not apply to those items which were originally specified as commerical quality.

2.2.3 Regulatory Commitments

A listing is maintained of commitments to regulatory requirements. Each new or revised USNRC Regulatory Guide will be evaluated for applicability and acceptability to TMI. The TMI Generation Group position on each is documented stating the method and degree of compliance or the justification for lack of compliance.

2.2.4 Safety Review

Safety review groups have been established with primary responsibilities for review of operational phase activities. In addition to performing regulatory required reviews, these groups provide management with visability and recommendations for improved plant safety.

2.2.4.1 Generation Review Committee (GRC):

The GRC is an off-site organization reporting to the Director-Technical Functions. This group is responsible to provide independent safety review of operations, nuclear engineering, chemistry and radiochemistry, metallurgy, nondestructive examination, instrumentation and control, radiological safety,

mechanical and electrical engineering, radwaste, administrative controls, quality assurance and other appropriate fields associated with the unique characteristics of TMI. The GRC is responsible for reviewing the following specific subjects:

a. Written safety evaluations of changes in the facility as described in the Safety Analysis Report, changes in procedures as described in the Safety Analysis Report and tests or experiments not described in the Safety Analysis Report which are completed without prior NRC approval under the provisions of 10 CFR 50.59(a)(1). This review is to verify that such changes, tests or experiments did not involve a change in the technical specifications or an unreviewed safety question as defined in 10 CFR 50.59 (a)(2).

b. Proposed changes in procedures, proposed changes in the facility, or proposed tests or experiments, any of which involves a change in the technical specifications or an unreviewed safety question as defined in 10 CFR 50.59(c). Matters of this kind shall be referred to the GRC by PORC following its review, or by other functional organizational units within the TMI Generation Group, prior to implementation.

c. Changes in the technical specifications or license amendments relating to nuclear safety prior to submittal to the Commission for approval and prior to implementation, except in those cases where the change is identical to a previously reviewed, proposed change.

d. Violations, deviations and reportable events which require reporting to the NRC in writing within 24 hours, such as:

    1. Violations of applicable codes, regulations, orders, technical specifications, license requirements, internal procedures or

instructions having safety significance.

2. Significant operating abnormalities or deviations from normal or expected performance of plant structures, systems, or components important to safety.

3. Reportable events, which require reporting to the NRC in writing within 24 hours, as defined in the plant technical specifications.

Review of events covered under this subsection shall include the results of any investigations made and the recommendations resulting from such investigations to prevent or reduce the probability of recurrence of the event.

e. Any other matter involving safe operation of the nuclear power plant which an idependent reviewer deems appropriate for consideration, or which is referred to the independent reviewers by the onsite operating organization or by other functional organizational units within the owner organization.

2.2.4.2 Plant Operations Review Committee (PORC):

The PORC is an on-site operations review organization functionally reporting to the Director-TMI UnIt 1. This group screens subjects of potential concern by reviewing and/or performing preliminary investigations of subjects requiring independent review.

PORC shall provide, as part of the normal duties of plant supervisory personnel, timely and continuing monitoring of operating activities to assist the Director - TMI Unit 1 in keeping abreast of general plant conditions and to verify that the day-to-day operating activities are conducted safely and in accordance with applicable administrative controls. These continuing monitoring activities are considered to be an integral part of the routine supervisory function and are important to

the safety of plant operation. The Director - TMI Unit 1 in carrying out his responsibility for overall safety of plant operations, shall be responsible for timely referral of appropriate matters to management and independent reviewers.

2.2.4.3 Nuclear Safety Evaluation Department (NSED):

NSED is an independent organization reporting to the Director-Reliability Engineering. It perform evaluations and investigations as assigned by Director-Reliability Engineering. NSED evaluates information from external sources for applicability to TMI. They are responsible for evaluations of hardware and software systems which affect the safe reliable operation of the plant. NSED personnel support the activities of GRC and PORC and contribute as requested. They interface with the QAD audit section to assure complete coverage and utilization of the audit program.

2.2.4.4 Quality Assurance Department:

The normal audit program conducted by the Quality Assurance Department and described in Section 9.0 also provides management with assessment of program status and effectiveness.

2.2.5 Indoctrination and Training

Indoctrination and training programs are established for both on-site and off-site personnel performing important to safety activities by the organizational units responsible for the activities. These programs are implemented by appropriate training plans and procedures which assure that:

a. Personnel responsible for performing important to safety activities are instructed as to the purpose, scope, and implementation of manuals, procedures, and instructions;

b. Personnel performing important to safety activities are trained and qualified in the principles and techniques of the activity being performed;

c. Proficiency of personnel performing important to safety activities is maintained by retraining, re-examining, or recertifying;

d. The scope, method and objective of the training is documented;

e. Records of training sessions are prepared and maintained, including identification of the content, the attendees, and the date the training was conducted.

2.3 Responsibilities

2.3.1 Sr. Vice President - Met-Ed

The Senior Vice President - Met-Ed shall regularly assess the scope, status, adequacy and compliance of the Quality Assurance Program to the requirements of 10 CFR 50, Appendix B. This assessment shall be the combined result of:

a. Frequent contact with Quality Assurance Program status through attendance at meetings, and review of periodic status reports on the effectiveness and implementation of the Quality Assurance Program.

b. Performance at least once a year of a preplanned and documented assessment of the effectiveness of the Quality Assurance Program to assure that the program meets regulatory requirements, and the policies and directives of TMI. This assessment may be performed utilizing the safety review groups, an independent consultant, or his own staff. Any corrective action which may be deemed necessary as a result of these assessments shall be formally identified and tracked through resolution.

2.3.2 Director - Reliabilty Engineering

The Director-Reliability Engineering has overall responsibility for establishment of the Operational Quality Assurance Program. He

also has overall responsibiliity for establishment and management of the Nuclear Safety Audit Department and the Quality Assurance Department, including Methods/Operation/Audit Section. He shall provide periodic status reports to the Sr. Vice President Met-Ed on the Quality Assurance Program.

2.3.3 Manager - Quality Assurance

The Manager-Quality Assurance, has the direct responsibility for verifying the effective implementation of the Quality Assurance Program. He shall establish and implement a formally documented and procedurally controlled program to evaluate and report to the Director-Reliability Engineering on the adequacy and continued effectiveness of the overall TMI Operational Quality Assurance Program. Reports of audits performed by the Quality Assurance Department or their agents, and quality trend analyses based on nonconformance and deficiency reports will provide the basis for this evaluation. Corrective action shall be implemented by responsible management as deemed appropriate when analyses reveal adverse quality trends. These actions may involve specific actions to provide compliance with the Quality Assurance Program, and may include follow-up system attribute audits and even revision to the TMI Operational Quality Assurance Program. Implementation and closeout of corrective actions shall be effectively monitored by the Manager-Quality Assurance to assure timely correction and compliance.

The Manager-Quality Assurance is responsible for the contents of Quality Assurance Plan and for ensuring that the Quality Assurance Plan is modified and updated as standards, regulation, requirements and experience dictate. Proposed revisions to the Plan may be suggested by TMI Generation Group personnel by submitting the request, in writing, to the Manager-Quality Assurance for review and action, as applicable. The Manager-Quality Assurance is responsible for the monitoring, surveillance and auditing of Quality Assurance Program implementation.

He is also responsible to provide the required training and qualification of Quality Assurance Department personnel.

2.3.4 Manager - Engineering & Design

The Manager of Engineering & Design is responsible for development and maintenance of the QCL. He solicits input and coordinates with affected organizations to assure a uniform approach to classification of items and activities important to safety.

2.3.5 Generation Group Directors and Managers

Management personnel in each department are responsible for Quality Assurance Program implementation. They are further responsible for development of procedures, for scope of involvement, for activities important to safety, and for training and indoctrination of personnel.

2.3.6 External Organizations

Quality Assurance Programs and implementing procedures for suppliers or contractors providing materials and services for the TMI Nuclear Station which are covered under the scope of this Quality Assurance Program shall be subject, when required, to review and acceptance by the Quality Assurance Department prior to the commencement of any important to safety activity. Procurement documents shall require, and the Quality Assurance Department shall assure, through their review and audit, that supplier and contractor Quality Assurance programs comply with the commitments of this document.

2.4 Resolution of Disputes

Resolution of disputes involving quality, arising from a difference of opinion between QA/QC personnel and other organization (engineering, procurement, manufacturing, construction, operation, maintenance, etc.) personnel shall, if possible, be accomplished at the level such disputes occur. If this is not possible the difference of opinion shall be escalated through supervisory/management

levels until resolution is achieved. The Manager-Quality Assurance shall be the arbitrator on differences of opinion involving conformance of items, components and systems to specified requirements and interpretation of the Quality Assurance Program. The Sr. VP Met-Ed/VP GPUSC and/or Vice President - Material Management shall be the final arbitrator.

3.0 Control of Documents and Records

3.1 Instructions, Procedures, and Drawings

3.1.1 Policy

The TMI Quality Assurance Program requires that activities important to safety be prescribed by documented procedures, instructions, and/or drawings and that these quality-affecting activities be accomplished through the implementation of these documents.

3.1.2 Requirements

TMI procedures, instructions, and/or drawings which prescribe the performance of activities important to safety shall:

a. Include quantitative (such as dimensions, tolerances, and operating limits) and qualitative (such as workmanship samples) acceptance criteria sufficient for determining that important activities have been satisfactorily accomplished.

b. Require approval of appropriate personnel prior to the initiation of the quality-affecting activity.

c. Describe the sequence of action to be accomplished.

d. Define the responsibilities and authorities of personnel performing the activity.

e. Describe interfaces with other company elements or other organizations.

f. Require indoctrination of user personnel prior to implementation.

g. Be distributed in a controlled manner to preclude the use of obsolete documents.

h. Be distributed with sufficient controlled copies to assure availability to responsible personnel.

3.1.3 Responsibilities

3.1.3.1 Department Managers

The Manager of each department performing activities important to safety is responsible for the preparation, approval and implementation of procedures, instructions and/or drawings. He is responsible to assure that provisions are made for interface controls for internal and external lines of communications among participating organizations and technical disciplines. Additionally, he is responsible to insure that the procedures reference the documents used in their preparation and the extent to which the procedures meet the requirements of the references.

3.1.3.2 Quality Assurance Department

The QAD shall review and approve those administrative policies, procedures, instructions and/or drawings which delineate the methods of complying with the requirements of this manual.

Contractor Quality Assurance program documents specified in the applicable procurement documents shall be reviewed and accepted by QAD. Compliance with detailed procedures and instructions shall be audited at specified frequencies.

Vendor Quality Assurance Plans/Manuals, special process procedures, and inspection and test procedures shall be reviewed and approved by QAD prior to releasing the vendor to implement such documents. Contractor Quality Assurance Plans/Manuals, work plans, selected drawings, instructions and procedures shall be reviewed and approved by QAD prior to releasing the contractor to start work. Adequacy shall be verified by audit and inspection programs.

3.1.3.3 TMI Unit Management

The Management of each TMI Unit is responsible for assuring that instructions, drawings, and procedures associated with the administrative controls, operation, fuel handling, inservice inspection, calibration, maintenance, modification, repair and operational testing of

structures, systems, and components important to safety are prepared, reviewed, approved and in accordance with approved written procedures which conform to the requirements of the TMI Quality Assurance Program. All activities important to safety accomplished by the plant staff shall be performed in accordance with approved procedures, instructions, or drawings.

3.1.3.4 Delegated Authorities

Those activities important to safety which are performed by contractors, agents, contractors, or vendors shall be delineated by documented, approved, and controlled procedures, instructions or drawings.

3.2 Document Control

3.2.1 Policy

Measures shall be established and documented to control the issuance of documents, such as program documents, design documents, instructions, procedures, and drawings, including changes thereto, which prescribe activities important to safety. These measures shall assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel and are distributed to, and used at, the location where the prescribed activity is performed.

