Type of Review

A postimplementation review will be performed.

Documentation Required

For operating licenses applicants, provide a description of the installed high-range containment monitors and specify the locations of these monitors inside containment. The description of the monitors should include:

(1) The description of or name of manufacturer and model number of the monitors;

(2) Verification that the monitors meet the specifications of Table II.F.1-3;

(3) Verification that the monitors will be operable on January 1, 1982; and,

(4) A plant layout drawing showing the location of the monitors.

Operating Reactors--By July 1, 1981 have available for review the final design details of the implementation of the above position and clarifications. If deviations to the above position or clarifications are necessary, provide a detailed explanation of and justification for the deviations by July 1, 1981.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.8.b

NUREG-0660

Regulatory Guide 1.97, Revision 2

Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Power Plants, dated September 13, 1979.

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

TABLE II.F.1-3

CONTAINMENT HIGH-RANGE RADIATION MONITOR

REQUIREMENT	-	The capabilit. detect and measure the radiation level within the reactor containment during and following an accident.
RANGE	-	1 rad/hr to 10^8 rads/hr (beta and gamma) or alternatively 1 R/hr to 10^7 R/hr (gamma only).
RESPONSE	-	60 keV to 3 MeV photons, with linear energy response ± 20%) for photons of 0.1 MeV to 3 MeV. Instruments must be accurate enough to provide usable information.
REDUNDANT	-	A minimum of two physically separated monitors (i.e., monitoring widely separated spaces within containment).
DESIGN AND QUALIFICATION	-	Category 1 instruments as described in Appendix A, except as listed below.
SPECIAL CALIBRATION		In situ calibration by electronic signal substitution is acceptable for all range decades above 10 R/hr. In situ calibration for at least one decade below 10 R/hr shall be by means of calibrated radiation source. The original laboratory calibration is not an acceptable position due to the possible differences after in situ installation. For high-range calibration, no adequate sources exist, so an alternate was provided.
SPECIAL ENVIRONMENTAL QUALIFICATIONS	-	Calibrate and type-test representative specimens of detectors at sufficient points to demonstrate linearity through all scales up to 10^6 R/hr. Prior to initial use, certify calibration of each detector for at least one point per decade of range between 1 R/hr and 10^3 R/hr.

II.F.1, ATTACHMENT 4, 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 contaiments.

Changes to Previous Requirements And Guidance

Regulatory Guide 1.97, Rev. 2 has been referenced since the October 30, 1979 letter as the guide for the design and qualification criteria for the containment pressure monitor. However, there have been many changes made to this proposed revision and it has not yet been made final. Therefore, the appropriate sections of the latest version of Regulatory Guide 1.97 has been added to this letter, Appendix A, and this is to be considered a new requirement.

The implementation date has been changed because of the new requirements and because of equipment procurement problems. The new implementation schedule is intended to allow licensees enough time to complete design modifications with a minimum number of plant shutdowns.

Clarification

(1) Design and qualification criteria are outlined in Appendix A.

(2) Measurement and indication capability shall extend to 5 psia for subatmospheric containments.

(3) Two or more instruments may be used to meet requirements. However, instruments that need to be switched from one scale to another scale to meet the range requirements are not acceptable.

(4) Continuous display and recording of the containment pressure over the specified range in the control room is required.

(5) The accuracy and response time specifications of the pressure monitor shall be provided and justified to be adequate for their intended function.

APPLICABILITY

This requirement applies to all operating reactors and all applicants for operating licenses.

Implementation

For operating reactors, design modifications should be completed by January 1, 1982.

Operating license applicants with an operating license dated before January 1, 1982 must have design changes completed by January 1, 1982; those applicants with license dated after January 1, 1982 must have all design modifications completed before they can receive their operating license.

Type of Review

A postimplementation review will be performed for operating reactors.

Documentation Required

The licensees shall inform the NRC when the required design modifications have been completed. Applicants with operating license dates beyond January 1, 1982 shall provide the required design information at least 6 months before the expected date of operation.

Technical Specification Changes Required

Changes to technical specification will be required.

References

NUREG-0660

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

II.F.1, ATTACHMENT 5, 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 5 feet above the normal water level of the suppression pool.

Changes to Previous Requirements and Guidance

Regulatory Guide 1.97, Rev. 2 was referenced in the October 30, 1979 letter as the guide for the design and qualification criteria for the wide range containment water level monitor. However, there have been many changes made to this proposed revision and it has not yet been made final. Therefore, the appropriate sections of the latest version of Regulatory Guide 1.97 has been added to this letter (Appendix A) and this is to be considered a new requirement.

The implementation date has been changed because of the new requirements and because of equipment procurement problems. The new implementation schedule is intended to allow licensees enough time to complete design modifications with a minimum number of plant shutdowns.

Clarification

(1) The containment wide-range water level indication channels shall meet the design and qualification criteria as outlined in Appendix A. The narrow-range channel shall meet the requirements of Regulatory Guide 1.89.

(2) The measurement capability of 600,000 gallons is based on recent plant designs. For older plants with smaller water capacities, licensees may propose deviations from this requirement based on the available water supply capability at their plant.

(3) Narrow-range water level monitors are required for all sizes of sumps but are not required in those plants that do not contain sumps inside the containment.

(4) For BWR pressure-suppression containments, the emergency core cooling system (ECCS) suction line inlets may be used as a starting reference point for the narrow-range and wide-range water level monitors, instead of the bottom of the suppression pool.

(5) The accuracy requirements of the water level monitors shall be provided and justified to be adequate for their intended function.

Applicability

This requirement applies to all operating reactors and all operating licenses for applicants

Implementation

For operating reactors, design modifications should be completed by January 1, 1982.

Operating license applicants with an operating license date before July 1, 1981 must have design changes completed by July 1, 1981, whereas those applicants with license dates past July 1, 1981 must have all design modifications completed before they can receive their operating license.

Type of Review

A postimplementation review will be performed for operating reactors and applicants for an operating license prior to January 1, 1982.

A preimplementation review will be performed for applicants for an operating license after January 1, 1982.

Documentation Required

Submittals from operating reactors licensees and applicants for operating licenses (with an operating license date before January 1, 1982) shall be provided by January 1, 1982. Applicants with operating license dates beyond January 1, 1982 shall provide the required design information at least 6 months before the expected date of operation.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0660

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

II.F.1, ATTACHMENT 6, CONTAINMENT HYDROGEN MONITOR

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% hydrogen concentration under both positive and negative ambient pressure.

Changes to Previous Requirements and Guidance

Regulatory Guide 1.97, Rev. 2 was referenced in the October 30, 1979 letter as the guide for the design and qualification criteria for the containment hydrogen monitor. However, there have been many changes made to this proposed revision and it has not yet been made final. Therefore, the appropriate sections of the latest version of Regulatory Guide 1.97 have been added to this letter (Appendix A) and, therefore, this is to be considered a new requirement.

The implementation date has been changed due to equipment procurement problems. The new implementation schedule is intended to allow licensees enough time to complete design modifications with a minimum number of plant shutdowns.

Clarification

(1) Design and qualification criteria are outlined in Appendix A.

(2) The continuous indication of hydrogen concentration is not required during normal operation.

If an indication is not available at all times, continuous indication and recording shall be functioning within 30 minutes of the initiation of safety injection.

(3) The accuracy and placement of the hydrogen monitors shall be provided and justified to be adequate for their intended function.

Applicability

This requirement applies to all operating reactors and all applicants for operating licenses.

Implementation

For operating reactors, design modifications should be completed by January 1, 1982.

Operating license applicants with an operating license date before January 1, 1982 must have design changes completed by January 1, 1982, whereas those applicants with license dates past January 1, 1982 must have all design modifications completed before they can receive their operating license.

Type of Review

A postimplementation review for operating reactors and applicants for an operating license prior to January 1, 1982 will be performed.

A preimplementation review for applicants for an operating license after January 1, 1982 will be performed.

Documentation Required

Operating reactors and applicants for operating license receiving an operating license before January 1, 1982 will submit documentation before January 1, 1982. Applicants with operating license issued after January 1, 1982 shall provide the required design information at least 6 months prior to the expected date of operation.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0660

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

II.F.2 INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

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.

Changes to Previous Requirements and Guidance

(1) Specify the "Design and Qualification Criteria" for the final ICC monitoring system in section, "Clarification" (items 7, 8, and 9), Attachment 1, and Appendix A.

(2) Specify complete documentation package to allow NRC evaluation of the final ICC monitoring systems to begin on January 1, 1981.

(3) No preimplementation review is required but pcstimplementation review of installation and preimplementation review before use as a basis for operator decisions are required.

(4) Installation of additional instrumentation is now required by January 1, 1982.

(5) Clarification item (6) has been expanded to provide licensees/applicants with more flexibility and diversity in meeting the requirements for determining liquid level indication by providing possible examples of alternative methods.

Previous guidance on the design and qualification criteria for upgrading of existing instrumentation was based on Regulatory Guide 1.97, which is still being developed. Detailed design requirements for incore thermocouples and additional instrumentation were not specified. The pertinent portions of draft Regulatory Guide 1.97 have now been included as Appendix A. Design requirements for incore thermocouples used in the ICC monitoring system are specified in Attachment 1. The only significant change in design requirements involves a relaxation of qualification requirements for display systems amenable to computer processing. This facilitates procurement of computer systems and makes feasible the use of cathode ray tube (CRT) displays that may be needed for proper interpretation of some reactor-water-level systems under development. This relaxation can be accomplished without compromise of ICC monitoring reliability by requiring 99% availability for the display systems, by requiring postaccident maintenance accessibility for nonredundant portions of the system, and by relying on diverse methods of ICC monitoring that include completely qualified display systems.

The staff has concluded that the previous installation requirement of January 1, 1981 for additional instrumentation is unrealistic for most licensees, due to procurement and development problems associated with proposed measurement methods. Further, the staff cannot find the proposed methods acceptable for use until development programs have been completed.

Clarification

(1) Design of new instrumentation should provide an unambiguous indication of ICC. This may require new measurements or a synthesis of existing measurements which meet design criteria (item 7).

(2) The evaluation is to include reactor-water-level indication.

(3) Licensees and applicants are required to provide the necessary design analysis to support the proposed final instrumentation system for inadequate core cooling and to evaluate the merits of various instruments to monitor water level and to monitor other parameters indicative of core-cooling conditions.

(4) The indication of ICC must be unambiguous in that it should have the following properties:

    (a) It must indicate the existence of inadequate core cooling caused by various phenomena (i.e., high-void fraction-pumped flow as well as stagnant boil-off); and,

    (b) It must not erroneously indicate ICC because of the presence of an unrelated phenomenon.

(5) The indication must give advanced warning of the approach of ICC.

(6) The indication must cover the full range from normal operation to complete core uncovery. For example, water-level instrumentation may be chosen to provide advanced warning of two-phase level drop to the top of the core and could be supplemented by other indicators such as incore and core-exit thermocouples provided that the indicated temperatures can be correlated to provide indication of the existence of ICC and to infer the extent of core uncovery. Alternatively, full-range level instrumentation to the bottom of the core may be employed in conjunction with other diverse indicators such as core-exit thermocouples to preclude misinterpretation due to any inherent deficiencies or inaccuracies in the measurement system selected.

(7) All instrumentation in the final ICC system must be evaluated for conformance to Appendix A, "Design and Qualification Criteria for Accident Monitoring Instrumentation," as clarified or modified by the provisions of items 8 and 9 that follow. This is a new requirement.

(8) If a computer is provided to process liquid-level signals for display, seismic qualification is not required for the computer and associated

hardware beyond the isolator or input buffer at a location accessible for maintenance following an accident The single-failure criteria of item 2, Appendix A, need not apply to the channel beyond the isolation device if it is designed to provide 99% availability with respect to functional capability for liquid-level display. The display and associated hardware beyond the isolation device need not be Class 1E, but should be energized from a high-reliability power source which is battery backed. The quality assurance provisions cited in Appendix A, item 5, need not apply to this portion of the instrumentation system. This is a new requirement.

(9) Incore thermocouples located at the core exit or at discrete axial levels of the ICC monitoring system and which are part of the monitoring system should be evaluated for conformity with Attachment 1, "Design and Qualification Criteria for PWR Incore Thermocouples," which is a new requirement.

(10) The types and locations of displays and alarms should be determined by performing a human-factors analysis taking into consideration:

(a) the use of this information by an operator during both normal and abnormal plant conditions,

(b) integration into emergency procedures,

(c) integration into operator training, and

(d) other alarms during emergency and need for prioritization of alarms.

Applicability

This requirement applies to all operating reactors and applicants for operating license.

Implementation

This requirement must be implemented by January 1, 1982.

Type of Review

A postimplementation review will be performed for installation, and a preimplementation review will be performed prior to use.

Documentation Required

By January 1, 1981, the licensee shall provide a report detailing the planned instrumentation system for monitoring of ICC. The report should contain the necessary information, either by inclusion or by reference to previous submittals including pertinent generic reports, to satisfy the requirements which follow:

(1) A description of the proposed final system including:

(a) a final design description of additional instrumentation and displays;

(b) a detailed description of existing instrumentation systems (e.g., subcooling meters and incore thermocouples), including parameter ranges and displays, which provide operating information pertinent to ICC considerations; and

(c) a description of any planned modifications to the instrumentation systems described in item 1.b above.

(2) The necessary design analysis, including evaluation of various instruments to monitor water level, and available test data to support the design described in item 1 above.

(3) A description of additional test programs to be conducted for evaluation, qualification, and calibration of additional instrumentation.

(4) An evaluation, including proposed actions, on the conformance of the ICC instrument system to this document, including Attachment 1 and Appendix A. Any deviations should be justified.

(5) A description of the computer functions associated with ICC monitoring and functional specifications for relevant software in the process computer and other pertinent calculators. The reliability of nonredundant computers used in the system should be addressed.

(6) A current schedule, including contingencies, for installation, testing and calibration, and implementation of any proposed new instrumentation or information displays.

(7) Guidelines for use of the additional instrumentation, and analyses used to develop these procedures.

(8) A summary of key operator action instructions in the current emergency procedures for ICC and a description of how these procedures will be modified when the final monitoring system is implemented.

(9) A description and schedule commitment for any additional submittals which are needed to support the acceptability of the proposed final instrumentation system and emergency procedures for ICC.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.3.b

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

II.F.2, ATTACHMENT 1, DESIGN AND QUALIFICATION CRITERIA FOR PRESSURIZED-WATER REACTOR INCORE THERMOCOUPLES

(1) Thermocouples located at the core exit for each core quadrant, in conjunction with core inlet temperature data, shall be of sufficient number to provide indication of radial distribution of the coolant enthalpy (temperature) rise across representative regions of the core. Power distribution symmetry should be considered when determining the specific number and location of thermocouples to be provided for diagnosis of local core problems.

(2) There should be a primary operator display (or displays) having the capabilities which follow:

(a) A spatially oriented core map available on demand indicating the temperature or temperature difference across the core at each core exit thermocouple location.

(b) A selective reading of core exit temperature, continuous on demand, which is consistent with parameters pertinent to operator actions in connecting with plant-specific inadequate core cooling procedures. For example, the action requirement and the displayed temperature might be either the highest of all operable thermocouples or the average of five highest thermocouples.

(c) Direct readout and hard-copy capability should be available for all thermocouple temperatures. The range should extend from 200°F (or less) to 1800°F (or more).

(d) Trend capability showing the temperature-time history of representative core exit temperature values should be available on demand.

(e) Appropriate alarm capability should be provided consistent with operator procedure requirements.

(f) The operator-display device interface shall be human-factor designed to provide rapid access to requested displays.

(3) A backup display (or displays) should be provided with the capability for selective reading of a minimum of 16 operable thermocouples, 4 from each core quadrant, all within a time interval no greater than 6 minutes. The range should extend from 200°F (or less) to 2300°F (or more).

(4) The types and locations of displays and alarms should be determined by performing a human-factors analysis taking into consideration:

(a) the use of this information by an operator during both normal and abnormal plant conditions.

(b) integration into emergency procedures,

(c) integration into operator training, and

(d) other alarms during emergency and need for prioritization of alarms.

