April 20, 1981

RULEMAKING ISSUE
(Affirmation)

SECY-81-246

FOR: The Commissioners

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: PROPOSED RULE RELATED TO TMI-2 REQUIREMENTS FOR OPERATING LICENSE APPLICATIONS

Purpose: To obtain Commission approval for publication of proposed amendments in the Federal Register for comment.

Discussion: A proposed Rule has been developed that incorporates into 10 CFR Part 50 a set of TMI-2 related requirements for operating license applications. These requirements are the same set contained in NUREG-0737. The approach taken in the proposed rule is similar to that taken for TMI-2 requirements for near-term construction permit applications.

There are two aspects of this proposed Rule that warrant highlighting.

I. Several of the items applicable to operating license application reviews, as contained in NUREG-0737 and listed below, are being addressed in other decisional processes. All NUREG-0737 items are nevertheless itemized in this proposed Rule for completeness and clarity.

1. Certain NUREG-0737 items related to Emergency Preparedness are listed only for completeness, since they have already been issued as effective regulations.

2. Several items from NUREG-0737 are being included in the Interim Degraded Core Cooling final Rule and in proposed rule changes regarding operator qualification, shift manning, and overtime that are under preparation. Since these rules have not yet been finalized, the enclosed rule includes those same items and, we

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have used the same language that appears in the latest staff draft of these rules. For the affected items, they could remain in either one of the Rules when issued.

3. The requirements as shown in the proposed rule are divided into two groups:

   a. Items that must be satisfied as a prerequisite to receiving an OL.

   b. "Dated" items that must be met by a specific date. For licenses issued before a particular dated item is required, that item and its implementation schedule would become a license condition. For defining "dated" items in this rule we used July 15, 1981, as the cut-off date.

II. In addition to the considerations in Item I, the staff recognizes that considerable experience in applying the NUREG-0737 requirements has been gained since drafting of them began about a year ago. In addition, the development over the last year, as prescribed in the Action Plan, of several other requirements has been completed such that they can now be incorporated into NUREG-0737. The enclosed proposed Rule identifies examples of such possible changes and requests comments on such actions. Included are:

   1. Changes to shift manning and overtime requirements.

   2. Changes to operator training requirements.

3. Several items for which the ongoing staff reviews, based on evaluations and reports submitted on operating reactors, may resolve our concern such that these evaluations and reports need not be submitted by operating license applicants.

4. Several items for which the ongoing staff reviews have given preliminary indications that the system modifications previously called for may not now be needed.

This evolving process has led the staff to conclude that it should consider changing a number of the requirements of NUREG-0737. It would be the staff's intent to thoroughly conduct a re-review of each item in NUREG-0737 during the public comment period on the proposed rule, and make appropriate changes in the final rule based on such a review and the public comment received.

It should be noted that the impact on ongoing licensing proceedings of codifying the existing NUREG-0737 requirements should be beneficial, possibly eliminating contentions regarding items which will be included in the rules. Should the requirements of NUREG-0737 be changed, however, slippage in OL reviews and the related hearing process appears inevitable.

Recommendation: That the Commission approve publication of the proposed Rule and set a 90 day comment period.

TAR for WJD

William J. Dircks
Executive Director for Operations

Commissioners' comments or consent should be provided directly to the Office of the Secretary by c.o.b. Monday, May 4, 1981.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT April 27, 1981, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

This paper is tentatively scheduled for affirmation at an open meeting during the week of May 11, 1981. Please refer to the appropriate Weekly Commission Schedule, when published, for a specific date and time.

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NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Licensing Requirements for Pending Operating License Applications

AGENCY: Nuclear Regulatory Commission

ACTION: Proposed rule

SUMMARY: The Nuclear Regulatory Commission is proposing to add to its power reactor safety regulations a set of licensing requirements applicable to operating license applications. The requirements stem from the Commission's ongoing effort to apply the lessons learned from the accident at Three Mile Island to power plant licensing. Each applicant covered by the rule would have to meet these requirements, together with the existing regulations, in order to obtain an operating license. Comments are sought on the requirements contained in the proposed rule.

DATE: Comments must be received on or before (90 days after publication.)

ADDRESSES: Comments should be sent to the Secretary of the Commission, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Docketing and Service Branch.

FOR FURTHER INFORMATION CONTACT: John A. Olshinski, Chief, Operating Reactors Assessment Branch, Division of Licensing, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Telephone 301-492-8069.

Supplemental Information

On March 28, 1979, the Three Mile Island Unit 2 (TMI-2) nuclear power plant experienced a loss of feedwater transient, complicated by a set of circumstances and events, culminating in the equivalent of a small-break loss-of-coolant accident with substantial core damage. The circumstances and events that caused the feedwater transient to develop into an accident include design deficiencies, equipment failures, and human errors.

In April 1979, the Commission established the Bulletin and Orders Task Force as the focal point for those TMI-2 related staff activities necessary to assure the immediate safety of all other operating power reactors. During May 1979, the efforts of this group resulted in the issuance of several IE Bulletins and Commission Orders covering a wide range of topics.

