Wisconsin Electric POWER COMPANY
231 W. MICHIGAN, P.O. BOX 2046, MILWAUKEE, WI 53201

April 26, 1982

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Dear Mr. Denton:

DOCKET NOS. 50-266 AND 50-301
RESPONSE TO NUREG-0737
UPDATE TO SCHEDULE REQUIREMENTS
AND IMPLEMENTATION STATUS
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

This letter provides additional information related to schedule requirements and implementation status of NUREG-0737, "Clarification of TMI Action Plan Requirements", items as requested in Mr. Darrell G. Eisenhut's letter, dated March 17, 1982. In order to provide a consistent update for all NUREG-0737 items, the information is presented as Revision 3 to our Schedule Table and Notes which provides schedule and implementation status, as of April 26, 1982. Enclosure 1 of Mr. Eisenhut's letter identified specific items for which a firm schedule needs to be established. By referring to our Schedule Table and Notes, Revision 3 dated April 26, 1982, for the item of interest, the schedule for that item, as established by Wisconsin Electric Power Company, can be found.

Your review of this response should be made with reference to prior Wisconsin Electric submittals. Each item is addressed relative to that item as completed, updated, or otherwise modified since our September 14, 1981 response. We have not repeated the pertinent notes referenced in the Schedule Table for those items whose status has not changed since our December 23, 1980, March 31 and September 14, 1981 submittals.

The total scope of work involved in the TMI backfit effort is not always apparent from a list of schedule commitments. It does not show the effort involved to properly plan and schedule the implementation of all the items. Our September 14, 1981 submittal listed some of the major projects that are being implemented at the same time in order to properly accomplish the TMI backfit modifications. We emphasized that arbitrarily scheduling one item ahead of another may disrupt the overall plan, cause wasted effort, and possibly result in delays to other items or final implementation of that item. A careful review of our

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commitments and the magnitude of the work in progress will demonstrate that we are attempting to implement NUREG-0737 requirements in a very careful and deliberate manner that meaningfully improves the operation of the plant.

Representatives from Wisconsin Electric met with members of your Staff on October 1, 1981 to discuss an integrated approach to completing the items required by NUREG-0737. The approach that was discussed would allow each utility to determine the best schedule for their plant to implement the required items. This approach was consistent with our September 14, 1981 submittal and Mr. Eisenhut's March 17, 1982 letter is being treated as a confirmation of this approach. The attached Schedule Table and Notes represents the implementation schedule we feel we can meet.

We would be pleased to respond to any questions you may have regarding the schedule or the accompanying explanatory notes.

Very truly yours,

CW Fay

C. W. Fay

Assistant Vice President

Enclosures

Subscribed and sworn to before me
This 26th day of April, 1982.

Dorothy Fleischmann
Notary Public, State of Wisconsin

My Commission Expires: July 1, 1984.

Copy to NRC Resident Inspector

RESPONSE TO NUREG-0737

POST-TMI REQUIREMENTS

FOR OPERATING PLANTS

Point Beach Nuclear Plant, Units 1 and 2
Docket Nos. 50-266 and 50-301

Schedule Table and Notes
Revision 3 - April 26, 1982

KEY TO SCHEDULE TABLE

N.A. = Schedule Not Applicable to Point Beach Nuclear Plant (PBNP)

N.R. = Not Required

TBD = To Be Determined at a later date per the Remarks

* = Schedule is based of delivery of equipment on schedule

Marginal Revision Notation:

None = Original issue dated December 23, 1980 (Reference 12)

1 = Revision 1 dated March 31, 1981 (Reference 23)

2 = Revision 2 dated September 14, 1981 (Reference 47)

3 = Revision 3 dated April 26, 1982 (updated in this submittal)

SCHEDULE TABLE
POST-TMI REQUIREMENTS FOR OPERATING REACTORS

Clarification Item	Shortened Title	Description	NRC Implementation Schedule	PBNP Applicability	PBNP Schedule	Remarks	Rev.
I.A.1.1	Shift Technical Advisor	1. On duty	1/1/80	Yes	Completed	On duty since 1/1/80 - Reference 1 and 12a 25a, and 59	3
		2. Tech Specs	12/15/80	Yes	Completed	Reference 12a and 15	
		3. Trained per LL Cat B	1/1/81	Yes	Completed	Note I.A.1.1.3 and Reference 25a	2,3
		4. Describe long-term program	1/1/81	Yes	Completed	Reference 17	
I.A.1.2	Shift Supervisor Responsibilities	Delegate non-safety duties	1/1/80	Yes	Completed	References 1 and 25a	3
I.A.1.3	Shift Manning	1. Limit overtime	11/1/80	Yes	Completed	PBNP Approved Procedure 4.3, Operations Division Personnel Assignments and Scheduling, Rev. 0, and References 25a and 60	3
		2. Min. Shift Crew	7/1/82	Yes	TBD	References 4b, 7, 54, 62, 66, and 70	3
I.A.2.1	Immediate Upgrading of RO and SRO Training and Qualifications	1. SRO Experience	5/1/80	Yes	Completed		
		2. SROs be ROs 1 yr.	12/1/80	Yes	Completed		2
		3. Three mo. training on shift	8/1/80	Yes	Completed	Note I.A.2.1.1/4 and Reference 9a	
		4. Modify training	8/1/80	Yes	Completed	Reference 25a	3
		5. Facility Certification	5/1/80	Yes	Completed	Note I.A.2.1.5	
I.A.2.3	Administration of Training Programs	Instructors Complete SRO Exam	8/1/80	Yes	Completed	Note I.A.2.3	
I.A.3.1	Revise Scope and Criteria for Licensing Exams	1. Increase scope	5/1/80	Yes	Completed		
		2. Increase passing grade	5/1/80	Yes	Completed		
		3. Simulator exam	6/1/80	N.A.	---	Note I.A.3.1.3 and Reference 68	3

SCHEDULE TABLE
POST-TMI REQUIREMENTS FOR OPERATING REACTORS

Clarification Item	Shortened Title	Description	NRC Implementation Schedule	PBNP Applicability	PBNP Schedule	Remarks	Rev.
I.A.1.1	Shift Technical Advisor	1. On duty	1/1/80	Yes	Completed	On duty since 1/1/80 - Reference 1 and 12a 25a, and 59	3
		2. Tech Specs	12/15/80	Yes	Completed	Reference 12a and 15	
		3. Trained per LL Cat B	1/1/81	Yes	Completed	Note I.A.1.1.3 and Reference 25a	2,3
		4. Describe long-term program	1/1/81	Yes	Completed	Reference 17	
I.A.1.2	Shift Supervisor Responsibilities	Delegate non-safety duties	1/1/80	Yes	Completed	References 1 and 25a	3
I.A.1.3	Shift Manning	1. Limit overtime	11/1/80	Yes	Completed	PBNP Approved Procedure 4.3, Operations Division Personnel Assignments and Scheduling, Rev. 0, and References 25a and 60	3
		2. Min. Shift Crew	7/1/82	Yes	TBD	References 4b, 7, 54, 62, 66, and 70	3
I.A.2.1	Immediate Upgrading of RO and SRO Training and Qualifications	1. SRO Experience	5/1/80	Yes	Completed		
		2. SROs be ROs 1 yr.	12/1/80	Yes	Completed		2
		3. Three mo. training on shift	8/1/80	Yes	Completed	Note I.A.2.1.1/4 and Reference 9a	
		4. Modify training	8/1/80	Yes	Completed	Reference 25a	3
		5. Facility Certification	5/1/80	Yes	Completed	Note I.A.2.1.5	
I.A.2.3	Administration of Training Programs	Instructors Complete SRO Exam	8/1/80	Yes	Completed	Note I.A.2.3	
I.A.3.1	Revise Scope and Criteria for Licensing Exams	1. Increase scope	5/1/80	Yes	Completed		
		2. Increase passing grade	5/1/80	Yes	Completed		
		3. Simulator exam	6/1/80	N.A.	---	Note I.A.3.1.3 and Reference 68	3

Clarification Item	Shortened Title	Description	NRC Implementation Schedule	PBNP Applicability	PBNP Schedule	Remarks	Rev.
I.C.1	Short-Term Accident and Procedures Review	1. SB LOCA	6/1/80	Yes	Completed	Reference 25a	3
		2. Inadequate Core Cooling					
		a. Reanalyze and propose guidelines	1/1/81	Yes	Completed	Generic procedures already submitted to NRC - Reference 36, 53, and 64	3
		b. Revise procedures	TBD	Yes	TBD	Note I.C.1	3
		3. Transients and accidents					
		a. Reanalyze and propose guidelines	1/1/81	Yes	Completed	Same as I.C.1.2.a	3
		b. Revise procedures	TBD	Yes	TBD	Note I.C.1	3
I.C.2	Shift and Relief Turnover Procedures	Implement shift turnover checklist	1/1/80	Yes	Completed	References 9b and 25a	3
I.C.3	Shift-Supervisor Responsibility	Clearly define superv and oper responsibilities	1/1/80	Yes	Completed	Reference 25a	3
I.C.4	Control-Room Access	Establish authority limit access	1/1/80	Yes	Completed	References 9b and 25a	3
I.C.5	Feedback of Operating	Licensee to implement procedures	1/1/81	Yes	Completed	PBNP Administrative Procedure 3.15.7, Rev. 0, approved 12/19/80, "Procedure for Feedback of Operating Experience to Plant Staff" and References 25a and 54	3
I.C.6	Verify Correct Performance of Operating Activities	Revise performance procedures	1/1/81	Yes	Completed	PBNP Administrative Procedure 4.13, Rev. 9, effective 6/20/80, "Equipment Isolation Procedure" and References 25a and 54	3
I.D.1	Control Room Design Reviews	Preliminary assessment and schedule for correcting deficiencies	TBD	Yes	TBD	Note I.D.1	2

