NUREG-0737

Clarification of TMI Action Plan Requirements

Manuscript Completed: November 1980
Date Published: November 1980

**Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555**

ABSTRACT

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

CONTENTS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OCT 31 1980

TO ALL LICENSEES OF OPERATING PLANTS AND APPLICANTS FOR OPERATING LICENSES AND HOLDERS OF CONSTRUCTION PERMITS

Gentlemen:

SUBJECT: POST-TMI REQUIREMENTS

On September 5, 1980, the NRC staff sent you a draft clarification letter regarding approved TMI Action Plan items. During the week of September 22, 1980, four regional meetings were held to provide a more detailed explanation of these requirements and to obtain industry comments concerning these items. Based on these discussions and other comments received, the NRC has revised its requirements regarding these items. It is the purpose of this letter to set forth those requirements.

This letter incorporates in one document, all TMI-related items approved for implementation by the Commission at this time. This document is being published as NUREG-0737. Enclosures 1 and 2 contain an itemized listing of OR and OL requirements including implementation schedules, applicability, method of implementation review and licensee submittal dates. Enclosure 3 contains more detailed clarifications of most of the NRC positions including the identification of any changes from previous requirements and guidance.

Most of the items in the attached document have already been issued as requirements by previous correspondence. Those items that are being issued as requirements for the first time by this letter are identified by an asterisk in Enclosures 1 and 2. Additional guidance on the Emergency Response Facilities, Section III.A.1.2, will be forwarded separately in the near future.

Licensees and applicants should note that the set of requirements identified in the enclosures do not constitute the total set of TMI-related actions in the TMI-2 Action Plan, NUREG-0660. Rather, as noted above, the enclosures are a compilation of those items that have been specifically approved by the Commission for implementation. Upon further staff development of criteria and planning, additional items will be issued. For example, in the relatively near future, the staff expects to issue further criteria on emergency operational facilities (NUREG-0696), auxiliary feedwater system improvements (derived from NUREG-0667), and instrumentation (Regulatory Guide 1.97, Revision 2). In general, the implementation of those requirements will be carefully examined to ensure that they do not unnecessarily impact any of the requirements in this letter.

The requirements herein (which include the requirements from NUREG-0694) are applicable to applicants for operating licenses and such applicants are expected to meet the same schedule of implementation as indicated for operating reactors. Operating license reviews being finalized over the next few months will be handled on a case-by-case basis. Any item for which the implementation date is prior to the expected date of issuance of an operating license will be considered to be a prerequisite to obtaining that license. For such items, applicants must submit information or documentation four months prior to the staff's scheduled issuance of its Safety Evaluation Report or four months prior to the listed implementation date, whichever is later.

A large number of post-TMI requirements require the installation of a number of control room indications. It is important that licensees and applicants give consideration to human factor engineering considerations in planning for the installation of such new control room equipment. In the coming months, the NRC will be requiring human factors engineering reviews of control room designs as part of Action Plan Item I.D.1, and such an effort at this time may reduce the potential for later modifications. As an example of possible considerations, licensees and applicants might well consider at this time whether some control panel indications are of lesser safety significance and can be moved to other locations in the control room.

It is expected that the requirements contained herein will be met. However, it is recognized that licensees have proceeded with implementation of some of these items prior to issuance of these clarifying criteria. The staff will consider requests for relief from various aspects of these criteria. Such requests should explain the need for relief, include a clear description of design features of the proposed installation, and provide a safety rationale supporting the adequacy of the proposed installation. A licensee or applicant seeking relief from any element of our criteria should submit a request for relief, along with supporting justification, in response to this letter.

Accordingly, pursuant to §50.54(f) operating reactor licensees are requested to furnish, within forty-five (45) days of this letter, confirmation that the implementation dates indicated in Enclosure 1 will be met. For any date that cannot be met, furnish a proposed revised date, justification for the delay, and any planned compensating safety actions during the interim. After our evaluation of your response the NRC staff will take action, as necessary to assure that such reqirements and commitments are appropriately enforceable. This may include, as needed, issuance of a Confirmatory or Show-Cause Order.

Sincerely,

Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosures:
As stated

ENCLOSURE 1

POST-TMI REQUIREMENTS

FOR OPERATING REACTORS

ENCLOSURE 1

POST-TMI REQUIREMENTS FOR OPERATING REACTORS

[For postimplementation reviews, licensees shall comply with 10 CFR 50.59. If it is determined that an unreviewed safety question exists or a change to the facility's existing technical specifications is required, NRC approval is required before implementation.]

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Preimplementation Approval	Postimplementation Review	Tech Spec. Req.	Licensee Submittal Req. by	Remarks
I.A.1.1	Shift technical advisor	1. On duty	1/1/80	All	9/13/79	10/30/79	No	Yes	Yes	1/1/80	Complete
		2. Tech specs	12/15/80	All	7/2/80	7/2/80	Yes	No	Yes	9/1/80	
		3. Trained per LL Cat B	1/1/81	All	9/13/79	Encl. 3	No	Yes	Yes	1/1/81	
		4. Describe long-term program	1/1/81	All	*	Encl. 3	No	No	No	1/1/81	
I.A.1.2	Shift supervisor responsibilities	Delegate non-safety duties	1/1/80	All	9/13/79	10/30/80	No	Yes	No	1/1/80	Complete
I.A.1.3	Shift manning	1. Limit overtime	11/1/80	All	7/31/80	7/31/80	No	Yes	No	11/1/80	
		2. Min shift crew	7/1/82	All	7/31/80	7/31/80	No	Yes	Yes	11/1/80	Amend TS on shift manning
I.A.2.1	Immediate upgrading of RO & SRO training and qualifications	1. SRO exper	5/1/80	All	3/28/80	3/28/80	No	Yes	No	None	Completion to be verified
		2. SROs be ROs 1 yr	12/1/80	All	3/28/80	Encl 3	No	Yes	No	None	Completion by OIE
		3. Three mo trng on shift	8/1/80	All	3/28/80	3/28/80	No	Yes	No	None	Completion by OIE
		4. Modify training	8/1/80	All	3/28/80	3/28/80	No	Yes	No	8/1/80	NRR staff to review
		5. Facility certification	5/1/80	All	3/28/80	3/28/80	No	Yes	No	None	OIE verification
I.A.2.3	Administration of training programs	Instructors complete SRO exam.	8/1/80	All	3/28/80	3/28/80	No	Yes	No	None	NRR to verify conformance
I.A.3.1	Revise scope & criteria for licensing exams	1. Increase scope	5/1/80	All	3/28/80	3/28/80	No	No	No	None	
		2. Increase passing grade	5/1/80	All	3/28/80	3/28/80	No	No	No	None	
		3. Simulator exams	6/1/80	Plants having simulator	*	None	No	No	No	None	
			10/1/81	All	*	Encl 3	No	No	No	None	Plants w/o simulators

Note: For complete reference citation of NUREG reports, see Appendix A.

*Requirement formally issued by this letter.

ENCLOSURE 1 (CONTINUED)

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Preimplementation Approval	Postimplementation Review	Tech Spec. Req.	Licensee Submittal Req. by	Remarks
I.C.1	Short-term accident & procedures review	1. SB LOCA	6/1/80	All	9/13/79	10/30/79	No	Yes	No	None	Complete
		2. Inadequate core cooling									
		a. Reanalyze & propose guidelines	1/1/81	All	9/13/79	Encl 3	Yes	No	No	1/1/81	
		b. Revise procedures	First refueling outage after 1/1/82	All	9/13/79	Encl 3	Yes	No	No	Not determined	
		3. Transients & accidents									
		a. Reanalyze & propose guidelines	1/1/81	All	9/13/79	Encl 3	Yes	No	No	1/1/81	
		b. Revise procedures	First refueling outage after 1/1/82	All	9/13/79	Encl 3	Yes	No	No	Not determined	
I.C.2	Shift & relief turnover procedures	Implement shift turnover checklist	1/1/80	All	9/13/79	10/30/79	No	Yes	No	1/1/80	Complete
I.C.3	Shift-supervisor responsibility	Clearly define superv & oper responsibilities	1/1/80	All	9/13/79	10/30/79	No	Yes	No	1/1/80	Complete
I.C.4	Control-room access	Establish authority, limit access	1/1/80	All	9/13/79	10/30/79	No	Yes	No	1/1/80	Complete
I.C.5	Feedback of operating experience	Licensee to implement procedures	1/1/81	All	5/7/80	Encl 3	No	Yes	No	None	
I.C.6	Verify correct performance of operating activities	Revise performance procedures	1/1/81	All	*	Encl 3	No	Yes	No	None	
I.D.1	Control-room design reviews	Preliminary assessment & schedule for correcting deficiencies	TBD	All	6/26/80	NUREG/CR-1580 (Draft)				4/82	Final guidance will be issued 1981 as NUREG-0700

*Requirement formally issued by this letter.

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Preimplementation Approval	Postimplementation Review	Tech Spec. Req.	Licensee Submittal Req. by	Remarks
I.D.2	Plant-safety-parameter display console	1. Description	TBD	All	6/26/80	Encl 3				Later	Guidance per NUREG-0696 Rev. 2
		2. Installed	TBD	All	6/26/80	Encl 3					
		3. Fully implemented	TBD	All	6/26/80	Encl 3					
II.B.1	Reactor-coolant-system vents	1. Design vents	7/1/81	All	9/13/79	10/30/79	No	Yes	No	7/1/81	Complete
		2. Install vents (LL Cat B)	7/1/82	All	9/13/79	10/30/79	Yes	No	Yes	7/1/81	
		3. Procedures	1/1/82	All	9/13/79	Encl 3	Yes	No	Yes	1/1/81	
II.B.2	Plant shielding	1. Review designs	1/1/80	All	9/13/79	10/30/79	No	Yes	No	1/1/80	Complete
		2. Plant modifications (LL Cat B)	1/1/82	All	9/13/79 H376 -224	10/30/79 Encl 3	No	Yes	No	1/1/82	
		3. Equipment qualification	6/30/82	All	CLI-80-21	Encl 3	No	Yes	No	11/1/80	
II.B.3	Postaccident sampling	1. Interim system	1/1/80	All	9/13/79	10/30/79	No	Yes	No	1/1/80	Complete
		2. Plant modifications (LL Cat B)	1/1/82	All	9/13/79	10/30/79 Encl 3	No	Yes	Yes	1/1/81 submittal if deviation from position	
II.B.4	Training for mitigating core damage	1. Develop training program	1/1/81	All	3/28/80	3/28/80 Encl 3	No	Yes	No	1/1/81	
		2. Implement program									
		a. Initial	4/1/81	All	3/28/80	Encl 3	No	Yes	No		
		b. Complete	10/1/81	All	3/28/80	Encl 3	No	Yes	No		
II.D.1	Relief & safety-valve test requirements	1. Submit program	1/1/80	All	9/13/79	10/30/79	No	Yes	No	1/1/80	Complete
		2. RV & SV testing (LL Cat B)									
		a. Complete testing	7/1/81	All	9/13/79	10/30/79	No	No	No	7/1/81	
		b. Plant-specific report	10/1/81	All	9/13/79	Encl 3	Yes	Yes	TBD	1/1/82	
		3. Block-valve testing	7/1/82	PWR	*	Encl 3	Yes	Yes	TBD	7/1/82	
II.D.3	Valve position indication	1. Install direct indications of valve position	1/1/80	All	9/13/79	10/30/79	No	Yes	Yes	1/1/80	Complete
		2. Tech specs	12/15/80	All	7/2/79	7/2/80	Yes	No	Yes	9/1/80	

*Requirement formally issued by this letter.

ENCLOSURE 1 (CONTINUED)

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Preimplementation Approval	Postimplementation Review	Tech Spec. Req.	Licensee Submittal Req. by	Remarks
II.E.1.1	Auxiliary feedwater system evaluation	1. Short term	7/1/81	PWR	3/10/80	Encl 3	Yes	Yes	Item specific	Plant specific	
		2. Long term	1/1/82	PWR	4/24/80	Encl. 3	Yes	Yes	Item specific	Plant specific	
II.E.1.2	Auxiliary feedwater system initiation & flow	1. Initiation									
		a. Control grade	6/1/80	PWR	9/13/79	10/30/79	Yes	No	Yes	1/1/80	Complete
		b. Safety grade	7/1/81	PWR	9/13/80	10/30/79	No	Yes	Yes	1/1/81	
		2. Flow indication									
		a. Control grade	1/1/80	PWR	9/13/79	10/30/79	No	Yes	Yes	1/1/80	Complete
		b. LL Cat A tech specs	12/15/80	PWR	9/13/79	7/2/80	Yes	No	Yes	9/1/80	
		c. Safety grade	7/1/81	PWR	9/13/79	10/30/79	No	Yes	Yes	1/1/81	
II.E.3.1	Emergency power for pressurizer heaters	1. Upgrade power supply	1/1/80	PWR	9/13/79	10/30/79	No	Yes	Yes	1/1/81	Complete
		2. Tech specs	12/15/80	PWR	9/13/79	7/2/80 Encl 3	Yes	No	Yes	9/1/80	See II.G.1
II.E.4.1	Dedicated hydrogen penetrations	1. Design	1/1/80	All	9/13/79	10/30/79	Yes	No	No	1/1/80	Complete
		2. Install	7/1/81	All	9/13/79	10/30/79 Encl 3	No	Yes	No	7/1/81	
II.E.4.2	Containment isolation dependability	1-4. Imp diverse isolation	1/1/80	All	9/13/79	10/30/79	No	Yes	Yes	1/1/80	Complete
		5. Cntmt pressure setpoint									
		a. Specify pressure	1/1/81	All	*	Encl 3	No	Yes	No	1/1/81	
		b. Modifications	7/1/81	All	*	Encl 3	Yes	No	Yes	1/1/81	
		6. Cntmt purge valves	1/1/81	All	*	Encl 3	No	Yes	Yes	1/1/81	
		7. Radiation signal on purge valves	7/1/81	All	*	Encl 3	No	Yes	Yes	7/1/81	
		8. Tech specs	12/15/80	All	9/13/79	7/2/80	Yes	No	Yes	9/1/80	

*Requirement formally issued by this letter.

ENCLOSURE 1 (CONTINUED)

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Preimplementation Approval	Postimplementation Review	Tech Spec. Req.	Licensee Submittal Req. by	Remarks
II.F.1	Accident-monitoring	1. Noble gas monitor	1/1/82	All	9/13/79	10/30/79 Encl 3	No	Yes	Yes	1/1/81 Submittal if deviation from position	
		2. Iodine/ particulate sampling	1/1/82	All	9/13/79	10/30/79 Encl 3	No	Yes	Yes	1/1/81 submittal if deviation from position	
		3. Containment high-range monitor	1/1/82	All	9/13/79	10/30/79 Encl 3	No	Yes	Yes	7/1/81 submittal if deviation from position	
		4. Containment pressure	1/1/82	All	9/13/79	10/30/79 Encl 3	No	Yes	Yes	1/1/82	
		5. Containment water level	1/1/82	All	9/13/79	10/30/79 Encl 3	No	Yes	Yes	1/1/82	
		6. Containment hydrogen	1/1/82	All	9/13/79	10/30/79 Encl 3	No	Yes	Yes	1/1/82	
II.F.2	Instrumentation for detection of inadequate core cooling	1. Subcool meter	1/1/80	PWR	9/13/79	10/30/79	No	Yes	Yes	1/1/80	Complete
		2. Tech spec (LL Cat A)	12/15/80	PWR	7/2/79	7/2/80	Yes	No	Yes	9/1/80	
		3. Install level instruments (LL Cat B)	1/1/82	All	9/13/79	10/30/79 Encl 3	No	Yes	Yes	1/1/81	
II.G.1	Power supplies for pressurizer relief valves, block valves, & level indicators	1. Upgrade to emerg sources	1/1/80	PWR	9/13/79	10/30/79	No	Yes	Yes	1/1/80	Complete
		2. Tech specs	12/15/80	PWR	7/2/80	7/2/80	Yes	No	Yes	9/1/80	See II.E.3.1
II.K.1	IE Bulletins	79-05, 06, 08	Bulletin specific	All	4/79	NA	No	Yes	No	Bulletin specific	NRR evaluating licensee responses
II.K.2	Orders on B&W plants	8. Upgrade AFW system	See II.E.1.1	B&W	Per order	8/21/79, Encl 3	Yes	No	As required	See I.E.1.1	
		9. FEMA on ICS	TBD	B&W	Per order	11/7/79, Encl 3	No	Yes	No		Plant specific
		10. Safety-grade trip	7/1/81	B&W	Per order	12/20/79,	Yes	No	Yes	1/1/81	
		11. Operator training, drilling	Complete	B&W	Per order	Encl 3	No	Yes	No	Complete	
		13. Thermal-mechanical	1/1/81	B&W	8/21/79	Encl 3	No	Yes	As required	1/1/81	
		report	1/1/82	C-E, W	*	Encl 3	No	Yes	As required	1/1/82	

*Requirement formally issued by this letter.

ENCLOSURE 1 (CONTINUED)

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Preimplementation Approval	Postimplementation Review	Tech Spec. Req.	Licensee Submittal Req. by	Remarks
II.K.2	Orders on B&W plants (continued)	14. Lift frequency of PORVs & SVs	See II.K.3.7	B&W	9/28/79	NUREG-0565	No	Yes	No	See II.K.3.7	
		15. Effects of slug flow on OTSGS	Complete	B&W	11/21/79	Encl 3	No	Yes	No	Complete, under staff review	
		16. RCP seal damage	Complete	B&W	11/21/79	Encl 3	No	Yes	No	Complete, under staff review	
		17. Voiding in RCS	a. Complete	B&W	1/9/80	Encl 3	No	Yes	No	Complete, under staff review	
			b. 1/1/82	C-E, W	*	Encl 3	No	Yes	No	1/1/82	
		19. Benchmark analysis of seq AFW flow	a. Complete	B&W	8/21/79	Encl 3	No	Yes	No	Complete, under staff review	
			b. 1/1/82	C-E, W-	*	Encl 3	No	Yes	No	1/1/82	
		20. System response to SB LOCA	Complete	B&W	8/21/79	Encl 3	No	Yes	No	Complete, under staff review	
II.K.3	Final recommendations, B&O task force	1. Auto PORV isolation									
		a. Design	7/1/81	PWR	*	Encl 3	Yes	No	Yes	7/1/81	If required by II.K.3.2
		b. Test/install	1st refuel 6 mo after staff approval	PWR	*	Encl 3	Yes	No	Yes	7/1/81	
		2. Report on PORV failures	1/1/81	PWR	5/7/80	Encl 3	No	Yes	No	1/1/81	
		3. Reporting SV & RV failures & challenges	1/1/81	All	5/7/80	None	No	Yes	Yes	1/1/81	Initiate data beginning 4/1/80
		5. Auto trip of RCPs									
		a. Propose modifications	7/1/81	PWR	5/7/80	Encl 3	No	Yes	No	2/15/81	
		b. Modify	3/1/82	PWR	5/7/80	Encl 3	Yes	No	Yes	7/1/81	If required
		7. Eval of PORV opening probability	1/1/81	B&W	*	Encl 3	No	Yes	No	1/1/81	

*Requirement formally issued by this letter.

ENCLOSURE 1 (CONTINUED)

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Preimplementation Approval	Postimplementation Review	Tech Spec. Req.	Licensee Submittal Req. by	Remarks
II.K.3	Final recommendations, B&O task force (continued)	9. PID controller	1/1/81	W	5/7/80	Encl 3	No	Yes	No	12/1/80	Implementation to be verified.
		10. Proposed anticipatory trip modifications	Plant specific	Select W	5/7/80	Encl 3	Yes	No	Yes	Plant specific	
		11. Justify use of certain PORV	Plant specific	Plant specific	*	None	No	Yes	No	Plant specific	See Sect. 3.2.4.d of NUREG-0611
		12. Anticipatory trip on turbine trip									
		a. Confirmation or propose modifications	1/1/81	W	5/7/80	Encl 3	No	Yes	No	1/1/81	
		b. Modify	1st refuel or 6 mo after staff approval	W	5/7/80	Encl 3	Yes	No	Yes	1st refuel after tech spec amend request	
		13. HPCI & RCIC init levels									
		a. Analysis	1/1/81	BWR	5/7/80	Encl 3	No	Yes	No	1/1/81	
		b. Modify	7/1/81	BWR	5/7/80		Yes	No	Yes	1/1/81	
		14. Iso condenser isol modification	1/1/82	BWR w/ iso cond	5/7/80	Encl 3	No	Yes	Yes	7/1/81	
		15. Isolation of HPCI & RCIC modification	7/1/81	BWR	5/7/80	Encl 3	No	Yes	Yes	1/1/81	
		16. Challenges & failures to relief valves									
		a. Study	4/1/81	BWR	5/7/80	Encl 3	No	Yes	No	4/1/81	
		b. Modify	1st refuel or within 1 yr after approval	BWR	5/7/80	Encl 3	Yes	No	Yes	4/1/81	
		17. ECC system outages	1/1/81	All	5/7/80	Encl 3	No	Yes	As required	1/1/81	

*Requirement formally issued by this letter.

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Preimplementation Approval	Postimplementation Review	Tech Spec. Req.	Licensee Submittal Req. by	Remarks
II.K.3	Final recommendations, B&O task force (continued)	18. ADS actuation									
		a. Study	4/1/81	BWR	5/7/80	Encl 3	No	Yes	No	4/1/81	
		b. Propose mods	4/1/82	BWR	5/7/80	Encl 3	No	Yes	Yes	4/1/82	
		c. Modifications	1st refuel 6 mo after staff approval	BWR	5/7/80 Encl 3	Encl 3	Yes	No	Yes	Refuel	
		19. Interlock recirc pump modification	7/1/81	BWR	5/7/80	Encl 3	No	Yes	Yes	7/1/81	
		20. Loss of SVC water at BRP	7/1/81	Big Rock	5/7/80	Encl 3	No	Yes	TBD	7/1/81	
		21. Restart of CSS & LPCI									
		a. Design	1/1/81	BWR	5/7/80	Encl 3	No	Yes	No	1/1/81	
		b. Modifications	1st refuel 6 mo after staff approval	BWR	5/7/80	Encl 3	Yes	No	Yes	1/1/81	
		22. RCIC suction									
		a. Verify procedures	1/1/81	BWR	5/7/80	Encl 3	No	Yes	No	1/1/81	
		b. Modifications	1/1/82	BWR	5/7/80	Encl 3	No	Yes	Yes	1/1/82	
		24. Space cooling for HPCI/RCIC modifications	1/1/82	BWR	5/7/80	Encl 3	No	Yes	Yes	1/1/82	
		25. Power on pump seals									
		a. Propose mods	7/1/81	BWR	5/7/80	Encl 3	No	Yes	No	7/1/81	
			1/1/82	C-E & W	*	Encl 3	No	Yes	No	1/1/82	
		b. Modifications	1/1/82	BWR	5/7/80	Encl 3	Yes	No	No	1/1/82	
			7/1/82	C-E & W	*	Encl 3	Yes	No	No	7/1/82	
		27. Common ref level	7/1/81	BWR	5/7/80	Encl 3	No	Yes	Yes	1/1/81	
		28. Qual of ADS accumulators	1/1/82	BWR	5/7/80	Encl 3	No	Yes	Yes	1/1/82	
		29. Performance of isolation condensors	4/1/81	BWR w/iso cond	5/7/80	Encl 3	No	Yes	No	4/1/81	

*Requirement formally issued by this letter.

