1.9 THREE MILE, ISLAND - 2 (TMI-2) ACTION PLAN REQUIREMENTS FOR APPLICANTS FOR AN OPERATING LICENSE

On October 31, 1980, D. G. Eisenhut, Director, Division of Licensing, Office of Nuclear Reactor Regulation, issued a letter to "All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits" addressing Post TMI - requirements (NUREG-0737). Enclosure 2 to this document identified TMI Action Plan Requirements for Applicants for an Operating License approved for implementation by the Commission at the time of issuance.

In this section, those specific requirements of enclosure 2, as cited above, which affect Waterford 3 are identified and addressed. Additionally, Table 1.9-1 provides a reference to an FSAR section or sections where Waterford 3's method of compliance is described.

1.9.1 SHIFT TECHNICAL ADVISOR (I.A.1.1.)

Position

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor (STA) may serve more than one unit at a multiunit site if qualified to perform the advisor function for the various units.

The STA shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The STA shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the STAs that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

Response

Technical Advisors with engineering expertise and special training in plant dynamic response will be available on shift to advise and assist the Shift Supervisor in the event of an accident. Subsection 13.1.2.1.5.1 of the FSAR has been expanded to discuss the details of this commitment.

STA requirements are specified in the Technical Specifications.

1.9.2 SHIFT SUPERVISOR ADMINISTRATIVE DUTIES (I.A.1.2)

Position

Review the administrative duties of the shift supervisor and delegate functions that detract from or are subordinate to the management responsibility for assuring safe operation of the plant to other personnel not on duty in the control room.

Response

→(LBDCR 13-015, R308)
A review of the Shift Supervisor's (Shift Manager's) duties has been conducted to relieve him of those administrative functions that detract from or are subordinate to his management responsibility for assuring the safe operation of the plant. These administrative duties are delegated as appropriate to other operations personnel not on duty in the control room.
←(LBDCR 13-015, R308)

1.9.3 SHIFT MANNING (I.A.1.3)

Position

This position defines shift manning requirements for normal operation. The letter of July 31, 1980 from D. G. Eisenhut to all power reactor licensees and applicants sets forth the interim criteria for shift staffing (to be effective pending general criteria that will be the subject of future rulemaking). Overtime restrictions were also included in the July 31, 1980 letter.

Response

Shift manning requirements are specified in the Technical Specifications.

1.9.4 IMMEDIATE UPGRADING OF REACTOR OPERATOR AND SENIOR REACTOR OPERATOR TRAINING AND QUALIFICATIONS (I.A.2.1)

Position

Effective December 1, 1980, an applicant for a senior reactor operator (SRO) license will be required to have been a licensed operator for 1 year.

Response

Licensed personnel training qualification requirements are discussed in 2 Subsection 13.2.1.2 of the FSAR.

Licensed operators have been screened and their job positions analyzed using position task analyses to determine training requirements. FSAR Subsection 13.2.1 has been revised to reflect this commitment.

1.9.5 ADMINISTRATION OF TRAINING PROGRAMS (I.A.2.3)

Position

Pending accreditation of training institutions, licensees and applicants for operating licenses will assure that training center and facility instructors who teach systems, integrated responses, transient, and simulator courses demonstrate senior reactor operator (SRO) qualifications and be enrolled in appropriate requalification programs.

Response

→(DRN 01-758)

The associated Waterford 3 Training Programs are based on a systematic approach to training. The Licensed Operator and Shift Technical Advisor Programs were initially accredited by INPO February 25, 1987. Instructors who routinely teach systems important to plant safety, integrated responses, transient and simulator courses have demonstrated SRO qualifications and are enrolled in appropriate requalification programs.

←(DRN 01-758)

1.9.6 REVISE SCOPE AND CRITERIA FOR LICENSING EXAMINATIONS (I.A.3.1)

Position

All reactor operator license applicants shall take a written examination with a new category dealing with the principles of heat transfer and fluid mechanics, a time limit of nine hours, and a passing grade of 80 percent overall and 70 percent in each category.

All senior reactor operator license applicants shall take the reactor operator examination, an operating test, and a senior reactor operator written examination with a new category dealing with the theory of fluids and thermodynamics, a time limit of seven hours, and a passing grade of 80 percent overall and 70 percent in each category.

Simulator examinations will be included as part of the licensing examinations.

Response

The scope and criteria for requalification of operating personnel were to reflect the added requirements of:

a. Instruction in heat transfer, fluid flow, thermodynamics and mitigation of accidents involving a degraded core.

b. Accelerated requalification quiz and exam grade levels.

c. Specified reactivity manipulations.

1.9.7 INDEPENDENT SAFETY ENGINEERING GROUP (I.B.1.2)

Position

Each applicant for an operating license shall establish an onsite independent safety engineering group (ISEG) to perform independent reviews of plant operations.

The principal function of the ISEG is to examine plant operating characteristics, NRC issuances, Licensing Information Service advisories, and other appropriate sources of plant design and operating experience information that may indicate areas for improving plant safety. The ISEG is to perform independent review and audits of plant activities including maintenance, modifications, operational problems, and operational analysis, and aid in the establishment of programmatic requirements for plant activities. Where useful improvements can be achieved, it is expected that this group will develop and present detailed recommendations to corporate management for such things as revised procedures or equipment modifications.

Another function of the ISEG is to maintain surveillance of plant operations and maintenance activities to provide independent verification that these activities are performed correctly and that human errors are reduced as far as practicable. ISEG will then be in a position to advise utility management on the overall quality and safety of operations. ISEG need not perform detailed audits of plant operations and shall not be responsible for sign-off functions such that it becomes involved in the operating organization.

Response

➔(DRN 00-576, R11; LBDCR 13-015, R308)
The Independent Technical Review (ITR) function is no longer a specific designated function for plant operating oversight and reduction of human errors. Rather, the oversight function is performed as part of on-going processes for assessing plant operation at Waterford 3. Those functions include activities conducted by Oversight, Performance Improvement, Regulatory Assurance, and Engineering. Human performance improvement has been integrated into all site functions and is a goal for all departments. The combination of these various activities meets the intent for independent safety review for the commitment to NUREG-0737, Section I.B.1.2, as follows:
🡐(LBDCR 13-015, R308)

- An operations experience group evaluates and distributes in-house and industry information to appropriate EOI personnel for review. Recommendations resulting from these reviews are implemented to improved reliability and safety.
- Design engineering support for Waterford 3 is located on site, making the engineers readily available to address potential design basis issues.
- Plant engineering support for Waterford 3 is located on site and is responsible for optimizing system performance and reliability and for providing technical assistance to the Operations and Maintenance organizations.
- The corrective action program contains the essential process elements of problem reporting, root cause analysis, and corrective action.
- The use of assessments provides information on performance trends and improvements for EOI and Waterford 3 management.

➔(DRN 03-657, R12-C)
- Oversight committees (SRC and OSRC) review plant operations.

🡐(DRN 03-657, R12-C)
➔(LBDCR 13-015, R308)
- Management participation in the Performance Improvement program process (e.g., review of condition reports, grading the significance of condition reports, review of root cause analyses, and determination of which conditions relate to human performance) ensures that the quality and integrity of the program is maintained and that problems are visible to Waterford 3 management.

🡐(DRN 00-576, R11; LBDCR 13-015, R308)

1.9.8 SHORT-TERM ACCIDENT ANALYSIS AND PROCEDURE REVISION (I.C.1)

Position

Analyze small-break LOCAs over a range of break sizes, locations and conditions (including some specified multiple equipment failures) and inadequate core cooling due to both low reactor coolant system inventory and the loss of natural circulation to determine the important phenomena involved and expected instrument indications. Based on these analyses, revise as necessary emergency procedures and training.

Response

A small break LOCA analysis has been conducted on Waterford 3 and is discussed in Sections 6.3 and 15.6. Additionally, LP&L has participated in the CE Owners Group effort to develop Emergency Procedure Guidelines (EPGs) which are based, in part, on generic analyses of small-break LOCAs. The Waterford 3 emergency operating procedures implement the CE Owners Group EPGs.

Cold license candidate's onsite training includes training in heat transfer, fluid flow and thermodynamics. It also includes comprehensive training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged. Subsection 13.2.1 of the FSAR discusses this training. The "contingency plan" for cold license candidates calls for the review and discussion of Operating, Emergency and Abnormal Procedures by operational personnel (also in Subsection 13.2.1). This review and discussion when combined with the previously mentioned training should prevent procedural inadequacies prior to implementation of the procedure.

1.9.9 SHIFT RELIEF AND TURNOVER PROCEDURES (I.C.2)

Position

Revise plant procedures for shift relief and turnover to require signed checklists and logs to assure that the operating staff (including auxiliary operators and maintenance personnel) possess adequate knowledge of critical plant parameter status, system status, availability and alignment.

Response

→(DRN 02-559, R12, LBDCR 15-027, R309)

Entergy Operations Procedures, such as EN-OP-115 (and its associated progeny procedures), entitled "Conduct of Operations", define the responsibilities and methods to be used by the Operations Group to ensure that plant operations are conducted in conformance with applicable legal requirements and regulations and dictates of good operating practices.

←(DRN 02-559, R12, LBDCR 15-027, R309)

→(LBDCR 13-015, R308)

1.9.10 SHIFT MANAGER RESPONSIBILITIES (I.C.3)

←(LBDCR 13-015, R308)

Position

Issue a corporate management directive that clearly establishes the command duties of the shift supervisor and emphasizes the primary management responsibility for safe operation of the plant. Revise plant procedures to clearly define the duties, responsibilities and authority of the shift supervisor and the control room operators.

Response

→(DRN 02-559, R12. LBDCR 15-027, R309)

Entergy Operations Procedures, such as EN-OP-115 (and its associated progeny procedures), entitled "Conduct of Operations", define the responsibilities and methods to be used by the Operations Group to ensure that plant operations are conducted in conformance with applicable legal requirements and regulations and dictates of good operating practices. A corporate management directive concerning command duties of the shift supervisor is issued yearly in accordance with the Technical Specifications.

←(DRN 02-559, R12, LBDCR 15-027. R309)

→(LBDCR 13-015, R308)
and dictates of good operating practices. A corporate management directive concerning command duties of the shift manager is issued yearly in accordance with the Technical Specifications.
←(LBDCR 13-015, R308)

1.9.11 CONTROL ROOM ACCESS (I.C.4)

Position

Revise plant procedures to limit access to the control room to those individuals responsible for the direct operation of the plant, technical advisors, specified NRC personnel, and to establish a clear line of authority, responsibility, and succession in the control room.

Response

→(DRN 02-559, R12, LBDCR 15-027. R309)
Entergy Operations Procedures, such as EN-OP-115 (and its associated progeny procedures), entitled "Conduct of Operations", define the responsibilities and methods to be used by the Operations Group to ensure that plant operations are conducted in conformance with applicable legal requirements and regulations and dictates of good operating practices.
←(DRN 02-559, R12, LBDCR 15-027, R309)

1.9.12 PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF (I.C.5)

Position

Each applicant for an operating license shall prepare procedures to assure that operating information pertinent to plant safety originating both within and outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and retraining programs. These procedures shall:

(1) Clearly identify organizational responsibilities for review of operating experience the feedback of pertinent information to operators and other personnel, and the incorporation of such information into training and retraining programs;

(2) Identify the administrative and technical review steps necessary in translating recommendations by the operating experience assessment group into plant actions (e.g., changes to procedures; operating orders);

(3) Identify the recipients of various categories of information from operating experience (i.e., supervisory personnel, shift technical advisors, operators, maintenance personnel, health physics technicians) or otherwise provide means through which such information can be readily related to the job functions of the recipients;

(4) Provide means to assure that affected personnel become aware of and understand information of sufficient importance that should not wait for emphasis through routine training and retraining programs;

(5) Assure that plant personnel do not routinely receive extraneous and unimportant information on operating experience in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency;

(6) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to operators and other personnel until resolution is reached; and,

(7) Provide periodic internal audit to assure that the feedback program functions effectively at all levels.

Response

Administrative Procedures ensure that operating experience from within and outside the LP&L organization is provided to operators and other operating personnel and is incorporated in training programs in accordance with NRC instructions. Waterford 3 Nuclear Training Procedure NTP-102, entitled "Licensed Reactor Operator Requalification" provides the program description and provides instructions for the conduct and documentation of operator retraining. Prior to the cold license examination this program shall be utilized to maintain the proficiency and basic levels of knowledge of the licensed personnel.