(5) The instrumentation must be evaluated for conformance to Appendix B, "Design and Qualification Criteria for Accident Monitoring Instrumentation," as modified by the provisions of items 6 through 9 which follow.

(6) The primary and backup display channels should be electrically independent, energized from independent station Class IE power sources, and physically separated in accordance with Regulatory Guide 1.75 up to and including any isolation device. The primary display and associated hardware beyond the isolation device need not be Class IE, but should be energized from a high-reliability power source, battery backed, where momentary interruption is not tolerable. The backup display and associated hardware should be Class IE.

(7) The instrumentation should be environmentally qualified as described in Appendix B, item 1, except that seismic qualification is not required for the primary display and associated hardware beyond the isolater/input buffer at a location accessible for maintenance following an accident.

(8) The primary and backup display channels should be design to provide 99% availability for each channel with respect to functional capability to display a minimum of four thermocouples per core quadrant. The availability shall be addressed in technical specifications.

(9) The quality assurance provisions cited in Appendix B, item 5, should be applied except for the primary display and associated hardware beyond the isolation device.

II.G.1 EMERGENCY POWER FOR PRESSURIZER EQUIPMENT

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:

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.

(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.

(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.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

(1) Although the primary concern resulting from lessons learned from the accident at TMI is that the PORV block valves must be closable, the design should retain, to the extent practical, the capability to also open these valves.

(2) The motive and control power for the block-valve should be supplied from an emergency power bus different from the source supplying the PORV.

(3) Any changeover of the PORV and block-valve motive and control power from the normal offsite power to the emergency onsite power is to be accomplished manually in the control room.

(4) For those designs in which instrument air is needed for operation, the electrical power supply should be required to have the capability to be manually connected to the emergency power sources.

Applicability

This requirement applies to all PWR operating reactors and all applicants for a PWR operating license.

Implementation

Implementation is complete for operating reactors. This requirement shall be implemented by applicants for operating license prior to the issuance of a fuel-loading license.

Type of Review

No further review for operating reactors is necessary.

Documentation Required

Each applicant shall provide sufficient documentation to support a reasonable assurance finding by the NRC that each of the positions stated above are met. The documentation should include, as a minimum, supporting information including system design description, logic diagrams, electrical schematics, test procedures, and technical specifications.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.1

NUREG-0660, Item II.G.1

NUREG-0694, Part 1

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1980.

II.K.2.2 CONTROL OF AUXILIARY FEEDWATER INDEPENDENT OF THE INTEGRATED CONTROL SYSTEM

Position

For Babcock and Wilcox (B&W)-designed reactors, provide procedures and training to initiate and control auxiliary feedwater independent of the integrated control system (ICS).

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

No further clarification is required at this time.

Applicability

This requirement applies to all operating license applicants of B&W-designed reactors.

Implementation

All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Type of Review

A review is not applicable at this time.

Documentation Required

Applicants shall provide sufficient documentation at least 4 months prior to the issuance of the staff safety evaluation report for a full power license to support a reasonable assurance finding by the NRC that the position specified above has been met.

Technical Specification Changes Required

Changes to technical specifications will not be required.

Reference

NUREG-0660, Item II.K.2, Table C.2, Item 2

II.K.2.8 AUXILIARY FEEDWATER SYSTEM UPGRADING

Position

All operating Babcock and Wilcox (B&W) plants were ordered to be shut down shortly after the TMI-2 accident. The orders included both short-term and long-term actions. The NRR Bulletins and Orders Task Force reviewed the licensees' compliance with the short-term actions of the orders and issued safety evaluation reports which served as the basis for plant restart. Additional items were identified in the review of the long-term actions which require further work by the licensees.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

The licensees were required to comply with the Commission Orders regarding certain short-term and long-term auxiliary feedwater system (AFWS) modifications. The staff evaluated the short-term actions, and safety evaluations were prepared before the plants were allowed to return to operation. The staff evaluation of the additional (long-term) items will be performed in conjunction with item II.E.1.1 in NUREG-0660, Auxiliary Feedwater System Evaluation and item II.E.1.2, AFWS Automatic Initiation and Flow Indicator.

Applicability

This requirement applies to all B&W operating reactors.

Implementation

No separate implementation is required for this item. All AFWS upgrade modifications for B&W plants are being reviewed as part of Section II.E.1.1 and Section II.E.1.2 in NUREG-0660.

Type of Review

See Section II.E.1.1 and Section II.E.1.2 in NUREG-0660.

Documentation Required

See Section II.E.1.1 and Section II.E.1.2 in NUREG-0660.

Technical Specification Changes Required

Changes to technical specifications will be made as required.

References

NUREG-0645, Volume 1, Section 2.4.6

NUREG-0660, Items II.E.1.1, II.E.1.2, and II.K.2.

II.K.2.9 FAILURE MODE EFFECTS ANALYSIS ON THE INTEGRATED CONTROL SYSTEM

Position

For Babcock and Wilcox (B&W)-designed reactors provide a failure-mode-and-effects analysis (FMEA) of the integrated control system (ICS).

Changes to Previous Requirements and Guidance

There are no changes from those issued in the November 7, 1979 letter from R. W. Reid, NRC.

Clarification

A generic failure-mode-and-effects analysis of the ICS (BAW-1564) was submitted on August 17, 1979 by the operating plant licensees. This report was reviewed by the staff and Oak Ridge National Laboratory (ORNL). Requests for additional information, regarding the recommendations contained in the report, were sent to the licensees on November 7, 1979. The responses to the November 7, 1979 letter have been received and are under review.

Applicability

This requirement applies to all B&W operating reactors and operating license applicants.

Implementation

Operating Reactors--Staff recommendations are pending completion of staff review.

Operating License Applicants--All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Type of Review

A postimplementation review will be performed.

Documentation Required

Operating Reactors--Documentation has been completed.

Operating License Applicants--B&W applicants should provide the following:

(1) Identify whether the previous generic submittal (BAW-1564) is applicable to your plant, and

(2) Specify what actions have been taken at your facility to comply with the recommendations listed in BAW-1564.

Technical Specification Changes Required

Changes to technical specifications will be determined following staff review.

References

NUREG-0645, Volume 1, Section 2.4.6

NUREG-0694, Part 2

Commission Orders on B&W Plants

Babcock & Wilcox Co., "Integrated Control System Reliability Analysis," report BAW-1564.

Letter from R. W. Reid, NRC, to All B&W Operating Plants, dated November 7, 1979.

II.K.2.10 SAFETY-GRADE ANTICIPATORY REACTOR TRIP

Position

For Babcock and Wilcox (B&W)-designed reactors, install safety-grade, anticipatory reactor trip (ART) on loss-of-feedwater and turbine trip.

Changes to Previous Requirements and Guidance

New request for final design submittal. Extension of date for submittal of design from October 1, 1980 to January 1, 1981.

Clarification

Operating Reactors

(1) IE Bulletin 79-05B, Item 5, issued on April 21, 1979, directed B&W licensees to provide a design and schedule for implementation of a safety-grade reactor trip upon:

   (a) loss of feedwater;
   (b) turbine trip; and
   (c) significant reduction in steam generator level.

(2) In accordance with IE Bulletin 79-05B, the B&W licensees submitted a conceptual design for a safety-grade, anticipatory reactor trip which would be initiated upon turbine trip and loss of feedwater only. Included in the licensees' responses was a generic evaluation prepared by B&W which proposed that the anticipatory reactor trip on low steam generator level was not necessary.

(3) Staff review of these submittals resulted in a preliminary design approval for the safety-grade anticipatory reactor trip being issued to the B&W licensees on December 20, 1979. However, the approval letters also specified the additional information which would be required to be submitted prior to final staff approval of the design.

(4) The staff will complete its review of the generic evaluation by B&W which indicates that the proposed anticipatory trip on low steam generator level is unnecessary. Further clarification will be provided on this matter, if required, following completion of the staff review.

Operating License Applicants--Compliance with item II.K.1 of NUREG-0694 (C.1.21) satisfies this requirement.

Applicability

This requirement applies to all B&W operating reactors and applicants for operating license.

Implementation

Operating Reactors--Final design information will be submitted by January 1, 1981. Safety-grade trip will be installed by July 1, 1981.

Operating License Applicants--Implementation of NUREG-0694, II.K.1 (C.1.21) prior to the issuance of the fuel load satisfies this requirement.

Type of Review

A postimplementation review will be performed.

Documentation Required

The following information was identified as required by the staff:

(1) The final design submittal should include the final logic diagrams, electrical schematic diagrams, piping and instrumentation diagrams, and location layout drawings.

(2) For sensors located in nonseismic areas which have not previously contained reactor protection systems (RPS) inputs, perform and submit an analysis which shows that the installation (including circuit routing) is designed such that the effects of credible faults (i.e., grounding, shorting, application of high voltage, or electromagnetic interference) or failures in these areas could not be propagated back to the RPS and degrade the RPS performance or operability.

(3) Submit "Seismic and Environmental Qualification Summary Reports" for the equipment which have not been previously submitted. In addition, demonstrate that the environmental test conditions bound the actual worst-case accident conditions expected at the installed locations.

(4) Assure that the anticipatory reactor trip (ART) testability includes provisions to perform channel functional tests at power. Testing of this circuitry is to be included in the RPS monthly surveillance tests.

(5) Include in the final design submittal the RPS checkout procedure which will demonstrate both the operability of the new trip circuitry and the continued operability of the previous RPS.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0645, Volume 1, Section 2.4.6

NUREG-0694, Item II.K.1 (C.1.21)

Commission Orders on B&W Plants

IE Bulletin 79-05B, Item 5, April 21, 1979

Letter from R. W. Reid, NRC, to B&W Licensees, dated December 20, 1979.

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

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.

Changes to Previous Requirements and Guidance

Licensees of Babcock and Wilcox (B&W) operating reactors shall submit the results of their evaluations by January 1, 1981. The completion schedule has been changed to allow time to complete the results of the evaluation. Also, this requirement has been changed to include all operating pressurized-water reactors (PWRs) and applicants.

Clarification

The position deals with the potential for thermal shock of reactor vessels resulting from cold safety injection flow. One aspect that bears heavily on the effects of safety injection flow is the mixing of safety injection water with reactor coolant in the reactor vessel. B&W provided a report on July 30, 1980 that discussed the mixing question and the basis for a conservative analysis of the potential for thermal shock to the reactor vessel. Other PWR vendors are also required to address this issue with regard to recovery from small breaks with an extended loss of all feedwater. In particular, demonstration shall be provided that sufficient mixing would occur of the cold high-pressure injection (HPI) water with reactor coolant so that significant thermal shock effects to the vessel are precluded.

Applicability

This requirement applies to all PWR operating reactors and applicants for an operating license.

Implementation

Implementation of any modifications will be determined by to the results of NRC staff review of the report.

Type of Review

A postimplementation review will be performed.

Documentation Required

Licensees of B&W operating reactors shall submit the results of their evaluations by January 1, 1981. Other PWR licensees shall submit the results of their evaluation by January 1, 1982. Applicants for operating license shall submit the results of their evaluations at least 6 months prior to the issuance of the staff safety evaluation report for a full-power license.

Technical Specification Changes Required

Changes to technical specifications will be determined following staff review.

References

NUREG-0645, Volume 1, Section 2.4.5

Letter from D. F. Ross Jr., NRC, to All B&W Operating Plants, dated August 21, 1979.

Letter from D. G. Eisenhut, NRC, to All Licensees, dated May 7, 1980.

II.K.2.15 EFFECTS OF SLUG FLOW ON STEAM GENERATOR TUBES

Position

Although the staff believed that the potential for slug flow was not great in Babcock and Wilcox (B&W) plants because of the venting path provided by the internal vent valves, the staff required that a confirmatory evaluation of the effects of slug flow on steam generator tubes be performed by the licensees to assure that the tubes could withstand any mechanical loading which could result from slug flow.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

The request for this information was originally sent to the B&W licensees in a letter from R. W. Reid, NRC, to all B&W operating plants, dated November 21, 1979.

The results of this analysis have been submitted by the licensees and is presently undergoing NRC staff review.

Applicability

This requirement applies to all B&W operating reactors and operating license applicants.

Implementation

Implementation of any modifications will be determined by to the results of NRC staff review of the evaluation.

Type of Review

A postimplementation review will be performed.

Documentation Required

No additional documentation is required at this time from licensees. Applicants must supply the requested information at least 4 months before the staff safety evaluation report for a full-power license is scheduled to be issued.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0565, Recommendation 2.6.2.1

NUREG-0645, Volume 1, Section 2.4.6

NUREG-0694, Part 2

Letter from R. W. Reid, NRC, to All B&W Operating Plants, dated November 21, 1979.

II.K.2.16 REACTOR COOLANT PUMP SEAL DAMAGE

Position

Evaluate the impact of reactor coolant pump seal damage and leakage due to loss-of-seal cooling upon loss of offsite power. If damage cannot be precluded, licensees should provide an analysis of the limiting small-break loss-of-coolant accident (LOCA) with subsequent reactor coolant pump (RCP) seal damage.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

The request for this information was originally sent to the Babcock and Wilcox (B&W) licensees in a letter from R. W. Reid, NRC, to all B&W operating plants, dated November 21, 1979.

The results of these evaluations have been submitted by the licensees and are presently undergoing NRC staff review.

Applicability

This requirement applies to all B&W operating reactors and operating license applicants.

Implementation

Implementation of any modifications will be determined by to the results of NRC staff review of the evaluations.

Type of Review

A postimplementation review will be performed.

Documentation Required

No additional documentation is required at this time from licensees. Applicants shall submit the requested information at least 4 months before the staff safety evaluation report for a full-power license is scheduled to be issued.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0565, Recommendation 2.6.2.f

NUREG-0645, Volume 1, Section 2.4.6

NUREG-0694, Part 2

Letter from R. W. Reid, NRC, to All B&W Operating Plants, dated November 21, 1979.

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

Position

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

Changes to Previous Requirements and Guidance

The previous requirement has been changed to include all PWR operating reactors and applicants.

Clarification

The background for this concern and a request for this analysis was originally sent to the Babcock and Wilcox (B&W) licensees in a letter from R. W. Reid, NRC, to all B&W operating plants, dated January 9, 1980.

The results of this evaluation have been submitted by the B&W licensees and is presently undergoing staff review.

Applicability

This requirement applies to all PWR operating reactors and operating license applicants.

Implementation

Implementation of any modifications will be determined by to the results of NRC staff review of the licensees/applicants evaluation.

The analysis for all but B&W licenses should be submitted by January 1, 1982 or 6 months before the expected issuance date of the staff safety evaluation report for the license, whichever is later.

Type of Review

A postimplementation review will be performed.

Documentation Required

No additional documentation is required at this time from B&W licensee's. All others should submit analyses as indicated in "Implementation."

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0660, Item II.K.2 (C.17)

Letter from R. W. Reid, NRC, to All B&W Operating Plants, dated January 9, 1980.

II.K.2.19 SEQUENTIAL AUXILIARY FEEDWATER FLOW ANALYSIS

Position

Provide a benchmark analysis of sequential auxiliary feedwater (AFW) flow to the steam generators following a loss of main feedwater.

Changes to Previous Requirements and Guidance

The previous requirement has been changed to include all operating pressurized-water reactors (PWRs) and applicants for operating license.

Clarification

This requirement was originally sent to the Babcock and Wilcox (B&W) licensees in a letter from D. F. Ross, Jr., NRC, to all B&W operating plants, dated August 21, 1979.

The results of this analysis has been submitted by the B&W licensees and is presently undergoing staff review.

Applicability

This requirement applies to all PWR operating reactors and applicants for operating licenses.

Implementation

Implementation of any modifications will be determined by the results of NRC staff review of this analysis.

The analysis for all but B&W licensees should be submitted by January 1, 1982 or 6 months before the expected issuance date of the staff safety evaluation report for a license, whichever is later.

Type of Review

A postimplementation review will be performed.

Documentation Required

No additional documentation is required at this time from B&W licensees. All others should submit analyses as indicated in "Implementation."

