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NUREG-0694

TMI-RELATED REQUIREMENTS FOR NEW OPERATING LICENSES

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TMI-RELATED REQUIREMENTS FOR NEW OPERATING LICENSES

Background

The NRC Action Plan on the Accident at Three Mile Island (NUREG-0660) was developed to provide a comprehensive and integrated plan for the actions now judged necessary by the Nuclear Regulatory Commission to correct or improve the regulation and operation of nuclear facilities. The Action Plan was based on the experience from the accident at TMI-2 and the official studies and investigations of the accident. Activities and programs of the NRC not related to the accident at TMI-2 are not described in the Action Plan. They are contained in a separate resource and scheduling mechanism known as the NRC Operating Plan. Thus, the Action Plan complements the current NRC Operating Plan and the other important safety issues and programs addressed therein. The schedules and resources presented in the Action Plan and the NRC Operating Plan were adjusted in accordance with the relative priorities of the various elements of each to optimize the increase in safety consistent with the resources available to the agency in fiscal years 1980 and 1981.

Those who have investigated the accident include the Congress, the General Accounting Office, the President's Commission on the Accident at Three Mile Island, the NRC Special Inquiry Group, the NRC Advisory Committee on Reactor Safeguards (ACRS), the Lessons-Learned Task Force and the Bulletins and Orders Task Force of the NRC Office of Nuclear Reactor Regulation, the Special Review Group of the NRC Office of Inspection and Enforcement, the NRC staff Siting Task Force and Emergency Preparedness Task Force, and the NRC Offices of Standards Development and Nuclear Regulatory Research. Each of the investigating groups organized their recommendations in a different way. (The recommendations of the major investigations are cross-indexed to the Action Plan in Volume 2.) The development of the Action Plan served to collect those recommendations into five chapters, each of which covers one broad subject; namely, I. Operational Safety; II. Siting and Design; III. Emergency Preparedness and Radiation Effects; IV. Practices and Procedures; and V. NRC Policy, Organization, and Management. The chapters are further divided into sections that contain the actions related to a stated objective in an area of safety identified by the accident as being in need of improvement.

In the development of this Action Plan, NRC has transformed the recommendations into discrete, scheduled tasks that specify changes (or studies of possible future changes) in its regulatory requirements or its organization and procedures. The Plan also identifies the organizational elements responsible for the various actions and contains estimates of the resources and schedule necessary for both NRC and the industry to accomplish the actions. As is the nature of any plan, the actions, resources and schedules in the near term are more likely to be accurate than are those for the long term. In recognition of this, the overall plan is not intended to be inviolable -- changes in the specified actions will be made as necessary to reflect new information.

Some actions to improve the safety of nuclear power plants now operating were judged to be necessary immediately after the accident and could not be delayed until an action plan was developed, although they were subsequently included in the Action Plan. Such actions came from the Bulletins and Orders issued by the Commission immediately after the accident, the first report of the Lessons-Learned Task Force issued in July, the recommendations of the Emergency Preparedness Task Force and the NRC staff and Commission. Before these immediate actions were applied to operating plants, they were approved by the Commission. Many of the required immediate actions have already been taken by licensees and most are scheduled to be complete by the end of 1980.

Development of the Action Plan began after the immediate corrective actions were well under way and at the time when the principal external investigation, that of the President's Commission, was complete. In developing the Action Plan, the various recommendations and possible actions of all the principal investigations were assessed and either rejected, adopted or modified. These assessments and decisions were made under the direction of a TMI Action Plan Steering Group, which served to integrate and coordinate the development of the Action Plan by the various program offices of the agency. The Commission, the ACRS, the Executive Director for Operations, and the directors of the program offices reviewed and commented on the various drafts of the plan, and their guidance, decisions and directions were followed in refining the plan. The decisions on whether to include specific items in the plan were based primarily on whether they were necessary to respond to the recommendations of the principal investigations. However, decisions on the priority and resources to be afforded the various actions in the plan have been based primarily on their relative risk-reduction potential. Throughout the decision-making process, there has been general agreement that the accident demonstrated that improvements in safety are needed. There has also been general agreement among the various investigators as to the causes of the accident and the failures and errors that occurred before and during the event, both in the equipment and in the organizations that built, operated and regulated the plant. Therefore, there has also been general agreement as to the areas where improvements should be made. Where differences of opinion have occurred, they most often relate to the degree of improvement required and the best ways of achieving improvement. Having considered the various recommendations and various ways of responding to them, the Action Plan represents a collective NRC assessment of the types and degree of improvement that are necessary and describes the means and schedule for attaining the improvements.

When determining the schedules for developing and implementing changes in requirements, the primary concern was the perceived immediacy of the need for corrective actions. As discussed above, many actions were taken to improve safety immediately or soon after the accident. These actions were generally considered to be interim improvements until a better, more comprehensive or more desirable solution could be implemented. However, in scheduling the longer term improvements, the availability of both NRC and industry resources was also considered, as well as the safety significance of the actions. Thus, the Action Plan presents a sequence of actions that will result in a gradually increasing improvement in safety as individual actions are completed and the initial immediate actions are replaced or supplemented by longer term, more durable improvements. The first step in that sequence of actions for new plants that are now ready to load fuel is to specify a discrete set of TMI-related licensing requirements. Such a set of requirements was developed as a subset of the Action Plan and is summarized in the following section.