1.9.13 GUIDANCE ON PROCEDURES FOR VERIFYING CORRECT PERFORMANCE OF OPERATING ACTIVITIES (I.C.6)

Position

It is required that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations and maintenance activities independent of the people performing the activity or both.

Implementation of automatic status monitoring if required will reduce the extent of human verification of operations and maintenance activities but will not eliminate the need for such verification in all instances. The procedures adopted by the licensees may consist of two phases--one before and one after installation of automatic status monitoring equipment, if required.

Response

→(DRN 02-559, R12, LBDCR 15-027, R309)

Entergy Operations Procedures, such as EN-OP-115 (and its associated progeny procedures), entitled “Conduct of Operations”, define the responsibilities and methods to be used by the Operations Group to ensure that plant operations are conducted in conformance with applicable legal requirements and regulations and dictates of good operating practices.

←(DRN 02-559, R12, LBDCR 15-027, R309)

1.9.14 NSSS VENDOR REVIEW OF PROCEDURES (I.C.7)

Position

Obtain nuclear steam supply system (NSSS) vendor review of low-power testing procedures to further verify their adequacy.

Response

Emergency Operating Procedures were developed using the Combustion Engineering Emergency Procedure Guidelines (EPGs) as their basis. Use of the EPGs satisfies the I.C.7 requirement for vendor review.

A detailed review and independent analysis of both low-power and power ascension test procedures was conducted to verify the adequacy of these procedures. Additionally the NSSS vendor assisted in analyzing and interpreting the testing program results.

1.9.15 CONTROL ROOM DESIGN REVIEWS (I.D.1)

Position

All licensees and applicants for operating licenses will be required to conduct a detailed control room design review to identify and correct design deficiencies. This detailed control room design review is expected to take about a year. Therefore, the Office of Nuclear Reactor Regulation (NRR) requires that those applicants for operating licenses who are unable to complete this review prior to issuance of a license make preliminary assessments of their control rooms to identify significant human factors and instrumentation problems and establish a schedule approved by NRC for correcting deficiencies. These applicants will be required to complete the more detailed control room reviews on the same schedule as licensees with operating plants.

Response

In accordance with the requirements of NUREG 0737 Supplement 1 LP&L performed a Detailed Control Room Design Review (DCRDR) in 1984-85. The DCRDR results and proposed changes to address human engineering discrepancies were submitted to the NRC via W3P85-1015 dated April 30, 1985 and supplemented via W3P86-2557 dated October 14, 1986 and W3P88-1240 dated August 3, 1988. By letter dated June 13, 1989, the NRC submitted a Supplemental Safety Evaluation which concluded that LP&L meets all of the nine DCRDR requirements of Supplement 1 to NUREG-0737.

1.9.16 PLANT SAFETY PARAMETER DISPLAY SYSTEM (I.D.2)

Position

Each applicant and licensee shall install a Safety Parameter Display System (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status.

Response

Based on the requirement set forth in NUREG-0737, Supplement 1, the Waterford 3 SPDS has been designed to provide a concise display of critical plant variables to the control room operators to aid them in rapidly and reliably determining the safety status of the plant. Since the SPDS is a software implementation on the preexisting Plant Monitor Computer (PMC), no additional hardware, save dedicated SPDS terminals, was necessary.

Based on a review of the Waterford 3 FSAR, SER and draft Technical Specifications, it was determined that the implementation of the SPDS would have no adverse impact on the safe operation of existing instrumentation and equipment. Furthermore, the addition of the SPDS did not affect any FSAR analyses or Technical Specifications.

A Safety Analysis Report of the Waterford 3 SPDS was developed in April, 1984 in order to provide the details of the SPDS implementation and to satisfy the NUREG-0737, Supplement 1 request for a written safety analysis. Included as part of the Waterford 3 SPDS Safety Analysis Report is a description of the PMC hardware and software (to include SPDS), the details of the SPDS Parameter Selection, the hierarchy of the SPDS displays, a discussion of the PMC reliability, a description of the Human Factors Principles employed, the implementation of the SPDS Verification and Validation as accomplished through the PMC Startup Testing and a summary of the Waterford 3 SPDS compliance with NUREG-0737, Supplement 1.

The SPDS is further described in Appendix 7.7A.

1.9.17 TRAINING DURING LOW-POWER TESTING (I.G.1)

Position

Define and commit to a special low-power testing program approved by NRC to be conducted at power levels no greater than five percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training.

→(LBDCR 13-014, R309)

Response

Chapter 14 (Section 14.2) reflects the startup and low power tests that were conducted for training. Subsection 13.2.1.1 describes the involvement of plant staff personnel with the low power test training.

←(LBDCR 13-014, R309)

1.9.18 REACTOR COOLANT SYSTEM VENTS (II.B.1)

Position

Each applicant and licensee shall install reactor coolant system (RCS) and reactor vessel head high point vents remotely operated from the control room. Although the purpose of the system is to vent noncondensible gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10CFR Part 50, "General Design Criteria." The vent system shall be designed with sufficient redundancy that assures a low probability of inadvertent or irreversible actuation.

Each licensee shall provide the following information concerning the design and operation of the high point vent system:

(1) Submit a description of the design, location, size, and power supply for the vent system along with results of analyses for loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10CFR50.46.

(2) Submit procedures and supporting analysis for operator use of the vents that also include the information available to the operator for initiating or terminating vent usage.

Response

In accordance with the above position, Waterford 3 has included in its design a Reactor Coolant Gas Venting System to allow for remote venting of noncondensible gases, which may collect in the RCS, via a reactor vessel head vent or pressurizer steam space vent during post-accident situations. The design bases, a system description, and evaluation have been included in FSAR Subsection 5.4.15.

Test procedures were developed in accordance with subsection IWV of Section XI of the ASME Code for Category B valves.

Operating procedures were developed to address the use of the Reactor Coolant System Vents, defining the conditions under which the vents should be used or not used and including information pertaining to the initiating and terminating vent usage. Procedures for the use and non-use of RCS vents were prepared in accordance with the guidelines supplied by the NSSS vendor.

The vent system was reviewed as part of Item I.D.1 (see FSAR Subsection 1.9.15).

1.9.19 DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POST-ACCIDENT OPERATIONS (II.B.2)

Position

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50 percent of the core radioiodine, 100 percent of the core noble gas inventory, and one percent of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

Response

Refer to FSAR Appendix 12.3A

1.9.20 POST-ACCIDENT SAMPLING CAPABILITY (II.B.3)

Position

A design and operational review of the reactor coolant and containment atmosphere sampling line systems shall be performed to determine the capability of personnel to promptly obtain (less than one hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18-3/4 rem to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (in less than two hours) certain radionuclides that are indicators of the degree of core damage. Such radionuclides are noble gas (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and nonvolatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift).

Response

Waterford 3 complies with this requirement as clarified in NUREG 0737 October 31, 1980 (see Subsection 9.3.8).

1.9.21 TRAINING FOR MITIGATING CORE DAMAGE (II.B.4)

Position

Licensees are required to develop a training program to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged. They must then implement the training program.

Response

Comprehensive training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged is conducted during licensed operator training and requalification training. Subsection 13.2.2.4 reflects this commitment.

1.9.22 PERFORMANCE TESTING OF BOILING-WATER REACTOR AND PRESSURIZED-WATER REACTOR RELIEF AND SAFETY VALVES (II.D.1)

Position

Pressurized-water reactor and boiling-water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

Response

The compliance of the Waterford 3 plant with NUREG 0737 Item II.D.1 requirements is described in W3P82-4011, dated 12/29/82. NRC acceptance of Waterford 3 compliance with this item is documented in a Safety Evaluation Report dated 04/15/88. Also refer to Subsection 3.9.3.3.

Waterford 3 design does not have block valves.

1.9.23 DIRECT INDICATION OF RELIEF AND SAFETY-VALVE POSITION (II.D.3)

Position

Reactor coolant system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe.

Response

To provide the operator with an unambiguous indication of primary safety valve position, a non-safety grade acoustic monitoring system was installed. The following discussion provides the manner in which the requirements of NUREG 0737 Item II.D.3 has been met.

The system ensures high reliability and testability and discriminates against inadvertent actuation due to impact events, cross talk and background noise. Two acoustic sensors in two separate instrumentation channels (one is mandated in paragraph (3) of NUREG 0737/II.D.3) are installed downstream of and close to each pressurizer safety valve. Their qualification for radiation and temperature exceeds actual DBA requirements. A metal enclosure protects the sensors against mechanical damage and provides support for coaxial cable in flexible conduit. The enclosure is installed on a machined mounting block strapped to the pipe. Any failure of the mounting bracket, however, does not compromise the function of either the safety valve or any other safety-related component.

The coaxial cable from the sensors to the containment penetrations is qualified for DBA conditions and/or suitably protected.

The four converters/amplifiers are located in the RAB in a non-harsh environment area served by a safety-related HVAC system (SVS). The converted signal is processed and analyzed by four individual signal conditioners located in the Valve and Loose Parts Monitoring (V&LPM) Panel in the Control Room (see Subsection 4.4.6.1). The signal from all four monitors are combined into a single "open" position valve alarm on the RTG Board. In the lower part of this board, there is a pair of "open-closed" light indicators which enables the operator to determine which safety valve has opened.

The V&LPM Panel and converters/amplifiers are energized by a vital power source.

The utilization of information provided to the operator by this position monitoring system shall be integrated into alarm response and emergency procedures and into operator training.

Backup methods of determining valve position are provided which utilize each safety valve discharge line temperature and quench tank temperature and water level, as discussed in Subsection 5.2.5.1.3.

1.9.24 AUXILIARY FEEDWATER SYSTEM EVALUATION (II.E.1.1)

<u>Position</u>

The Office of Nuclear Reactor Regulation is requiring reevaluation of the auxiliary feedwater (AFW) systems for all PWR operating plant licensees and operating license applications. This action includes:

(1) Perform a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss-of-main-feedwater-transient conditions. Particular emphasis is given to determining potential failures that could result from human errors, common causes, single-point vulnerabilities, and test and maintenance outages;

(2) Perform a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 and associated Branch Technical Position ASB 10-1 as principal guidance; and

(3) Reevaluate the AFW system flowrate design bases and criteria.

<u>Response</u>

The Emergency Feedwater System requirements evaluation and reliability analysis appears as FSAR Appendices 10.4.9A and 10.4.9B.

→(EC-33720, R307)
Reference to Appendix 10.4.9B. Response to Position (1) appears as FSAR Appendix 10.4.9B.
←(EC-33720, R307)

Response to Positions (2) and (3) appear in table form in FSAR Appendix 10.4.9A as Tables 10.4.9A-1 and 10.4.9A-3 respectively.

Additionally, a review of the Waterford 3 EFS design Technical Specifications and Operating Procedures against generic short term and long term recommendations is provided in FSAR Appendix 10.4.9A as Table 10.4.9A-2.

1.9.25 AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION AND FLOW INDICATION (II.E.1.2)

PART 1: Auxiliary Feedwater System Automatic Initiation-

Position

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system (AFWS), the following requirements shall be implemented in the short term:

(1) The design shall provide for the automatic initiation of the AFWS.

(2) The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of AFWS function.

(3) Testability of initiating signals and circuits shall be a feature of the design.

(4) The initiating signals and circuits shall be powered from the emergency buses.

(5) Manual capability to initiate the AFWS from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

(6) The ac motor-driven pumps and valves in the AFWS shall be included in the automatic actuation (simultaneous and/or sequential) of the loads onto the emergency buses.

(7) The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

PART 2: Auxiliary-Feedwater System Flowrate Indication

Position

Consistent with satisfying the requirements set forth in General Design Criterion 13 to provide the capability in the control room to ascertain the actual performance of the AFWS when it is called to perform its intended function, the following requirements shall be implemented:

(1) Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

(2) The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

Response

Security Related Information Text Withheld Under 10 CFR 2.390

1.9.26 EMERGENCY POWER SUPPLY FOR PRESSURIZER HEATERS (II.E.3.1)

Position

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17 and 20 of Appendix A to 10 CFR Part 50 for the event of loss of offsite power, the following positions shall be implemented:

(1) The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

(2) Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

(3) The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

(4) Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

Response

Security Related Information
Text Withheld Under 10 CFR 2.390

1.9.27 DEDICATED HYDROGEN PENETRATIONS (II.E.4.1)

Position

Plants using external recombiners or purge systems for postaccident combustible gas control of the containment atmosphere should provide containment penetration systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single-failure requirements of General Design Criteria 54 and 56 of Appendix A to 10CFR50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

The procedures for the use of combustible gas control systems following an accident that results in a degraded core and release of radioactivity to the containment must be reviewed and revised, if necessary.