Clarification Item	Shortened Title	Description	NRC Implementation Schedule	PBNP Applicability	PBNP Schedule	Remarks	Rev.
I.D.2	Plant Safety Parameter Display Console	1. Description	TBD	Yes	Completed	Note I.D.2	3
		2. Installed	TBD	Yes	12/1/83* (Projected)		3
		3. Fully implemented	TBD	Yes	3/1/84* (Projected)		3
II.B.1	Reactor Coolant System Vents	1. Design vents	7/1/81	Yes	Completed	References 25a and 47	3
		2. Install Vents (LL Cat B)	7/1/82	Yes	1/1/84	Note II.B.1	3
		3. Procedures	1/1/82	Yes	3/1/83		
II.B.2	Plant Shielding	1. Review designs	1/1/80	Yes	Completed	Reference 25a	3
		2. Plant modifications (LL Cat B)	1/1/82	Yes	1/1/84	Note II.B.2.2 and References 25a, 47 and 68	3
		3. Equipment qualification	6/30/82	Yes	1/1/84	Note II.B.2.3 and Reference 68	3
II.B.3	Post Accident Sampling	1. Interim system	1/1/80	Yes	Completed	Reference 25a	
		2. Plant modifications (LL Cat B)	1/1/82	Yes	6/1/83*	Note II.B.3 and References 29 and 68 Reference 29	3
II.B.4	Training for Mitigating Core Damage	1. Develop training program	1/1/81	Yes	Completed		1
		2. Implement program					
		a. Initial	4/1/81	Yes	Completed	References 23, 47 and 68	3
		b. Complete	10/1/81	Yes	Completed		
II.D.1	Relief and Safety Valve Test Requirements	1. Submit program	1/1/80	Yes	Completed	References 10a and 50	3
		2. RV and SV Testing (LL Cat B)					
		a. Complete testing	4/1/82	Yes	Completed	References 50 and 71	3
		b. Plant-specific report	7/1/82	Yes	7/1/82	References 50 and 71	3
		3. Block-Valve testing	7/1/82	Yes	N.A.	Note II.D.1.3	

Clarification Item	Shortened Title	Description	NRC Implementation Schedule	PBNP Applicability	PBNP Schedule	Remarks	Rev.
II.D.3	Valve Position Indication	1. Install direct indications of valve valve position	1/1/80	Yes	Completed	Note II.D.3.1 and References 9b and 25a	3
		2. Tech Specs	12/15/80	Yes	Completed	Reference 15	1
II.E.1.1	Auxiliary Feedwater System Evaluation	1. Short term	7/1/81	Yes	TBD	Note II.E.1.1 and References 40, 49 and 67	3
		2. Long term	1/1/82	Yes	TBD		
II.E.1.2	Auxiliary Feedwater System Initiation and Flow	1. Initiation					
		a. Control grade	6/1/80	Yes	N.A.	Reference 25a	3
		b. Safety grade	7/1/81	Yes	Original Plant Design	References 1, 2, 3, 12a, and 25a	3
		2. Flow Indication					
		a. Control grade	1/1/80	Yes	Completed	Reference 25a	3
		b. LL A Tech Specs	12/15/80	Yes	Completed	Reference 12a and 15	1
		c. Safety grade	7/1/81	Yes	Completed	References 44a and 68	3
II.E.3.1	Emergency Power for Pressurizer Heaters	1. Upgrade power	1/1/80	Yes	Original Plant Design	References 1, 2, 3 12a, and 25a	3
		2. Tech Specs	12/15/80	Yes	Completed	Reference 15	1
II.E.4.1	Dedicated Hydrogen Penetrations	1. Design	1/1/80	Yes	Original Plant Design	References 1, 2, 3, 25a and 48	3
		2. Install	7/1/81	Yes	N.A.		
II.E.4.2	Containment Isolation Dependability	1-4. Imp. diverse isolation	1/1/80	Yes	1/1/84	Note II.E.4.2.1/4 and Reference 44a	3
		5. Cntmt pressure setpoint					
		a. Specify pressure	1/1/81	Yes	Completed	References 26, 42, 44a and 68	3
		b. Modifications	7/1/81	Yes	N.A.		
		6. Cntmt purge valves	1/1/81	Yes	Completed	Administratively closed - References 44a and 45	3

Clarification Item	Shortened Title	Description	NRC Implementation Schedule	PBNP Applicability	PBNP Schedule	Remarks	Rev.
II.E.4.2	Containment Isolation Dependability	7. Radiation signal on purge valves	7/1/81	Yes	Original Plant Design	Reference Point Beach FFDSAR Section 4.2 and Figure 5.2-8	
		8. Tech Specs	12/15/80	Yes	Completed	References 15 and 45	3
II.F.1	Accident Monitoring	1. Noble gas monitor	1/1/82	Yes	12/1/82*	Note II.F.1.1/2 and References 57, 65, and 68	3
		2. Iodine/particulate sampling	1/1/82	Yes	12/1/82*		3
		3. Containment high-range radiation monitor	1/1/82	Yes	6/1/83*	Note II.F.1.3 and References 61 and 68	
		4. Containment pressure	1/1/82	Yes	6/1/83*	Note II.F.1.4 and Reference 68	3
		5. Containment water level	1/1/82	Yes	6/1/83*	Note II.F.1.5 and Reference 68	3
		6. Containment hydrogen	1/1/82	Yes	6/1/83*	Note II.F.1.6 and Reference 68	3
II.F.2	Instrumentation of Detection of Inadequate Core Cooling	1. Subcool meter	1/1/80	Yes	Completed	Note II.F.2.1 and Reference 25a	1,2,3
		2. Tech Spec (LL Cat A)	12/15/80	Yes	Completed	Reference 15	
		3. Install level instruments (LL Cat B)	1/1/82	Yes	1/1/84*	Note II.F.2.3 and Reference 52	3
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	1. Upgrade to emerg sources	1/1/80	Yes	Original Plant Design	References 12a and 25a	3
		2. Tech Specs	12/15/80	Yes	Completed	Reference 15	1
II.K.1	IE Bulletins	79-05, -06, -08	Bulletin specific	Yes	Completed	Reference 33	2
II.K.2	Orders on B&W Plants	8. Upgrade AFW system	See II.E.1.1	N.A.	---		
		9. FEMA on ICS	8/17/79	N.A.	---		
		10. Safety-grade trip	7/1/81	N.A.	---		
		11. Operator training, drilling	Complete	N.A.	---		

Clarification Item	Shortened Title	Description	NRC Implementation Schedule	Applicability	PBNP Schedule	Remarks	Rev.
II.K.2	Orders on B&W Plants	13. Thermal-mechanical report	1/1/82	Yes	Completed	Reference 58	3
		14. Lift frequency of PORVs and SVs	See II.K.3.7				
		15. Effects of slug flow on OTSGs	Completed	N.A.	---		
		16. RCP seal damage	Completed	N.A.	---		
		17. Voiding in RCS	a. Complete	N.A.	---		
			b. 1/1/82	Yes	Completed	Note II.K.2.17 and References 25b and 53	3
		19. Benchmark analysis of seq. AFW flow	a. Complete	N.A.	---		2
			b. 1/1/82	Eliminated	N.A.	Reference 34	
		20. System response to SB LOCA	Complete	N.A.	---		
II.K.3	Final recommendations, B&O Task Force	1. Auto PORV isolation					
		a. Design	7/1/81	Yes	N.R.		
		b. Test/install	1st refueling 6 mos after staff approval	Yes	N.R.	Note II.K.3.1 and References 4 and 44a	3
		2. Report on PORV failures	1/1/81	Yes	Completed	Note II.K.3.2 and Reference 19	3
		3. Reporting SV and RV failures and challenges	1/1/81	Yes	Completed	Note II.K.3.3 and Reference 69	3
		5. Auto trip on RCPS					
		a. Propose modifications	7/1/81	Yes	TBD	Note II.K.3.5 and Reference 31	3
		b. Modify	3/1/82	Yes	TBD		
		7. Eval of PORV opening probability	1/1/81	N.A.	---		
		9. PID controller	1/1/81	Yes	Completed	Controller change made upon initial notification by vendor prior to TMI-2 (References 4, 25a, and 43)	3
		10. Proposed anticipatory trip modifications	Plant specific	Yes	Original Plant Design	References 4 and 43	

Clarification Item	Shortened Title	Description	NRC Implementation Schedule	PBNP Applicability	PBNP Schedule	Remarks	Rev.
II.K.3 (Continued)	Final Recommendations, B&O Task Force	11. Justify use of certain PORV	Plant specific	Yes	N.A.	As part of the original Plant design (different from TMI-2), Point Beach has Copes-Vulcan PORVs which corresponds to the Westinghouse data base, and, thus, no justification is needed.	
		12. Anticipatory trip on turbine trip					
		a. Confirmation or proposed modifications	1/1/81	Yes	Original Plant Desgn	Reactor trip caused by turbine trip bypassed below 50% power as detected by the power range detectors. (References 43 and 44a)	3
		b. Modify	1st refuel 60 mo. after staff approval	N.A.	---		
		13. HPCI & RCIC init levels					
		a. Analysis	1/1/81	N.A.	---		
		b. Modify	7/1/81	N.A.	---		
		14. Iso condenser isol modification	1/1/82	N.A.	---		
		15. Isolation of HPCI and RCIC modification	7/1/81	N.A.	---		
		16. Challenges and failures to relief valves					
		a. Study	4/1/81	N.A.	---		
		b. Modify	1st refueling or 1 yr after approval	N.A.	---		
		17. ECCS system outages	1/1/81	Yes	Completed	Note II.K.3.17 and Reference 23	1,2

