NUREG-0718
Rev. 1

Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
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ABSTRACT

The TMI-2 Action Plan, NUREG-0660, does not specifically address requirements for construction permit and manufacturing license applications. There are currently pending six construction permit applications for eleven units with light water reactors and one manufacturing license application for eight floating nuclear plants. Staff review of these applications had been suspended since the TMI 2 accident pending the formulation of a policy to appropriately reflect the lessons learned from the accident.

The Commission is considering a new rule which will state the TMI-related requirements to be applied to these applications. NUREG-0718 was issued, and has now been revised, to provide guidance that the NRC staff believes should be followed to account for the lessons learned from the TMI-2 accident. NUREG-0718 is not a substitute for the regulations, and compliance is not a requirement. However, an approach or method different from the guidance contained herein will be accepted only if the substitute approach or method provides an equivalent basis for meeting the requirements.

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I. INTRODUCTION

After the accident at Three Mile Island, Unit 2, on March 28, 1979, the Commission directed its technical review resources to assuring the safety of operating power reactors rather than to the issuance of new licenses. Furthermore, the Commission decided that power reactor licensing should not continue until the assessment of that accident had been substantially completed and comprehensive improvements in both the operation and regulation of nuclear power plants had been set in motion.

Following the accident at Three Mile Island, Unit 2, the President established a Commission to make recommendations regarding changes necessary to improve nuclear safety. In May 1979, the Nuclear Regulatory Commission established a Lessons Learned Task Force to determine what actions were required for new operating licenses and chartered a Special Inquiry Group to examine all facets of the accident and its causes. These groups have published their reports.

The Lessons Learned Task Force led to NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" and NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report." Following release of the report of the Presidential Commission, the Commission provided a preliminary set of responses to the recommendations in that report. This response provided broad policy directions for development of an NRC Action Plan, work on which was begun in November 1979. During the development of the Action Plan, the Special Inquiry Group Report was received, which had the benefit of review by panels of outside consultants representing a cross section of technical and public views. This report provided additional recommendations.

NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," was developed to provide a comprehensive and integrated plan for the actions judged appropriate by the Nuclear Regulatory Commission to correct or improve the regulation and operation of nuclear facilities based on the experience from the accident at Three Mile Island, Unit 2, and the official studies and investigations of the accident. In developing the Action Plan, the various recommendations and possible actions of all the principal investigations were assessed and either rejected, adopted or modified.

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. Such actions came from the Bulletins and Orders issued immediately after the the accident, the first report of the Lessons Learned Task Force issued in July 1979, the recommendations of the Emergency Preparedness Task Force, and the NRC staff and Commission. Before these immediate actions were applied to operating plants, they were approved by the Commission. Most of the required immediate actions have already been taken by licensees.

On February 7, 1980, based on its review of initial drafts of the Action Plan, the Commission approved a listing of near-term operating license (NTOL) requirements as being necessary, but not necessarily sufficient, TMI-related requirements for granting new operating licenses. The fuel load requirements on the NTOL list were used by the Commission in granting operating licenses for three plants, with limited authorizations for fuel loading and low power testing.

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," now contained in NUREG-0694, which the staff recommended for imposition on current operating license applications. That list was recast from the previous NTOL list and sets forth the TMI-related requirements and actions for new operating licenses. In a Statement of Policy issued on June 16, 1980, the Commission concluded that the list of TMI-related requirements for new operating licenses found in NUREG-0694 is necessary and sufficient for responding to the TMI-2 accident. The Commission has decided that current operating license applications should be measured against the regulations, as augmented by these requirements. Subsequently, the staff incorporated all of the TMI-related items for operating reactor licensees and operating license applicants in one document, NUREG-0737, which was reviewed and approved by the Commission on October 28, 1980. This report was issued by letter on October 31, 1980; and the Commission issued a Statement of Policy on December 16, 1980, adopting NUREG-0737 in place of NUREG-0694 for operating license applications.

The TMI-2 Action Plan, NUREG-0660 does not specifically address requirements for constru[illegible]n permits (CP) or manufacturing license (ML) applications. There are curren[illegible] pending six CP applications for eleven units with light water reactors and one ML application for eight floating nuclear plants. The NRC staff review of these applications had been suspended since the TMI-2 accident pending the formulation of a licensing policy to appropriately reflect the lessons learned from the accident. Therefore, the NRC staff initiated a program to propose for Commission approval a course of action that would lead to the establishment of TMI-2 related requirements for these applications. Those requirements are described in NUREG-0718, which was issued in draft for public comment in August 1980, and in the final report dated March 1981. Subsequently, some revisions in the requirements have been made by the staff in the course of developing a proposed rule for the pending applications. The items in this document include the revisions.

II. ASSESSMENT OF TMI-2 ACTION PLAN FOR PENDING CP AND ML APPLICATIONS

In order to assess the extent to which the TMI-2 Action Plan should be implemented on the seven pending CP and ML applications, the staff developed five requirement categories. Each of the TMI-2 Action Plan requirements was carefully evaluated and then assigned to one of these five categories. A discussion of each of the requirement categories follows.

Category 1

A requirement of a type not applicable to the pending CP or ML applications for any of the following reasons:

a. It can only be addressed in operating license applications or by licensees;

b. It is not directed to CP or ML applicants;

c. It does not apply to plants of the type now pending;

d. It has been (or will be) superseded by another requirement in the Action Plan or in the regulations;

e. It has already been completed.

Category 2

A requirement of the type customarily left for the operating license stage.

Category 3

Studies (and other research and development activities to provide design development information) of the type customarily left for review at the operating license stage. However, to satisfy 50.35(a)(3) the staff believes that items in this category should be completed as early as is practicable so that the results can be most effectively taken into account in developing final design details. The applicant should provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and a program to assure that the results of such studies are factored into the final design.

Category 4

A requirement to demonstrate that any additional design, development and implementation necessary to satisfy the requirement (or to satisfy the goals of the task whose requirements are to be developed in the future) will be satisfactorily completed by the operating license stage. This is the type of information customarily required at the construction permit stage to satisfy 50.35(a)(2), or to satisfy ALAB-444 with respect to generic issues.

Category 5

A requirement for information of the type customarily reviewed at the preliminary design stage for the following types of items:

a. Items for which the required information should be sufficient to demonstrate that the requirement has been satisfied by the application. This is the kind of information and degree of detail customarily provided at the preliminary design stage with respect to site and major systems and structures to satisfy 50.34(a)(1). This will also be applicable to items relating to technical qualifications of the applicant and its management for design and construction.

b. Items for which the required information should be sufficient to assure that the requirement will be met at the final design stage. This is the kind of information and degree of detail customarily provided at the preliminary design stage with respect to the preliminary design of the facility to satisfy 50.34(a)(3)(4), etc.

Tables 1, C.1, C.2, and C.3 from NUREG-0660 list each of the TMI-2 Action Plan requirements. Appendix A of this report is a reprint of these tables with the NRC staff's category assignments for the pending CP and ML applications.

Appendix B provides a description of the specific information to be provided by CP and ML applicants for each of the Action Plan requirements assigned to Categories 3, 4, and 5.

APPENDIX A

REQUIREMENT CATEGORY ASSIGNMENTS

FOR PENDING CONSTRUCTION PERMIT AND MANUFACTURING LICENSE APPLICATIONS

TABLE 1 - PRIORITIES AND STATUS OF ITEMS IN TMI-2 ACTION PLAN

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
I.	Operational Safety		
I.A	Operating Personnel		
I.A.1	Operating Personnel and Staffing		
	1. Shift Technical Advisor	2/1c	
	2. Shift Supervisor Admin. Duties	2/1c	
	3. Shift Manning	2/1/c	
	4. Long-Term Upgrading	1b/1b	Refer to Action Plan Item I.B.1.1
I.A.2	Training and Qualifications of Operating Personnel		
	1. Immediate Upgrading of Operating and Senior Operator Training and Qualifications	1d/1c	Refer to Action Plan Item I.B.1.1
	2. Training and Qualifications of Operations Personnel	1d/1c	Refer to Action Plan Item I.B.1.1
	3. Administration of Training Programs for Licensed Operators	1b/1c	

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
	4. NRR Participation in IE Inspector Training	1b/1c	
	5. Plant Drills	2/1c	
	6. Long-Term Upgrading of Training and Qualifications	1d/1c	Refer to Action Plan Item I.B.1.1
	7. Accreditation of Training Institutions	1b/1c	
I.A.3	Licensing and Requalification of Operating Personnel		
	1. Revise Scope and Criteria for Licensing Exams	2/1c	Refer to Action Plan Item I.A.3.2
	2. Operator Licensing Program Changes	1b/1c	
	3. Requirements for Operator Fitness	1b/1c	
	4. Licensing of Additional Operations Personnel	1b/1c	
	5. Establish Statement of Understanding with INPO and DOE	1b/1c	
I.A.4	Simulator Use and Development		
	1. Initial Simulator Improvement	1d/1c	Refer to Action Plan Item I.A.4.2

