FEB 19 1982

MEMORANDUM FOR: See Attached Distribution

FROM: Darrell G. Eisenhut, Director
Division of Licensing

SUBJECT: OL RULE FOR NUREG-0737

The Division of Licensing has been assigned lead responsibility for the NUREG-0737 rulemaking activities (memorandum from H. Denton to NRR Division Directors, dated July 24, 1981). As such, we have developed the enclosed final rule for OL applicants after considering comments from the public, ACRS, and the staff.

Please review the enclosed rule and provide your comments to the Operating Reactors Assessment Branch by February 25, 1982. The schedule requires us going to the ACRS in March 1982 and this is due to the Commission in April 1982.

Original signed by

Darrell G. Eisenhut, Director
Division of Licensing

Enclosure:
As Stated

cc: w/enclosure
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L-4-1, Pt.50

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NUCLEAR REGULATORY COMMISSION

10 CFR PART 50

LICENSING REQUIREMENTS FOR PENDING OPERATING LICENSE APPLICATIONS

AGENCY: U.S. Nuclear Regulatory Commission

ACTION: Final Rule

SUMMARY: The Nuclear Regulatory Commission is amending its regulations by adding 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 has to meet these requirements, together with the existing regulations, in order to obtain an operating license.

EFFECTIVE DATE:

FOR FURTHER INFORMATION CONTACT: David M. Verrelli, Operating Reactors Assessment Branch, Division of Licensing, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washignton, D.C. 20555, telephone 301-492-8434.

Supplementary Information

On May 13, 1981 the Nuclear Regulatory Commission published in the Federal Register (46 FR 26491) a proposed rule that would add a set of licensing requirements applicable to operating license applications. The requirements stem from the Commission's ongoing effort to apply to power plant licensing the lessons learned from the accident at Three Mile Island. Additional information on the development of this rule and opportunities for public involvement in the process are discussed in 46 FR 26491.

The proposed rule asked for public comments by August 12, 1981. We acknowledged that a number of items merited additional consideration prior to being included in the final rule, and specially solicited comments on such items.

Comments were received from forty-nine sources - thirty-seven from applicants, licensees, owner's groups; six from vendors of nuclear steam supply systems; and six from other sources (AIF, UCS, NRDC, citizens). The Advisory Committee on Reactor Safeguards (ACRS) was briefed on October 29, 1981 and also contributed comments.

The Federal Register Notice of May 13, 1981 also stated that a similar rule applicable to operating reactors would soon be published. However, on August 6, 1981 the Commission decided not to proceed with the Operating Reactors Rule. On the possibility that some licensees were awaiting publication of the proposed rule for operating reactors, the comment period was extended to November 30, 1981 (46 FR 54378). Two additional comments were received.

We briefed the ACRS on , 1982, concerning the content of the final rule. ACRS comments included , and, as a result the following changes were made.

The Commission was briefed on , 1982, and contributed the following comments

As a result, the following changes in the final rule were made.

It is the Commission's view that its requirements for issuance of an operating license will be met by conforming to this new rule, together with the existing regulations.

Summary of Comments

The majority of the commenters opposed the rule in general because it might limit flexibility, negate previous agreements, duplicate other parts of the regulations, and include criteria not yet finalized by the NRC.

In recognition of these comments, two items that are not yet finalized, automatic trip of reactor coolant pumps and restart of LPCS and LPCI, have been deleted from the final rule. Five items, listed in the next section, have been deleted because we agreed that they duplicate other parts of the regulations; and seven items have been deleted from this rule and included in other parts of the regulations. Since this rule will apply to future OL applicants, it is unlikely that it will negate any previous agreements. As a result of our review of comments we have revised the proposed rule to acknowledge that alternative methods for satisfying the TMI-related requirements can be proposed by applicants. We believe the final rule provides sufficient flexibility; for example, it allows minimum staffing for emergencies to be equivalent to the table provided, and it does not provide for specific overtime limits. Therefore, we still believe the rule for OL applicants is necessary to codify the appropriate items from NUREG-0737 that must be described in future applications. As revised, it further provides for flexibility.

In addition to the general comments, many of the comments were directed toward specific items. Some commenters felt that the items concerning the Shift Technical Advisor and Independent Safety Evaluation Groups (ISEG) should be deleted since these may be only temporary requirements. After consideration of comments we determined that these items should be retained in the Final Rule; if the requirements change in the future, the rule can be revised.