Clarification Item	Shortened Title	Description	NRC Implementation Schedule	PBNP Applicability	PBNP Schedule	Remarks	Rev.
II.K.3 (Continued)	Final Recommendations, B&O Task Force	18. ADS actuation					
		a. Study	4/1/81	N.A.	---		
		b. Propose mods	4/1/82	N.A.	---		
		c. Modification	1st refuel 6 mo after staff approval	N.A.	---		
		19. Interlock recirc pump modification	7/1/81	N.A.	---		
		20. Loss of SVC	7/1/81	N.A.	---		
		21. Restart of CCS and LPCI					
		a. Design	1/1/81	N.A.	---		
		b. Modification	1st refueling 60 mo after staff approval	N.A.	---		
		22. RCIC suction					
		a. Verify procedures	1/1/81	N.A.	---		
		b. Modification	1/1/82	N.A.	---		
		24. Space cooling for HPCI/RCIC modifications	1/1/82	N.A.	---		
		25. Power on pump seals					
		a. Propose mods	1/1/82	Yes	None		3
		b. Modification	7/1/82	Yes	N.A.	Note II.K.3.25	
		27. Com ref. level	7/1/81	N.A.	---		
		28. Qual of ADS accumulators	1/1/82	N.A.	---		
		29. Performance of isolation condensers	4/1/81	N.A.	---		
		30. SB LOCA methods					
		a. Schedule outline	11/15/80	Yes	TBD		
		b. Model	1/1/82	Yes	TBD	Note II.K.3.30	
		c. New analyses	1/1/83 or 1 yr after staff approval	Yes	TBD		

Clarification Item	Shortened Title	Description	NRC Implementation Schedule	PBNP Applicability	PBNP Schedule	Remarks	Rev.
II.K.3 (Continued)	Final Recommendations, B&O Task Force	31. Compliance with CFR 50.46	1/1/83 or 1 yr after staff approval	Yes	TBD	Note II.K.3.31	
		40. RCP seal damage	See II.K.2.16	N.A.	---		
		43. Effects of slug flow	See II.K.2.15	N.A.	---		
		45. Manual depressurization	1/1/81	N.A.	---		
		46. Michelson concerns	Completed	N.A.	---		
		57. Manual act of ADS	TBD	N.A.	---		
III.A.1.1	Emergency Preparedness, Short Term	Short-term improvements	Completed	Yes	Completed		
III.A.1.2	Upgrade Emergency Support Facilities	1. Interim TSC, OSC and EOF	1/1/80	Yes	Completed	Note III.A.1.2 and References 9b and 25a	3
		2. Design	Reference 15b	Yes	Completed	Reference 28	2
		3. Modifications	Reference 15b	Yes	Ref. 56	Reference 13, 28, 38, and 56	3
III.A.2	Emergency Preparedness	1. Upgrade emergency plans to App. E, 10 CFR 50	4/1/81	Yes	Completed	Note III.A.2.1 and Reference 12b, 17a, and 55	3
		2. Meteorological data	6/1/83	Yes	Ref. 56	Note III.A.2.2 - References 13, 28, 38 and 56	3
III.D.1.1	Primary Coolant Outside Containment	1. Leak reduction	Completed	Yes	Completed	Currently changing to a yearly testing schedule for both units coincident with refueling outages (References 1, 2, 3, and 25a)	3
		2. Tech Specs	12/15/80	Yes	Completed	Reference 15	1
III.D.3.3	Inplant Iodine Monitoring	1. Provides means to determine presence of radioiodine	Completed	Yes	Completed	References 12a and 25a	3

Clarification Item	Shortened Title	Description	NRC Implementation Schedule	PBNP Applicability	PBNP Schedule	Remarks	Rev.
		2. Modifications to accurately measure I_2	1/1/81	Yes	Completed	Note III.D.3.3 - Reference 12a, 25a, and 63	3
III.D.3.4	Control Room Habitability	1. Review	1/1/81	Yes	Completed	Note III.D.3.4	1
		2. Modification	1/1/83	Yes	1/1/83	Note III.D.3.4	

I.A.3.1.3 REVISED SCOPE AND CRITERIA FOR LICENSING EXAMS - SIMULATOR EXAMS

Since Point Beach does not have a simulator, the date of October 1, 1981 is the effective date for initiation of the requirement for applicants to be examined on a simulator. It is our intention to utilize other plant simulators to meet this requirement in the future. Under these conditions, the role of the simulator in the examination process must be more clearly defined by the NRC. It is our assumption that the major use of a simulator will be in the area of transient and accident recognition. Other examination requirements related to the control room, e.g., control board familiarity, can be performed at the plant in a manner similar to the present examinations. Therefore, at the end of May 1982, Wisconsin Electric's first group of license candidates since the effective date of October 1, 1982, will be examined on a Westinghouse Nuclear Training Center simulator at Zion, Illinois.

I.C.1 SHORT-TERM ACCIDENT AND PROCEDURES REVIEW

The program to achieve compliance with NUREG-0737, Item I.C.1, was presented to the NRC in a meeting held on June 18, 1981, and submitted to the NRC by Owners Group letter, OG-61, dated July 7, 1981, R. W. Jurgensen to S. H. Hanauer (Reference 36). The previous program submittal to the NRC (Reference 20) resulted in several basic concerns which could not be easily resolved within the scope of the material submitted. These NRC concerns and the need to better organize the set of emergency procedures resulted in a new configuration which was presented to the NRC at the June 18, 1981 meeting. The new procedure set, referred to as Emergency Response Guidelines (ERGs), was transmitted to the NRC by Owners Group letter, OG-61, dated November 11, 1981 (Reference 53). Representatives of the Westinghouse Owners Group met with the NRC on February 9, 1982 to discuss the submitted ERGs. A summary of the meeting is presented in Reference 64. The requirements for procedure implementation is being formulated by the NRC. When this is made known to the industry, Wisconsin Electric will respond with an appropriate implementation program and schedule.

I.D.2 PLANT SAFETY PARAMETER DISPLAY CONSOLE

Two projects are underway that relate to the safety parameter display console. The first project is a joint effort with eleven other utilities and Quadrex Corporation to design a Safety Assessment System and demonstrate its effectiveness. This group of utilities met with the NRC on May 14, 1981 for the purpose of describing the program and showing some of the detailed information about the project. The presented material was submitted to the NRC via Ward Wogsland to Leo Beltracchi letter dated May 19, 1981 (Reference 26a). The second project deals with the procurement and installation of the hardware system. Wisconsin Electric on August 24, 1981, signed an agreement with Electronics Associates, Inc. (EAI) for the supply of a computer system and associated software on which to implement the safety parameter display console. The description of these two projects was submitted to the NRC as Attachment "C" from Sol Burstein to H. R. Denton letter dated June 1, 1981 (Reference 28). The implementation of the NUREG-0696 requirements and our position on Regulatory Guide 1.97 were also included in that submittal (Reference 28).

The schedule for installation and initial operation of the SPDS is projected to be completed by December of 1983. The system should be fully implemented by March of 1984. The schedule requires a few additional months because the computer system purchased from EAI to implement the SPDS will replace the presently installed plant computer and will be performing all of process computer functions as well as providing the total data base to the Technical Support Center.

II.B.1 REACTOR COOLANT SYSTEM VENTS - DESIGN

The Point Beach reactor coolant system gas venting systems were purchased from Combustion Engineering. The system design is shown in the attached schematic and provides for parallel vent paths from both the reactor vessel head vent and/or the pressurizer steam space to the pressure relief tank (PRT) and/or containment atmosphere. The parallel paths ensure vent capability from both vessels even considering a single failure. Each path is provided with an orifice to limit flow in the event of a downstream leak or break. The orifice size is 7/32" diameter. Leakage through a hole this size (<3/8" diameter which is below that size "corresponding to the definition of a LOCA" [10 CFR 50 Appendix A]) is within the capability of makeup from the charging system. The isolation valves used are pilot-operated Target Rock solenoid valves which fail closed. A solenoid valve in each vent path provides series valving to ensure isolation. Appropriate manual isolation, vent, drain, and test connection valves are provided to permit on-line maintenance, meet all testing requirements and allow flexibility.

Power for the solenoid valves will be obtained from 125V DC safety-grade busses normally supplied by a charger with battery backup. Two busses will be used for the gas vent system solenoid valves with one bus supplying power to one valve in each vessel vent path and one series vent valve. The other bus will supply the other (parallel) vessel vent valve and the other series vent valve. A loss of either power supply does not prevent venting of either vessel.

The connecting piping between the reactor vessel head vent and the pressurizer steam space vent will provide a continuously sloped path which allows for gas (or steam) venting from the vessel to the pressurizer.

Redundant, direct multi-position indication is available for each solenoid valve. One train of indication (both the open and closed position) will be provided, and displayed adjacent to the control switches on the control board, since separation cannot be assured (the magnetic reed switches are in the same solenoid housing).

A wide-range pressure transmitter will be connected to the manifold of piping which connects to each solenoid valve. This will allow monitoring of the system for leaks and during operation. The transmitter signal will be processed in the Foxboro Spec 200 racks. All controls and indications will be in the control room on the ASIP.

Installation of the gas vent system in each unit will require a cold shutdown of the unit for access to the reactor vessel head, refueling cavity, and pressurizer. Removal of the existing vent systems must be considered in the

overall work scheduling since returning to power without a vent system would take long, be more difficult and increase personnel exposure. The construction must, therefore, be scheduled into a refueling outage of sufficient length to accommodate removal, installation, and testing.