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
	2. Long-Term Training Simulator Upgrade	4/1c	Refer to Appendix B
	3. Feasibility Study of Procurement of NRC Training Simulator	1b/1b	
	4. Feasibility Study of NRC Engineering Computer	1b/1b	
I.B	Support Personnel		
I.B.1	Management for Operations		
	1. Organization and Management Long-Term Improvements	2/1a	
	2. Evaluation of Organization and Management Improvements of NTOL Applicants	1d/1c	Refer to Action Plan Item I.B.1.1
	3. Loss of Safety Function	1b/1c	
I.B.2	Inspection of Operating Reactors		
	1. Revise IE Inspection Program	1b/1c	
	2. Resident Inspector at Operating Reactors	1b/1b	
	3. Regional Evaluations	1b/1b	
	4. Overview of Licensee Performance	1b/1b	

TABLE 1 (Contirued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
I.C	Operating Procedures		
	1. Short-Term Accident Analysis and Procedures Revision	2/1c	
	2. Shift and Relief Turnover Procedures	2/1c	
	3. Shift Supervisor Responsibilities	2/1c	
	4. Control Room Access	2/1c	
	5. Procedures for Feedback of Operating Experience	5/5	Refer to Appendix B
	6. Procedures for Verification of Correct Performance of Operating Activities	2/1c	
	7. NSSS Vendor Review of Procedures	2/1c	
	8. Pilot Monitoring of Selected Emergency Procedures for NTOL Applicants	1b/1c	
	9. Long-Term Program Plan for Upgrading of Procedures	4/1c	Refer to Appendix B
I.D	Control Room Design		
	1. Control Room Design Reviews	4/4	Refer to Appendix B
	2. Plant Safety Parameter Display Console	4/4	Refer to Appendix B

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
	3. Safety System Status Monitoring	4/4	Refer to Appendix B
	4. Control Room Design Standard	lb/lb	
	5. Improved Control Room Instrumentation Research	lb/lb	
	6. Technology Transfer Conference	lb/lb	
I.E	Analysis and Dissemination of Operating Experience		
	1. Office for Analysis and Evaluation of Operation Data	lb/lb	
	2. Program Office Operational Data Activities	lb/lb	
	3. Operational Safety Data Analysis	lb/lb	
	4. Coordination of Licensee, Industry, and Regulatory Programs	ld/ld	
	5. Nuclear Plant Reliability Data System	lb/lb	
	6. Reporting Requirements	lb/lb	
	7. Foreign Sources	lb/lb	
	8. Human Error Rate Analysis	lb/lb	

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
I.F	Quality Assurance		
	1. Expand QA List	5/5	Refer to Appendix B
	2. Develop More Detailed QA Criteria	5/5	Refer to Appendix B
I.G	Preoperational and Low-Power Testing		
	1. Training Requirements	1b/1c	
	2. Scope of Test Program	1b/1c	
II.	Siting and Design		
II.A	Siting		
	1. Siting Policy Reformulation	1b/1c	
	2. Site Evaluation of Existing Facilities	1d/1c	Refer to Appendix B
II.B	Consideration of Degraded or Melted Cores in Safety Review		
	1. Reactor Coolant System Vents	4/4	Refer to Appendix B
	2. Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-accident Operation	4/4	Refer to Appendix B
	3. Post-accident Sampling	4/4	Refer to Appendix B

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
	4. Training for Mitigating Core Damage	2/1c	
	5. Research on Phenomena Associated with Core Degradation and Fuel Melting	1b/1b	
	6. Risk Reduction for Operating Reactors at Sites with High Population Densities	1c/1c	
	7. Analysis of Hydrogen Control	1d/1d	Refer to Action Plan Item II.B.8
	8. Rulemaking Proceeding on Degraded Core Accidents	5/5	Refer to Appendix B
II.C	Reliability Engineering and Risk Assessment		
	1. Interim Reliability Evaluation Program (IREP)	1b/1b	
	2. Continuation of IREP	1b/1b	
	3. Systems Interaction	1c/1c	
	4. Reliability Engineering	1d/1d	
II.D	Reactor Coolant System Relief and Safety Valves		
	1. Testing Requirements	4/4	Refer to Appendix B

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
	2. Research on Relief and Safety Valve Test Requirements	1b/1b	
	3. Relief and Safety Valve Position Indication	4/4	Refer to Appendix B
II.E	System Design		
II.E.1	Auxiliary Feedwater System		
	1. Auxiliary Feedwater System Evaluation	3/3	Refer to Appendix B
	2. Auxiliary Feedwater System Automatic Initiation and Flow Indication	4/4	Refer to Appendix B
	3. Update Standard Review Plan and Develop Regulatory Guide	1b/1b	
II.E.2	Emergency Core Cooling System		
	1. Reliance on ECCS	2/1b	
	2. Research on Small Break LOCAs and Anomalous Transients	1b/1b	
	3. Uncertainties in Performance Predictions	1b/1b	

TABLE 1 (Continued)

ACTION ITEM		REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
II.E.3	Decay Heat Removal		
	1. Reliability of Power Supplies for Natural Circulation	4/4	Refer to Appendix B
	2. Systems Reliability	1b/1b	
	3. Coordinated Study of Shutdown Heat Removal	1d/1d	Refer to Action Plan II.C.4
	4. Alternate Concepts Research	1b/1b	
	5. Regulatory Guide	1b/1b	
II.E.4	Containment Design		
	1. Dedicated Penetrations	5/5	Refer to Appendix B
	2. Isolation Dependability	4/4	Refer to Appendix B
	3. Integrity Check	2/1c	
	4. Purging	4/4	Refer to Appendix B
II.E.5	Design Sensitivity of B&W Reactors		
	1. Design Evaluation	4/1c	Refer to Appendix B
	2. B&W Reactor Transient Response Task Force	1d/1c	Deleted - future requirements will be established if necessary

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
II.E.6	In-situ Testing of Valves		
	1. Test Adequacy Study	1b/1b	
II.F	Instrumentation and Control		
	1. Additional Accident Monitoring Instrumentation	4/4	Refer to Appendix B
	2. Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	4/4	Refer to Appendix B
	3. Instrumentation for Monitoring Accident Conditions (Reg. Guide 1.97)	4/4	Refer to Appendix B
	4. Study of Control and Protective Action Design Requirements	1b/1b	
	5. Classification of Instrumentation, control and Electric Equipment	1b/1b	
II.G	Electrical Power		
	1. Power Supplies for Pressurized Relief Valves, Block Valves, and Level Indicators	4/4	Refer to Appendix B
II.H	TMI-2 Cleanup and Examination		
	1. Maintain Safety of TMI-2 and Minimize Environmental Impact	1b/1b	

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
	2. Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	1b/1b	
	3. Evaluate and Feedback Information Obtained from TMI	1b/1b	
	4. Determine Impact of TMI on Socioeconomic and Real Property Values	1b/1b	
II.J	General Implications of TMI for Design and Construction Activities		
II.J.1	Vendor Inspection Program		
	1. Establish a Priority System for Conducting Vendor Inspections	1b/1b	
	2. Modify Existing Vendor Inspection Program	1b/1b	
	3. Increase Regulatory Control Over Present Non-licenses	1b/1b	
	4. Assign Residents Inspectors to Vendors and Architect-Engineers	1b/1b	
II.J.2	Construction Inspection Program		
	1. Reorient Inspection Program	1b/1b	

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
	2. Increase Emphasis on Independent Measurement in the Construction Inspection Program	1b/1b	
	3. Assign Resident Inspectors to all Construction Sites	1b/1b	
II.J.3	Management for Design and Construction		
	1. Organization and Staffing to Oversee Design and Construction	5/5	Refer to Appendix B
	2. Issue Regulatory Guide	1b/1b	
II.J.4	Revise Deficiency Reporting Requirements		
	1. Revise Deficiency Reporting Requirements	1b/1b	
II.K	Measures to Mitigate Small-Break LOCAs and Loss of Feedwater Accidents		
	1. IE Bulletins	See Table 1A	
	2. Commission Orders on B&W plants	See Table 1B	
	3. Final Recommendations of B&O Task Force	See Table 1C	

TABLE 1 (Continued)

ACTION ITEM		REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
III.	Emergency Preparations and Radiation Effects		
III.A	NRC and Licensee Preparedness		
III.A.1	Improve Licensee Emergency Preparedness - Short-term		
	1. Upgrade Emergency Preparedness	1b/1b	
	2. Upgrade Licensee Emergency Support Facilities	4/4	Refer to Appendix B;
	3. Maintain Supplies of Thyroid Blocking Agent (Potassium Iodide)	2/1c	Refer to NUREG-0654, Rev. 1; Stockpiling for the general public is under consideration
III.A.2	Improving Licensee Emergency Preparedness - Long-term		
	1. Amend 10 CFR 50 and 10 CFR 50, Appendix E	1d/1d	Complete. Refer to amended regulations.
	2. Development of Guidance and Criteria	1e/1b	
III.A.3	Improving NRC Emergency Preparedness		
	1. NRC Role in Responding to Nuclear Emergencies	1b/1b	
	2. Improve Operations Centers	1b/1b	