Development of Requirements for New Operating Licenses

Near-term operating license (NTOL) requirements were defined in the past few months as those actions in the TMI Action Plan that were proposed by the staff for implementation prior to granting a new operating license because they are needed, are sufficiently characterized and studied at this time, and are known to have significant safety improvement potential. A list of NTOL requirements was given preliminary approval by the Commission on February 7, 1980. It is described in Table A.1 of the May 1980 version of the Action Plan. On February 7, that list was approved by the Commission as necessary but not sufficient for granting full-power operating licenses. Additional study of that list was undertaken by the Commission and ACRS, as described below. That work culminated in the Commission's approval on May 15, 1980 of the TMI-related requirements for new operating licenses. Those requirements are listed in the last section of this document.

From the inception of the TMI Action Plan, primary emphasis was placed on developing and implementing the necessary changes in requirements for operating reactors and changes in NRC practices and procedures to diminish the risk of present operations. By and large, the actions of this sort described in the first draft of the TMI Action Plan were already being implemented as a result of the short-term recommendations of the TMI-2 Lessons Learned Task Force (NUREG-0578, July 1979) and the requirements of the Bulletins and Orders Task Force. The first draft of the Action Plan also contained requirements that were to be applied in licensing reviews of new plants that would be ready to load fuel within the near future; i.e., the so-called near-term operating license facilities. Four new plants fell into the category of being ready to load fuel in 1980 (Sequoyah, North Anna 2, Diablo Canyon, Salem 2).

The NTOL list was refined several times between the first draft of the Action Plan and its final version. Throughout the process, the list has contained all the TMI-related requirements levied on operating reactors plus a few more. Also, in some instances, the requirements for the near-term operating licenses have implementation deadlines that are more stringent in some cases than the comparable requirements for operating plants.

This was done when there was a significant advantage to have the new procedure or equipment in place during fuel loading or power-ascension testing. As a general rule, however, implementation schedules for near-term operating license requirements were established with the intent of providing adequate safety improvement without incurring significant additional schedule and construction delays.

The first major effort to systematically review and refine the NTOL list occurred shortly after issuance of Draft 1 of the TMI Action Plan. The Action Plan Steering Group, in consultation with its Task Managers, discussed additions and refinements of the specific actions recommended in Draft 1 for near-term operating license applicants. A revised list of actions was then discussed, further refined and approved by the NRC Office Directors. This list of approximately 50 actions was forwarded to the Commission on January 5, 1980.

On January 10, 1980 the Action Plan Steering Group met with the Advisory Committee on Reactor Safeguards (ACRS) to discuss Draft 1 of the Plan. A copy of the proposed NTOL requirements was also provided to the Committee, although the focus of the meeting was on the entire Plan, not the NTOL list. A primary concern expressed by ACRS at that time was the lack of priority assignments within Draft 1 of the Action Plan and the likelihood that, without better delineation of priorities, NRC and the utilities could not focus on the most important actions.

In its review of the January 5 version of the NTOL list, the Commission also expressed a need for more assurance that the NTOL list contained the most important things to do first. They asked that the staff gain a reactor operator's perspective on the safety implications of the proposed requirements. In order to get operator and industry assessments of the impact on safety of implementing the near-term operating license actions, several site visit teams were created by the Steering Group to conduct onsite meetings with operating personnel and utility management. These teams were composed of IE Regional Branch Chiefs, the NRR licensing project managers for the plants, the resident inspectors, and various senior NRC managers and directors. Meetings were held at the four near-term operating license facilities and at four operating facilities. The operating facilities were included to ensure that actual operating experience, not just planning for operations, would be reflected in the overall safety assessment. During the site visits, the NRC team met separately with the licensed operators or license candidates, as well as with site and corporate managers. The primary objective was to identify actions on the NTOL list which, if implemented, might result in a less safe, rather than more safe, operation. As a result of the visits, the review teams concluded that no single near-term operating license requirement would, of itself, produce a negative safety or quality impact if implemented. However, in the aggregate, if all the requirements were imposed on the utility engineering and technical support staffs, they might be unduly diverted from necessary and ongoing routine safety-related tasks and overall safety might be diminished. As a result of discussions with operators and managers, the review teams recommended that four actions be removed from the January 5 version of the list and rescheduled for future action.

While such refinements of the NTOL list were under way, the NRC Special Inquiry Group (SIG) issued its report on Three Mile Island on January 24, 1980. The SIG recommendations were reviewed by the Steering Group, task managers and NRC Offices for incorporation into the Action Plan and, where appropriate, the list of near-term operating license requirements. This review identified a number of suggestions that were considered for addition to the NTOL list. Two of these suggestions were approved for the final list (Control Room Design Review - Item I.D.1, and Power Ascension Test Schedule - Item IV.F.1).

Based on information received from the site visits, ACRS meetings, and SIG recommendations, it was clear that a close review of the January 5 NTOL list was appropriate to ensure that requirements were not being levied that did not have a high safety payoff. Additionally, the Steering Group had completed a detailed estimation of priorities of all the actions in the Plan that could be used to evaluate the relative importance of specific requirements. A comprehensive review by the Steering Group identified twelve items in Draft 2 of the Action Plan that, after closer evaluation, were not considered to be essential for near-term operating licenses and were deleted from the NTOL list. These items continue to be included in the Action Plan for future action. Before an item was removed from the NTOL list, the basis for its removal was developed and reviewed by the Action Plan Steering Group and NRC Office Directors. Typical reasons for removing actions from the NTOL list were: the primary concern in the action was already being addressed by another interim requirement and the specific action could be addressed better in a more comprehensive manner in the long term; requirements were not well defined and could place a heavy resource demand on near-term operating license facilities with uncertain benefits that could be counter to safety in some extreme cases; implementation before fuel loading or full-power operation was not critical and the item could be implemented on the same schedule as operating reactors.