Some commenters recommended deleting specific items since they are already covered in other parts of the regulations. As previously discussed, we agreed to delete five items for this reason. However, several other items were kept in the final rule because we did not agree that they were adequately covered in other parts of the regulations. For example, deletion of proposed rule Item (1)iv, accident and procedures review, was suggested because it is covered in Section 50.46. We do not agree that Section 50.46 adequately addresses all transients and accidents that could lead to inadequate core cooling, and therefore decided to retain this item in the final rule. Other examples are: space cooling for HPCI/RCIC (proposed rule item (1)xli), and common reference water level (proposed rule item (1)xlii), claimed to be covered in 10 CFR 50, Appendix A; qualification of automatic depressurization system (proposed rule item (2)xi), claimed to be covered in 10 CFR 50.46(b)(5). We reviewed these comments and have decided to retain these items in the final rule because they are not adequately covered in the other parts of the regulations.

Changes from the Proposed Rule

After reviewing the comments, word changes were made to clarify many items. The items changed include valve position indication; ISEG; reference water level; safety injection on low pressurizer pressure; reactor trip on loss of main feedwater, turbine trip, and significant decrease in steam general level.

Many comments suggested that some items be deleted, included in other parts of the regulations, or combined with the other items. As a result, the final rule has been refined to thirty items from sixty-eight items in the proposed rule. This was done as follows:

A. Deleted twenty-one items -

1. Items deleted because they only requested information or analyses to determine if design changes should be made, thus they need not be addressed by future applicants:

   Aux Feedwater Evaluation [Proposed Rule Item (1)xiii]

   Description of Vessel Level Indications in BWRs [(1)xxiv]

   FMEA on ICS [(1)xxvii]

   Effects of Slug Flow on OTSG [(1)xxviii]

   RCP Seals [(1)xxix]

   RV Challenges and Failures [(1)xxxix]

   Manual Depressurization [(1)xlv]

   ADS Actuation [(1)liii]

   Voiding in RCS [(2)viii]

   Analysis of Sequential AFW Flow [(2)ix]

2. Items deleted because they are already sufficiently codified:

   SB LOCA Methods [(1)xliii] - covered in Appendix K

   Report SV and RV Failures [(1)xxxii] - covered in LER system

   Emergency Response Facilities [(1)l] - covered in Appendix E

   Emergency Plans and Facilities [(1)li] - covered in Appendix E

   Plant-specific SB LOCA Analysis [(2)xii] - covered in 50.46

3. Items deleted because they are presently under staff review and final determination as to whether the modifications are essential has not been made:

   Auto Trip of RCP's [(1)xxxiii]

   Restart of LPCS and LPCI [(2)xv]

4. Items deleted because they no longer apply:

   Aux Feedwater Ind. of ICS [(1)xxv] - aux feedwater now considered a safety system, and, therefore, covered elsewhere.

   RCIC Suction [(1)xl] - discusses manual transfer of RCIC system suction. Automatic transfer is now required of new applicants.

5. Item deleted because its application is too limited to merit codifying in the regulations:

   Justification for use of certain PORV's [(1)xxxv] - applies only to PORV's that failed during testing.

6. Item deleted because it went beyond the original intent of TMI Action Plan:

   Design of Aux Heat Removal Systems [(1)xxiii] - Item II.K.1.22 of NUREG-0660 says "describe" system, not "design" as stated in proposed rule.

B. Shifted seven items to other parts of the regulations:

Operator Training [(1)ii] - Section 55.10 and 55.21

Control Room Design [(1)x] - Part 50, App. A, GDC 19

Dedicated Hydrogen Penetrations [(1)xvi] - Hydrogen Control Rule, Section 50.44 (published December 2, 1981, 46 FR 58484)

Control Room Habitability [(1)xlix] - Part 50, App. A, GDC 19

RCS Vents [(2)ii] - Hydrogen Control Rule, Section 50.44 (published December 2, 1981, 46 FR 58484)

Shielding [(2)iii] - Part 50, App. A, GDC 4

Post-Accident Sampling [(2)iv] - Section 50.47

C. Seventeen items combined into seven items:

Proposed Rule	Final Rule
(1)iv and (1)vii	(1)iii
(1)v and (2)i	(1)iv
(1)xviii and (1)xix	(1)xiii
(1)xxx and (1)xxxvi	(1)xix
(1)xxxi, (1)xxxiv and (2)x	(1)xviii
(1)xxxvii, (1)xxxviii, (1)xli and (2)xiii	(1)xx
(1)xlvi and (1)xlvii	(1)xxvi

Regulatory Flexibility Act

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that this rule will not, if promulgated, have a significant economic impact on a substantial number of small entities. This rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 121. Since these companies are dominant in their service areas, this rule does not fall within the purview of the Act.