Final system design and construction drawings have recently been received from the system vendor which include some changes made to the layout for each unit based on a detailed field review to ensure access and operability of all components. Additional thermal and hydraulic analyses must be performed for the redesign configurations to ensure that the design criteria for the piping and supports are met. Installation of the gas vent systems should thus be able to be completed on Unit 2 by June 1, 1983 and on Unit 1 by January 1, 1984, the next scheduled refueling outages of sufficient length. Full operability of the systems, based on the installation schedule of the ASIP panels, is consistent with this schedule.

Use of the gas vent system is addressed as part of the accident response procedures referenced in I.C.1. These procedures will be completed and available for training and operational review by March 1, 1983, prior to startup of the system on the lead unit.

II.B.2.2. PLANT SHIELDING

The initial post TMI-2 shielding and plant access calculations for Point Beach Nuclear Plant were performed using the source term specified in NUREG-0578 (Reference 1 and Reference 2). Since the clarification in NUREG-0737 changed the assumptions used to develop the source terms, a preliminary re-evaluation was performed to estimate the impact on shielding requirements (References 12, 23, and 47). A minor additional shielding need was identified in the control room habitability study (Reference 16). Conceptual design engineering is nearing completion to determine exact shielding material and thickness requirements for each area requiring shield upgrading. A summary of the status of each of those areas is as follows:

C59 Control Panel

Proposals for the evaluation and design of portable and permanent shielding in the area of the C-59 control panel (References 1 and 2) are still being evaluated. Installation was thus not possible by the January 1, 1982 NRC implementation date. Conceptual design engineering to determine material and thickness requirements will be completed by May 31, 1982. It is anticipated that portable shielding will solve some of the concerns, and purchase orders for these items can be placed this year. The permanent shielding changes must be designed to be structurally compatible with the existing structure. Projected completion of this work is January 1, 1983.

Motor Control Centers

Basic design evaluations were made for relocation of portions of the Unit 1 safety injection lines which could eliminate the major radiological contributor to exposure of adjacent electrical equiment and reduce the dose rates for personnel access in the corridor beneath the existing piping. It was concluded that it is not in the best interest of safety to make a change in an operating

plant to a basic safety system which alters the initial design. Hence, additional lead or concrete shielding will be provided for portions of both the Unit 1 and Unit 2 safety injection lines in the vicinity of the 1B32 and 2B32 motor control centers. Conceptual design engineering to determine material and thickness requirements will be completed by May 31, 1982. Final completion of modifications will be consistent with II.B.2.3, below.

Wall Penetrations

Wall penetrations for piping and electrical runs between the auxiliary and control buildings were also identified as requiring shielding. This was made necessary when implementation of IE Bulletin 80-11 required the removal of concrete block which filled the unused portions of these wall openings. New shielding for these penetrations has been designed and completely installed.

Control Room Windows

A need for shielding over the windows of the control building to reduce the contribution from the theoretical surrounding semi-infinite cloud was recognized in Reference 16. Conceptual design engineering to determine material and thickness requirements will be completed by June 30, 1982. If portable shielding will suffice as anticipated, final purchase orders will be placed by the end of 1982.

II.B.2.3 PLANT SHIELDING - EQUIPMENT QUALIFICATION

Safeguards motor control centers (MCCs) 1B32 and 2B32 located in the auxiliary building are potentially subject to radiation exposure during the ECCS recirculation phase following a design-basis LOCA. The qualification of these MCCs is being addressed by adding shielding near the source to reduce the radiation doses to significantly below damage thresholds for the component materials. The shielding modifications are planned to be completed prior to the proposed new deadline for IE Bulletin 79-01, "Environmental Qualification of Safety-Related Electrical Equipment", (i.e., second refueling outage starting after March 31, 1982). Other equipment is being qualified in accordance with the requirements of IE Bulletin 79-01B.

II.B.3 POST-ACCIDENT SAMPLING

As indicated in previous submittals and reviewed by the NRC, modifications have been completed to permit coolant sampling under accident conditions. While the existing sampling system meets the requirements of NUREG-0737, we have elected to further upgrade the system to a configuration similar to that used by Connecticut Yankee. The new system will improve sampling speed, facilitate mechanical connection, and improve the ease of obtaining accurately diluted samples. While we expect to have the new system installed within twelve months, we believe no firm commitment is required since the current system meets the requirements of NUREG-0737.

Wisconsin Electric currently meets the requirements of NUREG-0737 for chloride analysis capability. A liquid ion chromatograph has been obtained by the Company's Laboratory Services Division and is maintained in a state of readiness

by the Division at Corporate headquarters in Milwaukee. If radiological characteristics of a sample so dictate, the instrument can be transported to Point Beach Nuclear Plant on a virtually immediate basis.

The reciprocal agreement with Kewaunee Nuclear Power Plant for backup analytical laboratory services, including chloride analysis, will be continued.

II.D.3.1 DIRECT INDICATION OF RELIEF AND SAFETY VALVE POSITION

Adapters and lift indicating switch assemblies will be mounted on the pressurizer safety valves to provide direct indication of valve position. The assemblies are being purchased from Crosby for use on Crosby valves. These switch assemblies use magnetically operated reed switches to provide separately powered redundant open, midpoint and closed indications. These indications will be located in the control room.

The lift indicating switch assemblies are not presently qualified but Crosby is expected to qualify the assemblies in the future. The reason the lift indicating switch assemblies will be installed is that they are easier to environmentally qualify than the presently installed acoustic monitoring system. The presently installed acoustic monitoring system will continue to be used until the new assemblies are installed.

Installation of the lift indicating switch assemblies is presently planned for the fall 1982 refueling of Unit 1 and the spring 1983 refueling of Unit 2.

II.E.1.1 AUXILIARY FEEDWATER SYSTEM EVALUATION

Over the past several years, the NRC has requested information regarding the design, instrumentation and operation of the auxiliary feedwater (AFW) systems for the Point Beach Nuclear Plant, Units 1 and 2 and Wisconsin Electric has responded to these requests and provided the necessary information to satisfy each point raised (References 1, 2, 3, 4a, 12, 24, 40, 49, and 67). The only remaining item to be addressed is that of automatic protection of the AFW pumps if the AFW system water supplies are not completely protected from damage following a seismic event or tornado.

Wisconsin Electric continues to maintain the position that the condensate storage tanks, although not Class I water sources, are reliable and designed to specific, adequate seismic criteria. In the absence of NRC acceptance of this as sufficient, we have subsequently agreed to study the feasibility and desirability of providing pump protection through automatic AFW pump trips on low suction pressure as an alternative to automatic switchover. However, since we believe this would decrease rather than increase system reliability, we will not install this equipment until the NRC acceptance is received.

Wisconsin Electric has reluctantly decided to implement the automatic AFW pump trips as part of the instrumentation effort in progress. We have completed the basic design of the AFW suction pressure instrumentation channels. The procurement of safety-grade, environmentally qualified equipment is in progress and installation will be integrated with the other instruments, equipment and cable routing. Implementation requires the Foxboro Spec 200 racks for both

units to be installed and operational, which is scheduled to be completed by June 1, 1983 with initial operation during the summer of 1983. The design of the modifications to be made to the AFW pump control circuitry has not yet begun. A complete and thorough review of any design will ensure that no decrease in the reliability or operability of the AFW system results from these changes. A schedule for this work has yet to be determined.

II.E.4.2.1/4 CONTAINMENT ISOLATION DEPENDABILITY - IMPROVED DIVERSE ISOLATION

As originally designed, Point Beach met the NRC criteria for diversity of isolation and manual, single valve restoration (Reference 3: NUREG-0578, Item 2.1.4, LL Cat. A). Two out of three areas of NRC concern have been previously completed. These were revisions to administrative and operating procedures and modification to the remote control switch for outboard purge valve CV-3212 (Reference 2).

The remaining item is the addition of isolation valves inside containment to piping for letdown, seal water return, and steam generator blowdown. Note that redundant, diversely actuated isolation outside of containment already exists. Auxiliary charging has been removed from this list since it already has inside isolation and the normally closed manual valve outside will be locked closed or administratively controlled if opened. The blowdown valves were added to this list due to the existing isolation valves being located outside of containment in an area which is postulated to be subject to falling debris from block walls, per the criteria of IE Bulletin 80-11.

The addition of inside containment isolation valves on the letdown, seal water return, and steam generator blowdown lines is in the process of being implemented utilizing the services of an outside consultants for the piping and support design and analysis as well as the controls and indication circuitry, conduit layout, and cable routing.

The remote air-operated isolation valves have been ordered from Copes-Vulcan. The initial vendor stated delivery schedule for the valves of 48 weeks has not been able to be met and installation in 1982 is not scheduled. Installation could next take place during the spring 1983 refueling outage for Unit 2 and the fall 1983 refueling outage for Unit 1 with completion by January 1, 1984.

The manual isolation valves required to test the auto-isolation valves have recently been purchased with an estimated delivery time of thirty weeks, which Wisconsin Electric has not been able to confirm with the vendor (Powell). We believe, however, that delivery of these valves will not delay installation of the isolation valves. Vent and drain valves will come from plant stock when available.