In May 1979, the Commission established the TMI-2 Lessons Learned Task Force to identify and evaluate safety concerns requiring prompt licensing actions for operating reactors (beyond the immediate actions taken as a result of the Bulletins and Orders Task Force effort), for pending operating license applications. A set of short-term recommendations offered by this task force was published as NUREG-0578 in July 1979.

In addition to these special NRC task forces, several other official groups have investigated the accident at TMI-2 and developed recommendations. These groups 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, the Special Review Group of the NRC Office of Inspection and Enforcement, the NRC's staff's Task Force on Emergency Planning, and the NRC Office of Standards Development and Nuclear Regulatory Research. Each of the investigating groups, acting independently, organized their recommendations in a different way. A steering group was appointed to organize and assess the many recommendations and to develop the "TMI-2 Action Plan", which would provide a comprehensive and integrated plan for all actions necessary to correct or improve the regulation and operation of nuclear facilities. The items identified by the Lessons Learned Task Force and many longer term generic items identified by the Bulletins and Orders Task Force were included in the Action Plan program. This Action Plan was published as NUREG-0660 in May 1980.

In reviewing the technical, schedular and cost aspects of the numerous items of the TMI Action Plan, the Commission has approved a number of actions that provide substantial additional protection which is required for public health and safety. The Commission asked the staff to obtain industry comments on the approved Action Plan items and to make appropriate revisions prior to finalizing the requirements.

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 the Action Plan was developed, although they were subsequently included in the Action Plan. 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 completed in the near future.

On May 15, 1980, after review of the last version of the Action Plan, the Commission approved a list of "Requirements for New Operating Licenses", contained in NUREG-0694. On October 28, 1980, the Commission approved a "Clarification of TMI Action Plan Requirements", now contained in NUREG-0737, which supersedes NUREG-0694. On December 16, 1980, the Commission issued a statement of policy, "Further Commission Guidance for Power Reactor Operating Licenses", which replaced a previous policy statement issued on June 16, 1980.

On September 5, 1980, the NRC sent letters regarding the new requirements approved by the Commission in its consideration of the TMI Action Plan to all licensees of operating reactors, applicants for operating licenses, and holders of construction permits. During the week of September 22, 1980 regional meetings were held to provide more detailed explanation of the new requirements and to obtain industry comments. Based upon these discussions, the finalized Action Plan requirements were issued on October 31, 1980, as NUREG-0737, which included a summarizing letter. The letter noted that NUREG-0737 includes in tabular form and with technical clarification all the post-TMI-2 requirements that had been approved at that time by the Commission, but does not constitute the totality of the TMI-2 Action Plan.

Since NUREG-0737 was issued, the Commission has determined that the new requirements should be codified into the Commission's regulations. While there is no intent to change the technical content of these requirements, the NUREG-0737 items have been re-written in language appropriate for the Commission's regulations.

Substance of the Rule

This rule, which addresses the same set of items as in NUREG-0737, imposes new safety requirements for operating license applications. The Commission has determined that these requirements must be met by all applicants for operating licenses. It should be noted, however, that there are many elements in the TMI Action Plan (NUREG-0660) not included in NUREG-0737, that have not yet been developed by the staff or acted upon by the Commission. There are also items that the Commission has directed to be the subject of further study. This rule will be augmented in the future to add new requirements as they are approved. Opportunity for public comment will be provided when such additional requirements are contemplated.

For the sake of completeness, all of the basic requirements of NUREG-0737 are incorporated in this proposed rule. It is recognized that some of the items individually are the subject of other ongoing rulemakings. The Commission does not intend to issue duplicative rules - - consolidation or other appropriate action will be taken before final rulemaking to be sure that each subject matter is addressed in only one place in the rules.

While this rule contains the basic requirements set out in NUREG-0737, it does not incorporate the entirety of the document. In particular, the rule does not contain the detailed criteria, staff positions, and guidance contained in NUREG-0737 for satisfying many of the requirements. To have included such detail would have resulted in an excessively detailed and restrictive rule. However, the Commission has reviewed NUREG-0737 and has concluded that the positions contained therein provide a basis for responding to the experience of the TMI-2 accident. Applicants may, of course, propose to satisfy the rule's requirements by a method other than that detailed in NUREG-0737, but in such cases must provide a basis for determining that the requirements of the rule have been met.

In developing this proposed rule, the Commission has recognized that there are a number of items from NUREG-0737 that merit additional study prior to being included in a final rule. For example, there are several items for which the ongoing Commission review, based on submittals by operating reactors, may resolve the concern and no additional information on these items will be needed for operating license applications. Some items may be redundant with existing regulations. Some items that are presently under Commission review with preliminary indications that either the requirement may not be needed or the specific criteria in NUREG-0737 for meeting the requirement may be revised. Finally, some items are so specific and of limited applicability that their inclusion in the regulations may not be warranted. Accordingly, while the proposed rule presently contains all items from NUREG-0737 applicable to operating license applications, comment is specifically solicited on items that may not need to be included for the reasons discussed above. The following are examples of items that have been identified as candidates for such reconsideration.

a. Generic items for which sufficient information may have already been received and no additional information may be needed from OL applicants:

   II.K.2.15 Effects of Slug Flow on OTSG Tubes

   II.K.2.17 Voiding in RCS (complete for B&W only)

   II.K.2.19 Benchmark Analysis in Sequential AFW Flow to the OTSG.

b. Items that may already be sufficiently codified in the regulations:

II.K.3.30 Upgrade of SBLOCA Model

II.K.2.31 Plant Specific Analysis to Show Conformance with 10 CFR 50.46

III.A.1.2 Upgraded Emergency Support Facilities

III A.2 Emergency Preparedness - Long Term

III.D.3.4 Control Room Habitability

c. Items for which the Commission position on acceptability in NUREG-0737 may be revised:

I.A.1.3 Overtime Limitations

I.C.6 Verify correct performance of Operating Procedures

d. Items that are presently under Commission staff review and reconsideration as to whether the modifications are needed:

II.K.3.5 Automatic RCP Trip for PWRs

II.K.1.21 Anticipatory Trip on LOFW, Turbine Trip and Low S/G Level (B&W)

II.K.2.10 Same as II.K.1.21

II.K.1.20 Procedures for Manual Trip on Specific Events (B&W)

e. Items that may be too detailed or of limited applicability to merit codifying in the regulations:

II.K.2.2 Initiation and Control of AFW Independent of ICS (B&W)

II.K.2.9 FMEA of the ICS (B&W)

II.K.3.9 Modifications to the PID Controller for W-designed Plants

II.K.3.10 Anticipatory Reactor Trip Bypass Setpoint

II.K.3.11 PORVs Manufactured by CCI, Inc.

The proposed rule includes a provision that the Commission may, for good cause shown, grant relief from the required implementation schedules on a case-by-case basis.

Based upon its extensive review and reconsideration of the issues arising as a result of the Three Mile Island accident, the Commission has decided that applications for an operating license should be measured by the NRC staff and Presiding Officers in adjudicatory proceedings against the existing regulations, as augmented by this rule. It is the Commission's view that this new rule, together with the existing regulations, form a set of regulations, conformance with which meets the requirements of the Commission for issuance of an operating license. The Commission seeks public comment on the requirements contained in this rule.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and Section 552 and 553 of Title 5 of the United States Code, the Commission proposes to amend Part 50 of Chapter I, Title 10 of the Code of Federal Regulations as follows:

1. A new paragraph (f) is added to 50.34 to read as follows:

50.34 Contents of applications, technical information

* *

(f) Additional TMI-related requirements for applications for an operating license

In addition to the requirements of paragraph (b) of this section, each application for an operating license that is to be issued after (...insert effective date of this rule...) shall meet the requirements in paragraphs (1) and (2) below.

If the applicant contends that implementation of an item on the schedule set forth in this rule is impractical for its facility, the applicant may provide information to support this contention. The Commission will evaluate this information and, based on its determination of earnest effort and good cause shown, may grant relief from the implementation schedule, on a case-by-case basis. In such cases, the Commission will impose alternative requirements suitable for that facility.

(1) For the following requirements, the application shall describe how each requirement will be implemented or satisfied prior to issuance of an operating license.

(i) The minimum shift staffing for operators, licensed and non-licensed, shall be as shown in Table 1. In addition to the staffing requirements stated in the Table, each operating shift, except during periods of cold shutdown, shall include a qualified Shift Technical Advisor (STA). In addition to the staffing requirements stated above, shift crew assignments shall include a licensed senior reactor operator to directly supervise core alterations. This licensed senior reactor operator may have fuel handling duties but shall not have other concurrent operational duties. The amount of overtime worked by plant staff members performing safety-related functions shall be limited. Other onshift staffing and emergency response capabilities shall be as shown in Table 2. The capability for augmentation of resources for emergency response functions shall be equivalent to that shown in Table 2. (I.A.1.1;* I.A.1.3; III.A.1.2)

(ii) The operator initial training and requalification programs shall include: heat transfer, fluid flow, and thermodynamics; emphasis on reactor and plant transients; and the use of all

*Alphanumeric designations correspond to the related action plan items in NUREG-0737, "Clarification of the TMI Action Plan Requirements" and NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident. They are provided herein for information only.

TABLE 1

REQUIRED SHIFT MANNING

Operating Status	One Unit, One Control Room	Two Units, One Control Room	Two Units, Two Control Rooms	Three Units, Two Control Rooms
One Unit Operating*	1 SS (SRO) 1 SRO 2 RO 2 AO	1 SS (SRO) 1 SRO 3 RO 3 AO	1 SS (SRO) 1 SRO 3 RO 3 AO	1 SS (SRO) 1 SRO 4 RO 4 AO
Two Units Operating*	NA	1 SS (SRO) 1 SRO 3 RO 3 AO	1 SS (SRO) 2 SRO 4 RO 4 AO	1 SS (SRO) 2 SRO) Only 1 SRO & 4 ROs 5 RO) required if both) units are operated) from one control) room 5 AO
All Units Operating*	NA	1 SS (SRO) 1 SRO 3 RO 3 AO	1 SS (SRO) 2 SRO 4 RO 4 AO	1 SS (SRO) 2 SRO 5 RO 5 AO
All Units Shut Down	1 SS (SRO) 1 RO 1 AO	1 SS (SRO) 2 RO 3 AO	1 SS (SRO) 2 RO 3 AO	1 SS (SRO) 3 RO 5 AO

SS - shift supervisor
SRO - licensed senior reactor operator
RO - licensed reactor operator
AO - auxiliary operator

NOTE: (1) In order to operate or supervise the operation of more than one unit, an operator (SRO or RO) must hold an appropriate, current license for each such unit.