Accordingly, notice is hereby given that 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 following amendments to 10 CFR Part 50 are published as a document subject to codification.

The authority citation for Part 50 reads as follows:

Authority: Sections 103, 104, 161, 182, 183, 68 Stat. 936, 937, 948, 953, 954, as amended (42 U.S.C. 2133, 2134,2201, 2232, 2233); secs. 202, 206, 88 Stat. 1244, 1246 (42 U.S.C. 5842, 5846), unless otherwise noted. Section 50.78** also issued under sec. 122, 68 Stat. 939, 42 U.S.C. 2152. Sections 50.80-50.81 also issued under sec. 184, 68 Stat. 954, as amended; 42 U.S.C. 2234. Sections 50.100-50.102 issued under sec. 186, 68 Stat. 955; 42 U.S.C. 2236. For the purposes of sec. 223, 68 Stat. 958, as amended; 42 U.S.C. 2273, § 50.54 (i) issued under sec. 161i, 68 Stat. 949; 42 U.S.C. 2201(i), and §§ 50.70-50.71 and § 50.78** issued under sec. 161o, 68 Stat. 950, as amended; 42 U.S.C. 2201(o) and the Laws referred to in Appendices.

PART 50 - DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

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

§50.34 Contents of application, technical information.

*   *   *   *   *   *

(g) Additional TMI-Related Requirements for an Operating License.

dition to the requirements of paragraph (b) of this section, an operating license to be issued after (*** insert effective date of this rule ***), the application shall describe how the requirements of subparagraphs (1) and (2) below are satisfied. If the applicant contends that a requirement is satisfied by an alternative, the applicant shall provide support for the contention. Such an alternative may be acceptable if the staff concludes that a suitable basis has been established.

(1.) Each licensee shall be in conformance with each of the following requirements not later than when power operation is authorized by the Commission.

Item No. (1)(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) to the senior operator in command for the shift, to provide prompt professionally qualified technical support in the diagnosis of off-normal events and to assess the effectiveness of safety actions to terminate or mitigate the consequences of such events. In addition to the staffing requirements stated above, shift crew assignments shall include a licensed senior reactor operator to directly supervise core alterations. 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)

[1] The alphanumeric designations correspond to the associated items of NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," and of NUREG-0737, "Clarification of the TMI Action Plan Requirements," which provides guidance on implementation of these items.

Item No. (1)(1)

The minimum shift staffing for operators, licensed and non-licensed, shall be 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) to the senior operator in command for the shift, to provide prompt professionally qualified technical support in the diagnosis of off-normal events and to assess the effectiveness of safety actions to terminate or mitigate the consequences of such events. In addition to the staffing requirements stated above, shift crew assignments shall include a licensed senior reactor operator to directly supervise core alternations. ~~This licensed senior reactor operator may have fuel handling duties~~ 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)[1]

Table 1.—Required Shift Manning

Operating mode	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 SS (SRO)	1 SS (SRO)	1 SS (SRO)
	1 SRO	1 SRO	1 SRO	1 SRO
	2 RO	3 RO	3 RO	4 RO
	2 AO	3 AO	3 AO	4 AO
Two Units Operating*	NA	1 SS (SRO)	1 SS (SRO)	1 SS (SRO)
		1 SRO	2 SRO	†2 SRO
		3 RO	4 RO	†5 RO
		3 AO	4 AO	
				5 AO
All Units Operating*	NA	1 SS (SRO)	1 SS (SRO)	1 SS (SRO)
		1 SRO	2 SRO	2 SRO
		3 RO	4 RO	5 RO
		3 AO	4 AO	5 AO
All Units Shut Down	1 SS (SRO)	1 SS (SRO)	1 SS (SRO)	1 SS (SRO)
	1 RO	2 RO	2 RO	3 RO
	1 AO	2 AO	3 AO	3 AO

SS—shift supervisor.
SRO—licensed senior reactor operator.
RO—licensed reactor operator.
AO—auxiliary operator.
†Only 1 SRO and 4 ROs required if both units are operated from one control room.
NOTE: 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.
(1) In addition to the staffing requirements indicated in the table, a licensed senior operator will be required to directly supervise any core alteration activity.
(2) See item I.A.1.1 for shift technical advisor requirements.
*Modes 1 through 4 for PWRs.
Modes 1 through 3 for BWRs.