3.2.2 Requirements

Written document control procedures shall be established to provide for control of the following documents as a minimum:

a. As-built Drawings

b. Quality Assurance Plans/Manuals, and Instructions

c. Operating Procedures & Instructions

d. Maintenance Procedures & Instructions

e. Design Documents (e.g., calculations, drawings, specifications, analyses) including documents related to computer codes.

f. Manufacturing, Construction and Installation Drawings

g. Manufacturing, Construction Modification, Installation, Test, and Inspection Procedures and Instructions

h. Procurement Documents

i. FSAR and Related Design Criteria Documents

j. Nonconformance Reports

k. Design Change Documents

l. Test Specifications

m. Operating and Special Orders

n. Equipment & Material Control Procedures

o. Refueling Procedures

p. QCL

q. Topical Reports

All procedures established for document control shall meet the following requirements:

a. Review, approval and issuance criteria for documents and their revisions shall be specified to assure adequate technical and quality requirements are met prior to issue.

b. The individuals or elements responsible for reviewing, approving and issuing documents and their revision shall be specified.

c. Changes must be documented, approved and included in the appropriate revision document prior to being implemented.

d. Revisions shall be reviewed and approved by the same organizations that performed the original review and approval or by other qualified, responsible, and designated organizations.

e. Document distribution must be sufficient to assure that the documents are readily available at convenient locations to plant personnel prior to commencement of work.

f. Appropriate document transmittal and maintenance measures shall be incorporated in document control systems to prevent inadvertent use of voided, superseded or obsolete documents. Holders of controlled documents are responsible for maintaining their assigned copies in a current status. Documents distributed for information only will not be considered a controlled copy, and, as such, will not be used in performing an activity important to safety since they will not be maintained current. Exceptions to this requirement must be approved, in writing, by QAD.

g. A master list or equivalent will be established and maintained to identify the current revision number of instructions, procedures, specifications, drawings, and procurement documents. This list will be distributed to predetermined responsible personnel to preclude the use of superceded documents.

h. Status indication measures shall be established for all controlled documents. Status lists or logs shall be maintained and be made available to project personnel to identify the current revision levels of controlled documents.

i. Maintenance, modification and inspection procedures shall be reviewed by the responsible Quality Assurance organization to determine:

   1. The need for inspection, the identification of inspection personnel and the documentation of inspection results.

2. That necessary inspection requirements, methods, and acceptance criteria have been identified.

3.2.3 Responsibility

3.2.3.1 Manager - Management Services

Responsible to approve the TMI Generation Group procedures for document control.

3.2.3.2 Manager - Administration and Services

Responsible for implementation of the document control system for all instructions, procedures, drawings and other controlled documents prepared for TMI in administration, operation, testing, maintenance, and modification of structures, systems and components important to safety.

3.2.3.3 Manager - Quality Assurance

Responsible for the review and approval of document control procedures for quality assurance requirements and document control measures; to evaluate the document control system effectiveness through review and audit.

3.2.3.4 Department Managers

Responsible to ensure that documents are available when required; to properly review and approve documents such as procedures, instructions, specifications, drawings, etc. to ensure that changes to documents are reviewed and approved by the same organization that performed the original review and approval of the document; to ensure that approved changes are promptly transmitted for incorporation into documents; to ensure that obsolete or superseded documents are eliminated from the system.

3.2.3.5 Delegated Authorities

Vendor, contractor, and agent QA programs shall be reviewed to assure compliance with the requirements of this section.

3.3 Quality Assurance Records

3.3.1 Policy

Quality Assurance records for items and activities covered under the scope of the TMI QA Program shall be identified, reviewed, retained, and retrievable. These requirements are imposed on all organizations performing activities important to safety. Quality Assurance record systems shall be described and controlled by approved written procedures and instructions, adequately implemented, and verified by QAD through inspections and audits.

3.3.2 Requirements

The procedures established for the generation, collection, storage, maintenance, and retrieval of the TMI Quality Assurance records shall meet the following minimum requirements:

a. The applicable design specification, procurement documents, test procedures, operational procedures and other documents shall specify the records to be generated, supplied and maintained by or for the owner. These records shall include results of reviews, inspections, tests, audits, and material analysis; monitoring of work performance; qualification of personnel, procedures, and equipment; and other documentation such as calculations design verifications, drawings, specifications, procurement documents, calibration procedures and reports; nonconformance reports; and corrective action reports.

b. Sufficient records and documentation shall be maintained to provide evidence of the quality of items or activities important to safety. Inspection and test records shall contain the following where applicable:

    1. A description of the type of observation.

2. The date and results of the inspection or test.

3. Information related to conditions adverse to quality.

4. Inspector or data recorder identification.

5. Evidence as to the acceptability or the results.

6. Action taken to resolve any discrepancies noted.

c. Documented and approved measures shall be established for complying with the applicable requirements of codes, standards, and procurement documents regarding record transmittal, retention, and maintenance subsequent to completion of work.

d. Record storage facilities shall be established and utilized to prevent destruction of quality records by fire, flooding, theft and deterioration by environmental conditions such as temperature or humidity in compliance with the applicable standards, codes and regulatory guides endorsed in Section 2 of the TMI Quality Assurance Plan.

3.3.3 Responsibilities

3.3.3.1 Manager - Quality Assurance

a. Responsible for reviewing and approving major participating organizations procedures for the maintenance of Quality Assurance records; establishing a program for the identification, storage, retrieval, and maintenance of Quality Assurance records generated by QAD, until they are turned over for storage, and performing planned and periodic audits to verify adequacy and implementation of Quality Assurance records requirements by both internal TMI organizations and external suppliers.

3.3.3.2 Manager - Administration and Services

a. Responsible for the collection, maintenance, and storage of records at the plant site in accordance with approved written procedures which conform to the requirements and policy of this section.

b. Responsible for providing procedures which ensure the maintenance of records sufficient to furnish objective evidence that activities affecting quality are in compliance with the applicable standards, codes and regulatory guides endorsed in Section 2 of the TMI Quality Assurance Plan.

3.3.3.3 Manager - Management Services

a. Responsible for the collection, maintenance, and storage of records at the home office in accordance with approved written procedures which conform to the requirements and policy of this section.

b. Responsible for providing procedures which ensure the maintenance of records sufficient to furnish objective evidence that activities affecting quality are in compliance with the applicable standards, codes and regulatory guides endorsed in Section 2 of the TMI Quality Assurance Plan.

3.3.3.4 Delegated Authorities

Records generated by site contractors shall be controlled according to contractor procedures until such time as they are turned over to the QAD for review, acceptance, and transmittal to the permanent records file. Purchased equipment records shall be retained by the vendor until the equipment is released for shipment.

When required by the procurement documents, contractors and vendors shall establish procedures to control Quality Assurance records. Implementation of these procedures shall be assured by performance of source surveillance by QAD and through audits performed by QAD.

Records to be submitted with the shipment or retained by the vendor will be specifically identified in procurement documents. These records will be reviewed as necessary by QAD to provide the required degree of confidence in the adequacy of compliance of the vendor with the requirements of this section.

4.0 Design Control

4.1 Policy

Measures shall be established and documented to assure that the applicable specified design requirements, such as design bases, regulatory requirements, codes and standards are correctly translated into specifications, drawings, procedures or instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included or referenced in design documents for design of systems and structures; external design of systems and structures; and assessment of damage.

4.2 Requirements

4.2.1 Design control measures require that:

a. The organizational structure be defined, and authority and responsibility of personnel involved in preparing, reviewing, approving and verifying design documents be delineated.

b. The FSAR design bases, FSAR safety analysis, design regulations, codes and standards and Plant Technical Specifications be adhered to in design work, except where the changes will be the subject of an operating license amendment application.

c. The materials, parts and processes selected by design are reviewed to assure that they are suitable for the intended application, including compatibility of materials, accessibility for inservice inspection, maintenance and repair, associated computer programs, and quality standards. The review will also evaluate suitability with regard to human factors which may effect safe operation.

d. Internal and external design interface controls, procedures, and lines of communication among participating design organizations and across technical disciplines are established and

described for the review, approval, release, distribution, and revision of documents involving design interfaces.

e. Errors and deficiencies in approved design documents, including design methods (such as computer codes) that could adversely affect items and activities important to safety shall be documented, and action shall be taken to assure that these errors or deficiencies are corrected.

f. Deviations in specified quality standards shall be identified and procedures shall be established to assure their control.

g. Review of standard "off the shelf" commercial materials, parts, and equipment for suitability of application with structures, systems, and components important to safety shall be conducted prior to selection.

h. Design verification methods (design review, alternate calculations or qualification testing) shall be established. Guidelines shall be established for determining the appropriate methods.

i. Design verification procedures shall be established which assure the following:

- The verifier is qualified and is not directly responsible for the design.

- Verification shall be complete prior to relying upon the component, system, or structure to perform its function during plant operations.

- Procedural control is established for design documents that reflect the commitments of the SAR. Design documents subject to procedural control include, but are not limited to, specifications, calculations, computer programs,

system descriptions, and drawings, including flow diagrams, piping and instrument systems for major facilities, site arrangements, and equipment locations.

- The responsibilities of the verifier, the areas and features to be verified, the pertinent considerations to be verified, and the extent of documentation shall be identified in procedures.

j. When verifications may be accomplished by test:

- Prototype, component or feature testing shall be performed as early as possible prior to installation of plant equipment, or prior to the point when the installation would become irreversible.

- Verification by test shall be performed under conditions that simulate the most adverse design conditions as determined by analysis.

k. Procedures shall be established to assure that computer codes are verified prior to use.

l. Design and specification changes, including field changes, will be subject to design control measures commensurate with those applied to the original design. Design changes shall be reviewed and approved by the organization responsible for the original design or by another organization with comparable expertise designated to review and approve changes.

m. Measures shall be provided to assure that responsible plant personnel are made aware of design changes and/or modifications, which may affect the performance of their duties.

4.3 Responsibilities

4.3.1 Plant Engineering

The Plant Engineering Department is responsible for providing technical support to operations and maintenance personnel. This is accomplished by providing on-shift engineers to the operations staff for direct technical coverage of the plant systems performance. In addition, Plant Engineering is responsible for in-plant fuel management and accountability, control rod programs, core calculations, providing technical assistance to unit management and the preparation and maintenance of procedures related to the activities of the department.

Plant Engineering is also responsible for engineering activities related to routine maintenance and minor plant modifications.

4.3.2 Project Engineering Manager

The Project Engineering Manager is responsible for coordination, staffing and directing of engineering tasks which are outside of the normal scope of activities of Plant Engineering. To fulfill these responsibilities he will:

a. Maintain a listing of all identified design tasks and the person(s) or organization assigned. For each outside design task, a Cognizant Engineer will be identified.

b. Maintain schedule and status information for each task.

c. Coordinate the efforts of the System Engineering Department and the Engineering and Design Department. Personnel from these departments will be utilized to perform the assigned tasks.

d. Control and coordinate the activities of A/E's providing direct engineering service.

e. Coordinate with the engineering management personnel of contractors with design responsibility.

f. Review and approve baseline design documents such as design criteria, flow diagrams, system descriptions, arrangement drawings, one-line diagrams and logic diagrams, as appropriate.

g. Review and approve final design documents including specifications and drawings (as required).

Note: This design review does not replace or eliminate the need for design verification by the organization who performed the design.

4.3.3 Systems Engineering Department

The Systems Engineering Department is responsible for providing conceptual and analytic engineering service to other engineering groups as required. They are directly responsible for technical administration of nuclear fuel-related engineering activities.

The Systems Engineering Department is responsible for implementation of the design control program for their own activities which are important to safety.

