NUREG-1066

Comparison of Implementation of Selected TMI Action Plan Requirements on Operating Plants Designed by Babcock & Wilcox

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

J. O. Thoma

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ABSTRACT

This report provides the results of a study conducted by the U. S. Nuclear Regulatory Commission staff to compare the degree to which eight Babcock and Wilcox (B&W) designed licensed nuclear power plants have complied with the requirements in NUREG-0737, "Clarification of TMI Action Plan Requirements". The eight licensed operating plants examined are as follows: Arkansas Nuclear One Unit 1 (ANO-1), Crystal River Unit 3, Davis Besse, Oconee Units 1, 2, and 3, Rancho Seco, and Three Mile Island Unit 1 (TMI-1). The purpose of this audit was to establish the progress of the TMI-1 licensee, General Public Utilities (GPU) Nuclear Corporation, in completing the long-term requirements in NUREG-0737 relative to the other B&W licensees examined.

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Executive Summary

This report provides the results of a study conducted by the U. S. Nuclear Regulatory Commission staff to compare the degree to which eight Babcock and Wilcox (B&W) designed licensed nuclear power plants have complied with the requirements in NUREG-0737, "Clarification of TMI Action Plan Requirements". The eight licensed operating plants examined are as follows: Arkansas Nuclear One, Unit 1 (ANO-1), Crystal River Unit 3, Davis Besse, Oconee Units 1, 2, and 3, Rancho Seco, and Three Mile Island Unit 1 (TMI-1). The purpose of this audit was to establish the progress of the TMI-1 licensee, General Public Utilities (GPU) Nuclear Corporation, in completing the long-term requirements in NUREG-0737 relative to the other B&W licensees examined.

The staff reviewed with each B&W licensee the scope of completion and current status of design, procurement, installation, testing, procedure preparation, and operator training for 19 items in NUREG-0737 which NRC records indicated were not complete as of January 23, 1984 for at least one of the eight operating B&W units. Two additional items were added at the request of GPU Nuclear during our initial site visit. The staff made site visits to all units in a three-week period to observe and discuss the status of implementation of each item, both with licensee management and operational personnel and with the NRC resident inspectors. Each licensee responded positively in providing the requested data and in arranging tours.

For the purpose of this examination, an item was considered complete if it had been designed and installed, was declared operable by the licensee, had written procedures implemented, and had all training requirements completed. Post-implementation issues raised as a result of NRC staff reviews, while a measure of the quality of the design or installation, are outside the scope of this audit in declaring items complete.

The staff concludes that no one B&W licensee stands out as superior to all others for all items audited for either the timing or the quality of the items implemented. Each licensee has at least some areas where its work is superior to the others and some areas where it has not performed as well as the others.

All B&W licensees either have met or plan on meeting their formal schedules for implementing these requirements. However, GPU Nuclear appears to be ahead of the other B&W licensees for item I.D.1, Control Room Design Reviews. Although a Task Analysis using ATOG procedures is not yet complete, GPU Nuclear has completed all modifications they presently feel are necessary whereas the other B&W licensees have not yet started their modifications. Of course, the NRC review may indicate the need for further modifications. On the other hand, for item II.B.2, Plant Shielding, GPU Nuclear is the only B&W licensee experiencing a long delay in completing the necessary modifications. A similar delay for TMI-1 may be experienced for III.D.3.4, Control Room Habitability. For all the other items audited, GPU Nuclear's commitments for implementation schedules are on a par with the other B&W licensees.

All B&W licensees claim that their systems, once declared operable, will meet or exceed the minimum requirements of NUREG-0737. Regarding the quality of work performed, GPU Nuclear appears to be ahead of the other B&W licensees for item III.A.2.2, Meterological Data, not so much from the meterological aspects but from the dose assessment aspect because it has installed additional equipment and capabilities beyond NRC requirements; and for item III.E.4.1.2, Dedicated Hydrogen Penetrations, where GPU Nuclear has gone beyond NRC requirements by installing two external hydrogen recombiners. For item II.B.3, Post Accident Sampling, GPU Nuclear is utilizing a manually operated system, whereas all the other B&W licensees have installed remotely operated sampling systems to reduce personnel exposure. For all the other items audited, GPU Nuclear's quality is on a par with the other B&W licensees.

In the aggregate, GPU Nuclear is on a par with all the other B&W licensees in meeting NUREG-0737 requirements.

1.0 INTRODUCTION

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. Those items that the Commission had approved for implementation as of October 1980 are contained in "Clarification of TMI Action Plan Requirements", NUREG-0737. On December 17, 1982 the NRC staff provided additional guidance through the issuance of "Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability (Generic Letter No. 82-33)".

This report provides a status of completion of selected NUREG-0737 items for operating nuclear power plants utilizing the Babcock and Wilcox (B&W) design. The purpose of this review was to address the relative progress of the TMI-1 licensee, General Public Utilities (GPU) Nuclear Corporation, in completing the NUREG-0737 long term requirements. The specific items selected for the comparison are listed in Table 1. The TMI Action Plan Requirements from NUREG-0737 audited included all items not yet completed for at least one of the eight operating B&W units as of January 23, 1984. In addition, two items, specifically II.D.1 Relief and Safety Valve Test Requirements and II.E.4.1.2 Dedicated Hydrogen Penetrations, were added at the request of the TMI-1 licensee. The staff desired to determine the current schedule, progress and level of effort necessary to complete the remaining items, to identify significant problems related to design and installation, and to provide a description of how each plant is meeting the requirements for each item. The eight operating units of B&W design audited are as follows: Arkansas Nuclear One, Unit 1 (ANO-1), Crystal River Unit 3, Davis Besse, Oconee Units 1, 2, and 3, Rancho Seco, and Three Mile Island, Unit 1 (TMI-1). References in this report to "all licensees" refer only to the six licensees associated with these eight operating units.

The staff made site visits to all units in a three-week period to observe and discuss the status of implementation of each item, both with licensee management and operational personnel and with the NRC resident inspectors. Each licensee responded in a positive manner in providing the requested data and tours. Dates for future action on these items are based on the licensees' best estimates. Some future dates represent formal commitments whereas other dates are based on the licensees' best estimates and do not represent formal commitments by licensee management. Those dates which represent formal commitments have been verified by the staff to be within the limits of Confirmatory Orders or Board Decisions issued for each licensee or appropriate corrective action is being taken. In several instances, future actions for TMI-1 are scheduled at the next refueling outage. The decision to allow TMI-1 to resume operations and to specify what actions may be delayed until the next refueling outage will, of course, be the prerogative of the Commission.

For the purpose of this audit, an item was considered complete if it had been physically designed and installed, declared operable by the licensee, had written procedures implemented, and had all training requirements completed.

Table 2 provides a summary for every item audited utilizing this definition of complete. Additionally, the status of applicable technical specifications and post-implementation reviews were also determined for each item.

This study was not intended as a detailed review of the acceptability of the individual items. The staff already has or will in the future conduct detailed specific reviews separately. Where available, the results of staff reviews have been provided. However, post-implementation issues raised during these reviews, while a measure of the quality of the design or installation, are outside the scope of this audit in declaring items complete.

Section 2.0 of this report provides an overall status of licensee compliance with each item audited and a comparison among licensees of how they complied. Section 3.0 provides some generalized conclusions about the status of compliance among the licensees examined for these long term actions.

Appendix A contains a detailed description of how each B&W licensee addressed these NUREG-0737 items. Appendix B contains photographs of how licensees at selected B&W plants complied with NUREG-0737 requirements.

<u>TABLE 1</u>

I.A.1.3	Shift Manning
I.C.1*	Guidance for the Evaluation and Development of Procedures for Transients and Accidents.
I.D.1*	Control Room Design Reviews
I.D.2*	Plant Safety Parameter Display Console
II.B.1	Reactor Coolant System Vents
II.B.2	Plant Shielding
II.B.3	Post-Accident Sampling
II.D.1	Relief and Safety Valve Test Requirements
II.E.1.1	Auxiliary Feedwater System Evaluation
II.E.1.2	Auxiliary Feedwater System Initiation and Flow
II.E.4.1.2	Dedicated Hydrogen Penetrations
II.F.1	Accident Monitoring Equipment (Six Major Subsections)
	II.F.1.1 Noble Gas Monitors
	II.F.1.2 Iodine/Particulate Sampling
	II.F.1.3 Containment High-Range Monitor
	II.F.1.4 Containment Pressure
	II.F.1.5 Containment Water Level
	II.F.1.6 Containment Hydrogen
II.F.2	Instrumentation for Detection of Inadequate Core Cooling
III.A.1.2*	Upgrade Emergency Support Facilities
III.A.2.2*	Meteorological Data
III.D.3.4	Control Room Habitability

*Indicates a Supplement 1 item.

TABLE 2

NUREG 0737 ITEM	ANO-1	CRYSTAL RIVER	DAVIS BESSE	OCONEE	RANCHO SECO	TMI
I.A.1.3, Shift Manning	C	C	C	C	June 1985	C
I.C.1, ATOG	C	C	November 1984	October 1985	November 1984	C
I.D.1, CRDR	NS	NS	NS	NS	NS	May 1984
I.D.2, SPDS	C	August 1985	C	July 1985	November 1984	January 1985
II.B.1, RC Vents	C	C	C	October 1985	C	C
II.B.2, Plant Shielding	C	C	C	C	C	December 1984**
II.B.3, Post-Accident Sampling	May 1984	C	C	May 1984	C	C
II.D.1, Relief and Safety Valves	C	C	C	C	C	C
II.E.1.1, AFW Evaluation	Jan. 1985	August 1985	C	C	June 1986	Fall 1985*
II.E.1.2, AFW Initiation and Flow	Jan. 1985	August 1985	C	C	C	Fall 1985*
II.E.4.1.2, Hydrogen Penetrations	C	C	C	C	C	C

FOOTNOTES ARE ON NEXT PAGE

NUREG 0737 ITEM	ANO-1	CRYSTAL RIVER	DAVIS BESSE	OCONEE	RANCHO SECO	TMI
II.F.1.1, Noble Gas	C	C	C	C	C	C
II.F.1.2, Iodine/ Particulate	C	C	C	C	C	C
II.F.1.3, Containment Radiation	C	C	C	Jan. 1985	C	C
II.F.1.4, Containment Pressure	C	C	C	C	C	C
II.F.1.5, Containment Water Level	C	C	C	C	C	C
II.F.1.6, Containment Hydrogen	C	C	C	C	C	C
II.F.2, ICC	Summer 1986	August 1985	Fall 1986	Jan. 1987	Oct. 1986	Fall 1985*
II.A.1.2, Emergency Facilities	C	C	C	C	Nov. 1984	C
III.A.2.2, Met. Data	June 1984	July 1984	C	Dec. 1984	C	C
III.D.3.4, Control Room Habitability	C	C	C	NS	Nov. 1984	Fall 1985*

C = Complete
Date = Expected completion date (generally depends on refueling outage scheduled to start or end around the approximate date indicated). For TMI-1 all dates shift to next refueling outage if allowed to start-up. For Oconee, dates indicate when all three units will be complete.
NS = not scheduled yet. Physical modifications will be done after the human factors review is complete.
*Estimated period of the first refueling following restart assuming TMI-1 restart occurs in July 1984.
**Estimated schedule of eddy-current inspections of Steam Generators assuming TMI-1 restart occurs in July 1984.

2.0 STATUS AND COMPARISONS

This section provides a brief statement of the requirements of each individual NUREG-0737 item audited and a description of the status and major differences in how the B&W licensees complied with the guidelines. Each topic is followed by a tabular summary which displays the projected schedule for incomplete items.

I.A.1.3 Shift Manning

NUREG-0737 established interim requirements for minimum shift staffing of licensed personnel and administrative limits on overtime for critical personnel. Subsequently 10 CFR Part 50.54 (m) "Conditions of Licensees" was modified on July 11, 1983 to establish minimum shift manning levels and to specify the plant location for licensed operator personnel on shift.

All units except Rancho Seco presently meet the minimum staffing requirements in the control room specified in 10 CFR Part 50.54(m). Rancho Seco has requested an interim exemption from this requirement until June 1, 1985, based on selected compensatory actions until enough licensed operators can be qualified to meet the rule. Currently, the Rancho Seco plant is on a four shift basis with reduced manning, and has proposed going to five shifts with reduced manning by July 1984, and then to six shifts with full manning by June 1985. This proposal is currently under review by the NRC. Davis-Besse meets the requirements for the number of licensed operators, but is requesting an exemption to allow the shift supervisor to continue to occupy his office which is located about 20 feet outside the control room rather than to move to the control room. ANO-1 and Oconee meet the minimum staffing requirements on a five-shift basis. TMI-1 and Crystal River meet the minimum staffing requirements on a six-shift basis.

All licensees enforce administrative limits on overtime. However, ANO-1 uses overtime to meet its minimum staffing requirements for operational personnel on its current five shift operations. The overtime procedures used at Oconee are of some relatively minor concern to the NRC staff because of their unique staffing system, described in more detail later in this report.

Most units either have implemented or are trying to implement a six-shift rotation for personnel. A six shift rotation is desirable since it would minimize overtime and allow personnel adequate time for necessary training. ANO-1 contemplates requesting an exemption to the shift staffing rule to allow them to go to a six-shift rotation of personnel now.

Oconee uses a unique shift staffing system built around a five-shift rotation. First, all operating personnel are qualified on all three units and rotate through the units in a normal rotation with their shift section. Secondly, they normally work a sequential 48 hour work week followed by a 36-hour work week on a 12-hour workday basis. Overtime is done during "days off". Two weeks out of every ten weeks is spent in training in a normal forty hour work week. This operating schedule has been in effect for about a year and is currently being evaluated by the NRC staff. The staff has some relatively minor concerns about a normal 12-hour workday and its affect on operator performance. However, the licensee claims that operator performance has improved since they implemented the 12 hour workday due to increased morale.

The licensee for TMI-1 has performed well in comparison to other B&W licensees in meeting this requirement. Several other licensees are performing at the same level at TMI-1.

I.A.1.3 SHIFT MANNING

Plant	Meet Rule	Have Requested an Exemption
ANO-1	Yes	No - but are thinking of requesting one for reducing the number of licensed operators per shift to permit 6 shift operation now.
Crystal River	Yes	No
Davis Besse	Yes	Yes, desire exemption for location of SRO
Oconee 1, 2, 3	Yes	No
Rancho Seco	No	Yes, for interim period until June 1, 1985
TMI 1	Yes	No

I.C.1 Guidance For The Evaluation And Development Of Procedures For Transients And Accidents (Also Known As Emergency Operating Procedures Or Abnormal Transient Operability Guidelines-ATOG)

Licensees were required to develop symptom-oriented procedures to help operators cope with abnormal transients and emergencies. Each licensee was to (1) submit a Procedures Generation Package consisting of plant-specific guidelines, a writer's guide, a procedure validation program, and a training program description; (2) upgrade EOPs to be consistent with Technical Guidelines and an appropriate procedure writer's guide; (3) provide appropriate training of operating personnel on the use of upgraded EOPs prior to implementation of the EOPs; and (4) implement upgraded EOPs.

Each licensee has submitted a Procedures Generation Package. The NRC staff had indicated in NUREG-0737 its preference to review a generic set of guidelines and to have these form the basis for licensee preparation of plant-specific guidelines. Consistent with this approach, and by agreement with the B&W Owners Group, NRC reviewed the Oconee 3 guidelines submittal as though they would be applied generically to all B&W plants. At different times during the NRC review of the Oconee 3 guidelines, each of the B&W licensees began preparing its own plant-specific guidelines. At present, TMI-1, ANO-1 and Crystal River 3 have written the new procedures, completed all training, and implemented the new procedures. The licensee for Davis Besse will complete its procedure writing and personnel training by the end of 1984 and Rancho Seco will implement the procedures and complete personnel training by start-up from their 1984 refueling outage. Only Oconee, the "lead" plant, does not expect to complete their new written procedures until April 1985 and their training by October 1985.

The B&W Owners Group is currently developing a generic document to ensure that the emergency operating procedures of each licensee reflects the most up-to-date information in addressing new issues and in incorporating new operational methods. This document will be used for long term maintenance of the procedures required by the NRC. One example of such a new issue is the procedures for operators in the event of multiple steam generator tube ruptures. TMI-1 has developed guidelines for this event, which the B&W Owners Group and other licensees are examining for eventual adoption by all. NRC has reviewed the guidelines on this subject and the benefit of the review is available to all the licensees.

All licensees, as members of the B&W Owners Group, have proposed a program for resolving the long term open issues. This proposal is currently under review by the staff.

The licensee for TMI-1 has performed well in meeting this requirement. However most of the other B&W licensees are also expeditiously completing this effort.

I.C.1 EMERGENCY OPERATING PROCEDURES (ATOG)

Plant	Procedures Written	Training Complete**
ANO-1	C	C
Crystal River	C	C
Davis Besse	November 1984*	November 1984*
Oconee 1, 2, 3	October 1985	October 1985
Rancho Seco	November 1984*	November 1984*
TMI-1	C	C - in additon they have already implemented a multiple steam generator tube rupture procedure (part of the long term items to be resolved)

* This item will be completed during a refueling outage scheduled for this approximate date.
** Except for long term items as discussd in the NRC review of the Oconee submittal which may be generic to all B&W plants.
C Means complete

I.D.1 Detailed Control Room Design Review

Each licensee is to conduct a review of their control room to identify human engineering discrepancies. Based on this review, physical modifications are to be implemented when they would result in a significant reduction in risk. Each licensee is required to submit two separate reports in this review process. The first report is the Program Plan which describes in detail how the review is to be conducted. The second report is a Summary Report due at the end of the human factors review, which is to include any physical modifications necessary and a proposed schedule for their implementation.

The licensee for TMI-1 appears to be ahead of the other B&W licensees in completing this effort, based primarily on the early effort made soon after the TMI-2 accident. The licensee has submitted a combined Program Plan and Summary Report and asserts that all physical modifications it feels are necessary to date are complete. The major efforts left to complete under the licensee's program are to conduct a Task analysis and walk down of the control room with the ATOG procedures (I.C.1), a process that will be completed in April 1984 and described in a Supplemental Summary Report scheduled for a May 1984 submittal. These efforts may result in further physical modifications but the licensee does not expect any major modifications.