(2) In addition to the staffing requirements indicated in the table, a licensed senior operator will be required to directly supervise any core alteration activity.

(3) See item I.A.1.1 for shift technical advisor requirements.

* Modes 1 through 4 for PWRs.
Modes 1 through 3 for BWRs.

Table 2

MINIMUM STAFFING REQUIREMENTS FOR NRC LICENSEES
FOR NUCLEAR POWER PLANT EMERGENCIES

Major Functional Area	Location	Major Tasks	Position Title or Expertise	On Shift*	Capability for Additions 30 min	Capability for Additions 60 min
Plant Operations and Assessment of Operational Aspects			Shift Supervisor (SRO)	1	--	--
			Shift Foreman (SRO)	1	--	--
			Control Room Operators	2	--	--
			Auxiliary Operators	2	--	--
Emergency Direction and Control (Emergency Coordinator)***			Shift Technical Advisor, Shift Supervisor or designated facility manager	1**	--	--
Notification/ Communication****		Notify licensee, State local and Federal personnel & maintain communication		1	1	2
Radiological Accident Assessment and Support of Operational Accident Assessment		Emergency Operations Facility (EOF) Director	Senior Manager	--	--	1
		Offsite Dose Assessment	Senior Health Physics (HP) Expertise		1	--
		Offsite Surveys		--	2	2
		Onsite (out-of-plant)		--	1	1
		In-plant surveys	HP Technicians	1	1	1
		Chemistry/Radio-chemistry	Rad/Chem Technicians	1	--	1
Plant System Engineering, Repair and Corrective Actions		Technical Support	Shift Technical Advisor	1	--	--
			Core/Thermal Hydraulics	--	1	
			Electrical	--	--	1
			Mechanical	--	--	1
		Repair and Corrective Actions	Mechanical Maintenance/	1**	--	1
			Rad Waste Operator			1
			Electrical Maintenance/	1**	1	1
			Instrument and Control (I&C) Technician	--	1	--

Table 2 (contd)

Major Functional Area	Major Tasks	Position Title or Expertise	On Shift*	Capability for Additions 30 min	60 min
Protective Actions (In-Plant)	Radiation Protection: a. Access Control b. HP Coverage for repair, corrective actions, search and rescue first-aid & firefighting c. Personnel monitoring d. Dosimetry	HP Technicians	2**	2	2
Firefighting	--	--	Fire Brigade per Technical Specifications	Local Support	
Rescue Operations and First-Aid	--	--	2**	Local Support	
Site Access Control and Personnel Accountability	Security, firefighting communications, personnel accountability	Security Personnel	All per Security plan		
		Total	10	11	15

Notes:

* For each unaffected nuclear unit in operation, maintain at least one shift foreman, one control room operator and one auxiliary operator except that units sharing a control room may share a shift foreman if all functions are covered.

** May be provided by shift personnel assigned other functions.

*** Overall direction of facility response to be assumed by EOF director when all centers are fully manned. Director of minute-to-minute facility operations remains with senior manager in technical support center or control room.

**** May be performed by engineering aide to shift supervisor.

available plant systems to control or mitigate accidents involving severe core damage. Additionally, intensive and comprehensive training exercises are to be conducted during low-power testing programs to provide experience for each operating shift. The principal instructors shall be qualified at the senior reactor operator level and shall periodically thereafter demonstrate their continued competency. An applicant for a senior reactor operator license shall participate in an NRC approved training program. In addition to the written examination and the oral examination administered in the plant, operational examinations on an appropriate simulator will be administered by the NRC. The minimum passing grade shall be 80% overall with a minimum in each technical category of 70%. The training program for all operating personnel shall include training to recognize, control and mitigate the consequences of accidents in which the core is severely damaged. In addition, each applicant shall support the development of its training program, emergency procedures and control room hardware, with applicable human engineering data. The training must include the use of all available structures, systems and components that can control or mitigate degraded core accidents. (I.A.2.1; I.A.2.3; I.A.3.1; I.G.1; II.B.4)

(iii) Corporate management directives shall be issued that emphasize the shift supervisor's role in the control room as the primary onsite manager responsible for safe operation of the plant under all conditions. Such directives shall clearly define his responsibilities and authority including his command decision authority, relative to other plant management personnel, over plant operations personnel. The shift supervisor's responsibilities shall incluoe limiting personnel access to the control room during emergencies; his administration duties shall be such that they do not detract from or are subordinate to the management responsibility for assuring the safe operation of the plant.