4.3.4 Engineering and Design Department

The Engineering and Design Department provides detailed mechanical and electrical engineering as well as design service to support TMI engineering activities. They are responsible for classification of items and activities important to safety and for preparation and maintenance of the Quality Classification List (QCL). The department is also responsible for implementation of the design control program for their own activities which are important to safety.

4.3.5 Other Design Organizations

All design organizations performing design activities for TMI shall have quality programs

which include design control provisions equivalent to those provided in the TMI Quality Assurance Program.

4.3.6 Quality Assurance Department

The Manager - Quality Assurance is responsible for providing Quality Assurance review and concurrence with design and engineering documents relating to items and activities important to safety to assure that appropriate quality requirements have been included. In addition, Quality Assurance will perform planned and periodic audits of responsible design organizations to verify implementation of design control measures.

5.0 Procurement and Material Control

5.1 Control of Procurement

5.1.1 Policy

Procurement of material, equipment and services which are considered important to safety shall be performed in accordance with written policies, procedures and instructions. These shall establish methods for preparation, review approval, and control of procurement documents and shall provide measures to comply with applicable regulatory requirements. Appropriate measures shall be established to evaluate procurement sources, monitor the activities of consultants, vendors and contractors, and confirm that purchased items conform to procurement document requirements. The programs of all participants shall be in accordance with the applicable requirements of the TMI Quality Assurance Program.

The general and specific requirements for the quality assurance program of all vendors and contractors, including their subvendors and subcontractors supplying material, equipment, or services which are considered important to safety, are delineated by procurement documents. The procurement documents impose quality program requirements that are commensurate with the degree of complexity, the uniqueness, and the importance to safety of the items and services being performed.

Quality Assurance measures shall apply to the procurement of materials including spare parts, replacement parts, off-the-shelf items and consumables. Procurement of spare or replacement parts for structures, systems, and components shall be subject to current Quality Assurance program controls and to codes, standards, and technical requirements equal to, or better than, original technical requirements, or in accordance with an approved engineering document.

5.1.2 Requirements

5.1.2.1 Procurement Documents

The sequence of actions for the preparation, review, approval and control of procurement documents shall be delineated in detailed procedures. These procedures shall delineate requirements to asssure that procurement documents:

a. Specify applicable quality assurance requirements.

b. Require applicable quality program requirements to be passed on to sub-vendors and subcontractors.

c. Specify or reference design bases technical requirements, including applicable regultory requirements, material, and component identification requirements, drawings, specifications, codes and standards, test and inspection requirements, and special process instructions.

d. Identify the documentation to be prepared, maintained, and suomitted for review, approval and record information as applicable.

e. Include an identification of those systems and activities important to safety.

f. Identify those records which vendors or contractors shall retain, maintain, and control; and those which vendors or contractors shall deliver prior to use or installation of the item.

g. Include right of access to vendors or contractors and their subtier vendor and contractor facilities and records for source inspection and/or audit.

h. For spare or replacement parts, contain requirements at least equivalent to those used for the original procurement. The original procurement document may be used as the technical requirements for purchase of spare or replacement parts.

i. Include the provision that suppliers shall refrain from implementing procedures which require owner approval prior to ootaining such approval.

Measures shall be established for the review, approval, and release of procurement documents and subsequent revisions. The reviews shall assure the inclusion of the applicable technical, quality, and administrative requirements in procurement documents prior to their use. Requisitions for professional service agreements for services covered by the scope of this Quality Assurance Program shall be reviewed by the QAD to assure inclusion of quality requirements.

QAD personnel shall review and concur with the adequacy of quality requirements to determine that they are correctly stated, inspectable and controllable; that there are adequate acceptace criteria; and that procurement documents have been processed in accordance with established requirements.

Review of procurement documents shall be documented to provide objective evidence of their approval prior to their release.

5.1.2.2 Qualification and Selection of Suppliers

The TMI Quality Assurance Program requires documented evaluations of prospective suppliers which demonstrate qualifications based upon one or more of the following criteria:

a. Review of performance histories which provide records of suppliers previous capability to provide similar products or services.

b. Review of the supplier's capability to comply with the criteria of 10 CFR 50, Appendix B, applicable to the items or services to be supplied.

c. A pre-award survey of supplier's facilities and Quality Assurance program to determine his capability to supply the items or services that meet the design

and quality requirements of the specification.

Procedures shall be established to accomplish the evaluation and selection of suppliers of equipment, material or services. Contracts or purchase orders for material, equipment or services covered by the scope of the Quality Assurance Program shall be awarded only to vendors or contractors who have been qualified by the QAD as having a Quality Assurance program commensurate with the equipment or services to be provided. When a supplier quality program is required, it shall be reviewed and approved prior to initiation of the activity affected by their program. For certain services, the supplier may be required, by procurement documents, to work under the direct control of the TMI Quality Assurance Program. In these instances, the supplier will not be required to have a separate quality assurance program, but will be required to work within the applicable requirements of this Quality Assurance Program and will require the approval of the Quality Assurance Department.

5.1.2.3 Manufacturing Assurance

Measures shall be established to provide control of manufacturing activities of vendors. These methods shall be described in detailed written procedures. The extent to which these specific controls will be applied to vendors will be described in individual vendor inspection plans. A vendor inspection plans will be prepared for each major contract within the scope of the TMI Quality Assurance Program.

The attributes of the manufacturing assurance program shall include:

a. Provisions for the review, approval and status tracking of the vendor's drawings, Quality Assurance manual and selected manufacturing and quality procedures prior to fabrication. Vendors may not implement procedures until written notice of approval is received, if applicable.

b. Established vendor inspection plans that delineate, as required the hold

and/or witness points in the manufacturing process for specified review, inspection, verification and test.

c. Methods for resolution of nonconformance where the vendor's suggested disposition is "Use-as-is" or "Repair". Such nonconformances require approval by the responsible engineer.

d. Planned and systematic audit and surveillance of vendor quality activities. Scope of coverage and frequency shall be determined by the criticality of the furnished items and the evaluated results of vendor qualifications, including pre-award surveys and quality procedure reviews. Revisions to surveillance plans shall be made as warranted by vendor performance.

e. Control of vendor document package including review for completeness and acceptability. Inadequate records shall be sufficient cause to reject the items furnished due to their indeterminate quality status.

f. Assessments of vendor control of quality shall be made at a frequency and depth commensurate with the importance, complexity and quantity of the items furnished. These assessments shall utilize the qualitative and quantitative information provided by vendor noncompliance documents; surveillance, inspection and audit reports; and receiving inspection and test records.

g. Receiving inspection procedures assure that:

    1. The material, component, or equipment is clearly identified and that the identification and quantity correspond to the information on the shipping documents and quality records.

    2. The item's handling and shipping, requirements have been met by the

vendor and maintained by the carrier.

3. The item's quality record package or compliance certificate is complete, and adequate.

4. Items delivered, which are not in compliance with requirements are documented in accordance with the nonconformance procedure, tagged, segregated (if possible), and prevented from being inadvertently issued for installation or use.

5. Items accepted and released are identified as to their inspection status prior to forwarding them to a controlled storage area or releasing them for installation or further work.

5.1.3 Responsibilities

5.1.3.1 Materials Management

Materials Management is responsible for complying with the requirements of this Plan and for the administration and operation of procurement and warehousing associated with the operation of the TMI Nuclear Station. In this regard, they are responsible for assuring that the technical and quality requirements, as established by the Generation Group, are incorporated into procurement documents without revision. Furthermore, Materials Management is responsible for assuring that the contractual, legal and commercial requirements are incorporated into the procurement documents in a manner which will not alter the technical or quality requirements. The Manager - Field Warehousing, TMI is responsible for the operation and maintenance of the company warehouses and storerooms at the TMI Nuclear Station.

5.1.3.2 Manager - Quality Assurance

The Manager - Quality Assurance is responsible for assuring that QAD procedures for the control of purchased equipment, material, and services are established, approved, implemented and effective. He is also responsible for

the approval of all TMI procedures necessary for the control of purchased equipment, material, and services within the scope of the TMI Quality Assurance Program. He is responsible for approval of suppliers' Quality Assurance Program to the extent required in the procurement documents. He is also responsible for review on acceptance of supplier document record packages. He is responsible for establishing and implementing an adequate program of source inspection, surveillance and receipt inspection to assure supplier compliance with contract requirements.

5.1.3.3 Responsible Engineer

A responsible engineer is that engineer assigned responsibility for the design and/or procurement of each structure, system, or item. He shall review, approve and control procedures, drawings and other quality-related documents submitted by the supplier of the specified equipment. He shall maintain a status reference of all documents requiring approval and distribute such information as required.

5.2 Indentification and Control of Materials, Parts and Components

5.2.1 Policy

Measures shall be established to provide for the identification and control of materials, parts and components important to safety. These measures shall assure that incorrect or nonconforming items are identified and controlled in order to prevent their inadvertant installation or use at TMI. Where required by design documents, the system established shall provide traceability of components from the receipt of material through fabrication and testing. Verification shall include review of objective evidence of inspections and tests which demonstrate that product identification and control is maintained at various stages of manufacture, installation, or erection. Identification requirements shall be specified in the applicable design and procurement documents.

5.2.2 Requirements

a. Identification requirements shall be included in specifications and drawings.

b. Material, parts, and components, including partially fabricated subassemblies or subdivided materials shall be identified to preclude the use of incorrect or defective items.

c. Materials and parts important to safety shall be identified so that they can be traced to the appropriate documentation, including, but not limited to:

1. Specifications
2. Drawings
3. Procurement Documents
4. Physical and Chemical Test Reports
5. Nonconformance Reports
6. Inspection Reports and Checklists
7. Storage Maintenance Instructions
8. NDE Reports
9. Vendor Certificates of Compliance

d. The location and method of identification shall be specified so as not to affect the form, fit, function or quality of the item being identified.

e. Correct identification of materials, parts and components shall be verified prior to release for fabrication, shipping, installation, and testing.

f. Where physical identification is either impractical or insufficient, physical separation, procedural control, or other approved means may be employed.

g. A receipt inspection at the site warehouse verifies that identification for

received items is complete and accompanied by appropriate documentation.

5.2.3 Responsibility

5.2.3.1 Responsible Engineer

a. Responsible for ensuring that procurement documents contain appropriate requirements for the identification and control of materials, parts, or components.

5.2.3.2 Manager - Quality Assurance

a. Responsible for Quality Assurance review and concurrence of procedures for maintaining identification in accordance with the requirements of this section.

b. Responsible for verification of identification during receipt inspection.

c. Responsible for monitoring and conducting inspections, surveillances and audits to verify conformance to the requirements of this section.

5.2.3.3 Manager - Site Warehousing, TMI

a. Responsible for maintaining identification and control of materials, parts or components received and stored at TMI in accordance with written procedures.

6.0 Control of of Station Activities

6.1 Policy

Station activities considered important to safety shall be conducted in accordance with the requirements of this Plan. These activities include design changes, procurement, fabrication, handling, shipping, storage, cleaning, erecting, installation, inspection, testing, operation, maintainance, repair, refueling and modification.

6.2 Requirements

The Quality Assurance requirements for station activities are contained in this Plan and include compliance with applicable USNRC Regulatory Guides and ANSI Standards indicated in Appendix C. These requirements shall be implemented in appropriate TMI procedures governing station activities. The requirements of the Plan apply to all individuals or organizations performing functions which affect the quality of structures, systems, components, or activities important to safety.