[1] The alphanumeric designations correspond to the associated items of NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," and of NUREG-0737, "Clarification of the TMI Action Plan Requirements," which provides guidance on implementation of these items.

Table 2.—Minimum Staffing Requirements for NRC Licensees for Nuclear Power Plant Emergencies

Major functional area and 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 and 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/Rad Waste Operator	1**		1
		Electrical Maintenance/Instrument and Control (I&C) Technician	1**	1	1
		I&C Technician		1	
Protective Actions (In-Plant)	Radiation Protection: a. Access Control b. HP Coverage for repair, corrective actions, search and rescue first-aid and firefighting c. Personnel monitoring d. Dosimetry	HP Technicians	2**	2	2
Firefighting			[illegible]	[illegible]	[illegible]
Rescue Operations and First-Aid			[illegible]	[illegible]	[illegible]
Site Access Control and Personnel Accountability	Security, firefighting communications, personnel accountability	Security Personnel	[illegible]		
		Total	10	11	15

†Fire Brigade per Technical Specifications.
‡All per security plan.
§Local support.

*For each unlicensed 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. Direction 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.

** On site, capable of being in control room within 10 minutes.

Item No. (1)(ii)

The shift supervisor's responsibilities shall include assuring that personnel access to the control room is limited during emergencies; his administrative duties shall be such that they do not detract from, and they are subordinate to, the management responsibility for assuring the safety 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)

Item No. (1)(iii)

An independent safety engineering group of technically qualified personnel shall be provided to perform continuing systematic onsite 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 independent high-level corporate technical officer. In addition, the licensee shall provide a mechanism to: (A) review operating experience information originating both within and outside the facility; (B) promptly supply information pertinent to plant safety, including 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.B.1.2; I.C.5)

Item No. (1)(iv)

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. Guidelines for emergency procedures to mitigate these transients and accidents shall be submitted for approval. Emergency procedures shall be implemented. (I.C.1)

Item No. (1)(v)

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)

Item No. (1)(vi)

A 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 may include automatic status monitoring, operational testing, or verification by a second qualified individual. (I.C.6)

Item No. (1)(vii)

In carrying out his responsibility of the adequacy of procedures, the licensee shall obtain the review and comments of the nuclear-steam-system-supplies on the initial low-power tests, power ascension tests, and emergency procedures. (I.C.7)

Item No. (1)(viii)

Positive and unambiguous position indication (open or closed) shall be provided in the control room for the relief and safety valves of the reactor coolant system. (II.D.3)

Item No. (1)(ix)

The protection system shall include automatic and manual initiation of the auxiliary feedwater system and control room indication of system flow. (Applicable to Pressurized Water Reactors (PWRs) only.) (II.E.1.2)

Item No. (1)(x)

The design shall include the capability to promptly provide 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 and associated controls for the pressurizer power-operated relief valves, and (D) pressurizer water level instrumentation. (Applicable to PWRs only.) (II.E.2.1; II.G.1)

Item No. (1)(xi)

The containment isolation system design shall provide that: (A) all non-essential systems are initially isolated automatically, (B) each non-essential penetration (except instrument lines) has two isolation barriers in series, (C) the overriding ("resetting") of an isolation signal, in order to re-activate a non-essential system during a plant transient or accident, shall require deliberate operator actions of at least two steps and no single sequence of operator override actions shall permit 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, and (E) all containment purge and vent isolation valves shall receive diverse automatic closure signals including containment high radiation. (II.E.4.2)

Item No. (1)(xii)

Accident monitoring instrumentation shall be provided having 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) Performing their function following an accident characterized by the radioactive material release terms described in Criterion 4 of Appendix A to 10 CFR 50. (II.F.1)

Item No. (1)(xiii)

The removal-from-service and the return-to-service of a safety related system shall be performed under procedures that assure that the reactor operator is kept advised of the operability status of such systems. Procedures shall assure that safety-related valves are returned to the positions needed for proper operation of ESF systems following manipulations necessary for test, maintenance, etc.; valve positioning requirements and periodic surveillance actions shall assure that safety-related valves are maintained in the proper positions during all operational modes. (II.K.1.5; II.K.1.10)

Item No. (1)(xiv)