Response

This position is not applicable to Waterford 3. Waterford 3 has redundant recombiners permanently installed inside containment (see Table 1.9-1).

Plant procedures for the use of combustible gas control systems following an accident resulting in a degraded core and release of radioactivity to the containment were reviewed and revised as necessary.

For additional information relating to the use of the hydrogen recombiners, refer to FSAR Subsection 6.2.5.

1.9.28 CONTAINMENT ISOLATION DEPENDABILITY (II.E.4.2)

Position

(1) Containment isolation system designs shall comply with the recommendations of Standard Review Plan Section 6.2.4 (i.e., that there be diversity in the parameters sensed for the initiation of containment isolation).

(2) All plant personnel shall give careful consideration to the definition of essential and nonessential systems, identify each system determined to be essential, identify each system determined to be nonessential, describe the basis for selection of each essential system, modify their containment isolation designs accordingly, and report the results of the reevaluation to the NRC.

(3) All nonessential systems shall be automatically isolated by the containment isolation signal.

(4) The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

(5) The containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions.

(6) Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979 must be sealed closed as defined in SRP 6.2.4, item II.3.f during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days.

(7) Containment purge and vent isolation valves must close on a high radiation signal.

Response

(1) All containment isolation valves receive actuation signals from diverse sensed parameters. FSAR Table 6.2-32 reflects compliance to this requirement. As indicated in the table all isolation valves receive actuation from one or more of the following signals:

(a) Containment Isolation Actuation Signal (CIAS) - This signal is actuated by either high containment pressure or low pressurizer pressure

(b) Safety Injection Actuation Signal (SIAS) - This signal is generated by different circuitry and relays than CIAS, but as CIAS, is actuated by either high containment pressure or low pressurizer pressure

(c) Main Steam Isolation Signal (MSIS) - This signal is generated by low steam generator pressure or high containment pressure

(d) Containment Purge Isolation Signal,(CPIS) - This signal is generated by high containment radiation. This signal is used in combination with CIAS to isolate the Containment Purge valves (see FSAR Table 6.2-32 items 10 and 11).

(2) FSAR Table 6.2-32 indicates whether each system receiving the above signals is considered essential or non-essential. Essential systems are those necessary to assure:

(a) the integrity of the reactor coolant pressure boundary

(b) the capability to shutdown the reactor and maintain it in a safe shutdown condition

→(DRN 04-1619, R14)

(c) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline in 10CFR50.67.

←(DRN 04-1619, R14)

The capability to manually override the isolation signal and reopen containment isolation valves on those non-essential systems which may be desirable to operate after the accident has been provided.

This feature is noted as a footnote in the Actuation Signal column of FSAR Table 6.2-32 and is in place on the valves listed in Table 1.9-3.

A review of the classification of essential and non-essential systems against the requirements of Revision 2 of Regulatory Guide 1.141 shall be conducted upon its issuance.

(3) As indicated in FSAR Table 6.2-32, all non-essential systems shall be automatically isolated by one or more of the above signals.

(4) Resetting of the isolation signals discussed above will not result in automatic. reopening of any containment isolation valves. Reopening of these valves can only be effected by deliberate operator action after reset of the signal. Each valve must then be opened individually by the operator.

(5) Containment Setpoint Pressure

The containment pressure trip setpoint and allowable value for initiating containment isolation have been derived using the explicit setpoint methodology. This methodology applies a statistical combination of the individual uncertainty components (instrument loop error, setpoint variance, instrument drift, etc.) to establish a total instrument channel uncertainty. The trip setpoint established, therefore, ensures sufficient margin between the technical specification limit and the nominal trip setpoint. At the same time the setpoint is high enough to minimize inadvertent actuation of containment isolation.

(6) Containment Purge Isolation Valve

→(DRN 01-758)

An analysis has been performed to determine the operability of the containment purge valves. This analysis has shown that the valves are capable of closing against the most severe design basis accident flow conditions when the valve opening is limited to 52 degrees. Modifications have been made to limit purge valve opening.

←(DRN 01-758)

In addition, the purge valve isolation signals are designed such that they cannot be locked, reset, or overridden.

The FSAR has been revised to reflect consistency with the changes to actuation signals for the containment isolation system indicated in FSAR Table 6.2-32.

(7) As indicated in FSAR Table 6.2-32, the Containment Purge Isolations Valves (Pens. Nos. 10 and 11) are automatically isolated on CPIS (high radiation).

Waterford 3 Plant Operating Procedures entitled "Loss of Coolant" and "High Airborne Activity" contain details of isolation initiation and subsequent action to be taken. Additionally, diverse isolation signals have been installed for the purpose of containment isolation.

1.9.29 ADDITIONAL ACCIDENT-MONITORING INSTRUMENTATION (II.F.1) NOBLE GAS EFFLUENT MONITOR

Position

Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions. Multiple monitors are considered necessary to cover the ranges of interest.

→(DRN 01-758)

(1) Noble gas effluent monitors with an upper range capacity of 10^5 μCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

←(DRN 01-758)

→(DRN 01-758)

(2) Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (as low as reasonably achievable (ALARA) concentrations to a maximum of 10^5 μCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of ten.

←(DRN 01-758)

SAMPLING AND ANALYSIS OF PLANT EFFLUENTS

Position

Because iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

CONTAINMENT HIGH-RANGE RADIATION MONITOR

Position

In containment radiation-level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be developed and qualified to function in an accident environment.

CONTAINMENT PRESSURE MONITOR

Position

A continuous indication of containment pressure shall be provided in the control room of each operating reactor. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and -5 psig for all containments.

CONTAINMENT WATER LEVEL MONITOR

Position

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWRs and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided for PWRs and shall cover the range from the bottom of the containment, to the elevation equivalent to a 600,000 gallon capacity. For BWRS, a wide range instrument shall be provided and cover the range from the bottom to five feet above the normal water level of the suppression pool.

→(DRN 01-758)

CONTAINMENT HYDROGEN MONITOR

←(DRN 01-758)

Position

A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the Control Room. Measurement capability shall be provided over the range of 0 to 10 percent hydrogen concentration under both positive and negative ambient pressure.

Response

Noble Gas Effluent Monitors/Containment High Range Radiation Monitor/Sampling and Analysis of Plant Effluent

The accident - radiation monitoring instrumentation enables plant operators to better follow the course of a major accident and thereby assist them in making decisions with reference to mitigating the effects of a major accident. The instrumentation consists of the following monitors:

(1) One (1) Plant Vent Stack Monitor

(2) One (1) Condenser Vacuum Pump Effluent Monitor

(3) One (1) Fuel Handling Bldg Emergency Exhaust Effluent Monitor

(4) Two (2) Main Steam Line Monitors

(5) Two (2) High Range Containment Monitors

Readout of all monitor items for all of the accident radiation monitors is available from the Radiation Monitoring System Computer Remote Console CRT and from separate control room readouts.

Methods for converting instrument readings to release rates for unit time, based on exhaust air flow and considering radionuclide spectrum distribution as a function of time after shutdown, are described in Waterford 3 plant procedures. These procedures, numbered EP-2-050 and EP-2-051 respectively, are entitled "Offsite Dose Assessment" and "Offsite Dose Assessment (computerized)."

Additionally, the two (2) High Range Containment Monitors have separate readout via safety-related remote display/control devices. These remote display/control devices are mounted on the existing safety-related radiation monitoring cabinet (CP-14) located in the Control Room.

Equipment associated with monitors (1) through (4) above are qualified environmentally to IEEE 323-1974, IEEE 344-1975 and NUREG 0588. The High Range Containment Monitoring System including the display devices is qualified to IEEE 323-1974 and IEEE 344-1975 and NUREG 0588. The above is in adherence to Regulatory Guide 1.97 Revision 3, and NUREG 0737 requirements. The source of power for monitors (1) through (3) above are from a non-Class IE interruptible 120V ac source; the High Range Containment radiation monitor are powered from the Class IE interruptible 120V ac source; the Main Steam Line radiation monitors are powered from a non-class IE uninterruptible 120V ac source.

The detectors associated with monitors (1) through (5) have a primary calibration report supplied by the vendor. This calibration report establishes the linearity of the detector. Field calibration of these detectors involves using a single calibration source which when placed in the proper geometry relative to the detector verifies a single point on the detector calibration curve. The frequency of field calibration for the monitors is every 18 months.

Table 1.9-4 summarizes the accident-radiation monitoring instrumentation data required per NUREG 0737.

Individual monitor descriptions follow:

1. High Range Noble Gas Plant Vent Monitor

→(DRN 03-2054, R14)

In adherence to NUREG 0737 and Regulatory Guide 1.97 Rev 3 one (1) High Range Noble Gas Monitor is installed to supplement the range of the existing plant vent stack radiation monitor. The high range noble gas monitor was purchased from General Atomic Company. The particular model which was purchased is entitled "Wide-Range Gas Monitor". This high range noble gas monitor makes use of separate isokinetic nozzles for isokinetic sampling over a flow range of 13,000 SCFM (±20%). This high range noble gas plant vent stack monitor is normally operating.

←(DRN 03-2054, R14)

In order to assure that plant personnel have access to certain assemblies of the monitor (such as the particulate and iodine sample filters) during a highrange release condition, the monitor is divided into separate assemblies that are located in such a way as to minimize personnel exposure to the postulated high levels of radiation. Figure 1.9-3 is a block diagram of the monitor showing the various assemblies of the system and their interconnections.

There are five assemblies: (1) Isokinetic Nozzles; (2) Sample Conditioner; (3) Wide-Range Gas Detectors; (4) Electronics; and (5) Readouts. Each of these assemblies are described below. Skid assemblies (2) and (3) are of open design to allow access to parts and to allow cooling by natural convection. All plumbing and piping are stainless steel and all connections are leak tested prior to shipment. All electrical power needed is distributed from the Wide-Range Gas Detector assembly.

(1) Isokinetic Nozzles

Two sets of isokinetic nozzles are normally used - one for normal and one for high-range conditions. Isokinetic nozzles are used to ensure representative particulate and iodine grab samples (see below). One isokinetic nozzle is mounted inside the duct; the other is mounted inside the sample stream coming from the duct-mounted nozzle. This second nozzle permits drawing of a 0.06 cfm sample under high activity conditions. The normal isokinetic nozzles operate at 1.67 ft^3/min, whereas the high-range isokinetic nozzles operate at 0.06 ft^3/min to minimize activity buildup. Included in this assembly are flow rate tranducers that are connected to a microprocessor to facilitate isokinetic flow control. The location for the nozzle assembly in the effluent stack was chosen in accordance with ANSI N13.1.

(2) Sample Conditioner

This skid assembly is located downstream from the isokinetic nozzles. Its purpose is to provide representative particulate and iodine grab samples for laboratory analysis (in accordance with NUREG 0737), and to prevent contamination of the gas monitor by filtering out large concentrations of radioiodines and particulates. Without the sample conditioner, the monitor would become contaminated and remain upscale even when actual radioactive gas levels decreased. To provide enough filtering material to contain the radioiodines and particulates for the duration of the measured period, special multiple filters are used. Filters on the high-activity flowpath have full 4 X solid lead shielding to minimize personnel exposure. Fast disconnect fittings are provided for the grab sample filters. Grab sample actuation and duration are non-safety RMS control room panel functions (see below). The sample conditioning assembly shall be accessible during a high-range condition to retrieve grab samples. The instrumentation will function in an environment based on the Design Basis Shielding envelope assumptions presented in Table H.F.1-2 of NUREG 0737. The radiation exposures of personnel retrieving the samples will not exceed GDC 19 criteria. Detailed analysis of these filters for particulates and iodines is provided for via the gamma spectroscopy system described in FSAR Subsection 1.9.38. Particulate and iodine filter efficiencies are typically 99 percent.