II.F.1.1/2 ACCIDENT MONITORING - NOBLE GAS, IODINE, AND PARTICULATE

Wisconsin Electric is currently in the process of upgrading the entire radiation monitoring system for the Point Beach Nuclear Plant. Wisconsin Electric has purchased an Eberline radiation monitoring system and anticipates that it

will be installed and operational by December 1, 1982. Accident monitoring for effluents has been integrated into the upgraded system. The effluent monitoring equipment which was purchased from Eberline consists of three model SPING-4 monitors, one model SPING-3 monitor and four model SA-11 monitors. The SPING monitors will be used for monitoring exhaust stacks and the SA-11 monitors will be used to monitor steam releases. It was originally envisioned that the SPING monitors would satisfy the exhaust stack sampling and monitoring criteria. However, due to changes issued in the NRC clarifications, the SPING monitors will not satisfy the particulate and iodine sampling criteria of II.F.1.2. They will, however, still provide the full range of noble gas monitoring required. In view of this, a separate sampling system (referred to as the Isokinetic Stack Sampling System [ISSS]) is being purchased from another vendor and will be integrated into the upgraded system with installation scheduled for December 1, 1982. An overall system description is provided. The details of each of the above effluent monitors (SPING, SA-11 and ISSS) follows the overall system description.

System Description

The upgraded radiation monitoring system is a microprocessor-based radiation detection system. The heart of the system is eight Data Acquisition Modules (DAM) and four SPING monitors. Each SPING has a DAM built into it. Each DAM has a microcomputer which performs the tasks of data acquisition, history file management, operational status check, and alarm determination. Each DAM is capable of serving nine detector inputs and six analog inputs. The operator interface with the system is through a Control Terminal (CT-2) and a color graphics minicomputer (CRT-1).

System Operation

Radiation is detected and the signals processed by the electronics in a component called an interface box. The output signals from the interface boxes are input to the microcomputer. These signals are converted to count rate by the microcomputer which then performs all mathematical calculations and control functions. The mathematical calculations include the application of conversion factors and background subtraction. Each microcomputer maintains a history file for each of its input channels. The history file consists of the last four hours of ten-minute averages, the last twenty-four hours of one-hour averages and the last twenty-four days of one-day averages. Each DAM (and SPING) has a local readout panel. The local readout device is capable of accessing the current status of any channel associated with that particular DAM. The DAM's are connected to two CT-2's. Each CT-2 drives a CRT-1. Each CT-2 has its own keyboard, printer and system status annunciator. The annunciator consists of an audible alarm and status lights. Each CT-2 includes a microcomputer which performs the functions of polling each DAM for operational status and data, logging any changes in status and associated data, logging history files automatically, performing calculations on history file data and annunciating alarm conditions and communication error messages. The CRT-1 operates independently of the CT-2 and has access to all of the CT-2 information. The CRT-1 constantly displays all channels which are in alarm and gives the operator the capability of graphically trending any channel. In addition, each DAM can be locally interrogated for current status and history files by a portable terminal. The Point Beach control room will be equipped with one CT-2 and one CRT-1. The Point Beach Technical Support Center will be equipped with either a CT-2 and/or CRT-1.

The entire upgraded radiation monitoring system will be powered from the vital (instrument) busses. The instrument b... provides power to each DAM. The DAM provides the power to each of its associated channels. In addition, each DAM is equipped with a battery which provides for eight hours of continuous operation in the event of a power failure.

A. SPING Monitor

The Eberline SPING monitor is a self-contained microprocessor-based radiation detection system used to sample and monitor particulates, iodine and noble gases in the air. The sample intake goes through a filter paper on which particulates are deposited, then through a charcoal cartridge to trap iodines and then into the gas chamber for low and medium range noble gas measurement. The sample then passes through a high-range noble gas chamber, through the pump and to the sample outlet. The SPING-3 is the only SPING which is not equipped with a high-range noble gas chamber. The high-range noble gas chamber is the only difference between a SPING-3 and a SPING-4.

The upper counting range of the particulate, iodine, and low-range noble gas channels is 10^6 cpm. The beta particulate channel is approximately 12% (4π) efficient for Cs-137 beta particules. The I-131 gamma scintillation channel is approximately 4% (4π) efficient for the 364 keV gamma from I-131 decay. The low-range noble gas channel's useable range is from 1×10^{-7} to 4×10^{-2} µCi/cc for Xe-133. The medium-range noble gas channels range is from 2.5×10^{-2} to 1×10^{3} µCi/cc for Xe-133. The high-range gross gamma channel has a range of 1×10^{0} to 1×10^{5} µCi/cc for Xe-133. An area monitor measures ambient radiation levels and has a range of 1 µR/hr to 100 mr/hr.

The SPING features stainless steel plumbing through the sampler stages, a photohelic flow indicator with low and high flow setpoints, remote flush valves, a manual grab sample port with hose barbs, a sealed diaphram air pump and a connection plug for a portable terminal.

1. Instrumentation

   a. The particulate filter is monitored by a beta scintillation detector from one side (Eberline model RDA-3A) and a solid-state alpha detector on the other side (Eberline model RDS-1). Counts from the beta detector are a measure of the amount of beta-emitting isotopes on the filter. The alpha detector measures the radon levels in the sample to provide a measure of the contribution of these daughter products to the beta particulate levels. This contribution can then be corrected by subtracting a factored amount of the alpha measurement from the beta measurement. This technique is used to compensate for the effects of fluctuating radon/thoron levels in the beta particulate measurement.

   b. The charcoal cartridge is monitored by a 2" x 2" NaI gamma scintillation detector. This detector (Eberline model RDA-2A) is gain stabilized to minimize the effects of drift caused by fluctuations in temperature and/or aging. The measurement is

accomplished using a single channel analyzer (SCA) with its window calibrated to the 364 keV energy of I-131. There is an additional SCA with its window calibrated to an energy above the I-131 energy to provide a measure of the background in the iodine window. The effects of a fluctuating background are compensated for, by subtracting the background.

c. The low-range noble gas monitor is a beta scintillation detector (Eberline model RDA-3A). Background correction for this channel is derived from the gamma background detector which is an energy compensated Geiger-Mueller (G-M) detector (Eberline model 10450-B28). Since the external (ambient) gamma radiation has a measurable effect on the beta measurement (particulate and gas), the gamma background channel is used as a source of subtraction for both the gas measurement and the particulate measurement.

d. An energy compensated G-M detector (Eberline model 10450-B28) monitors the gas volume for the medium-range noble gas measurement. Its output is proportional to the gamma emission of the sample. An additional identical detector, the background detector, is provided in the sampler shield as a measure of external background at the sampler. Thus, the effect of a fluctuating external background on the medium-range gas channel are nullified by measuring and subtracting the background.

e. An energy compensated G-M detector (Eberline model 10450-B28) monitors the gas volume of a section of 1" stainless steel tubing in the SA-9 high-range noble gas sampler of the SPING-4. Its output is proportional to the gamma emission of the sample.

f. Each SPING is equipped with a local area monitor. This detector is an energy compensated G-M tube (Eberline model DAI-1-CC) which is calibrated in radiation dose rate and provides a measure of the gamma field at the instrument.

g. Radioactive check sources are provided to enable periodic checking of the detectors and electronics for proper response. The following list summarizes the channels with check sources.

CHANNEL	CHECK SOURCE
Beta Particulate	30 μCi Cs-137
Iodine	0.5 μCi Ba-133
Low-Range Noble Gas	30 μCi Cs-137
Area Monitor	0.5 μCi Sr-90, Y-90
High-Range Noble Gas	0.5 μCi Sr-90, Y-90

2. The SPING-4 monitors will be used to monitor the exhaust flow of the Unit 1 containment purge exhaust stack, Unit 2 containment purge exhaust stack and the auxiliary building exhaust stack. The containment purge exhaust stacks are monitored even though the purge valves are closed during normal operation. These monitors

will be located in the electrical equipment rooms. The sample connect point for these monitors will be downstream of the stack filters. The SPING-3 monitor will be used to monitor the exhaust flow of the Point Beach radwaste packaging area exhaust stack. The SPING-3 will be located in the radwaste packaging area. The sample connect point for the SPING-3 will be located downstream of the filters. ANSI N13.1-1969 will be used as the guidelines for the design of nozzles and piping layout.

B. SA-11 Monitor

The upgraded radiation monitoring system includes a SA-11 monitor for each steam line for each unit (four total). The SA-11 is comprised of a lead-shielded detector which views the main steam line upstream of the safety valves for gamma radiation. The detector is an energy compensated G-M tube (Eberline model 10450-B28). The detector output is proportional to the gamma emission from the steam line. The detector output is input to a single channel on a DAM. The SA-11 monitor is equipped with a remotely actuated check source (0.5 µCi Sr-90, Y-90). The range of the SA-11 is from $1x10^{-1}$ to $1x10^{4}$ mR/hr. The exact conversion to µCi/cc of Xe-133 has not yet been determined. The calibration procedure for this monitor is expected to yield the appropriate conversion factors. The calibration procedure is currently being developed.

C. Isokinetic Stack Sampling System (ISSS)

The ISSS is being designed to satisfy the requirements of II.F.1.2. However, due to the untimely response of vendors to the Wisconsin Electric request for proposal, exact design details cannot be presented at this time. The conceptual design and specifications will be presented. It is anticipated that this system will be installed and operational by December 1, 1982. This system will be installed on the auxiliary building exhaust stack and the radwaste packaging area exhaust stack since they are the only viable early release paths in an accident situation.

A representative sample will be drawn at an isokinetic rate with respect to stack flow. The unit will maintain isokinetic sampling capability up to the design flow of 62,000 cfm with an error of less than ±20%. The system will be capable of continuous operation. The stack flow sensor and sample flow monitor will be high quality devices. The flow controller will be capable of both automatic and manual operation. The stack flow transmitter will provide a 4-20 ma signal for input to the DAM analog input boards. The Flow Control Valve (FCV) will be of high quality. System temperature control will be provided. Local indicators for stack velocity and flow and sample velocity and flow will be provided. The length of the line between the sample withdrawal and the sample collector will be as short as possible. The collector must be shielded. Quick changeout of the collection media is required to minimize personnel exposure.