Safety injection shall be actuated when the pressurizer low pressure setpoint is reached, regardless of the pressurizer water level. (Applicable to PWRs only.) (II.K.1.17)

Item No. (1)(xv)

The reactor protection system shall include reactor trip for each of the following: loss of main feedwater, turbine trip, and significant decrease in steam generator level. Procedures and associated operator training shall be provided to ensure prompt manual (or alternatively, automatic) reactor trip for each of the following: main steamline isolation valve closure, loss of offsite power, and low pressurizer level. (Applicable to B&W-designed PWRs only.) (II.K.1.20, II.K.1.21, and II.K.2.10)

Item No. (1)(xvi)

A detailed analysis of thermal-hydraulics 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)

Item No. (1)(xvii)

The system design shall be such that the power-operated relief valves on the pressurizer will open during less than five percent of all anticipated over-pressure transients for the range of plant conditions which might occur during a fuel cycle. (Applicable to B&W-designed PWRs only.) (II.K.2.14 and II.K.3.7)

Item No. (1)(xviii)

The control system shall be designed and operated so as to minimize the spurious opening of the PORV. If the probability of a small-break LOCA caused by a stuck-open power-operated relief-valve (PORV) is a significant contributor to small-break LOCAs from all causes, an automatic PORV isolation system shall be provided that would operate when the reactor coolant system pressure falls after the PORV has opened. (Applicable to PWRs only.) (II.K.3.2; II.K.3.1; II.K.3.9)

Item No. (1)(xix)

A reactor-trip on turbine-trip shall be provided. If the reactor-trip on a turbine-trip is to be bypassed at low power levels, an evaluation shall be provided to verify that the probability of a small-break LOCA resulting from a stuck-open PORV is not significantly increased. (II.K.3.12 and II.K.3.10)

Item No. (1)(xx)

The reactor core isolation cooling system and the high pressure core cooling system shall be designed so as to restart automatically if, after the initial actuation, the reactor water level should fall again to the initial actuation setpoint. The design of the HPCI/RCIC steam line pipe-break-detection circuitry shall be such that inadvertent isolation of these systems is minimized. The high pressure core cooling and RCIC systems and their essential support features (e.g., room cooling) shall be designed such that these core cooling systems can operate satisfactorily for at least two hours following a complete loss of offsite power. The RCIC system shall automatically transfer its suction to the suppression pool when the condensate storage tank level is low. (Applicable to Boiling Water Reactors (BWRs) only.) (II.K.3.13; II.K.3.15; II.K.3.22; II.K.3.24)

Item No. (1)(xxi)

Collection of data shall be provided that will establish for ECCS and other ESF systems and equipment: (A) outage dates and durations, (B) cause of the outage, (C) systems or components involved, (D) specific corrective actions taken, and (E) changes that may improve ECCS and ESF equipment availability. (II.K.3.17)

Item No. (1)(xxii)

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

Item No. (1)(xxiii)

The automatic depressurization system, valves, accumulators and associated equipment instrumentation shall be capable of performing their intended safety functions during and following exposure 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) (II.K.3.28)

Item No. (1)(xxiv)

For anticipated operational occurrence including those that may result in a stuck-open relief valve, combined with a single failure in any system but presuming proper actions, the licensee shall show that no significant fuel damage occurs. (II.K.3.44)

Item No. (1)(xxv)

The design shall assure capability of natural circulation, if required, in the event of depressurization of the reactor vessel during a small-break LOCA, and that this capability is not significantly impaired by non-condensible gases. (II.K.3.46)

Item No. (1)(xxvi)

Design and preventive maintenance measures shall assure that leakage outside containment from systems that could contain highly radioactive fluids following an accident are limited and are minimized to the maximum extent practicable. (III.D.1.1)

Item No. (1)(xxvii)

Instrumentation, equipment and associated training and procedures shall be provided 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)

(2) Each licensee shall be in conformance with each of the following requirements not later than the date indicated or when power operation is authorized by the Commission, whichever is later.

Item No. (2)(i)

Reactor coolant system relief and safety valves and, for PWRs, block valves, shall be qualified by type testing for all fluid conditions under normal 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. (II.D.1) (July 1, 1982)

Item No. (2)(ii)

The design of automatic depressurization system shall be such that any operation of this system needed to assure adequate core cooling will be initiated automatically. For licenses issued prior to April 1, 1983, the design shall be installed not later than the first refueling outage that is at least six months subsequent to staff approval of the design. (II.K.3.18) (Applicable to BWRs only)

Item (2)(iii)

(A) The 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 instrumentation available in the control room. (B) Each pressurized water 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 power reactor shall be provided with an instrumentation system, for example, 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 in the control room to provide 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 Criterion 4 of Appendix A to 10 CFR 50. (II.F.2) (First refueling outage after January 1, 1983.)