ENCLOSURE 1 (CONTINUED)

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Preimplementation Approval	Postimplementation Review	Tech Spec. Req.	Licensee Submittal Req. by	Remarks
II.K.3	Final recommendations, B&O task force (continued)	30. SB LOCA methods									
		a. Schedule outline	11/15/80	All	5/7/80	Encl 3	No	Yes	No	11/15/80	
		b. Model	1/1/82	All	5/7/80	Encl 3	Yes	No	No	1/1/82	
		c. New analyses	1/1/83 or 1 yr after staff approval	All	5/7/80	Encl 3	Yes	No	No	1/1/83 or 1 yr after staff approval	
		31. Compliance with CFR 50.46	1/1/83 or 1 yr after staff approval	All	5/7/80	Encl 3	Yes	No	TBD	1/1/83	
		40. RCP seal damage	See II.K.2.16								
		43. Effects of slug flow	See II.K.2.15								
		44. Eval transient with single failure	1/1/81	BWR	5/7/80	Encl 3	No	Yes	TBD	1/1/81	
		45. Manual depressurization	1/1/81	BWR	5/7/80	Encl 3	No	Yes	TBD	1/1/81	
		46. Michelson concerns	Complete	BWR	5/7/80	None	No	Yes	No	7/1/80	NRR to verify compliance
		57. Manual act of ADS	TBD	BWR	5/7/80	Encl 3	No	Yes	No	TBD	No licensee action until guidelines approved by staff
III.A.1.1	Emergency preparedness, short term	Short-term improvements	Complete	All	10/10/79	NUREG-0654	No	Yes	No	Complete	
III.A.1.2	Upgrade emergency support facilities	1. Interim TSC OSC & EOF	1/1/80	All	9/13/79		No	Yes	No	Complete	
		2. Design	TBD	TBD	TBD		TBD	TBD	TBD	TBD	
		3. Modifications	TBD	TBD	TBD		TBD	TBD	TBD	TBD	
III.A.2	Emergency preparedness	1. Upgrade emergency plans to App. E, 10 CFR 50	3/1/81	All	8/19/80	NUREG-0654	No	Yes	Yes	1/2/81	Procedures submitted 3/1/81
		2. Meteorological data	6/1/83	All	8/19/80	NUREG-0654	No	Yes	Yes	1/2/81	Staged implementation
III.D.1.1	Primary coolant outside containment	1. Leak reduction	Complete	All	9/13/79	10/30/79	No	Yes	Yes	Complete	
		2. Tech specs	12/15/80	All	7/2/79	7/2/80	Yes	No	Yes	9/1/80	

ENCLOSURE 1 (CONTINUED)

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Preimplementation Approval	Postimplementation Review	Tech Spec. Req.	Licensee Submittal Req. by	Remarks
III.D.3.3	Inplant radiation monitoring	1. Provide means to determine presence of radioiodine	Complete	All	9/13/79	10/30/79	No	Yes	No	Complete	
		2. Modifications to accurately measure I_2	1/1/81	All	9/13/79	Encl 3	No	Yes	Yes	1/1/81	
III.D.3.4	Control-room habitability	1. Review	1/1/81	All	5/7/80	Encl 3	No	Yes	No	1/1/81	
		2. Modifications	TBD	All	5/7/80	Encl 3	No	Yes	Yes	1/1/81	

ENCLOSURE 2

TMI ACTION PLAN REQUIREMENTS

FOR APPLICANTS FOR AN OPERATING LICENSE

ENCLOSURE 2

TMI ACTION PLAN REQUIREMENTS FOR APPLICANTS FOR AN OPERATING LICENSE

[If implementation date is earlier than issuance of operating license, the implementation date will be the licensing date.]

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Tech Spec. Req.	Remarks
I.A.1.1	Shift technical advisor	1. On shift	Fuel load	All	9/27/79	11/9/79	Yes	
		2. Training per LL Cat B	Fuel load	All	9/27/79	Encl. 3	No	
		3. Describe long-term program	Consistent with OL review schedule	All	*	Encl. 3	No	
I.A.1.2	Shift supervisor responsibilities	Delegate nonsafety duties	Fuel load	All	9/27/79	11/9/79	No	
I.A.1.3	Shift manning	1. Limit overtime	Fuel load	All	6/26/80	7/31/80 Enc. 3	No	
		2. Minimum shift crew	Fuel load	All	6/26/80	7/31/80 Enc. 3	Yes	Case by case
I.A.2.1	Immediate upgrade of RO & SRO training and qualifications	1. SRO experience	Fuel load	All	3/28/80	None	No	
		2. SROs be ROs, 1 yr	Initial criticality	All	3/28/80	3/28/80 Encl. 3	No	
		3. 3 mo training on-shift	Fuel load	All	3/28/80	None	No	
		4. Modify training	Fuel load	All	3/28/80	None	No	
		5. Facility certification	Fuel load	All	3/28/80	None	No	
I.A.2.3	Administration of training programs	Instructors complete SRO exam	2 mo prior to issuance of license	All	3/28/80	Encl. 3	No	
I.A.3.1	Revise scope & criteria for licensing exams	1. Increase scope	10/1/80	All	3/28/80	None	No	
		2. Increase passing grade	10/1/80	All	3/28/80	None	No	
		3. Simulator exams						
		a. Plants with simulators	Prior to fuel load	Plants having simulators	*	Encl. 3	No	
		b. All plants	Prior to fuel load or 10/1/81 whichever is later	All	*	Encl. 3	No	

Note: For complete reference citiation of NUREG reports, see Appendix A.

*Requirement formally issued by this letter.

ENCLOSURE 2 (CONTINUED)

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Tech Spec. Req.	Remarks
I.B.1.2	Evaluation of organization & management	Organization, resources tng. & qualifications for operators & accidents	Fuel load	All	6/26/80	None	Yes	Draft guideline available.
I.C.1	Short-term accident & procedure review	1. SB LOCA	Fuel load	All	9/27/79	11/9/79	No	
		2. Inadequate core cooling a. Reanalyze & propose guidelines	Fuel load	All	9/27/79	Encl. 3	No	
		b. Revise procedures	First refueling outage after 1/1/82	All	9/27/79	Encl. 3	No	
		3. Transients & accidents a. Reanalyze & propose guidelines	Fuel load	All	9/27/79	Encl. 3	No	
		b. Revise procedures	First refueling outage after 1/1/82	All	9/27/79	Encl. 3	No	
I.C.2	Shift & relief turnover procedures	Revise procedures to assure plant status known by new shift	Fuel load	All	9/27/79	11/9/79	No	
I.C.3	Shift supervisor responsibility	Corporate directive to establish command duties & revise plant procedures	Fuel load	All	9/27/79	11/9/79	Yes	
I.C.4	Control-room access	Establish authority & limit access	Fuel load	All	9/27/79	11/9/79	No	
I.C.5	Feedback of operating experience	Review & revise procedures	1/1/81 or prior to issuance of OL	All	6/26/80	Encl. 3	No	
I.C.6	Verify correct performance of operating activities	Revise performance procedures	1/1/81 or prior to fuel load	All	*	Encl. 3	No	
I.C.7	NSSS vendor rev of proc	1. Low-power test program	Fuel load	All	6/26/80	None	No	
		2. Power ascension & emergency procedures	Full power	All	6/26/80	None	No	

*Requirement formally issued by this letter.

ENCLOSURE 2 (CONTINUED)

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Tech Spec. Req.	Remarks
I.C.8	Pilot mon of selected emergency proc for NTOLs	Correct procedure based on NRC sample audit	Full power	All	6/26/80	None	No	
I.D.1	Control-room design reviews	Preliminary assessment & schedule for correcting deficiencies	Prior to issuance of OL	All	6/26/80	NUREG-1580 (Draft)	No	Guidance and schedule being developed.
I.D.2	Plant-safety-parameter display console	1. Description 2. Installed 3. Fully implemented	TBD TBD TBD	All All All	6/26/80 6/26/80 6/26/80	Encl. 3 Encl. 3 Encl. 3	No No No	Guidance and schedule being developed in NUREG-0696.
I.G.1	Training during low-power testing	1. Propose tests 2. Submit analysis and procedures. 3. Training & results	Fuel load Fuel load Full power	All All All	6/26/80 6/26/80 6/26/80	 None 	No Yes No	
II.B.1	Reactor-coolant-system vents	1. Design & analyses 2. Install 3. Procedures	Full power 7/1/82 1/1/82 or prior to issuance of OL	All All All	9/27/79 9/27/79 9/27/79	11/9/79 Encl. 3 Encl. 3	No Yes Yes	
II.B.2	Plant shielding	1. Radiation & shielding review 2. Corrective actions to assure access 3. Complete mods 4. Equipment qualification	Δ Full power 1/1/82 Δ	All All All All	9/27/79 9/27/79 9/27/79 CLI-80-21	Encl. 3 Encl. 3 Encl. 3 Encl. 3	No No No No	
II.B.3	Postaccident sampling	1. Design review 2. Corrective actions 3. Procedures 4. Complete actions	Δ Full power Full power 1/1/82	All All All All	9/27/79 9/27/79 9/27/79 9/27/79	Encl. 3 Encl. 3 Encl. 3 Encl. 3	No Yes Yes Yes	
II.B.4	Training for mitigating core damage	1. Develop training program 2. Complete training	Fuel load Full power	All All	3/28/80 6/26/80	3/28/80 None	No No	

[Δ]Four months before operating license is issued or 4 months before date indicated.

ENCLOSURE 2 (CONTINUED)

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Tech Spec. Req.	Remarks
II.D.1	Relief & safety-valve test requirements	1. Describe program & schedule	Fuel load	All	9/27/79	11/9/79	No	
		2. RV & SV tests	Fuel load	All	9/27/79	11/9/79	To be determined	
		3. Block valve tests	Fuel load or by 7/1/82, whichever is later	PWR	*	11/9/79 Enc. 3		
II.D.3	Valve position indication	Install in control room	Δ	All	9/27/79	11/9/79 Encl. 3	Yes	
II.E.1.1	Auxiliary feedwater system evaluation	1. Analysis	Full power	C-E & W B&W	3/10/80 4/24/80	None None	No No	See 3/10/80 and 4/24/80 letters.
		2. Modification	Full power	PWR	4/24/80	None	As required	
II.E.1.2	Auxiliary feedwater system initiation and flow	1. Initiation						
		(a) Control grade	Fuel load	PWR	9/27/79	11/9/79	Yes	
		(b) Safety grade	Δ	PWR	9/27/79	11/9/79	Yes	
		2. Flow indication						
		(a) Control grade	Fuel load	PWR	9/27/79	11/9/79	Yes	
		(b) Safety grade	Δ	PWR	9/27/79	11/9/79	Yes	
II.E.3.1	Emergency power for pressurizer heaters	Installed capability	4 mo prior to issuance of SER	PWR	9/27/79	11/9/79 Encl. 3	Yes	
II.E.4.1	Dedicated hydrogen penetrations	1. Design	Δ	All	9/27/79	11/9/79	No	
		2. Review & revise H_2 control proc	Fuel load	All	9/27/79	Encl. 3	No	
		3. Install	7/1/81 or prior to issuance of OL	All	9/27/79	Encl. 3	No	

ΔFour months before operating license is issued or 4 months before date indicated.

*Requirement formally issued by this letter.

ENCLOSURE 2 (CONTINUED)

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Tech Spec. Req.	Remarks
II.E.4.2	Containment isolation dependability	1-4 Implement diverse isolation	Prior to issuance of OL	All	9/27/79	11/9/79 Encl. 3	Yes	
		5. Containment press setpoint	7/1/81 or prior to issuance of OL	All	*	Encl. 3	Yes	
		6. Containment purge valves	1/1/81 or prior to issuance of OL	All	*	Encl. 3	Yes	
		7. Radiation signal on purge valves	7/1/81 or prior to issuance of OL	All	*	Encl. 3	Yes	
II.F.1	Accident-monitoring instrumentation	1. Procedures	Fuel load	All	9/27/79	11/9/79 Encl 3	No	
		2. Install instrumentation						
		a. Noble gas monitor	1/1/82 Δ	All	9/27/79	11/9/79 Encl. 3	Yes	
		b. Iodine/particulate sampling	1/1/82 Δ	All	9/27/79	11/9/79 Encl. 3	Yes	
		c. Containment high range monitor	1/1/82 Δ	All	9/27/79	11/9/79 Encl. 3	Yes	
		d. Containment pressure	6 mo prior to issuance of OL	All	9/27/79	11/9/79 Encl. 3	Yes	
		e. Containment water level	7/1/82 or prior to issuance of OL	All	9/27/79	11/9/79 Encl. 3	Yes	
		f. Containment hydrogen	1/1/82 or prior to issuance of OL	All	9/27/79	11/9/79 Encl. 3	Yes	
II.F.2	Instrumentation for detection of inadequate core-cooling	1. Procedures instruments	Fuel load	PWR	9/27/79	11/9/79 Encl. 3	No	
		2. Subcooling meter	Fuel load	PWR	9/27/79	11/9/79 Encl. 3	Yes	
		3. Describe other instrumentation	Fuel load	All	9/27/79	11/9/79 Encl. 3	No	
		4. Install additional instrumentation	1/1/82	All	9/27/79	11/9/79 Encl. 3	Yes	

*Requirement formally issued by this letter.

ΔFour months before operating license is issued or 4 months before date indicated.

ENCLOSURE 2 (CONTINUED)

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Tech Spec. Req.	Remarks
II.G.1	Power supplies for pressurizer relief valves, block valves, & level indicators	Power supply from emergency buses	Fuel load	PWR	9/27/79	11/9/79	Yes	
II.K.1	IE Bulletins	5. Review ESF valves	Fuel load	All	IEB 79-05 79-05A 79-06A 79-06B 79-08 6/26/80	None	Yes	
		10. Operability status	Fuel load	All	IEB 79-05A 79-06A 79-06B 79-08 6/26/80	None	No	
		17. Trip per low-level B/S	Fuel load	W	IEB 79-06A 6/26/80	None	Yes	Also see II.K.2.10.
		20. Prompt manual reactor trip	Fuel load	B&W	IEB 79-05B 6/26/80	None	No	
		21. Auto SG anticipatory reactor trip	Fuel load	B&W	IEB 79-05B 6/26/80	None	Yes	
		22. Aux heat rem system, proc	Fuel load	BWR	IEB 79-08 6/26/80	None	No	
		23. RV level, procedures	Fuel load	BWR	IEB 79-08 6/26/80	None	Yes	
II.K.2	Orders on B&W plants	2. Procedures to control AFW ind of ICS	Δ	B&W	6/26/80	None	No	
		9. FMEA on ICS system	Δ	B&W	6/26/80	Encl. 3	As required	
		10. Safety-grade trip anticipatory	Fuel load	B&W	6/26/80	Encl. 3	Yes	
		13. Thermal mechanical report	#	B&W	6/26/80	Encl. 3	As required	
		14. Lift frequency of PORV & SVs	Fuel power	B&W	6/26/80	None	No	
		15. Effects of slug flow on OTSGS	Fuel power Δ	B&W	6/26/80	Encl. 3	No	

ΔFour months before operating license is issued or 4 months before date indicated.

#Six months before full-power license.

ENCLOSURE 2 (CONTINUED)

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Tech Spec. Req.	Remarks
II.K.2	Orders on B&W Plants (continued)	16. RCP seal damage	Fuel power Δ	B&W	6/26/80	Encl. 3	No	
		17. Voiding in RCS	1/1/82°	C-E & W	*	Encl. 3	No	
		19. Benchmark analysis seq AFW flow	1/1/82°	C-E & W	*	Encl. 3	No	
II.K.3	Final recommendations, B&O task force	1. Auto PORV isolation	1st refuel 6 mo. after staff approval	PWR	*	Encl. 3	Yes	
		2. Report on PORV failures	1/1/81 Δ	PWR	*	Encl. 3	No	
		3. Reporting SV & RV failures & challenges	Δ	All	6/26/80	None	Yes	See 5/7/80 letter to ORS.
		5. Auto trip of RCPs						
		a. Propose mods	Prior to OL	PWR	*	Encl. 3	No	
		b. Modify	Full power	PWR	*	Encl. 3	Yes	
		7. Evaluation of PORV opening probability	Full power	PWR	6/26/80	None	No	
		9. PID controller	Δ	W	6/26/80	Encl. 3	No	
		10. Applicant's propose anticipatory trip at high power	Δ	Selected W	6/26/80	Encl. 3	Yes	
		11. Justification use of certain PORVs	Fuel load	Plant specific	6/26/80	None	No	See NUREG-0611, Sect. 3.2.4.d.
		12. Confirm anticipatory trip						
		a. Propose modifications	Δ	W	6/26/80	Encl. 3	No	
		b. Modify	Δ	W	*	Encl. 3	Yes	
		13. HPCI & RCIC init levels						
		a. Analysis	Δ	BWR	*	Encl. 3	Yes	
		b. Modify	Δ	BWR	*	Encl. 3	Yes	
		15. Isolation of HPCI and RCIC	Δ	BWR	*	Encl. 3	Yes	
		16. Challenges to & failure of relief valves						
		a. Study	4/1/81	BWR	*	Encl. 3	No	
		b. Modify	1st refueling after staff approval	BWR	*	Encl. 3	Yes	

ΔFour months before operating license is issued or 4 months before date indicated.

°Or 6 months before fuel load.

*Requirement formally issued by this letter.

ENCLOSURE 2 (CONTINUED)

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Tech Spec. Req.	Remarks
II.K.3	Final recommendations, B&O task force (continued)	17. ECCS outages	In accordance with review schedule for licensing	All	*	Encl. 3	As required	
		18. ADS actuation						
		a. Study	1 yr prior to OL	BWR	*	Encl. 3	No	
		b. Proposed mods	4 mo prior to OL	BWR	*	Encl. 3	Yes	
		c. Modify	1st refuel 6 mo after staff approval	BWR	*	Encl. 3	Yes	
		21. Restart of LPCS & LCPI						
		a. Design	1/1/81 Δ	BWR	*	Encl. 3	No	
		b. Modification	1st refuel 6 mo after staff approval	BWR	*	Encl. 3	Yes	
		22. RCIC suction						
		a. Procedures	1/1/81 Δ	BWR	*	Encl. 3	No	
		b. Modification	1/1/82 Δ	BWR	*	Encl. 3	Yes	
		24. Space cooling for HPCI/RCIC, modifications	Δ	BWR	*	Encl. 3	Yes	
		25. Power on pump seals						
		a. Propose mods	7/1/81 or 6 mo prior to SER	BWR C-E & W	*	Encl. 3	No	
		b. Modifications	Full power	BWR	*	Encl. 3	Yes	
				C-E & W	*	Encl. 3	Yes	
		27. Common reference level	7/1/81 Δ	BWR	*	Encl. 3	Yes	
		28. Qual of ADS accumlators	1/1/82 Δ	BWR	*	Encl. 3	As required	
		30. SB LOCA methods						
		a. Schedule outline	In accordance with review schedule	All	*	Encl. 3	No	
		b. Model	In accordance with review schedule	All	*	Encl. 3	No	

*Requirement formally issued by this letter.

ΔFour months before operating license is issued or 4 months before date indicated.

ENCLOSURE 2 (CONTINUED)

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Tech Spec. Req.	Remarks
II.K.3	Final recommendations, B&O task force (continued)	c. New analyses	In accordance with review schedule	All	*	Encl. 3	No	
		31. Plant-specific analysis	1/1/83Δ	All	*	Encl. 3	No	
		44. Evaluate transients with single failure	1/1/81Δ	BWR	*	Encl. 3	As required	
		45. Manual depressurization	1/1/81Δ	BWR	*	Encl. 3	No	
		46. Michelson concerns	Fuel load	BWR	*	Encl. 3	No	
III.A.1.1	Emergency preparedness, short term	Short-term improvements	Fuel load	All	8/19/80	NUREG-0654	No	Use NUREG-0654 until Rev. I is issued (due out 10/80).
III.A.1.2	Upgrade emergency support facilities	1. Establish TSC, OSC, EOF (interim basis)	TBD	All	9/27/79	11/9/79	No	
		2. Design	TBD	TBD	TBD	TBD	TBD	
		3. Modifications	TBD	TBD	TBD	TBD	TBD	
III.A.2	Emergency preparedness	1. Upgrade emergency plans to App E, 10 CFR 50	Fuel load	All	8/19/80	NUREG-0654	No	
		2. Meteorological data	Fuel load	All	6/26/80	NUREG-0654	No	
III.D.1.1	Primary coolant outside containment	Measure leak rates & establish program to keep leakage ALARA	Full power	All	9/27/79	11/9/79 Encl. 3	Yes	
III.D.3.3	Inplant I_2 radiation monitoring	1. Provide means to determine presence of radioiodine	Fuel load	All	9/27/79	11/9/79 Encl. 3	Yes	
		2. Modifications to accurately measure radioiodine	1/1/81 or prior to licensing	All	9/27/79	11/9/79 Encl. 3	Yes	
III.D.3.4	Control-room habitability	1. Identify and evaluate potential hazards	Full power	All	6/26/80	Encl. 3	No	
		2. Schedule for modifications	Full power	All	6/26/80	Encl. 3	No	
		3. Modifications	Full power	All	6/26/80	Encl. 3	Yes	

ΔFour months before operating license is issued or 4 months before date indicated.

*Requirement formally issued by this letter.

ENCLOSURE 3

CLARIFICATION OF

TMI ACTION PLAN REQUIREMENTS

I.A.1.1 SHIFT TECHNICAL ADVISOR

Position

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor (STA) may serve more than one unit at a multiunit site if qualified to perform the advisor function for the various units.

The STA shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The STA shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the STAs that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements resulting from NUREG-0660 and the October 30, 1979 letter from H. R. Denton to all operating nuclear power plants.

Clarification

The letter of October 30, 1979 clarified the short-term STA requirements. That letter indicated that the STAs must have completed all training by January 1, 1981. This paper confirms these requirements and requests additional information.

The need for the STA position may be eliminated when the qualifications of the shift supervisors and senior operators have been upgraded and the man-machine interface in the control room has been acceptably upgraded. However, until those long-term improvements are attained, the need for an STA program will continue.

The staff has not yet established the detailed elements of the academic and training requirements of the STA beyond the guidance given in its October 30, 1979 letter. Nor has the staff made a decision on the level of upgrading required for licensed operating personnel and the man-machine interface in the control room that would be acceptable for eliminating the need of an STA. Until these requirements for eliminating the STA position have been established, the staff continues to require that, in addition to the staffing requirements specified in its July 31, 1980 letter (as revised by item I.A.1.3 of this enclosure), an STA be available for duty on each operating shift when a plant is being operated in Modes 1-4 for a PWR and Modes 1-3 for a BWR. At other times, an STA is not required to be on duty.

Since the October 30, 1979 letter was issued, several efforts have been made to establish, for the longer term, the minimum level of experience, education, and training for STAs. These efforts include work on the revision to ANS-3.1, work by the Institute of Nuclear Power Operations (INPO), and internal staff efforts.