Training programs for shift supervisors shall strengthen both management and operational capabilities. (I.A.1.2; I.C.3; I.C.4)

(iv) An onsite independent safety engineering group of technically qualified personnel shall be provided to perform continuing systematic reviews of plant activities, including operating experience information that may indicate areas for improving plant safety. This group shall also provide recommendations and advice to an offsite high level corporate technical officer, not in the management chain for power production. (I.B.1.2)

(v) Analyses of small-break loss-of-coolant accidents and of transients and accidents that involve postulated multiple failures, consequential failures, and operator errors, which if unmitigated could lead to inadequate core cooling, shall be provided. The analyses shall be carried sufficiently into the event to identify all significant thermal/hydraulic/neutronic phenomena and to address possible failures and operator errors during the long-term cooling phase. Emergency procedure guidelines to mitigate these transients and accidents shall be provided. (I.C.1)

(vi) Administrative controls shall be provided to ensure adequate exchange of plant status information between control room operations personnel during shift and relief turnover. As a minimum, the exchanged information shall include: values of key plant parameters, availability and alignment of systems important to safety, identification of systems and components in an acceptable degraded mode of operation, and identification of systems out of service for maintenance or test. (I.C.2)

(vii) A management system shall be provided to perform the following functions: (a) review operating experience information originating both within and outside the facility; (b) promptly supply information pertinent to plant safety, including proposed procedural changes and plant modifications, to operators and other appropriate plant personnel; and (c) assure that such information is incorporated into training and requalification programs. (I.C.5)

(viii) A management system shall be provided to independently verify the proper performance of operational and maintenance activities, as a means of reducing errors that could result in or contribute to accidents. The system shall include automatic status monitoring or verification by two qualified individuals (I.C.6)

(ix) Reviews of the proposed procedures for low-power tests, power ascension tests, and emergency procedures to verify the adequacy of procedures shall be obtained from the nuclear steam system supplier. (I.C.7)

(x) Detailed reviews of the final design shall be performed to ensure that the design of the control room and control boards are in conformance with good human factors engineering principles and that information for the control room operators is presented in a manner that facilitates recognition of developing off-normal conditions, and mitigation of accidents. (I.D.1)

(xi) RESERVED

(xii) Direct position indications (open or closed) for the relief and safety valves shall be provided in the control room. (II.D.3)

(xiii) The auxiliary feedwater system (AFW) shall be evaluated including: (a) A simplified AFW reliability analysis using event-tree and fault-tree logic techniques; (b) A design review of AFW; (c) An evaluation of AFW flow design bases and criteria. (Applicable to PWRs only). (II.E.1.1)

(xiv) The design of the auxiliary feedwater system shall be such that it can be automatically and manually initiated. Indication of system flow shall be provided in the control room. (Applicable to PWRs only.) (II.E.1.2)

(xv) The design shall include the capability to promptly connect onsite electric power to: (a) pressurizer heater and associated controls sufficient to establish and maintain natural circulation in hot standby conditions, (b) pressurizer power-operated relief valves, (c) the block valves for the pressurizer power-operated relief valves, and (d) pressurizer water level instrumentation. (Applicable to PWRs only.) (II.E.3.1 & II.G.1)

(xvi) Each power reactor that relies upon external recombiners or purge repressurization systems to satisfy the requirements of 50.44 of this part shall be provided with containment penetrations for the external recombiners or purge/repressurization systems that either: (a) are dedicated to that service only, conform to the requirements of Criteria 54 and 56 of Appendix A of this part, are designed against postulated single failures and are sized to satisfy the flow requirements of the external recombiners or purge/repressurization systems, or (b) are of a combined design for use by either external recombiners or purge/

repressurization systems and other systems, conform to the requirements of Criteria 54 and 56 of Appendix A of this part, are designed against postulated single failures both for containment isolation purposes and for operation of the external recombiners or purge/pressurization systems, and are sized to satisfy the flow requirements of the external recombiners or purge/pressurization systems. (II.E.4.1)

(xvii) The containment isolation system design shall provide that: (a) all non-essential systems are isolated automatically, (b) each non-essential penetration (except instrument lines), has two isolation barriers in series, (c) the overriding (resetting) of the isolation signal shall require deliberate operator actions of at least two steps and no single sequence of operator override actions shall cause the reopening of the containment penetrations associated with more than one system or more than one purge or vent isolation valve, (d) the containment high pressure set point for initiating containment isolation is as low as is compatible with normal operation, (e) all containment purge and vent isolation valves will receive an automatic closure signal on containment high radiation. (II.E.4.2)

(xviii) A review shall be provided of all valve positions and positioning requirements and positive controls and all related test and maintenance procedures to assure proper ESF funtioning. (II.K.1.5)

(xix) Procedures for removing safety-related systems from service (and restoring to service) shall be provided to ensure that operability status will always be known by the control room operators. (II.K.1.10)

(xx) Safety injection shall be initiated when the pressurizer low pressure setpoint is reached regardless of the pressurizer level. (Applicable to Westinghouse-designed reactors only.) (II.K.1.17)

(xxi) The reactor protection system shall include anticipatory reactor trip for loss of main feedwater, turbine trip, or significant decrease in steam generator level. Procedures and associated operator training shall be provided to ensure prompt manual reactor trip for main steamline isolation valve closure, loss of offsite power, and low pressurizer level. (Applicable to Babcock & Wilcox-designed reactors only.) (II.K.1 20, II.K.1.21, and II.K.2.10)