The Directors of NRR, IE, SD, and RES reviewed the revised NTOL list with the Steering Group on February 5, 1980 and concurred in the requirements. The Commission met on February 7, 1980 and approved the list as being necessary to implement but did not approve the list as being sufficient for issuing new operating licenses. The EDO directed the responsible NRC program offices to implement their portions of the requirements by memo of February 19, 1980. Each of the requirements was to be specifically addressed in the Safety Evaluation Reports for the affected plants. Three near-term operating license applicants that were granted restricted operating licenses (fuel loading and low-power testing) revised their applications to demonstrate conformance with applicable portions of the NTOL list (Salem 2, Sequoyah, and North Anna 2).

In a February 19, 1980 memorandum to the ACRS, the NRC Chairman requested that the ACRS specifically consider the NTOL list in its March meeting and provide the Commission with ACRS views on whether the list was necessary and sufficient for authorizing operating licenses. The ACRS provided comments to the Commission on March 11, 1980 regarding thirteen specific areas of the Action Plan and noted that, subject to these comments, the NTOL items identified in Draft 3 of the Action Plan provided a satisfactory basis for resumption of licensing. The staff reviewed the ACRS comments and provided a point-by-point response to the Commission on April 1, 1980 describing how the Action Plan would be modified to account for ACRS comments. The staff also held meetings with the ACRS subcommittee on TMI on April 1 and 2, 1980 and with the full ACRS on April 10, 1980 to discuss the Action Plan in general, including modifications made by the staff in response to the ACRS letter of March 11. The April 17, 1980 letter from the ACRS provided specific comments on some elements of the Action Plan, plus a general agreement by the Committee that the plan was satisfactory for dealing with the issues identified by the accident at TMI-2.

On May 2, 1980, the staff transmitted an Action Paper to the Commission to seek a set of decisions on the TMI Action Plan (SECY-80-230). It contained a recommendation for the Commission to approve a list of "Requirements for New Operating Licenses." That list was a recast of the NTOL list described above into four sections, as summarized in the next section. On May 15, the NRC approved the list of new operating license requirements.

Requirements for New Operating Licenses

The TMI-related requirements and actions approved by the Commission for new operating licenses are of four types: (1) those required to be completed by a license applicant prior to receiving a fuel-loading and low-power testing license, (2) those required to be completed by a license applicant prior to receiving a license to operate at appreciable power levels up to full power, (3) those the NRC will take prior to issuing a fuel-loading and low-power testing or a full-power operating license, and (4) those required to be completed by a licensee prior to a specified date. For purposes of this discussion, only those dated requirements that have already been issued are of interest. Other dated requirements are expected to be issued in the future as work progresses on the Action Plan. The several parts of the list of TMI-related requirements approved by the Commission for new operating licenses are summarized below.

1. Fuel-Loading and Low-Power Testing Requirements

The first part of the list consists of those requirements that must be met by an applicant for an operating license prior to NRC issuance of a license to load fuel and conduct low-power testing. These fuel-loading (FL) requirements have been implemented in the staff reviews of Sequoyah, North Anna 2, and Salem 2 and were found by the Commission to provide an adequate basis for a fuel-loading and low-power testing license insofar as TMI-2 implications were concerned. Other normal operating license review requirements (non-TMI requirements) were also applied to these plants. The same list of fuel-loading requirements is being applied by the NRC staff to other applicants for operating licenses, such as Diablo Canyon.

2. Full-Power Requirements

The second part of the new requirements is that which must be met by operating license applicants prior to NRC issuing to them a license to operate at appreciable power levels, up to full power. These full-power (FP) requirements have been provided to the near-term operating license applicants and are being implemented by the staff in its ongoing reviews of Sequoyah, North Anna 2 and Salem 2 while those plants are undergoing low-power testing. No full-power license has been issued since these requirements were established. For later operating license applicants, the staff intends to conduct one operating license review by combining the fuel-loading and full-power testing requirements into a single, consistent set of operating license requirements.

3. Internal NRC Actions

The third category of new requirements consists of the actions that NRC plans to take prior to issuing licenses, either for fuel-loading and low-power testing or for full-power operation. The internal NRC actions designated to be complete before fuel loading (FL) were accomplished prior to issuing the fuel-loading licenses to Sequoyah, North Anna 2, and Salem 2. The staff actions designated to be complete before granting a new full-power license are under way, and are scheduled to be completed prior to any near-term operating plant licensee being permitted to operate beyond the low-power testing range.

4. Dated Requirements

A fourth class of new, TMI-related requirements that will affect new operating licenses is the set of requirements that was previously approved by the Commission, that has been issued to presently operating reactors, and that is required to be implemented by specified deadlines by all licensees or applicants. The attached list of summary descriptions of these requirements is in addition to the requirements listed above as prerequisites for fuel-loading or full-power licenses. Experience with implementation of the dated requirements on operating reactors is indicating to NRR that the January 1, 1981 deadline may be too tight in some cases to allow reasonable time for completion of the work required. This experience may prove to be the case for some of the dated requirements for NTOLs. The staff would intend to allow case-by-case exceptions to the deadlines if good cause is shown. The dated requirements are not preconditions for licensing of new plants. That is, if a completion deadline falls later than the operating license date for a new plant, then that requirement need not be met by the newly licensed plant until the completion deadline. If in the future a completion deadline falls before an operating license issuance date, then that requirement is a prerequisite for the new operating license, except when a good cause is shown for exception. These requirements were issued in letters to operating plant licensees on September 13 and October 30, 1979; to operating license applicants on September 27 and November 9, 1979; to licensees of plants under construction and construction permit applicants on October 10 and November 9, 1979; and to all licensees and applicants on March 28, 1980.

The list of TMI-related requirements for new operating licenses follows.