(3) Wide-Range Gas Detectors

This skid assembly contains the three radioactive gas detectors; this assembly also contains the necessary pumps, flow control valves, flowmeters, etc. Each detector has a solenoid-actuated, checksource to verify proper operation and is full 4X cast-lead shielded to reduce background effects. The 11 decades of Noble gas concentrations are monitored continuously by the three detectors with at least one decade overlap between ranges of the individual detectors. Table 1.9-4 shows the ranges of the three detectors for Xe-133. The low-range detector utilizes a plastic scintillator, whereas the mid-range and high-range detector are solid-state (Cd Te). As above, there are two flow paths through the detectors. During normal operation only the low-range detector is used and the mid-range and high-range detectors are bypassed. As the low-range detector begins to saturate, the flow path is automatically changed to the mid and high-range detectors and the low-range detector is purged. This prevents contamination of the low-range detector so that it will be available when it is automatically returned to service to measure radioactive gas concentrations as they return to low levels. The only solid-state electronics mounted on the skid are the detector preamplifiers which are provided with full 4X lead shielding to minimize radiation exposure. Without shielding, these electronics could not survive accident levels of background radiation for the duration of the accident condition.

(4) Electronics

The monitor is controlled by a microprocessor. The microprocessor performs flow control, valve actuations, engineering conversions, and other calculations and control functions, in addition to data storage. It is remotely located from the detectors, in a low radiation area. It contains the microprocessor, memory, high-voltage power supplies, preamplifiers, battery backup, etc. Mounted adjacent the microprocessor is a junction box for termination of user cables between the RM-80 and other assemblies of the monitor.

(5) Readouts

→(DRN 01-758)

The readout is from the Computer Remote Console CRT (CP-6) and from CP-14, in the Control Room. It is microprocessor based and provides a display of all monitored parameters. These include channel activity in μCi/cc flow rates, alarm status, etc. The local microprocessor also maintains history files of twenty-four 10-min, twenty-four 1-hour and twenty-eight 1-day averages of 4 activity that are available for recall via the CRT. Purge and grab sample control as well as effluent activity recording shall be provided for via labeled control switches and recorders, which will be located in CP-52 control cabinet. From the CP-52 cabinet, the operator is able to select clean prefilters, should one set be loaded to the points where appreciable concentrations of xenon off-gas is produced by iodine decay. In this manner, the Noble gas detector will not interpret iodine-daughter xenon as Noble gas going out the stack. Additionally from this same cabinet the operator may also take grab samples of 1 to 99 minutes duration for the low-range flow path and 0 to 99 seconds for the high-range flow path.

←(DRN 01-758)

2. Condenser Vacuum Pumps Effluent Monitor

This monitor is installed to monitor noble gas effluents from the condenser vacuum pump. Its operation shall be as described for the high-range plant vent stack monitor. Notable exception will be that it shall operate at all times that the condenser vacuum pumps are operating. Due to the high humidity conditions of the sample stream, a large amount of particulate plateout is expected in the sample lines of this monitor. As a result of this expected high plateout, representative samples cannot be assured even with isokinetic sampling. Thus there are no provisions made for isokinetic sampling for this monitor. Filters are used to prevent particulate and iodine activity buildup in the Noble gas monitor. Additional details on the functions provided by this monitor are given in FSAR Subsection 11.5.2.4.1.5.

3. Fuel Handling Building Emergency Exhaust Effluent Monitor

A monitor is installed to monitor the Fuel Handing Building Emergency Exhaust Effluent path. It is of the same type as the Plant Vent Stack monitor.

The operation of the FHB Emergency Exhaust Effluent Monitor shall be as follows:

a) Upon a high radiation signal the potentially contaminated area of the Fuel Handling Building shall be automatically isolated to prevent an unrestricted release of airborne effluents from occurring (see Subsection 9.4.2). This isolation signal shall cause both emergency exhausts to begin operating.

b) The Fuel Handling Building Emergency Exhaust Effluent Monitor shall begin operating at the same time that the Fuel Handling Building emergency exhaust begins operation. The monitor at this time shall be drawing a sample from both emergency exhausts via isokinetic nozzles located inside each emergency exhaust duct. Sampling at this time shall not be isokinetic.

c) Control Room operators after the automatic startup of both emergency exhausts shall select only one emergency exhaust for further use.

d) Selection of one emergency exhaust shall cause a solenoid operated valve to isolate the sample stream coming from the isokinetic nozzle located in the other FHB emergency exhaust. Thus, the monitor shall now be drawing an isokinetic sample from the operator selected FHB emergency exhaust duct.

e) Failure in one emergency exhaust system shall cause the isokinetic nozzle sample stream isolation valve associated with that emergency exhaust system to close, thereby assuring true isokinetic sampling from the other emergency exhaust.

The sequence described above shall be dependent only upon operator selection of one FHB emergency exhaust system. Other monitor functions shall be as described for the plant vent stack monitor.

4. Main Steam Line Monitors

In order to estimate the releases which may occur as a result of the actuation of steam generator secondary relief valves (SRV) and atmospheric steam dump valves (ADV) one collimated GM tube is installed to view the activity of each main steam line. The monitors are mounted within a three inch thick lead shield with a "window" at the front of the detector. The detector reading is in mr/hr, and shall be recorded by the microprocessor as a minimum of twenty-four 10 min, 24 -one hour and 28 -one day averages.

→(DRN 01-758)

Calculational methods are employed to quantify radiological releases based on monitor dose rates. In order to obtain concentrations of 10^{-1} to 10^{3} μCi/cc of Xe-133 in the main steam line a large primary to secondary leak must be present coincident with a large amount of cladding failure. Present with Xe-133 will be other nuclides. Based on the postulated primary system accident scenarios (e.g., cladding failure, fuel failure) and on the assumed steam generator isotope partition factors, isotopic concentrations in the main steam lines can be calculated. Two conversion factors have been developed for the main steam line monitors in terms of mr/hr per μCi/cc of pressurized steam. These factors are based on isotopic fractions arising from gross fuel failure and one percent cladding failure, respectively. A decontamination factor (DF) of 100 is assumed for iodine passing through the steam generators. Noble gases are assumed to pass through the steam generators unhindered. All the applicable noble gases and iodines were considered in the conversion factors. An average energy per disintegration per isotope was used to determine the dose-rate, and the primary dose rate contributors were Kr-87 and Kr-88. The methodology used to determine the conversion factors is taken from the Reactor Shielding Design Manual, 1956, by T Rockwell III. This model accounts for the thickness of the main steam line wall.

←(DRN 01-758)

Estimating radiological releases through the SRV's and the ADV's is calculated by summing the products of the a) concentrations and b) the mass of steam released. Main steam concentrations are obtained from a calculated conservative conversion factor. The mass of steam released can be monitored by using the Main Steam Flow Recorder located between the steam generator and the main steam safety valves.

5. High Range Containment Monitors

The High-Range Radiation Monitor consists of a gamma detector (General Atomic model number RD-23) and cable suitable for use in a containment environment, and support electronics including a readout located in the Control Room. The detector is encased in stainless steel to protect it from containment sprays and high temperatures. The monitor is a safety monitor (Class 1E) and is qualified under LOCA conditions to IEEE 323-1974. Radiation levels of up to 10^8 R/hr are displayed in the control room on a front panel meter. Two level trips are provided for alert and high radiation levels and are independently adjustable over the full range. A failure trip is provided to actuate upon loss of power, high voltage, or signal from the detector. Automatic self-testing is provided to continuously verify detector operation. Outputs are provided for recorders, remote alarm relays, and meters. A separate local display of radiation levels and alarm conditions exists. Energy response is uniform (± 20 percent) for photons in the range of 80 Kev to 3 Mev.

Readout control and data recording for these monitors is provided for as described for Class IE area monitors in FSAR Subsection 12.3.4. The monitors shall be located in containment in a manner as to provide a reasonable assessment of area radiation conditions inside containment. The monitors shall be widely separated so as to provide independent measurements and shall "view" a large fraction of containment volume. Monitors shall not be placed in areas which are protected by massive shielding and shall be reasonably accessible for replacement, maintenance and calibration.

A sustaining signal is generated within the detector corresponding to a predetermined value. A failure alarm will occur if the signal from the detector falls below this value. This feature assures knowledge of the monitor's integrity at all times.

Containment Hydrogen Monitor

See FSAR Subsection 6.2.5.1.

Containment Pressure Monitor

Response

Waterford 3 complies with this requirement.

A continuous recording of containment wide range (-5 to +195 psig) pressure is provided in the control room. This recorded range is approximately four times the design pressure of Waterford 3's steel containment.

→(EC-12329, R306)

Containment wide range pressure monitoring instrumentation consists of 2 redundant Class IE channels. Each channel consists of a pressure transmitter, which utilizes penetration #54 and is physically located outside the containment in the Auxiliary Building, approximately at EL. +5 feet. The accuracy of the transmitter to be used is ± 5% of the span. The pressure transmitter output signal is processed by a process analog control system (PAC) which in turn furnishes signals for the recorder in the main control room and the plant monitoring computer. The entire range of -5 to +195 psig is recorded by one trace of the recorder. A visual indicator is part of the recorder.

←(EC-12329, R306)

Qualification is in accordance with the criteria for Class IE transmitters located outside the containment building. The Containment pressure transmitters meet the requirements of Appendix B to NUREG-0737.

Containment Water Level Monitor

Response

Waterford 3 complies with this requirement.

Two redundant Class IE channels of instrumentation are provided to monitor containment sump level (narrow range). The narrow range monitors meet the recommendations of Regulatory Guide 1.89. Each channel of instrumentation consists of the following:

→ (DRN 99-1035)

a) A level transmitter with a range of 0 - 15', located inside the containment sump. The total depth of the containment sump is 14'. The measurement covers the range from 1.5' at the bottom of the sump to 16.5', thus, total approximate range is 0 to 15'. The containment sump will begin to overflow at an indicated level of 12.5'.

← (DRN 99-1035)

b) A process analog control system (PAC) to monitor level transmittal signal and to develop output signals to the plant monitoring computer and recorder or indicator.

c) A recorder/indicator, mounted on the main control board, for registering the containment sump level. One channel is provided with a recorder and one channel is provided with an indicator.

Two redundant Class IE channels of instrumentation are provided to monitor containment flood level (wide range). The wide range monitors meet the requirements of Appendix B to NUREG-0737. Each channel of instrumentation consists of the following:

a) A level transmitter with an approximate range of 0 - 16' located inside the containment. The measurement covers the range from the floor level at elevation -14.8 feet to elevation + 1.4 feet. The maximum flood level established by calculation is within this range.

b) A process analog control system (PAC) to monitor level transmittal signal and to develop output signals to the plant monitoring computer and recorder or indicator.

c) A recorder/indicator, mounted on the main control board, for registering the containment flood level. One channel is provided with a recorder and one channel is provided with an indicator.

→ (DRN 99-1035)

← (DRN 99-1035)

1.9.30 INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (II.F.2)

Position

Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

Response

See Appendix 1.9A.

1.9.31 EMERGENCY POWER FOR PRESSURIZER EQUIPMENT (II.G.1)

Position

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17, and 20 of Appendix A to 10CFR Part 50 for the event of loss-of-offsite power, the following positions shall be implemented.

Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

(1) Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

→(DRN 01-758)

(2) Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

←(DRN 01-758)

(3) Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

(4) The pressurizer level indication instrument channels shall be powered from the vital instrument buses. The buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

Response

(1) Waterford 3 has spring loaded pressurizer safety valves; therefore, this position requirement is not applicable.

(2) Waterford 3 has no pressurizer block valves; therefore, this position requirement is not applicable.

(3) See (1) and (2) above.

(4) Waterford 3 pressurizer level indication instrument channels are powered from the Class IE uninterruptible 120V ac power system. This system can be supplied from either the offsite power source or the standby power supply (Emergency Diesel Generators). (See Table 1.9-1).

For CWDs and Electrical Schematics, see FSAR Table 1.7-1.

1.9.32 IE BULLETINS ON MEASURES TO MITIGATE SMALL-BREAK LOCAs AND LOSS OF FEEDWATER ACCIDENTS (II.K.1)

Position

Review all valve positions, positioning requirements, positive controls and related test and maintenance procedures to assure proper ESF functioning. (C.1.5)

Response

A review of all ESF valve positions, controls and test and maintenance procedure is conducted during procedure preparation to ensure proper ESF functioning. This review assures normal lineup in the mode required for ESF operation. Test and maintenance procedures are reviewed so as to be "stand alone" procedures with independent verification for valve positioning and restoration.