All vendors were advised that particulates and radioiodines were the species to be collected. Vendors were directed to propose a collector which meets the requirements of NUREG-0737. Vendors were supplied with a list of references which included NUREG-0737, ANSI N13.1 and Regulatory Guide 1.97, Revision 2.

D. Procedures for Release Rate Calculations

The current procedures and calculational methods for quantifying releases are part of the Point Beach Emergency Plan Implementing Procedures. These procedures will be updated as required in order to reflect the upgraded radiation monitoring system.

II.F.1.3 ACCIDENT MONITORING - CONTAINMENT HIGH-RANGE RADIATION MONITOR

Additional in-containment high-range radiation monitoring will be provided for Point Beach Nuclear Plant by six gamma ionization chambers supplied by General Atomic Company. Three RD-23 detectors with an eight decade range will be located on floor- or beam-mounted seismic supports located on the 66' El. operating level in each unit's containment. Power for each detector and signal processing will be provided by RP-2C readout modules mounted in new auxiliary instrumentation racks located in the computer room above the control room. Each detector will be supplied power from a separate safety-grade instrument bus via an individual power supply associated with the RP-2C readout module. Power for each bus originates with a battery/charger-supplied inverter to ensure continuous operation. Separation and seismic support provide IE qualification for the detector channels.

The output of the individual detector channel will be electronically isolated and processed by Foxboro SPEC 200 instrumentation currently being installed in the same area. The SPEC 200 outputs will also be isolated and routed to the ASIP for display and high-level alarm annunciation. Additional outputs will be provided for the plant computer and instrumentation systems.

Installation of the in-containment monitors can be accomplished in fall 1982 for Unit 1, and spring 1983 for Unit 2. Operation of the entire system is dependent upon delivery and installation of the ASIP and the auxiliary racks (summer 1982). Power for the SPEC 200 instrumentation racks will be provided from new battery-inverter power supplies which should be available by June 1, 1983 (see II.F.1.4.4 below).

II.F.1.4 CONTAINMENT PRESSURE MONITOR

New high-range pressure transmitters are being installed to provide containment wide-range pressure indication. Indication and recording will be provided in the control room.

This indication system is typical of many that are being installed. It consists of several components, each with their own schedule, as follows:

1. Foxboro pressure transmitters and transmitter mounts meeting seismic design criteria and Quality Assurance standards were installed for Unit 1 during the fall 1981 refueling outage and are being installed for Unit 2 during the current spring 1982 refueling outage.

2. Foxboro Spec 200 analog process racks have been ordered, delivered and are in the process of being mounted seismically.

3. Interconnecting cables must be pulled and terminated. These cables connect the transmitters to the Foxboro racks, the racks to the main control board in the control room, the racks to the Auxiliary Safety Instrumentation Panel (ASIP) in the control room, and the racks to the computer system. These cables will be installed during 1982 and 1983 as equipment is installed.

4. Power for the white and yellow powered Foxboro SPEC 200 racks will be supplied by new battery-inverter power supplies. The batteries will be located in new battery rooms which will be constructed. It is estimated that the new power supplies will be available June 1, 1983.

5. Readouts in the control room will be mounted on the existing main control board (panel C01) and the new ASIPs (panels 1C20 and 2C20). The new panels are each fifteen feet long and their installation in the control room must be carefully planned and implemented. The ordering of these panels has been delayed due to the review and layout of instrumentation and cable and conduit routing and the identification of qualified instruments and controls. The design of these panels is being finalized and the procurement of these panels is in process with vendor bids being evaluated. The C20 panels will be ordered shortly and it is expected that they can be designed, fabricated, and delivered by early 1983 with operation scheduled for June 1, 1983.

Although portions of the high-range containment pressure indicating system may be operating sooner on an interim basis, it is currently expected that the final configuration of the analog indications in the control room will be operational during the summer of 1983. The same constraints that apply to the high-range containment pressure system also apply to most of the other indication systems that require display in the control room.

II.F.1.5 CONTAINMENT WATER LEVEL

DeLaval-Gems transmitters, receivers and seismically designed transmitter mounts were installed for Unit 1 during the fall 1981 refueling. The Unit 2 transmitters are being installed during the current spring 1982 refueling. The receivers will be wall mounted in the computer room. Installation of the receivers is expected during the latter part of 1982.

The Foxboro SPEC 200 racks will be used to isolate and distribute the receiver outputs to the computer and to the ASIP. The constraints and implementation schedule mentioned in II.F.1.4 with respect to the racks, cables, power supplies, and ASIP also apply to the containment water level indications. Operation of this system is therefore currently scheduled for June 1, 1983.

II.F.1.6 CONTAINMENT ATMOSPHERE HYDROGEN ANALYZER

Redundant hydrogen sensors powered by independent power supplies will be provided in each containment. The analyzers that have been ordered are manufactured by Exo Sensors, Inc. The final acceptability of this equipment is based on the outcome of Exo Sensor's qualification test program which is in progress.

It is planned to mount the sensors in the Unit 1 containment during the fall 1982 refueling and to mount the sensors in the Unit 2 containment during the spring 1983 refueling. Cable runs were installed in the Unit 1 containment during the fall 1981 refueling. Cables have already been installed in the Unit 2 containment. Local routing will be done after final installation of the transmitters.

The microprocessors will be installed in auxiliary racks in the computer room. These auxiliary racks have been ordered and delivery is expected during mid-1982 coincident with the seismic qualification testing of a similar rack.

The Foxboro SPEC 200 racks will be used to isolate and distribute the analyzer outputs to the computer and to the ASIP. The constraints mentioned in II.F.1.4 with respect to racks, cables, power supplies and the ASIP also apply to the containment atmosphere hydrogen analyzer indications. It is expected that the system from sensors to indicators in the control room can be operational for both units by June 1, 1983

II.F.2.1 INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING - SUBCOOLING METER

The final subcooling system which will meet all of the NRC requirements has been designed as part of the qualified instrumentation system being added to the plant. This added system is intended to provide the means by which the large number of instrumentation changes can be properly incorporated into the existing plant. This new system consists of redundant channels of instrumentation for each unit. The ASIP is located in the control room at a location which allows for easy viewing of the panel-mounted display devices. Subcooling display meters will be located on the ASIP. The operability of the final subcooling meter is dependent upon the delivery, installation and operational checkout of the new racks and panels. It is our plan to have the ASIP panels delivered in early 1983 and the displays located on these panels should be operational by June 1, 1983. This includes the final configuration of the subcooling display. The subcooling capability described in References 1 and 2 is judged to be sufficient until the final system is operational in the plant.

II.F.2.3 REACTOR VESSEL WATER LEVEL

The reactor vessel water level system uses redundant, separately powered wide- and narrow-range Foxboro differential pressure transmitters located inside containment, four transmitters per unit. Installation of the Unit 1 transmitters and portions of the tubing system were completed during the fall 1981 refueling outage. The Unit 2 transmitters and portions of the tubing system will be installed during the current spring 1982 refueling.

The top fluid connection to the reactor vessel head is being made by Westinghouse Electric Corporation. This connection was installed in Unit 1 during the fall 1981 refueling and is being installed in Unit 2 during the current spring 1982 refueling.

The bottom fluid connection to an incore instrumentation thimble guide tube coupling is being made by NUS Corporation. This connection was installed in Unit 1 during the fall 1981 refueling and is being installed in Unit 2 during the current spring 1982 refueling.

At the high point of the top tap fluid lines, a chamber will be installed with sufficient capacity to refill the fluid lines to the transmitters once. This chamber acts as a water reservoir; it does not contain a bellows or diaphram. The chamber, its supports, and supports for the fluid lines are being seismically designed and will be fabricated with the proper Quality Assurance provisions. It is expected that these elements of the system will be installed in Unit 2 during the spring 1983 refueling outage and in Unit 1 during the fall 1983 refueling outage.

The top tap fluid line has been designed to penetrate through the refueling canal wall in order to maintain proper elevations. This fluid penetration through the refueling canal wall will be fabricated and a hole will be drilled through the wall to accept the penetration. It is expected that this penetration will be installed in Unit 2 during the spring 1983 refueling and in Unit 1 during the fall 1983 refueling.

Chromel-alumel thermocouples will be attached to the vertical runs of the fluid lines to provide temperature compensation. It is expected that these thermocouples will be installed in Unit 2 during the spring 1983 refueling and in Unit 1 during the fall 1983 refueling.

The reactor vessel incore thermocouples and the thermocouples on the vertical runs of the fluid lines will be processed by input cards in the new computer multiplexing system. The computer multiplexing system will output weighted-average signals to the Foxboro racks. It is expected that the computer multiplexing system will be installed by November 1983.