INPO recently made available a document entitled "Nuclear Power Plant Shift Technical Advisor--Recommendations for Position Description, Qualifications, Education and Training." A copy of Revision 0 of this document, dated April 30, 1980, is attached as Appendix C. Sections 5 and 6 of the INPO document describe the education, training, and experience requirements for STAs. The NRC staff finds that the descriptions as set forth in Sections 5 and 6 of Revision 0 to the INPO document are an acceptable approach for the selection and training of personnel to staff the STA positions. (Note: This should not be interpreted to mean that this is an NRC requirement at this time. The intent is to refer to the INPO document as acceptable for interim guidance for a utility in planning its STA program over the long term (i.e., beyond the January 1, 1981 requirement to have STAs in place in accordance with the qualification requirements specified in the staff's October 30, 1979 letter).)

No later than January 1, 1981, all licensees of operating reactors shall provide this office with a description of their STA training program and their plans for requalification training. This description shall indicate the level of training attained by STAs by January 1, 1981 and demonstrate conformance with the qualification and training requirements in the October 30, 1979 letter. Applicants for operating licenses shall provide the same information in their application, or amendments thereto, on a schedule consistent with the NRC licensing review schedule.

No later than January 1, 1981, all licensees of operating reactors shall provide this office with a description of their long-term STA program, including qualification, selection criteria, training plans, and plans, if any, for the eventual phaseout of the STA program. (Note: The description shall include a comparison of the licensee/applicant program with the abovementioned INPO document. This request solicits industry views to assist NRC in establishing long-term improvements in the STA program. Applicants for operating licenses shall provide the same information in their application, or amendments thereto, on a schedule consistent with the NRC licensing review schedule.)

Applicability

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

Implementation

(1) Training that meets the lessons-learned requirements shall be completed by January 1, 1981 or by the time the fuel-loading license is issued, whichever is later.

(2) A description of the current training program and demonstration of conformance with the October 30, 1979 letter shall be submitted

(a) no later than January 1, 1981 for licensees of operating reactors; and

(b) on a schedule consistent with review schedule for applicants for operating licenses.

(3) A description of the long-term STA program shall be submitted

(a) no later than January 1, 1981 for licensees of operating reactors; and

(b) on a schedule consistent with review schedule for applicants for operating licenses.

<u>Type of Review</u>

Operating reactors will undergo postimplementation review.

Applicants for operating licenses will be reviewed as part of the licensing review.

<u>Documentation Required</u>

Documentation will be required as noted above.

<u>Technical Specification Changes Required</u>

Changes to technical specifications will be required.

<u>References</u>

NUREG-0578, Recommendation 2.2.1.b

NUREG-0660

INPO Document, see Appendix C

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

Letter from D. G. Eisenhut, NRC, to All Licensees and Applicants, dated July 31, 1980.

I.A.1.3 SHIFT MANNING

Position

This position defines shift manning requirements for normal operation. The letter of July 31, 1980 from D. G. Eisenhut to all power reactor licensees and applicants (copy attached) sets forth the interim criteria for shift staffing (to be effective pending general criteria that will be the subject of future rulemaking). Overtime restrictions were also included in the July 31, 1980 letter.

Changes to Previous Requirements and Guidance

Errors were discovered in the last column of the table attached to the letter of July 31, 1980. A corrected table is enclosed; a bar in the margin indicates the correction. (See p. I.A.1.3-4.)

The overtime requirements have been rewritten to be more flexible.

Clarification

Page 3 of the July 31, 1980 letter is superseded in its entirety by the following:

Licensees of operating plants and applicants for operating licenses shall include in their administrative procedures (required by license conditions) provisions governing required shift staffing and movement of key individuals about the plant. These provisions are required to assure that qualified plant personnel to man the operational shifts are readily available in the event of an abnormal or emergency situation.

These administrative procedures shall also set forth a policy, the objective of which is to operate the plant with the required staff and develop working schedules such that use of overtime is avoided, to the extent practicable, for the plant staff who perform safety-related functions (e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, I&C technicians and key maintenance personnel).

IE Circular No. 80-02, "Nuclear Power Plant Staff Work Hours," dated February 1, 1980 (copy attached) discusses the concern of overtime work for members of the plant staff who perform safety-related functions.

The staff recognizes that there are diverse opinions on the amount of overtime that would be considered permissible and that there is a lack of hard data on the effects of overtime beyond the generally recognized normal 8-hour working day, the effects of shift rotation, and other factors. NRC has initiated studies in this area. Until a firmer basis is developed on working hours, the administrative procedures shall include as an interim measure the following guidance, which generally follows that of IE Circular No. 80-02.

In the event that overtime must be used (excluding extended periods of shutdown for refueling, major maintenance or major plant modifications), the following overtime restrictions should be followed:

(1) An individual should not be permitted to work more than 12 hours straight (not including shift turnover time).

(2) There should be a break of at least 12 hours (which can include shift turnover time) between all work periods.

(3) An individual should not work more than 72 hours in any 7-day period.

(4) An individual should not be required to work more than 14 consecutive days without having 2 consecutive days off.

However, recognizing that circumstances may arise requiring deviation from the above restrictions, such deviation shall be authorized by the plant manager or his deputy, or higher levels of management in accordance with published procedures and with appropriate documentation of the cause.

If a reactor operator or senior reactor operator has been working more than 12 hours during periods of extended shutdown (e.g., at duties away from the control board), such individuals shall not be assigned shift duty in the control room without at least a 12-hour break preceding such an assignment.

NRC encourages the development of a staffing policy that would permit the licensed reactor operators and senior reactor operators to be periodically assigned to other duties away from the control board during their normal tours of duty.

If a reactor operator is required to work in excess of 8 continuous hours, he shall be periodically relieved of primary duties at the control board, such that periods of duty at the board do not exceed about 4 hours at a time.

The guidelines on overtime do not apply to the shift technical advisor provided he or she is provided sleeping accommodations and a 10-minute availability is assured.

Operating license applicants shall complete these administrative procedures before fuel loading. Development and implementation of the administrative procedures at operating plants will be reviewed by the Office of Inspection and Enforcement beginning 90 days after July 31, 1980.

See section III.A.1.2 for minimum staffing and augment capabilities for emergencies.

Applicability

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

Implementation

(1) Overtime administrative procedures shall be established for operating reactors by November 1, 1980 and by fuel loading for applicants for operating licenses.

(2) Staffing requirements shall be completed by July 1, 1982 for operating reactors and by fuel load for operating license applicants.

Type of Review

A postimplementation review will be performed on operating reactors.

Applicants for operating licenses will be reviewed prior to implementation.

Documentation Required

The documentation required is as noted in the letter of July 31, 1980.

Technical Specification Changes Required

Changes to technical specifications will be required for minimum shift crew manning.

References

NUREG-0660

IE Circular No. 80-02, "Nuclear Power Plant Staff Work Hours," February 1, 1980

Letter from D. G. Eisenhut, NRC, to All Power Reactor Licensees, July 31, 1980.

NEW GUIDANCE FOR INTERIM REQUIRED SHIFT STAFFING

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

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

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

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

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

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

SSINS No.: 6830
Accession No.:
7912190657

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

February 1, 1980

IE Circular No. 80-02

NUCLEAR POWER PLANT STAFF WORK HOURS

Description of Circumstances:

Studies indicate that with fatigue, especially because of loss of sleep, an individual's detection of visual signals deteriorates markedly, the time it takes for a person to make a decision increases and more errors are made, and reading rates decrease. Other studies show that fatigue results in personnel ignoring some signals because they develop their own subjective standards as to what is important, and as they become more fatigued they ignore more signals.

Inspections of personnel performance and training since the accident at Three Mile Island, have shown that in certain situations facility personnel are either required or allowed to remain on duty for extended periods of time. Also, complaints have been received from some licensed nuclear power plant operators concerning the number of continuous hours they have been on duty.

Licensee management is responsible for providing a sufficient number of trained personnel who are in the proper physical condition to operate and maintain the plant. Licensee management should review their administrative procedures covering the working hours of nuclear power plant staff. These procedures should establish a sound policy covering working hours for plant staff who perform safety related functions (e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, I&C technicians, key maintenance personnel, etc.)

Subcommittee ANS-3 is currently developing criteria to address the subject of operator work hours. These guidelines will become a part of ANSI N18.7. The NRC is also considering issuing requirements for administrative procedures that would control staff overtime. Until either the ANSI Standard is issued and endorsed by NRC (via a Regulatory Guide) or separate requirements are issued by NRC, it is recommended that the following guidance be used. The guidance should be applied to all personnel performing a safety related function:

1. Scheduled work should be limited to the following maximum work hours:

    a. An individual should not be permitted to work more than 12 hours straight.

b. There should be at least a 12-hour break between all work periods.

c. An individual should not work more than 72-hours in any 7-day period.

d. An individual should not work more than 14 consecutive days without having 2 consecutive days off.

2. In the event that special circumstances arise that require deviation from the above, such deviations should be authorized by the Station Manager with appropriate documentation of the cause. Plants should be staffed and schedules developed to operate such that exceptions are not required.

3. If an operator is required to work in excess of 12 continuous hours, his duties should be carefully selected. It is preferable that he not be assigned any task that affects core reactivity or could possibly endanger the safe operation of the plant.

No written response to this Circular is required. If you desire additional information regarding this matter, contact the Director of the appropriate NRC Regional Office.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

July 31, 1980

TO ALL LICENSEES OF OPERATING PLANTS AND APPLICANTS FOR OPERATING LICENSES AND HOLDERS OF CONSTRUCTION PERMITS

SUBJECT: INTERIM CRITERIA FOR SHIFT STAFFING

This is to provide you with the shift manning requirements as indicated in item (1) of our letter of May 7, 1980. Pending completion of the long-term development of criteria for shift staffing and administrative controls, the NRC staff has developed interim criteria for licensees of operating plants and applicants for operating licenses. Except for senior reactor operators, these interim criteria for shift staffing shall remain as described in the Standard Review Plan, Section 13.1.2, NUREG 75/087. Special requirements regarding the utilization and qualifications of an on-shift technical advisor to the shift supervisor were provided in our letter of October 30, 1979.

We have changed the previous requirements for senior reactor operators and now require that there be one licensed senior reactor operator in the control room at all times, other than during cold shutdown conditions. This will therefore require that there be a minimum of two senior reactor operators at each site at all times, other than during cold shutdown conditions, to assure the availability of one senior reactor operator in the control room without affecting the freedom of the shift supervisor to move about the site as needed. The criteria for reactor and auxiliary operators are stated below and the required staffing levels for selected station configurations and various plant operating modes are summarized in the enclosed table.

At any time a licensed nuclear unit is being operated in Modes 1-4 for a PWR (Power Operation, Startup, Hot Standby, or Hot Shutdown respectively) or in Modes 1-3 for a BWR (Power Operation, Startup, or Hot Shutdown respectively), the minimum shift crew shall include two licensed senior reactor operators (SRO), one of whom shall be designated as the shift supervisor, two licensed reactor operators (RO) and two unlicensed auxiliary operators (AO). For a multi-unit station, depending upon the station configuration, shift staffing may be adjusted to allow credit for licensed senior reactor operators (SRO) and licensed reactor operators (RO) to serve as relief operators on more than one unit; however, these individuals must be properly licensed on each such unit. At all other times, for a unit loaded with fuel, the minimum shift crew shall include one shift supervisor who shall be a licensed senior reactor operator (SRO), one licensed reactor operator (RO) and one unlicensed auxiliary operator.

Adjunct requirements to the shift staffing criteria stated above are as follows:

a. A shift supervisor with a senior reactor operator's license, who is also a member of the station supervisory staff, shall be onsite at all times when at least one unit is loaded with fuel.

b. A licensed senior reactor operator (SRO) shall, at all times, be in the control room from which a reactor is being operated. The shift supervisor may from time-to-time act as relief operator for the licensed senior reactor operator assigned to the control room.

c. For any station with more than one reactor containing fuel, the number of licensed senior reactor operators onsite shall, at all times, be at least one more than the number of control rooms from which the reactors are being operated.

d. In addition to the licensed senior reactor operators specified in a., b., and c. above, for each reactor containing fuel, a licensed reactor operator (RO) shall be in the control room at all times.

e. In addition to the operators specified in a., b., c., and d. above, for each control room from which a reactor is being operated, an additional licensed reactor operator (RO) shall be onsite at all times and available to serve as relief operator for that control room. As noted above, this individual may serve as relief operator for each unit being operated from that control room, provided he holds a current license for each unit.

f. Auxiliary (non-licensed) operators shall be properly qualified to support the unit to which assigned.

g. In addition to the staffing requirements stated above, shift crew assignments during periods of core alterations shall include a licensed senior reactor operator (SRO) to directly supervise the core alterations. This licensed senior reactor operator may have fuel handling duties but shall not have other concurrent operational duties.

These criteria do not relieve licensees of any special requirements for additional operators which may have been imposed for individual units.

General application of revised shift staffing criteria will be the subject of a rulemaking proceeding. However, these interim criteria will be effective for plants receiving operating licenses during the interim period (including TMI-1). Licensees of plants already holding operating licenses shall examine their current staffing practices and capabilities in light of these interim criteria and advise this office within 90 days of receipt of this letter of the date by which their shift staffing could be in compliance with these criteria. Licensees of operating plants shall take steps to meet the revised criteria as soon as practical, but no later than July 1, 1982. In your response to this letter, you are requested to discuss your plans, schedules and commitments to meet these staffing criteria. Holders of construction permits who have not as yet applied for an operating license should factor these criteria into their recruitment and crew training plans.

In addition, licensees of operating plants and applicants for operating licenses shall include in their administrative procedures (required by license conditions) provisions governing required shift staffing and movement of key individuals about the plant. These provisions are required to assure that qualified plant personnel to man the operational shifts are readily available in the event of an abnormal or emergency situation.

The administrative procedures shall also set forth a policy concerning overtime work for the senior reactor operators, reactor operators, and shift technical advisor required by these interim criteria. These procedures shall stipulate that overtime shall not be routinely scheduled to compensate for an inadequate number of personnel to meet the shift crew staffing requirements. In the event that overtime must be used, due to unanticipated or unavoidable circumstances, the following overtime restrictions shall be followed:

~~(1) An individual shall not be permitted to work more than 12 hours straight (not including shift turnover time).~~

~~(2) An individual shall not be permitted to work more than 24 hours in any 48 hour period.~~

SUPERSEDED

~~(3) An individual shall not work more than 72 hours in any 7 day period.~~

~~(4) An individual shall not work more than 14 consecutive days without having two consecutive days off.~~

However, recognizing that circumstances may arise requiring deviation from the above restrictions, such deviation may be authorized by the plant manager or higher levels of management in accordance with published procedures and with appropriate documentation of the cause.

The limitations on overtime follow the guidance provided in IE Circular 80-02, except for the requirement noted above on the restriction on use of overtime in circumstances that are unavoidable.

Operating license applicants shall complete these administrative procedures before fuel loading. Development and implementation of the administrative procedures at operating plants will be reviewed by the Office of Inspection and Enforcement beginning 90 days after the date of this letter.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing

Enclosures:
As stated

cc: OR Licensees, and OL Applicants
CP Holders Service Lists

INTERIM REQUIRED SHIFT STAFFING

SUPERSEDED

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

SS - Shift Supervisor
SRO - Licensed Senior Reactor Operator
RO - Licensed Reactor Operator
AO - Auxiliary Operator

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

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

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

I.A.2.1 IMMEDIATE UPGRADING OF REACTOR OPERATOR AND SENIOR REACTOR OPERATOR TRAINING AND QUALIFICATIONS

Position

Effective December 1, 1980, an applicant for a senior reactor operator (SRO) license will be required to have been a licensed operator for 1 year.

Changes to Previous Requirements

Changes to the previous requirements will permit various paths to provide experience equivalent to 1 year's experience as a licensed operator.

Clarification

Applicants for SRO either come through the operations chain (C operator to B operator to A operator, etc.) or are degree-holding staff engineers who obtain licenses for backup purposes.

In the past, many individuals who came through the operator ranks were administered SRO examinations without first being an operator. This was clearly a poor practice and the letter of March 28, 1980 requires reactor operator experience for SRO applicants.

However, NRC does not wish to discourage staff engineers from becoming licensed SROs. This effort is encouraged because it forces engineers to broaden their knowledge about the plant and its operation.

In addition, in order to attract degree-holding engineers to consider the shift supervisor's job as part of their career development, NRC should provide an alternate path to holding an operator's license for 1 year.

The track followed by a high-school graduate (a nondegreed individual) to become an SRO would be 4 years as a control room operator, at least one of which would be as a licensed operator, and participation in an SRO training program that includes 3 months on shift as an extra person.

The track followed by a degree-holding engineer would be, at a minimum, 2 years of responsible nuclear power plant experience as a staff engineer, participation in an SRO training program equivalent to a cold applicant training program, and 3 months on shift as an extra person in training for an SRO position.

Holding these positions assures that individuals who will direct the licensed activities of licensed operators have had the necessary combination of education, training, and actual operating experience prior to assuming a supervisory role at that facility.

The staff realizes that the necessary knowledge and experience can be gained in a variety of ways. Consequently, credit for equivalent experience should be given to applicants for SRO licenses.

Applicants for SRO licenses at a facility may obtain their 1-year operating experience in a licensed capacity (operator or senior operator) at another nuclear power plant. In addition, actual operating experience in a position that is equivalent to a licensed operator or senior operator at military propulsion reactors will be acceptable on a one-for-one basis. Individual applicants must document this experience in their individual applications in sufficient detail so that the staff can make a finding regarding equivalency.

Applicants for SRO licenses who possess a degree in engineering or applicable sciences are deemed to meet the above requirement, provided they meet the requirements set forth in sections A.1.a and A.2 in enclosure 1 in the letter from H. R. Denton to all power reactor applicants and licensees, dated March 28, 1980, and have participated in a training program equivalent to that of a cold senior operator applicant.

NRC has not imposed the 1-year experience requirement on cold applicants for SRO licenses. Cold applicants are to work on a facility not yet in operation; their training programs are designed to supply the equivalent of the experience not available to them.

Applicability

This requirement applies to all operating reactors and applicants for operating licenses (after initial criticality).

Implementation

This requirement applies to applicants for senior reactor operator licenses received after December 1, 1980.

Type of Review

A postimplementation review will be performed.

Documentation Required

No documentation is required from the facility. Information will be contained in individual applications.

Technical Specification Changes Required

Changes to technical specifications will not be required.

Reference

Letter from H. R. Denton, NRC, to All Power Reactor Applicants and Licensees, dated March 28, 1980.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 2 8 1980

ALL POWER REACTOR APPLICANTS AND LICENSEES

Gentlemen:

SUBJECT: QUALIFICATIONS OF REACTOR OPERATORS

In a letter dated September 13, 1979, we informed you of NRR requirements established as of that date based on our review of the TMI-2 accident. Enclosure 9 to the letter outlined the staff recommendations concerning improvements in the area of operator training for your information. Since that time, the Commission has acted on the staff recommendations.

It is the purpose of this letter to set forth the revised criteria to be used by the staff in evaluating reactor operator training and licensing that can be implemented under the current regulations and to establish an effective date for their implementation. Other criteria that will be established require additional staff work are also addressed. However, implementation dates cannot be provided at this time. Commission review in the area of operator training and qualification is continuing and can be expected to result in additional criteria. Finally, requirements will be established through rule making proceedings.

Enclosure 1 details the revised criteria and the effective date for their implementation. Your attention is specifically directed to Sections A, B and C of Enclosure 1 since these call out new criteria that will be implemented in the near future; therefore, your plans regarding training and licensing activities should be promptly revised to conform to these criteria.

Enclosures 2 and 3 provide guidance for establishing training programs in heat transfer, fluid flow and thermodynamics; and mitigating core damage. Enclosure 4 details control manipulations for requalification programs.

Based on our understanding of the industry's reasons for establishing the Institute of Nuclear Power Operations and our review of the latest revisions to applicable ANSI standards, we believe you share our desire to significantly upgrade the requirements for operations personnel.

Therefore, we urge you to effect the implementation of the criteria as soon as practicable rather than wait for the stated implementation date. Also, we urge you to start planning for the long range requirements so that they may be rapidly implemented upon completion of the rule making procedure.

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures
1. Requirements for Reactor Operator Training and Licensing
2. Training in Heat Transfer, Fluid Flow and Thermodynamics
3. Training Criteria For Mitigating Core Damage
4. Control Manipulations

ENCLOSURE 1

CRITERIA FOR REACTOR OPERATOR

TRAINING AND LICENSING

A. Eligibility Requirements to be Administered an Examination.

1. Experience*

a. Applicants for senior operator licenses shall have 4 years of responsible power plant experience. Responsible power plant experience should be that obtained as a control room operator (fossil or nuclear) or as a power plant staff engineer involved in the day-to-day activities of the facility, commencing with the final year of construction. A maximum of 2 years power plant experience may be fulfilled by academic or related technical training, on a one-for-one time basis. Two years shall be nuclear power plant experience. At least 6 months of the nuclear power plant experience shall be at the plant for which he seeks a license.

Effective date: Applications received on or after May 1, 1980.

---

*Precritical applicants will be required to meet unique qualifications designed to accommodate the fact that their facility has not yet been in operation.

b. Applicants for senior operator licenses shall have held an operator's license for 1 year.

Effective date: Applications received after December 1, 1980.

2. Training

a. Senior operator*: Applicants shall have 3 months of shift training as an extra man on shift.

b. Control room operator*: Applicants shall have 3 months training on shift as an extra person in the control room.

Effective date: Applications received after August 1, 1980.

c. Training programs shall be modified, as necessary, to provide:

1) Training in heat transfer, fluid flow and thermodynamics.

2) Training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged.

3) Increased emphasis on reactor and plant transients.

---

*Precritical applicants will be required to meet unique qualifications designed to accommodate the fact that their facility has not yet been in operation.

Effective date: Present programs have been modified in response to Bulletins and Orders. Revised programs should be submitted for OLB review by August 1, 1980.

d. Training center and facility instructors who teach systems, integrated responses, transient and simulator courses shall demonstrate their competence to NRC by successful completion of a senior operator examination.

Effective date: Applications should be submitted no later than August 1, 1980 for individuals who do not already hold a senior operator license.

e. Instructors shall be enrolled in appropriate requalification programs to assure they are cognizant of current operating history, problems, and changes to procedures and administrative limitations.

Effective date: Programs should be initiated May 1, 1980. Programs should be submitted to OLB for review by August 1, 1980.

3. <u>Facility Certifications</u>

Certifications completed pursuant to Sections 55.10(a)(6) and 55.33a(4) and (5) of 10 CFR Part 55 shall be signed by the highest level of corporate management for plant operation (for example, Vice President for Operations).

Effective date: Applications received on or after May 1, 1980.

B. NRC Examinations

1. Increased Scope of Examinations

a. A new category shall be added to the operator written examination entitled, "Principles of Heat Transfer and Fluid Mechanics."

b. A new category shall be added to the senior operator written examination entitled, "Theory of Fluids and Thermodynamics."

c. Time limits shall be imposed for completion of the written examinations:

1. Operator: 9 hours.
2. Senior Operator: 7 hours.

d. The passing grade for the written examination shall be 80% overall and 70% in each category.

e. All applicants for senior operator licenses shall be required to be administered an operating test as well as the written examination.

f. Applicants will grant permission to NRC to inform their facility management regarding the results of the examinations for purposes of enrollment in requalification programs.