The licensees for ANO-1, Crystal River, Davis Besse, and Oconee have submitted their Program Plans. The licensee for Oconee has also completed their study and submitted a Summary Report and will submit their schedule for implementing the physical modifications shortly. The licensee for Rancho Seco submitted its Program Plan but has committed to provide an improved Program Plan by April 1984. Schedules are indicated in the following table for submitting Summary Reports.

The extent of physical modifications are still unknown for a majority of these plants. Even so, it is realistic to assume that major control room modifications will require at a minimum one refueling outage (two years) and possibly even longer.

The licensee for TMI-1 appears to be further along on this item for physical modifications than the other B&W licensees. However, the staff review of these human factors modifications may result in more physical modifications in the future.

I.D.1 DETAILED CONTROL ROOM DESIGN REVIEW

Plant	Program Plan	Summary Report	Physical Modifications
ANO-1	S	August 1985	Not Scheduled.
Crystal River	S	April 1985	Not Scheduled.
Davis Besse	S	June 1984***	Not Scheduled.
Oconee	S	S	Not Scheduled.
Rancho Seco	April 30, 1984	December 31, 1985	Not Scheduled.
TMI-1	S	May 1984*	C**

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* A Supplemental Summary Report
** Further modifications may be required in the future.
*** Initially the licensee scheduled this item for completion by April 1984 but an extension request until June has been submitted to the NRC staff.
C Means complete
S Means submitted

I.D.2 Plant Safety Parameter Display Console (SPDS)

Each licensee shall establish a Safety Parameter Display System which will continuously display information from which the plant safety status can be readily and reliably assessed by control room personnel. This unit shall be located so that it can be monitored easily by control room personnel. The minimum information to be provided shall be sufficient to provide information to plant operators about the following functional areas:

(1) Reactivity control

(2) Reactor core cooling and heat removal from the primary system

(3) Reactor coolant system integrity

(4) Radioactivity control

(5) Containment conditions.

The specific paramenters to be displayed shall be determined by the licensee.

Implementation of the SPDS may be affected by the implementation of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident". However, licensees have been instructed to implement an SPDS with the best available parameters to support the five primary safety functions.

Only two licensees, Davis Besse and ANO-1, have installed operational systems and have training completed for a minimum capability SPDS, and both have plans for upgrading their installed system. At Davis-Besse a contractor originally designed and installed their SPDS; although the licensees now feels that the equipment chosen has too slow a response time. They are upgrading the equipment using in-house expertise to ensure that the final system will be adequately responsive to their needs. ANO-1 has installed a rather complex SPDS which they designed in-house and refer to as their "basic" system. They plan to further upgrade the equipment, a process they expect to complete by June 30, 1984, which will include implementation of Regulatory Guide 1.97. Both ANO-1 and Davis Besse are developing computer software programs using in-house expertise.

Rancho Seco has installed a SPDS system for which the vendor has completed testing and is now familiarizing operators. Training on this new system will not be completed until startup after the next refueling outage in late 1984. The Rancho Seco SPDS, bought as a package from a vendor, is designed to satisfy four of the five functional areas of interest to the NRC. Data for displaying radioactivity control,the fifth area of interest, is supplied by a separate computer system already installed.

Licensees for TMI-1, Oconee, and Crystal River expect to install an SPDS system by no later than their next refueling outage, as indicated in the following chart. Oconee expects to follow ANO-1 and Davis-Besse in developing the necessary computer software in-house. The remaining licensees (TMI-1 and Crystal River) are in the process of purchasing a SPDS software package. TMI-1 has one computer generated pressure/temperature plot already implemented, but the rest of their system and plots are still in the design phase.

The licensee for TMI-1 is performing at a level equal to the average performance of other B&W owners for this requirement. Some licensees are ahead of TMI-1 in this area.

I.D.2 SPDS

Plant	Installation	Operability	Procedures	Training
ANO-1	C	C	C	C
Crystal River	Spring 1985***	Spring 1985***	Spring 1985***	Spring 1985***
Davis Besse	C	C	C	C
Oconee 1	Jan. 1985	Jan. 1985	*	*
2	July 1985	July 1985	Oct. 1984*	Oct. 1984****
3	Nov. 1984	Nov. 1984		
Rancho Seco	C	C	Nov. 1984***	Nov. 1984***
TMI-1	Jan. 1985	Jan. 1985	Jan. 1985	Jan. 1985

* Oconee operators rotate through all units. Once installed on one unit, procedures are implemented and training is completed for all operators.
*** Actually tied in with start-up date from a refueling outage.
C means complete.

II.B.1 Reactor Coolant System Vents

Each licensee was required to install reactor coolant system and reactor vessel head high point vents capable of being remotely operated from the control room. Initially the licensees were to install the vents and not make them operational until after NRC review. However, rule 10 CFR Part 50.44(c)(3)(iii), "Standards for Combustible Gas Control Systems in Light-Water-Cooled Power Reactors", was modified to require installation of and operability for the vents by end of the first outage of sufficient duration after July 1, 1982.

The licensees for all eight units have installed vents on the hot legs and pressurizer (one is using the PORV as the pressurizer vent) and declared them to be fully operable. Reactor head vents have been installed on ANO-1, Oconee 1, 2 and 3, and TMI-1. NRC has granted the licensees for Davis-Besse, Rancho Seco, and Crystal River an exemption until the first outage of sufficient duration after December 1985 from installing a reactor head vent. The licensee for Davis Besse is proposing to install a 2-1/2-inch continuous vent from the reactor head to one steam generator instead of installing a head vent system. If the Davis Besse design is approved, it will not be installed at the earliest until the summer of 1986 and the installation could extend for two refueling outages. The licensees for Rancho Seco and Crystal River have concluded that no head vent is necessary for B&W design plants. The integral system test program supported by the B&W Owners Group and the NRC includes, among other things, tests to demonstrate whether a reactor head vent is needed to maintain natural circulation under certain voiding conditions in the reactor coolant system for B&W designed plants. This program will also include tests to validate the Davis Besse proposed design.

With respect to hot leg vents, only one unit is different from the rest. To comply with this requirement, most licensees simply tapped into the hot leg and within a few feet completed the hot leg vent piping directly to the containment building atmosphere. However, at TMI-1, the hot leg vent piping goes a considerable distance (perhaps 20 to 30 feet) before venting to the building atmosphere. GPU utilized this design for personnel safety reasons; i.e., so the vents would exhaust to the atmosphere away from where personnel might be located. But this long a run of piping did have to be seismically mounted, a process which involved some extra effort.

Except at Oconee, all licensees have written procedures implemented and have conducted training on the use of these vents. The procedures are being written for Oconee as part of their ATOG (I.C.1), an effort that will not be completed until October 1985, even though the licensee declares the vents to be fully operable.

The licensee for TMI-1 will complete this effort prior to start-up and has complied with all our guidelines. But of course, these vents are required by rule and all licensees either meet the rule or have been granted an exemption.

II.B.1 REACTOR COOLANT SYSTEM VENTS

Plant	Installation	Operability	Procedures	Training	Technical Specifications
ANO-1	C	C	C	C	To be submitted in March 1984
Crystal River	Hot legs - C PZR - C Head-exemption request for no head vents	C	C	C	Submitted
Davis Besse	Hot legs - C PZR - C Head-exemption request to allow installation of 2½" vent to S/G	C	C	C	Submitted
Oconee 1, 2, 3	Complete all 3 units	Complete all 3 units	Oct. 1985 (tied to ATOG) at present they have none for operating the system	Oct. 1985	No changes proposed.
Rancho Seco	Hot legs - C PZR - C Head-exemption request for no head vents	C	C	C	Submitted
TMI-1	Hot legs - C PZR - C Head - C	C	C	C	Submitted

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* Indicates date is associated with the end of a refueling outage.
C Means complete

II.B.2 Plant Shielding

NRC requirements, in general, called for licensees to review all their emergency procedures and determine what additional shielding or procedure changes were necessary to prevent over exposure to personnel under accident conditions. The accident conditions postulated are those described in Regulatory Guides 1.3 and 1.4. Under these conditions, releases were assumed equivalent to 50% of the core radioiodine and 10% of the core noble gases. These conditions also assumed that 1% of the core solids are contained in the primary coolant. The goal under these circumstances is to limit personnel exposure to a maximum of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident (i.e., GDC 19 limits). Our requirements did not specify any particular accident scenario nor did the staff post-implementation review concentrate on any one accident scenario.

All licensees have completed their plant shielding study. Physical modifications necessary for this task included the installation of long extensions (i.e., reach rods) or remote operators on some valves and the building of extensive shield walls to protect critical areas. On some plants, these modifications have resulted in a reduction in the radiation exposure of operating personnel on a normal or routine basis.

With the exception of TMI-1, all units have completed and declared operable the physical modifications and/or procedure modifications and associated personnel training necessary to complete this task. The remaining efforts on TMI-1 are the modifications requiring a bypass pipe and remote valve operators and controls in the decay heat removal system. These modifications will permit operators to implement the emergency procedures necessary to maintain core flow and avoid boron precipitation following a postulated cold leg LOCA. The TMI-1 licensee is proposing to complete this effort by December 1984 subject to plant availability.

Based on the need for TMI-1 modifications to satisfy emergency procedures related to the boron precipitation problem, all of the other B&W licensees were asked whether or not they specifically evaluated this event. All B&W licensees, except for Florida Power Corporation (for Crystal River), stated that the particular accident scenario was addressed in their design or the plant shielding study. The capability to mitigate the effects of this scenario without over-exposing personnel exists at Crystal River but must be added to the operating procedures. Florida Power has committed to upgrade this procedure by July 1984.

The licensee for TMI-1 is behind the other B&W licensees in completing this effort.

II.B.2 PLANT SHIELDING

Plant	Installation	Operability	Procedures	Training	Technical Specifications
ANO-1	C	C	C	C	NA
Crystal River**	C	C	C**	C	NA
Davis Besse	C	C	C	C	NA
Oconee 1, 2, 3	C	C	C	C	NA
Rancho Seco	C	C	C	C	NA
TMI-1	Dec. 1984*	Dec. 1984*	Dec. 1984*	Dec. 1984*	NA

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* Subject to plant availability. This is the estimated schedule of eddy-current inspections of Steam Generators assuming TMI-1 restart occurs in July 1984.

** Crystal River is complete subject to the need for procedure changes related to measures needed to avoid the boron precipitation problem. These procedure changes will be complete and implemented by July 1984.

C Means complete

II.B.3 Post-Accident Sampling

The NRC requirement is that each licensee have the capability of sampling and analyzing the reactor coolant and containment building atmosphere under accident conditions and be able to predict core damage in a three hour time frame without radiation exposures to any individual exceeding the criteria of GDC-19 (Appendix A of 10 CFR Part 50 - i.e., 5 rem whole body and 75 rem to extremities). Meeting the three-hour limit provided the most difficulty for the licensees and required the design of new systems.

All units except Oconee Unit 3 have installed a post-accident sampling system, but only TMI-1, Crystal River, Davis-Besse, Oconee 1 and 2, and Rancho Seco have systems that are operational. Oconee Unit 3 will have a fully installed and operational system by the end of its next refueling outage (approximately May 1984).

All licensees except TMI-1 have installed extensive new sampling systems to meet this requirement. The most extensive and elaborate systems have been designed to serve as both the normal and post accident sampling systems. Both ANO-1 and Crystal River have installed this type of equipment. At ANO, a new building was constructed to house just the sampling systems for both ANO-1 and ANO-2. The sampling systems at ANO-1 and Crystal River, which can be operated entirely from a remote location, have the capability for conducting inline analysis as well as drawing undiluted grab samples. ANO-1 is having problems with their Orion analysis equipment for chloride and boron and hopes to resolve the issue by April 1984. ANO-1 does, however, have grab sample capability installed and operational.

Total completion of the Crystal River system is pending delivery of a new on-line dissolved hydrogen analyzer to replace the analyzer presently in use. The present interim analyzer was installed to meet the provisions of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations". The new analyzer is scheduled for delivery in April 1984. The licensee considers the post-accident system fully operational.

The ANO-1 and Crystal River systems go to extreme measures to minimize personnel exposure even when taking grab samples. At Crystal River, for example, there are six uniquely designed lead pigs which automatically mate with their sample system when put into place. Once the sample is drawn, the lead pig can be moved around on its own dolly until a crane can be attached to put it on a truck.

The licensees for Oconee, Davis Besse, and Rancho Seco have installed a totally separate sampling system to be utilized under post-accident conditions. These systems are all designed for remote operation to reduce personnel exposure. The system at Rancho Seco conducts inline analysis and has the capability of taking diluted grab samples if necessary. Diluted samples reduce personnel exposure, although they make the analytical techniques more difficult. Although the licensee for Rancho Seco declares this system is operable, the present system has reliability problems with its installed analytical equipment. (The ion chromatograph, in particular, because of its location in an area which can have high ambient temperature).

Both the Davis Besse and Oconee designs take grab samples for analysis on site. Unlike other licensees who bought custom systems from vendors, Duke Power designed the Oconee systems in-house for taking diluted grab samples. For Oconee Units 1 and 2, the licensee claims the system is fully operable, although they are experiencing problems in verifying design values on the systems as-built. The licensee for Davis Besse states their system is fully operable with no problems.

The licensee for TMI-1 utilized a design that resulted in minimum changes to the plant. The modifications entailed rerouting sample lines to the normal sample sink and providing additional shielding around these lines. Under accident conditions, an operator, or series of operators, would manually draw an undiluted sample and carry it to a lead pig for transfer to the laboratory about 15 feet away. There is extensive laboratory equipment available on site to analyze the sample once it is obtained. The licensee has performed person-motion analysis of operator exposure that would result from this sampling scheme and concluded that GDC 19 exposure limits will not be exceeded. The licensee declares the system fully operable. Also, the TMI-1 system is not presently capable of drawing a sample from the containment building sump or from a depressurized reactor vessel. All the other licensees have installed this capability. After several discussions with the NRC staff, the licensee has agreed to modify their PASS to permit sampling from the Decay Heat System to provide the capability of obtaining a reactor coolant sample from a depressurized reactor coolant system.

NRC is conducting a post-implementation review at the headquarters, with a follow-up by Regional staff to ensure that the design values described in the safety evaluation are met. The major components of this requirement include obtaining a potentially highly radioactive sample, conducting a sophisticated analysis on it, and predicting core damage based on the results.

All licensees concentrated on sampling and analysis techniques and assumed they would know how to predict core damage based on the results. As part of the post-implementation review, the staff is requiring all licensees to develop procedures or a methodology for predicting core damage from anticipated analytical results. Providing the rest of the system is satisfactory, NRC considers the requirement completed if the licensee makes a commitment to developing a procedure to estimate core damage within a reasonable time frame (generally one year or less) provided an interim procedure based on radionuclide measurements is implemented.

Compared to the other B&W licensees, TMI-1 has installed a simplified sampling system which meets our minimum requirements, including personnel exposure. However, in the event of a core degrading accident, the TMI-1 system will result in more personnel exposure compared to the designs used by the other B&W licensees.

II.B.3 POST ACCIDENT SAMPLING

Plant	Installation	Operability	Procedures***	Training	Technical Specifications
ANO-1	C	No-problems with equipment for Cl and boron. Schedule unknown but probably 6-8 weeks (May 1984)	C	C	To be submitted in March 1984
Crystal River	C	C****	C	C	To be submitted in April 1984
Davis Besse	C	C	C	C	Submitted
Oconee 1	C	C but having problems in verifying some design values	C	May 1984**	To be submitted in April 1984
2	C		C	May 1984**	To be submitted in April 1984
3	May 1984*	May 1984*	C	May 1984**	To be submitted in April 1984
Rancho Seco	C	C-but ion chromato graph is not reliable due to heat problems	C	C	Submitted
TMI-1	C	C	C	C	To be submitted in April 1984

* Actual date depends on end of refueling outage.
**Oconee personnel are qualified for all 3 units. Unit 3 PASS is identical to Units 1 and 2.
***Does not include procedures for predicting core damage.
****Crystal River is replacing the installed dissolved hydrogen analyzer and expects to complete this effort by May 1984.
C means complete.

II.D.1 Relief And Safety Valve Testing Program

The NRC requirement is that each licensee conduct testing to qualify the reactor coolant system relief and safety valves under anticipated operating conditions for design-basis transients and accidents. This requirement includes the necessary testing and qualification of associated piping and the block valves.

All B&W units joined in and completed the EPRI test program to qualify their valves. Rancho Seco, ANO-1, Crystal River and Oconee had to make some valve adjustments and some plants added more piping supports to qualify their valves. Both TMI-1 and Davis-Besse had to remove a loop seal system as a result of this testing program. In addition, the Davis Besse valves were physicaly some distance from the pressurizer. As a result of the testing, two safety valves were relocated closer to the pressurizer and associated discharge piping eliminated. All modifications are complete. The staff review is still in progress and no final safety evaluation has been issued to any licensee.

The licensee for TMI-1 has performed consistent with all the other B&W licensees in meeting this effort.

II.D.1 RELIEF AND SAFETY VALVE TESTING PROGRAM

Plant	Part of EPRI*	End Results
ANO-1	Yes	Made ring adjustments and added some piping supports.
Crystal River	Yes	Made ring adjustments
Davis Besse	Yes	Removed loop seals and relocated valves closer to pressurizer and eliminated some discharge piping.
Oconee 1, 2, 3	Yes	Made ring adjustments and added some piping supports.
Rancho Seco	Yes	Made ring adjustments.
TMI-1	Yes	Removed loop seals.

*The EPRI test program is complete. All licensees have submitted plant specific evaluations based on the EPRI test program. Staff review is in progress.

II.E.1.1 Auxiliary Feedwater Evaluations

The NRC requirement is that each licensee evaluate its auxiliary feedwater (AFW) systems using the Standard Review Plan and associated Branch Technical Position and provide justification for any differences. They are also required to provide a simplified AFW reliability analysis and reevaluate their AFW system design flow rate basis to ensure adequate AFW flow under various accident conditions. Any area for which insufficient justification exists must be upgraded. Pre-implementation reviews have been completed on all the plants and a post-implementation review will be conducted in accordance with NUREG-0737.