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0645, Volume 1, Section 2.4.6

Letter from D. F. Ross, Jr., NRC, to All B&W Operating Plants, dated August 21, 1979.

II.K.2.20 SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WHICH REPRESSURIZES THE REACTOR COOLANT SYSTEM TO THE POWER-OPERATED RELIEF VALVE SETPOINT

Position

Provide an analysis which shows the plant response to a small-break loss-of-coolant accident (LOCA) during which the reactor coolant system (RCS) is repressurized to the power-operated relief valve (PORV) setpoint with subsequent failure of the PORV to close.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

This requirement was originally sent to the Babcock and Wilcox (B&W) licensees in a letter from D. F. Ross, Jr., NRC, to all B&W operating plants, dated August 21, 1979.

The results of this analysis has been submitted by the B&W licensees and is presently undergoing staff review.

Applicability

This requirement applies to all B&W operating reactors.

Implementation

Implementation of any modifications will be determined by the results of NRC staff evaluation of this analysis.

Type of Review

A postimplementation review will be performed.

Documentation Required

No additional documentation is required at this time.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0565, Recommendation 2.6.2.c

NUREG-0645, Volume 1, Section 2.4.6

Letter from D. F. Ross, Jr., NRC, to All B&W Operating Plants, dated August 21, 1979.

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

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.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

Implementation of this action item was modified in the May 1980 version of NUREG-0660. The change delays implementation of this action item until after the studies specified in TMI Action Plan item II.K.3.2 have been completed, if such studies confirm that the subject system is necessary.

Applicability

This requirement applies to all PWR operating reactors and applicants for operating license.

Implementation

If required by action plan item II.K.3.2, licensees shall implement modifications and perform confirmatory tests at the next refueling outage following staff approval of the design, unless this outage is scheduled within 6 months of the approval date. In this event, modifications will be completed during the following refueling outage.

Type of Review

A preimplementation review will be performed.

Documentation Required

If modifications are required, documentation shall include piping and instrumentation diagrams, electrical schematics of design modifications, and an analysis of conformance to IEEE 279-1971 requirements. Documentation shall be submitted by July 1, 1981.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0565, Recommendation 2.1.2.a

NUREG-0611, Recommendations 3.2.4.e and 3.2.4.f

NUREG-0635, Recommendations 3.2.4.a and 3.2.4.b

NUREG-0660

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

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.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

Based on its review of feedwater transients and small LOCAs for operating plants, the Bulletins and Orders Task Force in the Office of Nuclear Reactor Regulation recommended that a report be prepared and submitted for staff review which documents the various actions that have been taken to reduce the probability of a small-break LOCA caused by a stuck-open PORV and show how these actions constitute sufficient improvements in reactor safety. Action Item II.K.3.2 of NUREG-0660, published in May 1980, changed the implementation of this recommendation as follows: In addition to modifications already implemented on PORVs, the report specified above should include safety examination of an automatic PORV isolation system identified in Task Action Plan item II.K.3.1.

Modifications to reduce the likelihood of a stuck-open PORV will be considered sufficient improvements in reactor safety if they reduce the probability of a small-break LOCA caused by a stuck-open PORV such that it is not a significant contributor to the probability of a small-break LOCA due to all causes. (According to WASH-1400, the median probability of a small-break LOCA S_2 with a break diameter between 0.5 in. and 2.0 in. is 10^{-3} per reactor-year with a variation ranging from 10^{-2} to 10^{-4} per reactor-year.)

The above-specified report should also include an analysis of safety-valve failures based on the operating experience of the pressurized-water-reactor (PWR) vendor designs. The licensee has the option of preparing and submitting either a plant-specific or a generic report. If a generic report is submitted, each licensee should document the applicability of the generic report to his own plant.

Based on the above guidance and clarification, each licensee should perform an analysis of the probability of a small-break LOCA caused by a stuck-open PORV or safety valve. This analysis should consider modifications which have been made since the TMI-2 accident to improve the probability. This analysis shall evaluate the effect of an automatic PORV isolation system specified in Task

Action Plan item II.K.3.1. In evaluating the automatic PORV isolation system, the potential of causing a subsequent stuck-open safety valve and the overall effect on safety (e.g., effect on other accidents) should be examined.

Actual operational data may be used in this analysis where appropriate. The bases for any assumptions used should be clearly stated and justified.

The results of the probability analysis should then be used to determine whether the modifications already implemented have reduced the probability of a small-break LOCA due to a stuck-open PORV or safety valve a sufficient amount to satisfy the criterion stated above, or whether the automatic PORV isolation system specified in Task Action item II.K.3.1 is necessary.

In addition to the analysis described above, the licensee should compile operational data regarding pressurizer safety valves for PWR vendor designs. These data should then be used to determine safety-valve failure rates.

The analyses should be documented in a report. If this requirement is implemented on a generic basis, each licensee should review the appropriate generic report and document its applicability to his own plant(s). The report and the documentation of applicability (where appropriate) should be submitted for NRC staff review by the specified date.

Applicability

This requirement applies to all operating PWRs and operating license applicants.

Implementation

The report documenting the specified analyses and the licensee's documentation of applicability (where appropriate) should be submitted for staff review by January 1, 1981.

All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Type of Review

A postimplementation review will be performed.

Documentation Required

The licensee is to submit for staff review a report on the probability of small-break LOCA and safety-valve failure rates along with documentation of applicability (where appropriate) by January 1, 1981.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

WASH-1400 (NUREG-75/014)

NUREG-0565, Recommendation 2.1.2.d

NUREG-0611, Recommendations 3.2.4.g and 3.2.4.i

NUREG-0635, Recommendation 3.2.4.c

NUREG-0660, Items II.K.3.1, II.K.3.2

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

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.

Changes to Previous Requirements and Guidance

Implementation dates are changed to be consistent with test schedule for LOFT test (L3-6) and to provide for blind posttest analysis.

Clarification

This action item has been revised in the May 1980 version of NUREG-0660 to provide for continued study of criteria for early reactor coolant pump trip. Implementation, if any is required, will be delayed accordingly. As part of the continued study, all holders of approved emergency core cooling (ECC) models have been required to analyze the upcoming LOFT test (L3-6). The capability of the industry models to correctly predict the experimental behavior of this test will have a strong input on the staff's determination of when and how the reactor coolant pumps should be tripped.

Applicability

This requirement applies to all PWR operating reactors and operating license applicants.

Implementation

(1) Document models are to be used for analysis prior to December 3, 1980.

(2) DOE/NRC is to run the LOFT test (L3-6) from December 3, 1980 to December 17, 1980.

(3) NRC will distribute initial conditions approximately 4 weeks after the test.

(4) Prediction results will be submitted approximately 4 weeks after receipt of initial conditions.

(5) NRC determination of model acceptability is due April 1, 1981.

(6) Proposed design modifications (if necessary) are due by July 1, 1981.

(7) Modification (if necessary) is due by March 1, 1982.

Type of Review

An NRC preimplementation review will be performed (if any modifications are required).

Documentation Required

Prediction by vendor analysis of LOFT test (L3-6) is required. Additional information needed will depend upon prediction results

Technical Specification Changes Required

Changes to technical specifications are to be determined.

References

NUREG-0565, Recommendation 2.3.2.a

NUREG-0611, Recommendation 3.2.2.a

NUREG-0623

NUREG-0635, Recommendation 3.2.2.a

NUREG-0660

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

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% 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.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

Based on its review of best-estimate calculations performed by Babcock and Wilcox (B&W), the NRC staff believes that the frequency of PORV challenges has been reduced using the revised PORV and high-pressure reactor trip setpoints and assuming that the anticipatory reactor trips function as designed. At this time, however, the staff is unable to make a quantitative judgment of the expected frequency. Therefore, licensees with B&W-designed plants should perform additional analyses of anticipated transients which indicate the sensitivity of PORV challenges to (1) the variation in core physics parameters which may occur in the plant cycle; (2) single failures in mitigating systems; and (3) transients which do not actuate the anticipatory reactor trips. Analytical assumptions should include those specified in the plant final safety analysis reports (FSARs). The results of these more-detailed and extensive analyses should be used to determine the expected frequency of PORV openings for overpressure transients. This frequency should be less than 5% of the total number of overpressure transients, thereby confirming the findings of the staff's review.

The results of this study should be documented and submitted for staff review by the scheduled date.

Applicability

This requirement applies to all B&W operating reactors.

Implementation

The licensee's report documenting the specified analyses should be submitted for staff review by January 1, 1981.

Type of Review

A postimplementation review will be performed.

Documentation Required

The licensee should perform the specified analyses and submit the documentation of the results for staff review by the scheduled date.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0565, Recommendation 2.1.2.b

NUREG-0645, Recommendation 2.4.5, Item 27

Letter from D. F. Ross, Jr., NRC, to All B&W Operating Plants, dated August 21, 1979.

II.K.3.9 PROPORTIONAL INTEGRAL DERIVATIVE CONTROLLER MODIFICATION

Position

The Westinghouse-recommended modification to the proportional integral derivative (PID) controller should be implemented by affected licensees.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

The Westinghouse-recommended modification is to raise the interlock bistable trip setting to preclude derivative action from opening the power-operated relief valve (PORV). Some plants have proposed changing the derivative action setting to zero, thereby eliminating it from consideration. Either modification is acceptable to the staff. This represents a newly available option.

Applicability

This requirement applies to all Westinghouse operating reactors and operating license applicants.

Implementation

Operating Reactors--For operating reactors, modifications will be completed by January 1, 1981.

Operating License Applicants--All applicants for operating license should submit documentation 4 months prior to the expected issuance of an operating license or 4 months prior to the listed implementation date, whichever is later.

Type of Review

A postimplementation review will be performed.

Documentation Required

The licensee and applicant shall inform the NRC when the modification has been completed.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0611, Recommendation 3.2.4.b

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

II.K.3.10 PROPOSED ANTICIPATORY TRIP MODIFICATION

Position

The anticipatory trip modification proposed by some licensees to confine the range of use to high-power levels should not be made until it has been shown on a plant-by-plant basis that the probability of a small-break loss-of-coolant accident (LOCA) resulting from a stuck-open power-operated relief valve (PORV) is substantially unaffected by the modification.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

This evaluation is required for only those licensees/applicants who propose the modification.

Applicability

This requirement applies to selected Westinghouse operating reactors and operating license applicants.

Implementation

Operating Reactors--Completion date for meeting requirements will be dictated by plant schedule for proposed modification.

Operating License Applicants--All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Type of Review

A preimplementation review will be performed.

Documentation Required

(1) The licensee is to submit the required analysis and document-proposed change for staff approval prior to implementation. Documentation is to be submitted as proposed by the licensee.

(2) Modification schedule is to be determined on a plant-specific basis.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0611, Recommendation 3.2.4.c

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

II.K.3.12 CONFIRM EXISTENCE OF ANTICIPATORY REACTOR TRIP UPON TURBINE TRIP

Position

Licensees with Westinghouse-designed operating plants should confirm that their plants have an anticipatory reactor trip upon turbine trip. The licensee of any plant where this trip is not present should provide a conceptual design and evaluation for the installation of this trip.

Changes to Previous Requirements and Guidance

The date for submittal of design has been extended from July 1, 1980 to January 1, 1981.

Clarification

No further clarification is required.

Applicability

This requirement applies to all Westinghouse operating reactors and operating license applicants.

Implementation

Operating Reactors--Confirmation or proposed modification is to be completed by January 1, 1981. The modifications should be completed by the first refueling outage following staff approval of the design unless this outage is scheduled within 6 months of the approval date. In this event, modifications will be completed during the next refueling outage.

Operating License Applicants--All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evalution report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Type of Review

A preimplementation review will be performed (if design modifications are required).

Documentation Required

(1) Licensees should submit confirmation of existence of anticipatory reactor trip upon turbine trip or submit proposed design changes and schedule for implementation by January 1, 198.

(2) A commitment to implement modifications should be provided (if required)

(3) All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0611, Recommendation 3.2.4.a

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

II.K.3.13 SEPARATION OF HIGH-PRESSURE COOLANT INJECTION AND REACTOR CORE ISOLATION COOLING SYSTEM INITIATION LEVELS--ANALYSIS AND IMPLEMENTATION

Position

Currently, the reactor core isolation cooling (RCIC) system and the high-pressure coolant injection (HPCI) system both initiate on the same low-water-level signal and both isolate on the same high-water-level signal. The HPCI system will restart on low water level but the RCIC system will not. The RCIC system is a low-flow system when compared to the HPCI system. The initiation levels of the HPCI and RCIC system should be separated so that the RCIC system initiates at a higher water level than the HPCI system. Further, the initiation logic of the RCIC system should be modified so that the RCIC system will restart on low water level. These changes have the potential to reduce the number of challenges to the HPCI system and could result in less stress on the vessel from cold water injection. Analyses should be performed to evaluate these changes. The analyses should be submitted to the NRC staff and changes should be implemented if justified by the analyses.

Changes to Previous Requirements and Guidance

(1) Analysis and proposed modifications are required by Janaury 1, 1981.

(2) Implementation of modifications are required by July 1, 1981 (if applicable).

Clarification

No further clarification is required.

Applicability

This requirement applies to all operating BWRs and operating license applicants with RCIC and HPCI systems.

Implementation

Analysis and proposed modifications are required by January 1, 1981. Implementation of modifications is required by July 1, 1981 (if applicable)

Type of Review

A preimplementation review will be performed if required.

Documentation Required

(1) The licensee is to provide results of evaluation and proposed modifications (if necessary) to NRC staff by January 1, 1981. The licensee is to provide sufficient supporting analysis to demonstrate that the systems, as modified, would not degrade proper system functions.

(2) The licensee is to implement modifications (if necessary) by July 1, 1981.

(3) All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Technical Specification Changes Required

Changes to technical specifications will be required.

Reference

NUREG-0626, Recommendation A.1

II.K.3.14 ISOLATION OF ISOLATION CONDENSERS ON HIGH RADIATION

Position

Isolation condensers have radiation monitors on their vents. These monitors provide alarms in the control room but do not isolate the isolation condenser. The isolation condensers are currently isolated on a high-radiation signal in the steam line leading to the isolation condensers. The design should be modified such that the isolation condensers are automatically isolated upon receipt of a high-radiation signal at the vent rather than at the steam line. The purpose of the change is to increase the availability of the isolation condensers as heat sinks.

Changes to Previous Requirements and Guidance

There has been no change in the requirements for this task action item from the final recommendations of the Bulletins and Orders (B&O) Task Force. The schedule has been extended to allow completion of design and procurement.

Clarification

No further clarification is required.

Applicability

This requirement applies to all operating BWRs that have isolation condensers.

Implementation

Design modifications shall be completed by January 1, 1982.

Type of Review

A postimplementation review will be performed.

Documentation Required

The licensee shall inform the NRC when the required design modifications have been completed.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0626, Recommendation A.2.

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

II.K.3.15 MODIFY BREAK-DETECTION LOGIC TO PREVENT SPURIOUS ISOLATION OF HIGH-PRESSURE COOLANT INJECTION AND REACTOR CORE ISOLATION COOLING

Position

The high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems use differential pressure sensors on elbow taps in the steam lines to their turbine drives to detect and isolate pipe breaks in the systems. The pipe-break-detection circuitry has resulted in spurious isolation of the HPCI and RCIC systems due to the pressure spike which accompanies startup of the systems. The pipe-break-detection circuitry should be modified so that pressure spikes resulting from HPCI and RCIC system initiation will not cause inadvertent system isolation.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

No further clarification is required.

Applicability

This requirement applies to all operating boiling water reactors (BWRs) and applicants for operating license with HPCI and RCIC systems.

Implementation

Operating Reactors--For operating reactors, these requirements will be completed by July 1, 1981.

Operating License Applicants--All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Type of Review

A postimplementation review will be performed.

Documentation Required

Submit sufficient documentation to support a reasonable assurance finding by the NRC that the modifications, as implemented, have resulted in satisfying the concerns expressed in the "Position" statement above.

Technical Specifications Changes Required

Changes to technical specifications will be required.