Effective date: Examinations administered on or after May 1, 1980 for items a. through e. Applications received on or after May 1, 1980 for Item f.

C. Requalification Programs

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

   Effective date: May 1, 1980.

2. The criteria for requiring a licensed individual to participate in accelerated requalification shall be modified to be consistent with the new passing grade for issuance of a license; 80% overall and 70% each category.

   Effective date: Concurrent with the next facility administered annual requalification examination after the issue date of this letter.

3. Programs should be modified to require the control manipulations listed in Enclosure 4. Normal control manipulations, such as plant or reactor startups, must be performed. Control manipulations during abnormal or emergency operations must be walked through with, and evaluated by, a member of the training staff at a minimum. An appropriate simulator may be used to satisfy the requirements for control manipulations.

   Effective date: Programs modified by August 1, 1980. Renewal applications received after November 1, 1980 must reflect compliance with the program.

D. Long Range Criteria and/or Requirements

The following require additional staff work and/or rulemaking prior to their implementation.

1. Qualifications

   a. Shift supervisors shall have an engineering degree or equivalent qualifications.

   b. Senior operators shall have successfully completed a course in appropriate engineering and scientific subject equal to 60 credit hours of college level subjects.

2. Training

   a. All applicants shall attend simulator training programs. Required control manipulations and exercises to be performed shall be the same for "cold" and "hot" applicants.

   b. Eligibility requirements shall be developed for instructors, in addition to that listed in A.2 above.

3. NRC Examinations

   a. NRC shall administer the certification examinations that are presently administered at the conclusion of the off-site portion of the cold training programs.

   b. All applicants shall be required to be administered a simulator examination in addition to the written examinations and plant oral tests.

   c. NRC shall administer the requalification program annual examination.

4. Requalification Programs

All licensees shall participate in simulator programs as part of the requalification programs. Control manipulations shall be performed pursuant to Enclosure 4.

ENCLOSURE 2

TRAINING IN HEAT TRANSFER, FLUID FLOW AND THERMODYNAMICS

1. Basic Properties of Fluids and Matter. This section should cover a basic introduction to matter and its properties. This section should include such concepts as temperature measurements and effects, density and its effects, specific weight, buoyancy, viscosity and other properties of fluids. A working knowledge of steam tables should also be included. Energy movement should be discussed including such fundamentals as heat exchange, specific heat, latent heat of vaporization and sensible heat.

2. Fluid Statics. This section should cover the pressure, temperature and volume effects on fluids. Example of these parametric changes should be illustrated by the instructor and related calculations should be performed by the students and discussed in the training sessions. Causes and effects of pressure and temperature changes in the various components and systems should be discussed as applicable to the facility with particular emphasis on safety significant features. The characteristics of force and pressure, pressure in liquids at rest, principles of hydraulics, saturation pressure and temperature and subcooling should also be included.

3. Fluid Dynamics. This section should cover the flow of fluids and such concepts as Bernoulli's principle, energy in moving fluids, flow measure theory and devices and pressure losses due to friction and orificing. Other concepts and terms to be discussed in this section are NPSH, carry over, carry under, kinetic energy, head-loss relationships and two phase

flow fundamentals. Practical applications relating to the reactor coolant system and steam generators should also be included.

4. Heat Transfer by Conduction, Convection and Radiation. This section should cover the fundamentals of heat transfer by conductions. This section should include discussions on such concepts and terms as specific heat, heat flux and atomic action. Heat transfer characteristics of fuel rods and heat exchangers should be included in this section.

   This section should cover the fundamentals of heat transfer by convection. Natural and forced circulation should be discussed as applicable to the various systems at the facility. The convection current patterns created by expanding fluids in a confined area should be included in this section. Heat transport and fluid flow reductions or stoppage should be discussed due to steam and/or noncondensible gas formation during normal and accident conditions.

   This section should cover the fundamentals of heat transfer by thermal radiation in the form of radiant energy. The eletromagnetic energy emitted by a body as a result of its temperature should be discussed and illustrated by the use of equations and sample calculations. Comparisons should be made of a black body absorber and a white body emitter.

5. Change of Phase - Boiling. This section should include descriptions of the state of matter, their inherent characteristics and thermodynamic properties such as enthalpy and entropy. Calculations should be performed involving steam quality and void fraction properties. The types

of boiling should be discussed as applicable to the facility during normal evolutions and accident conditions.

6. Burnout and Flow Instability. This section should cover descriptions and mechanisms for calculating such terms as critical flux, critical power, DNB ratio and hot channel factors. This section should also include instructions for preventing and monitoring for clad or fuel damage and flow instabilities. Sample calculations should be illustrated by the instructor and calculations should be performed by the students and discussed in the training sessions. Methods and procedures for using the plant computer to determine quantitative values of various factors during plant operation and plant heat balance determinations should also be covered in this section.

7. Reactor Heat Transfer Limits. This section should include a discussion of heat transfer limits by examining fuel rod and reactor design and limitations. The basis for the limits should be covered in this section along with recommended methods to ensure that limits are not approached or exceeded. This section should cover discussions of peaking factors, radial and axial power distributions and changes of these factors due to the influence of other variables such as moderator temperature, xenon and control rod position.

ENCLOSURE 3

TRAINING CRITERIA FOR MITIGATING CORE DAMAGE

A program is to be developed to ensure that all operating personnel are training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged. The training program should include the following topics.

A. Incore Instrumentation

1. Use of fixed or movable incore detectors to determine extent of core damage and geometry changes.
2. Use of thermocouples in determining peak temperatures; methods for extended range readings; methods for direct readings at terminal junctions.
3. Methods for calling up (printing) incore data from the plant computer.

B. Excore Nuclear Instrumentation (NIS)

1. Use of NIS for determination of void formation; void location basis for NIS response as a function of core temperatures and density changes.

C. Vital Instrumentation

1. Instrumentation response in an accident environment; failure sequence (time to failure, method of failure); indication reliability (actual vs indicated level).

2. Alternative methods for measuring flows, pressures, levels, and temperatures.
   a. Determination of pressurizer level if all level transmitters fail.
   b. Determination of letdown flow with a clogged filter (low flow).
   c. Determination of other Reactor Coolant System parameters if the primary method of measurement has failed.

D. Primary Chemistry

1. Expected chemistry results with severe core damage; consequences of transferring small quantities of liquid outside containment; importance of using leak tight systems.
2. Expected isotopic breakdown for core damage; for clad damage.
3. Corrosion effects of extended immersion in primary water; time to failure.

E. Radiation Monitoring

1. Response of Process and Area Monitors to severe damages; behavior of detectors when saturated; method for detecting radiation readings by direct measurement at detector output (overranged detector); expected accuracy of detectors at different locations; use of detectors to determine extent of core damage.
2. Methods of determining dose rate inside containment from measurements taken outside containment.

F. Gas Generation

1. Methods of H_2 generation during an accident; other sources of gas (Xe, Ke); techniques for venting or disposal of non-condensibles.

2. H_2 flammability and explosive limit; sources of O_2 in containment or Reactor Coolant System.

Suggested References:

Collier, J. G. Convection Boiling and Condensation. New York: McGraw-Hill, 1972.

Eckert, E. R. G. and Drake, R. M., Jr. Analysis of Heat and Mass Transfer. New York: McGraw-Hill, 1973.

El-Wakil, M. M. Nuclear Heat Transport. Scranton, PA: International, 1971.

Gebhart, B. Heat Transfer. 2nd ed. New York: McGraw-Hill, 1971.

Mooney, D. Mechanical Engineering Thermodynamics. Prentice Hall, 1953.

ENCLOSURE 4

CONTROL MANIPULATIONS

The following control manipulations and plant evolutions where applicable to the plant design are acceptable for meeting the reactivity control manipulations required by Appendix A, Paragraph 3.a. of 10 CFR Part 55. The starred items shall be performed on an annual basis; all other items shall be performed on a two-year cycle. However, the requalification programs shall contain a commitment that each individual shall perform or participate in a combination of reactivity control manipulations based on the availability of plant equipment and systems. Those control manipulations which are not performed at the plant may be performed on a simulator. The use of the Technical Specifications should be maximized during the simulator control manipulations. Personnel with senior licenses are credited with these activities if they direct or evaluate control manipulations as they are performed.

PWR/BWR/HTGR

*(1) Plant or reactor startups to include a range that reactivity feedback from nuclear heat addition is noticeable and heatup rate is established.

(2) Plant shutdown.

*(3) Manual control of steam generators and/or feedwater during startup and shutdown.

(4) Boration and or dilution during power operation.

*(5) Any significant (> 10%) power changes in manual rod control or recirculation flow.

(6) Any reactor power change of 10% or greater where load change is performed with load limit control or where flux, temperature, or speed control is on manual (for HTGR).

*(7) Loss of coolant including:

1. significant PWR steam generator leaks
2. inside and outside primary containment
3. large and small, including leak-rate determination
4. saturated Reactor Coolant response (PWR).

(8) Loss of instrument air (if simulated plant specific).

(9) Loss of electrical power (and/or degraded power sources).

*(10) Loss of core coolant flow/natural circulation.

(11) Loss of condenser vacuum.

(12) Loss of service water if required for safety.

(13) Loss of shutdown cooling.

(14) Loss of component cooling system or cooling to an individual component.

(15) Loss of normal feedwater or normal feedwater system failure.

*(16) Loss of all feedwater (normal and emergency).

(17) Loss of protective system channel.

(18) Mispositioned control rod or rods (or rod drops).

(19) Inability to drive control rods.

(20) Conditions requiring use of emergency boration or standby liquid control system.

(21) Fuel cladding failure or high activity in reactor coolant or offgas.

(22) Turbine or generator trip.

(23) Malfunction of automatic control system(s) which affect reactivity.

(24) Malfunction of reactor coolant pressure/volume control system.

(25) Reactor trip.

(26) Main steam line break (inside or outside containment).

(27) Nuclear instrumentation failure(s).

I.A.2.3 ADMINISTRATION OF TRAINING PROGRAMS

Position

Pending accreditation of training institutions, licensees and applicants for operating licenses will assure that training center and facility instructors who teach systems, integrated responses, transient, and simulator courses demonstrate senior reactor operator (SRO) qualifications and be enrolled in appropriate requalification programs.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements included in the letter of March 28, 1980 from H. R. Denton to all power reactor applicants and licensees.

Clarification

The above position is a short-term position. In the future, accreditation of training institutions will include review of the procedure for certification of instructors. The certification of instructors may, or may not, include successful completion of an SRO examination.

The purpose of the examination is to provide NRC with reasonable assurance during the interim period, that instructors are technically competent.

The requirement is directed to permanent members of training staff who teach the subjects listed above, including members of other organizations who routinely conduct training at the facility. There is no intention to require guest lecturers who are experts in particular subjects (reactor theory, instrumentation, thermodynamics, health physics, chemistry, etc.) to successfully complete an SRO examination. Nor is it intended to require a system expert, such as the instrument and control supervisor teaching the control rod drive system, to sit for an SRO examination.

Applicability

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

Implementation

The requirements for operating reactors have been completed. Applications for SRO examinations should be submitted. All applicants for operating license should submit documentation 2 months prior to the expected issuance of an operating license.

Type of Review

A postimplementation review will be performed.

Documentation Required

No documentation is required.

Technical Specification Changes Required

Changes to technical specifications will not be required.

Reference

Letter from H. R. Denton, NRC, to All Power Reactor Applicants and Licensees, dated March 28, 1980.

I.A.3.1 REVISE SCOPE AND CRITERIA FOR LICENSING EXAMINATIONS--SIMULATOR EXAMS (ITEM 3)

Position

Simulator examinations will be included as part of the licensing examinations.

Changes to Previous Requirements and Guidance

The administration of simulator examinations will be deferred for applicants whose facilities do not have simulators on site as of October 1, 1980. These deferred simulator examinations will be initiated by October 1, 1981.

Clarification

The clarification does not alter the staff's position regarding simulator examinations.

The clarification does provide additional preparation time for utility companies and NRC to meet examination requirements as stated. A study is under way to consider how similar a nonidentical simulator should be for a valid examination. In addition, present simulators are fully booked months in advance.

Application of this requirement was stated on June 1, 1980 to applicants where a simulator is located at the facility. Starting October 1, 1981, simulator examinations will be conducted for applicants of facilities that do not have simulators at the site.

NRC simulator examinations normally require 2 to 3 hours. Normally, two applicants are examined during this time period by two examiners.

Utility companies should make the necessary arrangements with an appropriate simulator training center to provide time for these examinations. Preferably these examinations should be scheduled consecutively with the balance of the examination. However, they may be scheduled no sooner than 2 weeks prior to and no later than 2 weeks after the balance of the examination.

Applicability

This requirement applies to all applicants for operator and senior operator licenses at power reactors.

Implementation

The schedule for operating reactors is October 1, 1981 for licensees without simulators and June 1980 for licensees with simulators.

The schedule for applicants for operating license without simulators is October 1, 1981 or prior to fuel load, whichever is later, including cold examinations.

The schedule for applicants for operating license with simulators is prior to full load including cold examination.

Type of Review

No review will be performed. Arrangements will be made during the normal scheduling of examinations.

Documentation Required

No documentation is required. Arrangements will be made during the normal scheduling of examinations.

Technical Specification Changes Required

Changes to technical specifications will not be required.

Reference

Letter from H. R. Denton, NRC, to All Power Reactor Applicants and Licensees, dated March 28, 1980.

I.B.1.2 INDEPENDENT SAFETY ENGINEERING GROUP

Position

Each applicant for an operating license shall establish an onsite independent safety engineering group (ISEG) to perform independent reviews of plant operations.

The principal function of the ISEG is to examine plant operating characteristics, NRC issuances, Licensing Information Service advisories, and other appropriate sources of plant design and operating experience information that may indicate areas for improving plant safety. The ISEG is to perform independent review and audits of plant activities including maintenance, modifications, operational problems, and operational analysis, and aid in the establishment of programmatic requirements for plant activities. Where useful improvements can be achieved, it is expected that this group will develop and present detailed recommendations to corporate management for such things as revised procedures or equipment modifications.

Another function of the ISEG is to maintain surveillance of plant operations and maintenance activities to provide independent verification that these activities are performed correctly and that human errors are reduced as far as practicable. ISEG will then be in a position to advise utility management on the overall quality and safety of operations. ISEG need not perform detailed audits of plant operations and shall not be responsible for sign-off functions such that it becomes involved in the operating organization.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements, however further guidance is provided in the "Clarification" section that follows.

Clarification

The new ISEG shall not replace the plant operations review committee (PORC) and the utility's independent review and audit group as specified by current staff guidelines (Standard Review Plan, Regulatory Guide 1.33, Standard Technical Specifications). Rather, it is an additional independent group of a minimum of five dedicated, full-time engineers, located onsite, but reporting offsite to a corporate official who holds a high-level, technically oriented position that is not in the management chain for power production. The ISEG will increase the available technical expertise located onsite and will provide continuing, systematic, and independent assessment of plant activities. Integrating the shift technical advisors (STAs) into the ISEG in some way would be desirable in that it could enhance the group's contact with and knowledge of day-to-day plant operations and provide additional expertise. However, the STA on shift is necessarily a member of the operating staff and cannot be independent of it.

It is expected that the ISEG may interface with the quality assurance (QA) organization, but preferably should not be an integral part of the QA organization.

The functions of the ISEG require daily contact with the operating personnel and continued access to plant facilities and records. The ISEG review functions can, therefore, best be carried out by a group physically located onsite. However, for utilities with multiple sites, it may be possible to perform portions of the independent safety assessment function in a centralized location for all the utility's plants. In such cases, an onsite group still is required, but it may be slightly smaller than would be the case if it were performing the entire independent safety assessment function. Such cases will be reviewed on a case-by-case basis.

At this time, the requirement for establishing an ISEG is being applied only to applicants for operating licenses in accordance with Action Plan item I.B.1.2. The staff intends to review this activity in about a year to determine its effectiveness and to see whether changes are required. Applicability to operating plants will be considered in implementing long-term improvements in organization and management for operating plants (Action Plan item I.B.1.1).

Applicability

This requirement applies to all applicants for operating license.

Implementation

This requirement shall be implemented prior to issuance of an operating license (or fuel-loading license).

Type of Review

A preimplementation review will be performed.

Documentation Required

Each applicant for an operating license shall document in its application or amendments thereto, its plan for establishing and staffing the ISEG, including the qualifications of and the training to be given the ISEG staff.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0660

NUREG-0694, Item I.B.1.1 and Item I.B.1.2

I.C.1 GUIDANCE FOR THE EVALUATION AND DEVELOPMENT OF PROCEDURES FOR TRANSIENTS AND ACCIDENTS

Position

In letters of September 13 and 27, October 10 and 30, and November 9, 1979, the Office of Nuclear Reactor Regulation required licensees of operating plants, applicants for operating licenses and licensees of plants under construction to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, including procedures for operating with natural circulation conditions, and to conduct operator retraining (see also item I.A.2.1). Emergency procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Analyses of transients and accidents were to be completed in early 1980 and implementation of procedures and retraining were to be completed 3 months after emergency procedure guidelines were established; however, some difficulty in completing these requirements has been experienced. Clarification of the scope of the task and appropriate schedule revisions are being developed. In the course of review of these matters on Babcock and Wilcox (B&W)-designed plants, the staff will follow up on the bulletin and orders matters relating to analysis methods and results, as listed in NUREG-0660, Appendix C (see Table C.1, items 3, 4, 16, 18, 24, 25, 26, 27; Table C.2, items 4, 12, 17, 18, 19, 20; and Table C.3, items 6, 35, 37, 38, 39, 41, 47, 55, 57).

Changes to Previous Requirements and Guidance

A. Modification to Clarification

(1) Addresses owners' group and vendor submittals.

(2) References to task action plan items I.C.8 and I.C.9.

(3) Scope of procedures review is explained.

(4) Establishes configuration control of guidelines for emergency procedures.

B. Modification to Implementation

(1) Deleted reference to NUREG-0578, Recommendation 2.1.9 for item I.C.1(a)2, inadequate core cooling.

Clarification

The letters of September 13 and 27, October 10 and 30, and November 9, 1979, required that procedures and operator training be developed for transients and accidents. The initiating events to be considered should include the events presented in the final safety analysis report (FSAR) loss of instrumentation buses, and natural phenomena such as earthquakes, floods, and tornadoes. The purpose of this paper is to clarify the requirements and add additional requirements for the reanalysis of transients and accidents and inadequate core cooling.

Based on staff reviews to date, there appear to be some recurring deficiencies in the guidelines being developed. Specifically, the staff has found a lack of justification for the approach used (i.e., symptom-, event-, or function-oriented) in developing diagnostic guidance for the operator and in procedural development. It has also been found that although the guidelines take implicit credit for operation of many systems or components, they do not address the availability of these systems under expected plant conditions nor do they address corrective or alternative actions that should be performed to mitigate the event should these systems or components fail.

The analyses conducted to date for guideline and procedure development contain insufficient information to assess the extent to which multiple failures are considered. NUREG-0578 concluded that the single-failure criterion was not considered appropriate for guideline development and called for the consideration of multiple failures and operator errors. Therefore, the analyses that support guideline and procedure development should consider the occurrences of multiple and consequential failures. In general, the sequence of events for the transients and accidents and inadequate core cooling analyzed should postulate multiple failures such that, if the failures were unmitigated, conditions of inadequate core cooling would result.

Examples of multiple failure events include:

(1) Multiple tube ruptures in a single steam generator and tube rupture in more than one steam generator;

(2) Failure of main and auxiliary feedwater;

(3) Failure of high-pressure reactor coolant makeup system;

(4) An anticipated transient without scram (ATWS) event following a loss of offsite power, stuck-open relief valve or safety/relief valve, or loss of main feedwater; and

(5) Operator errors of omission or commission.

The analyses should be carried out far enough into the event to assure that all relevant thermal/hydraulic/neutronic phenomena are identified (e.g., upper head voiding due to rapid cooldown, steam generator stratification). Failures and operator errors during the long-term cooldown period should also be addressed.

The analyses should support development of guidelines that define a logical transition from the emergency procedures into the inadequate core cooling procedure including the use of instrumentation to identify inadequate core cooling conditions. Rationale for this transition should be discussed. Additional information that should be submitted includes:

(1) A detailed description of the methodology used to develop the guidelines;

(2) Associated control function diagrams, sequence-of-event diagrams, or others, if used;

(3) The bases for multiple and consequential failure considerations;

(4) Supporting analysis, including a description of any computer codes used; and

(5) A description of the applicability of any generic results to plant-specific applications.

Owners' group or vendor submittals may be referenced as appropriate to support this reanalysis. If owners' group or vendor submittals have already been forwarded to the staff for review, a brief description of the submittals and justification of their adequacy to support guideline development is all that is required.

Pending staff approval of the revised analysis and guidelines, the staff will continue the pilot monitoring of emergency procedures described in Task Action Plan item I.C.8 (NUREG-0660). For PWRs, this will involve review of the loss of coolant, steam-generator-tube rupture, loss of main feedwater, and inadequate core cooling procedures. The adequacy of each PWR vendor's guidelines will be identified to each NTOL during the emergency-procedure review. Since the analysis and guidelines submitted by the General Electric Company (GE) owners' group that comply with the requirements stated above have been reviewed and approved for trial implementation on six plants with applications for operating licenses pending, the interim program for BWRs will consist of trial implementation on these six plants.

Following approval of analysis and guidelines and the pilot monitoring of emergency procedures, the staff will advise all licensees of the adequacy of the guidelines for application to their plants. Consideration will be given to human factors engineering and system operational characteristics, such as information transfer under stress, compatibility with operator training and control-room design, the time required for component and system response, clarity of procedural actions, and control-room-personnel interactions. When this determination has been made by the staff, a long-term plan for emergency procedure review, as described in task action plan item I.C.9, will be made available. At that time, the reviews currently being conducted on NTOLs under item I.C.8 will be discontinued, and the review required for applicants for operating licenses will be as described in the long-term plan. Depending on the information submitted to support development of emergency procedures for each reactor type or vendor, this transition may take place at different times. For example, if the GE guidelines are shown to be effective on the six plants chosen for pilot monitoring, the long-term plan for BWRs may be complete in early 1981. Operating plants and applicants will then have the option of implementing the long-term plan in a manner consistent with their operating schedule, provided they meet the final date required for implementation. This may require a plant that was reviewed for an operating license under item I.C.8 to revise its emergency procedures again prior to the final implementation date for Item I.C.9. The extent to which the long-term program will include review and approval of plant-specific procedures for operating plants has not been established. Our objective, however, is to minimize the amount of plant-specific procedure review and approval required. The staff believes this objective can be acceptably accomplished by concentrating the staff review and approval on generic guidelines. A key element in meeting this objective is the use of staff-approved generic guidelines and guideline revisions by licensees to develop procedures. For this approach to be effective, it is imperative that, once the staff has issued approval of a guideline, subsequent revisions of the guideline should not be implemented by licensees until reviewed and approved by the staff. Any changes in plant-specific procedures based on unapproved guidelines could constitute an unreviewed safety issue under 10 CFR 50.59. Deviations from this approach on a plant-specific basis would be acceptable provided the basis is submitted by the licensee for staff review and approval. In this case, deviations from generic guidelines should not be implemented until staff approval is formally received in writing. Interim implementation of analysis and procedures for small-break loss-of-coolant accident and inadequate core cooling should remain on the schedule contained in NUREG-0578, Recommendation 2.1.9.