All B&W licensees have submitted their simplified AFW reliability analysis, have evaluated their AFW systems against the Standard Review Plan and associated Branch Technical Position, and have reevaluated their AFW system design flow rate criteria to ensure adequate AFW flow under various accident conditions. These evaluations identified a number of corrective actions.

Only the licensees for Oconee 1, 2, and 3, and Davis-Besse have completed all plant modifications required as a result of this review. The original AFW system at Davis Besse was designed as safety grade, so few modifications were necessary for this unit to meet this requirement. However, three open items remain in the NRC review regarding the contents of the Davis-Besse Technical Specifications for the AFW system. Duke Power chose to upgrade the AFW systems at Oconee early and obtained necessary materials from a plant they had under construction. Although the NRC has issued a safety evaluation report stating that Oconee satisfies the requirements of II.E.1.1, there was an open item in the Oconee review involving the capability to deliver AFW flow following a seismic event or tornado. This open item is now being pursued as a plant specific review topic, but is a direct result of the NRC review for II.E.1.1.

TMI-1, ANO-1, Rancho Seco and Crystal River still have modifications to make, all of which are expected to be complete by no later than a refueling outage as shown in the following table. Most of the plant modifications necessary at Rancho Seco are complete. The remaining modifications at Rancho Seco include missile protection and upgrading of flow control valves to seismic category I. The licensee is currently working with the staff in an attempt to resolve the staff decisions concerning missile protection. TMI-1, ANO-1, and Crystal River have a good number of physical modifications to complete. Crystal River may elect to install a new condensate storage tank rather than provide additional protection for the existing tank. This modification may take until mid-1987 to complete if this course of action is taken.

The licensee for TMI-1 is consistent with most of the other B&W licensees in meeting these requirements. Several of the B&W licensees have met these requirements earlier.

II.E.1.1 AUXILIARY FEEDWATER EVALUATION

Plant	Installation	Operability	Procedures	Training	Technical Specifications
ANO-1	Jan. 1985*	Jan. 1985*	Jan. 1985*	Jan. 1985*	To be determined
Crystal River	Refueling V* (Spring 1985) for most items. New condensate storage tanks if required=Refueling VI (mid-1987)	Spring 1985	Spring 1985	Spring 1985	To be determined
Davis Besse	C	C	C	C	Submitted***
Oconee 1, 2, 3	C	C	C	C	C
Rancho Seco	June 1986*	June 1986*	June 1986*	June 1986*	To be determined
TMI-1	Fall 1985**	Fall 1985**	Fall 1985**	Fall 1985**	To be determined

*Date tied to refueling outage for completion.
**Estimated period of the first refueling following restart assuming TMI-1 restart occurs in July 1984.
***Three staff positions concerning technical specifications have not been incorporated.
C means complete.

II.E.1.2 Auxiliary Feedwater System Initiation and Flow

The NRC requirement is for all licensees to provide a control grade automatic initiation feature for their AFW system and then upgrade it to meet safety grade requirements. Licensees are also required to install safety grade indication of auxiliary feedwater flow to each steam generator in the control room.

The licensees for Davis-Besse, Rancho Seco and Oconee state that they have installed and tested the system modifications and their systems are operable. They have also implemented operating procedures and completed operator training.

ANO-1, Crystal River, and TMI-1 will complete the necessary modifications at their next refueling outage. Each licensee has a control grade automatic initiation system which must be upgraded to safety grade requirements. TMI-1 currently has a control grade initiation system that was reviewed and found acceptable for restart by the Licensing and Appeal Boards and the licensee has committed to long term upgrades of their system. TMI-1 has installed safety grade AFW flow indication.

The licensees for ANO-1, Crystal River and Rancho Seco are installing an Emergency Feed Initiation and Control (EFIC) system. The EFIC system is an AFW control system which goes beyond our requirements by automatically controlling AFW flow based on steam generator level and by using automatic logic circuits to prevent certain accidents, such as overfill or feeding a steam generator with a ruptured feed line. The licensee for Rancho Seco has concluded that they have met the NRC requirements for II.E.1.2 and is installing the EFIC system to provide additional safety grade protection. The NRC staff is reviewing the licensees claim but, nevertheless, agree that once EFIC is installed it will more than satisfy NRC concerns in this area. ANO-1 and Crystal River are installing their EFIC as part of an integrated program to upgrade their AFW system to meet NRC requirements.

As far as schedule is concerned, the TMI-1 licensee is consistent with half of the B&W licensees. TMI-1 will meet our minimum requirements. However, several B&W licensees are installing systems which exceed our requirements.

II.E.1.2 AFW INITIATION AND FLOW

Plant	Installation	Operability	Procedures	Training	Technical Specifications
ANO-1	Jan. 1985* Installing EFIC	Jan. 1985*	Jan. 1985*	Jan. 1985*	To be determined
Crystal River	August 1985* Installing EFIC	August 1985*	August 1985*	August 1985*	To be submitted in Feb. 1985
Davis Bessel	C	C	C	C	Submitted
Oconee 1, 2, 3	C	C	C	C	C
Rancho Seco***	C-but installing EFIC	C	C-but will be re-written after EFIC	C	Present tech. specs still valid
TMI-1	Fall 1985**	Fall 1985**	Fall 1985**	Fall 1985**	To be determined.

*Date depends on a refueling outage and is the estimated start-up date from that outage.
**Estimated period of the first refueling following restart assuming TMI-1 restart occurs in July 1984.
***NRC is evaluating whether the NUREG-0737 requirements for safety grade initiation have been met. NRC agrees that once EFIC is installed these requirements will be satisfied.
C means complete.

II.E.4.1.2 Dedicated Hydrogen Penetrations

The NRC requirement in this area is that any plant not having an internal combustible gas control system must install a dedicated penetration for an external hydrogen recombiner or purge system for combustible gas control. All B&W plants have a dedicated penetration and meet NRC requirements. For the most part, these penetrations already existed and the licensees met our requirement by simply declaring them dedicated for an external hydrogen recombiner.

The licensee for TMI-1 has installed two external hydrogen recombiners, one came from TMI-2. Duke Power owns two recombiners and keeps one in storage at the Oconee site. The other licensees have evaluated the availability of hydrogen recombiners from either a vendor or another utility in case an accident should occur to their plant. Based upon these evaluations, the remaining licensees have determined that they can obtain and install and make operable an external hydrogen recombiner, if necessary, after an accident but before an uncontrolled hydrogen-oxygen recombination would take place which results in a loss of safety function.

The licensee for TMI-1 is definitely well ahead of the other B&W licensees in this effort because it has installed two hydrogen recombiners.

II.E.4.1.2 DEDICATED H_2 PENETRATIONS

Plant	Dedicated Penetration	Status of Recombiners
ANO-1	Yes	Will obtain one from offsite.
Crystal River	Yes	Will obtain one from offsite.
Davis Besse	Yes	Will obtain one from offsite.
Oconee 1, 2, 3	Yes	Duke owns two - one is on site and the other is elsewhere, both are in storage.
Rancho Seco	Yes	Will obtain one from offsite.
TMI-1	Yes	Two installed recombiners.

II.F.1 Accident Monitoring Equipment - This Item Has Six Major Subsections As Follows:

II.F.1.1 Noble Gas Monitors

The NRC requirement is that each licensee install noble gas effluent monitors with an extended range designed to function during accident conditions as well as during normal operations. Multiple monitors are considered necessary to cover the ranges of interest.

This item is complete for all plants, where complete means physically installed, operable, with procedures implemented, and training completed.

ANO-1, Crystal River, Davis Besse, and Rancho Seco all have similar systems. Basically a system for sampling for both II.F.1.1 Noble Gas and II.F.1.2 Iodine and Particulates is contained in a single skid, permanently attached to one sampling point. These skid mounted systems are found at each vent location on site. However, at Rancho Seco the licensee has not installed a noble gas monitor on the containment purge line (they are installed at other vent locations). The NRC has found this acceptable since the purge valves are not opened during reactor operation. Each skid contains a pump which draws an air sample through permanently installed piping, passes it through several particulate filters (for II.F.1.2), and then collects it in a sample chamber that monitors for noble gas. TMI-1 has a different system which samples directly for noble gas. All systems are operable and have no reliability problems.

The licensee for Oconee met this requirement by installing a general area radiation monitor beside each unit's exhaust stack. These monitors have a reliability problem, which the licensee believes is caused because moisture gets into the connection between the detector and pre-amp. In all the other applications, this radiation monitor has been very reliable; normally, the pre-amp is connected directly to the detector. At Oconee, the detector and pre-amp have to be separated by about a foot, which allows moisture problems to occur. The licensee declares these detectors fully operable, even though they must be replaced about every three months. The licensee is working with the vendor to solve this problem and believes that its latest solution will work permanently.

The licensee for TMI-1 has performed consistent with all the other B&W licensees in meeting this requirement.

II.F.1.1 NOBLE GAS MONITOR

Plant	Installation	Operability	Procedures	Training	Technical Specifications
ANO-1	C	C	C	C	March 1984.
Crystal River	C	C	C	C	Submitted
Davis Besse	C	C	C	C	No changes proposed.
Oconee 1, 2, 3	C	C	C	C	To be submitted in April 1984
Rancho Seco	C	C	C	C	April 1984.
TMI-1	C	C	C	C	Submitted

C means complete.

II.F.1.2 Iodine/Particulate Sampling

The NRC requirement is that each licensee provide the capability for effluent monitoring of radioiodines under accident conditions by absorption on charcoal or other sampling media, followed by onsite laboratory analysis.

This item has been installed, operationally tested, with procedures implemented, and training completed on all units. No reliability problems have been reported. Most licensees obtain this sample in conjunction with sampling for Noble gases as described in II.F.1.1. At Rancho Seco, the licensee has not installed these collection units on the containment purge line (they are installed at other vent locations). The NRC has found this acceptable since the purge valves are not opened during reactor operation. Some licensees are upgrading their systems to provide for automatic inputs to a computer but all claim to already meet our requirements.

The licensee for TMI-1 has performed consistent with all the other B&W licensees in meeting this requirement.

II.F.1.2 IODINE/PARTICULATE SAMPLING

Plant	Installation	Operability	Procedures	Training	Technical Specifications
ANO-1	C	C	C	C	March 1984
Crystal River	C	C	C	C	Submitted
Davis Besse	C	C	C	C	C
Oconee 1, 2, 3	C	C	C	C	To be submitted in April 1984
Rancho Seco	C	C	C	C	Submitted
TMI-1	C	C	C	C	Submitted

C means complete.

II.F.1.3 Containment High-Range Monitor

The NRC requirement is that each licensee install high-range radiation monitors inside the containment building. A minimum of two such monitors are required, which are to be physically separated and environmentally qualified.

These monitors have been installed on all units and are fully operational, except for Oconee 1, and Oconee 3. Procedures have been implemented and training of personnel is complete at all units.

At Crystal River both high-range detectors are fully operable, but one electronics train has developed internal problems and was returned to the vendor for trouble shooting. This instrument will be reinstalled by May 1984. The other electronics train is fully operable and can be utilized for either detector if necessary. Therefore, the licensee declares this system operational with two working detectors and one operable electronics train.

The licensee for Oconee discovered an electrical interference problem between the high voltage power supply and the detector output signal where they pass through a common connector. This electrical interference problem is of sufficient magnitude that the licensee declared the systems inoperable. It has been corrected on Oconee Unit 2 and will be corrected at the next refueling outage for Units 1 (January 1985) and 3 (May 1984).

There are no reported reliability problems for any other licensee. The licensee for TMI-1 has performed consistent with all the other B&W licensees in meeting this requirement.

II.F.1.3 CONTAINMENT HIGH-RANGE MONITOR

Plant	Installation	Operability	Procedures	Training	Technical Specifications
ANO-1	C	C	C	C	To be submitted in March 1984
Crystal River	C	C**	C	C	Submitted
Davis Besse	C	C	C	C	No changes proposed
Oconee 1	Jan. 1985*	Jan. 1985*	C	C	To be submitted in April 1984
2	C	C	C	C	To be submitted in April 1984
3	May 1984*	May 1984*	C	C	To be submitted in April 1984
Rancho Seco	C	C	C	C	Submitted
TMI-1	C	C	C	C	Submitted

*Date actually depends on end of a refueling outage.
**Only one electronic channel is fully operable. The second channel had to be sent back to the vendor for repairs. Reinstallation is expected to be completed in May 1984.
C means complete.

II.F.1.4 Containment Pressure

The NRC requirement is that each licensee provide a continuous indication of containment building pressure in the control room for each operating reactor. The measurement and indication capability is to provide a range of information to include three times the design pressure of the containment structure for concrete, four times the design pressure for steel, and -5 psig for all containment structures.

This item has been installed, operationally tested, with procedures implemented, and training completed on all units. There are no reported reliability problems or major differences in the quality of the installed systems.

The licensee for TMI-1 has performed consistent with all the other licensees in meeting this requirement.

II.F.1.4 CONTAINMENT PRESSURE

Plant	Installation	Operability	Procedures	Training	Technical Specifications
ANO-1	C	C	C	C	March 1984 to be submitted
Crystal River	C	C	C	C	Submitted
Davis Besse	C	C	C	C	Submitted
Oconee 1, 2, 3	C	C	C	C	To be submitted by April 1984
Rancho Seco	C	C	C	C	Submitted
TMI-1	C	C	C	C	Submitted

C means complete.

II.F.1.5 Containment Water Level

The NRC requirement is that each licensee provide a continuous indication of containment building water level in the control room for all plants. Two ranges of level indication are required for PWRs. The narrow range instrument must have a range from the bottom to the top of the containment building sump. The wide range instrument must have a range from the containment building floor to the containment building elevation equivalent to a 600,000 gallon capacity.

Both wide and narrow range containment building water-level monitoring devices have been installed in all the units. All systems have been operationally tested, all procedures have been implemented, and all training is completed. At Oconee Units 1, 2, and 3, a narrow-range monitor is installed but is not environmentally qualified. The licensee will replace these narrow range monitors with qualified equipment at each unit's next refueling outage: Unit 1 - January 1985; Unit 2 - January 1986; Unit 3 - May 1984. There are no reported reliability problems.

The licensee for TMI-1 has performed consistent with all the other B&W licensees in meeting this requirement.

II.F.1.5 CONTAINMENT WATER LEVEL

Plant	Installation	Operability	Procedures	Training	Technical Specifications
ANO-1	C	C	C	C	To be submitted in March 1984
Crystal River	C	C	C	C	Submitted
Davis Besse 1	C	C	C	C	Submitted
Oconee 1 2 3	Wide range - C Narrow range - one exists but not environmentally qualified. It will be replaced: Unit 1 - Jan. 85* Unit 2 - Jan. 86* Unit 3 - May 84*	C	C	C	To be submitted in April 1984
Rancho Seco	C	C	C	C	Submitted
TMI-1	C	C	C	C	Submitted

*Date depends on refueling outage.
C means complete.

II.F.1.6 Containment Hydrogen

The NRC requirement is that each licensee provide a continuous indication of hydrogen concentration in the containment building atmosphere in the control room. The monitoring device must be capable of detecting a range of between 0 and 10% hydrogen concentration under both positive and negative ambient pressure.

Containment building hydrogen monitors have been installed on all units and are fully operable. These systems have been operationally tested, procedures have been implemented, and all training is complete. There are no reported reliability problems and no major differences in the quality of the installations.

II.F.1.6 CONTAINMENT HYDROGEN

Plant	Installation	Operability	Procedures	Training	Technical Specifications
ANO-1	C	C	C	C	To be submitted in March 1984
Crystal River	C	C	C	C	Submitted
Davis Besse	C	C	C	C	No changes proposed
Oconee 1, 2, 3	C	C	C	C	To be submitted in April 1984
Rancho Seco	C	C	C	C	Submitted
TMI-1	C	C	C	C	Submitted

C means complete.

II.F.2 Instrumentation For Detection of Inadequate Core Cooling

The NRC requirement is that each licensee propose supplemental instrumentation to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). This requirement can be divided into three parts as follows: (1) saturation margin monitors (SMM), (2) core exit thermocouples (CETs), and (3) reactor coolant inventory tracking system. During the Commission review of ICC instrumentation requirements (SECY-82-407), it was concluded that differential pressure (Dp) measurement systems for B&W reactors should have a minimum range extending from the top of the candy cane to the low point in the hot leg and from the vessel upper head to the bottom of the hot leg.

First every plant was required to install instrumentation to monitor the margin to saturation temperatures as part of short-term lessons learned requirements. All plants have installed this instrumentation, made it operable, implemented procedures, and completed training. Typically, each unit monitors both hot leg temperatures and pressure, and instrumentation automatically displays a margin-to-saturation temperature for the given pressure for each loop. Crystal River and Davis Besse have added a separate input for incore thermocouple temperatures to measure a margin to saturation of the core exit coolant.

Secondly, all plants are required to provide displays in the control room for core exit thermocouples in selected guadrants and to upgrade the CET channels and saturation margin monitors to meet NUREG-0737 (II.F.2) design requirements. TMI-1 has a control grade resistance temperature detector (RTD) for the hot leg temperature indication, and the licensee is having troubles finding a suitable safety-grade replacement. At present, the licensee is reviewing whether or not they should upgrade the present RTD as an alternate to replacement. Also the digital indicators in the control room are not seismically qualified at TMI-1. The licensee will either qualify the indicators or purchase qualified replacements. A core exit thermocouple system has been installed at TMI-1 but it has not been seismically or environmentally qualified yet. Otherwise, TMI-1 design provisions to meet SMM and CET upgrading requirements have been approved by the staff. Upgrading provisions for the SMM and CETs for Rancho Seco and Crystal River 3 and for the ANO-1 SMM have also been approved. All of the other B&W licensees have submitted upgrading plans that are under review by the staff. The licensee for ANO-1 committed to use existing core exit thermocouples until the gamma-thermometer system is installed in 1986 as discussed later in this report. However the NRC has presently stated that the gamma-thermometer system will not satisfy the NUREG-0737 requirements for core exit thermocouples.