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
	3. Communications	1d/1c	Refer to Appendix B
	4. Nuclear Data Link	1b/1b	
	5. Training, Drills, and Tests	2/1c	
	6. Interaction of NRC with Other Agencies	1b/1b	
III.B	Emergency Preparedness of State and Local Governments		
	1. Transfer of Responsibilities to FEMA	1b/1b	
	2. Implementation of NRC's and FEMA's Responsibilities	1b/1b	
III.C	Public Information		
	1. Have Information Available for the News Media and the Public	1b/1b	
	2. The Office of Public Affairs will Develop Agency Policy and Provide Training for Interfacing with the News Media and Other Interested Parties	1b/1b	

TABLE 1 (Continued)

ACTION ITEM		REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
III.D	Radiation Protection		
III.D.1	Radiation Source Control		
	1. Primary Coolant Sources Outside the Containment Structure	4/4	Refer to Appendix B
	2. Radioactive Gas Management	1b/1b	
	3. Ventilation System and Radioiodine Adsorber Criteria	1b/1b	
	4. Radwaste System Design Features to Aid in Accident Recovery and Decontamination	1b/1b	
III.D.2	Public Radiation Protection Improvement		
	1. Radiological Monitoring of Effluents	1b/1b	
	2. Radioiodine, Carbon-14, and Tritium Pathway Dose Analysis	1b/1b	
	3. Liquid Pathway Radiological Control	2/1b	
	4. Offsite Dose Measurements	2/1c	
	5. Offsite Dose Calculation Manual	2/1b	

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
	6. Independent Radiological Measurements	1b/1b	
III.D.3	Worker Radiation Protection Improvements		
	1. Radiation Protection Plans	2/2	
	2. Health Physics Improvements	1b/1b	
	3. Inplant Radiation Monitoring	4/4	Refer to Appendix B
	4. Control Room Habitability	4/4	Refer to Appendix B
	5. Radiation Worker Exposure Data Base	1a/1c	
IV.	Practices and Procedures		
IV.A	Strengthen Enforcement Process		
	1. Seek Legislative Authority	1b/1b	
	2. Revise Enforcement Policy	1b/1b	
IV.B	Issuance of Instructions and Information to Licensees		
IV.B.1	Revise Practices for Issuance of Instructions and Information to Licensees	1b/1b	

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
IV.C	Extend Lessons Learned to Licensed Activities Other than Power Reactors		
IV.C.1	Extend Lessons Learned from TMI to other NRC Programs	1b/1b	
IV.D	NRC Staff Training		
IV.D.1	NRC Staff Training	1b/1b	
IV.E	Safety Decision-Making		
	1. Expand Research on Quantification of Safety Decision-Making	1b/1b	
	2. Plan for Early Resolution of Safety Issues	1b/1b	
	3. Plan for Resolving Issues at Construction Permit Stage	1b/1b	
	4. Resolve Generic Issues by Rulemaking	1b/1b	
	5. Assess Currently Operating Reactors	1b/1b	
IV.F	Financial Disincentive to Safety		
	1. Increased IE Scrutiny of Power Ascension Test Program	1b/1b	

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
	2. Evaluate the Impact of Financial Disincentives to the Safety of Nuclear Power Plants	1b/1b	
IV.G	Improve Safety Rulemaking Procedures		
	1. Develop a Public Agenda for Rulemaking	1b/1b	
	2. Periodic and Systematic Reevaluation of Existing Rules	1b/1b	
	3. Improve Rulemaking Procedures	1b/1b	
	4. Study Alternative for Improved Rulemaking Process	1b/1b	
IV.H	NRC Participation in the Radiation Policy Council		
V.	NRC Policy, Organization and Management		
	1. Develop NRC Policy Statement on Safety	1b/1b	
	2. Study Elimination on Non-safety Responsibilities	1b/1b	
	3. Strengthen Role of ACRS	1b/1b	
	4. Study Need for Additional Advisory Committees	1b/1b	

TABLE 1 (Continued)

ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
5. Improve Public and Intervenor Participation in Hearing Process	lb/lb	
6. Study Construction-During-Adjudication Rules	lb/lb	
7. Study Need for TMI-Related Legislation	lb/lb	
8. Study the Need to Establish an Independent Nuclear Safety Board	lb/lb	
9. Study the Reform of the Licensing Process	lb/lb	
10. Study NRC Top Management Structure and Process	lb/lb	
11. Reexamine Organization and Functions of NRC Offices	lb/lb	
12. Revise Delegations of Authority to Staff	lb/lb	
13. Clarify and Strengthen the Respective Roles of Chairman, Commission, and EDO	lb/lb	
14. Authority to Delegate Emergency Response Functions to a Single Commissioner	lb/lb	

TABLE 1 (Continued)

ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
15. Achieve Single Location - Long-term	1b/1b	
16. Achieve Single Location - Interim	1b/1b	
17. Reexamine Commission Role in Adjudication	1b/1b	

TABLE 1A OFFICE OF INSPECTION AND ENFORCEMENT BULLETINS

	Requirement	Source for Operating Reactors	Applicability	CP/ML Requirement Category Assignment	Comments
1.	Review TMI-2 PNs and detailed chronology of TMI-2 accident	79-50&-5A (Item 1) 79-06&06A (Item 1) 79-06&06B (Item 1)	BWR and PWR	1d/1d	Refer to Action Plan Items 1.A.2.2 and I.A.3.1
2.	Review transients similar to TMI-2 that have occurred at other facilities and NRC evaluation of Davis-Besse transient.	79-05&05A (Item 2)	B&W	1d/1c	Refer to Action Plan Items I.A.2.2 and I.A.3.1
3.	Review operating procedures for recognizing, preventing, and mitigating void formation in transients and accidents	79-05&05A (Item 3) 79-06&06A (Item 2) 79-06&06B (Item 2)	PWR	1d/1c	Refer to Action Plan Item I.C.1
4.	Review operating procedures and training instructions to ensure that: a. Operators to not override ESF actions unless continued operation is unsafe;	79-05&05A (Item 4.a) 79-05B (Item 2) 79-06A (Item 7.a) 79-06B (Item 6.a) 79-08 (Item 5.a)	PWR and BWR	1d/1c	Refer to Action Plan Items I.C.1, I.C.7, I.C.8, and I.G.1
	b. HPI system operation	NUREG-0645 (App. G) NUREG-0565 (Rec. 104) 69-110 6002-00 (11/1/79) 69-110 6003-00 (11/20/79) 69-110 6001-00 (11/1/79)	W,CE B&W ANO-1 Davis-Besse 1 Oconee 1, 2 & 3 Crystal River 3 Rancho Seco 1	1d/1c	Refer to Action Plan Item I.C.1

TABLE 1A (Continued)

Requirement	Source for Operating Reactors	Applicability	CP/ML Requirement Category Assignment	Comments
c. RCP operation	NUREG-0623	PWR	ld/lc	Refer to Action Plan Items I.A.1.3 and I.C.1
d. Operators are in cructed not to rely on level indication alone in evaluating plant conditions.	79-05A (Item 4.d) 79-06A (Item 7.d) 79-06B (Item 6.d) 79-08 (Item 5.b)	PWR and BWR	ld/lc	Refer to Action Plan Items I.C.1, I.A.3.1, and II.F.2
5. Safety-related valve position	79-05&05A (Item 5)	PWR and BWR	ld/ld	Refer to Action Plan Items I.C.2 and I.C.6
a. Review all valve positions and positioning requirements and positive controls and all related test and maintenance procedures to assure proper ESF functioning, if required.	79-06A (Item 8) 79-06B (Item 7) 79-008 (Item 6)			
b. Verify that AFW valves are in open position. See Requirement 8 below	79-05A (Item 5)	B&W	ld/ld	Refer to Action Plan Items I.C.2 and I.C.6
6. Review containment isolation initiation design and procedures. Assure isolation of all lines that do not degrade safety features or cooling capability upon automatic initiation of SI.	79-05A (Item 6) 79-06A (Item 4) 79-06B (Item 3) 79-08 (Item 2)	PWR and BWR	ld/ld	Refer to Action Plan Item II.E.4.2
7. Implement positive position controls on valves that could compromise or defeat AFW flow.	79-05A (Item 7)	B&W	ld/lc	Refer to Action Plan Item II.E.1.1

TABLE 1A (Continued)