Part 1 - Fuel-Loading and Low-Power-Testing Requirements

I.A.1.1 SHIFT TECHNICAL ADVISOR

A technical advisor to the shift supervisor shall be present on all shifts and available to the Control Room within 10 minutes. Although minimum training requirements have not been specified, shift technical advisors should enhance the accident assessment function at the plant.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.2.1.b, and letters of September 27 and November 9, 1979.)

I.A.1.2 SHIFT SUPERVISOR ADMINISTRATIVE DUTIES

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

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.2.1a, Item (4), and letters of September 27 and November 9, 1979.)

I.A.1.3 SHIFT MANNING

The minimum shift crew for a unit shall include three operators, plus an additional three operators when the unit is operating. Shift staffing may be adjusted at multi-unit stations to allow credit for operators holding licenses on more than one unit.

In each control room, including common control rooms for multiple units, there shall be at all times a licensed reactor operator for each reactor loaded with fuel and a senior reactor operator licensed for each reactor that is operating. There shall also be onsite at all times, an additional relief operator licensed for each reactor, a licensed senior reactor operator who is designated as shift supervisor, and any other licensed senior reactor operators required so that their total number is at least one more than the number of control rooms from which a reactor is being operated.

Administrative procedures shall be established to limit maximum work hours of all personnel performing a safety-related function to no more than 12 hours of continuous duty with at least 12 hours between work periods, no more than 72 hours in any 7 day period, and no more than 14 consecutive days of work without at least 2 consecutive days off.

These requirements shall be met before fuel loading. (Detailed guidance to licensees is expected to be issued in the early summer of 1980.)

I.A.3.1 REVISE SCOPE AND CRITERIA FOR LICENSING EXAMINATIONS

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

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

These requirements shall be met before fuel loading.* (See letter of March 28, 1980.)

I.B.1.2 EVALUATION OF ORGANIZATION AND MANAGEMENT IMPROVEMENTS OF NEAR-TERM OPERATING LICENSE APPLICANTS

The licensee organization shall comply with the findings and requirements generated in an interoffice NRC review of licensee organization and management. The review will be based on an NRC document entitled Draft Criteria for Utility Management and Technical Competence. The first draft of this document was dated February 25, 1980, but the document is changing with use and experience in ongoing reviews. These draft criteria address the organization, resources, training, and qualifications of plant staff, and management (both onsite and offsite) for routine operations and the resources and activities (both onsite and offsite) for accident conditions.

Establish a group that is independent of the plant staff but is assigned on site to perform independent reviews of plant operational activities and a capability for evaluation of operating experiences at nuclear power plants.

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*In the case of Sequoyah, North Anna 2, Salem and McGuire, the operators were not required to take the new written examination, but were required to meet the new passing-grade requirement. However, any license applicants who must be reexamined are being required to take the new examination. The licensed operators and senior operators for all other new operating licenses will be required to take the new examination.

Organizational changes are to be implemented on a schedule to be determined prior to fuel loading.

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

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

These requirements shall be met before fuel loading. (See NUREG-0578, Sections 2.1.3b and 2.1.9, and letters of September 27 and November 9, 1979.)

I.C.2 SHIFT RELIEF AND TURNOVER PROCEDURES

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

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.2.1c, and letters of September 27 and November 9, 1979.)

I.C.3 SHIFT SUPERVISOR RESPONSIBILITIES

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

These requirements shall be met before fuel loading. (See NUREG-0578, Section 2.2.1a, Items 1, 2, and 3, and letters of September 27 and November 9, 1979.)

I.C.4 CONTROL ROOM ACCESS

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

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.2.2a, and letters of September 27, and November 9, 1979.)

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

Review and revise, as necessary, procedures to assure that operating experiences are fed back to operators and other personnel.

This requirement shall be met before fuel loading.

I.C.7 NSSS VENDOR REVIEW OF PROCEDURES

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

This requirement must be met before fuel loading.

I.D.1 CONTROL ROOM DESIGN

Perform a preliminary assessment of the control room to identify significant human factors deficiencies and instrumentation problems and establish a schedule approved by the NRC for correcting deficiencies.

This requirement shall be met before fuel loading.

I.G.1 TRAINING DURING LOW-POWER TESTING

Define and commit to a special low-power testing program approved by NRC to be conducted at power levels no greater than 5 percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training.

This requirement shall be met before fuel loading.

II.B.4 TRAINING FOR MITIGATING CORE DAMAGE

Develop a training program to instruct all operating personnel in the use of installed systems, including systems that are not engineered safety features, and instrumentation to monitor and control accidents in which the core may be severely damaged.

This requirement shall be met before fuel loading.

II.D.1 RELIEF AND SAFETY VALVE TEST REQUIREMENTS

Describe a test program and schedule for testing to qualify reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.1.2, and letters of September 27, and November 9, 1979.)

II.D.3 RELIEF AND SAFETY VALVE POSTION INDICATION

Install positive indication in the control room of relief and safety valve position derived from a reliable valve position detection device or a reliable indication of flow in the valve discharge pipe.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.1.3a, and letters of September 27, and November 9, 1979.)

II.E.1.2 AUXILIARY FEEDWATER INITIATION AND INDICATION

Install a control-grade system for automatic initiation of the auxiliary feedwater system that meets the single-failure criterion, is testable, and is powered from the emergency buses, and control-grade indication of auxiliary feedwater flow to each steam generator that is powered from emergency buses.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.1.7a and b, and letters of September 27 and November 9, 1979.)

II.E.4.1 CONTAINMENT-DEDICATED PENETRATIONS

Provide a design of the containment isolation system for external recombiners or purge systems for postaccident combustible gas control, if used, that is dedicated to that service only and meets the single-failure criterion.