There is a possible generic problem in seismically qualifying some of the connectors located inside containment for the incore thermocouple system. The B&W owners group is working on a solution to this problem. This effort will be conducted in accordance with the environmental qualification program.

Third, every plant is required to install a system for tracking reactor vessel coolant inventory. An order was issued in December 1982 which directed the B&W licensees to propose a design and implementation schedule for such a system. Dates for specific submittals were included in the orders. Plant-specific reviews by the staff of the tracking system conceptual designs are complete. A status report of licensees compliance with the orders was provided in SECY-83-259 dated June 29, 1983.

The following describes how each B&W licensee is proposing to meet the reactor coolant inventory tracking system rquirements of II.F.2:

(1) The licensee for ANO-1 is developing a RADCAL gamma thermometer, including a program of confirmatory testing and qualification for application to the ANO-1 B&W design. Initial engineering on a system using the Combustion Engineering (CE) Heated Junction Thermocouple (HJTC) system for in-vessel monitoring has also been performed. A level of effort on the latter approach will continue until the gamma thermometer development program is completed and its acceptability is confirmed. Using this approach, installation is expected in June 1986 for ANO-1, corresponding to a projected refueling outage. A decision on the acceptability of the RADCAL system must be reached by the licensee and the NRC by July 1984 so that if it is a negative decision there will still be time to procure and install the alternate HJTC system during the same refueling outages.

The proposed instrument probes will traverse both the reactor core and the reactor vessel (RV) head and thus provide a more complete range for inventory tracking compared to the heated junction thermocouple (HJTC) which terminates at the top of the core. The sensors above the core will be differential heated thermocouples operating on the same principle as the Combustion Engineering (CE) HJTC design. These probes have potential installation advantage since they are of small diameter and mechanically very similar to existing in-core instrument assemblies, but the adequacy of the differential heated thermocouples above the core to give an unambiguous indication of liquid/steam interface must be evaluated and demonstrated by testing. The performance of the gamma thermometers within the core is of less concern since the information to be provided by these instruments is in excess of the data needed to satisfy the functional requirements for ICC monitoring, so long as the information they provide is not misleading to the operators.

For hot leg inventory monitoring of ANO-1, the licensee has proposed the installation of a stillwell consisting of a vertically run small diameter pipe or section of tubing between the hot and cold legs of the reactor coolant system (RCS) piping with the upper end connected to the existing high point vent and the lower end connected to the existing cold leg drain line of each steam generator loop. A gamma thermocouple probe with differential thermocouple sensors will be installed in the stillwell to monitor water level in the hot leg (approximately 40-50 feet span).

The present ANO-1 Core Exit Thermocouple (CET) system is not Class 1E design and display is available only on the plant computer. The licensee has proposed to incorporate thermocouples at the core exit on four gamma thermometer probes in lieu of upgrading the existing CET system. This is not consistent with the NUREG-0737 requirement of four thermocouples per quadrant.

(2) The licensee for Crystal River Unit 3 will install a Reactor Coolant Inventory Tracking System (RCITS) based on Dp measurements across vertical elevations of the hot leg and the RV to infer coolant level when the reactor coolant pumps (RCPs) are tripped, plus the use of RCP motor current measurements and pump inlet temperatures to infer coolant inventory trends when the RCPs are running. The concept also includes density compensation for Dp measurements due to temperature effects on reference leg and reactor coolant liquid density.

Differential pressure measurements cover a wide range measurement from the top of the candy cane to the bottom of the hot leg, plus a narrow range measurement from the top of the RV head to the bottom of the hot leg. A total of four Dp transmitters will be used to provide redundancy. Each pair of wide and narrow range transmitters will be independently powered by Class 1E instrumentation power. They will be mounted within the containment area. Seal chambers will be located at the high point of each reference leg to keep the legs full of water.

(3) The licensee for Davis Besse Unit 1 proposes to use a B&W design hot leg level measurement system (HLLMS) based on Dp measurement, a reactor coolant pump monitoring program (RCPMP), and a RV head to hot leg vent line with a thermocouple installed in the vent line for use in maintaining subcooling in the vent line under any controlled natural circulation cooldown and depressurization condition.

The licensee will complete installation of its total system by installation of the RV head to hot leg vent line during the 1986 refueling outage, if the staff approves it for use as a RV head vent and also agrees that the vent line can also be used in lieu of the head level monitoring system. Refer to the write-up of II.B.1, Reactor Coolant System Vents, for the testing being done to evaluate the effects of this vent line on transient safety analysis.

(4) The licensee for Oconee has proposed a full range inventory monitor utilizing a Dp cell reading between the high point vent and the decay heat removal drop line. A reactor vessel inventory monitor will use the Dp between a control rod drive mechanism and the decay heat drop line. The licensee has stated that the instrument is likely to be of the

Westinghouse variety, although the decision on the vendor has not yet been made. A pump power monitor will be utilized for when the Reactor Coolant Pumps are running.

Tentatively the schedule for the three units is likely to be as follows:

Unit 1: Completion by mid-1986

Unit 2: Completion by late-1986

Unit 3: Completion by late 1985 to subsequent refueling outage, depending on vendor supply timing.

(5) The licensee for Rancho Seco proposes to use a B&W designed Dp level system to measure level in the hog leg (bottom to top) and the CE HJTC system for monitoring the reactor vessel head inventory. However, the CE HJTC system has not been finally selected, and is being evaluated versus two other options, a Dp system in conjunction with a pump power monitor and a gamma thermometer. The licensee has committed to make a final selection by June 1984.

The hot leg level indication system is scheduled for installation during the next refueling outage (November 1984). Based on the committments in the licensee's submittal, one of the three head level options will be installated during the subsequent refueling outage.

(6) The licensee for TMI-1 will provide a full hot leg differential pressure coverage with an upper tap on the top of the hot leg and the lower tap on the decay heat removal drop line. Full reactor vessel coverage is provided with the upper tap on the top of the vessel and the lower tap using the same decay heat removal drop line tap. Portions of the system are presently installed. The remainder of the system will be installed at the first refueling outage after restart.

If the reactor coolant pumps are running, the level trending subsystem will measure void fraction from a pump power algorithm. These measurements will be performed every six minutes (pre-reactor trip) or every 30 seconds (post-reactor trip).

There are some subsections of this requirement where the licensee for TMI-1 is ahead of the other B&W licensees. However, in the aggregate the licensee for TMI-1 will perform consistent with all the other B&W licensees when this task is complete.

<u>II.F.2 INADEQUATE CORE COOLING</u>

Plant	Installation	Operability	Procedures	Training	Technical Specifications
ANO-1					
Sat. monitor	C	C	C	C	C
Thermocouples	Summer 1986**	Summer 1986**	Summer 1986**	Summer 1986**	To be determined
Level	"	"	"	"	"
Crystal River					
Sat. monitor	C	C	C	C	C
Thermocouples	Aug. 1985**	Aug. 1985**	Aug. 1985**	Aug. 1985**	To be determined
Level	"	"	"	"	"
Davis Besse					
Sat. monitor	C	C	C	C	C
Thermocouples	Aug. 1986**	Aug. 1986**	Aug. 1986**	Aug. 1986**	To be determined
Level	Aug. 1986**	Aug. 1986**	Aug. 1986**	Aug. 1986**	To be determined
Oconee 1, 2, 3					
Sat. monitor	C	C	C	C	C
Thermocouples	Jan. 1, 1987***	Jan. 1, 1987***	Jan. 1, 1987***	Jan. 1, 1987***	To be determined
Level	"	"	"	"	"
Rancho Seco					
Sat. monitor	C	C	C	C	C
Thermocouples	June 1986**	June 1986**	June 1986**	June 1986**	To be determined
Level	"	"	"	"	"
TMI-1					
Sat. monitor	C-monitor not seismic	C	C	C	C
Thermocouples	C****	C	C	C	C
Vessel level	Fall 1985*	Fall 1985*	Fall 1985*	Fall 1985*	To be determined

* Providing plant is shutdown for an adequate period. Otherwise, next refueling outage. For TMI-1, this is the estimated period of the first refueling outage following restart assuming TMI-1 restart occurs in July 1984.

**Actual date depends on start-up from a refueling outage.

***For all 3 units - this means the final one is complete.

****Except for seismic and environmental qualification

C means complete.

III.A.1.2 <u>Emergency Facilities Upgrade</u>

The NRC requirement is that each licensee establish three emergency response facilities: the Operational Support Center (OSC), the Technical Support Center (TSC), and Emergency Operations Facility (EOF). The OSC is basically a staging area to which qualified people can report in an emergency and all units have their designated staging areas. The EOF and TSC are rooms or buildings which had to be built to specific habitability requirements.

In the long run, completion dates for these facilities are affected by completion dates for implementing Regulatory Guide 1.97. Each licensee has a committed date by which it will propose a schedule for implementing Regulatory Guide 1.97. However, NRC has informed licensees that their completion of modifications for Regulatory Guide 1.97 will be considered as a separate issue and that these EOF and TCS facilities must be established immediately.

All units, except Rancho Seco, have an operational TSC. For the most part, the TSCs are dedicated rooms located near the control room. However at Oconee, the TSC is normally the I&C shop and must be converted (by obtaining equipment from designated cabinets, etc.) in an emergency. Rancho Seco installed a TSC but it does not meet NRC ventilation requirements. The TSC ventilation is directly tied into the control room ventilation at Rancho Seco and both are being upgraded in accordance with the control room habitability review. This modification is expected to be completed for startup following the refueling outage scheduled at the end of 1984.

All units except for Oconee have a fully qualified and operational EOF. Although there is an EOF at the Oconee site, the licensee has requested an exemption to locate their EOF 125 miles away at corporate headquarters. The Commission is currently reviewing the need to upgrade the protective factor of the ANO-1 EOF. The main problem with the existing Oconee EOF and ANO-1 EOF is that they both are located less than one mile from their respective units.

For the most part, licensees have spent a considerable amount of money developing their EOFs.

The licensee for TMI-1 has performed consistent with the other B&W licensees in meeting this requirement.

III.A.1.2 EMERGENCY FACILITIES UPGRADE

Plant	Installation	Operability	Procedures	Training
ANO-1	C	C	C	C
Crystal River	C-except for SPDS and EDAS	C	C	C
Davis Besse	C	C	C	C
Oconee 1, 2, 3	Licensee says this is complete except for SPDS. However, they have requested an exemption for locating their EOF, to have it 125 miles away. If this is not granted, their present EOF does not meet NRC habitability requirements	C	C	C
Rancho Seco	Nov. 1984*	Nov. 1984*	Nov. 1984*	Nov. 1984*
TMI-1	C	C	C	C

*Actual date tied in with start-up following a refueling outage.
C means complete.

III.A.2.2 Meterological Upgrade

The NRC requirement is that each licensee assure that adequate methods, systems, and equipment exists for assessing on a real time basis and monitoring the meterological aspects of actual or potential offsite consequences of a radiological emergency. This requirement is composed of four essential elements that can be categorized into three functions: meterological measurements, diffusion assessment and incorporation into dose assessment, and communications. The assessment capabilities should provide inputs to determine the necessity for notification, sheltering, and evacuation of the general public. Strictly speaking, III.A.2.2 is only concerned with the meterological aspects of emergency response capabilities. But overall, the meterological upgrades are very closely associated with dose assessment. Therefore, this audit also included the licensees capabilities for dose assessment utilizing the data from the meterological upgrades. NRC reviews will be done in the future as part of an overall appraisal of emergency responses facilities. The implementation of sections of Regulatory Guide 1.97 via the issuance of supplement 1 of NUREG-0737 may affect some of the detectors utilized for this upgrade. However, the implementation of Regulatory Guide 1.97 will be done on a separate schedule.

The meterological upgrade consists of appropriate meterological sensors (wind speed, wind direction, and measurement of atmospheric stability) and an appropriate means to model the diffusion and transport of potential releases in the area of the site. Alternate sources of input for environmental data must be provided in case the primary meterological tower is inoperable. When a source term (provided either by another program or by the main program itself) is inputed into this model, the end result will predict the plume pathway and the necessity for notification, sheltering, and/or evacuation of the general public. Again III.A.2.2 is primarily associated only with the meterological inputs to the model, but the major item of importance is dose assessment.

TMI-1, Davis Besse, and Rancho Seco have installed meterological capabilities which basically provides for modeling the environment and which the licensees claim will satisfy NRC requirements. Oconee, Crystal River, and ANO-1 will complete this requirement in 1984 as indicated in the table following this section.

Going beyond the limited scope of III.A.2.2 and emphasizing dose assessment, the licensees can meet these requirements by having environmental and source term sensors with a manual input into a computer program. Most of the licensees are upgrading their system to provide some automatic sensor inputs directly to the computer program. However, none of the licensees probably will have a system as extensive as TMI-1 for dose assessment. Not only does the TMI-1 model allow for initial predictions of the plume pathway, but it is sophisticated enough to allow modifications to the model based on field monitoring by radiological sensors (beyond our requirements). Also TMI-1 has permanently installed field radiation monitors which are connected via telephone to a central computer to provide a direct readout of exposures in a given area. All licensees have permanently installed field monitors, but none of the B&W licensees other than TMI-1 can read field sensors remotely, and this goes beyond our requirements.

The licensee for TMI-1 appears to be ahead of the other B&W licensees in meeting this requirement; not from a meterological aspect but from a dose assessment aspect because it has installed systems which go beyond the basic NRC requirements. However, the NRC appraisal of emergency response facilities has not been conducted yet so our assessment of their overall capabilities is unknown.

III.A.2.2 METEROLOGICAL UPGRADE

Plant	Installation	Operability	Procedures	Training
ANO-1	June 1984	June 1984	June 1984	June 1984
Crystal River	June 1984	July 1984	July 1984	July 1984
Davis Besse	C	C	C	C
Oconee 1, 2, 3	May 1984	December 1984	December 1984	December 1984
Rancho Seco	C	C	C	C
TMI-1	C	C	C	C

C means complete.

III.D.3.4 Control Room Habitability

The NRC requirement is that licensees 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 (Criteria 19, "Control Room", of Appendix A, "General Design Criteria for Nuclear Power Plants", to 10 CFR Part 50).

Davis Besse, ANO-1 and Crystal River all state that this effort is complete. Davis Besse concluded in its study that no physical modifications are necessary. ANO-1 and Crystal River had a few modifications to install, generally in the form of toxic gas detectors which could isolate the control room on an alarm condition. These modifications are complete and operational, procedures are implemented, and training is complete. Crystal River had a problem with frequent breaking of the detector sampling paper causing the control room to be isolated, but have solved this problem with a weekly maintenance check of the detectors.

After initially committing to certain modifications, the licensee for TMI-1 did a probabilistic risk assessment (PRA) which subsequently concluded that no modifications are necessary for their plant. However, the NRC Standard Review Plan does not accept probabilistic arguments for on-site hazards and the staff requested GPU Nuclear to reevaluate. GPU Nuclear recently produced an outline of their reevaluation program which the staff is now evaluating.

Rancho Seco and potentially Oconee have relatively major plant modifications to complete this effort. Basically Rancho Seco is having to install a new ventilation system for the control room. Because of these upgrades and other site needs, the Rancho Seco licensee has decided to concurrently install a new site diesel system to handle the increased electrical loads. These modifications are planned to be completed at Rancho Seco by the end of their next refueling outage, approximately November 1984.

At one time, Duke Power stated that this activity was complete for the Oconee Unit 1/2 and Unit 3 control rooms. However after conducting tests requested by NRC, Duke Power realized that they could not put a positive pressure on the Unit 1/2 control room and could only put a marginal positive pressure in the Unit 3 control room. Thus, Oconee is not able to totally isolate the Unit 1/2 control room. Duke Power is presently in the engineering/design phase of solving this problem so no real schedules are known. If major modifications are necessary, it may take as long as two refueling outages for each unit before they are complete at Oconee.

In the final analysis, the licensee for TMI-1 is slower than some of the other B&W licensees in completing this effort, in part, because the NRC staff did not accept one of the licensees proposed solutions.

III.D.3.4 CONTROL ROOM HABITABILITY

Plant	Installation	Operability	Procedures	Training	Technical Specifications
ANO-1	C	C	C	C	C
Crystal River	C	C	C	C	Submitted
Davis Besse	Study complete, no physical mods necessary	NA	C	C	No new ones necessary
Oconee 1, 2, 3	At one time they thought it was closed but now they know they cannot pressurize the Unit 1/2 control room and Unit 3 is marginal. They are still in the engineering/design phase and have no idea of an installation schedule. It may take as long as a two refueling outages for Unit 1 (Fall 1986) and Unit 2 (Fall 1987) before completion.				
Rancho Seco	Nov. 84* (Putting in new diesels, new ventilation system)	Nov. 84*	Nov. 84*	Nov. 84*	Not determined if needed yet
TMI-1	Fall 1985**	Fall 1985**	Fall 1985**	Fall 1985**	To be determined

*Date really tied in with start-up following a refueling outage.
C means complete.
**Estimated period of the first refueling following restart assuming TMI-1 restart occurs in July 1984.

3.0 Conclusions

After reviewing the efforts of the eight B&W licensees, the staff concludes that each licensee has some areas where they have done a better job than other licensees and areas where they have not done as well as other licensees. No licensee stands out as superior to all others for all items audited.

All B&W licensees either have met or plan on meeting their formal schedules for implementing these requirements. However, GPU Nuclear appears to be ahead of the other B&W licensees for item I.D.1, Control Room Design Reviews, in that GPU Nuclear has completed all modifications to date they feel are necessary and the other B&W licensee have not yet started their modifications. GPU Nuclear must still conduct a Task Analysis using the ATOG procedures to complete the Control Room Design Review. Of course, the NRC review may indicate the need for further modifications. However for II.B.2, Plant Shielding, GPU Nuclear is the only B&W licensee experiencing a long delay in completing the necessary modifications. A similar delay may be experienced at TMI-1 for III.D.3.4, Control Room Habitability as a result of NRC review of the licensees proposal. For all the other items audited, GPU Nuclear's commitments for implementation schedules are on a par with the other B&W licensees.