	Requirement	Source for Operating Reactors	Applicability	CP/ML Requirement Category Assignment	Comments
8.	Immediately implement procedures that assure two independent 100% AFW flow paths, or specify explicitly LCO with reduced AFW capacity.	79-05A (Item 8)	B&W	1d/1c	Refer to Action Plan Item II.E.1.1
9.	Review procedures to assure that radioactive liquids and gases are not transferred out of containment inadvertently (especially upon ESF reset). List all applicable systems and interlocks.	79-05A (Item 9) 79-06A (Item 9) 79-06B (Item 8) 79-08 (Item 7)	PWR and BWR	1d/1c	Refer to Action Plan Item II.E.4.2
10.	Review and modify (as required) procedures for removing safety-related systems from service (and restoring to service) to assure operability status is known.	79-05A (Item 10) 79-06A (Item 10) 79-06B (Item 9) 79-08 (Item 8)	PWR and BWR	1d/1c	Refer to Action Plan Items I.C.2 and I.C.6
11.	Make all operating and maintenance personnel aware of the seriousness and consequences of the erroneous actions taken leading up to, and in early phases of, the TMI-2 accident.	79-05A (Item 11) 79-06A(Item 1.a) 79-06B (Item 1.a)	PWR and BWR	1d/1c	Refer to Action Plan Items I.A.3.a and I.A.2.2
12.	One hour notification requirement, and continuous communications channel.	79-05B (Item 6) 79-06A (Item 11) 79-06B (Item 10) 79-08 (Item 9)	PWR and BWR	1d/1c	Refer to Action Plan Items I.E.6 and III.A.3.3

TABLE 1A (Continued)

	Requirement	Source for Operating Reactors	Applicability	CP/ML Requirement Category Assignment	Comments
13.	Propose Technical Specification changes reflecting implementation of all Bulletin items, as required.	79-05B (Item 7) 79-06A & Rev. 1 (Item 13) 79-06B (Item 12) 79-08 (Item 11)	PWR and BWR	1a/1c	
14.	Review operating modes and procedures to deal with significant amounts of hydrogen.	79-06A (Item 12) 79-06B (Item 11) 79-08 (Item 10)	W, CE GE	1d/1c	Refer to Action Plan Items II.B.4, II.B.7, II.E.4.1 and II.F.1
15.	For facilities with non-automatic AFW initiation, provide dedicated operator in continuous communication with CR to operate AFW.	79-06A (Item 5) 79-06B (Item 4)	W & CE	1d/1c	Refer to Action Plan Item II.E.1.2
16.	Implement (immediately) procedures that identify PRZ PORV "Open" indications and that direct operator to close manually at "RESET" setpoint.	79-06A (Item 6) 79-06B (Item 5)	W & CE	1d/1c	Refer to Action Plan Items I.C.1 and II.D.3
17.	Trip PZR Level Bistable so that PZR Lo Press. (rather than PZR Lo Press. and PZR Lo Level coincidence) will initiate safety injection. For test, reset Lo Level bistable.	79-06A & Rev. 1 (Item 3)	W	1c/1c	
18.	Develop procedures and train operators on methods of establishing and maintaining natural circulation	79-05B (Item 1)	B&W	1d/1c	Refer to Action Plan Items I.C.1 and I.G.1

TABLE 1A (Continued)

	Requirement	Source for Operating Reactors	Applicability	CP/ML Requirement Category Assignment	Comments
19.	Describe design an procedure modifications (based on analysis) to reduce likelihood of automatic PZR PORV actuation in transients.	79-05B (Item 3)	B&W	1d/1c	Refer to Action Plan Item II.E.5
20.	Provide procedures and training to operators for prompt manual reactor trip for LOFW, TT, MSIV closure, LOOP, LOSG Level, & Lo PZR Level.	79-05B (Item 4)	B&W	2/1c	
21.	Provide automatic safety-grade anticipatory reactor trip for LOFW, TT, or significant decrease in SG level.	79-05B (Item 5)	B&W	1d/1c	Refer to Action Plan Item II.K.2.10
22.	Describe automatic and manual actions for proper functioning of auxiliary heat removal systems when FW system not operable.	79-08 (Items 3)	BWR	4/1c	Refer to Appendix B
23.	Describe uses and types of RV level indication for automatic and manual initiation safety systems. Also, describe alternative instrumentation.	79-08 (Item 4)	BWR	1d/1d	
24.	Perform LOCA analyses for a range of small-break sizes and a range of time lapses between reactor trip and RCP trip.	79-05C (short-term Item 2) 79-06C (short-term Item 2)	PWR	1d/1c	Refer to Action Plan Item I.C.1

TABLE 1A (Continued)

Requirement	Source for Operating Reactors	Applicability	CP/ML Requirement Category Assignment	Comments
25. Develop operator action guidelines (based on analyses in Requirement 24 above).	79-05C (short-term Item 3) 79-06C (short-term Item 3)	PWR	1d/1d	Refer to Action Plan Item I.C.1
26. Revise emergency procedures and train ROs and SROs based on guidelines developed in Requirement 25 above.	79-05C (short-term Item 4) 79-06C (short-term Item 4)	PWR	1d/1c	Refer to Action Plan I.C.1, I.A.3.a, and I.G.1
27. Provide analyses and develop guidelines and procedures for inadequate core cooling conditions. Also, define RCP restart criteria.	79-05C (short-term Item 5) 79-06C (short-term Item 5)	PWR	1d/1d	Refer to Action Plan Items I.C.1 and II.F.2
28. Provide design that will assure automatic RCP trip for all circumstances where required.	NUREG-0623	PWR	1d/1d	Refer to Action Plan Item II.K.3.5

TABLE 1B REQUIREMENTS FOR NEW B&W PLANTS DERIVED FROM COMMISSION ORDERS ON OPERATING B&W PLANTS

	Requirement	Source for Operating Reactors	Applicability	CP/ML Requirement Category Assignment	Comments
1.	Upgrade timeliness and reliability of AFW system.	Commission Order	B&W	1d/1c	Refer to Action Plan Item II.E.1
2.	Procedures and training to initiate and control AFW independent of integrated control system.	Commission Order	B&W	1d/1c	
3.	Hard-wired control-grade anticipatory reactor trips.	Commission Order	B&W	1d/1c	Refer to Action Plan Item II.K.2.10
4.	Small-break LOCA analysis, procedures and operator training.	Commission Order	B&W	1d/1c	Refer to Action Plan Items I.A.3.1 and I.C.1
5.	Complete TMI-2 simulator training for all operators.	Commission Order	B&W	1d/1c	Refer to Action Plan Item I.A.2.6
6.	Reevaluate analysis for dual-level setpoint control.	Commission Order	Davis-Besse 1	1c/1c	
7.	Reevaluate transient of September 24, 1977.	Commission Order	Davis-Besse 1	1c/1c	
8.	Continued upgrading of AFW system.	Commission Order	B&W	1d/1c	Refer to Action Plan Item II.E.1

TABLE [illegible] (Continued)

	Requirement	Source for Operating Reactors	Applicability	CP/ML Requirement Category Assignment	Comments
9.	Analysis and upgrading of integrated control system.	Commission Order	B&W	4/1c	Refer to Appendix B
10.	Hard-wired safety-grade anticipatory reactor trips.	Commission Order	B&W	4/1c	Refer to Appendix B
11.	Operator training and drilling.	Commission Order	B&W	1d/1c	Refer to Action Plan Items I.A.3.1, I.A.2.2, I.A.2.5, and I.G.a
12.	Transient analysis and procedures for management of small breaks.	Commission Order	B&W	1d/1c	Refer to Action Plan Item I.C.1
13.	Thermal-mechanical report -- effect of HPI on vessel integrity for small-break LOCA with no AFW.	Letter, D. Ross to B&W operating plants, 8/121/79	B&W	1b/1c	
14.	Demonstrate that predicted lift frequency of PORVs and SVs is acceptable.	Letter, D. Ross to B&W operating plants, 8/21/79	B&W	1d/1d	
15.	Analysis of effects of slug flow on once-through steam generator tubes after primary system voiding.	Letter, D. Ross to B&W operating-plants, 8/21/79	B&W	1e/1e	

TABLE 1B (Continued)

	Requirement	Source for Operating Reactors	Applicability	CP/ML Requirement Category Assignment	Comments
16.	Impact of RCP seal damage following small-break LOCA with loss of offsite power.	Letter, D. Ross to B&W operating Plants, 8/21/79	All	3/3	Refer to Appendix B
17.	Analysis of potential voiding in RCS during anticipated transients.	Letter, R. Reid to all B&W operating plants 1/9/80	All B&W	1d/1c	Refer to Action Plan Item I.C.1
18.	Analysis of loss of feedwater and other anticipated transients.	Letter, D. Ross to B&W operating plants, 8/21/79	All B&W	1d/1c	Refer to Action Plan Item I.C.1
19.	Benchmark analysis of sequential AFW flow to once-through steam generator	Letter, D. Ross to B&W operating plants, 8/21/79	All B&W	1d/1c	Refer to Action Plan Item I.C.1
20.	Analysis of system response to small-break LOCA that causes system pressure to exceed PORV setpoint.	Letter, D. Ross to B&W operating plants 8/21/79	All B&W	1d/1c	Refer to Action Plan Item I.C.1
21.	LOFT 3-1 predictions.	Letter, D. Ross to B&W operating plants, 8/21/79	All B&W	1e/1c	