Reference

NUREG-0626, Recommendation A.3

II.K.3.16 REDUCTION OF CHALLENGES AND FAILURES OF RELIEF VALVES--FEASIBILITY STUDY AND SYSTEM MODIFICATION

Position

The record of relief-valve failures to close for all boiling-water reactors (BWRs) in the past 3 years of plant operation is approximately 30 in 73 reactor-years (0.41 failures per reactor-year). This has demonstrated that the failure of a relief valve to close would be the most likely cause of a small-break loss-of-coolant accident (LOCA). The high failure rate is the result of a high relief-valve challenge rate and a relatively high failure rate per challenge (0.16 failures per challenge). Typically, five valves are challenged in each event. This results in an equivalent failure rate per challenge of 0.03. The challenge and failure rates can be reduced in the following ways:

(1) Additional anticipatory scram on loss of feedwater,

(2) Revised relief-valve actuation setpoints,

(3) Increased emergency core cooling (ECC) flow,

(4) Lower operating pressures,

(5) Earlier initiation of ECC systems

(6) Heat removal through emergency condensers,

(7) Offset valve setpoints to open fewer valves per challenge,

(8) Installation of additional relief vales with a block- or isolation-valve feature to eliminate opening of the safety/relief valves (SRVs), consistent with the ASME Code,

(9) Increasing the high steam line flow setpoint for main steam line isolation valve (MSIV) closure,

(10) Lowering the pressure setpoint for MSIV closure,

(11) Reducing the testing frequency of the MSIVs,

(12) More-stringent valve leakage criteria, and

(13) Early removal of leaking valves.

An investigation of the feasibility and contraindications of reducing challenges to the relief valves by use of the aforementioned methods should be conducted. Other methods should also be included in the feasibility study. Those changes which are shown to reduce relief-valve challenges without compromising the performance of the relief valves or other systems should be implemented. Challenges to the relief valves should be reduced substantially (by an order of magnitude).

Changes to Previous Requirements and Guidance

The schedule for plant modifications has been changed to allow time for staff review of evaluation and purchase of required hardware.

Clarification

Failure of the power-operated relief valve (PORV) to reclose during the TMI-2 accident resulted in damage to the reactor core. As a consequence, relief valves in all plants, including BWRs, are being examined with a view toward their possible role in a small-break LOCA.

The safety/relief valves (SRV) are dual-function pilot-operated relief valves that use a spring-actuated pilot for the safety function and an external air-diaphragm-actuated pilot for the relief function.

The operating history of the SRV has been poor. A new design is used in some plants but the operational history is too brief to evaluate the effectiveness of the new design Another way of improving the performance of the valves is to reduce the number of challenges to the valves. This may be done by the methods described above or by other means. The feasibility and contraindications of reducing the number of challenges to the valves by the various methods should be studied. Those changes which are shown to decrease the number of challenges without compromising the performance of the valves or other systems should be implemented.

The failure of an SRV to reclose will be the most probable cause of a small-break LOCA. Based on the above guidance and clarification, results of a detailed evaluation should be submitted to the staff. The licensee shall document the proposed system changes for staff approval before implementation.

Applicability

This requirement applies to all operating BWRs and BWR operating license applicants.

Implementation

Results of the evaluation shall be submitted by April 1, 1981 for staff review. The actual modification shall be accomplished during the next scheduled refueling outage following staff approval or no later than 1 year following staff approval. Modification to be implemented should be documented at the time of implementation.

Type of Review

A preimplementation review will be performed.

Documentation Required

By April 1, 1981, licensees must submit the results of the feasibility study for reducing SRV challenges and propose any necessary modifications for reducing SRV challenges.

Technical Specification Changes Required

Modification may include testing frequency or leakage criteria which may require technical specification changes.

Reference

NUREG-0625, Recommendations A-2.8, F-3.4

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

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 5 years of operation. The report should also include the causes of the outages (i.e., controller failure, spurious isolation).

Changes to Previous Requirements and Guidance

This clarification adds the requirement to propose changes that will improve and control availability.

Clarification

The present technical specifications contain limits on allowable outage times for ECC systems and components. However, there are no cumulative outage time limitations on these same systems. It is possible that ECC equipment could meet present technical specification requirements but have a high unavailability because of frequent outages within the allowable technical specifications.

The licensees should submit a report detailing outage dates and length of outages for all ECC systems for the last 5 years of operation, including causes of the outages. This report will provide the staff with a quantification of historical unreliability due to test and maintenance outages, which will be used to determine if a need exists for cumulative outage requirements in the technical specifications.

Based on the above guidance and clarification, a detailed report should be submitted. The report should contain (1) outage dates and duration of outages; (2) cause of the outage; (3) ECC systems or components involved in the outage; and (4) corrective action taken. Test and maintenance outages should be included in the above listings which are to cover the last 5 years of operation. The licensee should propose changes to improve the availability of ECC equipment, if needed.

Applicant for an operating license shall establish a plan to meet these requirements.

Applicability

This requirement applies to all operating reactors and applicants for operating license.

Implementation

Licensees should submit detailed report by January 1, 1981.

Applicants for operating license should submit their plan for data collection in accordance with the review schedule for licensing.

Type of Review

A postimplementation review will be performed.

Documentation Required

(1) Licensees shall submit a report containing the items noted in the above sections.

(2) Licensees shall submit suggested changes to improve the availability of ECC equipment, if needed.

Technical Specification Changes Required

Changes depend on results of the licensee study.

References

NUREG-0626, Recommendation A.6

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

II.K.3.18 MODIFICATION OF AUTOMATIC DEPRESSURIZATION SYSTEM LOGIC--FEASIBILITY FOR INCREASED DIVERSITY FOR SOME EVENT SEQUENCES

Position

The automatic depressurization system (ADS) actuation logic should be modified to eliminate the need for manual actuation to assure adequate core cooling. A feasibility and risk assessment study is required to determine the optimum approach. One possible scheme that should be considered is ADS actuation on low reactor-vessel water level provided no high-pressure coolant injection (HPCI) or high-pressure coolant system (HPCS) flow exists and a low-pressure emergency core cooling (ECC) system is running. This logic would complement, not replace, the existing ADS actuation logic.

Changes to Previous Requirements and Guidance

The schedule has been changed to accommodate the vendor-projected completion date and staff review of a very complex change.

Clarification

No further clarification is required.

Applicability

This requirements applies to all operating boiling-water reactors (BWRs) and BWR applicants for operating license.

Implementation

Operating Reactors--The feasibility study shall be completed by April 1, 1981. Proposed modifications shall be submitted by April 1, 1982. The licensee will implement modifications at the next refueling outage following staff approval of the design unless this outage is scheduled within 6 months of the approval date. In this event, modifications will be completed during the following refueling outage.

Operating License Applicants--All applicants for operating license should submit documentation 1 year prior to the expected issuance of an operating license or 1 year prior to the listed implementation date, whichever is later.

Type of Review

A preimplementation review of modifications will be performed.

Documentation Required

Operating Reactors--The Licensee shall provide results of feasibility study to NRC staff by April 1, 1981. Licensee shall describe the proposed modifications for staff approval by April 1, 1982.

Operating License Applicants--Applicants for operating license shall provide results of feasibility study 1 year prior to issuance of operating license. A description of the proposed modification for staff approval is required 4 months prior to issuance of an operating license.

Technical Specification Changes Required

Changes to technical specifications will be required.

Reference

NUREG-0626, Recommendation A.7

Letter from D. G. Eisenhut, NRC, to All Licensees, dated May 7, 1980.

II.K.3.19 INTERLOCK ON RECIRCULATION PUMP LOOPS

Position

Interlocks should be installed on nonjet pump plants (other than Humboldt Bay) to assure that at least two recirculation loops are open for recirculation flow for modes other than cold shutdown. This is to assure that the level measurements in the downcomer region are representative of the level in the core region.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

No clarification is required.

Applicability

This requirement applies to all operating nonjet-pump boiling-water reactors (BWRs), except for Humboldt Bay.

Implementation

For operating reactors, these requirements will be completed by July 1, 1981.

Type of Review

A postimplementation review of modifications will be performed.

Documentation Required

Licensees shall submit sufficient documentation by July 1, 1981 to support a reasonable assurance finding by the NRC that the modifications, as implemented, have resulted in satisfying the "Position" statement above.

Technical Specification Changes Required

Changes to technical specifications will be required.

Reference

NUREG-0626, Recommendation A.8

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

II.K.3.20 LOSS OF SERVICE WATER FOR BIG ROCK POINT

Position

The service water system for Big Rock Point has only one cooling train and is powered from normal alternating current power. The Big Rock Point licensee should verify the acceptability of the consequences of a loss-of-service-water supply to the essential plant components in the event of a loss of offsite power.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

Licensee will be required to submit an evaluation showing the acceptability of the consequences of a loss of service water to the essential plant components in the event of a loss of offsite power. The staff will review the licensee's submittal in order to determine whether plant modifications or procedural modifications will be required.

Applicability

This requirement applies only to Big Rock Point.

Implementation

For Big Rock Point this requirement will be completed by July 1, 1981.

Type of Review

A postimplementation review will be performed.

Documentation Required

The licensee shall submit an evaluation documenting the acceptability of the consequences of a loss of service water to the essential plant components in the event of a loss of offsite power.

Technical Specification Changes Required

Changes to technical specifications may be required, depending upon modifications (if any).

Reference

NUREG-0626, Recommendation A.9

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

II.K.3.21 RESTART OF CORE SPRAY AND LOW-PRESSURE COOLANT-INJECTION SYSTEMS

Position

The core-spray and low-pressure, coolant-injection (LPCI) system flow may be stopped by the operator. These systems will not restart automatically on loss of water level if an initiation signal is still present. The core spray and LPCI system logic should be modified so that these systems will restart, if required, to assure adequate core cooling. Because this design modification affects several core-cooling modes under accident conditions, a preliminary design should be submitted for staff review and approval prior to making the actual modification.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

Modification of system design should be made in accordance with those requirements set forth in Sections 4.12, 4.13, and 4.16 of IEEE Standard 279-1971 with regard to protective function bypasses and completion of protective action once initiated.

Applicability

This requirement applies to all BWR operating reactors and applicants for BWR operating license.

Implementation

Operating Reactors--Analysis and proposed design modifications shall be completed by January 1, 1981. Licensee shall implement modifications at the next refueling outage following staff approval of the design, unless this outage is scheduled within 6 months of the approval date. In this event, modifications will be completed during the following refueling outage.

Operating License Applicants--All applicants for operating license should submit documentation 4 months prior to the expected issuance of an operating license or 4 months prior to the listed implementation date, whichever is later.

Type of Review

A preimplementation review will be performed.

Documentation Required

Each licensee or applicant for operating license shall submit proposed design modifications and supporting analysis that will contain sufficient information to support a reasonable assurance finding by the NRC that the above position is met. The documentation should include as a minimum:

(1) A discussion of the design with respect to the above paragraphs of IEEE 279-1971;

(2) Support information including system design description, logic diagrams, electrical schematics, piping and instrument diagrams, test procedures, and technical specifications; and

(3) Sufficient documentation to demonstrate that the systems, as modified, would not degrade proper system functions.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0626, Recommendation A-10

IEEE Standard 279-1971, Section 4.12, 4.13, and 4.16.

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

II.K.3.22 AUTOMATIC SWITCHOVER OF REACTOR CORE ISOLATION COOLING SYSTEM SUCTION--VERIFY PROCEDURES AND MODIFY DESIGN

Position

The reactor core isolation cooling (RCIC) system takes suction from the condensate storage tank with manual switchover to the suppression pool when the condensate storage tank level is low. This switchover should be made automatically. Until the automatic switchover is implemented, licensees should verify that clear and cogent procedures exist for the manual switchover of the RCIC system suction from the condensate storage tank to the suppression pool.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements issued in the letter of May 7, 1980.

Clarification

No further clarification is required at this time.

Applicability

This requirement applies to all operating boiling-water reactors (BWRs) and applicants for operating license with a reactor core isolation cooling (RCIC) system.

Implementation

Operating Reactors--Procedures shall be verified by January 1, 1981. Design shall be modified by January 1, 1982.

Operating License Applicants--All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Verify procedures - January 1, 1981
Modify design - January 1, 1982

Type of Review

A postimplementation review of modifications will be performed.

Documentation Required

Operating Reactors--Licensee shall document procedure verification by January 1, 1981. Licensee shall submit supporting analysis and implemented design changes by January 1, 1982 and provide sufficient supporting evaluation to demonstrate that the system, as modified, will not degrade proper system function.

Operating License Applicants--Submit appropriate verification in accordance with the review schedule for licensing.

Technical Specification Changes Required

Changes to technical specifications will be required.

Reference

NUREG-0626, Recommendation B.1

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

II.K.3.24 CONFIRM ADEQUACY OF SPACE COOLING FOR HIGH-PRESSURE COOLANT INJECTION AND REACTOR CORE ISOLATION COOLING SYSTEMS

Position

Long-term operation of the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems may require space cooling to maintain the pump-room temperatures within allowable limits. Licensees should verify the acceptability of the consequences of a complete loss of alternating-current power. The RCIC and HPCI systems should be designed to withstand a complete loss of offsite alternating-current power to their support systems, including coolers, for at least 2 hours.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

No clarification is required.

Applicability

This requirement applies to all operating boiling-water reactors (BWRs) and BWR operating license applicants with RCIC and HPCI systems.

Implementation

Operating Reactors--For operating reactors, these requirements will be completed by January 1, 1982.

Operating License Applicants--All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Type of Review

A postimplementation review of modifications for operating reactors will be performed.

Documentation Required

Operating Reactors--Licensee should submit results of verification tests and modifications (if needed) by January 1, 1982.

Operating License Applicants--All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Applicants for operating licenses shall submit the evaluation and proposals by January 1, 1982 or no later than 6 months prior to expected issuance of the staff safety evaluation report in support of license issuance whichever is later.

Technical Specification Changes Requied

Changes to technical specifications will be required.

Reference

NUREG-0626, Recommendation B.3

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

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

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 2 hours. Adequacy of the seal design should be demonstrated.

Changes to Previous Requirements and Guidance

The evaluation and proposed modifications shall be submitted by July 1, 1981. The May 7, 1980 letter called for modifications by January 1, 1982. This clarification adds a documentation requirement for the evaluation to be submitted by July 1, 1981. The modification date remains unchanged. Additionally, this task has changed to include Westinghouse and Combustion Engineering operating reactors and operating reactor applicants.

Clarification

The intent of this position is to prevent excessive loss of reactor coolant system (RCS) inventory following an anticipated operational occurrence. Loss of ac power for this case is construed to be loss of offsite power. If seal failure is the consequence of loss of cooling water to the reactor coolant pump (RCP) seal coolers for 2 hours, due to loss of offsite power, one acceptable solution would be to supply emergency power to the component cooling water pump. This topic is addressed for Babcock and Wilcox (B&W) reactors in Section II.K.2.16.

Application

This requirement applies to all BWR, Westinghouse and Combustion Engineering operating reactors and applicants for operating license.

Implementation

For BWR operating reactors the evaluation and proposed modifications shall be submitted by July 1, 1981 and modifications shall be completed by January 1, 1982. Westinghouse and Combustion Engineering operating reactors shall submit the evaluation and proposed modifications by January 1, 1982 and complete modifications by July 1, 1982.

Type of Review

A preimplementation review of modifications will be performed.

Documentation Required

BWR licensees and Westinghouse and Combustion Engineering licensees shall provide results of evaluation and proposed modifications by July 1, 1981 and January 1, 1982, respectively.

Applicants for operating licenses shall submit the evaluation and proposals by January 1, 1982 or no later than 6 months prior to expected issuance of the staff safety evaluation report in support of license issuance, whichever is later.

Technical Specification Changes Required

Changes to technical specifications will not be required.

Reference

NUREG-0626, Recommendation B.4

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

II.K.3.27 PROVIDE COMMON REFERENCE LEVEL FOR VESSEL LEVEL INSTRUMENTATION

Position

Different reference points of the various reactor vessel water level instruments may cause operator confusion. Therefore, all level instruments should be referenced to the same point. Either the bottom of the vessel or the top of the active fuel are reasonable reference points.