All B&W licensees claim that their systems, once declared operable, will meet or exceed the minimum requirements of NUREG-0737. Regarding the quality of work performed, GPU Nuclear appears to be ahead of the other B&W licensees for item III.A.2.2, Meterological Data, not so much from the meterological aspect but from the dose assessment aspect because GPU Nuclear has installed additional equipment and capabilities beyond NRC requirements; and for item II.E.4.1.2, Dedicated Hydrogen Penetrations, where GPU Nuclear has gone beyond NRC requirements by installing two external hydrogen recombiners. For item II.B.3, Post Accident Sampling, GPU Nuclear is utilizing a manually operated system, whereas all the other B&W licensees have installed remotely operated sampling systems to reduce personnel exposure. For all the other items audited, GPU Nuclear's quality is on a par with the other B&W licensees.

In the aggregate, GPU Nuclear is on a par with all the other B&W licensees in meeting NUREG-0737 requirements.

APPENDIX A

Detailed Plant Status

A detailed discussion on the licensees effort to implement the guidelines of selected issues from NUREG-0737. A generalized description of the NUREG-0737 requirements is provided at the start of each section. However for a detailed description of the NRC criteria, refer to the appropriate section of NUREG-0737.

I.A.1.3 SHIFT MANNING

NUREG-0737 Position

The NUREG-0737 position on minimum shift staffing was changed by rulemaking. Basically, 10 CFR 50.54 paragraph (m) "Conditions of licensees" was modified on July 11, 1983 to establish minimum shift manning levels for licensed personnel. The effective date of the rule was January 1, 1984. The minimum shift complement for any condition other than cold shutdown (200°F) or refueling is two NRC licensed senior operators and two NRC licensed reactor operators for a single-unit nuclear plant. In addition, NUREG-0737 required each licensee to establish administrative controls on the use of overtime for critical personnel with the desire of limiting overtime to the extent practicable.

The following describes how each B&W licensee met these guidelines:

ANO-1

Description:

The licensee currently meets the requirements of 10 CFR 50.54 (m) operating on a five shift rotation; but intends to operate on a six shift schedule after more licensed personnel become available with SRO qualifications. The required number of personnel for six shifts are expected to be avilable subsequent to examinations scheduled for March 1984. The licensee has 18 STA's, who also perform other station duties.

Procedures:

Complete.

Technical Specifications:

Tech Specs reflecting the new requirements are in preparation.

NRC Review Status:

The Region will verify compliance.

Crystal River Unit 3

Description:

Crystal River Unit 3 has a six-shift operations organization with each shift meeting the staffing requirements of 10 CFR 50.54(m).

The licensee has incorporated limitations on the use of overtime into the administrative procedures for operation of Crystal River Unit 3. These provisions have been implemented in the Operations Section Implementation Manual (OS1M).

Amendment No. 38 was issued on April 17, 1981 to require a shift technical advisor (called operations technical advisor at Crystal River) on all shifts during modes 1, 2, 3 and 4. This amendment also stipulated, in Technical Specification Table 6.2-1, the shift manning requirements at Crystal River.

Procedures:

Complete

Technical Specifications:

Complete

NRC Review Status:

The Region will verify compliance.

Davis Besse, Unit No. 1

Description:

The licensee currently is operating the Davis Besse Plant in conformance with the requirement of 10 CFR 50.54(m), "Conditions of Licenses" concerning "Licensed Operators Staffing at Nuclear Power Plants", which requires at least two licensed senior operators and two licensed operators on shift at all times and one licensed senior operator in the control room whenever the plant is operating in any mode other than cold shutdown or refueling. In addition, the Technical Specifications require two non-licensed operators and one shift technical advisor on shift at all times. The licensee is currently operating with six shifts. The licensee has requested an exemption from the requirement of a licensed senior operator in the control room whenever the plant is operating in any mode other than cold shutdown or refueling. The reason for the request is that the Shift Supervisor, an SRO, spends part of his time in his office adjacent to the Control Room on shift workload matters.

The licensee's administrative procedures include the overtime requirements delineated in IE Circular 80-02 and NUREG-0737. The administrative procedures include licensed operators (RO & SRO), equipment operators, health physicists, and I&C and maintenance technicians.

Procedures:

Complete

Technical Specifications:

The licensee has been requested to submit proposed technical specifications relating to the overtime limits. The licensee plans to submit proposed technical specifications concerning the staffing following the resolution of the exemption request relating to the licensed senior operator in the control room. Completion of this item is estimated to be May 1984.

NRC Review Status:

The exemption request concerning the licensed senior operator in the control room is under staff review.

Oconee Units 1, 2 and 3

Description:

The licensee has implemented the NRC requirements for minimum shift crew as of January 1, 1984. The shift manning is designed around a 12 hour shift, five shift rotation. Each operator works a sequential 48 hour week followed by a 36 hour week. Overtime is worked on days off. Every operator cycles through all three units every three months. Two weeks out of 10 are spent in training. Every SRO since 1979 has received STA qualification. All new hires for STA's will have a baccalaureate degree. The licensee has backup SRO's on the operational engineering staff.

Procedures:

Complete.

Technical Specifications:

Tech Specs reflecting the new requirements are in preparation.

NRC Review Status:

The Regions will verify compliance.

Rancho Seco

Description

The licensee currently does not meet the shift manning requirements of 10 CFR 50.54. By letter dated September 22, 1983 the licensee requested an extension to the deadline to implement the requirements of 10 CFR 50.54 regarding shift staffing until June 1, 1985. The licensee currently has a 4 shift operation with 2 SROs and 1 RO on each shift. The licensee expects to have sufficient personnel hired and trained to staff 6 shifts with 2 SRO's and 2 RO's per shift by June 1, 1985. Presently the licensee is operating on a four shift basis, will gradually increase to a six shift rotation with reduced operator staff then increase staffing levels on each shift to meet the rule by June 1, 1985.

The current technical specification require that a shift technical advisor be available for all shifts other than during could shutdown. In addition, limitations on the use of overtime has been incorporated into the technical specifications.

Procedures

Complete.

Training

The licensee currently has 2 people in SRO training and 10 people in RO training.

Technical Specification

No proposed technical specification changes have been submitted.

NRC Review Status

As noted above by letter dated September 22, 1983 the licensee requested an extension to the deadline for implementing the requirements of 10 CFR 50.54 regarding shift staffing until June 1, 1985. On January 12, 1984 the licensee met with the staff to discuss the extension request. As a result of the meeting on January 20, 1984 the licensee provided further details and justification regarding the extension request. The staff is reviewing the extension request and expects to issue an SER before the end of May 1984.

Three Mile Island Unit 1

Description:

TMI-1 currently has a sufficient number of qualified personnel to comply with the requirements of 10 CFR 50.54(m)(2) and their Technical Specifications, operating on a six-shift rotation. TMI-1 Technical Specifications require that a shift technical advisor be on shift whenever Tave is greater than 200^{o}F. The Technical Specifications also require written procedures on Plant Staff Overtime to limit the amount worked by staff performing safety-related functions in accordance with NRC Policy Statement on working hours (Generic Letter 82-12).

Procedures:

Complete.

Technical Specifications:

Complete.

NRC Review Status:

Complete.

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

NUREG-0737 SUPPLEMENT 1 POSITION

The use of human factored, functional oriented, emergency operating procedures will improve human reliability and the ability to mitigate the consequences of a broad range of initiating events and subsequent multiple failures or operator errors, without the need to diagnose specific events. Licensees are required to reanalyze transients and accidents and prepare Technical Guidelines. Licensees are to upgrade their emergency operating procedures in accordance with the Technical Guidelines. The analysis will serve as the basis for integrating upgraded emergency operating procedures and the control room design review and verifying the SPDS design.

Each licensee was to (1) submit a Procedures Generation Package consisting of plant specific guidelines, a writers guide, a procedures validation program and a training program description, (2) upgrade EOPs to be consistent with Technical Guidelines and an appropriate procedure Writer's Guide, (3) provide appropriate training of operating personnel on the use of upgraded EOPs prior to implementation of the EOPs, and (4) implement upgraded EOPs.

The following describes how each licensee is meeting these guidelines:

ANO-1

Description:

The licensee has implemented procedures based on the Abnormal Transient Operators Guidelines (ATOG). The licensee submitted its Procedures Generation Package consisting of the plant specific ATOG together with the writers guide, validation program description and training program description in April 1983. The NRC staff has not yet reviewed this submittal. The licensee has implemented the ATOG into Emergency Operating Procedures which are currently available in the control room. The licensee believes that these procedures are approximately consistent with the Oconee 3 ATOG which has been formally reviewed by the NRC staff. Further upgrades to the Emergency Operating Procedures will proceed along the same lines as the Babcock and Wilcox Owners group efforts in this area. As an example of such upgrades, the Owners Group has endorsed the TMI-1 steam generator tube rupture guidelines as the recommended approach for other plants, and ANO-1 has indicated that the procedures will be modified as necessary. The ANO-1 procedures currently incorporated available functions from the safety parameter display system and installed R.G. 1.97 instruments.

Procedures:

Implemented per current symptom based guidelines.

Training:

Completed per current symptom based guidelines.

Technical Specifications:

Not applicable

NRC Review Status:

NRC staff reviewed the Oconee 3 ATOG as being representative of all Babcock and Wilcox reactors. The SER documenting the staff's review was issued in September 1983. An SSER was issued in December 1983 on the issues of ATWS and natural circulation. The staff review of the Procedure Generation Package is not yet completed. Selected plants procedures and training will be subject to an NRC audit.

Crystal River Unit 3

Description:

The licensee revised the Crystal River Unit 3 Abnormal and Emergency Operating Procedures in July 1983 using guidance provided in the "Abnormal Transient Operating Guidelines" (ATOG) prepared by the B&W Owners Group. The revised procedures are symptom-oriented rather than event-oriented for the most part. Prior to revising and implementing the procedures, the licensee submitted a "Procedures Generation Package" to the NRC in March 1983. Future changes to these procedures will follow resolution by the B&W Owners Group of comments published by the NRC in Generic Letter No. 83-31 dated September 19, 1983. The generic letter forwarded the staff's Safety Evaluation of the Oconee Nuclear Station Unit 3 ATOG to all owners of B&W pressurized water reactors.

Procedures:

Complete. Sufficient validation has been performed to give reasonable confidence that the procedures are effective in implementing the intent of NUREG-0737, Item I.C.1.

Training:

Complete for all licensed operators (included classroom and simulator instruction).

Technical Specifications:

N/A

NRC Review Status:

The NRC issued a Safety Evaluation of Oconee Unit 3 ATOG in Generic Letter 83-31 as a lead plant, which included a number of details which needed further work by the B&W Owners Group. The staff notes that one of these items, procedures for dealing with multiple steam generator tube ruptures (SGTR), has not yet been resolved with the Owners Group and is therefore not addressed in the Crystal River Unit 3 Emergency Operating Procedures. B&W will distribute the guidelines for the SGTR procedures, to include multiple tube failures, for licensee use by the end of 1984. The nature of this item dictates an ongoing effort by the NRC, B&W Owners Group, and individual licensees to achieve the intended end result. However, this item is considered implemented to the extent possible at this point in time for Crystal River Unit 3. A pre-implementation review of the Procedures Generation Package was conducted in May 1983, which did not identify any major deficiencies that would preclude implementation. A detailed NRC review of the licensee's Procedures Generation Package is continuing, with assistance from Battelle Northwest Laboratories. Results of that review are expected in March or April 1984. Action required on the part of the licensee will depend on the staff's comments resulting from this review.

Davis Besse, Unit No. 1

Description:

The Davis Besse Procedures Generation Package, which is required by Supplement 1 of NUREG-0737, was submitted March 1, 1984. The licensee has indicated that preparation of specific plant operating procedures is presently underway based on the NRC Safety Evaluation of ATOG for Oconee 1, 2 and 3 per NRC Generic Letter 83-31 dated September 19, 1983.

Procedures:

The preparation of Davis Besse specific operating procedures is presently underway with completion scheduled before the next refueling outage scheduled in late 1984.

Training:

Operator training is scheduled to be complete prior to startup following the next refueling outage which is scheduled for late 1984.

Technical Specifications

No proposed Technical Specifications have been identified yet.

NRC Review Status

The NRC staff provided a Safety Evaluation (SE) of the ATOG based on the lead plant review of the Oconee 1, 2 and 3 ATOG in September 1983. The SE requested the licensees to provide sufficient documentation in the form of plant-specific ATOG and Transient Information Documents so that the NRC can perform comparisons with the ATOG version evaluated for the Oconee facility. A number of items associated with the guidelines were identified in the SER which need further work by the B&W Owners. The licensee is participating with the B&WOG in this activity and the schedule is dependent upon this effort. The licensee plans to provide the staff with the steam generator tube rupture guidelines for multiple tube ruptures as soon as possible after the issue is resolved with the Owners Group.

Oconee Units 1, 2 and 3

Description:

The licensee has submitted the Procedure Generation Package for Oconee 3 consisting of the Abnormal Transient Operating Guidelines (ATOG) the writers guide, the validation program description and the training program description. The ATOG has been approved by NRC staff. The applicant has stated that the emergency operating procedures based on these guidelines are being written and will be implemented by October 1985, until which time the units will use existing procedures. The licensee is participating in the Babcock and Wilcox Owner's Group activity in developing the generic guidelines.

Procedures:

The symptom based procedures are scheduled to be implemented by October 1985.

Training:

The training program will include the symptom based procedures by October 1985.

Technical Specifications:

Not applicable.

NRC Review Status:

NRC staff reviewed the Oconee 3 ATOG as being representative of all Babcock and Wilcox reactors. The SER documenting the staff's review was issued in September 1983. An SSER was issued in Decmeber 1983 on the issues of ATWS and natural circulation. Selected plants procedures and training will be subject to an NRC audit.

Rancho Seco

Description:

The licensee participated in the B&W Owners Group (B&W OG) program in the development of Abnormal Transient Operating Guidelines (ATOG). The licensee is following the B&W OG schedule for the long range items.

On January 31, 1984 the licensee submitted their Abnormal Operating Procedures Generation Package which includes their plant-specific Technical Guidelines, the Emergency Operating Procedures (EOP) Writers Guide and a Description of the Program to validate EOP's

Emergency Procedures Guidelines for handling multiple tube failures in steam generator were not included in the submitted generation package. The licensee indicated that within approximately six-weeks they will provide us with a revision to the single tube failure ATOG which will include multiple tube failures.

Procedures

The licensee is currently preparing its plant specific Emergency Operating Procedures utilizing the submitted Procedures Generation Package. The licensee plans to have the procedures prepared, approved and in place prior to start up from the next refueling outage (November 1984).

Training

Simulator training utilizing Generic Procedures will start the end of February 1984. Plant specific Emergency Procedure Training will start in the September-October 1984 time frame.

NRC Review Status

The NRC issued a generic safety evaluation report based on the Oconee Unit 3 ATOG in Generic Letter 83-31. The report delineated a number of items (long term items) which needed further work by the B&W Owners Group.

The Procedures Generation Package is now under staff review.

Three Mile Island Unit 1

Description:

With respect to TMI-1, the plant-specific technical guidelines were developed from: (1) the TMI-1 ATOG developed by B&W from the Oconee ATOG, (2) additional information on anticipated transients without scram and natural circulation, and (3) GPU Nuclear tube rupture guidelines which were used in lieu of the B&W ATOG tube rupture guidelines. The other aspects of the program, specifically the writers guide, procedures validation program, and training program description have all been developed and submitted for staff review.

Procedures:

Written, validated, and implemented.

Training:

Classroom training of the operators was complete at the time of the January 1984 site visit; and training at the B&W simulator was completed in February. The simulator training concluded with sessions observed by senior plant management in which the operators responded to unannounced emergency scenarios.

Technical Specifications:

Not applicable.

NRC Review Status:

Review of the TMI-1 Procedures Generation Package consisting of the TMI-1 ATOG, technical guidelines, writers guide, validation plan and training program description is in progress and will be complete before restart. The NRC is also conducting a detailed review of the TMI-1 tube rupture guidelines and procedures principally stemming from the TMI-1 steam generator repair program review. Additional reviews will be continuing on the generic (Oconee) ATOG to resolve open items.

I.D.1 CONTROL ROOM DESIGN REVIEWS

NUREG-0737 Supplement 1 Position

Each licensee is to conduct a control room design review to identify human engineering discrepancies. The objective of the control room design review is to "improve the ability of nuclear power plant control room operators to prevent accidents or cope with accidents if they occur by improving the information provided to them" (from NUREG-0660, Item I.D.1). As a complement to improvements of plant operating staff capabilities in response to transients and other abnormal conditions that will result from implementation of the SPDS and from upgraded emergency operating procedures, this design review will identify any modifications of control room configurations that would contribute to a significant reduction of risk and enhancement in the safety of operation. Decisions on whether to make physical changes should be carefully reviewed by persons competent in human factors engineering and risk analysis.

Each licensee is required to submit two separate reports in this review process. First there is a Program Plan which describes in detail how the review is to be conducted. Secondly, there is a Summary Report due at the end of the human factors review which is to describe physical modifications necessary and a proposed schedule for their implmentation.

The following describes how each licensee is meeting these guidelines:

ANO-1

Description:

The licensee has established an organization and engaged the services of a human factors consultant to perform the review. A program plan was submitted to NRC in November 1983. The summary report from the detailed control room design review is scheduled to be submitted by August 1985. The licensee has estimated that actual modifications to the control room may extend into two refueling outages after the human engineering deficiencies have been identified.

Design/Installation:

Design modifications are expected to be noted in the summary report to be submitted in August 1985.

Operability Tests:

Not applicable at this time.

Procedures:

Changes to current procedures would proceed on a schedule consistent with control room modifications.

Training:

Changes to the current training program would proceed on a schedule consistent with control room modifications.

NRC Review Status:

Evaluation of the licensee's Program Plan is complete. Further review will be done after the summary report submittal.