Applicability

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

Implementation

Reanalysis of transients and accidents and inadequate core cooling and preparation of guidelines for development of emergency procedures should be completed and submitted to the NRC for review by January 1, 1981. The NRC staff will review the analyses and guidelines and determine their acceptability by July 1, 1981, and will issue guidance to licensees on preparing emergency procedures from the guidelines. Following NRC approval of the guidelines, licensees and applicants for operating licenses issued prior to January 1, 1982, should revise and implement their emergency procedures at the first refueling outage after January 1, 1982. Applicants for operating licenses issued after January 1, 1982 should implement the procedures prior to operation. This schedule supersedes the implementation schedule included in NUREG-0578, Recommendation 2.1.9 for item I.C.1(a)3, Reanalysis of Transients and Accidents. For those licensees and/or owners groups that will have difficulty in attaining the January 1, 1981 due date for submittal of guidelines, a comprehensive program plan, proposed schedule, and a detailed justification for all delays and problems shall be submitted in lieu of the guidelines.

Type of Review

A preimplementation review of guidelines will be performed.

A preimplementation review of procedures will be performed.

Documentation Required

See above, "Implementation."

Technical Specification Changes Required

Changes to technical specifications will not be required.

Reference

NUREG-0578, Recommendation 2.1.9

NUREG-0660, Item I.C.8 and Appendix C, Tables C.1, C.2, C.3

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

Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, dated September 27, 1979.

Letter from D. G. Eisenhut, NRC, to All Power Reactor Licensees, dated October 10, 1979.

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

Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, dated November 9, 1979.

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

Position

In accordance with Task Action Plan I.C.5, Procedures for Feedback of Operating Experience to Plant Staff (NUREG-0660), each applicant for an operating license shall prepare procedures to assure that operating information pertinent to plant safety originating both within and outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and retraining programs. These procedures shall:

(1) Clearly identify organizational responsibilities for review of operating experience, the feedback of pertinent information to operators and other personnel, and the incorporation of such information into training and retraining programs;

(2) Identify the administrative and technical review steps necessary in translating recommendations by the operating experience assessment group into plant actions (e.g., changes to procedures; operating orders);

(3) Identify the recipients of various categories of information from operating experience (i.e., supervisory personnel, shift technical advisors, operators, maintenance personnel, health physics technicians) or otherwise provide means through which such information can be readily related to the job functions of the recipients;

(4) Provide means to assure that affected personnel become aware of and understand information of sufficient importance that should not wait for emphasis through routine training and retraining programs;

(5) Assure that plant personnel do not routinely receive extraneous and unimportant information on operating experience in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency;

(6) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to operators and other personnel until resolution is reached; and,

(7) Provide periodic internal audit to assure that the feedback program functions effectively at all levels.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

Each utility shall carry out an operating experience assessment function that will involve utility personnel having collective competence in all areas important to plant safety. In connection with this assessment function, it is important that procedures exist to assure that important information on operating experience originating both within and outside the organization is continually provided to operators and other personnel and that it is incorporated into plant operating procedures and training and retraining programs.

Those involved in the assessment of operating experience will review information from a variety of sources. These include operating information from the licensee's own plant(s), publications such as IE Bulletins, Circulars, and Notices, and pertinent NRC or industrial assessments of operating experience. In some cases, information may be of sufficient importance that it must be dealt with promptly (through instructions, changes to operating and emergency procedures, issuance of special changes to operating and emergency procedures, issuance of special precautions, etc.) and must be handled in such a manner to assure that operations management personnel would be directly involved in the process. In many other cases, however, important information will become available which should be brought to the attention of operators and other personnel for their general information to assure continued safe plant operation. Since the total volume of information handled by the assessment group may be large, it is important that assurance be provided that high-priority matters are dealt with promptly and that discrimination is used in the feedback of other information so that personnel are not deluged with unimportant and extraneous information to the detriment of their overall proficiency. It is important, also, that technical reviews be conducted to preclude premature dissemination of conflicting or contradictory information.

Applicability

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

Implementation

Procedures governing feedback of operating experience to plant staff shall be completed and the procedures put into effect on or before January 1, 1981 or prior to issuance of an operating license, whichever is later.

Type of Review

A postimplementation review will be performed.

Documentation Required

No documentation is required.

Technical Specificaton Changes Required

Changes to technical specifications will not be required.

References

NUREG-0660, Item I.C.5

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

I.C.6 GUIDANCE ON PROCEDURES FOR VERIFYING CORRECT PERFORMANCE OF OPERATING ACTIVITIES

Position

It is required (from NUREG-0660) that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations and maintenance activities independent of the people performing the activity (see NUREG-0585, Recommendation 5), or both.

Implementation of automatic status monitoring if required will reduce the extent of human verification of operations and maintenance activities but will not eliminate the need for such verification in all instances. The procedures adopted by the licensees may consist of two phases--one before and one after installation of automatic status monitoring equipment, if required, in accordance with item I.D.3.

Changes to Previous Requirements and Guidance

Proposed requirement in NUREG-0660; this requirement is formally issued by this letter.

Clarification

Item I.C.6 of the U.S. Nuclear Regulatory Commission Task Action Plan (NUREG-0660) and Recommendation 5 of NUREG-0585 propose requiring that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided. An acceptable program for verification of operating activities is described below.

The American Nuclear Society has prepared a draft revision to ANSI Standard N18.7-1972 (ANS 3.2) "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." A second proposed revision to Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," which is to be issued for public comment in the near future, will endorse the latest draft revision to ANS 3.2 subject to the following supplemental provisions:

(1) Applicability of the guidance of Section 5.2.6 should be extended to cover surveillance testing in addition to maintenance.

(2) In lieu of any designated senior reactor operator (SRO), the authority to release systems and equipment for maintenance or surveillance testing or return-to-service may be delegated to an on-shift SRO, provided provisions are made to ensure that the shift supervisor is kept fully informed of system status.

(3) Except in cases of significant radiation exposure, a second qualified person should verify correct implementation of equipment control measures such as tagging of equipment.

(4) Equipment control procedures should include assurance that control-room operators are informed of changes in equipment status and the effects of such changes.

(5) For the return-to-service of equipment important to safety, a second qualified operator should verify proper systems alignment unless functional testing can be performed without compromising plant safety, and can prove that all equipment, valves, and switches involved in the activity are correctly aligned.

NOTE: A licensed operator possessing knowledge of the systems involved and the relationship of the systems to plant safety would be a "qualified" person. The staff is investigating the level of qualification necessary for other operators to perform these functions.

For plants that have or will have automatic system status monitoring as discussed in Task Action Plan item I.D.3, NUREG-0660, the extent of human verification of operations and maintenance activities will be reduced. However, the need for such verification will not be eliminated in all instances.

Applicability

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

Implementation

Licensees/applicants must review and revise procedures as necessary to reflect this position by January 1, 1981 or prior to fuel load, whichever is later.

Type of Review

A postimplementation review will be performed.

Documentation Required

No documentation is required.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0585, Recommendation 5

NUREG-0660, Item I.C.6, I.D.3

I.D.1 CONTROL-ROOM DESIGN REVIEWS

Position

In accordance with Task Action Plan I.D.1, Control Room Design Reviews (NUREG-0660), all licensees and applicants for operating licenses will be required to conduct a detailed control-room design review to identify and correct design deficiencies. This detailed control-room design review is expected to take about a year. Therefore, the Office of Nuclear Reactor Regulation (NRR) requires that those applicants for operating licenses who are unable to complete this review prior to issuance of a license make preliminary assessments of their control rooms to identify significant human factors and instrumentation problems and establish a schedule approved by NRC for correcting deficiencies. These applicants will be required to complete the more detailed control room reviews on the same schedule as licensees with operating plants.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

NRR is presently developing human engineering guidelines to assist each licensee and applicant in performing detailed control-room review. A draft of the guidelines has been published for public comment as NUREG/CR-1580, "Human Engineering Guide to Control Room Evaluation." The due date for comments on this draft document was September 29, 1980. NRR will issue the final version of the guidelines as NUREG-0700, by February 1981, after receiving, reviewing, and incorporating substantive public comments from operating reactor licensees, applicants for operating licenses, human factors engineering experts, and other interested parties. NRR will issue evaluation criteria, by July 1981, which will be used to judge the acceptability of the detailed reviews performed and the design modifications implemented.

Applicants for operating licenses who will be unable to complete the detailed control-room design review prior to issuance of a license are required to perform a preliminary control-room design assessment to identify significant human factors problems. Applicants will find it of value to refer to the draft document NUREG/CR-1580, "Human Engineering Guide to Control Room Evaluation," in performing the preliminary assessment. NRR will evaluate the applicants' preliminary assessments including the performance by NRR of onsite review/audit. The NRR onsite review/audit will be on a schedule consistent with licensing needs and will emphasize the following aspects of the control room:

(1) The adequacy of information presented to the operator to reflect plant status for normal operation, anticipated operational occurrences, and accident conditions;

(2) The groupings of displays and the layout of panels;

(3) Improvements in the safety monitoring and human factors enhancement of controls and control displays;

(4) The communications from the control room to points outside the control room, such as the onsite technical support center, remote shutdown panel, offsite telephone lines, and to other areas within the plant for normal and emergency operation.

(5) The use of direct rather than derived signals for the presentation of process and safety information to the operator;

(6) The operability of the plant from the control room with multiple failures of nonsafety-grade and nonseismic systems;

(7) The adequacy of operating procedures and operator training with respect to limitations of instrumentation displays in the control room;

(8) The categorization of alarms, with unique definition of safety alarms.

(9) The physical location of the shift supervisor's office either adjacent to or within the control-room complex.

Prior to the onsite review/audit, NRR will require a copy of the applicant's preliminary assessment and additional information which will be used in formulating the details of the onsite review/audit.

Applicability

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

Implementation

(1) Operating reactors and applicants for OLs:

Complete review, using NRC guidelines (NUREG-0700) issued in 1981, on a schedule that will be determined upon issuance of the guidelines.

(2) Applicants for OLs whose schedules do not permit a full review prior to licensing: Preliminary review complete and approved by NRC prior to issuance of the operating license.

Type of Review

Type of review for operating reactors will be determined upon issuance of the guidance. A preimplementation review will be performed for operating license applicants.

Documentation Required

Operating Reactors--To be determined upon issuance of the guidance.

Applicants for OLs with impacted schedules should report on results of preliminary review prior to licensing.

Technical Specification Changes Required

Changes to technical specifications will not be required unless there are modifications to the control room.

References

NUREG-0660, Item I.D.1

NUREG/CR-1580 (Draft)

NUREG-0700

I.D.2 PLANT SAFETY PARAMETER DISPLAY CONSOLE

Position

In accordance with Task Action Plan 1.D.2, Plant Safety Parameter Display Console (NUREG-0660), each applicant and licensee shall install a safety parameter display system (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status.

Changes to Previous Requirements and Guidance

There are no changes to previous guidance.

Clarification

These requirements for the SPDS are being developed in NUREG-0696, which is scheduled for issuance in November 1980.

Applicability

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

Implementation

Schedules for implementation will be issued in conjunction with issuance of NUREG-0696.

Type of Review

To be determined in conjunction with issuance of NUREG-0696.

Documentation Required

To be determined in conjunction with issuance of NUREG-0696.

Technical Specification Changes Required

To be determined in conjunction with issuance of NUREG-0696.

References

NUREG-0660, Item 1.D.2

NUREG-0696

II.B.1 REACTOR COOLANT SYSTEM VENTS

Position

Each applicant and licensee shall install reactor coolant system (RCS) and reactor vessel head high point vents remotely operated from the control room. Although the purpose of the system is to vent noncondensible gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the events shall conform to the requirements of Appendix A to 10 CFR Part 50, "General Design Criteria." The vent system shall be designed with sufficient redundancy that assures a low probability of inadvertent or irreversible actuation.

Each licensee shall provide the following information concerning the design and operation of the high point vent system:*

(1) Submit a description of the design, location, size, and power supply for the vent system along with results of analyses for loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10 CFR 50.46.

(2) Submit procedures and supporting analysis for operator use of the vents that also include the information available to the operator for initiating or terminating vent usage.

Changes to Previous Requirements and Guidance

(1) The probability of a valve failing to close, once opened, should be minimized.

(2) Establishes environmental qualification (Commission Order, May 23, 1980).

(3) Establishes provisions for testing.

(4) Delete requirements of September 27, 1979 letter from Vassallo to applicants stating that vents shall satisfy single-failure criteria of IEEE-279. Vent systems are not required to have redundant paths. A degree of redundancy should be provided by powering different vents from different emergency buses.

(5) Documentation date changed to July 1, 1981 and implementation date to July 1, 1982.

Clarification does not change NRC concept of requirement, but provides more detail on scope. The dates have been revised to provide time for procurement and installation.

*It was the intent of the October 30, 1979 letter to delete the requirement to meet the criteria of 10 CFR 50.44 and SRP 6.2.5 for beyond-design-basis events. The analysis requirements of Position 2 in the September 13, 1979 letter are therefore unnecessary.

Clarification

A. General

(1) The important safety function enhanced by this venting capability is core cooling. For events beyond the present design basis, this venting capability will substantially increase the plant's ability to deal with large quantities of noncondensible gas which could interfere with core cooling.

(2) Procedures addressing the use of the reactor coolant system vents should define the conditions under which the vents should be used as well as the conditions under which the vents should not be used. The procedures should be directed toward achieving a substantial increase in the plant being able to maintain core cooling without loss of containment integrity for events beyond the design basis. The use of vents for accidents within the normal design basis must not result in a violation of the requirements of 10 CFR 50.44 or 10 CFR 50.46.

(3) The size of the reactor coolant vents is not a critical issue. The desired venting capability can be achieved with vents in a fairly broad spectrum of sizes. The criteria for sizing a vent can be developed in several ways. One approach, which may be considered, is to specify a volume of noncondensible gas to be vented and in a specific venting time. For containments particularly vulnerable to failure from large hydrogen releases over a short period of time, the necessity and desirability for contained venting outside the containment must be considered (e.g., into a decay gas collection and storage system).

(4) Where practical, the reactor coolant system vents should be kept smaller than the size corresponding to the definition of LOCA (10 CFR 50, Appendix A). This will minimize the challenges to the emergency core cooling system (ECCS) since the inadvertent opening of a vent smaller than the LOCA definition would not require ECCS actuation, although it may result in leakage beyond technical specification limits. On PWRs, the use of new or existing lines whose smallest orifice is larger than the LOCA definition will require a valve in series with a vent valve that can be closed from the control room to terminate the LOCA that would result if an open vent valve could not be reclosed.

(5) A positive indication of valve position should be provided in the control room.

(6) The reactor coolant vent system shall be operable from the control room.

(7) Since the reactor coolant system vent will be part of the reactor coolant system pressure boundary, all requirements for the reactor pressure boundary must be met, and, in addition, sufficient redundancy should be incorporated into the design to minimize the probability of an inadvertent actuation of the system. Administrative procedures, may be a viable option to meet the single-failure criterion. For vents larger than the

LOCA definition, an analysis is required to demonstrate compliance with 10 CFR 50.46.

(8) The probability of a vent path failing to close, once opened, should be minimized; this is a new requirement. Each vent must have its power supplied from an emergency bus. A single failure within the power and control aspects of the reactor coolant vent system should not prevent isolation of the entire vent system when required. On BWRs, block valves are not required in lines with safety valves that are used for venting.

(9) Vent paths from the primary system to within containment should go to those areas that provide good mixing with containment air.

(10) The reactor coolant vent system (i.e., vent valves, block valves, position indication devices, cable terminations, and piping) shall be seismically and environmentally qualified in accordance with IEEE 344-1975 as supplemented by Regulatory Guide 1.100, 1.92 and SEP 3.92, 3.43, and 3.10. Environmental qualifications are in accordance with the May 23, 1980 Commission Order and Memorandum (CLI-80-21).

(11) Provisions to test for operability of the reactor coolant vent system should be a part of the design. Testing should be performed in accordance with subsection IWV of Section XI of the ASME Code for Category B valves.

(12) It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

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

(b) integration into emergency procedures,

(c) integration into operator training, and

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

B. BWR Design Considerations

(1) Since the BWR owners' group has suggested that the present BWR designs have an inherent capability to vent, a question relating to the capability of existing systems arises. The ability of these systems to vent the RCS of noncondensible gas generated during an accident must be demonstrated. Because of differences among the head vent systems for BWRs, each licensee or applicant should address the specific design features of this plant and compare them with the generic venting capability proposed by the BWR owners' group. In addition, the ability of these systems to meet the same requirements as the PWR vent system must be documented.

(2) In addition to RCS venting, each BWR licensee should address the ability to vent other systems, such as the isolation condenser which may be

required to maintain adequate core cooling. If the production of a large amount of noncondensible gas would cause the loss of function of such a system, remote venting of that system is required. The qualifications of such a venting system should be the same as that required for PWR venting systems.

C. PWR Vent Design Considerations

(1) Each PWR licensee should provide the capability to vent the reactor vessel head. The reactor vessel head vent should be capable of venting noncondensible gas from the reactor vessel hot legs (to the elevation of the top of the outlet nozzle) and cold legs (through head jets and other leakage paths).

(2) Additional venting capability is required for those portions of each hot leg that cannot be vented through the reactor vessel head vent or pressurizer. It is impractical to vent each of the many thousands of tubes in a U-tube steam generator; however, the staff believes that a procedure can be developed that assures sufficient liquid or steam can enter the U-tube region so that decay heat can be effectively removed from the RCS. Such operating procedures should incorporate this consideration.

(3) Venting of the pressurizer is required to assure its availability for system pressure and volume control. These are important considerations, especially during natural circulation.

<u>Applicability</u>

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

<u>Implementation</u>

Installation should take place by July 1, 1982. Until staff approval is obtained, installation may proceed; but operating procedures should not be implemented and valves should be placed in a condition so as to minimize the potential for inadvertent actuation (e.g., remove power).

<u>Type of Review</u>

A preimplementation review will be performed prior to authorizing use of the vent.

<u>Documentation Required</u>

By July 1, 1981, the licensee shall provide the following information on the reactor coolant vent system for staff review:

(1) The information requested in items 1 and 2 under "Position";

(2) A discussion of the design with respect to conformance to the design criteria discussed under "Clarification," including deviations, if any, with adequate justification for such deviations; and,

(3) Supporting information including logic diagrams, electrical schematics, piping and instrumentation diagrams, test procedures, and technical specifications.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0660

Commission Orders, May 23, 1980 (CLI-80-21)

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

Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, dated September 27, 1979.

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

II.B.2 DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POSTACCIDENT OPERATIONS

Position

With the assumption of a postaccident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during postaccident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or postaccident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

Changes to Previous Requirements and Guidance

This requirement was originally issued by letters to all operating nuclear power plants, dated September 13 and October 30, 1979, and was incorporated into NUREG-0660. Significant changes in requirements or guidance are:

(1) Adds several areas to be evaluated for access to ensure that these areas are not overlooked.

(2) Specifies that the source term for recirculated depressurized coolant need not be assumed to contain noble gas since this gas will be released from the liquid when it is depressurized.

(3) Specifies that certain systems be considered as potential sources and that leakage from systems outside of containment need not be considered as potential sources.

(4) Allows averaging over 30 days of the dose rate criteria for areas requiring continuous occupancy and that the control room and technical support center should be considered areas requiring continuous occupancy. This ensures that the dose rate criteria is applied correctly to these areas.

(5) Specifies source terms to be used in conjunction with Commission Order and Memorandum dated May 23, 1980 (CLI-80-21) on equipment qualification, and specifies schedule in above order.

(6) Because of difficulty in obtaining equipment (e.g., remote-operated valves), the implementation date is moved to January 1, 1982, or the first outage of sufficient duration thereafter, but no later than July 1, 1982.

Clarification

The purpose of this item is to ensure that licensees examine their plants to determine what actions can be taken over the short-term to reduce radiation levels and increase the capability of operators to control and mitigate the consequences of an accident. These actions should be taken pending conclusions resulting in the long term degraded core rulemaking, which may result in a need to consider additional sources.

Any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is designated as a vital area. For the purposes of this evaluation, vital areas and equipment are not necessarily the same vital areas or equipment defined in 10 CFR 73.2 for security purposes. The security center is listed as an area to be considered as potentially vital, since access to this area may be necessary to take action to give access to other areas in the plant.

The control room, technical support center (TSC), sampling station and sample analysis area must be included among those areas where access is considered vital after an accident. (See Item III.A.1.2 for discussion of the TSC and emergency operations facility.) The evaluation to determine the necessary vital areas should also include, but not be limited to, consideration of the post-LOCA hydrogen control system, containment isolation reset control area, manual ECCS alignment area (if any), motor control centers, instrument panels, emergency power supplies, security center, and radwaste control panels. Dose rate determinations need not be for these areas if they are determined not to be vital.

As a minimum, necessary modifications must be sufficient to provide for vital system operation and for occupancy of the control room, TSC, sampling station, and sample analysis area.

In order to assure that personnel can perform necessary postaccident operations in the vital areas, the following guidance is to be used by licensees to evaluate the adequacy of radiation protection to the operators:

(1) Source Term

The minimum radioactive source term should be equivalent to the source terms recommended in Regulatory Guides 1.3, 1.4, 1.7 and Standard Review Plan 15.6.5 with appropriate decay times based on plant design (i.e., you may assume the radioactive decay that occurs before fission products can be transported to various systems).

(a) Liquid-Containing Systems: 100% of the core equilibrium noble gas inventory, 50% of the core equilibirum halogen inventory, and 1% of all others are assumed to be mixed in the reactor coolant and liquids recirculated by residual heat removal (RHR), high- pressure coolant injection (HPCI), and low-pressure coolant injection (LPCI), or the equivalent of these systems. In determining the source term for recirculated, depressurized cooling water, you may assume that the water contains no noble gases.

(b) Gas-Containing Systems: 100% of the core equilibrium noble gas inventory and 25% of the core equilibrium halogen activity are assumed to be mixed in the containment atmosphere. For vapor-containing lines connected to the primary system (e.g., BWR steam lines), the concentration of radioactivity shall be determined assuming the activity is contained in the vapor space in the primary coolant system.

(2) Systems Containing the Source

Systems assumed in your analysis to contain high levels of radioactivity in a postaccident situation should include, but not be limited to, containment, residual heat removal system, safety injection systems, chemical and volume control system (CVCS), containment spray recirculation system, sample lines, gaseous radwaste systems, and standby gas treatment systems (or equivalent of these systems). If any of these systems or others that could contain high levels of radioactivity were excluded, you should explain why such systems were excluded. Radiation from leakage of systems located outside of containment need not be considered for this analysis. Leakage measurement and reduction is treated under Item III.D.1.1, "Integrity of Systems Outside Containment Likely To Contain Radioactive Material for PWRs and BWRs." Liquid waste systems need not be included in this analysis. Modifications to liquid waste systems will be considered after completion of Item III.D.1.4, "Radwaste System Design Features To Aid in Accident Recovery and Decontamination."