TABLE 1C FINAL RECOMMENDATIONS OF BULLETINS AND ORDERS TASK FORCE

Requirement	Source	Applicability	CP/ML Requirement Category Assignment	Comments
1. Install automatic PORV isolation system and perform operational test.	NUREG-0565 (2.1.2.1) NUREG-0611 (3.2.4.e) 3.2.4.f) NUREG-0635 (3.2.4.a) (3.2.4.b)	PWR	1d/1c	Refer to Action Plan Item II.K.3.2
2. Report on overall safety effect of PORV isolation system.	NUREG-0565 (2.1.2.d) NUREG-0611 (3.2.4.g) (3.2.4.i) NUREG-0635 (3.2.4.c)	PWRs	3/3	Refer to Appendix B
3. Report safety and relief valve failures promptly and challenges annually.	NUREG-0565 (2.1.2.c, 2.1.2.e) NUREG-0611 (3.2.4.h) NUREG-0626 (B.14) NUREG-0635 (3.2.4.d)	All	2/2	
4. Review and upgrade reliability and redundancy of nonsafety equipment for small-break LOCA mitigation	NUREG-0565 (2.3.2.b) NUREG-0611 (3.2.2.b) NUREG-0626 (B.12) NUREG-0635 (3.2.2.b)	All	1b/1b	Refer to Action Plan Items II.C.1, II.C.2, and II.C.3
5. Continue to study need for C.1.4.c and need for automatic trip of RCPs, then modify procedures or designs as appropriate.	NUREG-0565 (2.3.2.a) NUREG-0611 (3.2.2.a) NUREG-0635 (3.2.2.a) NUREG-0623	PWR	1b/1b	

TABLE 1C (Continued)

	Requirement	Source	Applicability	CP/ML Requirement Category Assignment	Comments
6.	Instrumentation to verify natural circulation.	NUREG-0565 (2.6.2.b) NUREG-0611 (3.2.3.b) NUREG-0635 (3.2.3.b)	PWR	ld/ld	Refer to Action Plan Item I.C.1, II.F.2, II.F.3
7.	Evaluation of PORV opening probability during overpressure transient.	NUREG-0565 (2.1.2.b)	B&W	ld/lc	Refer to Action Plan Item II.K.2.14
8.	Further staff consideration of need for diverse decay heat removal method independent of SGs	NUREG-0565 (2.5.2.a) NUREG-0635 (4.2.5), App. VIII) NUREG-0611 (4.2.5, App. VIII)	PWR	ld/ld	Refer to Action Plan Item II.C.1 and II.E.3.3
9.	Proportional integral derivative controller modification.	NUREG-0611 (3.2.4.b)	W	lc/2	
10.	Anticipatory trip modification proposed by some licensees to confine range of use to high power levels.	NUREG-0611 (3.2.4.c)	W	lc/2	
11.	Control use of PORV supplied by Control Components Inc., until further review complete.	NUREG-0611 (3.2.4.d)	All	ld/ld	Deleted - this is covered by Action Plan item II.D.1
12.	Confirm existence of anticipatory trip upon turbine trip.	NUREG-0611 (3.2.4.a)	W	lc/2	

TABLE 1C (Continued)

	Requirement	Source	Applicability	CP/ML Requirement Category Assignment	Comments
13.	Separation of HPCI and RCIC system initiation levels. Analysis and implementation.	NUREG-0626 (A.1)	GE	3/1c	Refer to Appendix B
14.	Isolation of isolation condensers on high radiation.	NUREG-0626 (A.2)	GE plants with isolation condenser	1c/1c	
15.	Modify break detection logic to prevent spurious isolation of HPCI and RCIC systems.	NUREG-0626 (A.3)	GE	2/1c	
16.	Reduction of challenges and failures of relief valves - feasibility study and system modification.	NUREG-0626 (A.4)	GE	3/1c	Refer to Appendix B
17.	Report on outage of ECC systems - licensee report and proposed technical specification changes.	NUREG-0626 (A.6)	GE	1a/1c	
18.	Modification of ADS logic - feasibility study and modification for increased diversity for some event sequences.	NUREG-0626 (A.7)	GE	3/1c	Refer to Appendix B

TABLE 1C (Continued)

Requirement	Source	Applicability	CP/ML Requirement Category Assignment	Comments
19. Interlock on recirculation pump loops.	NUREG-0626 (A.8)	GE Non-Jet Pump OPs	1c/1c	
20. Loss of service water for Big Rock Point.	NUREG-0626 (A.9)	Big Rock Point	1c/1c	
21. Restart of core spray and LPCI systems on low level - design and modification.	NUREG-0626 (A.10)	GE	3/1c	Refer to Appendix B
22. Automatic switchover of RCIC system suction - verify procedures and modify design.	NUREG-0626 (B.1)	GE	1c/1c	
23. Central water level recording.	NUREG-0626 (B.2)	GE	4/1c	Refer to Appendix B
24. Confirm adequacy of space cooling for HPCI and RCIC systems.	NUREG-0626 (B.3)	GE	3/1c	Refer to Appendix B
25. Effect of loss of AC power on pump seals.	NUREG-0626 (B.4)	GE	1d/1d	Refer to Item II.K.2.16

TABLE 1C (Continued)

	Requirement	Source	Applicability	CP/ML Requirement Category Assignment	Comments
26.	Study effect on RHR reliability of its use for fuel pool cooling.	NUREG-0626 (B.5)	GE	1d/1c	Refer to Action Plan Item II.E.2.1
27.	Provide common reference level for vessel level instrumentation.	NUREG-0626 (B.6)	GE	2/1c	
28.	Study and verify qualification of accumulators on ADS valves.	NUREG-0626 (B.7)	GE	3/1c	Refer to Appendix B
29.	Study to demonstrate performance of isolation condensers with noncondensibles.	NUREG-0626 (B.13)	GE Isolation Condenser ORs	1c/1c	
30.	Revised small-break LOCA methods to show compliance with 10 CFR 50, Appendix K.	NUREG-0565 (2.2.2.a) NUREG-0611 (3.2.1.a) NUREG-0626 (A.12) NUREG-0635 (3.2.1.a) (3.2.5.a)	All	1b/1c	
31.	Plant-specific calculations to show compliance with 10 CFR 50.46.	NUREG-0565 (2.2.2.b) NUREG-0611 (3.2.1.b) NUREG-0626 (A.13, B.10) NUREG-0635 (3.2.a.b)	All	1b/1c	

TABLE 1C (Continued)

	Requirement	Source	Applicability	CP/ML Requirement Category Assignment	Comments
32.	Provide experimental verification of two-phase natural circulation models.	NUREG-0565 (2.6.2.a) NUREG-0611 (3.2.3.a) NUREG-0635 (3.2.3.a)	PWR	1b/1b	Refer to Action Plan Item II.E.2.2
33.	Evaluate elimination of PORV function.	NUREG-0565 (3.5) NUREG-0611 (3.2.4.k) NUREG-0635 (3.2.4.e)	PWR	1b/1b	Refer to Action Plan Item II.C.1
34.	RELAP-4 model development.	NUREG-0611 (3.2.5) NUREG-0635 (3.2.5)	PWR	1b/1b	Refer to Action Plan Item II.E.2.2
35.	Evaluation of effects of core flood tank injection on small-break LOCAs.	NUREG-0565 (2.2.2.c)	B&W	1d/1c	Refer to Action Plan Item I.C.1
36.	Additional staff audit calculations of B&W small-break LOCA analyses.	NUREG-0565 (2.4.2.a)	B&W	1b/1c	Refer to Action Plan Item I.C.1
37.	Analysis of B&W plant response to isolated small-break LOCA.	NUREG-0565 (2.6.2.c)	B&W	1d/1c	Refer to Action Plan I.C.1
38.	Analysis of plant response to a small-break LOCA in the pressurizer spray line.	NUREG-0565 (2.6.2.d)	B&W	1d/1c	Refer to Action Plan Item I.C.1

TABLE 1C (Continued)