Crystal River Unit 3

Description:

As required by this NUREG-0737 item, the licensee is in the process of conducting a control room design review for the purpose of determining which human factors engineering deficiencies should be corrected and defining the specific method of correction in each case. The review is intended to result in an integrated and coordinated program to improve the human factors aspects of the main control room. The licensee submitted its Detailed Control Room Design Review Program Plan in October 1983 for review by the NRC. Evaluation of the Program Plan by the NRC was completed in December 1983. The licensee will submit a summary report documenting the results of this review and the planned implementation schedule in April 1985.

Design/Installation:

Schedule to be submitted in April 1985. No installation (modifications to the control room) will commence until the NRC has issued a Safety Evaluation following receipt of the summary report and the licensee has resolved any open items.

Operability Tests:

Schedule to be established on the basis of the installation schedule.

Procedures:

Schedule to be established on the basis of the installation schedule.

Training:

Schedule to be established on the basis of the installation schedule.

Technical Specifications:

To be determined.

NRC Review Status:

NRC comments on the licensee's design review program plan were issued to the licensee by letter dated February 7, 1984. A meeting to discuss and resolve those comments is scheduled with the licensee on April 5, 1984. Further NRC review will be contingent upon receiving the licensee's summary report in April 1985.

Davis Besse, Unit No. 1

Description

The Davis Besse DCRDR program plan was submitted June 15, 1983. The staff found that the plan is, in general, well structured and is responsive to the major requirements of the DCRDR. The staff provided comments to the plan in December 1983. The Summary Report, which the licensee scheduled for April 1984, will identify modifications necessary and a date for submitting the schedule for implementing the modifications. The licensee has subsequently requested that the submittal date be extended to June 1984, and this request is under staff review. The Summary Report will be built around the ATOG procedure but will not include all of the Regulatory Guide 1.97 requirements.

Design/Installation

Tentatively scheduled to begin June 1984. The schedule will be provided after the submittal of the Summary Report. Procedures and necessary training will be determined after identification of the necessary modifications.

NRC Review Status:

The NRC staff review and comments concerning the Program Plan was provided to the licensee December 1983. The staff indicated that the plan, in general, was well structured and responsive to the major requirements of the DCRDR. The staff review of the Summary Report will result in a Safety Evaluation Report.

Oconee Units 1, 2 and 3

Description:

The licensee has completed the detailed control room design review for the combined control room for Units 1 and 2, and the separate control room for Unit 3; and has submitted the Summary Report on February 16, 1984 describing the human engineering discrepancies revealed by the review. The modifications to the control rooms are to be made during the implementation phase of this program. The licensee has submitted an implementation schedule which is presently being reviewed by the NRC staff.

Design/Installation:

The Summary Report contains descriptions of the design changes which are to be made. Installation will follow consistent with the implementation schedule.

Operability Tests:

Not applicable at this time.

Procedures:

Changes to current procedures would proceed on a schedule consistent with control room modifications.

Training:

Changes to the current training program would proceed on a schedule consistent with control room modifications.

NRC Review Status:

The Summary Report and implementation schedule is under NRC review.

Rancho Seco

Description

In accordance with NUREG 0737 Item I.D.1, the licensee plans to conduct a control room design review to improve the human factors aspects of the main control room. On September 30, 1983, the licensee submitted its Detailed Control Room Design Review Program Plan. Subsequent to this the licensee decided to revise its submittal and has hired a contractor to prepare a revised plan which will be submitted to the NRC by April 30, 1984. The licensee Summary Report documenting the results of the Control Room Design Review is scheduled to be submitted by December 31, 1985.

Design/Installation

The licensee has constructed a full-size mockup of its control panel which will be used for trying proposed changes before they are made in the control panel. The licensee indicates that required control room modifications will be determined following NRC review of the report.

Operability Tests

Schedule to be established.

Procedures

The licensee proposes to have procedures modifications that would result from their Control Room Design Review in place by the time it submits the Summary Report.

Training

Training schedule to be established.

NRC Review Status

The NRC comments on the licensee's Detailed Control Room Design Review Program Plan submitted on September 30, 1983 were issued on December 22, 1983. In our letter, we suggested that the licensee meet with us to discuss our concerns. The licensee has requested the meeting be delayed until after April 30, 1984 by which time it will have prepared and submitted a revised program plan.

Three Mile Island Unit 1:

Description:

GPU Nuclear submitted the TMI-1 control room design review program description and Summary Report on January 16, 1984. This submittal describes the TMI-1 control room design review which began in early 1980 in conjunction with the restart proceedings and which GPU Nuclear now considers to be essentially complete. The review included fabrication of a full-scale control room markup for procedure walk-throughs, review of the control panels using human factors criteria, quantitative and qualitative measurements of the control room environment, review of operator responsibilities for fundamental plant control functions, and a review of the alarm system. The NRC staff also performed an audit of the TMI-1 control room in 1981 and documented its findings in NUREG-0752 and supplement 1. The GPU Nuclear and NRC reviews were entered into evidence in the TMI-1 Restart proceedings, and the licensing board subsequently prescribed a program of corrective actions. GPU Nuclear has since completed these actions and considers Item I.D.1 complete except for a task analysis of the ATOG emergency procedures which will be completed and submitted May 31, 1984. GPU Nuclear has in place an ongoing human factors review program for proposed plant modifications.

Design/Installation:

Complete.

Operability Tests:

April 1984.

Training:

Complete.

Technical Specifications:

Not applicable.

NRC Review Status:

Review of program report and summary description in progress.

I.D.2 PLANT SAFETY PARAMETER DISPLAY SYSTEM (SPDS)

NUREG-0737 Supplement 1 Position

Each plant shall be provided with a Safety Parameter Display System that is located convenient to the control room operators. This system will continuously display information from which the plant safety status can be readily and reliably assessed by control room personnel who are responsible for the avoidance of degraded and damaged core events.

The SPDS should provide a concise display of critical plant variables to the control room operators to aid them in rapidly and reliably determining the safety status of the plant. Although the SPDS will be operated during normal operations as well as during abnormal conditions, the principal purpose and function of the SPDS is to aid the control room personnel during abnormal emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core. This can be particularly important during anticipated transients and the initial phase of an accident.

The minimum information to be provided shall be sufficient to provide information to plant operators about the following functional areas:

(1) Reactivity control

(2) Reactor core cooling and heat removal from the primary system

(3) Reactor coolant system integrity

(4) Radioactivity control

(5) Containment conditions.

The specific paramenters to be displayed shall be determined by the licensee. Implementation of the SPDS may be affected by the implementation of Regulatory Guide 1.97 because this regulatory guide has a major effect on instruments necessary during accident conditions. However, licensees have been instructed to implement an SPDS with the best available parameters to support the five primary safety functions.

The following describes how each licensee is meeting these guidelines:

ANO-1

Description:

The SPDS computer system at ANO-1 is an integrated network designed to perform functions required for ANO-1 and ANO-2, the Technical Support Center data display system, and the Emergency Operations Facility data display system. The computer accesses necessary input parameters from sensors in ANO-1 and ANO-2, processes the signals and provides displays to each control room as well as to the TSC and the secondary TSC portion of the EOF. "Touch Screen" controls are used on color graphic CRTs for rapid information access. The SPDS displays have been integrated with the development of the Emergency Operating Procedures. The required reliability is obtained using redundant CPUs, redundant data acquisition hardware, redundant display generators and redundant CRTs in each control room. The design of the SPDS incorporates human factors principles; however, an evaluation of the design will be conducted as part of the control room design review.

The SPDS is installed and operating in the control rooms so that the operators may become familiar with the system. The system currently includes parameters for all the five functions required. After completion of the validation and training program, the system was declared to be operational with existing parameters on January 9, 1984. The completed SPDS would also include incorporation of parameters identified in the Regulatory Guide 1.97 evaluation. Completion is expected by June 1986.

The Safety Analysis for the SPDS will identify the critical safety functions, the parameters required to monitor these functions, and a basis on which the selected parameters are sufficient to assess the safety status of each identified function over a wide range of events. This Safety Analysis is expected to be complete by June 29, 1984.

Design/Installation:

The design/installation is complete except for the items which would be covered by the CRDR and R.G. 1.97 evaluation. Completion is expected by June 1986.

Operability Tests:

The operability tests are complete except for the items covered by the CRDR and R.G. 1.97 evaluation. Completion is expected by June 1986.

Procedures:

The procedures are complete with existing parameters.

Training:

The training is complete with existing parameters.

Technical Specifications:

The need for technical specifications on this system has not yet been determined.

NRC Review Status:

NRC staff will review the system after receipt of the Safety Analysis in June 1984.

Crystal River Unit 3

Description

The licensee has proposed a plant Safety Parameter Display System (SPDS) utilizing a dedicated, dual-channel 8 bit microprocessor system which will be independent of the present plant computer. The system will display selected parameters both in the main control room and the Technical Support Center (TSC). During the last refueling outage, a single-channel data acquisition system, designated as RECALL, was installed with displays in the main control room, TSC and Emergency Operations Facility (EOF). The SPDS will be an extension of the RECALL system with appropriate qualifications and isolation devices. The licensee is scheduled to submit a safety analysis of the proposed SPDS in August 1984 for NRC review as discussed in Generic Letter No. 82-33 for all five (5) functional categories described in the Generic Letter No. 82-33. Installation of the SPDS is presently planned during the next scheduled refueling outage.

Design/Installation:

Design engineering to be completed in September 1984. Procurement is in progress. Installation scheduled for next refueling outage which begins in March 1985 and is scheduled to be complete by August 1985. Preliminary installation will begin in September 1984.

Operability Tests:

August 1985.

Procedures:

July 1985.

Training:

August 1985.

Technical Specifications:

The need for technical specifications will be determined by March 1985 and, if required, will be submitted at that time.

NRC Review Status:

NRC active review of the proposed system will begin following receipt of the licensee's safety analysis in August 1984.

Davis Besse, Unit No. 1

Description:

In 1981 the licensee implemented a Data Acquisition and Display System (DADS) capable of providing simple graphical displays of plant information useful to the operators. This provided the foundation of the licensee's SPDS. The licensee is adding hardware which will speed up the acquisition of the data. The SPDS has the capability of displaying and providing real time copies of displayed information of a large amount of data. Typical parameters avilable in color graphic display form include temperature and pressure of hot and cold legs, pressurizer level, steam generator levels, feedwater flow, auxiliary feedwater flow, hydrogen concentration, and nuclear parameters. There are presently approximately 30 parameters capable of being displayed on CRTs in the control room and the technical support center. The system is fully operational and may be upgraded as other parameters are added as a result of the Regulatory Guide 1.97 review. The licensee will modify its system to increase the speed of the data processing capability. The licensee considers this item complete to satisfy the NUREG-0737 requirements. The licensee's submittal concerning this item was submitted in November 1983.

Design/Installation:

SPDS complete and functional with licensee's DADS capability - February, 1984. SPDS complete with additional hardware to increase the speed of data processing - November 1984.

Operability Testing:

Complete by February 1984.

Procedures:

No procedures required. The SPDS provides an aid to ATOG procedures.

Training:

The operators are trained on how to operate the system.

Technical Specifications:

No proposed technical specifications have been identified yet.

NRC Review Status:

NRC staff review in progress.

Oconee Units 1, 2 and 3

Description:

The licensees current computer system does not support the needs of an operational SPDS. Plans are underway to upgrade the computer. The licensee has stated that the SPDS design will meet the "intent" of Supplement 1, NUREG-0737. The display will be available in the main control rooms and the Technical Support Center. The five required functions will be displayed by the system. The system is fed by a dedicated static inverter normally receiving power from DC batteries. Inverter, battery, or charger failure transfers the supply to either of two independent regulated AC power sources. The verification and validation program will be developed using INPO guidance. The SPDS project is under the direction of the Control Room Review Steering Committee and will include the human factors review, ATOG procedures and R.G. 1.97 evaluation. The licensee has provided the following schedules for operational SPDS systems:

Unit 1: January 1985

Unit 2: July 1985

Unit 3: November 1984

Design/Installation:

The SPDS design with verification and validation is expected to be completed by April 1984. Installation would follow delivery of the new computer system, scheduled for June 1984.

Operability Tests:

The tests will be completed prior to declaring the system to be operational.

Procedures:

Expected to be consistent with training schedule below.

Training:

Training is scheduled to be completed in October 1984.

Technical Specifications:

No technical specifications have been identified yet on this item.

NRC Review Status:

The safety analysis required under Supplement 1 NUREG-0737 was submitted Feburary 16, 1983 and is under review.

Rancho Seco

Description

The Rancho Seco SPDS was designed as a diagnostic tool compatible with Abnormal Transient Operating Guidelines and Emergency Operating Procedures. The Rancho SPDS is designed to use a minimum of displays and parameters to concisely present information concerning the safety status of (1) Reactively Control, (2) Reactor Core Cooling and heat removal from the primary system, (3) Reactor Coolant System integrity, and (4) containment conditions. The radioactivity control function, the fifth item required by NUREG-0737 Supplement No. 1, is primarily performed by a separate installed system which consists of a dedicated computer based system utilizing seven remote sensors and a display console in the control room. The Rancho Seco SPDS System consists of two color video monitors with their associated controls installed in the main control room to provide displays of various transient and accident parameters or derived variables in the control room. The data is provided to the displays through a remote multiplexing system that takes information from various locations in the plant and processes this data through the SPDS process computer located in the Computer Room. On July 28, 1983 the licensee provided the NRC with its SPDS Safety Analysis and Implementation Plan.

Design/Installation

The SPDS has been designed constructed, vendor tested and installed in the Rancho Seco Control Room.

Operability Tests

The SPDS has been vendor tested and is currently being used for operator familiarization and in place testing.

Procedures

The procedures associated with SPDS will be incorporated into the upgraded Emergency Operating Procedures. This will be completed prior to start up from the next refueling outage (November 1984).

Training

Operator training will be completed prior to start up from the next refueling outage.

NRC Review

NRC review of the licensees reports submitted on July 28, 1983 is in progress.

Three Mile Island Unit 1

Description:

The TMI-1 safety parameter display system (SPDS) will be implemented in three steps. Step one, the post-trip pressure-temperature plot has already been implemented. This plot provides reactor coolant system and steam generator pressure-temperature information to support the symptom-based ATOG emergency procedures. Operators and shift technical advisors are given a computer-aided instruction course to learn how to interpret the pressure-temperature plot. Step two, the 'basic' SPDS will envelop those critical plant parameters currently available in the plant process computer base. The 'basic' SPDS will provide information on reactivity control, reactor coolant system inventory control, primary side heat removal and reactor building environment. It will not provide information on radiation control. Step three, the 'final' SPDS will include the 'basic' SPDS plus additional parameters available in the control room, but not currently on the plant process computer. The 'final' SPDS will add radiation control parameters and also include reactor vessel and hot leg water level information that will result from the Item II.F.2 modifications.

Beyond the requirements of NUREG-0737, GPU Nuclear has long range plans to replace the process computer with a new computer system providing two fully compatible and redundant computer systems which will be powered from an uninterruptible power supply. These improvements could enhance SPDS redundancy and reliability.

Design/Installation:

Pressure Temperature Plot: Complete.

'Basic' SPDS: January 1985 (subject to plant availability)

'Final' SPDS: First outage of sufficient duration following June 1985.

Operability Tests:

Pressure Temperature Plot: Complete.

'Basic' SPDS: January 1985 (Same as installation).

'Final' SPDS: First outage of sufficient duration following June 1985.

Procedures:

Pressure Temperature Plot: Complete.

'Basic' SPDS: Upon installation.

'Final' SPDS: Upon installation.

Training:

Pressure Temperature Plot: Complete.

'Basic' SPDS: Upon installation.

'Final' SPDS: Upon installation.

Technical Specifications:

Not applicable.

NRC Review Status:

Review pending receipt of SPDS safety analysis (April 1984).

II.B.1 REACTOR COOLANT SYSTEM VENTS

NUREG-0737 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 non-condensible 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. Subsequently, NUREG-0737 was superceeded by rulemaking which modified 50.44(c)(3)(iii) "Standards for combustible gas control systems in light-water-cooled power reactors" to require installation and operability of the vents by the end of the first outage of sufficient duration after July 1, 1982.

The following describes how each B&W licensee met these requirements:

ANO-1

Description:

The licensee has installed vents at the high points of the hot legs, the pressurizer and the reactor vessel head. Each RCS hot leg high point vent has two parallel sets of two solenoid-operated valves in series with a piping cross connect between the valves. The non-condensible gases, steam and/or liquids vented from the RCS hot leg high points are discharged directly to the containment atmosphere. The vent path from the pressurizer discharges to the quench tank and has two solenoid-operated valves in series. The reactor vessel head vent has two sets of two solenoid operated valves, discharging into the containment atmosphere. Each of the solenoid-operated valves is remotely controlled from the main control room and also provides indication of valve position. Parallel sets of valves in each hot leg vent is powered from different emergency buses.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Licensee submittal is scheduled for March 19, 1984.

NRC Review Status:

Complete.

Crystal River Unit 3

Description:

High point reactor coolant system (RCS) vents were installed on the pressurizer and both loop hot legs (candy canes) in July 1983. The licensee, in October 1982, requested an exemption to the requirement to install a reactor vessel head vent on the basis that other methods exist for venting the head area. On July 21, 1983, the staff issued an order granting the licensee a schedular exemption which changed the required date for installation of a head vent to the first scheduled outage of sufficient duration after December 31, 1985. The pressurizer and hot leg vent paths each consist of a short piping run which exhausts to the reactor building atmosphere and each path contains two solenoid operated isolation valves in series which are remotely operated from the main control room. Each valve is remotely operated from and has positive open/closed status lights in the main control room. The valves are powered from separate class 1E emergency power supplies and fail in the closed position in the event of loss of power. An alarm sounds in the control room if both valves in a vent path are open. The system meets the criteria of NUREG-0737, Item II.B.1 and the requirements of 10 CFR 50.44 paragraph (c)(3)(iii).

Design/Installation:

Pressurizer and loop vents: Complete

Reactor Vessel head vents: Schedule exemption issued

Operability Tests:

Pressurizer and loop vents were operationally tested in July 1983 and are considered operable.

Procedures:

Complete and implemented.

Training:

Complete

Technical Specifications:

Licensee submitted proposed technical specifications in June 1983. They are presently being evaluated by the staff using the guidance of Generic Letter 83-37. The technical specifications are scheduled for issuance in April 1984.