Review and revise, if necessary, the procedures for use of combustible gas control system following an accident resulting in a degraded core and release of radioactivity into the containment.

This requirement shall be met before fuel loading. (See NUREG-0578, Sections 2.1.5a and 2.1.5c, and letters of September 27 and November 9, 1979.)

II.F.1 ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION

Provide procedures for estimating noble gas, radioiodine, and particulate release rates if the existing effluent instrumentation goes off the scale.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.1.8b, and letters of September 27 and November 9. 1979.)

II.F.2 INADEQUATE CORE COOLING INSTRUMENTS

Develop procedures to be used by operators to recognize inadequate core cooling with currently installed instrumentation in PWRS. Install a primary coolant saturation meter. Provide a description of any additional instruments or controls needed to supplement installed equipment to provide unambiguous, easy-to-interpret indication of inadequate core cooling, procedures for use of this equipment, analyses used to develop these procedures, and a schedule for installing this equipment.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.1.3b, and letters of September 27 and November 9, 1979.)

II.G EMERGENCY POWER FOR PRESSURIZER EQUIPMENT

Motive and control components of the power-operated relief valves and associated block valves and the pressurizer level indication shall be capable of being supplied from the offsite power source or from the emergency power buses when offsite power is not available.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.1.1, and letters of September 27 and November 9, 1979.)

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

C.1.5* Review all valve positions, positioning requirements, positive controls and related test and maintenance procedures to assure proper ESF functioning. (See Bulletin 79-06A Item 8, 79-06B Item 7, 79-08 Item 6.)

C.1.10 Review and modify, as required, procedures for removing safety-related systems from service (and restoring to service) to assure operability status is known. (See Bulletin 79-05A Item 10, 79-06A Item 10, 79-06B Item 9, 79-08 Item 8.)

C.1.17 For Westinghouse-designed reactors, trip the pressurizer low-level coincident signal bistables, so that safety injection would be initiated when the pressurizer low-pressure setpoint is reached

*Table C.1 of the Action Plan lists all the requirements given in IE Bulletins.

regardless of the pressurizer level. (See Bulletin 79-06A and Revision 1, Item 3.)

C.1.20 For B&W-designed reactors, provide procedures and training to operators for prompt manual reactor trip for loss of feedwater, turbine trip, main steamline isolation valve closure, loss of offsite power, loss of steam generator level, and low pressurizer level. (See Bulletin 79-05B Item 4.)

C.1.21 For B&W-designed reactors, provide automatic safety-grade anticipatory reactor trip for loss of feedwater, turbine trip or significant decrease in steam generator level. (See Bulletin 79-05B, Item 5.)

C.1.22 For boiling water reactors, describe the automatic and manual actions necessary for proper functioning of the auxiliary heat removal systems systems that are used when the main feedwater system is not operable. (See Bulletin 79-08, Item 3.)

C.1.23 For boiling water reactors, describe all uses and types of reactor vessel level indication for both automatic and manual initiation of safety systems. Describe other instrumentation that might give the operator the same information on plant status. (See Bulletin 79-08, Item 4.)

These requirements shall be met before fuel loading.

II.K.3 FINAL RECOMMENDATIONS OF B&O TASK FORCE*

C.3.9** For Westinghouse-designed reactors, modify the pressure integral derivative controller, if installed on the PORV, to eliminate spurious openings of the PORV.

* The B&O recommendations were not specifically delineated as to fuel-loading or full-power requirements prior to the review of Sequoyah, North Anna 2, and Salem 2. The NRR staff is presently confirming compliance with these four items for these plants.

**Table C.3 of the Action Plan lists the requirements derived from final recommendations of the B&O Task Force.

C.3.10 For Westinghouse-designed reactors, if the anticipatory reactor trip upon turbine trip is to be modified to be bypassed at power levels less than 50 percent, rather than below 10 percent as in current designs, demonstrate that the probability of a small-break LOCA resulting from a stuck-open PORV is not significantly changed by this modification.

C.3.11 Demonstrate that the PORV installed in the plant has a failure rate that is not significantly less than the valves for which there is an operating history.

C.3.12 For Westinghouse-designed reactors, confirm that there is an anticipatory reactor trip on turbine trip.

These requirements shall be met before fuel loading.

III.A.1.1 UPGRADE EMERGENCY PREPAREDNESS

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

This requirement shall be met before fuel loading.

III.A.1.2 UPGRADE EMERGENCY SUPPORT FACILITIES

Establish an interim onsite technical support center separate from, but close to, the control room for engineering and management support of reactor operations during an accident. The center shall be large enough for the necessary utility personnel and five NRC personnel, have direct display or callup of plant parameters, and dedicated communications with the control room, the emergency operations center, and the NRC. Provide a description of the permanent technical support center.

Establish an onsite operational support center, separate from but with communications to the control room for use by operations support personnel during an accident.

Designate a near-site emergency operations facility with communications with the plant to provide evaluation of radiation releases and coordination of all onsite and offsite activities during an accident.

These requirements shall be met before fuel loading. (See NUREG-0578, Sections 2.2.2.b, 2.2.c, and letters of September 27 and November 9, 1979 and April 25, 1980).

III.D.3.3 INPLANT RADIATION MONITORING

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.

This requirement shall be met before fuel loading. (See NUREG-0578, Section 2.1.8c, and letters of September 27 and November 9, 1979)

Part 2 - Full-Power Requirements

I.C.7 NSSS VENDOR REVIEW OF PROCEDURES

Obtain NSSS vendor review of power-ascension test and emergency procedures to further verify their adequacy.

This requirement must be met before issuance of a full-power license.