	Requirement	Source	Applicability	TMI Requirement Category Assignment	Comments
39.	Evaluation of effects of water slugs in piping caused by HPI and CFT flows.	NUREG-0565 (2.6.2.e)	B&W	1d/1c	Refer to Action Plan Item I.C.1
40.	Evaluation of RCP seal damage and leakage during a small-break LOCA.	NUREG-0565 (2.6.2.f)	B&W	1d/1c	Refer to Action Plan Item II.K.2.16
41.	Submit predictions for LOFT Test L3-6 with RCPs running.	NUREG-0565 (2.6.2.g)	B&W	1d/1c	Refer to Action Plan Item I.C.1
42.	Submit requested information on the effects of non-condensible gases.	NUREG-0565 (2.6.2.h)	B&W	1d/1c	Refer to Action Plan Item I.C.1
43.	Evaluation of mechanical effects of slug flow on steam generator tubes.	NUREG-0565 (2.6.2.i)	B&W	1d/1c	Refer to Action Plan II.K.2.15
44.	Evaluation of anticipated transients with single failure to verify no significant fuel failure.	NUREG-0626 (A.14)	GE	1e/1c	Deleted - generic study was submitted
45.	Evaluate depressurization with other than full ADS.	NUREG-0626 (A.15)	GE	3/1c	Refer to Appendix B

TABLE 1C (Continued)

	Requirement	Source	Applicability	CP/ML Requirement Category Assignment	Comments
46.	Response to list of concerns from ACRS consultant.	NUREG-0626 (A.17)	GE	1d/1c	
47.	Test program for small-break LOCA model verification pretest prediction, test program and model verification.	NUREG-0626 (B.9)	GE	1d/1c	Refer to Action Plan Items I.C.1, and II.E.2.2
48.	Assess change in safety reliability as result of implementation B&OTF recommendations.	NUREG-0626 (B.15)	GE	1d/1c	Refer to Action Plan Items II.C.1 and II.C.2
49.	Review of Procedures (NRC).	NUREG-0611 (3.4.1) NUREG-0635 (3.4.1)	W, CE	1b/1b	Refer to Action Plan I.C.8 and I.C.9
50.	Review of Procedures (NSSS vendors)	NUREG-0611 (3.4.2) NUREG-0635 (3.4.2)	W, CE	1d/1c	Refer to Action Plan I.C.7 and I.C.9
51.	Sympton-based emergency procedures.	NURGE-0611 (3.4.3) NUREG-0626 (B.8) NUREG-0635 (3.4.3)	W, CE GE	1d/1c	Refer to Action Plan Item I.C.9
52.	Operator awareness of revised emergency procedures.	NUREG-0626 (A.11)	GE	1d/1c	Refer to Action Plan Items I.B.1, I.C.2, and I.C.5

TABLE 1C (Continued)

Requirement	Source	Applicability	CP/ML Requirement Category Assignment	Comments
53. Two operators in control room.	NUREG-0626 (A.16)	GE	1d/1c	Refer to Action Plan Item I.A.1.3
54. Simulator upgrade for small-break LOCAs.	NUREG-0565 (2.3.2.c) NUREG-0611 (3.3.1.b) NUREG-0626 (B.11) NUREG-0635 (3.3.1.b)	All	1d/1c	Refer to Action Plan Item I.A.4.1
55. Operator monitoring of control board.	NUREG-0611 (3.5.1) NUREG-0635 (3.5.1)	W, CE	1d/1c	Refer to Action Plan Items I.C.1, I.D.2 and I.D.3
56. Simulator training requirements.	NUREG-0611 (3.3.1.a) NUREG-0635 (3.3.1.a)	W, CE	1d/1c	Refer to Action Plan Items I.A.3.1, I.A.3.3, and I.A.2.6
57. Identify water sources prior to manual activation of ADS.	NUREG-0626 (A.5)	GE	1d/1c	Refer to Action Plan I.C.1

APPENDIX B
INFORMATION REQUIREMENTS
FOR
TMI-2 ACTION PLAN ITEMS
IN
CATEGORIES 3, 4, AND 5

I.A.4.2 LONG-TERM TRAINING SIMULATOR UPGRADE

Applicants shall describe their program for providing simulator capability for their plants. In addition, they shall describe how they will assure that their proposed simulator will correctly model their control room. Applicants shall provide sufficient information to permit the NRC staff to verify that they will have the necessary simulator capability to carry out the actions described in this Action Plan item as well as Action Plan Item II.K.3.54. Applicants shall submit, prior to the issuance of construction permits, a general discussion of how the requirements will be met. Sufficient details shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

I.C.5 PROCEDURES FOR FEEDBACK OF OPERATING, DESIGN AND CONSTRUCTION EXPERIENCE

Applicants shall submit a description of their administrative procedures for evaluating operating, design, and construction experience and describe how they will assure that applicable important industry experiences originating from both within and outside the applicant's construction organization will be provided in a timely manner to those designing and constructing the plant. Applicants shall submit a general discussion of how the requirments will be met. These procedures shall: (1) Clearly identify organization responsibilities for review and identification of these important experiences and the feedback of pertinent information to those responsible for designing and constructing the plant; (2) Identify the administrative and technical review steps necessary in implementing applicable important experiences; (3) Identify the recipients of various categories of information from these experiences or otherwise provide means through which such information can be readily related to the job functions of the recipients; (4) Assure that applicant and contractor personnel do not routinely receive extraneous and unimportant experience-related information in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency; (5) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to applicant and contractor personnel for implementation until resolution is reached; and (6) Provide practical interim audits to assure that the feedback program functions effectively at all levels. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of construction permits or manufacturing license.

I.C.9 LONG-TERM PROGRAM PLAN FOR UPGRADING OF PROCEDURES

Applicants shall describe their program plan, which is to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analysis, human factors engineering, crisis management and operator training. Applicants shall also insure that their program will be coordinated, to the extent possible, with INPO and other industry group efforts. Applicants will submit, prior to the issuance of construction permits, a general discussion of how the requirements will be met. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

I.D.1 CONTROL ROOM DESIGN REVIEWS

Applicants shall provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Applicants shall provide a general discussion of their approach to control room designs that reflect human factor principles by specifying the design concept selected and the supporting design bases and criteria. Cosmetic revisions to conventional (1960 technology) designs are unacceptable. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses. Applicants shall commit to control room designs reflecting human factors principles prior to issuance of a CP or ML and shall supply design information for review prior to committing to fabrication or revision of fabricated control room panels and layouts.

I.D.2 PLANT SAFETY PARAMETER DISPLAY CONSOLE

Applicants shall describe how they intend to meet the staff criteria contained in NUREG-0696 for a plant safety parameter display console. The console shall display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

I.D.3 SAFETY SYSTEM STATUS MONITORING

Applicants shall describe how their design conforms to Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

I.F.1 EXPAND QA LIST

Prior to issuance of the construction permits or manufacturing license, applicants shall revise their QA programs by expanding their QA lists to include all items and activities affecting safety as defined by Regulatory Guide 1.29 and Appendix A to 10 CFR Part 50, and shall provide a commitment to apply the revised QA program to all such items and activities.

I.F.2 DEVELOP MORE DETAILED QA CRITERIA

Applicants shall describe the changes to their QA programs that have resulted from their review of the accident at TMI-2. In addition, applicants shall address the appropriate matters discussed in this Action Plan item, including the establishment of a quality assurance (QA) program based on consideration of: (a) ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (b) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (c) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (d) establishing criteria for determining QA programmatic requirements; (e) establishing qualification requirements for QA and QC personnel; (f) sizing the QA staff commensurate with its duties and responsibilities; (g) establishing procedures for maintenance of "as-built" documentation; and (h) providing a QA role in design and analysis activities. Applicants shall submit, prior to the issuance of the construction permits or manufacturing license, a revised description of their QA program that includes consideration of these matters.

II.B.1 REACTOR COOLANT SYSTEM VENTS

Applicants shall modify their plant designs as necessary to provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting these requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.B.2 PLANT SHIELDING TO PROVIDE ACCESS TO VITAL AREAS AND PROTECT SAFETY EQUIPMENT FOR POST-ACCIDENT OPERATION

Applicants shall (1) perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID 14844* source

*TID 14844, U.S. Atomic Energy Commission, 1962.

term radioactive material and (2) implement plant design modifications necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.B.3 POST-ACCIDENT SAMPLING

Applicants shall (1) review the reactor coolant and containment atmosphere sampling system designs and the radiological spectrum and chemical analysis facility designs, and (2) modify their plant designs as necessary to provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID 14844* source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole-body or 75 rem to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.B.8 RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

Applicants shall:

(1) commit to performing a site/plant-specific probabilistic risk assessment and incorporating the results of the assessment into the design of the facility. The commitment must include a program plan, acceptable to the staff, that demonstrates how the risk assessment program will be scheduled so as to influence system designs as they are being developed. The assessment shall be completed and submitted to NRC within two years of issuance of the construction permit. The outcome of this study and the NRC review of it will be a determination of specific preventive and mitigative actions to be implemented to reduce these risks. A prevention feature that must be considered is an additional decay heat removal system whose functional requirements and criteria would be derived from the PRA study.