Changes to Previous Requirements and Guidance

The submittal date has been extended from October 1, 1980 to January 1, 1981.

Clarification

No further clarification is required at this time.

Applicability

This requirement applies to all operating BWRs and applicants for operating license.

Implementation

Operating Reactors--These requirements will be completed by July 1, 1981.

Operating License Applicants--All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Type of Review

A postimplementation review for operating reactors will be performed.

Documentation Required

Operating Reactors--The licensee shall implement actions and submit documentation of the modifications by January 1, 1981.

Operating License Applicants--All applicants for an operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0626, Recommendation B.6

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

II.K.3.28 VERIFY QUALIFICATION OF ACCUMULATORS ON AUTOMATIC DEPRESSURIZATION SYSTEM VALVES

Position

Safety analysis reports claim that air or nitrogen accumulators for the automatic depressurization system (ADS) valves are provided with sufficient capacity to cycle the valves open five times at design pressures. GE has also stated that the emergency core cooling (ECC) systems are designed to withstand a hostile environment and still perform their function for 100 days following an accident. Licensee should verify that the accumulators on the ADS valves meet these requirements, even considering normal leakage. If this cannot be demonstrated, the licensee must show that the accumulator design is still acceptable.

Changes to Previous Requirements and Guidance

No changes have been made to the previous requirement as specified in the letter from D. G. Eisenhut dated May 7, 1980 to all operating reactor licensees and in NUREG-0626.

Clarification

The ADS valves, accumulators, and associated equipment and instrumentation must be capable of performing their functions during and following exposure to hostile environments and taking no credit for nonsafety-related equipment or instrumentation. Additionally, air (or nitrogen) leakage through valves must be accounted for in order to assure that enough inventory of compressed air is available to cycle the ADS valves.

Applicability

This requirement applies to all operating BWR plants and all applicants for operating license.

Implementation

This requirement shall be completed by January 1, 1982.

Type of Review

A review of evaluation results or a postimplementation review of any accumulator design changes in operating reactors will be performed.

Documentation Required

All operating reactor licensees shall submit evaluation results for staff review to show that accumulators are qualified and shall implement actions, as required, by January 1, 1982. All applicants for operating license shall submit documentation 4 months before the expected issuance of the staff safety evaluation report for an operating license or 4 months before the listed implementation date, whichever is later.

Technical Specification Changes Required

Changes to technical specifications will be required.

Reference

NUREG-0626, Section A.2-15

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

II.K.3.29 STUDY TO DEMONSTRATE PERFORMANCE OF ISOLATION CONDENSERS WITH NONCONDENSIBLES

Position

If natural circulation plays an important role in depressurizing the system (e.g., in the use of isolation condensers), then the various modes of two-phase-flow natural circulation, including noncondensibles, which may play a significant role in plant response following a small-break loss-of-coolant accident (LOCA) should be demonstrated.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

Licensees should provide confirmatory verification, using applicable experimental data, of the analysis models used to calculate the various modes of single and two-phase natural circulation predicted to occur in their plants during transient and accident events.

Applicability

This requirement applies to all operating boiling water reactors (BWRs) with isolation condensers.

Implementation

For operating reactors, these requirements will be completed by April 1, 1981.

Type of Review

A postimplementation review will be performed.

Documentation Required

Licensees shall provide results of evaluation to NRC staff by April 1, 1981.

Technical Specification Changes Required

Changes to technical specifications will not be required.

Reference

NUREG-0626, Recommendation B.13

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

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

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 10 CFR 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.

Changes to Previous Requirements and Guidance

The changed requirement (1) allows for justification of acceptability of present small-break LOCA models by comparison with test date, and (2) requests each licensee to outline scope and schedule for model revision or comparison with test data by late fall, 1980. The original requirement did not allow provision for showing acceptability of present models by comparison with plant data.

Clarification

As a result of the accident at TMI-2, the Bulletins and Orders Task Force was formed within the Office of Nuclear Reactor Regulation. This task force was charged, in part, to review the analytical predictions of feedwater transients and small-break LOCAs for the purpose of assuring the continued safe operation of all operating reactors, including a determination of acceptability of emergency guidelines for operators.

As a result of the task force reviews, a number of concerns were identified regarding the adequacy of certain features of small-break LOCA models, particularly the need to confirm specific model features (e.g., condensation heat transfer rates) against applicable experimental data. These concerns, as they applied to each light-water reactor (LWR) vendor's models, were documented in the task force reports for each LWR vendor. In addition to the modeling concerns identified, the task force also concluded that, in light of the TMI-2 accident, additional systems verification of the small-break LOCA model as required by II.4 of Appendix K to 10 CFR 50 was needed. This included providing predictions of Semiscale Test S-07-10B, LOFT Test (L3-1), and providing experimental verification of the various modes of single-phase and two-phase natural circulation predicted to occur in each vendor's reactor during small-break LOCAs.

Based on the cumulative staff requirements for additional small-break LOCA model verification, including both integral system and separate effects verification, the staff considered model revision as the appropriate method for reflecting any potential upgrading of the analysis methods.

The purpose of the verification was to provide the necessary assurance that the small-break LOCA models were acceptable to calculate the behavior and consequences of small primary system breaks. The staff believes that this assurance can alternatively be provided, as appropriate, by additional justification of the acceptability of present small-break LOCA models with regard to specific staff concerns and recent test data. Such justification could supplement or supersede the need for model revision.*

The specific staff concerns regarding small-break LOCA models are provided in the analysis sections of the B&O Task Force reports for each LWR vendor, (NUREG-0635, -0565, -0626, -0611, and -0623). These concerns should be reviewed in total by each holder of an approved emergency core cooling system (ECCS) model and addressed in the evaluation as appropriate.

The recent tests include the entire Semiscale small-break test series and LOFT Tests (L3-1) and (L3-2). The staff believes that the present small-break LOCA models can be both qualitatively and quantitatively assessed against these tests. Other separate effects tests (e.g., ORNL core uncovery tests) and future tests, as appropriate, should also be factored into this assessment.

Based on the preceding information, a detailed outline of the proposed program to address this issue should be submitted. In particular, this submittal should identify (1) which areas of the models, if any, the licensee intends to upgrade, (2) which areas the licensee intends to address by further justification of acceptability, (3) test data to be used as part of the overall verification/upgrade effort, and (4) the estimated schedule for performing the necessary work and submitting this information for staff review and approval.

Applicability

This requirement applies to all operating reactors and applicants for operating license.

Implementation

Detailed outline of the scope and schedule for meeting this requirement should be submitted by each licensee and applicant by November 15, 1980. This submittal will form the basis for a meeting with the staff to review and approve the overall plan. Meetings with the staff to review this submittal are expected for late fall 1980.

The additional information requested should be submitted by January 1, 1982. The plant-specific analyses using the revised models should be submitted by January 1, 1983, or one year after any model revisions are approved.

Type of Review

A postimplementation review of the schedule will be performed. A preimplementation review will be performed by the staff to approve the model and analyses.

*As an example, a model that presently does not properly account for horizontal countercurrent two-phase flow in the hot leg piping should either be revised to properly account for the phenomenon, or demonstrated to produce a conservative result for the entire spectrum of small breaks considered.

Documentation Required

(1) Licensees shall submit outline of program for model justification/revision by November 15, 1980.

(2) Licensees shall submit additional information for model justification and/or revised analysis model for staff approval by January 1, 1982.

(3) Licensees shall submit their plant-specific analyses using the revised models by January 1, 1983 or one year after any model revisions are approved.

(4) Applicants shall submit appropriate information in accordance with the licensing review schedule.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0565, Recommendation 2.2.2a

NUREG-0611, Recommendation 3.2.1a

NUREG-0623

NUREG-0626, Recommendation A.12

NUREG-0635, Recommendation 3.2.1.a and 3.2.5.a

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

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

Position

Plant-specific calculations using NRC-approved models for small-break loss-of-coolant accidents (LOCAs) as described in item II.K.3.30 to show compliance with 10 CFR 50.46 should be submitted for NRC approval by all licensees.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

See "Clarification" for item II.K.3.30.

Applicability

This requirement applies to all operating reactors and applicants for operating license.

Implementation

Calculations shall be submitted by January 1, 1983 or 1 year after staff approval of LOCA analysis models, whichever is later, only if model changes have been made.

Type of Review

A review for conformance with 10 CFR 50.46 limits will be performed.

Documentation Required

Operating Reactors--Licensee shall provide results of evaluation to staff, in accordance with the schedule as indicated above.

Operating License Applicants--All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Technical Specification Changes Required

Changes to technical specifications are to be determined.

References

NUREG-0565, Recommendation 2.2.2.b

NUREG-0611, Recommendation 3.2.1.b

NUREG-0626, Recommendations A.13 and B.10

NUREG-0635, Recommendation 3.2.1.b

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

II.K.3.44 EVALUATION OF ANTICIPATED TRANSIENTS WITH SINGLE FAILURE TO VERIFY NO FUEL FAILURE

Position

For anticipated transients combined with the worst single failure and assuming proper operator actions, licensees should demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery. Transients which result from a stuck-open relief valve should be included in this category.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

No further clarification is required at this time.

Applicability

This requirement applies to all operating boiling-water reactors (BWRs) and BWR license applicants.

Implementation

Operating Reactors--For operating reactors, these requirements will be completed by January 1, 1981.

Operating License Applicants--All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Type of Review

A postimplementation review will be performed.

Documentation Required

Operating Reactors--Licensee shall provide results of evaluation to staff by January 1, 1981.

Operating License Applicants--All applicants for operating license should submit documentation 4 months prior to the expected issuance of an the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Technical Specification Changes Required

Changes to technical specifications will be determined following review of evaluation.

Reference

NUREG-0626, Recommendation A.14

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

II.K.3.45 EVALUATION OF DEPRESSURIZATION WITH OTHER THAN AUTOMATIC DEPRESSURIZATION SYSTEM

Position

Analyses to support depressurization modes other than full actuation of the automatic depressurization system (ADS) (e.g., early blowdown with one or two safety relief valves (SRVs)) should be provided. Slower depressurization would reduce the possibility of exceeding vessel integrity limits by rapid cooldown.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

No further clarification is required at this time.

Applicability

This requirement applies to all operating boiling-water reactors (BWRs) and BWR license applicants.

Implementation

Operating Reactors--For operating reactors, these requirements will be completed by January 1, 1981.

Operating License Applicants--All applicants for operating license should submit documentation 4 months prior to the expected issuance of an the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Type of Review

A postimplementation review will be performed.

Documentation Required

Operating Reactors--Licensee shall provide results of evaluation to staff by January 1, 1981.

Operating License Applicants--All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Technical Specification Changes Required

Changes to technical specifications will be determined following review of evaluation.

Reference

NUREG-0626, Recommendation A.15

II.K.3.57 IDENTIFY WATER SOURCES PRIOR TO ACTUATION OF AUTOMATIC DEPRESSURIZATION SYSTEM

Position

Emergency procedures should include verification that a source of cooling water, such as the core spray, low-pressure coolant injection (LPCI), or condensate systems, is available prior to manual actuation of the automatic depressurization system (ADS). Alternate water sources should be identified in the procedures, and reference should be made to procedures for startup and operation of systems that provide these sources. This is being implemented through the guidelines being developed to assure adequate core cooling.

Changes to Previous Requirements and Guidance

There are no changes to the previous position or requirements.

Clarification

Exceptions to the requirement that a source of cooling water be available prior to manual actuation of the ADS should be identified and justified.

Symptomatic guidelines have been developed by the BWR owners' group and are being implemented for trial use of the near-term operating licenses. Implementation of the symptomatic approach for operating reactors will be accomplished on a schedule compatible with that identified under item I.C.1.

Applicability

This requirement applies to all operating boiling-water reactors.

Implementation

Guidelines have been submitted by the BWR owners' group. Implementation will be consistent with item I.C.1. No additional licensee action is required until guidelines are approved by the staff.

Type of Review

A postimplementation review will be performed.

Documentation Required

The BWR owners' group has submitted guidelines that are being reviewed by the staff.

Technical Specification Changes Required

Changes to technical specifications will not be required.

Reference

NUREG-0626, Recommendation A.5

III.A.1.2 UPGRADE EMERGENCY SUPPORT FACILITIES

Additional clarification will be provided in the near future.

III.A.2 IMPROVING LICENSEE EMERGENCY PREPAREDNESS--LONG-TERM

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."

Changes to Previous Requirements and Guidance

The final regulations on emergency planning (45 FR 55401-55413) which become effective on November 3, 1980, require the submittal and implementation of the radiological emergency response plans of licensees and state and local entities within the plume exposure and ingestion emergency planning zones (EPZ) by January 2, 1981

NUREG-0654 has been revised to include changes developed from team reviews and comments obtained during the comment period.

The revised NUREG-0654 establishes the schedule for installation of meteorological equipment to meet a prescribed implementation date (also see proposed Revision 1 to Regulatory Guide 1.23). The NRC rule establishes July 1, 1981 as the date when the prompt notification capability is to be functional. Item III.A.1.2 establishes dates when emergency response facilities must be functional.

Clarification

In accordance with Task Action Plan item III.A.1.1, "Upgrade Emergency Preparedness," each nuclear power facility was required to immediately upgrade its emergency plans with criteria provided October 10, 1979, as revised by NUREG-0654 (FEMA-REP-1, issued for interim use and comment, January 1980). New plans were submitted by January 1, 1980, using the October 10, 1979 criteria. Reviews were started on the upgraded plans using NUREG-0654. Concomitant to these actions, amendments were developed to 10 CFR Part 50 and Appendix E to 10 CFR Part 50, to provide the long-term implementation requirements. These new rules were issued in the Federal Register on August 19, 1980, with an effective date of November 3, 1980. The revised rules delineate requirements for emergency preparedness at nuclear reactor facilities.

NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," provides detailed items to be included in the upgraded emergency plans and, along with the revised rules, provides for meteorological criteria, means for providing for a prompt notification to the population, and the need for emergency response facilities (see Item III.A.1.2).

Implementation of the new rules levied the requirement for the licensee to provide procedures implementing the upgraded emergency plans to the NRC for review. Publication of Revision 1 to NUREG-0654 (FEMA-REP-1) which incorporates the many public comments received is expected in October 1980. This is the document that will be used by NRC and FEMA in their evaluation of emergency plans submitted in accordance with the new NRC rules.

NUREG-0654, Revision 1; NUREG-0696, "Functional Criteria for Emergency Response Facilities;" and the amendments to 10 CFR Part 50 and Appendix E to 10 CFR Part 50 regarding emergency preparedness, provide more detailed criteria for emergency plans, design, and functional criteria for emergency response facilities and establishes firm dates for submission of upgraded emergency plans for installation of prompt notification systems. These revised criteria and rules supersede previous Commission guidance for the upgrading of emergency preparedness at nuclear power facilities.

Revision 1 to NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," provides meteorological criteria to fulfill, in part, the standard that "Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use" (see 10 CFR §50.47). The position in Appendix 2 to NUREG-0654 outlines four essential elements that can be categorized into three functions: measurements, assessment, and communications.

Proposed Revision 1 to Regulatory Guide 1.23, "Meteorological Measurements Programs in Support of Nuclear Power Plants," has been adopted to provide guidance criteria for the primary meteorological measurements program consisting of a primary system and secondary system(s) where necessary, and a backup system. Data collected from these systems are intended for use in the assessment of the offsite consequences of a radiological emergency condition.

Appendix 2 to NUREG-0654 delineates two classes of assessment capabilities to provide input for the evaluation of offsite consequences of a radiological emergency condition. Both classes of capabilities provide input to decisions regarding emergency actions. The Class A capability should provide information to determine the necessity for notification, sheltering, evacuation, and, during the initial phase of a radiological emergency, making confirmatory radiological measurements. The Class B capability should provide information regarding the placement of supplemental meteorological monitoring equipment, and the need to make additional confirmatory radiological measurements. The Class B capability shall identify the areas of contaminated property and foodstuff requiring protective measures and may also provide information to determine the necessity for sheltering and evacuation.