(3) Dose Rate Criteria

The design dose rate for personnel in a vital area should be such that the guidelines of GDC 19 will not be exceeded during the course of the accident. GDC 19 requires that adequate radiation protection be provided such that the dose to personnel should not be in excess of 5 rem whole body, or its equivalent to any part of the body for the duration of the accident. When determining the dose to an operator, care must be taken to determine the necessary occupancy times in a specific area. For example, areas requiring continuous occupancy will require much lower dose rates than areas where minimal occupancy is required. Therefore, allowable dose rates will be based upon expected occupancy, as well as the radioactive source terms and shielding. However, in order to provide a general design objective, we are providing the following dose rate criteria with alternatives to be documented on a case-by-case bases. The recommended dose rates are average rates in the area. Local hot spots may exceed the dose rate guidelines. These doses are design objectives and are <u>not</u> to be used to limit access in the event of an accident.

(a) Areas Requiring Continuous Occupancy: <15 mrem/hr (averaged over 30 days). These areas will require full-time occupancy during the course of the accident. The control room and onsite technical support center are areas where continuous occupancy will be required. The dose rate for these areas is based on the control room occupancy factors contained in SRP 6.4.

(b) Areas Requiring Infrequent Access: GDC 19. These areas may require access on an irregular basis, not continuous occupancy. Shielding should be provided to allow access at a frequency and duration estimated by the licensee. The plant radiochemical/chemical analysis laboratory, radwaste panel, motor control center, instrumentation locations, and reactor coolant and containment gas sample stations are examples of sites where occupancy may be needed often, but not continuously.

(4) Radiation Qualification of Safety-Related Equipment

The review of safety-related equipment which may be unduly degraded by radiation during postaccident operation of this equipment relates to equipment inside and outside of the primary containment. Radiation source terms calculated to determine environmental qualification of safety-related equipment consider the following:

(a) LOCA events which completely depressurize the primary system should consider releases of the source term (100% noble gases, 50% iodines, and 1% particulates) to the containment atmosphere.

(b) LOCA events in which the primary system may not depressurize should consider the source term (100% noble gases, 50% iodines, and 1% particulate) to remain in the primary coolant. This method is used to determine the qualification doses for equipment in close proximity to recirculating fluid systems inside and outside of containment. Non-LOCA events both inside and outside of containment should use 10% noble gases, 10% iodines, and 0% particulate as a source term.

The following table summarizes these considerations:

Containment	LOCA Source Term (Noble Gas/Iodine/ Particulate)	Non-LOCA High-Energy Line Break Source Term (Noble Gas/Iodine/Particulate)
	%	%
Outside	(100/50/1) in RCS	(10/10/0) in RCS
Inside	Larger of (100/50/1) in containment **or** (100/50/1) in RCS	(10/10/0) in RCS

Applicability

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

Implementation

(1) For Vital Area Access

By January 1, 1982 modifications should be completed: For operating plants, documentation should be completed by January 1, 1982. For OL applicants, documentation of the evaluation should be completed at least four months before the operating license is issued.

(2) For Equipment Qualification

All safety-related electrical equipment must be fully qualified by June 30, 1982. Documentation in accordance with:

(a) Operating Reactors and NTOL (operating license expected by February 1981): submittal to be received no later than November 1, 1980.

(b) Operating Licenses (operating license expected by June 30, 1982): submittal no later than 4 months before issuance of operating license. Operating licenses in accordance with review schedule.

Type of Review

A postimplementation review will be performed.

Documentation Required

For Vital Area Access--For operating license applicants provide a summary of the shielding design review, a description of the results of this review, and a description of the modifications made or to be made to implement the result of the review. Include in your submittal:

(1) Specification of source terms used in the evaluation; including time after shutdown that was assumed for source terms in systems;

(2) Specification of systems assumed in your analysis to contain high levels of radioactivity in a postaccident situation. If any of the systems listed in "Clarification," item 2, were excluded, explain why such systems are excluded from review;

(3) Specification of areas where access is considered necessary for vital system operation after an accident. If any of the areas listed in the "Clarification" section above were not considered to be areas requiring access after an accident, explain why they were excluded;

(4) The projected doses to individuals for necessary occupancy times in vital areas and a dose rate map for potentially occupied areas.

Documentation Required

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

For Equipment Qualification--Provide the information required by the Commission Memorandum and Order on equipment qualification (CLI-80-21).

Technical Specification Changes Required

Technical specifications will not be required.

References

NUREG-0578, Recommendation 2.1.6.b

NUREG-0660, Item II.B.2

Commission Order and Memorandum, May 23, 1980 (CLI-80-21)

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

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

Letter from D. G. Eisenhut, NRC, to All Power Reactor Licensees, dated April 25, 1980.

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

II.B.3 POSTACCIDENT SAMPLING CAPABILITY

Position

A design and operational review of the reactor coolant and containment atmosphere sampling line systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18-3/4 rem to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (in less than 2 hours) certain radionuclides that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and nonvolatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift).

Changes to Previous Requirements and Guidance

This requirement was originally issued to all operating plants by letters dated September 13 and October 30, 1979. Significant changes in requirements or guidance are:

(1) Allows combined time of 3 hours or less for sampling and analysis.

(2) Specifies that licensee may use online sampling and analysis to meet the 3-hour time requirement but must provide capability to remove grab samples of reactor coolant and containment atmosphere for separate analysis.

(3) Implementation date has been changed to January 1, 1982.

(4) Provides design guidance for sampling and analytical capability.

Clarification

The following items are clarifications of requirements identified in NUREG-0578, NUREG-0660, or the September 13 and October 30, 1979 clarification letters.

(1) The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.

(2) The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the 3-hour time frame established above, quantification of the following:

(a) certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases; iodines and cesiums, and nonvolatile isotopes);

(b) hydrogen levels in the containment atmosphere;

(c) dissolved gases (e.g., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids.

(d) Alternatively, have inline monitoring capabilities to perform all or part of the above analyses.

(3) Reactor coolant and containment atmosphere sampling during postaccident conditions shall not require an isolated auxiliary system [e.g., the letdown system, reactor water cleanup system (RWCUS)] to be placed in operation in order to use the sampling system.

(4) Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or H_2 gas in reactor coolant samples is considered adequate. Measuring the O_2 concentration is recommended, but is not mandatory.

(5) The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish water and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the licensee shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

(6) The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10 CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities). (Note that the design and operational review criterion was changed from the operational limits of 10 CFR Part 20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979 letter from H. R. Denton to all licensees).)

(7) The analysis of primary coolant samples for boron is required for PWRs. (Note that Revision 2 of Regulatory Guide 1.97, when issued, will likely specify the need for primary coolant boron analysis capability at BWR plants.)

(8) If inline monitoring is used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident and at least one sample per week until the accident condition no longer exists.

(9) The licensee's radiological and chemical sample analysis capability shall include provisions to:

(a) Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory Guide 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 μCi/g to 10 Ci/g.

(b) Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of ventilation system design which will control the presence of airborne radioactivity.

(10) Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

(11) In the design of the postaccident sampling and analysis capability, consideration should be given to the following items:

(a) Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The postaccident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

(b) The ventilation exhaust from the sampling station should be filtered with charcoal adsorbers and high-efficiency particulate air (HEPA) filters.

Applicability

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

Implementation

Installation should take place by January 1, 1982.

Type of Review

A postimplementation review will be performed.

Documentation Required

Operating Reactors--By January 1, 1982 have available for review the final design details of the implementation of the above position and clarifications. The final design includes piping and instrumentation diagrams (P&IDs), together with either (a) a summary description of procedures for sample collection, sample transfer or transport, and sample analysis, or (b) copies of procedures for sample collection, sample transfer or transport, and sample analysis. If deviations to the above position or clarification are necessary, provide detailed explanation and justification for the deviations by January 1, 1982.

Operating License Applicants--Provide a description of the implementation of the position and clarification including P&IDs, together with either (a) a summary description of procedures for sample collection, sample transfer or transport, and sample analysis, or (b) copies of procedures for sample collection, sample transfer or transport, and sample analysis, in a accordance with the proposed review schedule but in no case less than 4 months prior to the issuance of an operating license.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.8.a

NUREG-0660, Item II.B.3

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

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

II.B.4 TRAINING FOR MITIGATING CORE DAMAGE

Position

Licensees are required to develop a training program to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged. They must then implement the training program.

Changes to Previous Requirements and Guidance

Persons who must participate in the training program are to be defined.

The implementation schedule has been revised to reflect the TMI Action Plan schedule.

Clarification

Shift technical advisors and operating personnel from the plant manager through the operations chain to the licensed operators shall receive all the training indicated in Enclosure 3 to H. R. Denton's March 28, 1980 letter.

Managers and technicians in the Instrumentation and Control (I&C), health physics, and chemistry departments shall receive training commensurate with their responsibilities.

Applicability

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

Implementation

Licensees with operating reactors will develop a training program by January 1, 1981 and initiate the training program by April 1, 1981. The initial program should be complete by October 1, 1981. Applicants for operating licenses should develop a training program prior to fuel loading and complete the program prior to full-power operation.

Type of Review

A postimplementation review will be performed.

Documentation Required

Programs shall be available for review by January 1, 1981.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0660, Item II.B

Letter from H. R. Denton, NRC, to All Power Reactor Applicants and Licensees, dated March 28, 1980.

II.D.1 PERFORMANCE TESTING OF BOILING-WATER REACTOR AND PRESSURIZED-WATER REACTOR RELIEF AND SAFETY VALVES (NUREG-0578, SECTION 2.1.2)

Position

Pressurized-water reactor and boiling-water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents.

Changes to Previous Requirements and Guidance

A. Safety and Relief Valves and Piping--The types of documentation required for safety and relief valves and piping and the specific submittal dates are considered to be a clarification of item II.D.1 as described in NUREG-0660. The submittal of information was implied but not explicitly discussed in that report.

B. Block Valves--Qualification of PWR block valves is a new requirement. Since block valves must be qualified to ensure that a stuck-open relief valve can be isolated, thereby terminating a small loss-of-coolant accident due to a stuck-open relief valve. Isolation of a stuck-open power-operated relief valve (PORV) is not required to ensure safe plant shutdown. However isolation capability under all fluid conditions that could be experienced under operating and accident conditions will result in a reduction in the number of challenges to the emergency core-cooling system. Repeated unnecessary challenges to these system are undesirable.

C. ATWS Testing--Testing of anticipated transients without scram (ATWS) for later phases of the valve qualification program was noted in item II.D.1 of NUREG-0660. The clarification below provides updated information on PWR ATWS temperature and pressure conditions and clarifies that ATWS testing need not be accomplished by July 1981.

Clarification

Licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry, piping, and supports, as well as the valves themselves.

A. Performance Testing of Relief and Safety Valves--The following information must be provided in report form by October 1, 1981:

(1) Evidence supported by test of safety and relief valve functionability for expected operating and accident (non-ATWS) conditions must be provided to NRC. The testing should demonstrate that the valves will open and reclose under the expected flow conditions.

(2) Since it is not planned to test all valves on all plants, each licensee must submit to NRC a correlation or other evidence to substantiate that the valves tested in the EPRI (Electric Power Research Institute) or other generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions as prescribed in the final safety analysis report (FSAR). The effect of as-built relief and safety valve discharge piping on valve operability must also be accounted for, if it is different from the generic test loop piping.

(3) Test data including criteria for success and failure of valves tested must be provided for NRC staff review and evaluation. These test data should include data that would permit plant-specific evaluation of discharge piping and supports that are not directly tested.

B. Qualification of PWR Block Valves--Although not specifically listed as a short-term lessons-learned requirement in NUREG-0578, qualification of PWR block valves is required by the NRC Task Action Plan NUREG-0660 under task item II.D.1. It is the understanding of the NRC that testing of several commonly used block valve designs is already included in the generic EPRI PWR safety and relief valve testing program to be completed by July 1, 1981. By means of this letter, NRC is establishing July 1, 1982 as the date for verification of block valve functionability. By July 1, 1982, each PWR licensee, for plants so equipped, should provide evidence supported by test that the block or isolation valves between the pressurizer and each power-operated relief valve can be operated, closed, and opened for all fluid conditions expected under operating and accident conditions.

C. ATWS Testing--Although ATWS testing need not be completed by July 1, 1981, the test facility should be designed to accommodate ATWS conditions of approximately 3200 to 3500 (Service Level C pressure limit) psi and 700°F with sufficient capacity to enable testing of relief and safety valves of the size and type used on operating pressurized-water reactors.

<u>Applicability</u>

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

<u>Implementation</u>

See implementation schedules in the "Documentation Required" section.

<u>Type of Review</u>

Preimplementation review will be performed for EPRI and BWR test programs with respect to qualification of relief and safety valves. Also, the applicants' proposal for functional testing or qualification of PWR valves will be reviewed.

Postimplementation review will also be performed of the test data and test results as applied to plant-specific situations.

Documentation Required

Preimplementation review will be based on EPRI, BWR, and applicant submittals with regard to the various test programs. These submittals should be made on a timely basis as noted below, to allow for adequate review and to ensure that the following valve qualification dates can be met:

Final PWR (EPRI) Test Program--July 1, 1980
Final BWR Test Program--October 1, 1980
Block Valve Qualification Program--January 1, 1981

Postimplementation review will be based on the applicants' plant-specific submittals for qualification of safety relief valves and block valves. To properly evaluate these plant-specific applications, the test data and results of the various programs will also be required by the following dates:

PWR (EPRI)/BWR Generic Test Program Results--July 1, 1981
Plant-specific submittals confirming adequacy of safety and relief valves based on licensee/applicant preliminary review of generic test program results--July 1, 1981
Plant-specific reports for safety and relief valve qualification--October 1, 1981
Plant-specific submittals for piping and support evaluations--January 1, 1982
Plant-specific submittals for block valve qualification--July 1, 1982

Technical Specification Changes Required

No technical specification changes are required.

References

NUREG-0578

NUREG-0660, Item II.D.1

II.D.3 DIRECT INDICATION OF RELIEF-AND SAFETY-VALVE POSITION

Position

Reactor coolant system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

(1) The basic requirement is to provide the operator with unambiguous indication of valve position (open or closed) so that appropriate operator actions can be taken.

(2) The valve position should be indicated in the control room. An alarm should be provided in conjunction with this indication.

(3) The valve position indication may be safety grade. If the position indication is not safety grade, a reliable single-channel direct indication powered from a vital instrument bus may be provided if backup methods of determining valve position are available and are discussed in the emergency procedures as an aid to operator diagnosis of an action.

(4) The valve position indication should be seismically qualified consistent with the component or system to which it is attached.

(5) The position indication should be qualified for its appropriate environment (any transient or accident which would cause the relief or safety valve to lift) and in accordance with Commission Order, May 23rd, 1980 (CLI-20-81).

(6) It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

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

(b) integration into emergency procedures,

(c) integration into operator training, and

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

Applicability

This requirement applies to all reactor licenses and applicants for operating license. (Operating reactor licensees completed this requirement by January 1980.)

Implementation

Implementation will be completed prior to the issuance of a fuel-loading license.

Type of Review

A preimplementation review will be performed.

Documentation Required

Documentation should be provided that discusses each item of the clarification, as well as electrical schematics and proposed test procedures in accordance with the proposed review schedule, but in no case less than 4 months prior to the scheduled issuance of the staff safety evaluation report.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.3.a

NUREG-0660, Item II.D.3

NUREG-0694, Part 1

Commission Order and Memorandum, May 23, 1980 (CLI-20-81)

Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, dated September 27, 1979.

Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, dated November 9, 1979.

II.E.1.1 AUXILIARY FEEDWATER SYSTEM EVALUATION

Position

The Office of Nuclear Reactor Regulation is requiring reevaluation of the auxiliary feedwater (AFW) systems for all PWR operating plant licensees and operating license applications. This action includes:

(1) Perform a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss-of-main-feedwater-transient conditions. Particular emphasis is given to determining potential failures that could result from human errors, common causes, single-point vulnerabilities, and test and maintenance outages;

(2) Perform a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 and associated Branch Technical Position ASB 10-1 as principal guidance; and

(3) Reevaluate the AFW system flowrate design bases and criteria.

Changes to Previous Requirements and Guidance

Short-term requirements will be implemented by July 1, 1981. The date for implementation of short-term requirements has been slipped because staff review of submittals is not complete.

Clarification

Operating Plant Licenses--Items 1 and 2 above have been completed for Westinghouse (W), Combustion Engineering (C-E), and two Babcock and Wilcox (B&W) operating plants (Rancho Seco, short-term only, and TMI-1). As a result of staff review of items 1 and 2, letters were issued to these plants that required the implementation of certain short- and long-term AFW system upgrade requirements. Included in these letters was a request for additional information regarding item 3 above. The staff is now in the process of evaluating licensees' responses and commitments to these letters.

The remaining B&W operating plants (Oconee 1-3, Crystal River 3, ANO-1, and Davis-Besse 1) have submitted the analysis described in item 1 above. The analysis is presently undergoing staff review. When the results of the staff reviews are complete, each of the remaining B&W plants will receive a letter specifying the short- and long-term AFW system upgrade requirements based on item 1 above. Included in these letters will be a request for additional information regarding items 2 and 3 above.

Operating License Applicants--Operating license applicants have been requested to respond to staff letters of March 10, 1980 (W and C-E) and April 24, 1980 (B&W). These responses will be reviewed during the normal review process for these applications.

Applicability

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

Implementation

For operating reactors, the NRC staff will review and evaluate operating plant licensee responses to staff recommendations for improving AFW system reliability and requested information on AFW system flowrate design basis in time to support licensee implementation of the short-term requirements by July 1, 1981 and long-term requirements by January 1982.

Applicants for operating license should refer to letters of March 10, 1980 (W and C-E) and April 24, 1980 (B&W) for implementation schedule.

Type of Review

A preimplementation review will be performed.

Documentation Required

Licensees and applicants will be required to submit the information indicated above.

Technical Specification Changes Required

Changes to technical specifications will be determined by specific item.

Reference

NUREG-0660, Item II.E.1.1

Letter from D. F. Ross, Jr., NRC, to All Pending W and C-E License Applicants, dated March 10, 1980.

Letter from D. F. Ross, Jr., NRC, to All Pending B&W License Applicants, dated April 24, 1980.

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

PART 1: Auxiliary Feedwater System Automatic Initiation

Position

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system (AFWS), the following requirements shall be implemented in the short term:

(1) The design shall provide for the automatic initiation of the AFWS.

(2) The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of AFWS function.

(3) Testability of the initiating signals and circuits shall be a feature of the design.

(4) The initiating signals and circuits shall be powered from the emergency buses.

(5) Manual capability to initiate the AFWS from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

(6) The ac motor-driven pumps and valves in the AFWS shall be included in the automatic actuation (simultaneous and/or sequential) of the loads onto the emergency buses.

(7) The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

Changes to Previous Requirements and Guidance

There are no changes to the previous guidance issued in the H. R. Denton letter to licensees, dated October 30, 1979.

Clarification

The intent of this recommendation is to assure a reliable automatic initiation system. This objective can be met by providing a system which meets all the requirements of IEEE Standard 279-1971.

The staff has determined that the following salient paragraphs of IEEE 279-1971 should be addressed as a minimum:

IEEE 279-1971, Paragraph	
4.1*	General Functional Requirements
4.2*	Single Failure
4.3, & 4.4	Qualification
4.6	Channel Independence
4.7	Control and Protection System Interaction
4.9* & 4.10*	Capability for Testing
4.11	Channel Bypass
4.12	Operating Bypass
4.13	Indication of Bypass
4.17*	Manual Initiation

Applicability

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

Implementation

Final design information should be submitted by January 1, 1981. The safety-grade system will be installed by July 1, 1981.

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

Type of Review

A postimplementation review will be performed.

Documentation Required

Each licensee shall provide by January 1, 1981 sufficient documentation to support a reasonable assurance finding by the NRC that the above requirements are met. The documentation should include as a minimum

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

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

*These requirements were part of the short-term, control-grade requirements.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.7.a

NUREG-0660, Item II.E.1.2

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

PART 2: Auxiliary Feedwater System Flowrate Indication

Position

Consistent with satisfying the requirements set forth in General Design Criterion 13 to provide the capability in the control room to ascertain the actual performance of the AFWS when it is called to perform its intended function, the following requirements shall be implemented:

1. Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

2. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

Changes to Previous Requirements and Guidance

The requirements for Westinghouse (W) and Combustion Engineering (C-E) plants have been relaxed to require only a single-channel flow indication, instead of redundant channels. This single channel need not be seismically qualified nor need it be powered from a Class 1E power source.

The auxiliary feedwater flow indication requirements have been relaxed for PWRs with U-tube steam generators because flow indication is of secondary importance in assuring steam generator cooling capability for steam generators of this design.

Clarification

The intent of this recommendation is to assure a reliable indication of AFWS performance. This objective can be met by providing an overall indication system that meets the following appropriate design principles:

(1) For Babcock and Wilcox Plants

(a) To satisfy these requirements, B&W plants must provide as a minimum two auxiliary feedwater flowrate indicators for each steam generator.

(b) The flow indication system should conform to the following salient paragraphs of IEEE 279-1971:

IEEE 279-1971, PARAGRAPH

4.1*	General Functional Requirements
4.2*	Single Failure
4.3 & 4.4	Qualification
4.6	Channel Independence
4.7	Control and Protection System Interaction
4.9* & 4.10*	Capability for Testing

(2) For Westinghouse and Combustion Engineering Plants

(a) To satisfy these requirements, W and C-E plants must provide as a minimum one auxiliary feedwater flowrate indicator and one wide-range steam-generator level indicator for each steam generator or two flow-rate indicators.

(b) The flow indication system should be:

(i) environmentally qualified
(ii) powered from highly reliable, battery-backed non-Class IE power source
(iii) periodically testable
(iv) part of plant quality assurance program
(v) capable of display on demand

It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

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

(b) integration into emergency procedures,

(c) integration into operator training, and

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

Applicability

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

Implementation

Final design information should be submitted by January 1, 1981. The system will be installed by July 1, 1981. All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

*These requirements were part of the short-term, control-grade requirements.

Type of Review

A postimplementation review will be performed.

Documentation Required

By January 1, 1981 each licensee shall provide sufficient documentation to support a reasonable assurance finding by the NRC that the above-specified requirements have been met. The documentation should include as a minimum:

(1) A discussion of the design with respect to each of the requirements specified above; and

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.7.b

NUREG-0660, Item II.E.1.2

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

II.E.3.1 EMERGENCY POWER SUPPLY FOR PRESSURIZER HEATERS

Position

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17, and 20 of Appendix A to 10 CFR Part 50 for the event of loss of offsite power, the following positions shall be implemented:

(1) The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

(2) Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

(3) The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

(4) Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements in October 30, 1979 letter from H. R. Denton to all licensees.

Clarification

(1) Redundant heater capacity must be provided, and each redundant heater or group of heaters should have access to only one Class 1E division power supply.

(2) The number of heaters required to have access to each emergency power source is that number required to maintain natural circulation in the hot standby condition.

(3) The power sources need not necessarily have the capacity to provide power to the heaters concurrently with the loads required for loss-of-coolant accident.

(4) Any changeover of the heaters from normal offsite power to emergency onsite power is to be accomplished manually in the control room.

(5) In establishing procedure to manually load the pressurizer heaters onto the emergency power sources, careful consideration must be given to:

   (a) which ESF loads may be appropriately shed for a given situation;
   (b) reset of the safety injection actuation signal to permit the operation of the heaters; and
   (c) instrumentation and criteria for operator use to prevent overloading a diesel generator.