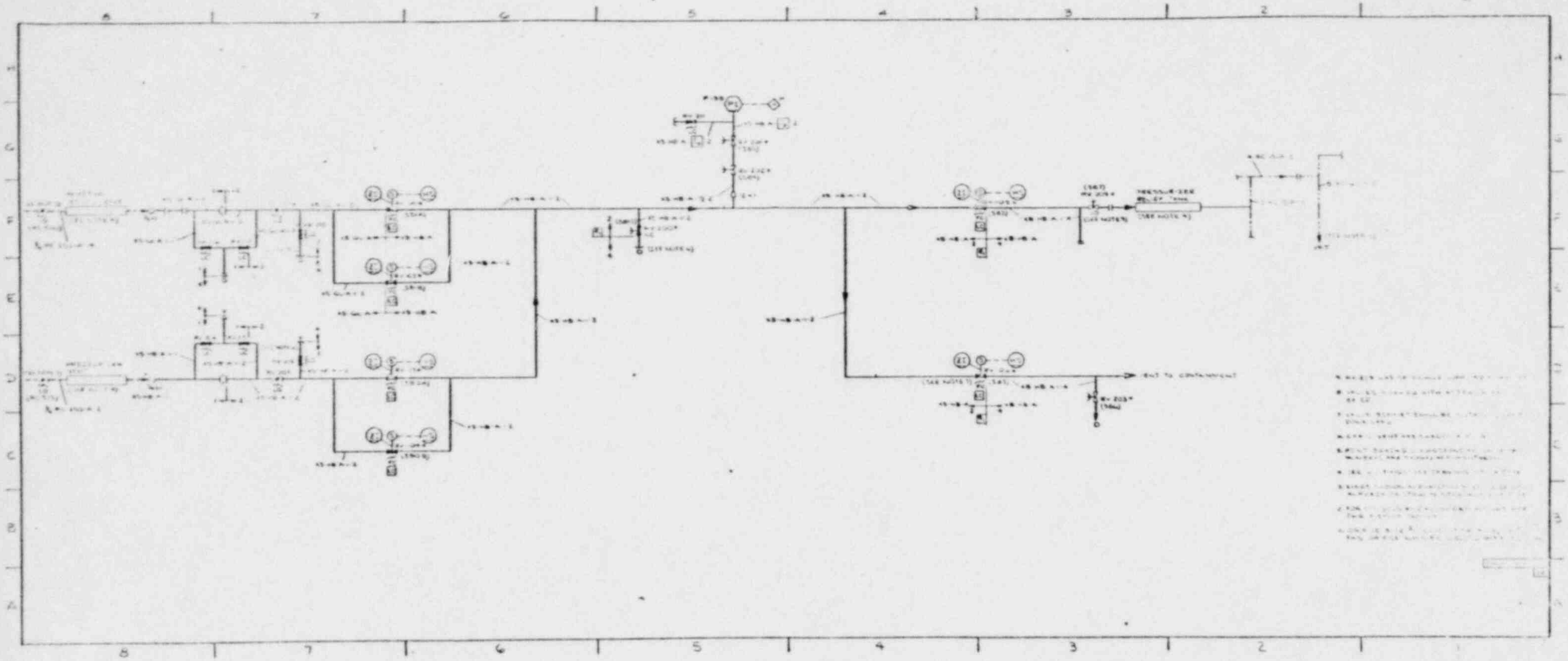
Foxboro SPEC 200 analog process racks will be used to power the differential pressure transmitters, perform the computations, and distribute the isolated output signals to the computer and ASIP. It is expected that the system from sensors to indicators in the control room can be operational for both units by the end of 1983.

A detailed description of the vessel level system showing the tap locations, fluid line routing, transmitter location, signal processing and parameter displays has been submitted to the NRC in C. W. Fay to H. R. Denton letter dated October 20, 1981 (Reference 52). The system utilizes vessel differential pressure to determine level but it is not a Westinghouse-designed or supplied system.

II.K.2.17 POTENTIAL FOR VOIDING IN THE RCS DURING TRANSIENTS

The potential for void formation in a Westinghouse-designed nuclear steam supply system during natural circulation cooldown/depressurization operation was addressed in a study and submitted to the NRC in Owners Group letter, OG-57, dated April 20, 1981. This study is applicable to Point Beach Nuclear Plant.

In addition, the Westinghouse Owners Group has developed a natural circulation cooldown guideline that takes the results of the study into account so as to preclude void formation in the upper head region during natural circulation cooldown/depressurization transients, and specifies those conditions under which upper head voiding may occur. These Westinghouse Owners Group generic guidelines have been submitted to the NRC in Reference 53. The generic guidance developed by the Westinghouse Owners Group (augmented as appropriate with plant-specific consideration) will be utilized in the implementation of Point Beach Nuclear Plant emergency operating procedures.



REACTOR COOLANT GAS VENT SYSTEM

II.B.1 Reactor Coolant System Vents

APPROVED
REFERENCE
DESIGN

REFERENCES

1. S. Burstein (WE) letter to H. R. Denton (NRC), December 31, 1979, "Implementation of NUREG-0578"

2. C. W. Fay (WE) letter to H. R. Denton (NRC), March 14, 1980, "Implementation of NUREG-0578"

3. A. Schwencer (NRC) letter to S. Burstein (WE), April 9, 1980, "Evaluation of Compliance with Category "A" Lessons Learned Requirements"

3a. D. G. Eisenhut (NRC) letter To All Operating Reactor Licensees, May 7, 1980, "Five Additional TMI-2 Related Requirements (Applicable) to Operating Reactors"

3b. C. W. Fay (WE) letter to H. R. Denton (NRC), May 2, 1980, "Implementation of NUREG-0578"

4. C. W. Fay (WE) letter to H. R. Denton (NRC), June 11, 1980, "Implementation of Five Additional TMI-2 Related Requirements"

4a. C. W. Fay (WE) letter to H. R. Denton (NRC), June 11, 1980, "Additional Information Auxiliary Feedwater System"

4b. D. G. Eisenhut (NRC) letter To All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits, July 31, 1980, "Interim Criteria for Shift Staffing"

4c. C. W. Fay (WE) letter to H. R. Denton (NRC), August 7, 1980, "Qualification of Reactor Operators"

5. C. W. Fay (WE) letter to H. R. Denton (NRC), November 3, 1980, "Status of Duty and Call Technical Advisor Training"

6. C. W. Fay (WE) letter to H. R. Denton (NRC), December 1, 1980, "Revised Emergency Plan"

7. C. W. Fay (WE) letter to H. R. Denton (NRC), November 3, 1980, "Operating Licenses DPR-24 and DPR-27, Interim Criteria for Shift Staffing"

8. C. W. Fay (WE) letter to H. R. Denton (NRC), September 22, 1980, "Comments on Draft NUREG-0696, Functional Criteria for Emergency Response Facilities"

9. S. Burstein (WE) letter to H. R. Denton (NRC), October 20, 1979, "Implementation of NUREG-0578" - including TMI Accident Review Task Force Report (Section 3.6.A)

9a. P. F. Collins (NRC) letter to C. W. Fay (WE), October 28, 1980, regarding Item I.A.2.1.3 of NUREG-0737.

9b. R. F. Heishman (NRC) letter to Sol Burstein (WE), November 20, 1980, Inspection Report No. 50-266/80-19 and No. 50-301/80-19.

10. R. W. Jurgensen (WOG) letter to S. H. Hanauer (NRC), OG-47, December 15, 1980, "Westinghouse Owners Group Response to Item I.C.1 of NUREG-0737"

10a. R. C. Youngdahl (EPRI) letter to D. G. Eisenhut (NRC), December 15, 1980, PWR Utilities Position on NUREG-0737, Item II.D.1, Performance Testing of BWR and PWR Relief and Safety Valves (NUREG-0778, Section 2.1.2)

11. S. H. Hanauer (NRC) letter to R. A. Newton (WE), December 17, 1980, request for a basis document for the emergency procedure guidelines.

12. C. W. Fay (WE) letter to H. R. Denton (NRC), December 23, 1980, "Response to NURG-0737, Schedule Requirements as Related to Point Beach Nuclear Plant, Units 1 and 2"

12a. R. F. Heishman (NRC) letter to Sol Burstein (WE), December 23, 1980, Inspection Report No. 50-266/80-20 and No. 50-301/80-20.

12b. C. W. Fay (WE) letter to H. R. Denton (NRC), December 30, 1981, "Point Beach Nuclear Plant Emergency Plan"

13. C. W. Fay (WE) letter to H. R. Denton (NRC), January 9, 1981, "Additional Response to NUREG-0737"

13a. C. W. Fay (WE) letter to H. R. Denton (NRC), January 19, 1981, "Response to NUREG-0737"

14. R. W. Jurgensen (WOG) letter to S. H. Hanauer (NRC), OG-48, January 28, 1981, "Emergency Operating Instruction Background Documents"

15. C. W. Fay (WE) letter to H. R. Denton (NRC), February 4, 1981, "Technical Specification Change Request No. 65"

15a. Sol Burstein (WE) letter to S. J. Chilk (NRC), February 18, 1981, "Emergency Operatings Facility"

15b. D. G. Eisenhut (NRC) letter To All Licensees of Operating Plants and Holders of Construction Permits, February 18, 1981, "Post-TMI Requirements for the Emergency Operations Facility (Generic Letter 81-10)"

16. C. W. Fay (WE) letter to H. R. Denton (NRC), February 23, 1981, "Additional Responses to NUREG-0737"

17. C. W. Fay (WE) letter to H. R. Denton (NRC), February 26, 1981, "Duty and Call Technical Advisor Training"

17a. C. W. Fay (WE) letter to H. R. Denton (NRC), February 27, 1981, "Emergency Plan Implementing Procedures"

18. C. W. Fay (WE) letter to H. R. Denton (NRC), March 4, 1981, "NUREG-0737 Schedule Requirements"

19. R. W. Jurgensen (WOG) letter to J. R. Miller (NRC), OG-52, March 13, 1981, WCAP-9804, "Probabilistic Analysis and Operational Data in Response to Item II.K.3.2 for Westinghouse NSSS Plants"

20. R. W. Jurgensen (WOG) letter to S. H. Hanauer (NRC), OG-54, March 18, 1981, "Westinghouse Owners Group Update on Item I.C.1 of NUREG-0737 Activities"

21. E. R. Mathews (WPS) letter to Sol Burstein (WE), March 19, 1981, "Letter of Intent to Provide Mutual Assistance"

22. C. W. Fay (WE) letter to H. R. Denton (NRC), March 31, 1981, "Emergency Plan"

23. C. W. Fay (WE) letter to H. R. Denton (NRC), March 31, 1981, "Response to NUREG-0737 Update to Schedule Requirements and Implementation Status"

24. C. W. Fay (WE) letter to H. R. Denton (NRC), April 9, 1981, "Requirements for Auxiliary Feedwater System"

25. C. W. Fay (WE) letter to H. R. Denton (NRC), April 14, 1981, "Post-TMI Requirements for the Emergency Operations Facility"

25a. R. F. Heishman (NRC) letter to Sol Burstein (WE), April 17, 1981, regarding routine inspection conducted on March 2-31, 1981, of activities at Point Beach Nuclear Plant, Units 1 and 2.