Proposed Revision 1 to Regulatory Guide 1.23 outlines the set of meteorological measurements that should be accessible from a system that can be interrogated; the meteorological data should be presented in the prescribed format. The results of the assessments should be accessible from this system; this information should incorporate human-factors engineering in its display to convey the essential information to the initial decision makers and subsequent management team. An integrated system should allow the eventual incorporation of effluent monitoring and radiological monitoring information with the environmental transport to provide direct dose consequence assessments.

Requirements of the new emergency-preparedness rules under paragraphs 50.47 and 50.54 and the revised Appendix E to Part 50 taken together with NUREG-0654 Revision 1 and NUREG-0696, when approved for issuance, go beyond the previous requirements for meteorological programs. To provide a realistic time frame for implementation, a staged schedule has been established with compensating actions provided for interim measures.

Applicability

This requirement applies to all operating reactors and applicants for operating licenses.

Implementation

Schedule for Operating Reactors -- For operating reactors the following implementation milestones shall be met to address the four basic elements of the introduction to Appendix 2 to NUREG-0654.

Milestones are numbered and tagged with the following code; a-date, b-activity, c-minimum acceptance criteria. They are as follows:

(1) a. January 2, 1981

b. Submittal of radiological emergency response plans

c. A description of the plan to include elements of NUREG-0654, Revision 1, Appendix 2

(2) a. March 1, 1981

b. Submittal of implementing procedures

c. Methods, systems, and equipment to assess and monitor actual or potential offsite consequences of a radiological emergency condition shall be provided

(3) a. April 1, 1981

b. Implementation of radiological emergency response plans

c. Four elements of Appendix 2 to NUREG-0654 with the exception of the Class B model of element 3, or

Alternative to item (3) requiring compensating actions:

A meteorological measurements program which is consistent with the existing technical specifications as the baseline or an element 1 program and/or element 2 system of Appendix 2 to NUREG-0654, or two independent element 2 systems shall provide the basic meteorological parameters (wind direction and speed and an indicator of atmospheric stability) on display in the control room. An operable dose calculational methodology (DCM) shall be in use in the control room and at appropriate emergency response facilities.

The following compensating actions shall be taken by the licensee for this alternative:

(i) if only element 1 or element 2 is in use:

- The licensee (the person who will be responsible for making offsite dose projections) shall check communications with the cognizant National Weather Service (NWS) first order station and NWS forecasting station on a monthly basis to ensure that routine meteorological observations and forecasts can be accessed.

- The licensee shall calibrate the meteorological measurements program at a frequency no less than quarterly and identify a readily available source of meteorological data (characteristic of site conditions) to which they can gain access during calibration periods.

- During conditions of measurements system unavailability, an alternate source of meteorological data which is characteristic of site conditions shall be identified to which the licensee can gain access.

- The licensee shall maintain a site inspection schedule for evaluation of the meteorological measurements program at a frequency no less than weekly.

- It shall be a reportable occurrence if the meteorological data unavailability exceeds the goals outlined in Proposed Revision 1 to Regulatory Guide 1.23 on a quarterly basis.

(ii) The portion of the DCM relating to the transport and diffusion of gaseous effluents shall be consistent with the characteristics of the Class A model outlined in element 3 of Appendix 2 to NUREG-0654.

(iii) Direct telephone access to the individual responsible for making offsite dose projections (Appendix E to 10 CFR Part 50(IV)(A)(4)) shall be available to the NRC in the event of a radiological emergency. Procedures for establishing contact and identification of contact individuals shall be provided as part of the implementing procedures.

This alternative shall not be exercised after July 1, 1982. Further, by July 1, 1981, a functional description of the upgraded programs (four elements) and schedule for installation and full operational capability shall be provided (see milestones 4 and 5).

(4) a. March 1, 1982

b. Installation of Emergency Response Facility hardware and software

c. Four elements of Appendix 2 to NUREG-0654, with exception of the Class B model of element 3.

(5) a. July 1, 1982

b. Full operational capability of milestone 4.

c. The Class A model (designed to be used out to the plume exposure EPZ) may be used in lieu of a Class B model out to the ingestion EPZ. Compensating actions to be taken for extending the application of the Class A model out to the ingestion EPZ include access to supplemental information (meso and synoptic scale) to apply judgment regarding intermediate and long-range transport estimates. The distribution of meteorological information by the licensee should be as follows by July 1, 1982:

Meteorological Information	CR	TSC	EOF	NRC and Emergency Response Organizations
Basic Met. Data (e.g., 1.97 Parameters)	X	X	X	X (NRC)
Full Met. Data (1.23 Parameters)		X	X	X
DCM (for Dose Projections)	X	X	X	X
Class A Model (to Plume Exposure EPZ)	X	X	X	X
Class B Model or Class A Model (to Ingestion EPZ)		X	X	X

(6) a. July 1, 1982 or at the time of the completion of milestone 5. whichever is sooner.

b. Mandatory review of the DCM by the licensee

c. Any DCM in use should be reviewed to ensure consistency with the operational Class A model. Thus, actions recommended during the initial phases of a radiological emergency would be consistent with those after the TSC and EOF are activated.

(7) a. September 1, 1982

b. Description of the Class B model provided to the NRC

c. Documentation of the technical bases and justification for selection of the type Class B model by the licensee with a discussion of the site-specific attributes.

(8) a. June 1, 1983

b. Full operational capability of the Class B model

c. Class B model of element 3 of Appendix 2 to NUREG-0654, Revision 1

Schedule for Near-Term Operating Licenses--For applicants for an operating license, at least milestones 1, 2, and 3 shall be met prior to the issuance of an operating license. Subsequent milestones shall be met by the same dates indicated for operating reactors. for the alternative to milestone 3, the meteorological measurements program shall be consistent with the NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 2.3.3 program as the baseline or element 1 and/or element 2 systems.

Type of Review

A postimplementation review will be performed for the April 1, 1981 requirement.

Documentation Required

Complete updated emergency plans shall be provided by January 2, 1981 and complete implementing procedures shall be submitted by March 1, 1981.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-75/087

NUREG-0654 (FEMA-REP-1), Revision 1

NUREG-0696

Regulatory Guide 1.23, Proposed Revision 1

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

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.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements

Clarification

Applicants shall provide a summary description, together with initial leak-test results, of their program to reduce leakage from systems outside containment that would or could contain primary coolant or other highly radioactive fluids or gases during or following a serious transient or accident.

(1) Systems that should be leak tested are as follows (any other plant system which has similar functions or postaccident characteristics even though not specified herein, should be included):

Residual heat removal (RHR)

Containment spray recirculation

High-pressure injection recirculation

Containment and primary coolant sampling

Reactor core isolation cooling

Makeup and letdown (PWRs only)

Waste gas (includes headers and cover gas system outside of containment in addition to decay or storage system)

Include a list of systems containing radioactive materials which are excluded from program and provide justification for exclusion.

(2) Testing of gaseous systems should include helium leak detection or equivalent testing methods.

(3) Should consider program to reduce leakage potential release paths due to design and operator deficiencies as discussed in our letter to all operating nuclear power plants regarding North Anna and related incidents, dated October 17, 1979.

Applicability

This requirement applies to all operating license applicants. Operating reactors satisfied the requirements of NUREG-0578, Recommendation 2.1.6a (Systems Integrity), by providing information required by January 1, 1980.

Implementation

This requirement shall be implemented by applicants for operating license prior to issuance of a full-power license.

Documentation Required

Applicants shall submit the information requested in the "Clarification" section of this position at least 4 months prior to issuance of a fuel-loading license.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.6.a

NUREG-0660, Item III.D.1.1

NUREG-0694, Part 2

Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Power Plants, dated October 17, 1979.

III.D.3.3 IMPROVED INPLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS

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.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

Effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments using sample media that will collect iodine selectively over xenon (e.g., silver zeolite) for the following reasons:

(1) The physical size of the auxiliary and/or fuel handling building precludes locating stationary monitoring instrumentation at all areas where airborne iodine concentration data might be required.

(2) Unanticipated isolated "hot spots" may occur in locations where no stationary monitoring instrumentation is located.

(3) Unexpectedly high background radiation levels near stationary monitoring instrumentation after an accident may interfere with filter radiation readings.

(4) The time required to retrieve samples after an accident may result in high personnel exposures if these filters are located in high-dose-rate areas.

After January 1, 1981, each applicant and licensee shall have the capability to remove the sampling cartridge to a low-background, low-contamination area for further analysis. Normally, counting rooms in auxiliary buildings will not have sufficiently low backgrounds for such analyses following an accident. In the low background area, the sample should first be purged of any entrapped noble gases using nitrogen gas or clean air free of noble gases. The licensee shall have the capability to measure accurately the iodine concentrations present on these samples under accident conditions. There should be sufficient samplers to sample all vital areas.

For applicants with fuel-loading dates prior to January 1, 1981, provide by fuel loading (until January 1, 1981) the capability to accurately detect the presence of iodine in the region of interest following an accident. This can be accomplished by using a portable or cart-mounted iodine sampler with attached single-channel analyzer (SCA). The SCA window should be calibrated to the 365 KeV of iodine-131 using the SCA. This will give an initial conservative estimate of presence of iodine and can be used to determine if respiratory protection is required. Care must be taken to assure that the counting system is not saturated as a result of too much activity collected on the sampling cartridge.

APPLICABILITY

This requirement applies to all operating reactors and all applicants for an operating license.

IMPLEMENTATION

Applicants for fuel-loading license prior to January 1, 1981 shall meet position 2 prior to fuel loading. Licensees and applicants shall meet position 1 by January 1, 1981, or prior to licensing, whichever is later.

Type of Review

A postimplementation review will be performed.

Documentation Required

For applicants for an operating license, provide a description of the in-plant airborne radioiodine sampling and analysis systems specifying the number and types of samplers, sample media, sample flushing methods, and sample analysis equipment type and location.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0578 Recommendation 2.1.8.c

NUREG-0660, Item III.D.3.3

Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Power Plants, dated September 13, 1979.

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

III.D.3.4 CONTROL-ROOM HABITABILITY REQUIREMENTS

Position

In accordance with Task Action Plan item III.D.3.4 and control room habitability, 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 safely operated or shut down under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50).

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

(1) All licensees must make a submittal to the NRC regardless of whether or not they met the criteria of the referenced Standard Review Plans (SRP) sections. The new clarification specifies that licensees that meet the criteria of the SRPs should provide the basis for their conclusion that SRP 6.4 requirements are met. Licensees may establish this basis by referencing past submittals to the NRC and/or providing new or additional information to supplement past submittals.

(2) All licensees with control rooms that meet the criteria of the following sections of the Standard Review Plan:

2.2.1-2.2.2	Identification of Potential Hazards in Site Vicinity
2.2.3	Evaluation of Potential Accidents;
6.4	Habitability Systems

shall report their findings regarding the specific SRP sections as explained below. The following documents should be used for guidance:

(a) Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of Regulatory Power Plant Control Room During a Postulated Hazardous Chemical Release";

(b) Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accident Chlorine Release"; and,

(c) K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," 13th AEC Air Cleaning Conference, August 1974.

Licensees shall submit the results of their findings as well as the basis for those findings by January 1, 1981. In providing the basis for the habitability finding, licensees may reference their past submittals. Licensees should, however, ensure that these submittals reflect the current facility design and that the information requested in Attachment 1 is provided.

(3) All licensees with control rooms that do not meet the criteria of the above-listed references, Standard Review Plans, Regulatory Guides, and other references.

These licensees shall perform the necessary evaluations and identify appropriate modifications.

Each licensee submittal shall include the results of the analyses of control room concentrations from postulated accidental release of toxic gases and control room operator radiation exposures from airborne radioactive material and direct radiation resulting from design-basis accidents. The toxic gas accident analysis should be performed for all potential hazardous chemical releases occurring either on the site or within 5 miles of the plant-site boundary. Regulatory Guide 1.78 lists the chemicals most commonly encountered in the evaluation of control room habitability but is not all inclusive.

The design-basis-accident (DBA) radiation source term should be for the loss-of-coolant accident LOCA containment leakage and engineered safety feature (ESF) leakage contribution outside containment as described in Appendix A and B of Standard Review Plan Chapter 15.6.5. In addition, boiling-water reactor (BWR) facility evaluations should add any leakage from the main steam isolation valves (MSIV) (i. e., valve-stem leakage, valve seat leakage, main steam isolation valve leakage control system release) to the containment leakage and ESF leakage following a LOCA. This should not be construed as altering the staff recommendations in Section D of Regulatory Guide 1.96 (Rev. 2) regarding MSIV leakage-control systems. Other DBAs should be reviewed to determine whether they might constitute a more-severe control-room hazard than the LOCA.

In addition to the accident-analysis results, which should either identify the possible need for control-room modifications or provide assurance that the habitability systems will operate under all postulated conditions to permit the control-room operators to remain in the control room to take appropriate actions required by General Design Criterion 19, the licensee should submit sufficient information needed for an independent evaluation of the adequacy of the habitability systems. Attachment 1 lists the information that should be provided along with the licensee's evaluation.

Applicability

This requirement applies to all operating reactors and operating license applicants.

Implementation

Licensees shall submit their responses to this request on or before January 1, 1981. Applicants for operating licenses shall submit their responses prior to issuance of a full-power license. Modifications needed for compliance with the control-room habitability requirements specified in this letter should be identified, and a schedule for completion of the modifications should be provided. Implementation of such modifications should be started without awaiting the results of the staff review. Additional needed modifications, if any, identified by the staff during its review will be specified to licensees

Type of Review

A postimplementation review will be performed.

Documentation Required

By January 1, 1981 licensees shall provide the information described in Attachment 1. Applicants for an operating license shall submit their responses prior to full-power licensing.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0660, Item III.D.3.4.

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

III.D.3.4. ATTACHMENT 1. INFORMATION REQUIRED FOR CONTROL-ROOM HABITABILITY EVALUATION

(1) Control-room mode of operation, i.e., pressurization and filter recirculation for radiological accident isolation or chlorine release

(2) Control-room characteristics

(a) air volume control room

(b) control-room emergency zone (control room, critical files, kitchen, washroom, computer room, etc.)

(c) control-room ventilation system schematic with normal and emergency air-flow rates

(d) infiltration leakage rate

(e) high efficiency particulate air (HEPA) filter and charcoal adsorber efficiencies

(f) closest distance between containment and air intake

(g) layout of control room, air intakes, containment building, and chlorine, or other chemical storage facility with dimensions

(h) control-room shielding including radiation streaming from penetrations, doors, ducts, stairways, etc.

(i) automatic isolation capability-damper closing time, damper leakage and area

(j) chlorine detectors or toxic gas (local or remote)

(k) self-contained breathing apparatus availability (number)

(l) bottled air supply (hours supply)

(m) emergency food and potable water supply (how many days and how many people)

(n) control-room personnel capacity (normal and emergency)

(o) potassium iodide drug supply

(3) Onsite storage of chlorine and other hazardous chemicals

(a) total amount and size of container

(b) closest distance from control-room air intake

(4) Offsite manufacturing, storage, or transportation facilities of hazardous chemicals

(a) identify facilities within a 5-mile radius;

(b) distance from control room

(c) quantity of hazardous chemicals in one container

(d) frequency of hazardous chemical transportation traffic (truck, rail, and barge)

(5) Technical specifications (refer to standard technical specifications)

(a) chlorine detection system

(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.

APPENDIX A

KEY TO REFERENCES

The final paragraph of each clarification item lists the reference materials related to that item. Those listed as NUREG-XXXX are NRC documents available for purchase from: GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 and the National Technical Information Service, Springfield, Virginia 22161. They are also available for inspection and copying for a fee in the NRC Public Document Room at 1717 H Street, N.W., Washington, D. C. to avoid frequent repetition within this document, the NUREG reports are listed only by number. A complete list with title and date of publication follows:

WASH-1400 (NUREG-75/014), "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," Executive Summary, Main Report, Appendices I-II, U.S. Nuclear Regulatory Commission, December 1975.

NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," U.S. Nuclear Regulatory Commission, 1975 (available only from the National Technical Information Service, Springfield, Virginia 22161).