(xxii) An analysis shall be provided to verify that the power operated relief valves on the pressurizer will open during less than five percent of all anticipated overpressure transients for the range of plant conditions which might occur during a fuel cycle. (Applicable to Babcock-Wilcox-designed reactors only.) (II.K.2.14 and II.K.3.7)

(xxiii) The design of the auxiliary heat removal systems shall be such that necessary automatic actions will occur, and manual actions can be taken, when the main feedwater system is not operable. (Applicable to BWRs only.) (II.K.1.22)

(xxiv) A description shall be provided of all reactor vessel level indications used for automatic or manual initiation of safety systems. Other instrumentation that might give the operator the same information on plant status shall also be described. (Applicable to BWRs only.) (II.K.1.23)

(xxv) Procedures and training shall be provided for operating personnel relative to initiation and control of auxiliary feedwater. (Applicable to Babcock & Wilcox-designed reactors only.) (II.K.2.2)

(xxvi) A failure modes and effects analysis of the integrated control system (ICS) shall be provided. (Applicable to Babcock and Wilcox-designed reactors only.) (II.K.2.9)

(xxvii) 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 cooler high-pressure injection system water, shall be provided to confirm that vessel integrity is not jeopardized. (Applicable to PWRs only.) (II.K.2.13) (xxviii) An analysis shall be provided of the effects of slug flow on the once-through steam generator tubes after primary system voiding. (Applicable to Babcock & Wilcox-designed plants only.) (II.K.2.15)

(xxix) An evaluation shall be provided of the potential for and impact of reactor coolant pump seal damage and leakage upon loss of offsite power. If such damage is indicated, an analysis shall be provided of the limiting small-break loss-of-coolant accident complicated by subsequent reactor coolant pump seal damage. (II.K.2.16 and II.K.3.25)

(xxx) For Westinghouse-designed facilities where the reactor trip is to be bypassed when operating below 50 percent power, an evaluation shall be provided to verify that the probability of a small break LOCA resulting from a stuck-open PORV is not significantly greater than the case where this trip is bypassed only when operating below 10 percent power. (II.K.3.10)

(xxxi) An analysis shall be provided that defines the probability of a small-break LOCA caused by a stuck-open power operated relief valve (PORV). If this probability is a significant contributor to small-break LOCAs from all causes, provide a design description for an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened. (Applicable to PWRs only.) (II.K.3.2 and II.K.3.1)

(xxxii) Any failure of a safety or relief valve shall be reported promptly to the NRC and all challenges to such valves shall be reported annually. (Applicable to PWRs only.) (II.K.3.3)

(xxxiii) An evaluation shall be provided of the automatic tripping of the reactor coolant pumps in the case of a small-break loss-of-coolant accident. (Applicable to PWRs only.) (II.K.3.5)

(xxxiv) If a proportional integral-derivative controller is installed in the power operated relief valve (PORV) control system, the control system shall be operated so as to preclude opening the PORV due to derivative action. (Applicable to Westinghouse-designed reactors only.) (II.K.3.9)

(xxxv) Complete justification shall be provided for the use of any type of pressure-operated relief valve that has failed during testing (such as those supplied by Control Components, Inc., that failed during hot functional testing at a plant). (Applicable to PWRs only.) (II.K.3.11)

(xxxvi) An anticipatory reactor-trip on turbine-trip shall be provided. (Applicable to Westinghouse-designed reactors only.) (II.K.3.12)

(xxxvii) An evaluation shall be provided of the safety effectiveness of initiating the reactor core isolation cooling system at a higher water level than that for the high pressure coolant injection system and of restarting both systems on low water level. (Applicable to BWRs only.) (II.K.3.13)

(xxxviii) The design of the HPCI/RCIC steam line pipe-break-detection circuitry shall be such that pressure spikes resulting from HPCI and RCIC system initiation will not cause inadvertent isolation of these systems. (Applicable to BWRs only.) (II.K.3.15)

(xxxix) An analysis shall be provided to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems, shall be provided. (Applicable to BWRs only.) (II.K.3.16)

(xl) The RCIC system shall automatically transfer suction to the suppression pool when the condensate storage tank level is low. (Applicable to BWRs only.) (II.K.3.22) (xli) The HPCI and RCIC systems shall be designed to withstand and operate satisfactorily during and following a complete loss of alternating current power to their support systems, including space coolers, for at least two hours. (Applicable to BWRs only.))II.K.3.24)

(xlii) The scales of the various reactor vessel water level instruments shall be referenced to the same point. (Applicable to BWRs only.))II.K.3.27)

(xliii) Small-break loss-of-coolant accident analysis methods used to comply with Appendix K to 10 CFR Part 50 shall be revised and provided that account for experimental data, including data from the Loss-of-fluid-test (LOFT) and Semiscale Test facilities. This evaluation shall consider LOFT test, (L3-6). (II.K.3.30)

(xliv) Analysis shall be provided to demonstrate that for anticipated transients complicated by the worst single failure, and assuming proper operator actions, the core remains covered or no significant fuel damage results from core uncovery. (Applicable to BWRs only). (II.K.3.44)

(xlv) Analysis shall be provided to support depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. (Applicable to BWRs only.) (II.K.3.45) (xlvi) (A) Each boiling and pressurized light-water nuclear power reactor applicant shall implement leak reduction measures to that leakage, from systems outside containment (systems that would or could contain highly radioactive fluids during and following a serious transient or accident), is eliminated or minimized to the maximum extent practicable to prevent the release of significant amounts of radioactive material during and following an accident. Consideration shall be given to reductions of potential release paths that could result from design or operator deficiencies.