Position

Review and modify, as required, procedures for removing safety-related systems from service (and restoring to service) to assure operability status is known. (C.1.10)

Response

Procedures dealing with all safety related systems and components require independent verification of all valve operations and breaker positions both for operational safeguards readiness and for realignment for the performance of maintenance or tests. At the completion of maintenance or tests, the same independent verification is performed for the restoration portion of the procedure to assure the operability status is known. Periodic tests and checks are performed to verify continued operability status.

1.9.33 ORDERS ON B&W PLANTS (II.K.2)

THERMAL MECHANICAL REPORT--EFFECT OF HIGH-PRESSURE INJECTION ON VESSEL INTEGRITY FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITH NO AUXILIARY FEEDWATER (II.K.2.13)

Position

A detailed analysis shall be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.

Response

This analysis has been performed by the CE Owners Group (CEOG) and is discussed in CEN-189, submitted to the NRC by the CEOG on 12/31/81. Appendix I of this report specifically addresses Waterford 3.

POTENTIAL FOR VOIDING IN THE REACTOR COOLANT SYSTEM DURING TRANSIENTS (II.K.2.17)

Position

Analyze the potential for voiding the reactor coolant system (RCS) during anticipated transients.

Response

Louisiana Power & Light Company sponsored the CE Owners Group evaluation of this item. The evaluation results are contained in CEN-199, "Effects of Vessel Head Voiding During Transients and Accidents in CE NSSS's".

It was found that voiding in the reactor vessel upper head region is not expected to occur for normal operational transients. For natural circulation cooldown transients voiding may occur in the reactor vessel upper head region. However, in the event that voids are formed, the operator guidance provided in CE's emergency procedure guidelines adequately address how to control and reduce the voids. These guidelines form the basis for the emergency operating procedures at Waterford 3.

Additionally, for FSAR Chapter 15 transients the impact of voiding will not result in violation of the Standard Review Plan requirements. Finally, the report concludes that any potential void formation during the plant transients addressed is not great enough to impair reactor coolant circulation or core coolability.

SEQUENTIAL AUXILIARY FEEDWATER FLOW ANALYSIS (II.K.2.19)

Position

Provide a benchmark analysis of Sequential Auxiliary Feedwater (AFW) flow to the steam generators following a loss of main feedwater.

Response

The concerns expressed in this item, and as clarified in NUREG-0737, are not considered applicable to Waterford 3 which utilizes vertical U-tube steam generators as designed by Combustion Engineering.

1.9.34 FINAL RECOMMENDATIONS OF B&O TASK FORCE (II.K.3)

INSTALLATION AND TESTING OF AUTOMATIC POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM (II.K.3.1)

Position

All PWR licensees should provide a system that uses the PORV block valve to protect against a small-break loss-of-coolant accident. This system will automatically cause the block valve to close when the reactor coolant system pressure decays after the PORV has opened. Justification should be provided to assure that failure of this system would not decrease overall safety by aggravating plant transients and accidents.

Each licensee shall perform a confirmatory test of the automatic block valve closure system following installation.

Response

The requirements of this position are not applicable to Waterford 3. Waterford 3 has no pressurizer block valves.

REPORT ON OVERALL SAFETY EFFECT OF POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM (II.K.3.2)

Position

(1) The licensee should submit a report for staff review documenting the various actions taken to decrease the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV) and show how those actions constitute sufficient improvements in reactor safety.

(2) Safety-valve failure rates based on past history of the operating plants designed by the specific nuclear steam supply system (NSSS) vendor should be included in the report submitted in response to (1) above.

Response

Waterford 3 has spring loaded pressurizer safety valves; therefore, the requirements of this position are not applicable to Waterford 3.

REPORTING OF SV AND RV FAILURES AND CHALLENGES (II.K.3.3)

Position

Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report.

Response

In the unlikely event of any failure of the spring loaded pressurizer safety valves to close, LP&L will report such failures to the NRC promptly. All challenges to the safety valves will be documented in the monthly report.

AUTOMATIC TRIP OF REACTOR COOLANT PUMPS DURING LOSS-OF-COOLANT ACCIDENT (II.K.3.5)

Position

Tripping of the reactor coolant pumps in case of a loss-of-coolant accident (LOCA) is not an ideal solution. Licensees should consider other solutions to the small-break LOCA problem (for example, an increase in safety injection flow rate). In the meantime, until a better solution is found, the reactor coolant pumps should be tripped automatically in case of a small-break LOCA. The signals designated to initiate the pump trip are discussed in NUREG-0623.

Response

LP&L has been a charter participant in the CE Owners Group development of CEN-152, CE Emergency Procedure Guidelines (EPGs). The EPGs form the basis, as approved by the NRC, for the Waterford 3 emergency operating procedures (EOPs). For Cycle 2 operation, LP&L implemented Revision 2 to CEN-152.

EVALUATION OF POWER-OPERATED RELIEF VALVE OPENING PROBABILITY DURING OVERPRESSURE TRANSIENT (II.K.3.7)

Position

Most overpressure transients should not result in the opening of the power-operated relief valve (PORV). Therefore, licensees should document that the PORV will open in less than 5 percent of all anticipated overpressure transients using the revised setpoints and anticipatory trips for the range of plant conditions which might occur during a fuel cycle.

Response

Waterford 3 has spring loaded pressurizer safety valves; therefore, the requirements of this position are not applicable to Waterford 3.

→(DRN 01-758)

REPORT ON OUTAGES OF EMERGENCY CORE-COOLING SYSTEMS LICENSEE REPORT AND PROPOSED TECHNICAL SPECIFICATION CHANGES (II.K.3.17)

←(DRN 01-758)

Position

Several components of the emergency core-cooling (ECC) systems are permitted by technical specifications to have substantial outage times (e.g., 72 hours for one diesel-generator; 14 days for the HPCI system). In addition, there are no cumulative outage time limitations for ECC systems. Licensees should submit a report detailing outage dates and lengths of outages for all ECC systems for the last five years of operation. The report should also include the causes of the outages (i.e., controller failure, spurious isolation).

Response

A formal reliability/availability program has been established for Waterford 3. The information will contain: 1) outage dates and duration of outages; 2) cause of the outage; 3) ECCS systems or components involved in the outage; and 4) corrective action taken. Requirements for collecting and analyzing data specific to the ECCS as well as many other plant systems is incorporated in the procedures that define the reliability/availability program. Data collection and analysis is integrated and interfaced with that required for LER and NPRDS reporting. The methodology is designed to provide a means for quick retrieval of ECCS information.

→(DRN 01-758)

By letter dated May 5, 1989, the NRC informed LP&L that Item II.K.3.17 of NUREG-0737 was an early action item to allow the NRC to quickly evaluate existing requirements and prepare followup actions. The requirements of 10CFR50.72 and industry efforts to report on the Equipment Performance Information and Exchange System (EPIX) are adequate for reporting ECCS outages. A special report from Waterford 3 is not required. Item II.K.3.17 is closed for Waterford 3.

←(DRN 01-758)

EFFECT OF LOSS OF ALTERNATING-CURRENT POWER ON PUMP SEALS (II.K.3.25)

Position

The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating-current (ac) power for at least two hours. Adequacy of the seal design should be demonstrated.

Response

Waterford 3 complies with this position by supplying emergency power (Diesel Generator automatic loading) to the component cooling water pump. Although the component cooling water lines to the reactor recirculation pump seal coolers are isolated on a CSAS (See FSAR Table 6.2-32), these lines and isolation valves are provided with a manual override.

→(DRN 01-758)

REVISED SMALL-BREAK LOSS-OF-COOLANT-ACCIDENT METHODS TO SHOW COMPLIANCE WITH 10CFR PART 50, APPENDIX K (II.K.3.30)

←(DRN 01-758)

Position

The analysis methods used by nuclear steam supply system (NSSS) vendors and/or fuel suppliers for small-break loss-of-coolant accident (LOCA) analysis for compliance with Appendix K to 10CFR Part 50 should be revised, documented, and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT Test and Semiscale Test facilities.

Response

LP&L has participated in a series of CE Owner's Group tasks in support of Item II.K.3.30.

In the summer of 1979, CE submitted two reports to the NRC, CEN-114-P, "Review of Small Break Transients in Combustion Engineering Nuclear Steam Supply Systems," July 1979 (Proprietary), and CEN-115-P, "Response to NRC IE Bulletin 79-06C, Items 2 and 3 for CE Nuclear Steam Supply Systems," August 1979 (Proprietary), which describe CE's Small Break LOCA Evaluation Model. These submittals were prepared in response to NRC requests following the TMI-2 accident. After review of these documents, the NRC identified a number of questions with some portions of the small break model. The NRC requested a response to these questions in the NRC TMI Action Plan, NUREG-0737, Item II-K-3.30. At a meeting held on January 26, 1981 with members of the NRC staff and representatives of the CE Owners Group and CE, the NRC staff described seven technical items which form the basis for the seven specific questions relative to the CE Small Break LOCA Evaluation Model. The NRC staff also indicated that responding to these seven questions would fulfill the response to Item II.K.3.30 of the NRC TMI Action Plan.

The seven questions were responded to in CEN-203-P Revision I-P, "Response to NRC Action Plan Item II.K.3.30 Justification of Small Break LOCA Methods," transmitted to the NRC by CE on April 15, 1982. The response shows that using the CE Small Break LOCA Evaluation Model results in-conservatively high cladding temperatures, and complies with Waterford 3's requirement under this item.

By letter dated July 12, 1985 the NRC found LP&L in compliance with the requirements of Item II.K.3.30 and that a plant specific analysis (Item II.K.3.31) was not required.

PLANT-SPECIFIC CALCULATIONS TO SHOW COMPLIANCE WITH 10CFR PART 50.46, (II.K.3.31)

Position

Plant-specific calculations using NRC-approved models for small-break loss-of-coolant accidents (LOCAS) to show compliance with 10CFR50.46 should be submitted for NRC approval by all licensees.

Response

See Response to Item II.K.3.30 above.

1.9.35 EMERGENCY PREPAREDNESS-SHORT TERM (III.A.1.1)

Position

Comply with Appendix E, "Emergency Facilities," to 10CFR Part 50, Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants," and for the offsite plans, meet essential elements of NUREG-75/111 or have a favorable finding from FEMA.

Provide an emergency response plan in substantial compliance with NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" except that only a description of and completion schedule for the means for providing prompt notification to the population, the staffing for emergencies in addition to that already required, and an upgraded meteorological program need be provided. NRC will give substantial weight findings on offsite plans in judging the adequacy against NUREG-0654. Perform an emergency response exercise to test the integrated capability and a major portion of the basic elements existing within emergency preparedness plans and organizations.

Response

The Waterford 3 Emergency Plan was updated using the criteria provided in NUREG-0654 with special attention to the establishment of the emergency action levels in accordance with NUREG-0610. Refer to FSAR Section 13.3.

The Technical Support Center, the Operational Support Center, and the near site Emergency Operations Facility have been established. Refer to FSAR Section 13.3.

Improved offsite radiological monitoring capability in accordance with NRR/RAB technical position has been developed.

Louisiana Power & Light coordinated with the State and local government in developing Radiological Emergency Response Plans for Waterford 3. Therefore, the State, local and facility plans are well coordinated.

Periodic exercises are conducted with Federal, State, and local government to evaluate major portions of their emergency response capability and to correct identified deficiencies. Refer to FSAR Section 13.3.

1.9.36 UPGRADE EMERGENCY SUPPORT FACILITIES (III.A.1.2)

Position

Each operating nuclear power plant shall maintain an onsite technical support center (TSC) separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the TSC. Records that pertain to the as-built conditions and layout of structures, systems, and components shall be readily available to personnel in the TSC.

An operational support center (OSC) shall be established separate from the control room and other emergency response facilities as a place where operations support personnel can assemble and report in an emergency situation to receive instructions from the operating staff. Communications shall be provided between the OSC, TSC, EOF, and control room.

An emergency operations facility (EOF) will be operated by the licensee for continued evaluation and coordination of all licensee activities related to an emergency having or potentially having environmental consequences.