2. Paragraph (b) of §50.47 is amended by the addition of the following to item (9).

§50.47 Emergency Plans

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(b)(9) * * * * * *

Each boiling and pressurized light-water nuclear power reactor shall be provided with the capability for personnel to obtain and 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 Criterion 4 of Appendix A to 10 CFR 50. (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 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 of 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 clad 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. (II.B.3)[1]

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[1]The alphanumeric designations correspond to the associated items of NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," and of NUREG-0737, "Clarification of the TMI Action Plan Requirements," which provides guidance on implementation of these items.

3. Paragraph (l) of §50.54 is amended to read as follows:

§50.54 Conditions of Licenses

* * * * * *

(l) The licensee shall designate individuals to be responsible for directing the licensed activities of licensed operators. The designated individual on duty shall be the primary onsite manager responsible for the safe operation of the plant under all conditions. His responsibilities and authorities including his command decision authority, relative to other plant management personnel, over plant operations personnel shall be clearly defined. During all emergency conditions, the designated individual shall be in the control room to direct plant operations. These individuals shall be licensed as senior operators pursuant to Part 55 of this chapter. (I.C.3)[1]

[1] The alphanumeric designations correspond to the associated items of NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," and of NUREG-0737, "Clarification of the TMI Action Plan Requirements," which provides guidance on implementation of these items.

4. Criterion 4 of Appendix A to 10 CFR 50 is amended to add the following paragraph.

Appendix A - General Design Criteria for Nuclear Power Reactors

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Criterion 4 - Environmental and missile design basis.

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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 radiation 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. (II.B.2)[1]

[1] The alphanmeric designations correspond to the associated items of NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," and of NUREG-0737, "Clarification of the TMI Action Plan Requirements," which provides guidance on implementation of these items.

5. Criterion 19 of Appendix A to 10 CFR 50 is amended by adding the following two paragraphs directly after the present first paragraph.

Appendix A - General Design Criteria for Nuclear Power Reactors

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Criterion 19 - Control Room

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Habitability systems shall be provided for the control room to protect personnel against the effects of accidental release of toxic fumes or radioactive gases such that the nuclear power unit can be either safely operated or safely shutdown under any condition, including loss-of-coolant accidents. (III.D.3.4)[1]

The final design of the control room and control boards shall conform with good human factors engineering principles and information for the operator shall be presented in a manner that facilitates recognition of developing off-normal conditions, and mitigation of accidents. (I.D.1)[1]

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[1] The alphanumeric designations correspond to the associated items of NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," and of NUREG-0737, "Clarification of the TMI Action Plan Requirements," which provides guidance on implementation of those items.

6. §55.10 of 10 CFR 55 is amended as follows.

§55.10 Contents of Application

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(a)(6) Evidence that the applicant has learned to operate the controls in a competent and safe manner and has need for an operator or senior operator license in the performance of his duties. The applicant must have completed a course of training that has been approved by the Commission. For an applicant for a senior operator license, the applicant shall also have obtained significant experience as a licensed operator, or shall also have earned a 4-year college degree in engineering or science (I.A.2)[1]. The Commission may accept as proof of this, a certification of an authorized representative of the facility license where the applicant's services will be utilized. ...; and

[1] The alphanmeric designations correspond to the associated items of NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," and of NUREG-0737, "Clarification of the TMI Action Plan Requirements," which provides guidance on implementation of these items.

7. Two new items (m) and (n) are added to §55.21 as follows:

§55.21 Content of Operator Written Examination

* * * * * *

(m) Heat transfer, fluid flow, thermo-dynamics; reactor and plant transisnts. (I.A.2)[1]

(n) Recognition, control, and mitigation of consequences of accidents in which the core is severely damaged. (I.A.2)[1]

[1] The alphanmeric designations correspond to the associated items of NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," and of NUREG-0737, "Clarification of the TMI Action Plan Requirements," which provides guidance on implementation of these items.

Samuel J. Chilk
Secretary of the Commission

Dated at Washington, D.C., this day
of , 1982