(B) Each boiling and pressurized light-water nuclear power reactor licensee shall establish and implement a program of preventive maintenance to eliminate or minimize, to the maximum extent practicable, leakage from systems outside containment. This program shall include periodic (integrated) leak tests of these systems at intervals not to exceed each refueling cycle and also include (as-well-as) the reduction of potential release paths by appropriate operator training. (III.D.1.1)

(xlvii) Each boiling and pressurized light-water power reactor shall be provided with instrumentation, equipment and associated training and procedures for determining, under accident conditions, the airborne radioiodine concentration in areas within the facility where plant personnel may be present during and following an accident. (III.D.3.3) (xlviii) The control room and associated habitability systems shall be designed to adequately protect the reactor operations staff against the effects of accidental release of toxic or radioactive gases such that the nuclear plant can be operated or safely shutdown under accident conditions. Analysis based upon the final as-built conditions shall be provided to demonstrate that airborne concentrations of such hazardous fumes will permit control room operators to remain in the control room to take appropriate safety actions. (III.D.3.4)

(xlix) Dedicated emergency response facilities shall be established and maintained for command and control, support, and coordination of onsite and offsite functions during reactor accident conditions. The Technical Support Center is to provide an appropriate near-the-control-room location for those individuals who are knowledgable of and responsible for engineering and management support of reactor operations, to diagnose and evaluate plant conditions and for more orderly conduct of plant activities during emergency conditions. The Operational Support Center is to provide an area separate from the control

(2) These facilities are discussed further in Appendix E to 10 CFR 50 and in NUREG-0696, "Functional Criteria for Emergency Response Facilities."

room for shift and other support personnel (e.g., auxiliary operator, technicians, health physics personnel) to report for instructions from the control room staff. The near-site Emergency Operations facility is to provide (a) a center for analysis of plant effluents, meteorological conditions, offsite radiation measurements and for offsite dose projections, and (b) a center for coordination of all licensee onsite and offsite activities and coordination with Federal, State, and local authorities for implementation of offsite emergency plans.

(l) Plans and facilities for coping with emergencies shall be in accordance with the requirements set forth in other sections of 10 CFR Part 50. (III.A.1.1; III.A.1.2; III.A.2)

(li) The design shall ensure the capability of natural circulation in the event that depressurization of the reactor vessel, during a small break LOCA, is required. (II.K.3.46)

(2) These requirements shall be implemented either by the date indicated or before the issuance of an Operating License, whichever is later. The application shall describe how each requirement will be implemented or satisfied.

(i) Emergency procedures shall be provided to mitigate small-break loss-of-coolant accidents, and transients and accidents that involve postulated multiple failures, consequential failures, and an operator errors, which, if unmitigated, could lead to inadequate core cooling. (January 1, 1982) (I.C.1)

(ii) Each boiling and pressurized light-water nuclear power reactor shall be provided with high point vents for the reactor coolant system and reactor vessel head and other systems required to maintain adequate core cooling if the accumulation of noncondensible gases would cause their loss of function, remotely operated from the control room, to provide improved operational capability to maintain adequate core cooling following an accident. High point vents are not required, however, for the tubes in U-tube steam generators. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents and associated controls, instruments and power sources must conform to the requirements of Appendix A and Appendix B of this part. In particular, the vent system shall be designed to ensure a low probability that (1) the vents will not perform their safety functions and (2) their would be inadvertent or irreversible actuation of a vent. Furthermore, the use of these vents during and following an accident must not aggravate the challenge to the containment or the course of the accident. (July 1, 1982) (II.B.1)

(iii) Each boiling and pressurized light-water nuclear power reactor shall be provided with both adequate access to areas that may be used during and following an accident and protection of safety equipment so that an accident that results in the release of large amounts of radioactive material will not limit personnel occupancy or degrade safety equipment by the radiaction fields that may exist during and following the accident to the extent that required safety functions cannot be accomplished.

(a) The facility design must be based on a release of radioactive material from the fuel to the primary coolant system that is not less than 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium halogen inventory, and 1% of the remaining core fission products. For equipment and areas affected by the reactor coolant, it shall be assumed that the above distribution of radioactive material is intimately mixed with the coolant water except that recirculated, depressurized coolant water may be assumed to contain no noble gases. For equipment and areas affected by the containment atmosphere, it shall be assumed that not less than 100% of the core equilibrium noble gas inventory and 25% of the core equilibrium halogen inventory are uniformly dispersed in the containment atmosphere and an additional 25% of the core equilibrium halogen inventory and 1% of the remaining core fission products are uniformly distributed on surfaces exposed to the containment atmosphere.