It is the aim of the Commission through these assessments to seek such improvements in the reliability of core and containment heat removal

*TID 14844, U.S. Atomic Energy Commission, 1962.

systems as are significant and practical and do not impact excessively on the plant. Applicants are encouraged to take steps that are in harmony with this aim.

(2) include provisions in the containment design for one or more dedicated penetrations, equivalent in size to a single 3-foot diameter opening. This shall be done in order not to preclude the installation of systems to prevent containment failure, such as filtered vented containment systems.

(3) provide a system for hydrogen control capable of handling hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction.

(4) provide preliminary design information at a level consistent with that normally required at the construction permit stage of review sufficient to demonstrate that:

(a) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Division 1, Subsubarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Division 2, Subsubarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent, depending upon which option is chosen for control of hydrogen. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

(b) The containment and associated systems will provide reasonable assurance that uniformly-distributed hydrogen concentrations do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

(c) The facility design will provide reasonable assurance that, based on a 100% fuel clad metal-water reaction, combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(d) If the option chosen for hydrogen control is post-accident inerting: (a) Containment structure loadings produced by an inadvertent full inerting (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Division 1, Subsubarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Division 2, Subsubarticle CC-3720, Service Load Category), (b) demonstrate that a pressure test, which is required, of the containments at 1.10 and 1.15 times for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting can be safely conducted, (c) demonstrate that inadvertent full inerting of the containment can be safely accommodated during plant operation.

(e) If the option chosen for hydrogen control is a distributed ignition system, equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity shall be designed to perform its function during and after being exposed to the environmental conditions created by activation of the distributed ignition system.

II.D.1 TESTING REQUIREMENTS

Applicants and their agents shall provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWR's, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transient without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed and not before issuance of an ATWS rule. Applicants shall submit, prior to the issuance of the construction permits or manufacturing license, a general explanation of how the testing requirements will be met. Sufficient detail should be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Applicants shall (1) demonstrate the applicability of the generic tests conducted under II.D.1 to their particular plants and (2) modify their plant designs as necessary. Applicants shall commit, prior to the issuance of the construction permits or manufacturing license, to comply with these requirements and shall submit within six months following the completion of the generic tests or the issuance of construction permits, whichever is later, a detailed explanation of how the test results will be incorporated in the plant design. Sufficient detail should be presented to provide reasonable assurance that the requirements resulting from the test will be implemented properly prior to the issuance of operating licenses.

II.D.3 RELIEF AND SAFETY VALVE POSITION INDICATION

Applicants shall modify their plant designs as necessary to provide direct indication of relief and safety valve position in the control room. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to issuance of operating licenses.

II.E.1.1 AUXILIARY FEEDWATER SYSTEM EVALUATION

Applicants shall perform a reevaluation of their proposed auxiliary feedwater (AFW) system. This reevaluation shall include the following:

(1) Performance of simplified auxiliary feedwater system reliability analyses using event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss of main feedwater transient conditions, with particular emphasis being given to determining potential failures that could result from human errors, common causes, single point vulnerabilities, and test and maintenance outages. The results of this evaluation shall be compared with the results of the NRC staff's generic AFW system evaluation published in Appendix III to NUREG-0611 and Appendix III to NUREG-0635. Applicants with plants with AFW systems with relatively low reliabilities shall submit proposed design changes and/or procedural actions which will improve the relative reliability of the AFW system to above average. Applicants whose plant designs do not include high head high pressure injection system pumps for use in the feed and bleed mode of decay heat removal in case of AFW system failure shall assure that their AFW system has a very high reliability relative to those AFW systems evaluated by the NRC and staff and reported in NUREG-0611 and NUREG-0635 respectively.

(2) Completion of a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 as principal guidance. This requirement applies to those plants where the Standard Review Plan was not used as criteria during the NRC staff's CP review.

(3) Reevaluation of the AFW system flow design bases and criteria. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

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

Applicants with PWR plants shall provide automatic and manual auxiliary feedwater (AFW) system initiation and auxiliary feedwater system flow indication in the control room. These systems shall be safety grade and meet the requirements specified in NUREG-0737. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.E.3.1 RELIABILITY OF POWER SUPPLIES FOR NATURAL CIRCULATION

Applicants shall (1) upgrade the power supplies for the pressurizer heaters and associated motive and control power interfaces to meet the applicable requirements specified in NUREG-0737 and (2) establish procedures and training for maintaining the reactor coolant system at hot standby conditions with only onsite power available.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent witn that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.E.4.1 DEDICATED PENETRATION

Applicants for plant designs with external hydrogen recombiners shall modify their applications as necessary to include redundant dedicated containment pentrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere. Applicants shall submit, prior to the issuance of construction permits or the manufacturing license, a detailed explanation of how the requirements will be met in order to provide reasonable assurance that the requirements will be implemented properly.

II.E.4.2 ISOLATION DEPENDABILITY

Containment isolation system designs shall comply with the recommendations of Standard Review Plan Section 6.2.4.

All plants shall give careful consideration to the definition of essential and non-essential systems, identify each system determined to be essential, identify each system determined to be non-essential, and describe the basis for selection of each essential system. All non-essential systems shall be automatically isolated by the containment isolation signal. Revision 2 to Regulatory Guide 1.141 will contain guidance on the classification of essential versus non-essential systems and is due to be issued by June 1981.

For post-accident situations, each non-essential penetration (except instrument lines) is required to have two isolation barriers in series that meet the requirements of General Design Criteria 54, 55, 56, and 57, as clarified by Standard Review Plan, Section 6.2.4. Isolation must be performed automatically (i.e., no credit can be given for operator action). Manual valves must be sealed closed, as defined by Standard Review Plan, Section 6.2.4, to qualify as an isolation barrier. Each automatic isolation valve in a non-essential penetration must receive diverse isolation signals.

The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action. Administrative provisions to close all isolation valves manually before resetting the isolation signals is not an acceptable method of meeting this requirement.

Ganged reopening of containment isolation valves is not acceptable. Reopening of isolation valves must be performed on a valve-by-valve basis, or on a line-by-line basis, provided that electrical independence and other single-failure criteria continue to be satisfied.

The containment setpoint pressure that initiates containment isolation for non-essential penetrations must be reduced to the minimum compatible with normal operating conditions. The containment pressure history during normal operation for similar operating plants should be used as a basis for arriving at an appropriate minimum pressure setpoint for initiating containment isolation. The pressure setpoint selected should be far enough above the maximum observed (or expected) pressure inside containment during normal operation so that inadvertent containment isolation does not occur during normal operation from instrument drift or fluctuations due to the accuracy of the pressure sensor. A margin of 1 psi above the maximum expected containment pressure should be adequate to account for instrument error. Any proposed values greater than 1 psi will require detailed justification.

All systems that provide a path from the containment to the environs (e.g., containment purge and vent systems) must close on a safety-grade high radiation signal.

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

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.E.4.4 PURGING

Applicants shall (1) provide a capability for containment purging/venting designed to minimize purging time, consistent with ALARA principles for occupational exposure, (2) evaluate the performance of purging and venting isolation valves against accident pressure, (3) address the interim NRC guidance on valve operability, (4) adopt procedures and restrictions consistent with the revised requirements; and (5) provide and demonstrate high assurance that the purge system will be reliably isolated under accident conditions.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.E.5.1 DESIGN EVALUATION

Applicants with B&W-designed reactors shall (1) identify the most severe overcooling events (considering both anticipated transients and accidents) that could occur at the facilities, (2) show, in view of the arrival rate for these events, that the design criterion for the number of actuation cycles of the emergency core cooling system and reactor protection system is adequate, (3) recommend changes to systems or procedures that would reduce primary system sensitivity. Applicants with B&W-designed reactors shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.F.1 ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION

Applicants shall provide instrumentation to measure, record and readout in the control room: (a) containment pressure, (b) containment water level, (c) containment hydrogen concentration, (d) containment radiation intensity (high level), and (e) noble gas effluents at all potential, accident release points. The requirements for the specific monitors are listed in NUREG-0737. Applicants shall also provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential, accident release points, and for onsite capability to analyze and measure these samples. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.F.2 IDENTIFICATION OF AND RECOVERY FROM CONDITIONS LEADING TO INADEQUATE CORE COOLING

Applicants shall describe their program for developing and implementing procedures to be used by the reactor operators to detect and recover from conditions leading to inadequate core cooling.