6.2.1 Details

The following subsections discuss typical activities which are representative of the broad scope of administrative controls and quality assurance requirements that are applicable to station activities. The organizational structures and functional responsibilities governing station activities shall be structured so that attainment of Quality Assurance Plan objectives is accomplished by those who have been assigned or delegated responsibility for performing the work and verification of conformance to established requirements is accomplished by qualified personnel who do not have direct responsibility for performing or directly supervising the work. Quality Assurance Department activities such as inspection, monitoring, surveillance, reviews and audits are performed to independently verify conformance to this plan, applicable station administration controls, and applicable regulatory and licensing commitments. These independent verifications are applied to station activities on a graded

approach and to the extent necessary to provide adequate confidence that structures, systems, components, and personnel perform satisfactorily to maintain the safety of the station. Station work functions such as routine and abnormal operations, maintenance, repair or rework, in-service inspections, technical specification compliance, fuel handling, radwaste handling, radiation protection, chemical analysis, housekeeping and cleanliness, fire protection, security, training, environmental requirements, health physics, and other activities considered important to safety which are discussed in the Quality Assurance Plan are controlled to an extent consistent with their importance to safety.

6.2.1.1 Control of Inspection

A program for inspection of activities affecting quality shall be established and executed by, or for, the organization performing the activity to verify conformance to the documented instructions, procedures, and drawings for accomplishing the activity. Design specifications, drawings, procedures, or instructions shall include the necessary inspection requirements. These requirements include acceptance criteria and reference to codes, standards, and regulatory documents. These requirements shall be further translated into inspection procedures, instructions, or checklists which shall contain, as required, the following:

a. Identification of characteristics and activities to be inspected.

b. Inspection methods.

c. Identification of organization responsible for performing the inspection.

d. Acceptance and rejection criteria.

e. Identification of applicable revisions or required procedures, drawings and specifications.

f. Documentation of inspection results including identification of the inspector.

g. Listing of necessary measuring and test equipment including their accuracy requirements.

Inspectors (including NDE personnel) shall be qualified in accordance with applicable codes, standards and TMI training programs and their qualification and certification shall be kept current and documented.

Individuals performing inspections shall be other than those who performed or directly supervised the activity being inspected and shall not report directly to the immediate supervisors who are responsible for the work activity being inspected. If the individuals performing inspections are not part of the responsible Quality Assurance organization, the inspection procedures and personnel qualification criteria shall be reviewed and concurred with by the responsible Quality Assurance organization prior to the initiation of the inspection activity. Inspection of activities as defined in ANSI N45.2.10 may be conducted by second line supervisory personnel or by other qualified personnel not assigned first line supervisory responsibility for the conduct of work. These inspections, i.e., those performed by individuals not assigned first line supervisory responsibility, are not intended to dilute or replace the clear responsibility of first line supervisors for the quality of work performed under their supervision. When inspections associated with normal operations of the plant (such as routine maintenance, surveillance and tests) are performed by individuals other than those who performed or directly supervised the work, but are within the same group, the following controls shall be met:

a. The quality of the work can be demonstrated through a functional test when the activity involves breaching a pressure retaining item.

b. The qualification criteria for inspection personnel are reviewed and found acceptable by the Quality Assurance organization prior to initiating the inspection.

Work authorization documents relating to work considered important to safety shall be reviewed by Quality Assurance Department personnel to determine the need for: a) inspection, b) identification of inspection organization, c) identification of inspection witness and hold points, d) documenting inspection results.

When hold points have been establisned, either contractually by procurement or internally by plant procedures, work may not proceed until either inspection is performed or waived by the responsible Quality Assurance organization.

Inspection of modifications, repairs, and replacements shall be by the same method and to the same criteria as the original inspection or by an approved, documented, engineering and QA alternate. Where direct inspection is not practicable, control of processing, equipment and personnel shall be based on statistically valid sampling plans. Inspection personnel shall be provided with suitable equipment and tools, which are calibrated as necessary, and controlled to assure that accuracy requirements are satisfied and that inspections are complete.

Inspection data and results shall be evaluated by designated personnel to assure that the inspection objectives have been met and that items requiring action or follow-up are identified and documented.

Records shall be kept in sufficient detail to provide adequate confirmation of an inspection program.

6.2.1.2 Control of Special Processes

Measures shall be established and documented to assure that special processes are accomplished under controlled conditions in accordance with applicable codes, standards, applications criteria, and other special requirements including the use of qualified personnel

and procedures. Special processes are those that require interim in process controls in addition to final inspection to assure quality including, but not limited to, such processes as welding, heat treating, chemical cleaning, and nondestructive examination. Procedures for special processes shall be established to meet the requirements or applicable codes and standards, where applicable, or to meet the requirements of special process specifications which may be produced for TMI. These procedures shall provide for recording evidence of acceptable accomplishment of special processes. Procedures and instructions for the control of special processes shall be reviewed and approved by qualified personnel. Procedures, equipment, and personnel performing special processes shall be qualified in accordance with applicable codes, standards, and specifications. Organizational responsibilities shall be delineated for the qualification of special processes, equipment and personnel. Qualification records of personnel equipment and procedures associated with special processes shall be established, maintained and kept current. For special processes not covered by the existing codes or standards, or when item quality requirements exceed the requirements of established codes or standards, the necessary qualifications of personnel, procedures and equipment shall be defined.

6.2.1.3 Test Control

A documented test program shall be established to assure that all testing required to demonstrate that the structure, system or component considered important to safety will perform satisfactorily in service. The tests shall be performed in accordance with written, approved, and controlled test procedures which incorporate or reference the requirements and acceptance standards contained in the applicable design documents. The extent of testing shall be based on the complexity of the modification, replacement, or repair. Testing, including proof tests prior to installation and preoperational tests, necessary to demonstrate that structures, systems and components will perform satisfactorily in service, shall be accomplished in accordance with written

approved procedures. These procedures shall be based on requirements and acceptance limits contained in applicable design and procurement documents. These test procedures or instructions shall provide for the following as required:

a. A description of the test objective.

b. Instructions for performing the test, including caution or safety notes in sufficient detail to avoid operator interpretation.

c. Test prerequisites such as calibrated instrumentation, adequate test equipment and instrumentation including accuracy requirements, completeness of item to be tested, suitable and controlled environmental conditions, and trained qualified and licensed or certified personnel.

d. Provisions for data collection and storage.

e. Acceptance and rejection criteria as specified in design and procurement documents.

f. Methods of documenting or recording test data and results, in sufficient detail to prevent misinterpretation.

g. Provisions for assuring that test prerequisites have been met.

h. Mandatory hold or witness points for inspection by TMI Quality Assurance and/or other designated personnel.

i. Provisions for control of jumpers, lifted leads and jurisdictional or safety tags.

j. Provisions for returning a system to normal configuration upon completion of the test.

Test results shall be documented, evaluated, and their acceptability determined by a responsible individual or group.

The test program shall cover all required tests including:

1. Tests during the preoperational period to demonstrate that plant performance is in accordance with the design intent.

2. Tests during the initial operational phase to demonstrate the performance of systems that could not be tested prior to operation to confirm that plant behavior conforms to design criteria.

3. Tests during the operational phase to provide assurance that failures or substandard performance do not remain undetected and that the required reliability of systems important to safety is maintained.

4. Tests during activities associated with plant maintenance during the operational phase and to demonstrate satisfactory performance following plant maintenance or procedural changes.

Tests performed following plant repairs or replacements shall be conducted in accordance with the original design and testing requirements or engineering approved, documented alternatives. Testing shall be sufficient to confirm that the changes reasonably produce expected results and that the change does not reduce safety of operations.

6.2.1.4 Control of Measuring and Test Equipment

Measures shall be established to assure that tools, gauges, instruments, and other measuring and testing devices used in activities affecting the function or quality of structures, systems, and components covered under the scope of the TMI Quality Assurance Program be properly controlled, calibrated, and adjusted at specified periods to maintain accuracy within specified limits. Additional measures shall be established to ensure the range, type and accuracy of test equipment conforms to the specified testing requirements.

Requirements for each control program shall include inspection and verification of accuracy upon receipt of equipment, identification of all gauges and instruments, calibration and scheduled recall for calibration and traceability to an accepted Standard. Procedures shall be established to implement the following requirements:

a. To establish the calibration technique and frequency maintenance, and control of all measuring and test equipment which are used in the measurement, inspection, and monitoring of components, systems, and structures covered under the scope of the TMI Quality Assurance Program (instruments, tools, gauges, fixtures, reference and transfer standards, and nondestructive examination equipment).

b. The identification of measuring and test equipment traceable to the calibration test data.

c. Installed operations measuring and test equipment requiring calibration shall be labelled, tagged or otherwise controlled in accordance with written, approved procedures to assure that approved calibration intervals are not exceeded. Portable measuring and test equipment may be similarly controlled; but shall, as a minimum, be clearly labelled to indicate the date on which the current calibration expires. Portable measuring and test equipment that has exceeded the approved calibration interval shall not be used for measurements or tests.

d. Establish calibration frequency for measuring and test equipment based on required accuracy, purpose, degree of usage, stability characteristics, and/or any other condition which may affect the measurement. A calibration recall system shall be implemented to assure recalibration within the required period for each piece of measuring and test equipment covered under the scope of this program.

e. Methods for determining the validity of previous inspections performed when the measuring and test equipment is found to be out of calibration. Inspections or tests are repeated on items determined to be suspect. Such determination is to be documented in suitable form. If any calibration, testing or measuring device is consistently found to be out of calibration, it shall be repaired or replaced.

f. Calibration shall be against standards that have an accuracy of at least four times the required accuracy of the equipment being calibrated. When this is not possible, standards shall have an accuracy that assures the equipment being calibrated will be within required tolerance and that the basis of acceptance is documented and authorized by the supervisor of the calibrating organization.

g. A status of all measuring and test equipment under the calibration program is to be maintained.

h. Utilization of reference and transfer standards traceable to nationally recognized standards. Where national standards do not exist, provisions shall be established to document the basis for the calibration.

i. NDE equipment, such as ultrasonic equipment, shall be controlled and calibrated in accordance with the ASME code governing its use.

6.2.1.5 Handling, Storage and Shipping

Measures shall be established and documented to control handling, storage, and shipping, including cleaning, packaging, and preservation of items important to safety in accordance with established instructions, procedures, and drawings to prevent damage, deterioration or loss.

Organizations performing special handling, preservation, storage, cleaning, packaging, and shipping activities shall do so in accordance with predetermined work and inspection procedures or instructions utilizing suitably trained individuals.

Procedures shall be established to control the cleaning, handling, storage, packaging, and shipping of materials, components, systems in accordance with design and procurement requirements to preclude damage loss or deterioration by environmental conditions such as temperature or humidity. These procedures shall include an assessment of, but not limited to, the following:

a. Packaging and preservation procedures to provide assurance of adequate protection against corrosion, contamination, physical damage or any effect which would lower the quality of the items or cause them to deteriorate during shipping, handling or storage. Special protective environments, special coverings, inert gas atmosphere, allowable moisture content, and temperature level shall be specified as required and their existence verified and documented.

b. Cleaning procedures to provide assurance that necessary cleaning operations are carried out prior to packaging, storage or installation. The level of cleanliness required, and verification and documentation requirements shall be specified in the procedures.

c. Detailed handling procedures to be provided for all items that require special handling. Special handling tools and equipment shall be provided and controlled to ensure safe and adequate handling. These tools and equipment shall be maintained, inspected and tested in accordance with written procedures at established intervals to ensure their reliability and availability for use.

d. Storage procedures to provide for methods of storage and the control of items in storage which will minimize the possibility of damage or deterioration during storage. Periodic inspections of storage areas shall be performed and documented to verify compliance with storage procedures. Release of items for installation shall also be procedurally controlled.

e. Procedures to be provided to assure that proper marking and labeling of items and containers is accomplished to provide identification and necessary instructions during packaging, shipment and storage.

f. Procedures for documenting and reporting noncompliance and nonconformance to handling, and shipping requirements.

g. Provisions for the storage of chemicals, reagents, lubricants and other consumable materials which will be used in conjunction with systems which are important to safety.

h. Provisions for "Limited Life" requirements (including "Shelf Life" and "Service Life" for applicable materials.