NUREG-0565, "Staff Report on the Generic Evaluation of Small-Break Loss-of-Coolant Accident Behavior for Babcock and Wilcox Operating Plants," U.S. Nuclear Regulatory Commission, January 1980.

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

NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report," U.S. Nuclear Regulatory Commission, August 1979.

NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," U.S. Nuclear Regulatory Commission, December 1979.

NUREG-0611, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants," U.S. Nuclear Regulatory Commission, January 1980.

NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small-Break Loss-of-Coolant Accidents in Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, November 1979.

NUREG-0625, "Report of the Siting Policy Task Force," U.S. Nuclear Regulatory Commission, August 1979.

NUREG-0626, "Staff Report on the Generic Assessment of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Boiling Water Reactors Designed by the General Electric Company," U.S. Nuclear Regulatory Commission, January 1980.

NUREG-0635, "Generic Assessment of Small-Break Loss-of-Coolant Accidents in Combustion Engineering Designed Operating Plants," U.S. Nuclear Regulatory Commission, January 1980.

NUREG-0645, "Final Report of Bulletins and Orders Task Force of the Office of Nuclear Reactor Regulation," Vols. 1 and 2, U.S. Nuclear Regulatory Commission, January 1980.

NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, January 1980.

NUREG-0660, Vols. 1 and 2, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980; Revision 1, August 1980.

NUREG-0667, "Transient Response of Babcock & Wilcox Designed Reactors," U.S. Nuclear Regulatory Commission, May 1980.

NUREG-0694, "TMI-Related Requirements for New Operating Licenses," U.S. Nuclear Regulatory Commission, June 1980.

NUREG-0696 (draft), "Functional Criteria for Emergency Response Facilities," U.S. Nuclear Regulatory Commission, July 1980.

NUREG-0700, "Guidelines for the Design Review of Nuclear Power Plant Control Rooms," U.S. Nuclear Regulatory Commission, to be published.

NUREG-CR-1580 (draft), "Human Engineering Guide for Control Room Evaluation," Essex Corporation, July 1980.

The following NRC letters are available for inspection and copy for a fee in the NRC Public Document Room at 1717 H Street, N.W., Washington, D. C.:

8/21/79 Letter from D. F. Ross, Jr., NRC, to All B&W Operating Plants, dated August 21, 1979, Subject: Identification and Resolution of Long-Term Generic Issues Related to the Commission Orders of May 1979.

9/13/79 Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Power Plants, dated September 13, 1979, Subject: Followup Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident.

9/27/79 Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, dated September 27, 1979, Subject: Followup Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident.

10/10/79 Letter from D. G. Eisenhut, NRC, to All Power Reactor Licensees, dated October 10, 1979, Subject: Emergency Planning.

10/17/79 Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Power Plants, dated October 17, 1979, Subject: Radioactive Release at North Anna Unit 1 and Lessons Learned.

10/30/79 Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979, Subject: Discussion of Lessons Learned Short-Term Requirements.

11/7/79 Letter from R. W. Reid, NRC, to All B&W Operating Plants, dated Novemeber 7, 1979, Subject: Request for Additional Information - BAW Report 1564, "Integrated Control System Reliability Analysis."

7/31/80 Letter from D. G. Eisenhut, NRC, to All Licensees and Applicants, dated July 31, 1980, Subject: Interim Criteria for Shift Staffing.

11/9/79 Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, dated November 9, 1979, Subject: Discussion of Lessons Learned Short-Term Requirements.

11/21/79 Letter from R. W. Reid, NRC, to All B&W Operating Plants, dated November 21, 1979, Subject: Request for Additional Information on Small-Break Loss-of-Coolant Accident.

12/20/79 Letter from R. W. Reid, NRC, to All B&W Licensees, dated December 20, 1979, Subject: Preliminary Design Approval for the Safety-Grade Anticipatory Reactor Trip (AFT) on Loss-of-Feedwater and Turbine Trip.

1/9/80 Letter from R. W. Reid, NRC to All B&W Operating Plants, dated January 9, 1980, Subject: Concern for Voiding During Transients on B&W Plants.

3/10/80 Letter from D. F. Ross, Jr., NRC, to All Pending W and C-E License Applicants, dated March 10, 1980, Subject: Actions Required from Operating License Applicants of Nuclear Steam Supply Systems Designed by W and C-E Resulting from the NRC Bulletins and Orders Task Force Review Regarding TMI-2 Accident.

3/28/80 Letter from H. R. Denton, NRC, to All Power Reactor Applicants and Licensees, dated March 28, 1980, Subject: Qualifications of Reactor Operators.

4/24/80 Letter from D. F. Ross, Jr., NRC, to All Pending B&W License Applicants, dated April 24, 1980, Subject: Actions Required from Operating License Applicants of Nuclear Steam Supply Systems Designed by B&W Resulting from the NRC Bulletins and Orders Task Force Review Regarding TMI-2 Accident.

4/25/80 Letter from D. G. Eisenhut, NRC, to All Power Reactor Licensees, dated April 25, 1980, Subject: Clarification of NRC Site Requirements for Emergency Response Facilities at Each Site.

5/7/80 Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980, Subject: Five Additional TMI-2 Related Requirements to Operating Reactors.

Documents with the following types of designation and other miscellaneous documents are available for inspection and copying for a fee in the NRC Public Document Room at 1717 H Street., N.W., Washington, D.C.:

Commission Order (CLI-80-21)

NRC Regulation (45 FR 55401-55413)

Inspection and Enforcement documents

Regulatory Guides

Standard Review Plan

Technical Specifications

Branch Technical Position

Staff Interim Position

Other documents that are national technical standards are available for inspection from public technical libraries:

ANSI Standards

IEEE Standards

Code of Federal Regulations

APPENDIX B

DESIGN AND QUALIFICATION CRITERIA FOR ACCIDENT MONITORING INSTRUMENTATION

Applicability

To the extent feasible and practical (in conformance with the stipulations of Appendix A and ancillary requirements), equipment is to be installed by the specified implementation dates. Where equipment is unavailable, precluding conformance with equipment qualification and schedular requirements, the implementation dates are to be met by installation of best available equipment. In such cases, deviations are to be described and a schedule for the feasible installation of equipment in conformance with the stipulations of Regulatory Guide 1.97 (when the guide is used) is to be provided.

Appendix A is consistent with our current draft version of Regulatory Guide 1.97. We expect no further revisions to our requirements.

Criteria

(1) The instrumentation should be environmentally qualified in accordance with Regulatory Guide 1.89 (NUREG-0588). Qualification applies to the complete instrumentation channel from sensor to display where the display is a direct-indicating meter or recording device. Where the instrumentation channel signal is to be used in a computer-based display, recording and/or diagnostic program, qualification applies to and includes the channel isolation device. The location of the isolation device should be such that it would be accessible for maintenance during accident conditions. The seismic portion of environmental qualification should be in accordance with Regulatory Guide 1.100. The instrumentation should continue to read within the required accuracy following, but not necessarily during, a safe shutdown earthquake. Instrumentation, whose ranges are required to extend beyond those ranges calculated in the most severe design basis accident event for a given variable, should be qualified using the following guidance.

The qualification environment shall be based on the design basis accident events, except the assumed maximum of the value of the monitored variable shall be the value equal to the maximum range for the variable. The monitored variable shall be assumed to approach this peak by extrapolating the most severe initial ramp associated with the design basis accident events. The decay for this variable shall be considered proportional to the decay for this variable associated with the design basis accident events. No additional qualificaton margin needs to be added to the extended range variable. All environmental envelopes except that pertaining to the variable measured by the information display channel shall be those associated with the design basis accident events.

The above environmental qualification requirement does not account for steady-state elevated levels that may occur in other environmental parameters associated with the extended range variables. For example, a sensor measuring containment pressure must be qualified for the measured process variable range, but the corresponding ambient temperature is not mechanistically linked to that pressure. Rather, the ambient temperature value is the bounding value for design basis accident events analyzed in

Chapter 15 of the final saety analysis report (FSAR). The extended range requirement is to ensure that the equipment will continue to provide information should conditions degrade beyond those postulated in the safety analysis. Since variable ranges are nonmechanistically determined, extension of associated parameter levels is not justifiable and has, therefore, not been required.

(2) No single failure within either the accident-monitoring instrumentation, its auxiliary supporting features or its power sources concurrent with the failure that are a condition or result of a specific accident should prevent the operator from being presented the information necessary for him to determine the safety status of the plant and to bring the plant to a safe condition and maintain it in a safe condition following that accident. Where failure of one accident-montoring channel results in ambiguity (that is, the redundant displays disagree) which could lead the operator to defeat or fail to accomplish a required safety function, additional information should be provided to allow the operator to deduce the actual conditions in the plant. This may be accomplished by: (a) providing additional independent channels of information of the same variable (addition of an identical channel), or (b) providing an independent channel which monitors a different variable bearing a known relationship to the multiple channels (addition of a diverse channel), or (c) providing the capability, if sufficient time is available, for the operator to perturb the measured variable and determine which channel has failed by observation of the response on each instrumentation channel. Redundant or diverse channels should be electrically independent, energized from station Class 1E power source, and physically separated in accordance with Regulatory Guide 1.75 up to and including any isolation device. At least one channel should be displayed on a direct-indicating or recording device. (NOTE: Within each redundant division of a safety system, redundant monitoring channels are not required.)

(3) The instrumentation should be energized from station Class 1E power sources.

(4) An instrumentation channel should be available prior to an accident except as provided in Paragraph 4.11, "Exemption," as defined in IEEE Std 279 or as specified in technical specifications.

(5) The recommendations of the following regulatory guides pertaining to quality assurance should be followed:

1.28 "Quality Assurance Program Requirements (Design & Construction)

1.30 "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment"

1.38 "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants"

1.58	"Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel"
1.64	"Quality Assurance Requirements for the Design of Nuclear Power Plants"
1.74	"Quality Assurance Terms and Definitions"
1.88	"Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records"
1.123	"Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants"
1.144	"Auditing of Quality Assurance Programs for Nuclear Power Plants"
Task RS 810-5	"Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants" (Guide number to be inserted.)

Reference to the above regulatory guides (except Regulatory Guides 1.30 and 1.38) are being made pending issuance of a regulatory guide endorsing NQA-1 (Task RS 002-5), now in progress.

(6) Continuous indication (it may be by recording) display should be provided at all times. Where two or more instruments are needed to cover a particular range, overlapping of instrument span should be provided.

(7) Recording of instrumentation readout information should be provided. Where trend or transient information is essential for operator information or action, the recording should be analog stripchart or stored and displayed continuously on demand. Intermittent displays, such as data loggers and scanning recorders, may be used if no significant transient response information is likely to be lost by such devices.

(8) The instruments should be specifically identified on the control panels so that the operator can easily discern that they are intended for use under accident conditions.

(9) The transmission of signals from the instrument or associated sensors for other use should be through isolation devices that are designated as part of monitoring instrumentation and that meet the provisions of the document.

(10) Means should be provided for checking, with a high degree of confidence, the operational availability of each monitoring channel, including its input sensor, during reactor operation. This may be accomplished in various ways; for example:

(a) By perturbing the monitored variable

(b) By introducing and varying, as appropriate, a substitute input to the sensor of the same nature as the measured variable

(c) By cross-checking between channels that bear a known relationship to each other and that have readouts available.

(11) Servicing, testing, and calibrating programs should be specified to maintain the capability of the monitoring instrumentation. For those instruments where the required interval between testing will be less than the normal time interval between generating station shutdowns, a capability for testing during power operation should be provided.

(12) Whenever means for removing channels from service are included in the design, the design should facilitate administrative conrol of the access to such removal means.

(13) The design should facilitate administrative control of the access to all setpoint adjustments, module calibration adjustments, and test points.

(14) The monitoring instrumentation design should minimize the development of conditions that would cause meters, annunciators, recorders, alarms, etc., to give anomalous indications potentially confusing to the operator.

(15) The instrumentation should be designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules.

(16) To the extent practical, monitoring instrumentaton inputs should be from sensors that directly measure the desired variables.

(17) To the extent practical, the same instruments should be used for accident monitoring as are used for the normal operations of the plant to enable the operator to use, during accident situations, instruments with which the operator is most familiar. However, where the required range of monitoring instrumentation results in a loss of instrumentation sensitivity in the normal operating range, separate instruments should be used.

(18) Periodic testing should be in accordance with the applicable portions of Regulatory Guide 1.118 pertaining to testing of instruments channels.

APPENDIX C

N U C L E A R   P O W E R

P L A N T   S H I F T   T E C H N I C A L   A D V I S O R

Recommendations for Position Description, Qualifications, Education and Training

THE
INSTITUTE OF NUCLEAR
POWER OPERATIONS

Revision 0
April 30, 1980

FOREWORD

The Shift Technical Advisor position is generally accepted by the industry and the NRC as being an interim position. Long range criteria (three to five years) require that the qualifications of shift supervisors and senior operators be upgraded with the shift supervisor required to have an engineering degree or equivalent qualifications.

In developing recommendations for the STA position and giving consideration to the current shortage of qualified engineering graduates to fill the interim positions, the working groups attempted to identify those areas of education and levels of experience considered necessary to effectively accomplish the position's most important function - accident assessment. Recognizing that many engineering or scientific degree programs do not normally include the range and depth of technical subjects required for accident assessment, the recommendations included identify the subject areas and depth of study necessary but do not specify through what programs they should be acquired.

The user is cautioned to ensure that the recommended education and training is conducted in a professional manner by competent instructors and at the proper level. Institutions and programs accredited by recognized agencies such as ECPD/ABET or others ensure that adequate standards are met.

The program identified should provide the technical depth necessary to meet long-term qualification requirements of both the Senior Reactor Operator and the Shift Supervisor at the time when the STA position is eliminated. Since the shift supervisor position normally is involved in a broader range of managerial responsibilities, additional training in non-technical subjects such as technical writing, oral communication, and decision making is recommended.

Development of the technical and language skills at the level recommended along with the applied fundamentals and practical training recommended is considered an acceptable equivalent to an engineering degree insofar as qualifications for Shift Supervisor are concerned.

TABLE OF CONTENTS

1. DEFINITIONS

The definitions given below are of a restricted nature for the purpose of these recommendations.

Academic Training - Successfully completed college-level work which may or may not lead to a recognized degree in a discipline related to the position.

Experience - Applicable work in design, construction, preoperational and startup testing activities, operation, maintenance, or technical services. Observation of others performing these functions shall not be considered acceptable experience.

Licensed Operator - Any individual who possesses an operator's license pursuant to Title 10, Code of Federal Regulations, Part 55, "Operators' Licenses".

Licensed Senior Operator - Any individual who possesses a senior operator's license pursuant to 10 CFR Part 55.

Manager of Nuclear Power - The individual in the utility organization who is directly responsible for the operation of that utility's nuclear power plants and will usually be the person to whom the Plant Manager reports.

Nuclear Power Plant - Any plant using a nuclear reactor to produce electric power, process steam or space heating.

Nuclear Power Plant Experience - Experience acquired in the preoperational and startup testing activities or operation of nuclear power plants. Experience in design, construction, maintenance, and instructing may be considered applicable nuclear power plant experience and should be evaluated on a case-by-case basis.

(1) Experience acquired at military or production nuclear plants may qualify as equivalent nuclear power plant experience.

(2) Nuclear power plant systems and operations training (classroom, on-the-job or simulator) may qualify as nuclear power plant experience if it applies to the plant at which the position is to be filled or a similar plant.

Nuclear Reactor - Any assembly of fissionable material which is designed to achieve a controlled, self-sustaining neutron chain reaction.

On-The-Job Training - Participation in nuclear power plant startup, operation, maintenance, or technical services under the direction of experienced personnel.

Related Technical Training - Formal training beyond the high school level in technical subjects associated with the position in question, such as acquired in training schools or programs conducted by the military, industry, utilities, universities, vocational schools, or others. Such training programs shall be of a scheduled and planned length and include text material and lectures.

Shall, Should and May - The word "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission - neither a requirement nor a recommendation.