(6) The Class IE interfaces for main power and control power are to be protected by safety-grade circuit breakers (see also Regulatory Guide 1.75).

(7) Being non-Class IE loads, the pressurizer heaters must be automatically shed from the emergency power sources upon the occurrence of a safety injection actuation signal (see item 5.b. above).

Applicability

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

Implementation

Implementation is complete for operating reactors.

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

Type of Review

A review will be performed as part of the licensing review process.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.1

NUREG-0660, Item II.E.3.1

NUREG-0694, Part 2

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

II.E.4.1 DEDICATED HYDROGEN PENETRATIONS

Position

Plants using external recombiners or purge systems for postaccident combustible gas control of the containment atmosphere should provide containment penetration systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single-failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

The procedures for the use of combustible gas control systems following an accident that results in a degraded core and release of radioactivity to the containment must be reviewed and revised, if necessary.

Changes to Previous Requirements and Guidance

Changes in the implementation date have been made because of equipment procurement problems and to minimize the number of plant shutdowns necessary must make to install equipment related to the TMI Action Plan.

Clarification

(1) An acceptable alternative to the dedicated penetration is a combined design that is single-failure proof for containment isolation purposes and single-failure proof for operation of the recombiner or purge system.

(2) The dedicated penetration or the combined single-failure proof alternative shall be sized such that the flow requirements for the use of the recombiner or purge system are satisfied. The design shall be based on 10 CFR 50.44 requirements.

(3) Components furnished to satisfy this requirement shall be safety grade.

(4) Licensees that rely on purge systems as the primary means for controlling combustible gases following a loss-of-coolant accident should be aware of the positions taken in SECY-80-399, "Proposed Interim Amendments to 10 CFR Part 50 Related to Hydrogen Control and Certain Degraded Core Considerations." This proposed rule, published in the Federal Register on October 2, 1980, would require plants that do not now have recombiners to have the capacity to install external recombiners by January 1, 1982. (Installed internal recombiners are an acceptable alternative to the above.)

(5) Containment atmosphere dilution (CAD) systems are considered to be purge systems for the purpose of implementing the requirements of this TMI Task Action item.

Applicability

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

Implementation

For operating reactors, design modifications shall be completed by July 1, 1981.

Operating license applicants must have design changes completed by July 1, 1981 or prior to issuance of an operating license, whichever is later.

Type of Review

For operating reactors review will take place after implementation.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will be required for plants that need to make modifications.

References

NUREG-0578

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

II.E.4.2 CONTAINMENT ISOLATION DEPENDABILITY

Position

(1) Containment isolation system designs shall comply with the recommendations of Standard Review Plan Section 6.2.4 (i.e., that there be diversity in the parameters sensed for the initiation of containment isolation).

(2) All plant personnel shall give careful consideration to the definition of essential and nonessential systems, identify each system determined to be essential, identify each system determined to be nonessential, describe the basis for selection of each essential system, modify their containment isolation designs accordingly, and report the results of the reevaluation to the NRC.

(3) All nonessential systems shall be automatically isolated by the containment isolation signal.

(4) 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.

(5) The containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions.

(6) 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.3.f during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days. (A copy of the Staff Interim Position is enclosed as Attachment 1.)

(7) Containment purge and vent isolation valves must close on a high radiation signal.

Changes to Previous Requirements and Guidance

Although there has been no change in the requirements since NUREG-0660 was issued, positions 5, 6, and 7 have not been previously transmitted to licensees. These three positions were not part of the original NUREG-0578 requirements of Recommendation 2.1.4; however they were added to item II.E.4.1 of NUREG-0660 as a result of further staff evaluation of features needed to improve containment isolation dependability. The schedule for implementing positions 5, 6, and 7 on operating plants has been changed from NUREG-0660. The design for position 5 shall be completed by January 1, 1981 with modifications completed by July 1, 1981. Position 6 shall be implemented by January 1, 1981. Position 7 shall be implemented by July 1, 1981 or during the following outage of sufficient duration, but no later than January 1, 1982.

Clarification

(1) The reference to SRP 6.2.4 in position 1 is only to the diversity requirements set forth in that document.

(2) For postaccident situations, each nonessential 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 nonessential penetration must receive the diverse isolation signals.

(3) Revision 2 to Regulatory Guide 1.141 will contain guidance on the classification of essential versus nonessential systems and is due to be issued by June 1981. Requirements for operating plants to review their list of essential and nonessential systems will be issued in conjunction with this guide including an appropriate time schedule for completion.

(4) Administrative provisions to close all isolation valves manually before resetting the isolation signals is not an acceptable method of meeting position 4.

(5) 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.

(6) The containment pressure history during normal operation 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. Applicants for an operating license and operating plant licensees that have operated less than one year should use pressure history data from similar plants that have operated more than one year, if possible, to arrive at a minimum containment setpoint pressure.

(7) Sealed-closed purge isolation valves shall be under administrative control to assure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator. Checking the valve position light in the control room is an adequate method for verifying every 24 hours that the purge valves are closed.

Applicability

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

Implementation

As part of Category "A" lessons-learned requirements, all operating plants were required to be in conformance with positions 1 through 4 by January 1, 1980.

Each licensee will provide, and justify, the minimum containment pressure that will be used to initiate containment isolation as stated in position 5 by January 1, 1981. By July 1, 1981, all operating plants must be in compliance with position 5. All operating plants must be in compliance with position 6 by January 1, 1981. All operating plants must be in compliance with position 7 by July 1, 1981.

Applicants for an operating license must be in compliance with positions 1 through 4 before receiving an operating license. Applicants must be in compliance with positions 5 and 7 by July 1, 1981, and position 6 by January 1, 1981 or before they receive their operating license, whichever is later for each position.

Applicants must provide, and justify, the minimum containment pressure that will be used for initiating containment isolation as stated in position 5.

Type of Review

A postimplementation review will be performed for operating reactors.

Documentation Required

The type and dates of documentation required are as previously stated.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.4

NUREG-0660, Item II.E.4.2

Standard Review Plan, Section 6.2.4

II.E.4.2, ATTACHMENT 1, OCTOBER 23, 1979* INTERIM POSITION FOR CONTAINMENT PURGE AND VENT VALVE OPERATION PENDING RESOLUTION OF ISOLATION VALVE OPERABILITY

Once the conditions listed below are met, restrictions on use of the containment purge and vent system isolation valves will be revised based on our review of your responses to the November 1978 letter on this subject justifying your proposed operational mode. The November 1978 letters to all licensees identified certain events related to containment purging of concern to the NRC and requested commitments to either cease purging or justify purging operations. The revised restrictions can be established separately for each system.

(1) Whenever the containment integrity is required, emphasis should be placed on operating the containment in a passive mode as much as possible and on limiting all purging and venting times to as low as achievable. To justify venting or purging, there must be an established need to improve working conditions to perform a safety-related surveillance or safety-related maintenance procedure. (Examples of improved working conditions would include deinerting, reducing temperature,** humidity, and airborne activity sufficiently to permit efficient performance or to significantly reduce occupational radiation exposures.)

(2) Maintain the containment purge and vent isolation valves closed whenever the reactor is not in the cold shutdown or refueling mode until such time as you can show that:

   (a) All isolation valves greater than 3-in. nominal diameter used for containment purge and venting operations are operable under the most severe design-basis-accident (DBA) flow-condition loading and can close within the time limit stated in the technical specifications, design criteria, or operating procedures. The operability of butterfly valves may, on an interim basis, be demonstrated by limiting the valve to be no more than 30° to 50° open (90° being full open). The maximum opening shall be determined in consultation with the valve supplier. The valve opening must be such that the critical valve parts will not be damaged by DBA-LOCA (loss-of-coolant accident) loads and that the valve will tend to close when the fluid dynamic forces are introduced, and

   (b) Modifications, as necessary, have been made to segregate the containment ventilation isolation signals to ensure that, as a minimum, at least one of the automatic safety injection actuation signals is uninhibited and operable to initiate valve closure when any other isolation signal may be blocked, reset, or overridden.

*Previously referred to as DOE Interim Position.
**Only when temperature and humidity controls are not in the present design.

II.F.1 ADDITIONAL ACCIDENT-MONITORING INSTRUMENTATION

Introduction

Item II.F.1 of NUREG-0660 contains the following subparts:

(1) Noble gas effluent radiological monitor;

(2) Provisions for continuous sampling of plant effluents for postaccident releases of radioactive iodines and particulates and onsite laboratory capabilities (this requirement was inadvertently omitted from NUREG-0660; see Attachment 2 that follows, for position);

(3) Containment high-range radiation monitor;

(4) Containment pressure monitor;

(5) Containment water level monitor; and

(6) Containment hydrogen concentration monitor.

NUREG-0578 provided the basic requirements associated with items (1) through (3) above. Letters issued to all operating nuclear power plants dated September 13, 1979 and October 30, 1979 provided clarification of staff requirements associated with items (1) through (6) above. Attachments 1 through 6 present the NRC position on these matters.

It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

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

(b) integration into emergency procedures,

(c) integration into operator training, and

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

References

NUREG-0660, item II.F.1

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

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

II.F.1, ATTACHMENT 1, NOBLE GAS EFFLUENT MONITOR

Position

Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions. Multiple monitors are considered necessary to cover the ranges of interest.

(1) Noble gas effluent monitors with an upper range capacity of 10^5 µCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

(2) Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (as low as reasonably achievable (ALARA)) concentrations to a maximum of 10^5 µCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of ten.

Changes to Previous Requirements and Guidance

This requirement was originally issued by letters to all operating power plants dated September 13 and October 30, 1979. Significant changes in requirements or guidance are:

(1) Deletion of specific range overlap requirement.

(2) Specifies that offline monitoring is not required for safety valve and dump valve discharge lines.

(3) Implementation date changed from January 1, 1981 to January 1, 1982.

(4) Specifies that inline sensors are acceptable for concentrations between 10^2 µCi/cc to 10^5 µCi/cc of noble gases.

Clarification

(1) Licensees shall provide continuous monitoring of high-level, postaccident releases of radioactive noble gases from the plant. Gaseous effluent monitors shall meet the requirements specified in the enclosed Table II.F.1-1. Typical plant effluent pathways to be monitored are also given in the table.

(2) The monitors shall be capable of functioning both during and following an accident. System designs shall accommodate a design-basis release and then be capable of following decreasing concentrations of noble gases.

(3) Offline monitors are not required for the PWR secondary side main steam safety valve and dump valve discharge lines. For this application, externally mounted monitors viewing the main steam line upstream of the valves are acceptable with procedures to correct for the low energy gammas the external monitors would not detect. Isotopic identification is not required.

(4) Instrumentation ranges shall overlap to cover the entire range of effluents from normal (ALARA) through accident conditions.

The design description shall include the following information.

(a) System description, including:

(i) instrumentation to be used, including range or sensitivity, energy dependence or response, calibration frequency and technique, and vendor's model number, if applicable;

(ii) monitoring locations (or points of sampling), including description of methods used to assure representative measurements and background correction;

(iii) location of instrument readout(s) and method of recording, including description of the method or procedure for transmitting or disseminating the information or data;

(iv) assurance of the capability to obtain readings at least every 15 minutes during and following an accident; and,

(v) the source of power to be used.

(b) Description of procedures or calculational methods to be used for converting instrument readings to release rates per unit time, based on exhaust air flow and considering radionuclide spectrum distribution as a function of time after shutdown.

<u>Applicability</u>

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

<u>Implementation</u>

Implementation must be completed by January 1, 1982.

<u>Type of Review</u>

A postimplementation review will be performed.

<u>Documentation Required</u>

Licensees and licensing applicants should have available for review the final design description of the as-built system, including piping and instrument diagrams together with either (1) a description of procedures for system operation and calibration, or (2) copies of procedures for system operation and calibration. Operating Reactors--By January 1, 1981 operating reactors should have available for review the final design details of the implementation of the above position and clarifications. If deviations to the above position or clarification are necessary, provide detailed explanation and justification for the deviations by January 1, 1981.

License applicants will submit the above details in accordance with the proposed review schedule, but in no case less than 4 months prior to the issuance of an operating license.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.8.b

American National Standard ANSI N13.1-1969, February 1969

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

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

TABLE II.F.1-1

HIGH-RANGE NOBLE GAS EFFLUENT MONITORS

REQUIREMENT - Capability to detect and measure concentrations of noble gas fission products in plant gaseous effluents during and following an accident. All potential accident release paths shall be monitored.

PURPOSE - To provide the plant operator and emergency planning agencies with information on plant releases of noble gases during and following an accident.

DESIGN BASIS MAXIMUM RANGE

Design range values may be expressed in Xe-133 equivalent values for monitors employing gamma radiation detectors or in microcuries per cubic centimeter of air at standard temperature and pressure (STP) for monitors employing beta radiation detector (Note: 1R/hr @1 ft = 6.7 Ci Xe-133 equivalent for point source). Calibrations with a higher energy source are acceptable. The decay of radionuclide noble gases after an accident (i.e., the distribution of noble gases changes) should be taken into account.

10^5 µCi/cc - Undiluted containment exhaust gases (e.g., PWR reactor building purge, PWR drywell purge through the standby gas treatment system).

- Undiluted PWR condenser air removal system exhaust.

10^4 µCi/cc - Diluted containment exhaust gases (e.g., > 10:1 dilution, as with auxiliary building exhaust air).

- BWR reactor building (secondary containment) exhaust air.

- PWR secondary containment exhaust air.

10^3 µCi/cc - Buildings with systems containing primary coolant or primary coolant offgases (e.g., PWR auxiliary buildings, BWR turbine buildings).

- PWR steam safety valve discharge, atmospheric steam dump valve discharge.

10^2 µCi/cc - Other release points (e.g., radwaste buildings, fuel handling/storage buildings).

TABLE II.F.1-1

(CONTINUED)

REDUNDANCY	-	Not required; monitoring the final release point of several discharge inputs is acceptable.
SPECIFICATIONS	-	(None) Sampling design criteria per ANSI N13.1.
POWER SUPPLY	-	Vital instrument bus or dependable backup power supply to normal ac.
CALIBRATION	-	Calibrate monitors using gamma detectors to Xe-133 equivalent (1 R/hr @ 1 ft = 6.7 Ci Xe-133 equivalent for point source). Calibrate monitors using beta detectors to Sr-90 or similar long-lived beta isotope of at least 0.2 MeV.
DISPLAY	-	Continuous and recording as equivalent Xe-133 concentrations or μCi/cc of actual noble gases.
QUALIFICATION	-	The instruments shall provide sufficiently accurate responses to perform the intended function in the environment to which they will be exposed during accidents.
DESIGN CONSIDERATIONS	-	Offline monitoring is acceptable for all ranges of noble gas concentrations. Inline (induct) sensors are acceptable for 10^2 μCi/cc to 10^5 μCi/cc noble gases. For less than 10^2 μCi/cc, offline monitoring is recommended. Upsteam filtration (prefiltering to remove radioactive iodines and particulates) is not required; however, design should consider all alternatives with respect to capability to monitor effluents following an accident. For external mounted monitors (e.g., PWR main steam line), the thickness of the pipe should be taken into account in accounting for low-energy gamma radiation.

II.F.1, ATTACHMENT 2 SAMPLING AND ANALYSIS OF PLANT EFFLUENTS

Position

Because iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

Changes to Previous Requirements and Guidance

This requirement was originally issued by letters to all operating power plants dated September 13, 1979 and October 30, 1979. This requirement was inadvertently omitted from NUREG-0660. Significant changes in requirements or guidance are:

(1) Changes implementation date to January 1, 1982.

(2) Specifies a shielding basis design envelope for design of samplers and sample transport devices.

(3) Specifies provisions for isokinetic sampling.

(4) Specifies representative sampling per criteria of ANSI N131-1969.

(5) Allows use of gamma radiation measurement and shielding/distance factors in lieu of analysis of highly radioactive samples.

Clarification

(1) Licensees shall provide continuous sampling of plant gaseous effluent for postaccident releases of radioactive iodines and particulates to meet the requirements of the enclosed Table II.F.1-2. Licensees shall also provide onsite laboratory capabilities to analyze or measure these samples. This requirement should not be construed to prohibit design and development of radioiodine and particulate monitors to provide online sampling and analysis for the accident condition. If gross gamma radiation measurement techiques are used, then provisions shall be made to minimize noble gas interference.

(2) The shielding design basis is given in Table II.F.1-2. The sampling system design shall be such that plant personnel could remove samples, replace sampling media and transport the samples to the onsite analysis facility with radiation exposures that are not in excess of the criteria of GDC 19 of 5-rem whole-body exposure and 75 rem to the extremities during the duration of the accident.

(3) The design of the systems for the sampling of particulates and iodines should provide for sample nozzle entry velocities which are approximately isokinetic (same velocity) with expected induct or instack air velocities. For accident conditions, sampling may be complicated by a reduction in stack or vent effluent velocities to below design levels, making it necessary to substantially reduce sampler intake flow rates to achieve the isokinetic condition. Reductions in air flow may well be beyond the

capability of available sampler flow controllers to maintain isokinetic conditions; therefore, the staff will accept flow control devices which have the capability of maintaining isokinetic conditions with variations in stack or duct design flow velocity of $\pm$ 20%. Further departure from the isokinetic condition need not be considered in design. Corrections for non-isokinetic sampling conditions, as provided in Appendix C of ANSI 13.1-1969 may be considered on an ad hoc basis.

(4) Effluent streams which may contain air with entrained water, e.g. air ejector discharge, shall have provisions to ensure that the adsorber is not degraded while providing a representative sample, e.g., heaters.

Applicability

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

Implementation

This requirement will be implemented by January 1, 1982.

Type of Review

A postimplementation review will be performed.

Documentation Required

By January 1, 1981 operating reactors should have available for review the final design details of the implementation of the above position and clarifications. If deviations to the above position or clarification are necessary, provide detailed explanation and justification for the deviations by January 1, 1981.

License applicants will submit the above details in accordance with the proposed review schedule, but in no case less than 4 months prior to the issuance of an operating license.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.8.b

American National Standard ANSI N13.1-1969, February 1969

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

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

TABLE II.F.1-2

SAMPLING AND ANALYSIS OR MEASUREMENT OF HIGH-RANGE RADIOIODINE AND PARTICULATE EFFLUENTS IN GASEOUS EFFLUENT STREAMS

EQUIPMENT - Capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluents during and following an accident. The capability to sample and analyze for radioiodine and particulate effluents is not required for PWR secondary main steam safety valve and dump valve discharge lines.

PURPOSE - To determine quantitative release of radioiodines and particulates for dose calculation and assessment.

DESIGN BASIS SHIELDING ENVELOPE - 10^2 μCi/cc of gaseous radioiodine and particulates, deposited on sampling media; 30 minutes sampling time, average gamma energy (E) of 0.5 MeV.

SAMPLING MEDIA

- Iodine > 90% effective adsorption for all forms of gaseous iodine.
- Particulates > 90% effective retention for 0.3 micron (μ) diameter particles.

SAMPLING CONSIDERATIONS

- Representative sampling per ANSI N13.1-1969.
- Entrained moisture in effluent stream should not degrade adsorber.
- Continuous collection required whenever exhaust flow occurs.
- Provisions for limiting occupational dose to personnel incorporated in sampling systems, in sample handling and transport, and in analysis of samples.

ANALYSIS

- Design of analytical facilities and preparation of analytical procedures shall consider the design basis sample.
- Highly radioactive samples may not be compatible with generally accepted analytical procedures; in such cases, measurement of emissive gamma radiations and the use of shielding and distance factors should be considered in design.

II.F.1, ATTACHMENT 3, CONTAINMENT HIGH-RANGE RADIATION MONITOR

Position

In containment radiation-level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be developed and qualified to function in an accident environment.

Changes to Previous Requirements and Guidance

This requirement was originally issued by letters to all operating power plants dated September 13 and October 30, 1979 and was incorporated into NUREG-0660. Significant changes in requirements or guidance are:

(1) Specifies a lower range so that the monitor can follow the radiation increase from lower levels of radiation for personnel safety up to the maximum expected in major accidents;

(2) Specifies that monitors be located in containment to view a large segment of the containment atmosphere which will more accurately reflect and monitor accident conditions;

(3) Requires monitors in both primary containment (drywell) and secondary containment for BWR Mark III, because under certain accident conditions the drywell and secondary containment are interconnected through the suppression pool resulting in high radiation in both containments following an accident;

(4) Specifies accuracy and energy response in order to ensure accurate measurements independent of the energy spectrum of an accident (this specification was referenced in the letter of October 30; 1979 in referencing Regulatory Guide 1.97, Rev. 2);

(5) Specifies design and qualification criteria to ensure that the monitor will function in an accident environment;

(6) Specifies that electronic calibration is acceptable for higher dose rate ranges because such methods are sufficient to provide acceptable accuracy;

(7) Deletes the requirement for NRR (Office of Nuclear Reactor Regulation) preimplementation review if the monitors meet the listed specifications because the monitor specifications ensure that adequate monitors will be installed;

(8) Moves the implementation date to January 1, 1982 because of the potential unavailability of appropriate equipment and because the qualification of monitors is incomplete;

(9) Requires documentation by July 1, 1981, of alternative proposals for monitors that do not meet the requirements of Table II.F.1-3.

Clarification

(1) Provide two radiation monitor systems in containment which are documented to meet the requirements of Table II.F.1-3.

(2) The specification of 10^8 rad/hr in the above position was based on a calculation of postaccident containment radiation levels that included both particulate (beta) and photon (gamma) radiation. A radiation detector that responds to both beta and gamma radiation cannot be qualified to post-LOCA (loss-of-coolant accident) containment environments but gamma-sensitive instruments can be so qualified. In order to follow the course of an accident, a containment monitor that measures only gamma radiation is adequate. The requirement was revised in the October 30, 1979 letter to provide for a photon-only measurement with an upper range of 10^7 R/hr.

(3) The monitors shall be located in containment(s) in a manner as to provide a reasonable assessment of area radiation conditions inside containment. The monitors shall be widely separated so as to provide independent measurements and shall "view" a large fraction of the containment volume. Monitors should not be placed in areas which are protected by massive shielding and should be reasonably accessible for replacement, maintenance, or calibration. Placement high in a reactor building dome is not recommended because of potential maintenance difficulties.

(4) For BWR Mark III containments, two such monitoring systems should be inside both the primary containment (drywell) and the secondary containment.

(5) The monitors are required to respond to gamma photons with energies as low as 60 keV and to provide an essentially flat response for gamma energies between 100 keV and 3 MeV, as specified in Table II.F.1-3. Monitors that use thick shielding to increase the upper range will underestimate postaccident radation levels in containment by several orders of magnitude because of their insensitivity to low energy gammas and are not acceptable.

Applicability

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

Implementation Date

Implementation for operating reactors must be completed by January 1, 1982. License applicants will submit the required documentation in accordance with the appropriate review schedule, but in no case less than 4 months prior to the issuance of the staff evaluation report for an operating license.

Type of Review

A postimplementation review will be performed.

Documentation Required

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

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

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

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

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

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.8.b

NUREG-0660

Regulatory Guide 1.97, Revision 2

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

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

TABLE II.F.1-3

CONTAINMENT HIGH-RANGE RADIATION MONITOR

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

II.F.1, ATTACHMENT 4, CONTAINMENT PRESSURE MONITOR

Position

A continuous indication of containment pressure shall be provided in the control room of each operating reactor. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and -5 psig for all contaiments.