I.C.8 PILOT MONITORING OF SELECTED EMERGENCY PROCEDURES FOR NEAR-TERM OPERATING LICENSE APPLICANTS

Correct emergency procedures, as necessary, based on the NRC audit of selected plant emergency operating procedures (e.g., small-break LOCA, loss of feedwater, restart of engineered safety features following a loss of ac power, steam-line break, or steam-generator tube rupture).

This action will be completed prior to issuance of a full-power license.

I.G.1 TRAINING DURING LOW-POWER TESTING

Supplement operator training by completing the special low-power test program. Tests may be observed by other shifts or repeated on other shifts to provide training to the operators.

This requirement shall be met before issuance of a full-power license.

II.B.1 REACTOR COOLANT SYSTEM VENTS

Provide a description of the design of reactor coolant system and reactor vessel head high point vents that are remotely operable from the control room and supporting analyses. This requirement shall be met before issuance of a full-power license. (See Enclosure 4 to letters of September 27 and November 9, 1979.)

II.B.2 PLANT SHIELDING

Provide (1) a radiation and shielding design review that identifies the location of vital areas and equipment in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by radiation during operations following an accident resulting in a degraded core, and (2) a description of the types of corrective actions needed to assure adequate access to vital areas and protection of safety equipment.

This requirement shall be met before issuance of a full-power license. (See NUREG-0578, Section 2.1.6b, and letters of September 27 and November 9, 1979.)

II.B.3 POSTACCIDENT SAMPLING

Provide (1) a design and operational review of the capability to promptly obtain and perform radioisotopic and chemical analyses of reactor coolant and containment atmosphere samples under degraded core accident conditions without excessive exposure, (2) a description of the types of corrective actions needed to provide this capability, and (3) procedures for obtaining and analyzing these samples with the existing equipment.

This requirement shall be met before issuance of a full-power license. (See NUREG-0578, Section 2.1.8a and letter of September 27 and November 9, 1979).

II.B.4 TRAINING FOR MITIGATING CORE DAMAGE

Complete the training of all operating personnel in the use of installed systems to monitor and control accidents in which the core may be severely damaged.

This requirement shall be met before issuance of a full-power license.

II.E.1.1 AUXILIARY FEEDWATER SYSTEM RELIABILITY EVALUATION

(1) Provide a simplified auxiliary feedwater system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFWS failure following a main feedwater transient, with particular emphasis on potential failures resulting from human errors, common causes, single point vulnerability, and test and maintenance outage.

(2) Provide an evaluation of the AFWS using the acceptance criteria of Standard Review Plan Section 10.4.9.

(3) Describe the design basis accident and transients and corresponding acceptance criteria for the AFWS.

(4) Based on the analyses performed modify the AFWS, as necessary.

These requirements shall be met before issuance of a full-power license.

II.E.3.1 EMERGENCY POWER FOR PRESSURIZER HEATERS

Install the capability to supply from emergency power buses a sufficient number of pressurizer heaters and associated controls to establish and maintain natural circulation in hot standby conditions.

This requirement shall be met before issuance of a full-power license. (See NUREG-0578, Section 2.1.1, and letters of September 27 and November 9, 1979.)

II.E.4.2 CONTAINMENT ISOLATION DEPENDABILITY

Provide (1) containment isolation on diverse signals, such as containment pressure or ECCS actuation, (2) automatic isolation of nonessential systems (including the bases for specifying the nonessential systems), (3) no automatic reopening of containment isolation valves when the isolation signal is reset.

These requirements shall be met before issuance of a full-power license. (See NUREG-0578, Section 2.1.4, and letters of September 27 and November 9, 1979.)

II.K.2 COMMISSION ORDERS ON BABCOCK & WILCOX PLANTS

C.2.2* For B&W-designed reactors, provide procedures and training to initiate and control auxiliary feedwater independent of the integrated control system.

*Table C.2 of the Action Plan lists all of the requirements of the Commission Orders.

C.2.9 For B&W-designed reactors, provide a failure mode and effects analysis of the integrated control system. (See Commission Order.)

C.2.10 For B&W-designed reactors, install safety-grade anticipatory reactor trip for loss of feedwater and turbine trip. (See Commission Order.)

C.2.13 For B&W-designed reactors, confirm by a detailed analysis of thermal-mechanical conditions in the reactor vessel during recovery from a small-break LOCA, with an extended loss of all feedwater requiring the use of the high-pressure injection system, that vessel integrity is not jeopardized. (See letter of August 21, 1979.)

C.2.14 For B&W-designed reactors, demonstrate that the power-operated relief valves on the pressurizer will open in less than five percent of all anticipated overpressure transients using revised setpoints and anticipatory trips for the range of plant conditions which might occur during a fuel cycle. (See letter of August 21, 1979.)

C.2.15 For B&W-designed reactors, analyze the effects of slug flow on once-through steam generator tubes after primary system voiding. (See letter of August 21, 1979.)

C.2.16 For B&W-designed reactors, evaluate the effect of reactor coolant pump damage and leakage following a small-break LOCA concurrent with a loss of offsite power that results in the loss of seal cooling. (See letter of August 21, 1979.)

These requirements shall be met before issuance of a full-power license.

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

C.3.3* Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report.

*Table C.3 of the Action Plan lists all of the recommendations of the B&O Task Force.

This requirement shall be met before issuance of a full-power license.

III.A.1.1 UPGRADE EMERGENCY PREPAREDNESS

Provide an emergency response plan in substantial compliance with NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (which may be modified after May 13, 1980 based on public comments) except that only a description of and completion schedule for the means for providing prompt notification to the population (App. 3), the staffing for emergencies in addition to that already required (Table B.1), and an upgraded meteorological program (App. 2) need be provided. NRC will give substantial weight to FEMA findings on offsite plans in judging the adequacy against NUREG-0654. Perform an emergency response exercise to test the integrated capability and a major portion of the ba[illegible]ments existing within emergency preparedness plans and organizations.