6.2.1.6 Inspection, Test, and Operating Status

Measures shall be established and documented to ensure that the required inspections and tests are performed and that the acceptability of items with regard to inspection and tests performed is known throughout manufacturing, installation, and operation. Status of items covered by the scope of the TMI Quality Assurance Plan shall be controlled in accordance with approved procedures. These procedures shall include the use of appropriate tags, markings, lists, logs, diagrams, or other suitable means, to assure that required inspections and tests are satisfactorily completed to prevent inadvertent bypassing of required inspections and tests and to prevent inadverent operation.

The requirements for an acceptable inspection, test and operating status program for structures, systems, and components throughout fabrication, installation and test include:

a. Design and quality documents which address the requirements for the identification of inspection, test, and operating status of structures, systems and components.

b. Procedures which include controls for the application and removal of inspection and welding stamps, and other status indicators such as tags, markings, labels, and stamps.

c. Bypassing or altering the sequence of required inspections, tests or other critical operations procedurally controlled by Engineering procedures with concurrence by the appropriate quality organization. Where necessary to preclude inadvertent bypassing of required inspections and tests, the procedures shall provide for the identification of items which have passed such inspections and test.

d. In cases where documentary evidence is not available to confirm that an item has passed required inspections and tests, that item shall be considered nonconforming until such evidence becomes available. Affected systems shall also be considered to be inoperable and reliance shall not be placed on such systems to fulfill their intended safety functions.

e. Procedures to be provided to require identification of the operating status of systems, components, controls, or support equipment in order to prevent inadvertent or unauthorized operation. These procedures shall require control measures such as locking or tagging to secure and identify equipment in a

controlled status. Independent verification shall be required, where appropriate, to ensure that necessary measures, such as tagging equipment, have been implemented correctly.

f. Temporary modifications shall be controlled by approved procedures which include a requirement for independent verification. A log shall be maintained of the current status of such temporary modification.

g. Nonconforming services and inoperative or malfunctioning structures, system, components or materials shall be identified, documented and controlled in accordance with the requirements of this Plan.

6.2.1.7 Fire Protection

The primary objective of a Fire Protection Program is to minimize both the probability and consequences of postulated fires. Fire Protection starts with design and must be carried through all phases of construction and operation. Therefore, Quality Assurance Program requirements in accordance with Regulatory Guide 1.120 and this Plan shall be established to assure the reliability of the TMI fire protection systems. Quality measures shall be established to ensure that the guidelines for design, measurement, installation, testing and administrative controls for the fire protection systems are satisfied.

6.2.1.8 Plant Security

Procedures shall be developed utilizing the guidelines of ANSI N18.17-1973 to supplement features and physical barriers designed to control access to the plant and, as appropriate, to vital areas within the plant. Information concerning specific design features and administrative provisions of the plant security programs shall be confidential and thus accorded limited distribution. Quality measures shall be established to ensure that the guidelines for design, measurement, installation, testing and administrative

controls for the plant security systems are satisfied.

6.2.1.9 Housekeeping and Cleanliness

Housekeeping practices on a regularly scheduled basis shall be utilized recognizing the requirements for the control of radiation zones and the control of work activities, conditions and environments that can affect the quality of important parts of the nuclear plant. Housekeeping encompasses all activities related to the control of cleanliness of facilities, materials, equipment fire prevention and protection including disposal of combustible material and debris and control of accesses to areas, protection of equipment, radioactive contamination control and storage of solid radioactive waste. Housekeeping practices shall assure that only proper materials, equipment processes, and procedures are utilized and that the quality of the item is not degraded as a result of housekeeping practices or techniques. During maintenance activities, certain portions of safety-related systems may be subject to potential contamination with foreign materials. To prevent such contamination, control measures, including measures for access control, shall be established. Additionally, immediately prior to closure, an inspection shall be conducted and documented to ensure cleanliness. Special housekeeping considerations shall be made for maintenance of radioactively contaminated systems for components.

6.2.1.10 Equipment Control

Permission to release equipment or systems for maintenance shall be granted by designated NRC SRO licensed operations personnel. Procedures shall be provided for control of equipment, as necessary, to maintain personnel and reactor safety, to avoid unauthorized operation of equipment, and to assure that operational equipment is a ready status. These procedures shall require:

a. Control measures such as locking or tagging or secure and identify equipment in a controlled status.

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b. Independent verifications, where appropriate, to ensure that necessary measures, such as tagging equipment, has been implemented correctly.

c. Control measures for temporary modifications, such as temporary by-pass lines, electrical jumpers, lifted electrical leads, and temporary trip point settings. Included shall be a requirement for independent verification. (A log shall be maintained of the current status of temporary modifications.)

d. Control of inspection and test status on individual items by the use of markings such as stamps, tags, labels, routing cards or other suitable means.

e. When equipment is ready to be returned to service, operating personnel shall place the equipment in operation and verify and document its functional acceptability.

6.2.1.11 Control of Construction, Maintenance (Preventive/Corrective) and Modifications

Construction, maintenance or modifications which has the potential to affect the functioning of structures, systems or components important to safety shall be performed in a manner to ensure quality at least equivalent to that specified in original design bases and requirements, materials specifications and inspection requirements. A suitable level of confidence in structures, systems or components on which maintenance or modifications have been performed shall be attained by appropriate inspection and performance testing. Construction, maintenance or modification of equipment shall be preplanned and performed in accordance with written procedures, documented instructions or drawings appropriate to the circumstances which conform to applicable codes, standards, specifications, and criteria. Skills normally possessed by qualified maintenance personnel may not require detailed step-by-step delineations in a written procedure but are subject to general administrative procedural controls that govern or define the following areas:

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a. Methods for obtaining permission and clearance for operation personnel to work and for logging such work.

b. Factors to be taken into account, including the necessity of maintaining occupational radiation exposure as low as is reasonably achievable (ALARA).

c. Method for identification of what procedural coverage is necessary for the maintenance, construction and modification activity.

d. Considerations for system/equipment cleaniness control.

e. Method for identification of post maintenance, construction or modification, testing, including system/equipment functional capability to meet operational requirements in all respects.

f. Method for ensuring that maintenance, contruction or modification activities, performed either on-site or off-site, are properly reviewed.

The following type of activities are among those that may not require detailed step-by-step written procedures:

a. Gasket replacement

b. Trouble shooting electrical circuits

c. Changing chart or drive speed gears or slide wires on recorder.

Means for assuring quality of maintenance, modifications or construction activities (for example, inspections, measurements, tests, welding, heat treatment, cleaning, nondestructive examination and worker qualifications in accordance with applicable codes and standards) and measures to document the performance thereof shall be established. Measures shall be established and documented to identify the inspection and test status of items to be used in maintenance modification, and construction activities. Normally, the

point of control for such items should be the plant storage area.

A corrective maintenance program shall be developed to maintain structures, systems and components important to safety at the quality required for them to perform their intended functions. Corrective maintenance shall be performed in a timely manner to insure that important to safety items are adequately maintained in the original, design, functional status.

A preventative maintenance program including procedures as appropriate for structures, systems, and components important to safety shall be established which prescribes the frequency and type of maintenance to be performed. In all cases, maintenance shall be scheduled and planned so as not to compromise the safety of the plant. Planning shall consider the possible safety consequences of concurrent or sequential maintenance, testing or operating activities. Preventive maintenance shall be performed in a timely manner to insure that important to safety items are adequately maintained in the original, design, functional status.

6.2.1.12 Procedural Requirements

Measures shall be established to control and coordinate the approval and issuance of documents, including changes, which prescribe all activities affecting quality. Those documents which are considered important to safety require a documented Quality Assurance Department review. This review is to provide an independent verification that the procedures have been prepared, reviewed and approved in accordance with established policy and program controls; they contain the necessary policy and program requirements including the inspection and verification requirements where applicable; and they contain clear descriptions related to the extent of documenting results of completed actions when required. These documents include operating and special orders, operating procedures, test procedures, equipment and material control procedures, maintenance or modification procedures, and refueling procedures. Each procedure shall be

reviewed and approved prior to initial use. Plant procedures shall be reviewed by an individual knowledgeable in the area affected by the procedure no less frequently than every two (2) years to determine if changes are necessary or desirable.

6.2.1.13 Control of Surveillance Testing and Inspection

A surveillance testing and inspection program shall be established to insure that important to safety structures, systems, and components will continue to operate, keeping parameters within normal bounds, or will act to put the plant in a safe condition if they exceed normal bounds.

Provisions shall be made for performing required surveillance testing and inspections, including inservice inspections. Such provisions shall include the establishment of a master surveillance schedule reflecting the status of all planned inplant surveillance tests and inspections. Frequently of surveillance tests and inspections may be related to the results of reliability analyses, the frequency and type of service, or age of the item or system, as appropriate.

Additional control procedures shall be instituted, as necessary, to assure timely conduct of surveillance tests and inspections and appropriate documentation, reporting, and evaluation of the results. Following the completion of testing, procedures shall be established to assure the return of systems to an operable status. These procedures shall include provisions for the documentation of authority, conduct, responsibility, and verification involved in returning the system to an operable status. Such provisions shall include the use of procedures, checklists, and independent verification as appropriate, considering the degree that system status was altered during the performance of the test.

6.2.1.14 Radiation Control

Procedures shall be provided for the implementation of a radiation control program. The radiation control program involves the acquisition of data and provision of equipment to

perform necessary radiation surveys, measurements and evaluations for the assessment and control of radiation hazards associated with TMI. Procedures shall be developed and implemented for quality assurance review of records and programs to insure the adequacy of measures taken to control radiation exposure of employees and others. Additionally, quality measures for radwaste management shall be implemented in accordance with 10 CFR 71, Appendix E.

6.3 Responsibilities

6.3.1 Senior Vice President - Met-Ed

The Senior Vice President - Met-Ed has overall responsibility for the management, supervision, and control of all station activities.

6.3.2 TMI Generation Group Station Management

The TMI Generation Group Station management is responsible for the implementation and compliance of the Quality Assurance Plan and directly responsible to insure their respective activities and responsibilities are conducted in accordance with applicable adminsitrative controls, requlatory and licensing requirements.

6.3.3 Delegated Authorities

Contractors or other agents outside the TMI Generation Group who are assigned or delegated responsibilities and/or activities governed by this Plan shall comply with the applicable requirements of the Plan.

7.0 Control of Radioactive Waste

7.1 Policy

Measures shall be established and documented to assure that the applicable requirements of the Code of the Federal Regulations, Title 10, Part 71 and Title 49, Parts 100 through 199 applicable to the packaging and transporting of radioactive wastes are satisfied. Appendix E to 10 CFR 71 identifies the quality assurance criteria applicable to the control of radioactive waste.

The applicable portions of this Plan that relate to the criteria in Appendix E to 10 CFR 71 describe to a large extent the administrative controls and quality requirements to be applied in the control of Radioactive Waste. Typically, Sections 6.2.1.1 thru 6.2.1.6 and 6.2.1.9 apply to Control of Radioactive Waste.

7.2 Requirements

7.2.1 Procedures shall be developed and implemented to cover the following:

a. Processing of radioactive wastes including the collection, handling and preparation for shipment of radioactive liquids and solids. These procedures shall be consistent with the ALARA program and shall clearly identify the administrative controls and organizational responsibilities.

b. Training and qualification of personnel operating radioactive waste processing equipment, health physics monitoring, packaging and shipping and other operations deemed appropriate by management.

c. The activities associated with the packaging of radioactive wastes to include the proper selection of the receptacles to be used for containing the waste materials, the selection of the shipping containers (structures used to contain and support the receptacle and its contents) Health Physics inspections of the packaging prior to release, proper markings on

the outside of the package and the preparation of shipping papers and certificates.

d. Movement of radioactive materials within and outside the protected area to assure personnel protection at all times.

e. The shipment of radioactive material from the Station to be in accordance with the regulations of the U.S. Department of Transportation for the transportation of hazardous materials (49 CFR) and of the NRC (10 CFR 71).

f. The packaging used for transporting of radioactive wastes, whether purchased from an outside supplier or designed by GPUSC, shall meet the applicable requirements of 10 CFR 71 and 49 CFR.