(b) The facility design basis must be such that an individual operator will not receive more than a 5 rem whole body dose, or its equivalent to any part of the body, while performing a necessary safety function during and following an accident. (January 1, 1982) (II.B.2)

(iv) Each boiling and pressurized light-watrer nuclear power reactor shall be provided with the capability for personnel to obtain and quantitatively analyze a reactor coolant or containment atmosphere sample during and following an accident.

(a) The facility design must be based on the radioactive material release terms described in paragraph (f)(1)(iii) of this section.

(b) The design basis for the plant equipment that provides the capability to obtain and analyze a sample must be based on the assumption that it will be done promptly, and without incurring a radiation exposure to any individual in excess of 5 rem to the whole body, or its equivalent to any part of the body.

(c) The capability to quantitatively analyze a sample must be based on the use of either in-line monitoring or an onsite radiological and chemical analysis facility. If in-line monitoring is chosen, a capability must be provided for backup sampling using grab samples, and must include the capability for analyzing the samples at either an onsite or offsite facility. The analysis capability must provide, as needed, quantification of the following:

(1) Those radioisotopes necessary to indicate the extent of core damage;

(2) Hydrogen in the containment atmosphere;

(3) Total dissolved gases or dissolved hydrogen gas in the reactor coolant;

(4) Boron in the reactor coolant; and

(5) Chloride in the reactor coolant. Chloride analyses may be performed offsite and are not required to be done promptly. (January 1, 1982) (II.B.3)

(v) Qualification tests shall be conducted on the reactor coolant system relief and safety valves and, for PWRs, block valves, for all fluid conditions under operating conditions, transients and accidents. Block valves for each relief valve shall be qualified to isolate not only a leaking relief valve under normal conditions, but also any fluid flow conditions generated by a stuck-open relief valve under normal operating or accident conditions. The results of the qualification tests shall be submitted. (Applicable to PWRs only) (July 1, 1982) (II.D.1)

(vi) Accident Monitoring Instrumentation shall be provided for each boiling and pressurized light-water nuclear power reactor and shall have the capability during and following an accident for:

(a) Providing and recording in the control room a continuous indication of:

(1) Containment pressure;

(2) Hydrogen concentration in the containment atmosphere;

(3) Containment water level;

(4) Containment radiation level; and

(5) Radioactive noble gas concentrations in the plant gaseous effluents at all potential accident release paths effective.

(b) Quantifying the concentration of radioiodines and radioactive particulates in plant gaseous effluents at all potential accident release paths.

(c) All the instruments and monitoring systems used for accident monitoring shall be designed and qualified (with extended ranges) to perform their function following an accident characterized by the radioactive material release terms described in paragraph (f)(2)(iii) of this section. (January 1, 1982) (II.F.1)

(vii) (a) Each boiling and pressurized light-water nuclear power reactor licensee shall develop and implement procedures and training to be used by the operators to recognize the existence of inadequate core cooling and low coolant level in the reactor core using available instrumentation.

(b) Each pressurized light-water nuclear power reactor shall be provided with a primary coolant saturation meter (subcooling meter) that provides in the control room a continuous, recorded, on-line indication of the primary coolant saturation condition.

(c) Each boiling and pressurized light-water nuclear power reactor shall be provided with an instrumentation system such as reactor vessel water level indicators for pressurized water reactors that augment the incore thermocouples; and incore thermocouples for boiling water reactors that augment the reactor vessel water level indicators. The instrumentation system must supply to the control room a recorded, unambiguous, easy-to-interpret, indication of inadequate core cooling. The indication must cover the complete range from normal operation to complete core uncovery and give advance warning of the approach of inadequate core cooling.

(d) All instruments used to detect the existence of inadequate core cooling shall be designed and qualified to perform their function following an accident characterized by the radioactive material release terms described in paragraph (f)(2)(iii) of this section.
(January 1, 1982) (II.F.2)

(viii) An analysis shall be provided that defines the potential for voiding in the reactor coolant system during anticipated transients. (Applicable to PWRs only) (January 1, 1982) (II.K.2.17)

(ix) An analysis shall be provided of sequential auxiliary feedwater flow to the steam generators following a loss of main feedwater. (Applicable to PWRs only) (January 1, 1982) (II.K.2.19)

(x) If determined necessary as a result of the analysis required by paragraph (1)(XLi) of this section, an automatic power-operated relief valve isolation system shall be installed that will automatically cause the block valve to close when the reactor coolant system pressure falls after the PORV has opened. (Applicable to PWRs only) (This requirement shall be implemented, if found to be necessary, by the end of the first refueling 6 months after staff approval of the design.) (II.K.3.1)

(xiv) The automatic depressurization system, valves, accumulators and associated equipment instrumentation shall show to be capable of performing their intended safety functions during and following expsosure to the hostile environment of an accident situation, taking no credit for non-safety related equipment or instrumentation, and taking account for air (or nitrogen) leakage through valves. (Applicable to BWRs only.) (Janaury 1, 1982) (II.K.3.28)

(xv) Plant-specific calculations for small break loss of coolant accidents shall be provided consistent with the requirement set forth in 1(f)(i)(xLiii)