Applicants shall provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWR's, and a suitable combination of signals from indicators of coolant level in the reactor vessel and incore thermocouples in PWR's and BWR's.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.F.3 INSTRUMENTATION FOR MONITORING ACCIDENT CONDITIONS (REG. GUIDE 1.97)

Applicants shall provide in their facility design instrumentation to monitor plant variables and systems during and following an accident in accordance with defined design bases and Regulatory Guide 1.97, Rev. 2, December 1980. Designs are already established for much of the instrumentation that will be required; some of the requirements, however, may involve state-of-the-art designs or designs which have yet to be developed.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.G.1 POWER SUPPLIES FOR PRESSURIZER RELIEF VALVES, BLOCK VALVES, AND LEVEL INDICATION

Applicants with PWR plants shall provide power supplies for the pressurizer relief valves, block valves, and pressurizer level indicators to meet the applicable requirements specified in NUREG-0737. Level indicators shall be powered from vital buses, motive and control power connections to emergency power sources shall be through devices qualified in accordance with requirements applicable to systems important to safety, and electric power shall be provided from emergency sources. Applicants with PWR plants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the support design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.J.3.1 ORGANIZATION AND STAFFING TO OVERSEE DESIGN AND CONSTRUCTION

Applicants shall describe their program for the management oversight of design and construction activities. Specific items to be addressed include: (1) the organizational and management structure which is singularly responsible for the direction of the design and construction of the proposed plant, (2) technical resources which are directed by the utility organization, (3) details of the interaction of design and construction within the utility organization and the manner by which the utility will assure close integration of the architect engineer and nuclear steam supply vendor, (4) proposed procedures for handling the transition to operation, and (5) the degree of top level management oversight and technical control to be exercised by the utility during design and construction, including the preparation and implementation of procedures necessary to guide the effort.

Draft NUREG-0731, "Guidelines for Utility Management Structure and Technical Resources" is the keystone for similar development of guidelines for this task. Therefore, the principal applicable elements of NUREG-0731 shall be used by CP and ML applicants in addressing this task.

Applicants shall submit detailed information in order to provide reasonable assurance that the requirements will be implemented properly prior to issuance of the construction permits or manufacturing license.

II.K.1.22 DESCRIBE AUTOMATIC AND MANUAL ACTIONS FOR PROPER FUNCTIONING OF AUXILIARY HEAT REMOVAL SYSTEMS WHEN FW SYSTEM NOT OPERABLE

Applicants with BWR plants shall design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. A general explanation of how this requirement will be met is required prior to issuance of the construction permits. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly.

II.K.2.9 ANALYSIS AND UPGRADING OF INTEGRATED CONTROL SYSTEM

Applicants with B&W-designed plants shall address the requirements set forth in the Commission Orders issued to operating B&W plants in May 1979 regarding the analysis and upgrading of the integrated control system (ICS), and perform a failure modes and effects analysis of the ICS to include considerations of failures and effects of input and output signals to the ICS. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.K.2.10 HARD-WIRED SAFETY-GRADE ANTICIPATORY REACTOR TRIPS

Applicants with B&W-designed plants shall provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.K.2.16 IMPACT OF RCP SEAL DAMAGE FOLLOWING SMALL-BREAK LOCA WITH LOSS OF OFFSITE POWER

Applicants shall perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss-of-coolant accident with subsequent reactor coolant pump seal damage. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.K.3.2 REPORT ON OVERALL SAFETY EFFECT OF PORV ISOLATION SYSTEM

Applicants with PWR plants shall address the requirements set forth in Item 3.2.4.e and 3.2.4.f of NUREG-0611 and perform an analysis of the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV). If this probability is a significant contributor to the probability of small-break LOCA's from all causes, provide a description and evaluation of the effect on small break LOCA probability of an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened. Applicants with PWR plants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.K.3.13 SEPARATION OF HPCI* AND RCIC SYSTEM INITIATION LEVELS - ANALYSIS AND IMPLEMENTATION

Applicants with BWR plants shall address the requirements set forth in Item A.1 of NUREG-0626 as they apply to HPCI and RCIC systems, and perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

*For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, "high pressure core spray" for "high pressure coolant injection" and "HPCS" for "HPCI".

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

Applicants with BWR plants shall address the requirements set forth in Item A.4 of NUREG-0626, and perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.K.3.18 MODIFICATION OF ADS LOGIC - FEASIBILITY STUDY AND MODIFICATION FOR INCREASED DIVERSITY FOR SOME EVENT SEQUENCES

Applicants with BWR plants shall address the requirements set forth in Item A.7 of NUREG-0626 and perform a feasibility and risk assessment study to determine the optimum automatic depressurization system (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.K.3.21 RESTART OF CORE SPRAY AND LPCI SYSTEMS ON LOW LEVEL - DESIGN AND MODIFICATION

Applicants with BWR plants shall address the requirements set forth in Item A.10 of NUREG-0626 and perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.K.3.23 CENTRAL WATER LEVEL RECORDING

Applicants with BWR plants shall provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. Applicants shall implement design modifications as necessary to meet the requirements. Applicants shall submit, prior to issuance of construction permits, a general explanation of how the requirements will be met. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.K.3.24 CONFIRM ADEQUACY OF SPACE COOLING FOR HPCI* AND RCIC SYSTEMS

Applicants with BWR plants shall address the HPCI and RCIC systems requirements set forth in Item B.3 of NUREG-0626, and perform a study to determine the need

*For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, "high pressure core spray" for "high pressure coolant injection" and "HPCS" for "HPCI".

for additional space cooling to ensure reliable long-term operation of these systems following a complete loss of offsite power to the plant for at least two hours. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.K.3.28 VERIFY QUALIFICATION OF ACCUMULATORS ON ADS VALVES

Applicants with BWR plants shall provide information to ensure that the ADS valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation while taking no credit for non-safety related equipment or instrumentation. Air (or nitrogen) leakage through valves must be accounted for to ensure that enough inventory of compressed air (or nitrogen) will be available to cycle the ADS valves. Applicants shall commit that these requirements will be met in the final design at the OL stage.

In addressing this item prior to CP issuance, applicants should note that safety analysis reports claim that air (or nitrogen) accumulators for the ADS valves provide sufficient capacity (inventory) to cycle these valves open five times at design pressures. Also, General Electric has stated that the emergency core cooling systems are designed to withstand a hostile environment and still perform their functions for 100 days following an accident.

II.K.3.45 EVALUATE DEPRESSURIZATION WITH OTHER THAN FULL ADS

Applicants with BWR plants shall address the requirements set forth in Item A.15 of NUREG-0626, and provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization sytem, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

III.A.1.2 UPGRADE LICENSEE EMERGENCY SUPPORT FACILITIES

Applicants shall address the requirements for a Technical Support Center, Operational Support Center and the Emergency Operations Facility. Applicants shall provide preliminary design information in accordance with the functional criteria in NUREG-0696 at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

III.D.1.1 PRIMARY COOLANT SOURCES OUTSIDE THE CONTAINMENT STRUCTURE

Applicants shall review the designs of such systems outside containment, and their provisions for leakage control and detection, overpressurization design, discharge points for waste gas venting systems, etc., with the goal of minimizing potential exposures to workers and public following an accident, and providing reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. Applicants shall provide for leakage control and detection in the design of systems outside containment that contain (or might containing) TID 14844* source term radioactive materials following an accident, and submit a leakage control program, including an initial test program and a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems.

In this regard, applicants shall submit, prior to the issuance of construction permits, a general discussion of their approach to minimizing leakage from such systems outside containment, in sufficient detail to provide reasonable assurance that this objective will be met satisfactorily prior to issuance of operating licenses.

III.D.3.3 IN-PLANT RADIATION MONITORING

Applicants shall review their designs to ensure that provisions for monitoring inplant radiation and airborne radioactivity are appropriate for a broad range of routine and accident conditions. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

III.D.3.4 CONTROL ROOM HABITABILITY

Applicants shall review the design of their facilities for conformance to requirements stated in the Action Plan. Applicants shall evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID 14844* source term release and make necessary design provisions to preclude such problems.

Applicants shall address prior to the issuance of the construction permits or manufacturing license, how they will implement the existing requirements set forth in this Action Plan item. Applicants shall also address the extent to which improvements have been made to prevent control room contamination via pathways not previously considered. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

*TID 14844, U.S. Atomic Energy Commission, 1962.

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16. ABSTRACT (200 words or less)

The TMI-2 Action Plan, NUREG-0660, does not specifically address requirements for construction permit and manufacturing license applications. There are currently pending six construction permit applications for eleven units with light water reactors and one manufacturing license application for eight floating nuclear plants. Staff review of these applications had been suspended since the TMI-2 accident pending the formulation of a policy to appropriately reflect the lessons learned from the accident.

The Commission is considering a new rule which will state the TMI-related requirements to be applied to these applications. NUREG-0718 was issued, and has now been revised, to provide guidance that the NRC staff believes should be followed to account for the lessons learned from the TMI-2 accident. NUREG-0718 is not a substitute for the regulations, and compliance is not a requirement. However, an approach or method different from the guidance contained herein will be accepted only if the substitute approach or method provides an equivalent basis for meeting the requirements.

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