Response

The Technical Support Center, Operational Support Center, and the near site Emergency Operations Facility have been established. Refer to FSAR Section 13.3.

1.9.36a IMPROVING LICENSEE EMERGENCY PREPAREDNESS-LONG-TERM (III.A.2)

Position

Each nuclear facility shall upgrade its emergency plans to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Specific criteria to meet this requirement is delineated in NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparation in Support of Nuclear Power Plants."

Response

The Emergency Plan was upgraded in accordance with the applicable criteria of NUREG-0654. Refer to FSAR Section 13.3.

1.9.37 INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIAL FOR PRESSURIZED-WATER REACTORS AND BOILING-WATER REACTORS (III.D.1.1)

Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate leak reduction

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

(b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Response

LP&L has instituted a program to maintain leakage rates of systems outside containment which could contain radioactivity to as low as practical. To support this program, a review of plant systems has identified the systems outside containment which could potentially contain highly radioactive fluids following a serious accident.

A. Systems included in the leak reduction program:

1) Containment Spray System - that portion of the system located outside containment that would be in use in the recirculation mode of operation including that suction piping from the Safety Injection Sump up through the pumps and heat exchangers to the containment isolation valve.

2) Low Pressure Safety injection - the piping outside of containment in use during operation in the shutdown cooling mode.

3) High Pressure Safety Injection - the piping from the recirculation suction header through the pump up to containment.

4) Hydrogen Analyzer System - that portion of piping from the outside containment isolation valve to the Hydrogen Analyzer Panels, along with the piping to the containment atmospheric grab sampler in the post-accident sampling area and return piping back to the containment isolation valve.

→(DRN 01-758)

5) Post-accident Sampling System (PASS) - for the liquid portion of the system, testing includes the piping from the connection to the Primary Sample System sample point 5A and 5B sample lines to the PASS skid packages back to the outside containment isolation valve. Also included is the piping from the RCS Hot leg Sample outside containment isolation valve to the PASS skid package and back to the containment isolation valve. For the gas portion of the system, also included is that portion of the tubing from the liquid/gas separator through the skid package back to the outside containment isolation valve.

←(DRN 01-758)

6) Containment Vacuum Relief (CVR) - the essential instrument tubing from outside containment to the differential pressure instruments.

→(DRN 01-758)

7) Primary Sampling System - that portion of primary sample point 5A and 5B sampling lines from the safety injection recirculation lines to their connection to the Post Accident Sampling System line.

←(DRN 01-758)

B. Systems excluded from the program (their isolation will not preclude any option of cooling the reactor core nor prevent the use of needed safety systems):

1) The Gaseous Waste Management System. This system isolates on CIAS and is not required for use post-accident. The Reactor Coolant Vent System provides RCS venting as discussed in Subsection 1.9.18.

2) The Chemical Volume and Control System (Charging and Letdown). On a CIAS or SIAS this system will isolate letdown flow which is not required after an accident nor is it needed to bring the reactor to a safe shutdown condition. In addition to letdown being isolated on an SIAS, the Volume Control Tank (VCT) outlet valve closes resulting in the charging pumps taking suction directly from the Boric Acid Makeup Tanks, or the charging pumps can be lined up to the Refueling Water Storage Pool (RWSP). Thus, no highly radioactive fluids are expected to flow through the portion of the CVCS outside of containment.

3) Reactor Coolant Pump Seal Bleed-off to the VCT. This system is isolated on a CIAS. If seal bleed-off is needed post-accident, the pressure in that portion of the system inside containment will increase and the header relief valve will open thus providing a flow path to the Quench Tank.

4) The Boron Management System. This system receives a CIAS which isolates the Reactor Drain Tank outlet, thus, when the tank is pressurized it relieves to the containment sump.

5) The Primary Sampling System. This system with the exception of the portions of sample point 5A and 5B sampling lines discussed in A. 6) above, isolates on a CIAS and would not be required because of the availability of the Post-Accident Sampling System.

6) The Shield Building Ventilation System. That portion of the system from the annulus through the filters and up to the fan is operated at a negative pressure. So, any leakage would be in the inward direction and not outward from the system. System leakage downstream of the fan is of no radiological significance since the SBVS filter exhaust is suitable for discharge to the atmosphere.

→(DRN 04-1619, R14)

7) The Controlled Ventilation Area System. Similar to the Shield Building Ventilation System, that portion of the system up to the fan is operated at a negative pressure, and the discharge of the fan is of no radiological significance since the CVAS filter exhaust is suitable for discharge of the environment.

←(DRN 04-1619, R14)

→(DRN 05-1265, R14-A)

C. For liquid systems, leakage detection is performed by visual inspection of all potential leak sources (e.g., valves, pump seals, etc.). Upon detection of a leak, the leak rate is determined. For gas systems, leakage detection is performed by pressurizing the system with an inert gas, nitrogen, or instrument air and visually inspecting potential leak sources with a soapy water solution (or equivalent method). Those leakage sources whose leak rates cannot be reduced to as low as practical, will be reported to the Plant Manager or his designee for resolution. Initial leakage rates were determined during plant startup testing prior to initial criticality and reported to the NRC in W3P85-0538 dated March 4, 1985. Future leak rate measurements will be performed at intervals not to exceed each refueling outage. Records of leakage rates and their sources will be retained in plant files.

←(DRN 05-1265, R14-A)

D. The potential release path identified in the NRC letter dated October 17, 1979 (Radioactive Release at North Anna Unit 1 and Lessons Learned) is not credible in the Waterford 3 design. High level in the volume control tank is alarmed in the main control room and automatically causes influent flow to be diverted to the Boron Management System. The overall program for prevention of unplanned radioactivity releases will incorporate the features of IE Circular 79-21. Aspects of the program and related features of the Waterford 3 design are:

1) All tanks outside of containment are provided with level indicators and high

level alarms to alert the operator of high level conditions, and loop seals on overflow lines to prevent the escape of radioactive gas. Generally, collection tanks and tanks which receive processed waste are provided with backup tanks. Tanks outside of containment which are not provided with a backup tank are:

- Primary Water Storage Tank (PWST) - The PWST is located outside the nuclear plant island and as such, the Technical Specifications place a strict limit on the amount of activity allowed in the tank. Therefore, a spill from the tank would not involve a significant amount of radioactivity.

→(DRN 00-803)

- Equipment Drain Tank (EDT) - The EDT does not have an overflow and thus there is no potential for spillage. In the event of high level, the EDT pump starts and pumps to the Holdup Tanks.

←(DRN 00-803)

- Spent Resin Tank (SRT) - High level in the SRT automatically causes the inlet valve to close. The SRT is vented to the vent gas collection header.

→(DRN 01-758)

- Condensate Storage Tank - Overflow from this tank is to the floor of its enclosed concrete area, which is provided with six inch curbs to limit the spread of liquid.

←(DRN 01-758)

2) Storm drains are located away from areas with a high potential for radioactive spills and there are no cross-connects between the floor and storm drainage systems.

3) Radioactive pumps are generally located in isolated compartments whose drains are designed to catch all potential leakage.

These drains are routed through the radioactive drainage systems to the waste management systems. Pumps whose potential for radioactive leakage is greatest are equipped with drip pans and lines piped to the floor drains. Discussion of the Equipment Drain System can be found in FSAR Subsection 9.3.3.

4) Cubicles where the potential for liquid leakage exists are generally provided with floor drains and/or equipment drains. Areas where flooding could be expected to cause a safety problem are provided with watertight doors.

The Waterford 3 Leak Reduction procedure addresses: (1) performance of inspections to verify integrity of systems that could cause an inadvertent release, and (2) implementation of a preventive maintenance program to promptly repair identified problems, such as leaking equipment and plugged floor drains.

Underground piping will be pressure tested as required by ASME Section XI or other regulatory requirements. New permanent piping systems will be pressure tested prior to first use in accordance with ASME.

Section XI or other applicable regulatory requirements. All temporary piping associated with vendor solidification equipment is hydrostatically tested by the vendor prior to shipment and installation in accordance with their QA procedures.

1.9.38 IMPROVED IMPLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS (III.D.3.3)

Position

(1) Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

(2) Each applicant for a fuel-loading license to be issued prior-to January 1, 1981 shall provide the equipment, training, and procedures necessary to accurately determine the presence of airborne radioiodine in areas within the plant where plant personnel may be present during an accident.

Response

Surveillance of inplant airborne Iodine concentration for the Waterford 3 SES is provided for via three different methods. The specific methods are:

1. In Plant Airborne Radiation Monitoring System (PARMS)

→(DRN 01-758)

2. Portable Continuous Air Monitors (PCAMS)

←(DRN 01-758)

3. Portable (hand-held) High Volume Air Samplers used in conjunction with the existing GE (Li) detector system which is located in the counting room

Method #1 is applicable for use only in the RAB whereas items 2 and 3 can be used by Health Physics Personnel anywhere in the plant where access may be required. An approximate indication of Fuel Handling Building airborne activities is provided for by two FHB normal exhaust monitors and four FHB isolation monitors described in FSAR Subsection 12.3.4.2.1.

→(DRN 02-406)

Initial indication of potentially high airborne iodine concentration is provided for via stationary In Plant airborne Particulate Iodine and Gas Monitors. Four of these stationary monitors draw isokinetic samples of air from RAB ductwork. The sample points for three of the four RAB monitors were picked such that common exhaust ducts are sampled which are collecting exhausts from various rooms in the RAB in which occupancy is periodic as defined in FSAR Section 12.3A.

←(DRN 02-406)

The fourth RAB In Plant Airborne Radiation Monitoring (IPSRM) draws a sample from the exhaust plenum of the RAB ventilation system and provides for overall monitoring of Particulate Iodine and Gas airborne concentrations in the RAB.

→ (DRN 00-1053; 02-406)

The remaining stationary In Plant Airborne Particulate Iodine and gas monitors provide for room specific particulate iodine and gas concentration readings. The specific areas are in the Hot Machine shop and Decontamination Area.

← (DRN 00-1053; 02-406)

The iodine detector portion for all of the stationary monitors consists of a charcoal cartridge assembly and lead plug on the front end of the shield and a NaI (Tl) detector assembly and lead plug on the opposite end. The sample enters the shield, passes through the replaceable charcoal cartridge, and then exits the shield. The charcoal cartridge absorbs iodine and can be manually changed (in less than one minute). In operation, the cartridge is viewed with the NaI (Tl) integral line gamma scintillation detector. A single-channel analyzer (SCA) in the signal processor monitors an adjustable (set at 10 percent) window around the 364.KeV I^{131} photo peak.

During accident conditions Noble gases are also adsorbed on the charcoal and can "swamp" the iodine detector due to the poor resolution of NaI and because no background subtraction is provided, use of the above detectors results in overly conservative estimates of the airborne iodine concentrations when large amounts of noble gases are present. In order to minimize unnecessary usage of respirators, by plant personnel and in order to assist Health Physics personnel in quickly localizing the area from which high airborne, iodine concentration is arising, additional portable hand held high volume air samplers utilizing silver zeolite filters are available for use under accident conditions. Silver zeolite will retain iodine as well as, if not better than, charcoal but has low retention of Noble gases.

After securing a sample from the area of interest through the use of the high volume samplers Health Physics personnel would then take the sampler to the counting room where the Ge (Li) spectroscopy system is located for detailed spectroscopic analysis. A clean air source will be available for purging the silver zeolite cartridges of noble gases prior to counting. Appropriate precautions will be taken to prevent the spread of contamination during purging. As stated in FSAR Subsection 12.3A.3.3 post accident sampling shall not be performed from the existing sampling panel. As a result, the counting room background activity will be 2.6 mr/m 20 minutes after the accident and less than 1 mr/hr after six hours. At this level of background activity the existing Ge (Li) spectroscopy system is usable for accurately analyzing the iodine content of the sample filter. Utilizing the information obtained from the spectroscopic analysis of the sampler filter the Health Physics personnel at the Waterford 3 SES can accurately determine whether the use of respirators is required by plant personnel who would be entering the area from which the sample was taken.

The gamma-spectrometer consists of a computer based multichannel analyzer (MCA) which is used in conjunction with Ge (Li) detectors and with which it is fully compatible, and also consists of a preamplifier/amplifier for signal conditioning. A computer is utilized for data processing and storage.