7.2.2 The carriers to be used for transporting of radioactive wastes shall be selected on the basis of their experience, knowledge of DOT regulations, control and maintenance of their equipment and the selection and control of their drivers. The carrier is required to have or shall be supplied documented procedures covering acceptance of materials from a shipper, certification requirements, placarding, stowage control, reporting of incidents and security.

7.2.3 Radwaste operations shall be controlled to minimize personnal exposures or environmental contamination consistent with ALARA.

7.2.4 Operations procedures shall be reviewed by QAD to establish any necessary witness or hold points or activities to be monitored.

7.3 Responsibilities

7.3.1 The Manager - TMI Unit 1 and Manager - Radiological Controls shall develop and implement procedures for processing activities and movement of radioactive materials.

7.3.2 The Operations Department shall be responsible for the processing and packaging of liquid wastes and for the packaging of solid wastes

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in preparation for shipment. Additionally, the Operations Department is responsible for the collection and identification of radioactive solids, such as rags, papers, boots, gloves, etc., and having them moved to the Radwaste facility for packaging.

7.3.3 The Radiological Controls Department is responsible for monitoring all activities associated with the processing and handling of radioactive wastes and for providing advice on radiological matters relating to processing, packaging and shipping.

7.3.4 The Operations Department is responsible for the selection of the proper packaging for the specific contents to be shipped, taking into consideration the radiation levels, contamination limits and shipping requirements. Health Physics inspects the packaging for radiation level and, if acceptable, the Operations Department marks the outside of the package with the appropriate markings, completes the shipping papers and certificates, attaches the security seal and advises the carrier that the shipment is ready.

7.3.5 Plant Engineering is responsible for reviewing and accepting the designs of packaging purchased from an outside supplier. If packaging is to be designed by GPUSC, the design, fabrication and licensing of the packaging shall be the responsibility of the Director - Technical Functions.

7.3.6 Each manager for this functional area related to the control of radioactive wastes, shall establish the requirements for personnel qualification and institute training and indoctrination to satisfy these requirements. Training requirements shall be consistent with the importance and complexity of the activity performed.

7.3.7 Quality Assurance Modifications/Operations Manager is responsible for review and concurrence with procedures describing control of radioactive waste. He is also responsible to monitor and/or inspect radioactive waste processing operations to the extent to verify

they are preformed in accordance with established procedures, applicable administrative controls and regulatory requirements.

The Operations Department shall review and accept carriers' documented procedures as specified by procurement documents covering acceptance of radioactive waste materials for shipment.

8.0 Control of Corrective Actions and Nonconformances

8.1 Policy

Nonconforming materials, parts, components, services or activities within the scope of the QA Plan shall be controlled to prevent their inadvertent utilization. As a result, measures shall be established which ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances be promptly identified and corrected. The cause of the condition adverse to quality shall be determined and appropriate action taken to preclude repetition. The identification, cause, and actions taken to correct conditions adverse to quality shall be documented and reported to the appropriate levels of management.

Significant conditions within the intent of 10CFR 50.55(e) or 10 CFR 21 shall be reported to appropriate management levels within the affected organization for review and evaluation.

8.2 Requirements

Procedures shall be established which detail and implement the following corrective action system measures:

a. Conditions adverse to quality shall be evaluated to determine the need for corrective action.

b. Corrective action documentation shall include identification, cause, and actions taken to correct and to preclude the similar recurrence for conditions adverse to quality.

c. Follow-up activities shall be conducted to verify implementation of corrective actions and to close out corrective action in a timely manner.

d. Significant deficiencies, nonconformances and defects, reportable under 10 CFR 50.55(e) or 10 CFR 21 shall be

reported to appropriate management levels for evaluation and possible reporting to the Nuclear Regulatory Commission.

e. Control of nonconforming materials, parts, components, services, or activities. These procedures shall address and detail measures to implement the following requirements:

1. Measures for the identification, documentation, segregation; and dispositions of nonconforming materials, parts or components.

2. Disposition of nonconformances shall be made by the organization that established the governing requirements or by other qualified individuals or committees authorized by the TMI Generation Group.

3. Nonconformance reports shall be used to identify materials, parts, components, and activities which are not in compliance with the requirements of specifications, codes, drawings, and detailed installation or manufacturing program requirements. This shall include use of nonconformance reports on items whose status is indeterminate due to the lack of documentation. Nonconformance reports on items shall contains the following minimum information:

(a) Identification of the nonconforming item and date of inspection.

(b) Identification of the initiator of the nonconformance report.

(c) Description of the nonconformance.

(d) Disposition of the nonconformance (repair, rework, use as is, or scrap).

(e) Inspection requirements.

(f) Required approval signatures of the disposition and the verification.

(g) Evidence of review for reporting per 10 CFR 50.55(e) or 10 CFR 21.

4. Reworked, repaired, and replacement items shall be reinspected and tested in accordance with the original inspection and test requirements or acceptable alternatives as determined by Engineering and Quality Assurance. All inspection, testing, rework, and repairs shall be by approved procedures and the results documented.

5. Identification of nonconforming items by appropriate means (tags, labels, etc.) and segregation, if practical, until disposition of the nonconforming item has been determined.

6. Prior to the initiation of a preoperational test on a safety-related item all nonconformances shall be evaluated for significance or impact on further testing or operation.

7. Nonconformance reports shall be periodically analyzed to show quality trends. Such analysis will be based upon severity, number, frequency of nonconformances, the causes of the nonconformances, and the timeliness and adequacy of the reporting and resolution of nonconformances. Significant results shall be reported to management for review and assessment.

8.3 Responsibilities

8.3.1 The Manager - Quality Assurance is responsible for the review and concurrence of all procedures for reporting and controlling of nonconformances for compliance with the requirements of the Operational Quality Assurance Plan.

8.3.2 Operations Management is responsible for ensuring that nonconformances are reported and corrected for plant personnel activities involving operation, maintenance, repair replacement, addition, modification, health physics, environmental monitoring, fuel handling, and inservice inspection. Plant items such as failures, malfunctions, deficiencies, deviations and defective materials, parts or components are handled in a manner consistent with their importance to safety and reviewed in accordance with appropriate procedures and the applicable Technical Specifications.

8.3.3 Each Manager is responsible for the disposition and corrective action of nonconformances identified as within the scope of his responsibilities. In the specific case of materials, parts, components, or systems which have not been installed or accepted as operational at the Station, the responsible Manager and the Manager - Quality Assurance approves the resolution of nonconformances.

9.0 Audits

9.1 Policy

A comprehensive system of planned and documented audits shall be established and executed:

a. To ensure that Quality Assurance requirements are adequate, effective and implemented.

b. To ensure than nonconformance and Quality Assurance deficiencies are identified and corrected.

c. To verify compliance with the TMI Quality Assurance Program.

In addition, this audit program shall provide data for a continuing evaluation of the effectiveness of the TMI Quality Assurance Program.

9.2 Requirements

A comprehensive system of audits shall be established for both internal and external functions which affect structures, systems, components, operations and activities covered by the scope of the TMI Quality Assurance Program.

Planned and scheduled audits shall verify compliance with the following:

a. TMI Quality Assurance Program

b. 10 CFR 50, Appendix B

c. Regulatory Guides, ANSI, and other codes and standards as endorsed in the TMI Quality Assurance Program.

d. Operating procedures

e. Plant technical specifications

f. Administrative procedures

g. Other procedures and instructions affecting quality

h. Procurement documents

9.2.1 Audit Program

Audits shall be performed in accordance with pre-established written procedures and checklists, and shall be conducted by trained and qualified personnel having no direct responsibilities in the areas being audited. The audit program shall include

a. Audit schedules

b. Procedures for preparation, performance and reporting of audits

c. Analysis of audit data and reporting results to appropriate levels of management

d. Follow-up action to be taken based upon individual and collective audit reports

e. Qualification of auditors

f. Delineation of the authority, responsibility, and organizational independence of those responsible for the audit program.

Audits shall be regularly scheduled based upon the status and safety importance of activities being performed and shall be initiated in a timely manner to assure the effectiveness during design, procurement, manufacturing, construction, installation, inspection, testing and as required by the technical specifications for TMI. In addition, audits may be scheduled and performed as required by management or the safety review groups for special evaluations. Implementation of corrective action shall be verified in a timely manner. Unscheduled audits may be conducted at any time on any aspect of this Quality Assurance Plan.

Both TMI Generation Group and organizations providing goods and/or services are subject to the audit requirements of this Program.

Audits will be performed by the Quality Assurance Methods/Operations/Audit group.

Each audit team shall be led by a qualified Audit Team Leader. Audit team members shall be utilized as required and will be classified as either auditors or technical specialists, depending on their function on the audit team.

Procurement documents shall include audit access requirements in insure vendor compliance to the audit program. Audited organizations shall cooperate with the auditing organization, providing whatever assistance is necessary in the performance of the audit. The audited organization shall take corrective action for findings and resolve observations in a timely manner.

9.2.2 Audit Frequency

Audit frequencies shall be based upon the status and safety importance of activities, degree of previous experience, consistency of overall coverage, unique testing/operating activities, and follow-up on previous audit findings.

9.2.3 Documentation

Audit results shall be documented in a written report to the audited organization. The Quality Assurance organization conducting the audit is responsible for conducting follow-up actions including re-audit of deficient areas, as required, to assure correction of the deficiencies.

9.2.4 Training

Audits shall be performed by personnel who are trained and qualified to the requirements defined in ANSI N45.2.23. These requirements provide the means to assure that audits are performed in a thorough and professional manner. Documented training programs shall be organized to provide auditors with the necessary training and knowledge of regulatory requirements, codes, standards, procedures, etc. applicable to the activities being audited.

9.3 Responsibilities

9.3.1 Senior Vice President - Met-Ed

Responsible for the performance of an independent review of the TMI Quality Assurance Program and related activities.

9.3.2 Manager - Quality Assurance

a. The Manager-Quality Assurance is responsible for establishing and implementing the overall Quality Assurance audit program. He assures that all applicable areas are audited and that the auditing organization meets the requirements of this Plan. He evaluated the effectiveness of the overall audit program, analyzes the reports and related information for quality trends and appraises the TMI Generation Group Management and the Director-Reliability Engineering of significant findings of the program. The Manager-Quality Assurance further ensures that an overall Quality Assurance Audit Program Schedule is established and implemented.

b. The Manager-Quality Assurance has the authority and organizational freedom to schedule and perform audits and to identify quality or management control problems and provide recommended solutions.