This requirement shall be met before issuance of a full-power license.

III.D.1.1 PRIMARY COOLANT SOURCES OUTSIDE CONTAINMENT

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, measure actual leak rates and establish a program to maintain leakage at as-low-as-practical levels and monitor leak rates.

This requirement shall be met before issuance of a full-power license. (See NUREG-0578, Section 2.1.6a, and letters of September 27 and November 9, 1979.)

III.D.3.4 CONTROL ROOM HABITABILITY

Identify and evaluate potential hazards in the vicinity of the site as described in SRP Sections 2.2.1, 2.2.2, and 2.2.3, confirm that operators in the control room are adequately protected from these hazards and the release of radioactive gases as described in SRP Section 6.4, and, if necessary, provide the schedule for modifications to achieve compliance with SRP Section 6.4.

This requirement shall be met by issuance of a full-power license.

Part 3 - NRC Actions

I.B.2.2 REACTOR INSPECTOR AT OPERATING REACTORS

An NRC resident inspector will be assigned to each site.

This action shall be completed before fuel loading.

I.D.1 CONTROL ROOM DESIGN REVIEW

NRC review of applicant's preliminary assessment of the control room design to determine whether the assessment is adequate and identify any necessary corrections and approve the schedule for correction of the deficiencies.

This action shall be completed prior to fuel loading.

II.B.7 ANALYSIS OF HYDROGEN CONTROL

Reach a decision on the immediate requirements, if any, for hydrogen control in small containments and apply, as appropriate, to new OLs pending completion of the degraded core rulemaking in II.B.8 of the Action Plan.

This action shall be completed before issuance of a full-power license.

II.B.8 DEGRADED CORE - RULEMAKING

Issue an advance notice of rulemaking on requirements for design and other features for accidents involving severely damaged cores.

This action shall be completed before issuance of a full-power license.

III.A.3.1 ROLE OF NRC IN EMERGENCY PREPAREDNESS

More explicitly define the role of the NRC in emergency situations involving NRC licenses.

This action was completed in a meeting between the staff and the Commission on February 6, 1980.

III.A.3.3 COMMUNICATIONS

Install direct dedicated telephone lines between each plant and the NRC Operations Center.

This action shall be completed prior to fuel loading.

III.B.2 IMPLEMENTATION OF NRC AND FEMA RESPONSIBILITIES

The applicant emergency plans shall meet the requirements of Appendix E to 10 CFR 50 and, the positions in Regulatory Guide 1.101 (Mar. 1977). Offsite plans shall meet the essential planning elements in NUREG-75/111 and Supplement 1 thereto or receive a favorable finding by FEMA.

This requirement shall be met prior to fuel loading.

III.D.2.4 OFFSITE DOSE MEASUREMENTS

The NRC will place approximately 50 thermoluminescent dosimeters (TLDs) around the site in coordination with the applicant and State environmental monitoring program.

This action shall be completed prior to issuance of a full-power license.

IV.F.1 POWER-ASCENSION TEST

IE will monitor the power-ascension test program to confirm that safety is not compromised because of the expanded startup test program and economic costs of the delay in commercial operation.

This action shall be taken during the startup and power-ascension test program.

Part 4 - Dated Requirements

I.A.1.1 SHIFT TECHNICAL ADVISOR

The Shift Technical Advisor shall have a technical education, which is taught at the college level and is equivalent to about 60 semester hours in basic subjects of engineering and science, and specific training in the design, function, arrangement and operation of plant systems and in the expected response of the plant and instruments to normal operation, transients and accidents including multiple failures of equipment and operator errors.

This requirement shall be met by January 1, 1981. (See NUREG-0578, Section 2.2.1b, and letters of September 27 and November 9, 1979.)

I.A.2.1 IMMEDIATE UPGRADING OF OPERATOR AND SENIOR OPERATOR TRAINING AND QUALIFICATION

Applicants for SRO license shall have 4 years of responsible power plant experience, of which at least 2 years shall be nuclear power plant experience (including 6 months at the specific plant) and no more than 2 years shall be academic or related technical training.

Certifications that operator license applicants have learned to operate the controls shall be signed by the highest level of corporate management for plant operation.

These requirements shall be met on or after May 1, 1980. (See March 28, 1980 letter.)

Revise training programs to include training in heat transfer, fluid flow, thermodynamics, and plant transients.

This requirement shall be met by August 1, 1980. (See March 28, 1980 letter.)

I.A.2.3 ADMINISTRATION OF TRAINING PROGRAMS FOR LICENSED OPERATORS

Training instructors who teach systems, integrated responses, transient and simulator courses shall successfully complete a SRO examination.

Applications shall be submitted by August 1, 1980. (See March 28, 1980 letter.)

Instructors shall attend appropriate retraining programs that address, as a minimum, current operating history, problems and changes to procedures and administrative limitations. In the event an instructor is a licensed SRO, his retraining shall be the SRO requalification program.

Programs shall be initiated by May 1, 1980. (See March 28, 1980 letter.)

I.A.3.1 REVISE SCOPE AND CRITERIA FOR LICENSING EXAMS

Applicants for operator licenses will be required to grant permission to the NRC to inform their facility management regarding the results of examinations.

Contents of the licensed operator requalification program shall be modified to include instruction in heat transfer fluid flow, thermodynamics, and mitigation of accidents involving a degraded core.

These requirements shall be met by May 1, 1980. (See March 28, 1980 letter.)