The Ge (Li) detectors are the closed-end-coaxial type used for gamma spectroscopy for radiochemistry and health physics, and are mounted inside a chamber shielded with six inches of lead to minimize the effects of background on sample analysis. Software programs are available in the spectroscopy system which can assist the Health Physics Personnel in accurately determining iodine concentrations in the sample filters. Specifically the programs are: spectrum smoothing, peak search, and nuclide identification.

The Ge (Li) detectors have the following characteristics which assure accurate readings:

a) Relative photopeak efficiency for 1.33 MeV photons: 15 percent for one detector and - 20 percent for the second detector.

b) Peak to compton ratio - 40:1

c) Resolution (@ FWHM of Co^{60} 1.33 peak MeV) £ 2.0 keV

d) Full width at one tenth maximum (FWTM), is less than or equal to double the full width at half maximum (FWHM), i.e., FWTM £ 2 FWHM.

Once occupancy constraints have been established by Health Physics personnel through the use of portable high volume air samplers and the GE (Li) Gamma Scintillation detectors, Portable continuous airborne activity monitors would be utilized by personnel occupying the area. The function of the aforementioned Portable Continuous Air Monitors (PCAM) would be to monitor habitability conditions of the occupied area.

If high concentrations of airborne activities are detected the PCAMs shall alarm in the area. Personnel on hearing the alarm will evacuate the area. Health Physicists will then have to reestablish occupancy constraints through the use of the portable high-volume air samplers and the Ge(Li) spectroscopy system.

In summary monitoring of overall RAB radioactive particulate Iodine and gas concentratures is provided for via the IPARMS. Using these monitors the presence of potentially unacceptability high airborne activity levels can be detected in the RAB.

If the existence of a potential problem area is shown by the IPARMS, portable high volume samplers shall be used by Health Physics personnel to establish initial occupancy constraints for personnel who shall be working in a given area. Analysis of the samples taken via the portable high volume samplers shall be performed by the existing Ge (Li) spectoscopy system located in the counting room.

Finally, if plant personnel shall have to work, in an area that was previously cleared by Health Physics personnel for occupancy, for any length of time the PCAM's shall be placed into the work area to continuously monitor airborne activity levels. If unusually high activity levels were to be detected, the personnel occupying the area would be warned by the PCAM alarms.

Portable high volume air samplers and filter cartridges, and clean air purge capability meet the requirements of NUREG-0737.

The presence of airborne noble gases in the vicinity of the gamma spectroscopy equipment should not normally interfere with its ability to analyze the iodine content of the sample filter. However, an alternate gamma spectroscopy system will be available onsite and will be located in a more habitable counting room.

1.9.39 CONTROL-ROOM HABITABILITY REQUIREMENTS (III.D.3.4)

Position

Licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safety operated or shut down under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10CFR Part 50).

Response

Waterford 3 has met the requirements of SRP 6.4, 2.2.1, 2.2.2, and 2.2.3 in previous submittals to the NRC. A control room habitability evaluation was performed in accordance with the requirements and clarifications of this position, and FSAR Table 1.9-2 identifies specific information requested or references FSAR sections where information has previously been submitted.

→(DRN 04-1619, R14)
The Waterford 3 control room is designed to be habitable during both toxic gas and radiological emergencies. The control room design meets the guidance of Regulatory Guide 1.78 (June 1974) and 1.95 (February 1975) as well as complies with GDC19 of Appendix A to 10CFR50 and 10CFR50.67. Control room habitability systems and equipment is discussed in Section 6.4 of the FSAR, while the toxic gas analysis is contained in Subsection 2.2.3.
←(DRN 04-1619, R14)

TMI-RELATED REQUIREMETNS FOR NEW OPERATING LICENSES

No.	Title	Description	FSAR Sections	References
I.A.1.1	Shift Technical Advisor Responsibilities	1. On Shift 2. Training 3. Describe long term program	Subsection 13.1.2.2.2	(1) item 2.2.lb, (2), (3), (4)
→(LBDCR 13-015, R308)				
I.A.1.2	Shift Manager Responsibilities	Delegate Non-Safety Duties	Subsection 1.9.2	(1) item 2.2.1a, (2), (3) (5)
←(LBDCR 13-015, R308)				
I.A.1.3	Shift Manning	1. Limit overtime 2. Minimum Shift Crew	1. Subsection 1.9.3 2. Table 16.6.2-1, & Table 6.2-1	
I.A.2.1	Immed. Upgrade of RO & SRO Training and Qualifications	1. SRO Experience 2. SROs be ROs 1 year 3. Three Mos. tng. on-shift 4. Modify Training 5. Facility Certification	Subsections 13.2.1, 13.2.2	(4), (6), (7)
I.A.2.3	Administration of Training Programs	Instructors complete SRO exam	Subsections 13.2.1, 13.2.2, 13.2.4, 13.3.7	(4), (6)
I.A.3.1	Revise Scope and Criteria for Lic. Exams	1. Increase Scope 2. Increase Passing Grade 3. Simulator Exams a) Plants with Simulators b) All Plants	Subsection 13.2.2	(4), (6), (7)
I.B.1.2	Evaluation of Organization	Organization, resources tng. and qualifications for operators and accidents	Subsection 1.9-7, Chapter 13	(7)
I.C.1	Short-term Accident and Procedure Review	1. SB LOCA 2. Inadequate Core Cooling a) Reanalyze and Propose Guidelines b) Revise Procedures 3. Transients & Accidents a) Reanalyze and Propose Guidelines b) Revise Procedures	Subsections 1.9.8, 6.3 13.2.1, 15.6.3	(1) item 2.1.3b 2.1.9, (2) (3), (4)
I.C.2	Shift and Relief Turnover Procedures	Revise procedures to assure plant status known by new shift	Subsection 1.9.9	(1) item 2.2.1c, (2), (3)
I.C.3	Shift Supervisor	Corporate directive to establish command duties and revise plant procedures	Subsection 1.9-10	(1) item 2.2.1a, (2), (3)

TABLE 1.9-1 (Sheet 2 of 5)

TMI-RELATED REQUIREMETNS FOR NEW OPERATING LICENSES

No.	Title	Description	FSAR Sections	References
I.C.4	Control Room Access	Establish authority and limit access	Subsection 1.9.11	(1) item 2.2.2a, (2), (3)
I.C.5	Feedback of Operating Experience	Review and Revise Procedures	Subsection 1.9.12	(4),(7)
I.C.6	Verify Correct Performance of Oper. Activites	Revise Performance Procedures	Subsection 1.9.13	(4)
I.C.7	NSSS Vendor Review of Procedures	1. Low Power Test Program 2. Power Ascension Procedures 3. Emergency Procedures	Subsections 1.9.14, 13.5.1.2, 14.2.2.5, 14.2.3	(4),(7)
I.D.1	Control Room Design Rev.	Preliminary Assessment and schedule for correcting deficiencies	Subsection 1.9.15	(7)
I.D.2	Plant-Safety- Parameter Display Console	1. Description 2. Installed 3. Fully Implemented	Subsection 1.9.16	
I.G.1	Training During Low Power Testing	1. Propose Tests 2. Submit Analysis and Procedures 3. Training Results	Section 14.2, & Subsection 13.2.1.2	
II.B.1	Reactor Coolant System Vents	1. Design Analysis 2. Install 3. Procedures	Subsection 5.4.15	(2), (3), (4)
II.B.2	Plant Sheilding	1. Radiation and Shielding Review 2. Corrective Actions to Assure Access 3. Complete Mods 4. Equipment Qualifications	Appendix 12.3A Subsection 9.3.8	(1) item 2.1.6.6, (2), (3), (4)
II.B.3	Post Accident Sampling	1. Design Review 2. Corrective Actions 3. Procedures 4. Complete Actions		(1) item 2.1.8a, (2),(3),(4)
II.B.4	Training for Mitigating Core Damage	1. Develop Training Program 2. Complete Training	Subsections 13.2.1, 13.2.2	(6),(7)
II.D.1	Relief and Safety Valve Test Requirements	1. Describe Program and Schedule	1. Subsection 1.9.22	(1) item 2.1.2,

TABLE 1.9-1 (Sheet 3 of 5)

TMI-RELATED REQUIREMETNS FOR NEW OPERATING LICENSES

No.	Title	Description	FSAR Sections	References
II.D.1	(Cont'd)	2. RV and SV Tests 3. Block Valve Tests	2. Subsection 1.9-22 3. (NA)	
II.D.3	Valve Position Indication	Install In Control Room	1.9.23	(1) item 2.1.3a (2),(3),(4)
II.E.1.1	AFV System Evaluation	1. Analysis 2. Modification	Appendix 10.4.9 A & 9B	(8)
II.E.1.2	AFW System Initiation and Flow	1. Initiation (a) Control grade (b) Safety grade 2. Flow Indication (a) Control grade (b) Safety grade	Subsection 7.3.1.1.6, Table 7.5-1, 10.4.9, Table 1.7-1	(1) item 2.1.7a & (2),(3)
II.E.3.1	Emergency Power for Pressurizer Heater	Installed Capability	Subsections 1.9.26 and 5.4.1.0.2, Figure 8.3-33	(1) item 2.1.1 (2),(3),(4)
II.E.4.1	Dedicated Hydrogen Penetrations	1. Design 2. Review and Revise H_2 Control Proc 3. Install	1. (NA) 2. Subsections 1.9.27, 6.2.5 3. (NA)	(1) item 2.1.5a 2.1.5c, (2),(3),(4)
II.E.4.2	Containment Isolation Dependability	1-4.Implement diverse isolation 5. Cont. pressure setpoint 6. Cont. purge valve 7. Radiation signal on purge valve	1) Subsection 1.9.28 Table 1.9-3, Table 6.2-32	(1) item 2.1.4, (2),(3),(4)
II.F.1	Accident Monitoring Instrumentation	1. Procedures 2. Install Instrumentation (a) Noble gas monitor (b) Iodine/particulate sampling (c) Containment high range monitor (d) Containment pressure (e) Containment water level (f) Containment hydrogen	1. Subsection 1.9.2.9 2. (a), (b), (c), (d), (e) Subsection 1.9.29 (f) Subsection 6.2.5.1	(1) item 2.1.8b, (2),(3),(4)

TABLE 1.9-1 (Sheet 4 of 5)

TMI-RELATED REQUIREMETNS FOR NEW OPERATING LICENSES

No.	Title	Description	FSAR Sections	References
II.F.2	Instrumentation for detection of inadequate core-cooling	1. Procedures 2. Subcooling meter 3. Describe other instrumentation 4. Install add'l instrumentation	Subsection 1.9.30	(1) item 2.1.sb, (2),(3),(4)
II.G.1	Power supplies for pressurizer relief valves,	Power supply from emergency buses	Table 1.7-1 Subsections 1.9.31 8.3.1.1.1c, 7.7.1.2.2	(1) item 2.1.1, (2), (3), (4)
II.K.1	IE Bulletins	5. Review ESF Valves 10. Operability Status	5. Subsection 1.9.32 10. Subsection 1.9.32	(7)
II.K.2	Orders on B & W Plants	13. Thermal-mechanical Report 17. Voiding in RCS 19. Benchmark analysis seq AFW flow	13. Subsection 1.9.33 17. Subsection 1.9.33 19. Subsection 1.9.33	
II.K.3	Final Recomendations, B & O Task Force	1. Auto PORV isolation 2. Report PORV failures 3. Reporting SV and RV failures & challenges 5. Auto trip of RCPs a) propose mods b) modify 7. Evaluation of PORV opening probability 17. ECCS outages 25. Power on pump seals a) propose mods b) mods	1. (NA) 2. (NA) 3. Subsection 1.9.34 5. Subsection 1.9.34 17. Subsection 1.9.34 25. Subsection 1.9.34	(4),(7),(8)
II.K.3	Cont'd	30. SB LOCA Methods a) schedule outline b) model c) new analysis 31. Plant Specific Analysis	30. Subsection 1.9.34 31. Subsection 1.9.34	
III.A.1.1	Emergency Preparedness, Short term	1. Comply with 1OCFR5O, APP. E 2. Comply with NUREG-0654 3. Conduct Exercise 4. Meteorotogical Data	Subsection 1.9.35 and Section 13.3	(7)

TABLE 1.9-1 (Sheet 5 of 5)

TMI-RELATED REQUIREMETNS FOR NEW OPERATING LICENSES

No.	Title	Description	FSAR Sections	References
III.A.1.2	Upgrade Emergency Support Facilities	1. Establish (Interim Basis) (a) TSC (b) OSC (c) EOF 2. Design 3. Modifications	Subsections 1.9.36, 13.3.6	(1) item 2.2.2b, 2.2.c, (2), (3)
III.A.2	Emergency Preparedness Long Term Long Term	1. Upgrade Emergency Plan to APP. E, 10CFR50 2. Meterorological Data	Subsection 1.9.36a & Section 13.3	
III.D.1.1	Primary Coolant Outside Containment	Measure leak rates and establish program to keep leakage ALARA	Subsections 1.9.37, 6.2.2.4.2, 6.3.4.3, 9.3.4.3.4, 9.3.6.4	(1) item 2.1.6a, (2),(3),(4)
III.D.3.3	Inplant 1, Radiation Monitoring	1. Provide means to determine presence of radio-iodine 2. Modifications to accurately measure radio-iodine	Subsection 1.9.38	(1) item 2.1.8c, (2),(3),(4)
III.D.3.4	Control Room Habitability	1. Identify and evaluate potential hazards 2. Schedule for Modifications 3. Modifications	Subsection 1.9.39, Table 1.9-2	

References To Table 1.9-1

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

2. Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, Subject: Followup Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident, dated September 27, 1979.

3. Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, Subject: Discussion of Lessons Learned Short-Term Requirements, dated November 9, 1979.

4. Letter from D. G. Eisenhut, NRC, to All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits, Subject: Preliminary clarification of TMI Action Plan Requirements, dated September 5, 1980.

5. Letter from D. G. Eisenhut, NRC, to All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits, Subject: Interim Criteria for Shift Staffing, dated July 31, 1980.

6. Letter from H. R. Denton, NRC, to All Power Reactor Applicants and Licensees, Subject: Qualification of Reactor Operators, dated March 28, 1980.

7. Nuclear Regulatory Commission, "TMI-Related Requirements for New Operating Licenses", USNRC Report NUREG-0694,June 1980.

8. Letters from D. F Ross, Jr., NRC, to All Pending W, CE, and B&W License Applicants, Subject: Actions Required from B&O Task Force Review, dated March 10, 1980 (April 24, 1980).

TABLE 1.9 2 (Sheet 1 of 4) Revision 301 (09/07)

TMI INFORMATION REQUIRED FOR CONTROL ROOM

HABITABILITY EVALUATION (TASK ACTION PLAN ITEM III.D.3.4)

ITEM NO.	Information Required	Information	Reference
1.	Control room mode of operation i.e. pressurization and filter recirculation for radiological accident isolation or chlorine release.	See reference	Subsection 6.4.3.3
→(EC 5000082445, R301)			
2.	Control room characteristics		
a)	air volume control room	220,000 ft^3 Maximum 168,500 ft^3 Minimum	Subsection 6.4.2.2 and Table 6.4 2
←(EC 5000082445, R301)			
b)	Control room emergency zone (control room, critical files and kitchen, washroom, computer room, etc).	The control room envelope is defined to include the main control room, computer room, computer room air conditioning equipment room, control room HVAC equipment room, emergency living quarters, emergency food and water storage rooms, toilets, locker room, kitchenette, supervisors office, corridors, conference room and vault (critical document reference file). Control room operators will require access to the above areas immediately after and during an emergency. The entire envelope floor is at elevation +46 ft. MSL inside the Reactor Auxiliary Building. Drawing G134 is a layout drawing showing the control room envelope, and the placement of equipment.	Subsection 6.4.2.1 and figure 1.2 8
→(DRN 07 241, R301)			
c)	Control room ventilation system schematics with normal and emergency air flowrates.	See reference	Figure 6.4 1 through 6.4 3
←(DRN 07 241, R301)			

TABLE 1.9 2 (Sheet 2 of 4) Revision 11 A (02/02)

TMI INFORMATION REQUIRED FOR CONTROL ROOM

HABITABILITY EVALUATION (TASK ACTION PLAN ITEM III.D.3.4)

ITEM NO.	Information Required	Information	Reference
d)	Infiltration leakage rate	200CFM (max)	Subsection 6.4.2.3 and Table 6.4.2
→ (DRN 01 570)			
e)	high efficiency particulate air (HEPA) filter and charcoal absorber efficiencies	HEPA filter 99.97% Charcol Absorber see referenced table	Table 9.4 2
← (DRN 01 570)			
f)	Closest distance between containment and air intake	Approximately 75 ft.	Figure 1.2 1
g)	Layout of control room, air intakes, containment building, and chlorine, or other chemical storage facility with dimensions.	See reference	Subsection 6.4.4.2, Table 2.2 3, 1.1 2, 1.2 8, 1.2 17, 1.2 18, 1.2 19, 1.2 20, 1.2 21 and 1.2 22
h)	control room shielding including radiation streaming from penetrations, doors, ducts, stairways, etc.	See reference	Subsection 6.4.2.5
i)	Automatic isolation capability damper closing time, damper leakage and area.	See reference	Subsection 6.4.4.2 and Table 6.4 1
j)	Chlorine detectors or toxic gas (local or remote)	Chlorine and Anhydrous Ammonia detectors are local detectors	Subsection 6.4.4.2
k)	self contained breathing apparatus availability (number)	7 units (for 5 men)	Subsection 6.4.4.2
l)	bottled air supply (hours supply)	6 hours (for 5 men)	Subsection 6.4.4.2

TABLE 1.9 2 (Sheet 3 of 4) Revision 11 (05/01)

TMI INFORMATION REQUIRED FOR CONTROL ROOM

HABITABILITY EVALUATION (TASK ACTION PLAN ITEM III.D.3.4)

ITEM NO.	Information Required	Information	Reference
m)	Emergency food and potable water supply (how many days and how many people)	5 days (for 5 men)	Subsection 6.4.4.2
n)	Control room personnel capacity (normal and emergency)	5 men	Subsections 6.4.1 and 6.4.4.2
o)	potassium iodide drug supply	See reference	Subsection 13.3.5.6.2
→ (DRN 99 1093) ← (DRN 99 1093)			
4.	Offsite manufacturing, storage or transportation facilities of hazardous chemicals.	See reference	Subsection 2.2.2, 2.2.3 and Tables 2.2 1 through 2.2 10
a)	identify facilities within a five mile radius	See reference	Table 2.2 1
b)	Distance from control room	See reference	Table 2.2 3
c)	Quantity of hazardous chemicals is one container	See reference	Table 2.2 3
d)	Frequency of hazardous chemical transportation traffic (truck, rail, and barge)	See reference	Subsections 2.2.2.1.2 2.2.2.2, 2.2.2.3, 2.2.2.4, Tables 2.2 2, 2.2 4, 2.2 5, 2.2 6, 2.2 8 and 2.2 9

TABLE 1.9 2 (Sheet 4 of 4) Revision 11 (05/01)

TMI INFORMATION REQUIRED FOR CONTROL ROOM

HABITABILITY EVALUATION (TASK ACTION PLAN ITEM III.D.3.4)

ITEM NO.	Information Required	Information	Reference
5.	Technical Specifications (refer to standard technical specifications)		
a)	Chlorine detection system	For chlorine or ammonia detectors, see reference	Subsection 16.2.3/4.3.3.7
b)	control room emergency filtration system including the capability to maintain the control room pressurization at 1/8 in. water gauge, verification of isolation by test signals and damper closure times, and filter testing requirements.	See reference	Subsection 16.2.3/4.7.7.1

TABLE 1.9-3

CONTAINMENT ISOLATION VALVES PROVIDED WITH CAPABILITY FOR MANUAL OVERRIDE

	System Name	Fluid	Valve Tag No.
1.	Instrument Air	Compressed Air	2IA-F601A/B
2.	Component Cooling Water Inlet to Reactor Coolant Pumps and CEDM Coolers	Demineralized Water	2CC-F146A/B
3.	Component Cooling Water Outlet from Reactor Coolant Pumps and CEDM Coolers	Demineralized Water	2CC-F147A/B 2CC-243A/B
4.	CVCS Letdown Line	Borated Water	2CH-F1518A/B 1CH-F2501A/B
5.	Sampling Line from RCS	Primary Coolant	2SL-F1504A/B 2SL-F15O1A/B
6.	Sampling Line from Pressurizer Surge	Primary Coolant	2SL-F1505A/B 2SL-F15O2A/B
7.	Sampling Line from Pressurizer Steam Space	Primary Coolant	2SL-F1506A/B 2SL-F1503A/B
8.	Containment Sump Pump Discharge	Borated Water	2WM-F105A/B 2WM-F104A/B
9.	CVCS from RCP Controlled Bleedoff	Primary Coolant	2CH-F1512A/B 2CH-F1513A/B
10.	Sampling from Steam Generator 1 Blowdown	Secondary Coolant	2SL-F602 2SL-F601
	Sampling from Steam Generator 2 Blowdown	Secondary Coolant	2SL-F604 2SL-F603
12.	Hydrogen Analyzer Supply and Return Lines	Hydrogen	2HA-E609A 2HA-E608A 2HA-E610A 2HA-E629B 2HA-E628B 2HA-E630B

TABLE 1.9-4 Revision 10 (10/99)

ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION

Radiation Monitoring Instrumentation	1 Detector Instru. Tag No.	2 Control Wiring Dia. B-424 Sh. No.	3 General Atomic Model No.	4 No. Of Channels	5 Power Source	6 Qualified to IEEE 323/74 344/75	7 LOCATION - FSAR Fig. No./Column Line Iso-nozzle	8 Filter	9 Detector	10 Micro-processor	11 Readout	12 Recording	13 Range	14 Sensi-tivity	15 Energy Depend-ence	16 Response cpm/ ?Ci/cc	17 Calibra-tion Frequency (Note 1)	18 Calibra-tion Method-ology
1. Plant Vent Stack	RD-HV-0110	2648	RD-52 RD-72	1	Note 3	Yes	EL. + 111 Plant Stack	1.2-17 5A	1.2-17 8A	Dwg. G135 7A-8A/ K-L	CR Panel CP-52 & CP-6	CR-CP-52 with rec.(Note 4)	10^{7}-10^{5} μCi/cc	NA	NA	4.32×10^{7} 2.84×10^{4} 1.19×10^{2}	1	Note 2
2. Condensor Vacuum Pump Effluents	RD-AE-0002	2649	RD-52 RD-72	1	Note 3	Yes	NA	1.2-3 7G	1.2-3 6G	1.2-3 5G	CR Panel Cp-51 CP-52 & CP-6	CR-CP-52 with rec.(Note 4)	10^{7}-10^{5} μCi/cc	NA	NA	4.32×10^{7} 2.84×10^{4} 1.19×10^{2}	1	Note 2
3. FHB Emergency Exhaust	RD-HV-3032	2639	RD-52 RD-72	1	Note 3	Yes	FHB EL. 9 ft. 6 in.Exh. Plenum	1.2-15	1.2-17 3FH/V-T	1.2-15 V	CR Panel CP-51 CP-52 & CP-6	CR-CP-52 with rec.(Note 4)	10^{7}-10^{5} μCi/cc	NA	NA	4.32×10^{7} 2.84×10^{4} 1.19×10^{2}	1	Note 2
4. Main Steam Line	RD-MS-5500 A, 5500 B	2690 2691		2 (1 for each M.S. line)	Note 4	Yes	NA	NA	1.2-17 140°	Dwg. G135 7A-8A/ K-L	CR Panel CP-51 CP-52& CP-6	CR-CP-52 with rec.		NA	100KeV -3MeV ± 20%	NA	1	Note 2
5. Containment High Range	RD-CA-5400 AS 5400 BS	2635 S 2636 S	RS 23A	2 (SA, SB)	Note 5	Yes	NA	NA	1.2-17 270°- 90°	Dwg. G135 7A-8A/ K-L	CR Panel CP-14 & CP-6	CR-CP-14 with rec.	10^{0}-10^{8} mr/hr	NA	0.1-3MeV ± 20%	NA	1	Note 2

NOTES
1. Calibration Frequency per refueling outage as recommended by NUREG 0472.
2. Methodology submitted by manufacturer - to be incorporated into Operating Procedures.
3. Power not available upon Loss of Offsite Power (LOOP).
4. Power from Static Uninterruptible Power Supply (SUPS).
5. Power from Emergency DG upon LOOP.