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APPENDICES

APPENDIX A — Comparison Chart of Quality Assurance Plan Requirements with those of various parts of the Code of Federal Regulations and Nuclear Industry Standards

APPENDIX B — Minimum Document Control Responsibility for "Important to Safety" Documents Quality Assurance Program

APPENDIX C — NRC Regulatory Guide Commitments and Exceptions

COMPARISON CHART OF QUALITY ASSURANCE PLAN REQUIREMENTS WITH THOSE OF VARIOUS PARTS OF THE CODE OF FEDERAL REGULATIONS AND NUCLEAR INDUSTRY STANDARDS

10 CFR 50, App. B	
Criterion	QA Plan
I	1.0
II	2.0
III	4.0
IV	5.1
V	3.1
VI	3.2
VII	5.1
VIII	5.2
IX	6.2.1.2
X	6.2.1.1
XI	6.2.1.3
XII	6.2.1.4
XIII	6.2.1.5
XIV	6.2.1.6
XV	8.0
XVI	8.0
XVII	3.3
XVIII	9.0

ANSI N45.2	
Paragraph	QA Plan
2.0	2.0
3.0	1.0
4.0	4.0
5.0	5.1
6.0	3.1
7.0	3.2
8.0	5.1
9.0	5.2
10.0	6.2.1.2
11.0	6.2.1.1
12.0	6.2.1.3
13.0	6.2.1.4
14.0	6.2.1.5
15.0	6.2.1.6
16.0	8.0
17.0	8.0
18.0	3.3
19.0	9.0

10 CFR 71, App. E	
Criterion	QA Plan
1	1.0
2	2.0
3	4.0
4	5.1
5	3.1
6	3.2
7	5.1
8	5.2
9	6.2.1.2
	7.0
10	6.2.1.3
	7.0
11	6.2.1.3
	7.0
12	6.2.1.4
	7.0
13	6.2.1.5
	7.0
14	6.2.1.6
	7.0
15	8.0
16	8.0
17	3.3
18	9.0

ANSI N18.7 - 1976			
Paragraph	QA Plan	Paragraph	QA Pla
3.1	1.0	5.2.11	8.0
3.2	1.0	5.2.12	3.3
3.3	1.0	5.2.13	5.0
3.4	1.0	5.2.14	8.0
3.5	2.0	5.2.15	3.0
4.1	2.0/9.0	5.2.16	6.2.1.4
4.2	2.0/9.0	5.2.17	6.2.1.1
4.3	2.0	5.2.18	6.2.1.2
4.4	2.0	5.2.19	6.2.1.3
4.5	9.0	5.3	6.2.1.1
5.1	2.0		
5.2.1	2.0		
5.2.2	3.1		
5.2.3	3.1		
5.2.4	3.1		
5.2.5	3.1		
5.2.6	6.2.1.10		
5.2.7	6.2.1.11		
5.2.8	6.2.1.13		
5.2.9	6.2.1.8		
5.2.10	6 2.1.10		

APPENDIX B

MINIMUM DOCUMENT CONTROL RESPONSIBILITY FOR "IMPORTANT TO SAFETY" DOCUMENTS

Document	Prepared By	Reviewed By	*Approved By/Concurrence	Issued By	(Notes) (1,2,3)
A.1 QA Plan (Significant Changes)	QA Department	Manager - QA	*President - Met Ed *President - GPUSC Sr. V.P. - Met Ed V.P. - Materials Management Director - Reliability Eng. Applicable Unit Directors Manager - QA	QA Department	
A.2 QA Plan (Insignificant Changes)	QA Department	Manager - QA	*Manager - QA Applicable Unit Director	QA Department	
B.1 QA Department Procedures	QA Department	QA Department	*Manager - QA	QA Department	
B.2 QA Department Section Procedures	QA Section	QA Section	*QA Section Manager Manager - QA	QA Department	
B.3 QA Department Section Instructions	QA Section	QA Section	QA Section Manager *QA Subsection Manager/Supervisor	QA Department	
C.1 TMI Generation Group Administrative Procedures	TMI Generation Group Administration	TMI Generation Group Department Heads Affected Director - Reliability Eng.	*Sr. V.P. - Met Ed Director - Reliability Eng.	TMI Generation Group Administration	
C.2 TMI Generation Group Function/Dept. Procedures	TMI Generation Group Function/Dept.	TMI Generation Group Function/Dept. Manager - QA	*TMI Generation Group Function/Dept. Manager Manager - QA	TMI Generation Group Administration	
C.3 TMI Station Administrative Procedures	TMI Station Organizations	Affected TMI Station Management QA Mod/Ops Manager	*Unit Manager	TMI Station Administration	
C.4 TMI Station Section Procedures	TMI Station Sections	TMI Station Section Manager/Supervisor QA Mod/Ops Manager	*TMI Station Section Manager/Supvr. Unit Manager	TMI Station Administration	

APPENDIX B

Document	Prepared By	Reviewed by	*Approved By/Concurrence	Issued By	(Notes) (1,2,3)
C.5 TMI Station Section Instructions	TMI Station Sections	TMI Station Sections QA Mod/Ops Manager	*TMI Station Section Manager/Supervisor QA Mod/Ops Manager	TMI Station Administration	
C.6 TMI Special Test Procedures (Per 10 CFR 50.59)	TMI Station Organizations	PORC GRC Unit Manager QA Mod/Ops Manager	*Unit Manager NRC	TMI Station Administration	
C.7 TMI Radiation Protection Procedures	Radiation Protection	Radiation Protection Engineering	*Radiation Protection Manager/ Supervisor	TMI Station Administration	
D.1 Procurement Requisition	Off Site Organizations	Applicable Section Managers QA Design/Procurement Manager	*Project Engineering Manager QA Design/Procurement Manager	Project Engineering Manager	
D.2 Procurement Requisition	Site Organizations	Applicable Section Managers QA Design/Procurement Supvr.	*Manager - Plant Engineering QA Design/Procurement Supvr.	Originating Organizations	
E.1 Engineering Change Memorandums	TMI Generation Group Engineering	Applicable Section Managers QA Design/Procurement Manager	*Project Engineering Manager QA Design/Procurement Manager Applicable Section Manager	TMI Generation Group Engineering	(Notes) (4,5,6)
E.2 Engineering Change Memorandums	TMI Plant Eng.	Applicable Department Managers QA Mod/Ops Manager	Applicable Department Managers *TMI Plant Engineering QA Design/Procurement Supvr.	TMI Plant Eng.	(Notes) (4,5,6)

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NOTE: 1) Responsible individuals or organizations may have documented, designated alternates who are authorized to perform the function.

2) Designated support organizations (within the TMI Generation Group or outside contractors) may be authorized and designated to perform certain of the functions.

3) This Appendix is a supplement of Section 3.0 of the QA Plan

4) Drawings will not be reviewed unless used in lieu of specifications.

5) TMI Generation Engineering is defined as those sections within the Technical Function Department.

6) Engineering Change Memoramdums are defined as formalized documents for the description and approval of changes prepared for incoproration at TMI. Engineering Change Memorandums will include:

a) Cover Sheet
b) Nuclear Safety Environmental Impact Evaluation
c) Safety Evaluation
d) Fire Hazards Analysis
e) Nuclear Safety Related Design Verifications
f) Index of Interim Drawings and Related Purchase Orders

APPENDIX C

QUALITY ASSURANCE PROGRAM NRC REGULATORY GUIDE COMMITMENTS AND EXCEPTIONS

Engineering, in establishing specific requirements for design will use regulatory guide positions controlled by Engineering in a project criteria document. Examples of positions taken relative to regulatory guides are listed. Those identified by an asterisk cover regulatory guides which are specifically quality related or impacted and are therefore controlled by this manual.

The TMI Quality Assurance Program complies with Section C of the NRC Regulatory Guides indicated below. Exceptions to NRC Regulatory Guide position are detailed in Part 2 of this Appendix.

This Appendix addresses additional Reg. Guides not listed in Rev. 7 of the Operational Quality Assurance Plan. Compliance with these added Reg. Guides will apply to modifications, additions and activities performed after issue of Rev. 8 and does not imply backfitting and/or retroactive compliance. It is also to be recognized that existing plant conditions, may prevent or preclude the satisfaction of all requirements of a specific design related regulatory guide. The deviation will be documented and, along with the justification, will be approved by the Manager of Design and Engineering.

APPENDIX C, PART I

JANUARY, 1980

COMMITMENT TO QUALITY ASSURANCE REGUALTORY GUIDES FOR THREE MILE ISLAND

REG. GUIDE		ANSI STD.		DEGREE OF COMPLIANCE	REMARKS
*1.8 5/77, Rev. 1-R	Personnel Selection and Training	N18.1	1971	Modified	Comply with "Regulatory Position".
*1.28 2/79, Rev. 2	Quality Assurance Program Requirements (Design and Construction)	N45.2	1977	Full	Comply with "Regulatory Position".
1.30 8/11/72	QA Requirements for the Installation, Inspection and Testing of Instrumentation and Electrical Equipment	N45.2.4	1972	Full	Comply with "Regulatory Position".
*1.33 2/78, Rev. 2	Quality Assurance Program Requirements (Operation)	N18.7	1976	Modified	See alternate method attached.
1.37 3/16/73	QA Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants	N45.2.1	1973	Modified	See alternate method attached.
1.38 5/77, Rev. 2	QA Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water Cooled Nuclear Power Plants	N45.2.2	1972	Modified	See alternate method attached.
1.39 9/77, Rev. 2	Housekeeping Requirements for Water Cooled Nuclear Power Plants	N45.2.3	1973	Full	Comply with "Regulatory Position".
1.54 6/73	QA Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants	101.4	1972	Modified	See alternate method attached.
*1.58 7/79 Proposed Rev. 1	Qualifications of Nuclear Power Plant Inspection, Examination and Testing Personnel	N45.2.6	1978	Modified	See alternate method attached.
*1.64 6/76 Rev. 2	Quality Assurance Requirements for the Design of Nuclear Power Plants	N45.2.11	1974	Modified	See alternate method attached.
*1.74 2/74	Quality Assurance Terms and Definitions	N45.2.10	1973	Full	Comply with "Regulatory Position".

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REG. GUIDE		ANSI STD.	DEGREE COMPLIANCE	REMARKS
*1.88 10/76, Rev. 2	Collection Storage and Maintenance of Nuclear Power Plant Quality Assurance Records	N45.2.9 1974	Modified	See alternate method attached.
1.94 4/76, Rev. 1	QA Requirements for Installation, Inspection and Testing of Structural Concrete & Steel during Nuclear Power Plant Construction	N45.2.5 1974	Modified	See alternate method attached.
1.116 5/77, Rev. 0-R	QA Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems	N45.2.8 1975	Modified	See alternate method attached.
*1.123 7/77, Rev. 1	QA Requirements for Control of Procurement of Items and Services for Nuclear Power Plants	N45.2.13 1976	Full	Comply with "Regulatory Position".
*1.144 1/79	Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants	N45.2.12 1977	Modified	See comments attached.
1.26 2/76, Rev. 3	QA Classifications and Standards for Water Stream and Radioactive Waste Containing Components of Nuclear Power Plants		Modified	See alternate method attached.
1.31 4/78, Rev. 3	Control of Ferrite Content in Stainless Steel Weld Metal		Modified	
1.63 8/78, Rev. 2	Electric Penetration Assemblies in Containment Structure for Light Water Cooled Nuclear Power Plants	I.EEE-317 1976	Modified	See clarification attached.

NRC Regulatory Guide 1.30, August 1972

Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electric Equipment

Met-Ed shall comply with the Regulatory Position established in this Regulatory Guide in that QA programmatic/administrative requirements included therein shall apply to maintenance and modification activities even though such requirements were not in effect originally. Technical requirements associated with maintenance and modifications shall be the original requirements or better (e.g., code requirements, material properties, design margins, manufacturing processes, and inspection requirements).

NCR Regulatory Guide 1.33, Rev. 2, February 1978

Quality Assurance Program Requirements (Operation)

The TMI QA Program complies with the regulatory position of this guide with the following clarifications:

1. Paragraph C.4.a is interpreted to mean audits will be made once each 6 months to verify the nonconformances and corrective action program is properly implemented and documented, particularly as related to actions taken to correct deficiencies that affect items important to safety.

2. Paragraph 5.2.8 of ANSI N18.7 - 1976 titled "Surveillance Testing and Inspection"

   In lieu of a "master surveillance" schedule, a surveillance testing schedule shall be established reflecting the status of all inplant surveillance tests and inspections.

3. Paragraph 5.2.15 of ANSI N18.7 - 1976 titled "Review, Approval and Control of Procedures"

   The third sentence of the third paragraph is interpreted to mean applicable procedures shall be reviewed following a reportable incident such as an accident, an unexpected transient, significant operator error, or equipment malfunction.