NRC Review Status:

The Safety Evaluation for this item was issued to the licensee on September 8, 1983 by NRR. Region II is conducting a post-implementation review, which will include confirmation of various items as discussed in the Safety Evaluation.

Davis Besse, Unit No. 1

Description:

Reactor coolant system vents have been installed and are operational on the pressurizer and the high points of the reactor coolant hot leg loops. The vent paths from reactor coolant hot legs to the containment atmosphere each contain solenoid operated valves in series. The vent path from the pressurizer to the pressurizer quench tank contains two motor operated valves in series. The valves of all three vent paths are remotely controlled from the control room and have position indication in the control room. Each vent path contains flow restrictors to limit the flow from a pipe rupture or from inadvertant actuation of the high point vent system to less than the capability of the reactor coolant make up system. The licensee was granted an interim exemption from the requirements for a reactor vessel head vent. Except for the interim exemption of the head vent requirement, the system design meets the criteria of NUREG-0737 Item II.B.1 and the requirements of 10 CFR 50.44 par (c)(3)(iii).

Design/Installation:

Pressurizer and loop vents are complete and operational. The licensee was granted an interim exemption from the head vent requirement.

Operability Tests:

Pressurizer and loop vents, testing complete.

Procedures:

Complete for pressurizer and loop vents.

Training:

Complete for pressurizer and loop vents.

Technical Specifications:

Proposed technical specifications for the pressurizer and loop vents were submitted by letter dated May 5, 1982.

NRC Review Status:

An NRC Safety Evaluation Report was issued by letter dated October 5, 1983. An interim exemption from the head vent requirement was granted. Resolution of the head vent issue is dependent upon the demonstration of the ability of B&W plants to manage the venting problem in the B&W Integrated Test Program. Toledo Edison Company has proposed to install a vessel head to hot leg high point connection which would function as a head vent if such a line would eliminate the need for a reactor vessel vent and level measurement system. This proposal is under staff review.

Oconee Units 1, 2 and 3

Description:

Oconee 1, 2 and 3 provide venting capability from the high points of the reactor vessel head and both RCS hot legs. The existing power operated relief valve is used to vent the pressurizer. The high point vent paths from the reactor vessel head and the RCS hot leg high points each contain solenoid-operated valves in series which are remotely controlled from the main control room, where positive indication of valve position is also provided. Each vent path is powered from a different emergency power train. The noncondensible gases, steam and/or liquids vented from the hot legs are separately piped to the basement of the reactor building and released into the discharge air stream of the reactor building coolers. The reactor vessel head vent ties into one of the hot leg vents downstream of the hot leg vent valves. Each high point vent path has a capacity of one half the RCS volume per hour.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Will be written as part of the ATOG (I.C.1) procedures which are scheduled for completion in October 1985.

Training:

Will be done as part of ATOG (I.C.1) which is scheduled for completion in October 1985.

Technical Specifications:

Complete.

NRC Review Status:

Complete.

Rancho Seco

Description:

Reactor Coolant System (RCS) high point vents have been installed on the pressurizer and the high points in both RCS hot legs. The non condensable gases, steam, and/or liquids can thus be vented from either the pressurizer or the RCS hot legs. The piping for each vent discharges to the containment atmosphere through spargers. The vent path from the pressurizer and RCS hot legs each contain two solenoid-operated valves in series which are remotely controlled from the control room. Positive indication of valve position is also provided in the control room. The pipe size has been selected such that it limits the flow from a pipe rupture or from inadverting operation of the vent valves to less than the capability of the reactor coolant make up system. The licensee was granted an interim exemption from the requirements of reactor system head vent requirement. Except for the interim exemption of the head vent requirement, the system design meets the criteria of NUREG 0737 Item II.B.1 and the requirements of 10 CFR 50.44 par (e)(3)(iii).

Design/Installation

Pressurizer and loop vents are complete and operational. The licensee was granted an interim exemption from the head vent requirement.

Operability Tests

Pressurizer and loop vents, testing complete.

Procedures

Complete for pressurizer and loop vents.

Training

Complete for pressurizer and loop vents.

Technical Specifications

Proposed technical specifications for the pressurizer and loop vents were submitten by letter dated February 17, 1983.

NRC Review Status

An NRC Safety Evaluation Report was issued by letter dated September 27, 1983. An interim schedular exemption from the head vent requirement was granted. Resolution of the head vent issue is dependent upon the depressurization results from the B&W Owners Group Integral Systems Test Program.

Three Mile Island Unit No. 1

Description:

Reactor coolant system vents have been installed on the pressurizer, loop "A" and "B" hot legs, and the reactor vessel head. The reactor vessel head and loop vent paths each have two solenoid-operated valves in series. The pressurizer high point vent path has one solenoid - and one motor-operated valve in series. Each valve has positive position indication and can be remotely operated from the control room. An alarm sounds in the control room if both valves in a vent path are open. Each vent path has a flow/no-flow indicator. The system design meets the criteria of NUREG-0737, Item II.B.1 and the requirements of 10 CFR 50.44 para(c)(3)(iii).

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Submitted.

NRC Review Status:

NUREG-0737 criteria subsumed by 10 CFR 50.44(c)(3)(iii). NRC safety evaluation provided for information only by letter dated October 17, 1983.

II.B.2 PLANT SHIELDING

NUREG-0737 Position

With the assumption of a post-accident release equivalent to that described in Regulatory Guides 1.3 and 1.4 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. Under these conditions, releases were assumed equivalent to 50% of the core radioiodine and 10% of the core noble gases. These conditions also assumed that 1% of the core solids are contained in the primary coolant. 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 post-accident 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 post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility. The goal under post-accident conditions is to limit personnel exposure to a maxinum of 5 REM whole body, or its equivalent to any part of the body, for the duration of the accident (i.e. GDC 19 limits)

The following describes how each B&W licensee met these guidelines:

ANO-1

Description:

The licensee reviewed the plant shielding and environmental qualification of equipment for spaces/systems which may be used in post-accident operation and found it necessary to make one modification. The return flow from the reactor coolant pump seals was changed from going into the let-down tank to the quench tank. The licensees review of the large cold leg loss-of-coolant accident indicated that no additional shielding was necessary from consideration of boron precipitation during operation of the decay heat removal system. The justification provided was that core circulation can be assured by operating valves located in habitable areas to run a redundant decay heat removal train in the event of an active failure in the normal train.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

No changes required.

NRC Review Status:

Complete.

Crystal River Unit 3

Description:

This item required the licensee to perform a shielding design study to ascertain the adequacy of shielding for vital areas requiring access during post-accident conditions. The licensee included in this study the main control room, diesel generator room, nuclear sample room, hydrogen purge equipment, containment air monitor instrumentation, radioactive waste control panel, radiochemistry laboratory and count room and the Technical Support Center. As a result of the study, remotely controlled motor operators were installed on four valves in the decay heat removal system, three motor-operated bypass valves were added to the makeup and purification system filter, provisions for remote venting of the waste gas holdup tank was installed and extensions were installed on the handwheels of a number of valves. Several emergency procedures were reviewed to determine adequacy of worker protection following an accident.

NUREG-0737, Item II.B.2 also included requirements for environmental equipment qualification as it relates to radiation exposures. This aspect of Item II.B.2 has been enveloped by efforts in progress to meet the requirements of the equipment qualification rule (10 CFR 50.49). The licensee has submitted all required information to demonstrate compliance with this rule.

The staff noted during the review of this item that shielding or other hardware modifications that may be needed to reduce exposures in the recirculation decay heat removal mode following a loss of coolant accident had apparently not been considered by the licensee from the standpoint of boron precipitation prevention. The licensee has reviewed this scenario and determined that the capability to mitigate the effects of this scenario without over exposing personnel exists at Crystal River but some operating procedures must be changed. Florida Power has committed to upgrade their procedure by July 1984.

Design-Installation:

All shielding design and installation is complete.

Operability Tests:

Complete.

Procedures:

Complete

Training:

Complete

Technical Specifications:

None required

NRC Review Status:

The NRC issued a Safety Evaluation on November 8, 1983 which concluded that the requirements of NUREG-0737, Item II.B.2.2 have been met. The staff is scheduled to issue a Safety Evaluation regarding environmental equipment qualification for Crystal River, Unit 3 in April 1984.

Davis Besse, Unit No. 1

Description

The licensee performed a Plant Shielding Review as required by NUREG-0737 Item II.B.2. The shielding review concuded that motor operators for valves DH63 and DH64, with control and position indication in the control room, were needed in order to remotely align the decay heat and high pressure injection pumps if required to provide adequate flow to meet varying reactor coolant system leak rates. The motor operators for the subject valves with position indication in the control room were installed and are operable.

The review also concluded that a remote reactor coolant sampling system was needed. The licensee has implemented a post accident sampling system which provides the capability of obtaining grab samples of the reactor coolant system (see Item II.B.3).

At the meeting with the licensee of February 3, 1984, the staff discussed the concern regarding the capability of providing redundant long-term flow path through the reactor to avoid build up of boron concentration that could lead to precipitation, thus adversely affecting long-term cooling. The licensee determined that the valves which would be operated to prevent boron precipitation would be located in a radioactive environment but are capable of remote control from the control room. Therefore, it would not be necessary for an individual to operate the valves at the valve location.

Design/Installation

The licensee has completed the modifications relating to the two valves for remotely aligning the decay heat and high pressure injection pumps. The original plant design provided for valve manipulation from the control room to prevent possible boron precipitation following an accident.

Operability Tests

Operability tests have been complete. The licensee considers this area is complete.

Procedures

As a result of the review, certain procedures were modified. The licensee considers this area complete.

Training

Training is complete.

NRC Review Status

A Safety Evaluation Report was provided the licensee by letter dated August 8, 1983. Additional questions concerning operation of valves on the boron precipitation were identified. At the meeting with the licensee of February 3, 1984, it was determined that these valves were remotely operated from the control room. The staff considers this items is closed.

Oconee Units 1, 2 and 3

Description:

The licensee has stated that all modifications required by this item have been completed. Some additional shielding was required to be placed around the control room and some reach rods needed to be installed. No major equipment changes were identified. The licensees review of the large cold leg loss-of-coolant accident and maintenance of long term core flow to preclude boron precipitation indicated that no modifications were required because all valve operations can be handled from the control room.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete. A few changes may be required subsequent to finalized PASS procedures.

Training:

Complete.

Technical Specifications:

No changes identified.

NRC Review Status:

Complete.

Rancho Seco

Description

The licensee completed the radiation and shielding design review required by NUREG-0737. Their review indicated that the following changes were necessary to meet the guidelines of this section of NUREG-0737.

(a) The addition of motor operators on two valves in the decay heat train return line from the hot leg. This was provided to allow the valves to be remotely operated from the control room to maintain long term core flow and prevent Boron precipitation in case of a single active failure in the Decay Heat Removal System. No other shielding or equipment modifications were required.

(b) Modifying some procedures to conform with the basis for its shielding calculations.

Remote operators were installed on the two valves in the decay heat train return line and the appropriate procedure modifications were made. The licensee considers this item completed

Design/Installation

Two remote valve operators were installed.

Procedures/Training

The appropriate procedures were modified and the operator training was completed.

Technical Specifications:

No changes were necessary as a result of this effort.

NRC Review

The NRC staff completed its post implementation review of this item and concluded that the licensee has met the NUREGO 0737 guidelines for this item. The NRC staff did not include in its review the Radiation Qualification of Safety Related Equipment. This topic is being sepately addressed under NRC Multi-Plant Action Item B-60 "Environmental Qualification of Electric Equipment for Nuclear Power Plants."

Three Mile Island Unit 1

Description

As a result of the radiation and scheduling review of Item II.B.2, modifications were determined to be necessary for adequate access to some motor control centers and for remote operation of six decay heat removal system valves. Operation of the six valves is necessary following a large cold leg loss of coolant accident (LOCA) to maintain redundant long-term flow path through the reactor core to avoid a buildup in boron concentration that could lead to precipitation, thus adversely affecting long-term cooling. The valves are located in an area of potential high radiation following an accident. Therefore, remote operation capability is considered necessary.

The modifications to permit access to the motor control centers are complete. However, based upon a good faith effort by GPU Nuclear and upon an acceptable justification for interim operation, the Commission found acceptable the GPU Nuclear proposal to defer the valve modifications until the first refueling. The good faith effort determination was made late 1982 on the basis of GPU Nuclear procurement activity then believed to be leading to a mid-1983 delivery of long-lead equipment and a March 1984 installation date. (See SECY-82-384 dated September 16, 1982 and SECY-82-384A dated December 6, 1982). By subsequent correspondence dated October 21, 1983 GPU Nuclear proposed an alternate for preventing excessive boron concentration following a large cold leg LOCA. The alternate solution involves installation of an additional piping run with one additional valve, and would be implemented in lieu of providing remote operators for the six decay heat removal system valves. GPU Nuclear believes the alternate solution will be less expensive and will reduce the complexity of post-accident operation. Installation would also be at the first refueling after restart. The staff has reviewed and approved the alternate solution to this problem

Design/Installation:

Motor Control Center: Complete.

Boron Precipitation Control (Alternate Solution): Engineering scheduled for April 1984 completion. Installation scheduled for fourth quarter 1984 completion (providing the plant is shutdown for a sufficient duration). If the plant is allowed to start-up in July 1984, a steam generator inspection outage would occur in December 1984.

Operability Tests (Alternate Solution):

Next outage or fourth quarter 1984.

Procedures (Alternate Solution):

Fourth quarter 1984, pending completion of installation.

Technical Specifications:

No changes are necessary at present.

NRC Review:

Complete.

II.B.3 POST-ACCIDENT SAMPLING CAPABILITY

NUREG-0737 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. The goal was to limit personnel exposure under post-accident conditions to 5 rem whole-body or its equivalent to any part of the body (i.e., GDC-19 limits).

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 chorlide sample analysis within a shift).

The following describes how each B&W licensee met these guidelines:

ANO-1

Description:

The PASS at the Arkansas Nuclear site uses identical sampling systems for each of the two units with a shared analytical capability. The system is housed in a separate shielded building. The PASS provides the ability to obtain samples from each reactor coolant hot leg, the pressurizer water and gas, the containment sump, and the containment atmosphere. The PASS valves not accessible after an accident are environmentally qualified for the conditions in which they need to operate. The lines are heat traced and pressurized. There is a capability to perform on-line gamma spectrometry for the full range of isotopes. In addition, there exists the capability of obtaining grab samples which could be analyzed on or off site. When the automatic system is fully operational, it is expected to be used for routine on-line analysis in addition to post-accident analysis. At present, the installed equipment for on-line boron, chloride and pH analysis is not providing accurate data. Repairs and testing are in progress.

Design/Installation:

Complete.

Operability Tests:

Conditionally operable while repairs and testing are in progress.

Procedures:

The PASS was used in the last emergency drill. Revisions to procedures may be required before complete operability is finalized.

Training:

Complete except for potential revisions.

Technical Specifications:

Technical specification revisions will be submitted by March 19, 1984.

NRC Review Status:

NRC staff has issued an SER with three open items. The licensee has indicated that the items will be closed out by Janury 31, 1985.

Crystal River Unit 3

Description:

Crystal River Unit 3 has a newly-installed Post Accident Sampling System (PASS) which goes far beyond the minimum requirements of NUREG-0737, Item II.B.3. The system is designed to facilitiate sample collection and analysis following an accident with emphasis on ALARA considerations but will also be used by the licensee to collect and analyze routine samples during routine plant operations. This has the advantage of maintaining a high degree of operator familiarity with and confidence in the system. The PASS at Crystal River Unit 3 features a remote control panel with a mimic board, computer-controlled on-line analyses for repeatability, a special design "grab sample" collection flask with "quick-disconnect" connections, an automated isotopic measurement system, an automated titration instrument for measurement of reactor coolant boron concentration and an ion chromatograph for measurement of reactor coolant chloride concentration. This system satisfies the requirements of NUREG-0737, Item II.B.3.

Design/Installation:

Complete, except for the on-line reactor coolant dissolved hydrogen analyzer, which is scheduled for delivery in April 1984. Installation of the analyzer is scheduled for May 1984.

Operability Tests:

Complete, except for dissolved hydrogen analyzer (see above).

Procedures:

Complete.

Training:

Complete, except for dissolved hydrogen analyzer.

Technical Specifications:

Proposed technical specifications for the new containment isolation valves required for the PASS were submitted by the licensee in June 1983 and are being evaluated by the NRC. No technical specifications for operation and surveillance of the PASS, per se, are required; however, the licensee has been requested to submit a brief discussion of the post-accident sampling capability for inclusion in the administrative section of the Crystal River Technical Specifications in accordance with the staff's Generic Letter No. 83-37.

NRC Review Status:

A final design analysis is to be submitted by the licensee in April 1984 on the PASS. The NRC will perform a post-implementation review based, in part, on this submittal and based, in part, upon field inspection.

Davis Besse, Unit No. 1

Description:

The post accident sampling system (PASS) consists of a control panel and a sample skid located on the bottom floor of the Auxiliary Building. The PASS is capable of obtaining highly radioactive reactor coolant grab samples following an accident. The grab samples can be obtained from (1) pressurizer liquid and vapor spaces, (2) reactor coolant cold leg, (3) containment emergency sump, and (4) the letdown system. This part of the PASS is not an in line system. The samples can be diluted or undiluted. Another part of the PASS, the Post Accident Containment Atmosphere Sample System has the capability of obtaining containment atmosphere grab samples after an accident. The sample can be obtained from the installed post accident containment radiation monitoring system or a location in the spent fuel area. Samples taken from either location will include particulate, noble gas and iodine. The licensee has a fully functional chemical and health physics laboratory equipped and staffed to perform the analyses required by NUREG-0737. Chlorine analyses are conducted off site. The licensee considers that this item meets the requirements of II.B.3 and is complete.

Design/Installation:

The PASS was declared operational December 31, 1983. The licensee is assessing potential improvements to the installed system.

Operability Tests:

The PASS was tested prior to declaring it operational.

Procedures:

All procedures to support operation of the PASS are in place.

Training:

Training is complete.