25b. R. W. Jurgensen (WOG) letter to P. S. Check (NRC), OG-57, April 20, 1981, "Natural Circulation Cooldown Report"

26. C. W. Fay (WE) letter to H. R. Denton (NRC), May 7, 1981, "Containment Pressure Setpoint, NUREG-0737 Item II.E.4.2"

26a. W. A. Wogsland (WOG) letter to L. Beltracchi (NRC), May 19, 1981, "Proprietary Handouts, SAS Meeting, May 14, 1981"

27. D. G. Eisenhut (NRC) letter to R. W. Jurgensen (WOG), May 28, 1981, "WOG Procedures Development and Evaluation Program"

28. Sol Burstein (WE) letter to H. R. Denton (NRC), June 1, 1981, "Emergency Response Facilities"

29. R. A. Clark (NRC) letter to Sol Burstein (WE), June 1, 1981, "Clarification of Post-Accident Sampling Requirements of NUREG-0737, II.B.3"

30. C. W. Fay (WE) letter to H. R. Denton (NRC), June 2, 1981, "Emergency Plan Public Notification System"

31. R. W. Jurgensen (WOG) letter to P. S. Check (NRC), OG-60, June 15, 1981, "Response to NRC Letter of May 30, 1981"

32. R. A. Clark (NRC) letter to Sol Burstein (WE), June 17, 1981, "Review of the Westinghouse Owners Group submittal for Action Plan Item I.C.1"

33. R. A. Clark (NRC) letter to Sol Burstein (WE), June 17, 1981, "Safety Evaluation by the Office of Nuclear Reactor Regulation" (RE: IEB 79-06A and 79-06A, Revision 1, Item II.K.1)

34. R. A. Clark (NRC) letter to Sol Burstein (WE), June 26, 1981, "Elimination of Item II.K.2.19"

35. C. W. Fay (WE) letter to H. R. Denton (NRC), July 6, 1981, "NUREG-0737 Schedule Update"

36. R. W. Jurgensen (WOG) letter to S. H. Hanauer (NRC), OG-61, July 7, 1981, "Summary of WOG Program to Address NUREG-0737, Item I.C.1."

37. R. A. Clark (NRC) letter to Sol Burstein (WE), July 10, 1981, "Order Confirming Licensee Commitments on Post-TMI Related Issues"

38. C. W. Fay (WE) letter to H. R. Denton (NRC), July 17, 1981, "Meteorological System Description"

39. R. C. Youngdahl (EPRI) letter to H. R. Denton (NRC), July 24, 1981, "Status of EPRI PWR Safety and Relief Valve Test Program, NUREG-0737, Item II.D.1"

40. R. A. Clark (NRC) letter to Sol Burstein (WE), July 28, 1981, "Request for Additional Information on Auxiliary Feedwater System"

41. R. A. Clark (NRC) letter to Sol Burstein (WE), August 5, 1981, "Westinghouse Reactor Vessel Level Instrumentation for Monitoring Inadequate Core Cooling"

41a. Sol Burstein (WE) letter to S. J. Chilk (NRC), August 12, 1981, "Comments on Incorporation of NUREG-0737 Licensing Requirements into 10 CFR Part 50"

42. R. A. Clark (NRC) letter to Sol Burstein (WE), August 14, 1981, "Containment Isolation Pressure Setpoint"

43. R. A. Clark (NRC) letter to Sol Burstein (WE), August 19, 1981, "TMI Task Plan Items II.K.3.9, II.K.3.10, and II.K.3.12 for Point Beach Nuclear Plant, Units 1 and 2"

44. C. W. Fay (WE) letter to Secretary of the Commission (NRC), August 25, 1981, "Comments on NUREG-0799, Draft Criteria for Preparation of Emergency Operating Procedures"

44a. R. L. Spessard (NRC) letter to Sol Burstein (WE), August 26, 1981, regarding routine inspection conducted on July 1-31, 1981, of activities at Point Beach Nuclear Power Plant, Units 1 and 2.

45. Sol Burstein (WE) letter to H. R. Denton (NRC), August 28, 1981, "Technical Specification Change Request No. 68, Containment Purge Valve Operability"

46. Sol Burstein (WE) letter to H. R. Denton (NRC), September 11, 1981, "Modification of Instrument Power Supply"

47. Sol Burstein (WE) letter to H. R. Denton (NRC), September 14, 1981, "Response to NUREG-0737, Update to Schedule Requirements and Implementation Status"

48. R. A. Clark (NRC) letter to Sol Burstein (WE), September 14, 1981, regarding acceptance of item II.E.4.1, Dedicated Hydrogen Penetrations in NUREG-0737.

49. C. W. Fay (WE) letter to H. R. Denton (NRC) September 16, 1981, "Additional Information, Auxiliary Feedwater System"

50. D. G. Eisenhut (NRC) to All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits, September 29, 1981, "Revised Schedule for Completion of TMI Action Plan Item II.D.1, Relief and Safety Valve Testing (Generic Letter No. 81-36).

51. T. G. Colburn (NRC) letter to Wisconsin Electric Power Company, October 7, 1981, "Summary of Meeting Held with Licensee to Discuss Integrated Scheduling of NUREG-0737 Items with Other NRC Requirements"

52. C. W. Fay (WE) letter to H. R. Denton (NRC), October 20, 1981, "Reactor Vessel Water Level Indication System Description"

53. R. W. Jurgensen (WOG) letter to D. G. Eisenhut (NRC), OG-64, November 30, 1981, "Emergency Response Guideline Program"

54. R. A. Clark (NRC) letter to Sol Burstein (WE), December 2, 1981, "TMI Action Plan Items I.A.1.3, I.C.5, and I.C.6 as Described in NUREG-0737"

55. C. W. Fay (WE) letter to H. R. Denton (NRC), December 16, 1981, "Prompt Notification System"

56. C. W. Fay (WE) letter to H. R. Denton (NRC), December 29, 1981, "Functional Description and Proposed Implementation Schedule - Meteorological Monitoring System"

57. C. W. Fay (WE) letter to H. R. Denton (NRC) December 29, 1981, "NUREG-0737 Items II.F.1.1 and II.F.1.2"

58. O. D. Kingsley (WOG) letter to H. R. Denton (NRC), OG-66, December 30, 1981, "Reactor Vessel Integrity"

59. R. A. Clark (NRC) letter to Sol Burstein (WE), January 13, 1982, "NUREG-0737 Item I.A.1.1 Shift Technical Advisor (STA)"

60. D. G. Eisenhut (NRC) letter To All Licensees of Operating Plants, Applicants for an Operating License, and Holders of Construction Permits, February 8, 1982, "Nuclear Power Plant Staff Working Hours (Generic Letter No. 82-02)"

61. R. A. Clark (NRC) letter to C. W. Fay (WE) February 11, 1982, "Status of NUREG-0737 Item II.F.3 (sic) for Point Beach Nuclear Plant Units 1 and 2"

62. C. W. Fay (WE) letter to J. G. Keppler (NRC), February 18, 1982, "Minimum Staffing Requirements for NRC Licensees for Nuclear Power Plant Emergencies (Table III.A.1.2-1 or Table B-1 of NUREG-0654, Revision 1) and Point Beach Nuclear Plant Conformance"

63. R. A. Clark (NRC) letter to C. W. Fay (WE) February 18, 1982, "NUREG-0737 Item III.D 3.3., Improved In-Plant Iodine Instrumentation Under Accident Condition"

64. H. B. Clayton (NRC) memorandum for D. L. Ziemann (NRC), February 24, 1982, "Meeting Summary, Westinghouse Owners' Group and Westinghouse Emergency Operating Procedure Guidelines"

65. R. A. Clark (NRC) letter to C. W. Fay (WE) February 25, 1982, "Status of NUREG-0737 Items II.F.1.1 and II.F.1.2 for Point Beach Nuclear Plant, Units 1 and 2"

66. C. W. Fay (WE) letter to J. G. Keppler (NRC), March 8, 1982, "Minimum Staffing Requirements for NRC Licensees for Nuclear Power Plant Emergencies (Table III.A.1.2-1 or Table B-1 of NUREG-0654, Revision 1) and Point Beach Nuclear Plant Conformance"

67. C. W. Fay (WE) letter to H. R. Denton (NRC), March 16, 1982, "Auxiliary Feedwater Automatic Initiation"

68. D. G. Eisenhut (NRC) letter To All Licensees of Operating Power Reactors, March 17, 1982, "Post-TMI Requirements (Generic Letter No. 82-05)"

69. R. A. Clark (NRC) letter to C. W. Fay (WE), March 24, 1982, "TMI Action Plan Item II.K.3.3, Reporting Relief Valve (RV) and Safety Valve (SV) Failures and Challenges"

70. J. G. Keppler (NRC) letter to Sol Burstein (WE), March 31, 1982, regarding minimum onsite emergency staffing.

71. C. W. Fay (WE) letter to H. R. Denton (NRC), April 8, 1982, "Further Response to NUREG-0737, Item II.D.1"