4. Paragraph 5.2.17 of ANSI N18.7 - 1976 titled "Inspections"

Not all inspections will require a separate inspection report. Inspection requirements may be integrated into appropriate procedures or other documents with the procedures or documents with the procedure or document serving as the record; however, records of inspections will be identified and retrievable.

NRC Regulatory Guide 1.3.7, March 16, 1973

Quality Assurance Requirements for Cleaning Fluids Systems and Associated Components of Water Cooled Nuclear Power Plants

The TMI Quality Assurance Program complies with the regulatory position of this guide with the following classifications:

1. The second sentence of paragraph C.3 should be amended to read:

   "The water quality for final flushes of fluid systems and associated components shall be at least equivalent to the operating systems water, except for the oxygen nitrogen content and this does not infer that chromates or other additives normally in the system water will be added to the flush water."

2. Paragraph C.4 should be amended to add:

   Expendable material such as inks, temperature indicating crayons, labels, wrapping materials (other than polyethylene), water soluble dam materials, lubricants, NDT penetrant materials and couplants, which contact stainless steel or nickle alloy material surfaces shall not contain lead, zinc, copper, mercury or other low melting alloys or compounds as basic essential chemical constituents. Prescribed maximum levels of water leachable chloride ions, total halogens and sulfur compounds shall be imposed on expendable products.

NRC Regulatory Guide 1.38, Rev. 2, May 1977

Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water Cooled Nuclear Power Plants

The TMI Quality Assurance Program complies with the regulatory position of this guide with the following modifications:

1. Section 3.6 of ANSI N45.2.2 - 1972 concerns prevention of halogenated materials from contacting stainless steel or nickle alloy materials. The position stated in Reg. Guide 1.37 also applies to this guide.

2. Section 3.7.1 of ANSI N45.2.2 - 1972

   Cleated, sheathed boxes will be used up to 1000 lbs. rather than 500 lbs. as specified. This type of box is safe for, and has been tested for, loads up to 1000 lbs. Other material standards (i.e., FED Spec. PPP-B-601) allow this. Special qualification testing shall be required for loads in excess of 1000 lbs.

3. Section 6/2/1 of ANSI N45.2.2 - 1972

   For storage of level D items access will be controlled and limited by posting. Other positive controls such as fencing or posting of guards will be provided for higher storage levels.

4. Section 7.3 of ANSI N45.2.2 - 1972

   Rerating of hoisting equipment will be considered only when necessary. Prior to performing any lift greater than the load rating, the equipment manufacturer will be contacted for his approval and direction. The manufacturer will be requested to supply a document granting approval for a limited number of lifts at the new rating and any restrictions involved such as modifications to be made to the equipment, number of lifts to be made at the new rating, and the test lift load. At all times the codes governing rerating of hoisting equipment will be observed.

5. Section A.3.4.1 Appendix to ANSI 45.2.2 - 1972

   The last sentence of A.3.4.1(4) and (5) should be corrected as follows:

   (4) "However, preservatives for inaccessible inside surfaces of pumps, valves and pipe systems containing, reactor coolant water shall be the water flushable type."

   (5) "The name of the preservative used shall be indicated to facilitate touch up."

6. With regard to Section A.3.5.2 of the Appendix to ANSI N45.2.2 - 1972 entitled "Tapes and Adhesives":

   Tapes will meet a sulphur limit of 0.30% by weight instead of 0.10% as specified in A.3.5.2(1)(a). This limit is reasonable based upon the chemical content of commercially available tapes will be of a contrasting color rather than "Brightly Colored" as required by A.3.5.1(3).

7. With regard to Section A.3.7.1 of the Appendix to ANSI N45.2.2 - 1972 entitled "Fiberboard Boxes":

   In lieu of A.3.7.1(3) and (4), the following will be imposed: Fiberboard boxes shall be securely closed either with a water resistant adhesive applied to the entire area of contact between the flaps, or all seams and joints shall be sealed with not less than 2-inch wide, water resistant tape.

NRC Regulatory Guide 1.39, Rev. 2, September 1977

Housekeeping Requirements for Water Cooled Nuclear Power Plants Endorses ANSI N45.2.3 - 1973

The Operational Quality Assurance Program complies with this guide with the following clarification:

1. With regard to SEctions 2.1 and 3.2 of ANSI N45.2.3 - 1973 entitled "Planning and Control of Facilities", respectively:

   The TMI Nuclear Station will not utilize the five level zone designation system, but will utilize standard janitorial and work practices to maintain a level of cleanliness commensurate with company

policy in the areas of housekeeping, plant and personnel safety, and fire protection.

Cleanliness will be maintained, consistent with the work being performed, so as to prevent the entry of foreign material into safety related systems. This will include as a minimum documented cleanliness inspections which will be performed immediately prior to system closure. Control of personnel, tools, equipment, and supplies will be established when major portions of the reactor system are opened for inspection, maintenance of repair.

Additional housekeeping requirements will be implemented as required for control or radioactive contamination.

NRC Regulatory Guide 1.54, June 1973

Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants

Endorses ANSI N101.4 - 1972

The Operational Quality Assurance Program complies with this guide with the following clarification:

1. Met-Ed shall comply with the Regulatory Position established in this Regulatory Guide in that QA programmatic/administrative requirements included therein shall apply to maintenance and modification activities even though such requirements were not in effect originally. Technical requirements associated with maintenance and modifications shall be the original requirements or better (e.g., code requirements, material properties, design margins, manufacturing processes, and inspection requirements).

2. The guidance or Regulatory Guide 1.54 shall be followed for organic protective coatings selected and evaluated in accordance with pertinent sections of ANSI N101.2 when applied to interior surfaces of the containment. The supplier's quality assurance program shall be approved prior to implementation. Quality Assurance documentation may not be similar to records and documents listed in Sections 7.4 through 7.8 of ANSI N101.4 but will be evaluated to assure that they provide at least the same degree of documentation as required by this standard.

NRC Regulatory Guide 1.58, August 1973

Qualifications of Nuclear Power Plant Inspection, Examination, and Testing Personnel

Endorses ANSI N45.2.6 - 1973

The Operational Quality Asssurance Program complies with this guide with the following clarification:

1. The guidance of Regulatory Guide 1.58 shall be followed as it pertains to the qualifications of personnel who verify conformance of work activities to quality requirements. The qualifications of plant operating personnel concerned with day-to-day operation, maintenance, and certain technical services shall conform to Regulatory Guide 1.8.

2. Not all personnel who approve inspection and test procedures will be certified as meeting the Level III capability requirements of ANSI N45.2.6 - 1973, but personnel who approve inspection and test procedures will be determined by management, through evaluation of their education, training and experience, to be fully qualified and competent to approve such procedures. The basis for the determination will be documented.

NRC Regulatory Guide 1.64, Rev. 2, June 1976

Quality Assurance Requirements for the Design of Nuclear Power Plants

Endorses ANSI N45.2.11 - 1974

Met Ed shall comply with the Regulatory Position established in this Regulatory Guide in that QA programmatic/administrative requirements included therin shall apply to maintenance and modification activities even though such requirements were not in effect originally. Technical requirements associated with maintenance and modifications shall be the original requirements or better (e.g., code requirements, material properties, design margins, manufacturing processes, and inspection requirements).

The Operational Quality Assurance Program complies with this guide with the following clarification to paragraph C.2(1) of Regulatory Guide 1.64: If in an exceptional circumstance the designer's immediate Supervisor is the only technically qualified individual available, this

review can be conducted by the Supervisor, providing that: (a) the other provisions of the Regulatory Guide are satisfied, and (b) the justification is individually documented and approved in advance by the Supervisor's management, and (c) quality assurance audits cover frequency and effectiveness of use of Supervisors as design verifiers to guard against abuse.

NRC Regulatory Guide 1.94, Rev. 1, April 1976

Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel during the Construction Phase of Nuclear Power Plants

Endorses ANSI N45.2.5 - 1974

The Operational quality Assurance Program complies with this guide with the following clarification:

> Met-Ed shall comply with the Regulatory Position established in this Regulatory Guide in that QA programmatic/administrative requirements included therin shall apply to maintenance and modification activities even though such requirements were not in effect originally. Technical requirements associated with maintenance and modifications shall be the original requirements or better (e.g., code requirements, material properties, design margins, manufacturing processes, and inspection requirements).

NRC Regulatory Guide 1.116, Rev. 0-R, June 1976

Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems

Endorses ANSI N45.2.8

The Operational Quality Assurance Program complies with this guide with the following clarification:

> Met-Ed shall comply with the Regulatory Position established in this Regulatory Guide in that QA programmatic/administrative requirements included therein shall apply to maintenance and modification activities even though such requirements were not in effect originally. Technical requirements associated with maintenance and modifications, shall be the original requirements or better (e.g., code requirements, material properties, design margins, manufacturing processes, and inspection requirements).

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NRC Regulatory Guide 1.26, Rev. 3, February 1976

Quality Group Classifications and Standard for Water, Steam and Radioactive Waste Containing Components of Nuclear Power Plants

Since the original design and construction of the Three Mile Island Plants was to different classification criteria than contained in this guide; Met-Ed will comply with the regulatory position of this guide with the following clarifications:

1. For modifications to existing plant systems and for new construction, iters will be classified according to this guide providing such action will improve the safety of the system being modified or make a significant improvement in overall plant safety. Otherwise the items will be classified the same as the original design and construction.

2. Tie-in's to existing plant systems will be made to the same or better code, standard and technical requirements which were applicable to the system to which the tie-in is to be made.

NRC Regulatory Guide 1.63, Rev. 2, July 1978

Electric Penetration Assemblies in Containment Structures for Light Water Cooled Nuclear Power Plants

Met-Ed will comply with the regulatory position of this guide with the following clarification:

For modifications to existing structures and to new constructions, this guide will be utilized providing its use will improve the safety of the structure being modified or make a significant improvement in overall plant safety. Otherwise, the code, standard and technical requirements applicable to the original design and construction will be utilized.

NRC Regulatory Guide 1.144, January 1979

Auditing of Quality Assurance Programs for Nuclear Power Plants

Met-Ed is in basic agreement with the position set forth by your staff in your draft response to the subject regulatory guide. Listed below are comments to the major points raised in your response:

1. Section C.3.a(2)

   The proposed scheduling requirement for internal audits appears to change the basis for having a rational, programmatic approach to auditing. In its place, the new regulatory guide requires mandatory auditing of all program elements on a yearly basis. The latter would require that all elements obtain the same attention regardless of importance, past performance, or to what extent other aspects of quality assurance measuring and evaluating techniques are used; as an example, the extent to which surveillance and process monitoring is used.

2. Section C.3.b(1)

   We agree that source inspection provides some controlled basis for replacing the need for external audits. It is recommended that the use of quality assurance program surveillance should also be viewed as another alternative.

3. Section C.3.b(2)

   We agree with the staff's position that the new regulatory guide wording will lead to "audit proliferation". While the licensee is responsible for procurement control. This can be exercised through an annual evaluation of the contractor's performance using pertinent results from manufacturing surveillance, source inspection, receiving inspection, and other applicable factors. The evaluation would include a recommendation as to the need for a scheduled program or problem area audit. Hence, auditing, like surveillance and inspection, should be treated as a quality assurance tool used for evaluation. Furthermore, the recommendation to audit should include provisions for reviewing the importance and impact of the particular contractor's scope and status.

NRC Regulatory Guide 1.88, Rev. 2, October, 1976

Collection, Storage, and Maintenance of Nuclear Power Plant Availability Assurance Records

Met-Ed will comply with this regulatory guide with the following clarification:

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APPENDIX C, PART 2

1. With regard to Section 5-6 of ANSI N45.2.6-1974 titled Facility, Met-Ed will comply with the requirements of Section 5-6 of the 1979 revision in lieu of the 1974 revision.

   In order to reach full compliance with this modified position, Met-Ed must contruct a new facility. This facility is scheduled for construction in 1980.