The criteria for requiring a licensed individual to participate in accelerated requalification shall be modified to be consistent with the new passing grade for issuance of a license.

This requirement shall apply to all annual requalification examinations conducted after March 28, 1980. (See March 28, 1980 letter.)

Requalification programs shall be modified to require specific reactivity control manipulations. Normal control manipulations, such as plant or reactor startups, must be performed. Control manipulations during abnormal or emergency operations shall be walked through and evaluated by a member of the training staff. An appropriate simulator may be used to satisfy the requirements for control manipulations.

This requirement shall be met by August 1, 1980. (See March 28, 1980 letter.)

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

Analyze the design basis transients and accidents including single active failures and considering additional equipment failures and operator errors to identify appropriate and inappropriate operator actions. Based on these analyses, revise, as necessary, emergency procedures and training.

This requirement was intended to be completed in early 1980, however some difficulty in completing this requirement has been experienced. Clarification of the scope and revision of the schedule are being developed and will be issued by July 1980. It is expected that this requirement will be coupled with Task I.C.9., Long-term Upgrading of Procedures. (See NUREG-0578, Sections 2.1.3b and 2.1.9, and letters of September 27 and November 9, 1979.)

II.B.1 REACTOR COOLANT SYSTEM VENTS

Install reactor coolant system and reactor vessel head high-point vents that are remotely operable from the control room.

This requirement shall be met before January 1, 1981. (See Enclosure 4 to letters of September 27 and November 9, 1979.)

II.B.2 PLANT SHIELDING

Complete modificatons to assure adequate access to vital areas and protection of safety equipment following an accident resulting in a degraded core.

This requirement shall be met by January 1, 1981. (See NUREG-0578, Section 2.1.6b, and letters of September 27 and November 9, 1979.)

II.B.3 POSTACCIDENT SAMPLING

Complete corrective actions needed to provide the capability to promptly obtain and perform radioisotopic and chemical analysis of reactor coolant and containment atmosphere samples under degraded-core conditions without excessive exposure.

This requirement shall be met by January 1, 1981. (See NUREG-0578, Section 2.1.8a, and letters of September 27 and November 9, 1979.)

II.D.1 RELIEF AND SAFETY VALVE TEST REQUIREMENTS

Complete tests to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

This requirement shall be met by July 1, 1981. (See NUREG-0578, Section 2.1.2, and letters of September 27 and November 9, 1979.)

II.E.1.2 AUXILIARY FEEDWATER INITIATION AND INDICATION

Upgrade, as necessary, automatic initiation of the auxiliary feedwater system and indication of auxiliary feedwater flow to each steam generator to safety-grade quality.

This requirement shall be met by January 1, 1981. (See NUREG-0578, Sections 2.1.7a and b, and letters of September 27 and November 9, 1979.)

II.E.4.1 CONTAINMENT DEDICATED PENETRATION

Install a containment isolation system for external recombiners or purge systems for postaccident combustible gas control, if used, that is dedicated to that service only and meets the single-failure criterion.

This requirement shall be met before January 1, 1981. (See NUREG-0578, Section 2.1.5a and 2.1.5c, and letters of September 27 and November 9, 1979.)

II.F.1 ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION

Install continuous indication in the control room of the following parameters:

a. Containment pressure from minus 5 psig to three times the design pressure of concrete containments and four times the design pressure of steel containments;

b. Containment water level in PWRs from (1) the bottom to the top of the containment sump, and (2) the bottom of the containment to a level equivalent to 600,000 gallons of water;

   Containment water level in BWRs from the bottom to 5 feet above the normal water level of the suppression pool;

c. Containment atmosphere hydrogen concentration from 0 to 10 volume percent;

d. Containment radiation up to 10^8 Rad/hr;

e. Noble gas effluent from each potential release point from normal concentrations to 10^5 μCi/cc (Xe-133).

Provide capability to continuously sample and perform onsite analysis of the radionuclide and particulate effluent samples.

This instrumentation shall meet the qualification, redundancy, testability and other design requirements of the proposed revision to Regulatory Guide 1.97.

This requirement shall be met by January 1, 1981. (See NUREG-0578, Section 2.1.8b, and letters of September 27 and November 9, 1979.)

II.F.2 INADEQUATE CORE COOLING INSTRUMENTS

Install, if required, additional instruments or controls needed to supplement installed equipment in order to provide unambiguous, easy-to-interpret indication of inadequate core cooling.

This requirement shall be met by January 1, 1981. (See NUREG-0578, Section 2.1.3b, and letters of September 27 and November 9, 1979.)

III.A.1.2 UPGRADE EMERGENCY SUPPORT FACILITIES

Provide radiation monitoring and ventilation systems, including particulate and charcoal filters, and otherwise increase the radiation protection to the onsite technical support center to assure that personnel in the center will not receive doses in excess of 5 rem to the whole body or 30 rem to the thyroid for the duration of the accident. Provide direct display of plant safety system parameters and call up display of radiological parameters.

For the near-site emergency operations facility, provide shielding against direct radiation, ventilation isolation capability, dedicated communications with the onsite technical support center and direct display of radiological and meteorological parameters.

This requirement shall be met by January 1, 1981, although the safety parameter information requirements will be staged over a longer period of time. (See NUREG-0578, Section 2.2.2b and 2.2.2c and letters of September 27 and November 9, 1979 and April 25, 1980.)

III.D.3.3 IN-PLANT RADIATION MONITORING

Provide the equipment, training, and procedures to accurately measure the radioiodine concentration in areas within the plant where plant personnel may be present during an accident.

This requirement shall be met before January 1, 1981. (See NUREG-0578, Section 2.1.8c, and letters of September 27 and November 9, 1979.)