Changes to Previous Requirements And Guidance

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

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

Clarification

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

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

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

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

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

APPLICABILITY

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

Implementation

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

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

Type of Review

A postimplementation review will be performed for operating reactors.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specification will be required.

References

NUREG-0660

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

II.F.1, ATTACHMENT 5, CONTAINMENT WATER LEVEL MONITOR

Position

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWRs and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided for PWRs and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool.

Changes to Previous Requirements and Guidance

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

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

Clarification

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

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

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

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

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

Applicability

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

Implementation

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

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

Type of Review

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

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

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0660

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

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

Position

A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

Changes to Previous Requirements and Guidance

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

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

Clarification

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

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

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

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

Applicability

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

Implementation

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

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

Type of Review

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

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

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0660

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

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

Position

Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

Changes to Previous Requirements and Guidance

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

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

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

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

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

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

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

Clarification

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(b) integration into emergency procedures,

(c) integration into operator training, and

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

Applicability

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

Implementation

This requirement must be implemented by January 1, 1982.

Type of Review

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

Documentation Required

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

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

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

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

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

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

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

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

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

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

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

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

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.3.b

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

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

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

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

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

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

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

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

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

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

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

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

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

(b) integration into emergency procedures,

(c) integration into operator training, and

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

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

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

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

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

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

II.G.1 EMERGENCY POWER FOR PRESSURIZER EQUIPMENT

Position

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17, and 20 of Appendix A to 10 CFR Part 50 for the event of loss-of-offsite power, the following positions shall be implemented:

Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

(1) Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

(2) Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

(3) Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

(4) The pressurizer level indication instrument channels shall be powered from the vital instrument buses. The buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

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

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

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

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

Applicability

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

Implementation

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

Type of Review

No further review for operating reactors is necessary.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.1

NUREG-0660, Item II.G.1

NUREG-0694, Part 1

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

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

Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

No further clarification is required at this time.

Applicability

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

Implementation

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

Type of Review

A review is not applicable at this time.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will not be required.

Reference

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

II.K.2.8 AUXILIARY FEEDWATER SYSTEM UPGRADING

Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

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

Applicability

This requirement applies to all B&W operating reactors.

Implementation

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

Type of Review

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

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will be made as required.

References

NUREG-0645, Volume 1, Section 2.4.6

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

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

Position

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

Changes to Previous Requirements and Guidance

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

Clarification

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

Applicability

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

Implementation

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

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

Type of Review

A postimplementation review will be performed.

Documentation Required

Operating Reactors--Documentation has been completed.

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

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

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

Technical Specification Changes Required

Changes to technical specifications will be determined following staff review.

References

NUREG-0645, Volume 1, Section 2.4.6

NUREG-0694, Part 2

Commission Orders on B&W Plants

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

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

II.K.2.10 SAFETY-GRADE ANTICIPATORY REACTOR TRIP

Position

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

Changes to Previous Requirements and Guidance

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

Clarification

Operating Reactors

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

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

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

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

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

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

Applicability

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

Implementation

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

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

Type of Review

A postimplementation review will be performed.

Documentation Required

The following information was identified as required by the staff:

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

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

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

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

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0645, Volume 1, Section 2.4.6

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

Commission Orders on B&W Plants

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

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

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

Position

A detailed analysis shall be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.

Changes to Previous Requirements and Guidance

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

Clarification

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

Applicability

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

Implementation

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will be determined following staff review.

References

NUREG-0645, Volume 1, Section 2.4.5

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

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

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

Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

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

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

Applicability

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

Implementation

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0565, Recommendation 2.6.2.1

NUREG-0645, Volume 1, Section 2.4.6

NUREG-0694, Part 2

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

II.K.2.16 REACTOR COOLANT PUMP SEAL DAMAGE

Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

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

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

Applicability

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

Implementation

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0565, Recommendation 2.6.2.f

NUREG-0645, Volume 1, Section 2.4.6

NUREG-0694, Part 2

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

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

Position

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

Changes to Previous Requirements and Guidance

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

Clarification

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

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

Applicability

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

Implementation

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

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

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

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

II.K.2.19 SEQUENTIAL AUXILIARY FEEDWATER FLOW ANALYSIS

Position

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

Changes to Previous Requirements and Guidance

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

Clarification

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

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

Applicability

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

Implementation

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

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0645, Volume 1, Section 2.4.6

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

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

Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

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

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

Applicability

This requirement applies to all B&W operating reactors.

Implementation

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

Type of Review

A postimplementation review will be performed.

Documentation Required

No additional documentation is required at this time.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0565, Recommendation 2.6.2.c

NUREG-0645, Volume 1, Section 2.4.6

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

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

Position

All PWR licensees should provide a system that uses the PORV block valve to protect against a small-break loss-of-coolant accident. This system will automatically cause the block valve to close when the reactor coolant system pressure decays after the PORV has opened. Justification should be provided to assure that failure of this system would not decrease overall safety by aggravating plant transients and accidents.

Each licensee shall perform a confirmatory test of the automatic block valve closure system following installation.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

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

Applicability

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

Implementation

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

Type of Review

A preimplementation review will be performed.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0565, Recommendation 2.1.2.a

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

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

NUREG-0660

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

Position

(1) The licensee should submit a report for staff review documenting the various actions taken to decrease the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV) and show how those actions constitute sufficient improvements in reactor safety.

(2) Safety-valve failure rates based on past history of the operating plants designed by the specific nuclear steam supply system (NSSS) vendor should be included in the report submitted in response to (1) above.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

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

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

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

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

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

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

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

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

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

Applicability

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

Implementation

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

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

WASH-1400 (NUREG-75/014)

NUREG-0565, Recommendation 2.1.2.d

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

NUREG-0635, Recommendation 3.2.4.c

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

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

Position

Tripping of the reactor coolant pumps in case of a loss-of-coolant accident (LOCA) is not an ideal solution. Licensees should consider other solutions to the small-break LOCA problem (for example, an increase in safety injection flow rate). In the meantime, until a better solution is found, the reactor coolant pumps should be tripped automatically in case of a small-break LOCA. The signals designated to initiate the pump trip are discussed in NUREG-0623.

Changes to Previous Requirements and Guidance

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

Clarification

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

Applicability

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

Implementation

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

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

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

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

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

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

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

Type of Review

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

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications are to be determined.

References

NUREG-0565, Recommendation 2.3.2.a

NUREG-0611, Recommendation 3.2.2.a

NUREG-0623

NUREG-0635, Recommendation 3.2.2.a

NUREG-0660

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

Position

Most overpressure transients should not result in the opening of the power-operated relief valve (PORV). Therefore, licensees should document that the PORV will open in less than 5% of all anticipated overpressure transients using the revised setpoints and anticipatory trips for the range of plant conditions which might occur during a fuel cycle.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

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

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

Applicability

This requirement applies to all B&W operating reactors.

Implementation

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0565, Recommendation 2.1.2.b

NUREG-0645, Recommendation 2.4.5, Item 27

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

II.K.3.9 PROPORTIONAL INTEGRAL DERIVATIVE CONTROLLER MODIFICATION

Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

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

Applicability

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

Implementation

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

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0611, Recommendation 3.2.4.b

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

II.K.3.10 PROPOSED ANTICIPATORY TRIP MODIFICATION

Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

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

Applicability

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

Implementation

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

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

Type of Review

A preimplementation review will be performed.

Documentation Required

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

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0611, Recommendation 3.2.4.c

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

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

Position

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

Changes to Previous Requirements and Guidance

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

Clarification

No further clarification is required.

Applicability

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

Implementation

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

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

Type of Review

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

Documentation Required

(1) Licensees should submit confirmation of existence of anticipatory reactor trip upon turbine trip or submit proposed design changes and schedule for implementation by January 1, 1981.

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

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0611, Recommendation 3.2.4.a

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

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

Position

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

Changes to Previous Requirements and Guidance

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

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

Clarification

No further clarification is required.

Applicability

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

Implementation

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

Type of Review

A preimplementation review will be performed if required.

Documentation Required

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

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

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

Technical Specification Changes Required

Changes to technical specifications will be required.

Reference

NUREG-0626, Recommendation A.1

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

Position

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

Changes to Previous Requirements and Guidance

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

Clarification

No further clarification is required.

Applicability

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

Implementation

Design modifications shall be completed by January 1, 1982.

Type of Review

A postimplementation review will be performed.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0626, Recommendation A.2.

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

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

Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

No further clarification is required.

Applicability

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

Implementation

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

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

Technical Specifications Changes Required

Changes to technical specifications will be required.

Reference

NUREG-0626, Recommendation A.3

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

Position

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

(1) Additional anticipatory scram on loss of feedwater,

(2) Revised relief-valve actuation setpoints,

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

(4) Lower operating pressures,

(5) Earlier initiation of ECC systems

(6) Heat removal through emergency condensers,

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

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

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

(10) Lowering the pressure setpoint for MSIV closure,

(11) Reducing the testing frequency of the MSIVs,

(12) More-stringent valve leakage criteria, and

(13) Early removal of leaking valves.

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

Changes to Previous Requirements and Guidance

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

Clarification

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

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

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

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

Applicability

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

Implementation

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

Type of Review

A preimplementation review will be performed.

Documentation Required

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

Technical Specification Changes Required

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

Reference

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

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

Position

Several components of the emergency core-cooling (ECC) systems are permitted by technical specifications to have substantial outage times (e.g., 72 hours for one diesel-generator; 14 days for the HPCI system). In addition, there are no cumulative outage time limitations for ECC systems. Licensees should submit a report detailing outage dates and lengths of outages for all ECC systems for the last 5 years of operation. The report should also include the causes of the outages (i.e., controller failure, spurious isolation).

Changes to Previous Requirements and Guidance

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

Clarification

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

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

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

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

Applicability

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

Implementation

Licensees should submit detailed report by January 1, 1981.

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

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

Technical Specification Changes Required

Changes depend on results of the licensee study.

References

NUREG-0626, Recommendation A.6

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

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

Position

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

Changes to Previous Requirements and Guidance

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

Clarification

No further clarification is required.

Applicability

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

Implementation

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

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

Type of Review

A preimplementation review of modifications will be performed.

Documentation Required

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

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

Technical Specification Changes Required

Changes to technical specifications will be required.

Reference

NUREG-0626, Recommendation A.7

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

II.K.3.19 INTERLOCK ON RECIRCULATION PUMP LOOPS

Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

No clarification is required.

Applicability

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

Implementation

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

Type of Review

A postimplementation review of modifications will be performed.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will be required.

Reference

NUREG-0626, Recommendation A.8

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

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

Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

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

Applicability

This requirement applies only to Big Rock Point.

Implementation

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

Technical Specification Changes Required

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

Reference

NUREG-0626, Recommendation A.9

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

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

Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

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

Applicability

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

Implementation

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

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

Type of Review

A preimplementation review will be performed.

Documentation Required

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

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

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

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0626, Recommendation A-10

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

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

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

Position

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

Changes to Previous Requirements and Guidance

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

Clarification

No further clarification is required at this time.

Applicability

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

Implementation

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

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

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

Type of Review

A postimplementation review of modifications will be performed.

Documentation Required

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

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

Technical Specification Changes Required

Changes to technical specifications will be required.

Reference

NUREG-0626, Recommendation B.1

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

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

Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

No clarification is required.

Applicability

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

Implementation

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

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

Type of Review

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

Documentation Required

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

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

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

Technical Specification Changes Requied

Changes to technical specifications will be required.

Reference

NUREG-0626, Recommendation B.3

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

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

Position

The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating-current (ac) power for at least 2 hours. Adequacy of the seal design should be demonstrated.

Changes to Previous Requirements and Guidance

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

Clarification

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

Application

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

Implementation

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

Type of Review

A preimplementation review of modifications will be performed.

Documentation Required

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

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

Technical Specification Changes Required

Changes to technical specifications will not be required.

Reference

NUREG-0626, Recommendation B.4

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

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

Position

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

Changes to Previous Requirements and Guidance

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

Clarification

No further clarification is required at this time.

Applicability

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

Implementation

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

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

Type of Review

A postimplementation review for operating reactors will be performed.

Documentation Required

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

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0626, Recommendation B.6

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

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

Position

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

Changes to Previous Requirements and Guidance

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

Clarification

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

Applicability

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

Implementation

This requirement shall be completed by January 1, 1982.

Type of Review

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

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will be required.

Reference

NUREG-0626, Section A.2-15

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

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

Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

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

Applicability

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

Implementation

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will not be required.

Reference

NUREG-0626, Recommendation B.13

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

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

Position

The analysis methods used by nuclear steam supply system (NSSS) vendors and/or fuel suppliers for small-break loss-of-coolant accident (LOCA) analysis for compliance with Appendix K to 10 CFR Part 50 should be revised, documented, and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT Test and Semiscale Test facilities.

Changes to Previous Requirements and Guidance

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

Clarification

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

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

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

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

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

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

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

Applicability

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

Implementation

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

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

Type of Review

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

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

Documentation Required

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

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

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

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

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0565, Recommendation 2.2.2a

NUREG-0611, Recommendation 3.2.1a

NUREG-0623

NUREG-0626, Recommendation A.12

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

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

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

Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

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

Applicability

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

Implementation

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

Type of Review

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

Documentation Required

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

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

Technical Specification Changes Required

Changes to technical specifications are to be determined.

References

NUREG-0565, Recommendation 2.2.2.b

NUREG-0611, Recommendation 3.2.1.b

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

NUREG-0635, Recommendation 3.2.1.b

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

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

Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

No further clarification is required at this time.

Applicability

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

Implementation

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

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

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

Technical Specification Changes Required

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

Reference

NUREG-0626, Recommendation A.14

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

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

Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

No further clarification is required at this time.

Applicability

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

Implementation

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

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

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

Technical Specification Changes Required

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

Reference

NUREG-0626, Recommendation A.15

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

Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous position or requirements.

Clarification

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

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

Applicability

This requirement applies to all operating boiling-water reactors.

Implementation

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will not be required.

Reference

NUREG-0626, Recommendation A.5

III.A.1.2 UPGRADE EMERGENCY SUPPORT FACILITIES

Additional clarification will be provided in the near future.

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

Position

Each nuclear facility shall upgrade its emergency plans to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Specific criteria to meet this requirement is delineated in NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparation in Support of Nuclear Power Plants."

Changes to Previous Requirements and Guidance

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

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

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

Clarification

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

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

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

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

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

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

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

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

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

Applicability

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

Implementation

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

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

(1) a. January 2, 1981

b. Submittal of radiological emergency response plans

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

(2) a. March 1, 1981

b. Submittal of implementing procedures

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

(3) a. April 1, 1981

b. Implementation of radiological emergency response plans

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

Alternative to item (3) requiring compensating actions:

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

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

(i) if only element 1 <u>or</u> element 2 is in use:

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

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

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

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

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

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

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

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

(4) a. March 1, 1982

b. Installation of Emergency Response Facility hardware and software

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

(5) a. July 1, 1982

b. Full operational capability of milestone 4.

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

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

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

b. Mandatory review of the DCM by the licensee

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

(7) a. September 1, 1982

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

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

(8) a. June 1, 1983

b. Full operational capability of the Class B model

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

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

Type of Review

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

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-75/087

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

NUREG-0696

Regulatory Guide 1.23, Proposed Revision 1

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

Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate leak reduction

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

(b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

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

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

Residual heat removal (RHR)

Containment spray recirculation

High-pressure injection recirculation

Containment and primary coolant sampling

Reactor core isolation cooling

Makeup and letdown (PWRs only)

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

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

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

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

Applicability

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

Implementation

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

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.6.a

NUREG-0660, Item III.D.1.1

NUREG-0694, Part 2

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

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

Position

(1) Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

(2) Each applicant for a fuel-loading license to be issued prior to January 1, 1981 shall provide the equipment, training, and procedures necessary to accurately determine the presence of airborne radioiodine in areas within the plant where plant personnel may be present during an accident.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

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

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

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

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

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

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

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

APPLICABILITY

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

IMPLEMENTATION

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0578 Recommendation 2.1.8.c

NUREG-0660, Item III.D.3.3

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

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

III.D.3.4 CONTROL-ROOM HABITABILITY REQUIREMENTS

Position

In accordance with Task Action Plan item III.D.3.4 and control room habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50).

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

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

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

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

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

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

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

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

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

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

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

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

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

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

Applicability

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

Implementation

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0660, Item III.D.3.4.

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

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

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

(2) Control-room characteristics

   (a) air volume control room

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

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

   (d) infiltration leakage rate

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

   (f) closest distance between containment and air intake

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

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

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

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

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

   (l) bottled air supply (hours supply)

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

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

   (o) potassium iodide drug supply

(3) Onsite storage of chlorine and other hazardous chemicals

   (a) total amount and size of container

   (b) closest distance from control-room air intake

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

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

    (b) distance from control room

    (c) quantity of hazardous chemicals in one container

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

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

    (a) chlorine detection system

    (b) control-room emergency filtration system including the capability to maintain the control-room pressurization at 1/8-in. water gauge, verification of isolation by test signals and damper closure times, and filter testing requirements.

APPENDIX A

KEY TO REFERENCES

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

WASH-1400 (NUREG-75/014), "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," Executive Summary, Main Report, Appendices 1-11, U.S. Nuclear Regulatory Commission, December 1975.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

10/10/79	Letter from D. G. Eisenhut, NRC, to All Power Reactor Licensees, dated October 10, 1979, Subject: Emergency Planning.
10/17/79	Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Power Plants, dated October 17, 1979, Subject: Radioactive Release at North Anna Unit 1 and Lessons Learned.
10/30/79	Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979, Subject: Discussion of Lessons Learned Short-Term Requirements.
11/7/79	Letter from R. W. Reid, NRC, to All B&W Operating Plants, dated Novemeber 7, 1979, Subject: Request for Additional Information - BAW Report 1564, "Integrated Control System Reliability Analysis."
7/31/80	Letter from D. G. Eisenhut, NRC, to All Licensees and Applicants, dated July 31, 1980, Subject: Interim Criteria for Shift Staffing.
11/9/79	Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, dated November 9, 1979, Subject: Discussion of Lessons Learned Short-Term Requirements.
11/21/79	Letter from R. W. Reid, NRC, to All B&W Operating Plants, dated November 21, 1979, Subject: Request for Additional Information on Small-Break Loss-of-Coolant Accident.
12/20/79	Letter from R. W. Reid, NRC, to All B&W Licensees, dated December 20, 1979, Subject: Preliminary Design Approval for the Safety-Grade Anticipatory Reactor Trip (ART) on Loss-of-Feedwater and Turbine Trip.
1/9/80	Letter from R. W. Reid, NRC to All B&W Operating Plants, dated January 9, 1980, Subject: Concern for Voiding During Transients on B&W Plants.
3/10/80	Letter from D. F. Ross, Jr., NRC, to All Pending <u>W</u> and C-E License Applicants, dated March 10, 1980, Subject: Actions Required from Operating License Applicants of Nuclear Steam Supply Systems Designed by <u>W</u> and C-E Resulting from the NRC Bulletins and Orders Task Force Review Regarding TMI-2 Accident.
3/28/80	Letter from H. R. Denton, NRC, to All Power Reactor Applicants and Licensees, dated March 28, 1980, Subject: Qualifications of Reactor Operators.
4/24/80	Letter from D. F. Ross, Jr., NRC, to All Pending B&W License Applicants, dated April 24, 1980, Subject: Actions Required from Operating License Applicants of Nuclear Steam Supply Systems Designed by B&W Resulting from the NRC Bulletins and Orders Task Force Review Regarding TMI-2 Accident.

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

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

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

Commission Order (CLI-80-21)

NRC Regulation (45 FR 55401-55413)

Inspection and Enforcement documents

Regulatory Guides

Standard Review Plan

Technical Specifications

Branch Technical Position

Staff Interim Position

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

ANSI Standards

IEEE Standards

Code of Federal Regulations

APPENDIX B

DESIGN AND QUALIFICATION CRITERIA FOR ACCIDENT MONITORING INSTRUMENTATION

Applicability

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

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

Criteria

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

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

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

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

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

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

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

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

1.28 "Quality Assurance Program Requirements (Design & Construction)

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

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

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

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

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

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

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

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

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

   (a) By perturbing the monitored variable

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

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

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

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

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

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

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

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

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

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

APPENDIX C

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

Recommendations for Position Description,
Qualifications, Education and Training

THE
INSTITUTE OF NUCLEAR
POWER OPERATIONS

Revision 0
April 30, 1980

FOREWORD

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

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

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

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

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

TABLE OF CONTENTS

1. DEFINITIONS

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

2. INTRODUCTION

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

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

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

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

3. OBJECTIVE

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

4. POSITION DESCRIPTION

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

Function

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

General Qualifications

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

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

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

General Duties

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

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

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

Typical Responsibilities

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

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

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

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

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

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

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

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

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

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

(10) Review abnormal and emergency procedures.

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

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

Accountability

The STA is accountable for the following end results:

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

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

5. GENERAL EDUCATION AND EXPERIENCE

5.1 EDUCATION AND TRAINING

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

5.2 EXPERIENCE

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

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

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

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

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

5.3 ABSENCES FROM STA DUTIES

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

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

6. EDUCATION AND TRAINING REQUIREMENTS

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

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

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

6.1 EDUCATION

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

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

Physics

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

TOTAL 270

6.1.2 College Level Fundamental Education

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

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

6.2 APPLIED FUNDAMENTALS - PLANT SPECIFIC

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

Subject/Topics	Contact Hours
Plant Specific Reactor Technology (including core physics data)	
Plant Chemistry and Corrosion Control	
Reactor Instrumentation and Control	
Reactor Plant Materials	
Reactor Plant Thermal Cycle	
TOTAL	120

6.3 MANAGEMENT/SUPERVISORY SKILLS

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

6.4 PLANT SYSTEMS

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

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

6.5 ADMINISTRATIVE CONTROLS

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

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

TOTAL	80

6.6 GENERAL OPERATING PROCEDURES

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

6.7 TRANSIENT/ACCIDENT ANALYSIS AND EMERGENCY PROCEDURES

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

6.8 SIMULATOR TRAINING

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

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

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

PWR Simulator Exercises

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

BWR Simulator Exercises

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

6.9 ANNUAL REQUALIFICATION TRAINING

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

NRC FORM 335 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER *(Assigned by DDC)*
NUREG-0737

4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)*
Clarification of TMI Action Plan Requirements

2. *(Leave blank)*

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)

5. DATE REPORT COMPLETED

MONTH	YEAR
October	1980

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Office of Nuclear Reactor Regulation
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

DATE REPORT ISSUED

MONTH	YEAR
November	1980

6. *(Leave blank)*

8. *(Leave blank)*

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Office of Nuclear Reactor Regulation
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

10. PROJECT/TASK/WORK UNIT NO.

11. CONTRACT NO.

13. TYPE OF REPORT
Technical Report

PERIOD COVERED *(Inclusive dates)*

15. SUPPLEMENTARY NOTES

14. *(Leave blank)*

16. ABSTRACT *(200 words or less)*

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

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT

19. SECURITY CLASS *(This report)*
Unclassified

21. NO. OF PAGES

20. SECURITY CLASS *(This page)*
Unclassified

22. PRICE
$