STA - Shift Technical Advisor - That position at a nuclear power plant established to evaluate plant conditions and provide advice to the Shift Supervisor during plant transients and accidents. Inherent in this function is the detection and reporting of potential safety problems.

Utility (Owner Organization) - The organization, including the on-site operating organization, which has overall legal, financial and technical responsibility for the operation of one or more nuclear power plants. This shall include contracted personnel (vendors, consultants, etc.).

2. INTRODUCTION

After the the accident at Three Mile Island, investigations by several committees and the Nuclear Regulatory Commission concluded that certain deficiencies may exist in the level of technical expertise generally available to the shift operating staff prior to, during, and immediately after an accident or severe plant transient. Although adequate expertise may be available some time later, the lack of skilled analytical capability during such occurances may contribute to equipment damage or danger to the plant staff and the public. Subsequent recommendations and regulations require that additional technical expertise be made available to each operating shift. Current regulatory requirements identify those individuals providing this expertise on shift as Shift Technical Advisors (STAs).

The purpose of this document is to describe the position and identify specific areas of formal education, plant training and experience necessary to assure an advanced level of analytical ability on shift. These recommendations will provide a level of technical ability that is essential to improved operational safety and are consistent with regulatory requirements. This Institute position was developed in conjunction with representatives of utilities, equipment vendors and engineering educators, giving consideration to specific contributions the function must make to shift operations.

For convenience, the necessary contributions are identified in the form of a position description. Although this format suggests that the function will be performed by a new position, it is not intended to pre-empt management's prerogative to accomplish the function through other qualified individuals within an existing organizational structure.

It should be noted that the areas of formal education identified are not normally included in any one course or in the courses for any one established engineering or related scientific degree program. Rather, the areas and depth of study are those needed to effectively perform the function. The areas identified do provide a basis for either exempting certain subject areas for qualified engineering graduates or for establishing developmental programs for non-graduates or graduates of a degree program that does not include the requisite subject areas.

3. OBJECTIVE

The objective of creating the STA position is to improve the quality of plant technical management and operation by providing additional on-shift expertise in the area of operational safety, thus reducing the probability of abnormal or emergency condition occurrences and mitigating the consequences of these conditions if they do occur.

4. POSITION DESCRIPTION

The responsibilities of the Shift Technical Advisor should be documented in such a way that the incumbent clearly understands the duties and responsibilities of the position. The following position description is a suitable method for describing the work to be performed and the measures of incumbent performance.

Function

Provide advanced technical assistance to the operating shift complement during normal and abnormal operating conditions.

General Qualifications

(1) That combination of educating, training and nuclear plant experience identified in Sections 5 and 6.

(2) An in-depth understanding of nuclear plant equipment, systems and operating practices and procedures.

(3) Well developed analytical skills and the ability to make sound judgements under stressful conditions.

General Duties

(1) During assigned tour of duty be cognizant of plant and equipment status.

(2) Maintain independence from normal plant operations as necessary to make objective evaluations of plant operations and to advise or assist plant supervision in correcting conditions that may compromise the safety of operations.

(3) Be readily available to provide appropriate assistance to the normal shift complement.

Typical Responsibilities

(1) During transients and accidents, compare existing critical parameters, (i.e. neutron power level; reactor coolant system level, pressure and temperature; containment pressure, temperature, humidity and radiation level; and plant radiation levels) with those predicted in the Plant Transient and Accident Analysis, to ascertain whether the plant is responding to the incident as predicted.

Report any abnormalities to the Shift Supervisor immediately and provide assistance in formulating a plan for appropriate corrective action.

(2) Make a qualitative assessment of plant parameters during and following an accident in order to ascertain whether core damage has occurred.

(3) During emergencies be observant of critical parameters, ascertain that there is adequate core cooling including availability of a heat sink for the coolant system, and, in the event that critical parameters become unavailable due to instrument failure, perform calculations or through other means determine approximate values for the parameters in question.

(4) Investigate the cause(s) of abnormal or unusual events occurring on assigned shift and assess any adverse affects therefrom. Recommend changes to procedures or equipment as necessary to prevent recurrence.

(5) Evaluate the effectiveness of plant procedures in terms of terminating or mitigating accidents and make recommendations to the Shift Supervisor when changes are needed.

(6) Assist the operations staff in interpreting and applying the requirements of Technical Specifications.

(7) Perform an early review of the planned activities for the upcoming shift to ascertain whether special considerations or precautions are warranted and make appropriate recommendations to the Shift Supervisor. This review should include scheduled surveillance tests and major maintenance items.

(8) Evaluate effectiveness of plant instructions and recommend needed changes to the appropriate Supervisor.

(9) Evaluate core power distribution during and following load changes. Perform hot channel factor and/or rod program analyses as required.

(10) Review abnormal and emergency procedures.

(11) Prepare special reports when requested by the Operations Superintendent.

(12) Provide an engineering evaluation of Licensee Event Reports from other plants as assigned.

Accountability

The STA is accountable for the following end results:

(1) Contributes to maximizing safety of operations by independently observing plant status and advising shift supervision of conditions that could compromise plant safety.

(2) Contributes to maximizing plant safety during transient or accident situations by independently assessing plant conditions and by providing the technical assistance necessary to mitigate the incident and minimize the effect on personnel, the environment, and plant equipment.

5. GENERAL EDUCATION AND EXPERIENCE

5.1 EDUCATION AND TRAINING

The Shift Technical Advisor shall meet the education and training requirements of Section 6.

5.2 EXPERIENCE

The Shift Technical Advisor shall have a minimum of 18 months of nuclear power plant experience, at least two months of which shall be at an operating nuclear plant.

A maximum of six months of this experience may be obtained in the military or at a production nuclear plant and should be evaluated on a case-by-case basis.

A maximum of three months of systems and operations training may be applied toward these experience requirements.

At least 12 months of this experience shall be at the station at which the position is to be filled. This may be waived in part when two essentially identical plants are involved.

Experience gained at a nuclear station prior to initial fuel loading is acceptable, if the individual actively participates in preparation and review of plant procedures and test programs, and is on-site for at least one year during the preoperational test phase.

5.3 ABSENCES FROM STA DUTIES

Persons not actively performing the STA functions for a period of thirty (30) days or longer shall, prior to assuming responsibilities of the position, as a minimum receive training sufficient to ensure he is cognizant of facility/procedure changes that occured during his absence.

Persons not performing the STA function for a period of six (6) months or longer shall, prior to assuming the responsibilities of the position, receive the annual requalification training described in this document.

6. EDUCATION AND TRAINING REQUIREMENTS

A waiver for any of the required education or training shall be granted only by the Manager of Nuclear Power and should be evaluated on a case-by-case basis. Such waivers may be considered when a candidate has documented accredited college courses or can demonstrate an acceptable level of knowledge through comprehensive examinations in the area to be waived.

For courses completed at an accredited college, a semester credit hour shall be considered equivalent to approximately 15 contact hours in a full-time training program.

When courses prescribed in Sections 6.1.2 and 6.2 are not administered by an accredited college or university the curriculum and instructor shall be certified by the INPO.

6.1 EDUCATION

6.1.1 Prerequisites Beyond High School Diploma It is assumed that many candidates may have received previous training and are qualified to begin the coursework prescribed in 6.1.2. Prerequisite education considered necessary for successful completion of the advanced coursework is identified below. This coursework may be waived without formal documentation of specific course completion.

	Contact Hours
Mathematics	
Trigonometry, Analytical Geometry, College Algebra	90
Chemistry	
Inorganic Chemistry	30

Physics

Engineering Physics (heat, mechanics, light sound, electricity and magnetism)	150
TOTAL	270

6.1.2 College Level Fundamental Education

	Contact Hours
Mathematics	90
Engineering mathematics through the introduction to ordinary differential equations and the utilization of Laplace transforms to interpret control response.	
Reactor Theory	100
Atomic and Nuclear Physics Statics, through 2-group Diffusion Theory Dynamics, Point Kinetics, Reactivity Feedback	
Reactor Chemistry	30
Inorganic Chemistry (as related to reactor systems) Corrosion - Reaction Rates	
Nuclear Materials	40
Strength of Materials Reactor Material Properties (phase diagrams, fuel densification)	

Thermal Sciences (for nuclear systems)	120
Thermodynamics	
Laws of Thermodynamics	
Properties of Water and Steam	
Steam Cycles and Efficiency	
Fluid Dynamics	
Bernoulli's Equation	
Fluid Friction and Head Loss	
Elevation Head	
Pump and System Characteristics	
Two Phase Flow	
Heat Transfer	
Methods of Heat Transfer	
Boiling Heat Transfer	
Heat Exchangers	
Electrical Sciences	60
Electronics (Circuit theory, digital electronics)	
Motors, Generators, Transformers, Switchgear	
Instrumentation and Control Theory	
Nuclear Instrumentation and Control	40
Radiation Detectors	
Reactor Instrumentation	
Reactivity Control and Feedback	
Nuclear Radiation Protection and Health Physics	40
Biological Effects	
Radiation Survey Instrumentation	
Shielding	
TOTAL	520

6.2 APPLIED FUNDAMENTALS - PLANT SPECIFIC

In addition to the general education require ments described in Section 6.1, all STAs shall complete the following training at the college level tailored to the specific plant at which the STA is assigned or a plant of similar design. It may be presented separately from or may be integrated with the education described in Section 6.1.

Subject Topics	Contact Hours
Plant Specific Reactor Technology (including core physics data)	
Plant Chemistry and Corrosion Control	
Reactor Instrumentation and Control	
Reactor Plant Materials	
Reactor Plant Thermal Cycle	
TOTAL	110

6.3 MANAGEMENT/SUPERVISORY SKILLS

Subject	Contact Hours
Leadership	
Interpersonal Communication	
Motivation of Personnel	
Problem and Decisional Analysis	
Command Responsibilities and Limits	
Stress	
Human Behavior	
TOTAL	40

6.4 PLANT SYSTEMS

The training program shall cover the following systems along with others considered necessary for a specific plant.

System	Contact Hours
Emergency Core Cooling	
Emergency Cooling Water	
Emergency Electrical Power, AC and DC	
Reactor Protection	
Reactor Coolant	
Reactor Coolant Inventory and Chemistry Control	
Containment System (including Containment Cooling)	
Closed Cooling Water	
Nuclear Instrumentation	
Non-Nuclear Instrumentation	
Reactor Control	
Containment Hydrogen Monitoring and Control	
Radioactive Waste Disposal (liquid, gas, solid)	
Emergency Control Air	
Condensate and Main Feedwater	
Auxiliary Feedwater	
Steam Generator Level Control (PWR)	
Reactor Vessel Water Level Control (BWR)	
Main Steam	
Loose Parts Monitoring (PWR)	
Status Monitoring (including Process Computer)	
Seismic Monitoring	
Residual Heat Removal	
Radiation Monitoring	
Plant Ventilation	
Main Turbine and Generator	
TOTAL	200

6.5 ADMINISTRATIVE CONTROLS

Subject	Contact Hours
Responsibilities for Safe Operation and Shutdown	
Equipment Outages and Clearance Procedures	
Use of Procedures	
Plant Modifications	
Shift Relief Turnover and Manning	
Containment Access	
Maintaining Cognizance of Plant Status	
Unit Interface Controls (multi-unit plants with one or more units still under construction)	

Physical Security
Control Room Access
Duties and Responsibilities of the STA
Radiological Emergency Plan
Code of Federal Regulations (appropriate sections)
Plant Technical Specifications (including bases)
Radiological Control Instructions

TOTAL 80

6.6 GENERAL OPERATING PROCEDURES

Subject	Contact Hours
Startup	
At Power Operations	
Shutdown	
Xenon Following While on Standby	
ECP and S.D. Margin Calculation	
TOTAL	30

6.7 TRANSIENT/ACCIDENT ANALYSIS AND EMERGENCY PROCEDURES

Subject	Contact Hours
Transient and Accident Analyses	
Plant Abnormal and Emergency Procedures	
TOTAL	30

6.8 SIMULATOR TRAINING

The plant evolutions, transients and events listed below shall be conducted along with any others deemed necessary. The primary objective should be to demonstrate plant and operator response to a given condition or event and not necessarily to develop the control manipulation expertise of the trainee. The trainee/ instructor ratio should not exceed 4:1.

Simulator exercises should be preceded by a period of discussion of the planned exercises addressing expected response of the plant and applicable plant procedures to be used. Approximately 100 contact hours are required with about 50 hours in the classroom and 50 hours on the simulator.

Following each exercise demonstrating a transient or emergency event, an incident critique discussion should be held to enhance the trainees' understanding of that particular exercise. When the simulator is not plant-specific, the training shall be tailored to the specific plant as much as practical.

PWR Simulator Exercises

Reactor and Plant Startup
Load Changes at Power
Shutdown to Cold Condition
Demonstration of Steam Generator Level Manual Control
Load Rejections of Greater than 10%
Failure of Rod Control System
Failure of Automatic Steam Generator Level Controls
Failure of Pressurizer Level and Pressure Automatic
   Controls
Turbine Trip from Full Power
Reactor Trip from Full Power
Loss of Normal Feedwater at Full Power
Failure Open of Power Operated Relief Valve
Stuck Open Pressurizer Safety Valve
Loss of Reactor Coolant Pumps at Full Power
   and Demonstration of Natural Circulation
Failure Open of One or More Turbine Bypass Valves
   While at a) Full Power, b) Hot Standby
Loss of All Feedwater (normal and emergency)
Loss of Reactor Coolant (small and DBA)
Steam Generator Tube Rupture (small and large)
Loss of RHR Shutdown Cooling with the RCS
   Temperature 200° to 300°F
Inadvertent Safety Injection While at Power
Loss of Offsite Electrical Power
Loss of One Train of Onsite Electrical Power

BWR Simulator Exercises

Reactor and Plant Startup
Load Changes at Power (using flow control
when applicable)
Shutdown
Load Rejection of Greater than 10%
Turbine Trip from Full Power
Turbine Bypass Valve Failure to Open Following Trip
Inadvertent Isolation of MSIV's While at Power
Reactor Scram from Full Power
Reactor Pressure Control Failure
Dropped Control Rod While at Power
Cold Water Transient at Power
Inadvertent Opening of Relief Valve
Loss of Main Feedwater Pumps at Power
Inadvertent Start of Idle Recirculation Pump
Inadvertent Trip of Recirculation Pump(s)
Loss of Reactor Coolant (small break - large break)
Steam Line Break (inside-outside containment)
Loss of Offsite Power
Loss of Shutdown Cooling with RCS Temperature
200° - 300°F
Demonstration of Natural Circulation Capabilities
Malfunction of Reactor Water Level Automatic Controls

6.9 ANNUAL REQUALIFICATION TRAINING

Subject Material	Hours Required
Review of transient and accident analyses of FSAR condition III and IV events emphasizing the individual's role in accident assessment. Review selected industry events and LERs that could have led to more serious incidents.	40 (Lecture)
Simulator exercises related to the transients in Section 6.8 conducted so as to emphasize the role of the STA.	40 (Simulator)
TOTAL	80

NRC FORM 335 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC)	NUREG-0737
4. TITLE AND SUBTITLE (Add Volume No., if appropriate)	Clarification of TMI Action Plan Requirements
2. (Leave blank)	
3. RECIPIENT'S ACCESSION NO.	
7. AUTHOR(S)	
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16. ABSTRACT (200 words or less)

This document, NUREG-0737, is a letter from D.G. Eisenhut, Director of the Division of Licensing, NRR, to licensees of operating power reactors and applicants for operating licenses forwarding post-TMI requirements which have been approved for implementation. Following the accident at Three Mile Island Unit 2, the NRC staff developed the Action Plan, NUREG-0660, to provide a comprehensive and integrated plan to improve safety at power reactors. Specific items from NUREG-0660 have been approved by the Commission for implementation at reactors. In this NRC report, these specific items comprise a single document which includes additional information about schedules, applicability, method of implementation review, submittal dates, and clarification of technical positions. It should be noted that the total set of TMI-related actions have been collected in NUREG-0660, but only those items that the Commission has approved for implementation to date are included in this document, NUREG-0737.

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