Technical Specifications:

The licensee is currently preparing proposed Technical Specifications.

NRC Review Status:

In July 1982, the staff requested information on the Davis Besse PASS. The licensee responded by letter dated April 25, 1983. The staff review is in progress. Estimated schedule for completion is June 1984.

Oconee Units 1, 2 and 3

Description:

Separate and identical sampling capability is provided for each of the three units. The PASS provides the capability to collect diluted or undiluted liquid and gaseous grab samples that can be transported to the radiochemical laboratory for hydrogen, pH, conductivity, boron, chloride and radionuclide analyses. It also provides in-line monitoring of hydrogen in the containment atmosphere. Pressurized reactor coolant samples are cooled, degassed, and diluted at the PASS station. The dissolved hydrogen and oxygen are measured by gas chromatography. Reactor collant samples are diluted to minimize personnel exposure. The containment atmosphere sampling line is heat traced. The gaseous portion of the PASS is complete in all three units. The installation of the liquid portion of the PASS is complete in Units 1 and 2 and is scheduled for completion in Unit 3 during the March 1984 refueling outage.

Design/Installation:

Complete for Units 1 and 2; March 1984 for Unit 3.

Operability Tests:

Four valves on Units 1 and 2 which had formerly been considered to be environmentally qualified were found to be unqualified for submerged operation. Procurement of qualified valves is in process. The licensee expects to install and test the qualified valves at the next refueling outage for each Unit 1 and 2 (approximately January 1985 for Unit 1 and January 1986 for Unit 2).

When the Unit 3 system is completed, it will have qualified valves installed and tested during the next refueling outage (approximate startup date is November 1984).

Procedures:

Scheduled for completion by April 1984.

Training:

Scheduled for completion by April 1984.

Technical Specifications:

A package is scheduled to be submitted to NRC by April 1984.

NRC Review Status:

An SER has been issued with an open item regarding the final plant specific core damage estimation procedure. The resolution of this item is expected to be addressed along with completion of all applicable procedures.

Rancho Seco

Description

The Rancho Seco Post Accident Sampling System (PASS) consists of an in line sampling and analysis system capable of sampling and analyzing reactor coolant system, the reactor building (containment) atmosphere, and the reactor building emergency sump within 3 hours or less from the time a decision is made to take a sample. The system also has the capability of providing a diluted or undiluted grab sample when required. The PASS system is capable of analyzing by remote controlled in line monitors noble gases, iodines, cesiums, non volatile isotopes, boron, chloride, pH and dissolved gases in the primary coolant. Hydrogen and gamma spectrum in the containment atmosphere sample can also be analyzed by in-line monitors. The licensee considers the system operational.

Design/Installation

The installation was completed during the last refueling outage.

Operability Tests

The system was tested and licensee considered the system operational in August 1983. Though the system appears to be capable of obtaining and analyzing samples if required, there are a number of operational problems with the system. For example the ion chromatograph microprocessor is temperature sensitive and is lcoated in an area subject to high ambient temperatures at times. The licensee is working on a modification that will provide cooling.

Procedures

All procedures except for the core damage procedure have been completed and are in place. A core damage procedure was prepared by a vendor for the licensee. The licensee is currently reviewing the document and experts to have the final procedure completed and in place in May, 1984.

Training

Training is underway with training completed for some of the licensee's staff. Training for the remainder of the licensee's staff is expected to be completed by the end of April 1984.

Technical Specifications

The licensee provided proposed Technical Specifications for the Post Accident Sampling System by letter dated Jan 23, 1984.

NRC Review Status

The staff review of the Post Accident Sampling System design criteria was completed and our SER's were issued by letters dated February 15, 1983 and September 2, 1983.

Region V has identified a number of hardware and operational problems associated with the system (see Region V Report No. 83-24 and 83-26) including those noted above.

Three Mile Island Unit 1

Description

The TMI-1 post-accident sampling system permits a reactor coolant grab sample for pH, dissolved oxygen and hydrogen, chloride, boron and radionuclide analysis in the chemistry lab. Containment atmosphere grab-samples provide measurements of containment atmosphere hydrogen and radionuclides. In line redundant monitoring of containment hydrogen is also provided. The reactor coolant sampling system was developed by modifying the existing system, principally by rerouting the sample line to the floor of the nuclear sampling room, thus permitting the line to be surrounded by lead shielding, and by adding certain sampling equipment, such as, long handled tools and a shielded sample transport cart. The GPU Nuclear system does not provide for drawing samples from a depressurized reactor coolant system.

GPU Nuclear has performed person-motion studies and concluded that post-accident samples can be obtained and analyzed without exceeding the applicable personnel radiation exposure limits. The sampling procedure uses a team of technicians operating in series in order to limit the exposure to any one individual to meet GDC-19 requirements.

Design/Installation:

Complete.

Operability Tests:

Complete, except for a demonstration of the chemical analysis procedures using a standard test matrix solution scheduled for August 1984.

Procedures:

Complete except for the final core damage procedure which is scheduled for August 1984 submittal to the staff.

Training:

Complete.

Technical Specifications:

Will be submitted in the near future (next two months).

NRC Review:

NRC staff has issued an SER with one open item regarding the final plant specific core damage estimation procedure which will be completed prior to August 1, 1984. NRC Region I has recently questioned whether the person-motion studies consider all potential contributers of radiation exposure such as airborne radioactivity originating from the sink, residual contamination during subsequent samples, etc. (See inspection report 50-289/84-03 dated February 9, 1984.) Region 1 met with the licensee to discuss these issues and their resolution on February 23, 1984 (see meeting report 50-289/84-78).

II.D.1 Relief and safety Valve Test Requirements

NUREG-0737 Position

Licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under anticipated operating conditions for design-basis transients and accidents. This requirement includes the necessary testing and qualification of associated piping. A further part of this program included qualification of block valves.

The following describes how the B&W licensees of operating plants met these guidelines:

All of the B&W licensees joined the EPRI generic test program to generically test each valve type. Following the EPRI test program, each licensee conducted a plant specific review and submitted the results to NRR for both relief valve and block valve qualification. For the most part, the only physical modifications resulting from this test program involved setting some ring adjustments on the safety valves and possibly adding some piping supports to the piping where appropriate. However, both TMI-1 and Davis-Besse had a loop seal on their safety valves which was removed as a direct result of this testing. Also, Davis-Besse had a design where two safety valves were some distance away from the pressurizer (about 35 feet) in a valve room. As a result of this testing, these two safety valves were moved closer to the pressurizer and the discharge piping runs were removed. The PORV was not moved at Davis Besse.

NRR Review:

The NRR review is in progress but has not been completed on any B&W plant. The NRR review has raised questions but none have involved additional testing to date.

II.E.1.1 AUXILIARY FEEDWATER SYSTEM EVALUATION

NUREG-0737 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 vunerabilities, 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.

The end result of these reviews was a series of short term and long term corrective actions. For the most part the short term corrective actions are done and only the long term items remain.

The above evaluations have been completed by all B&W licensees. The following discusses the physical modifications which have resulted from these reviews for each plant.

ANO-1

Description:

The ANO-1 AFW system incorporates a steam driven pump with a rating of 705 gpm at 1175 psi and a motor driven pump with a rating of 672 gpm at 1175 psi. Each pump feeds both steam generators giving parallel flow paths, each of which contains redundant safety grade modulating DC solenoid flow control valves. Automatic initiation of AFW flow occurs on (1) low steam generator level or pressure; (2) loss of reactor coolant pumps; (3) loss of main feedwater flow; and (4) ECCS actuation. The present system employs non-safety-grade instrumentation, specifically the startup range level signal from the steam generator, for automatic initiation. The Integrated Control System does not control AFW initiation or flow control. With the completion of installation of the Emergency Feedwater Initiation and Control (EFIC) system, the automatic initiation will be safety grade and comply with the single failure criterion of IEEE Standard 279-1971. The AFW flow indication is safety grade and is obtained from an orifice in each flow path, with display provided in the control room. The AFW system is seismically qualified; environmental qualification will be completed with the upgrading scheduled for the end of 1984. A major part of the upgrading will be installation of the EFIC system designed to:

- o automatically initiate AFW, as noted above for the present system,
- o controls AFW flow to maintain OTSG levels within appropriate setpoints,
- o provide level rate control,
- o isolate the main steam and main feed lines of a depressurized OTSG, and
- o control the atmospheric dump valves.

The long term improvement includes installation of a second condensate storage tank which is safety grade. Installation of the second condensate storage tank would not occur until the refueling outage after the next refueling outage scheduled in late 1984.

Design/Installation:

The design of all upgrades is almost complete with detailed review in progress. Installation of the safety grade EFIC will be completed by the end of 1984.

Operability Tests:

The licensee has indicated that the startup tests for EFIC will be a major effort.

Procedures:

The procedures for the EFIC system will be completed and implemented by the end of 1984.

Training:

Training will be completed on the EFIC system by the end of 1984.

Technical Specifications:

Technical specification changes will be submitted in June 1984.

NRC Review Status:

NRC pre-implementation review for the control grade modification is complete. An SER has been issued on the safety grade modifications to EFIC. Staff review of the new condensate storage tank is continuing.

Crystal River Unit 3

Description:

The licensee has a safety-grade auxiliary feedwater (AFW) system using a 740 gpm turbine-driven pump and a 740 gpm motor-driven pump as part of the original design. The AFW system consists of two trains, either of which can supply both steam generators. Control grade automatic initiation circuitry and flow detectors were installed to satisfy the short-term (Category A) TMI Action Plan requirements (see Item II.E.1.2 for further discussion in this area).

The licensee has performed a reliability study of the AFW system, as required, and has identified a number of improvements which will enhance the reliability of the system. A number of these improvements will be made during the next refueling outage. The licensee has also committed to an updated reliability study which will take into account the recommendations which have been or will be implemented as a result of the reliability study.

Design/Installation:

All applicable short-term recommendations have been implemented. The licensee has committed to complete most of the remaining improvements including the Emergency Feedwater Initiation and Control System during the next refueling outage, which is scheduled to begin in March 1985. At least one major improvement, protection of the condensate storage tank from external hazards, will not be completed until a subsequent refueling outage. An installation schedule for this item will be provided by the licensee in March 1984.

Operability Tests:

Complete for items already installed. August 1985 for items to be installed during the next refueling outage. Schedule to be provided for remaining items.

Procedures:

As required in conjunction with installation.

Training:

As required in conjunction with installation.

Technical Specifications:

As required in conjunction with installation. Technical Specifications have been issued for all completed modifications and short-term recommendations.

NRC Review Status:

The technical review for this item has been completed by the staff and a Safety Evaluation will be issued in March 1984. All open items discussed in the staff's safety evaluation have been previously addressed by the licensee, who has committed to appropriate plant modifications and technical specification changes.

Davis Besse, Unit No. 1

Description

The licensee has a safety grade system with two turbine driven AFW pumps rated at 1050 gpm at 1050 psi as part of the original plant design. Another motor driven AFW pump, which is not safety grade, rated at 400 to 500 gpm at normal Steam Generator pressure which is used for start up purposes is capable of connection to the emergency power source. The normal flow indication is provided by redundant level transmitters for each Steam Generator. However, to meet the staff positions, the licensee has installed redundant flow indicators to each Steam Generator. One flow indication to each Steam Generator is safety grade. The other flow indication to each Steam Generator is control grade. The safety grade initiation complies with the staff position. The licensee considers the existing system meets the NUREG-0737 requirements. The staff has concluded that the redundant flow indicators at Davis Besse meet NRC requirements.

Design/Installation

The safety grade AFW system is the original design which was licensed for the plant. The licensee concluded that no modifications were needed to meet the NUREG-0737 requirements except for the short term requirement regarding emergency power to the motor driven start up pump and redundant flow indication to each Steam Generator. These items have been completed.

Operability Tests

The AFW system was tested for operability during start up testing for operating license.

Procedures

All procedures consistant with the currently implemented guidelines are complete and implemented.

Training:

Training is complete for current implemented guidelines.

Technical Specifications

The current Technical Specifications and proposed Technical Specifications meet the staff positions except in three areas. The staff has requested proposed Technical Specifications to (1) require lock open valves and surveillance requirements of manual operated valves in the AFW lines, (2) require flow test of the AFW system which would pass water to the Steam Generators before startup after extended cold shutdowns, and (3) require a dedicated individual in communication with the control room to manipulate AFW line manual operated valves when conducting periodic testing of the AFW system.

NRC Review Status

The NRC review is complete except for review of proposed Technical Specifications as noted above. Schedule for completion is estimated at June 1984.

Oconee Units 1, 2 and 3

Description:

The Oconee units each have three safety grade pumps each to provide AFW; one steam driven with a capacity of 1100 gpm at 1127 psi and two motor driven each with a capacity of 500 gpm at 1127 psi. The installation assures that each steam generator has two normal and two emergency flow paths for feedwater. A safety grade flow indication is obtained from a flow orifice for each steam generator, each orifice having two transmitters for redundancy. Each AFW pump has an automatic initiation system independent of the integrated control system which operates on loss of main feed pumps. The steam generators have a safety grade level control system; however, AFW is not automatically initiated on low steam generator level. Additional redundancy is available at this three unit site because manual operation of valves can tie in any AFW pump to supply feedwater to any of the steam generators.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

The licensee has stated that NRC inspection report 83-36 closed this item for all three units per license amendments 110 and 107.

NRC Review Status:

Completed with the resident inspectors review of the licensees modifications. The SERs issued June 1981 and April 1982 concluded that the Oconee Units met the requirements of II.E.1.1 and II.E.1.2. However, there was an open item in the review of II.E.1.1 involving the capability to deliver AFW flow following a seismic event or a tornado. This open item is now being pursued as a plant specific review topic but this effort is a direct result of the NRC review for II.E.1.1.

Rancho Seco

Description

The Rancho Seco AFW is designed to supply an independent source of water to the steam generators during accident and transient condition in the event of a loss of main feedwater supply. The AFWS consists of two interconnected trains with one electric motor driven pump and one combination turbine-drive/motor driven AFW pump (both the motor drive and turbine drive on a common shaft). Each pump has a rated capacity of 840 GPM at 1150 PSIG. Each train is capable of supplying auxiliary feedwater to either or both steam generators. The feedwater flow is either controlled by the existing integrated control system (ICS) or by backup operator manual control from the control room.

All essential portions of the AFWS are designed to Seismic Category 1 criteria except for the use of non-seismically qualified, control valves. Presently, the AFW flow path to the Steam Generators contains a safety grade manually operated by-pass valve in parallel with the flow control valve. The licensee indicated that seismically qualified valves will be installed as part of the AFWS upgrade modifications. These valves have not been installed to date. The design of the upgraded system was reviewed against the guidelines of the Standard Review Plan for AFWS and was found acceptable except for protection against internally generated missiles. The licensee has provided a safety grade flow indication to indicate flow to each steam generator. In addition, the licensee has provided a safety grade automatic initiation for the auxiliary feedwater system which (1) starts the auxiliary feedwater pumps and opens the safety grade bypass valve to bypass the control grade control valve on SFAS (ECCS) initiation, and (2) starts the auxiliary feedwater pumps only on loss of all four Reactor Coolant Pumps. A control grade system provides for automatic initiation on loss of the main feedwater pumps. The motor-driven auxiliary feedwater pump is automatically loaded onto the diesel generator. The licensee now considers that they have met the requirements of NUREG 0737 for safety grade automatic initiation and control II.E.1.2. However, the licensee has proposed installing an Emergency Feedwater Initiation and Control System (EFIC) which the licensee claims to be well beyond the requirements of NUREG 0737. The licensee proposed replacing or augmenting current AFW system instrumentation and controls and other instrumentation and controls with a safety grade EFIC system independent of ICS which will:

1. Automatically initiate Auxiliary Feedwater flow on:

   (a) loss of main feedwater flow as indicated by the reactor power/main feedwater flow reactor trip;

   (b) low level in either steam generator;

   (c) loss of all four reactor coolant pumps;

   (d) low pressure in either SG and loss of main FW; and

   (e) SFAS ECCS actuation.

2. Control AFW flow to maintain SG level of appropriate set points;

3. Control level rate change when required to minimize overcooling;

4. Isolate the main feedwater lines of a depressurizer steam generator;

5. Select AFW to the appropriate steam generator(s) under conditions of (a) steam line break, (b) feedwater line break, or (c) AFW line break downstream of the last check valve .

6. Terminate main feedwater to the steam generators on approach to overfill.

7. Control the atmospheric dump valves to (a) minimize challenge to main steam safeties and (b) provide for cooldown from the main control room.

Design/Installation:

All the short term lessons learned were discussed during the Rancho Seco hearing and the required modifications were made at the time (1979). During the last refueling outage (completed in August 1983) the licensee (a) modified the AFWS piping so that the AFWS pumps could be tested without taking both AFWS trains out of service, (b) made provisions to automatically load the motor driven AFW pump on to the diesel (2) changed the AFW flow indication from control grade to safety grade.

The EFIC system modifications are now expected to be made in the 1986 refueling outage.

Procedures:

All procedures consistent with the current design of the AFW system are complete and implemented.

Operator Training:

Training is completed for the current procedures.

Technical Specifications:

The current Technical Specification are consistent with the current design of the AFW system except for testing of the AFW pumps. The pumps are currently using the level in the condensate storage tank for determining flows. The tech spec will be revised to utilize the safety grade flow indication for testing the pumps. When the EFIC system is installed, the Tech Specs will require modification.

NRC Review Status:

The review of the upgraded AFW systems against the criteria of the Standard Review Plan, the short term lessons learned and long term lessons learned, NUREG-0611 and NUREG-0635, the licensees reliability analysis and the licensees reevaluation of the AFWS flow requirement was completed. In addition the review of the safety grade initiation and flow indication (II.E.1.2) based on the EFIC system preliminary design was completed. Two items remain to be completed. These are the (1) evaluation of the upgrade AFW system for internally generated missiles and (2) review of final EFIC drawings to verify that the conclusions in the safety evaluation for II.E.1.2 are still valid.

Three Mile Island Unit 1

Description:

The TMI-1 emergency feedwater (EFW) system consists of two motor-driven pumps and one turbine-driven pump which discharge through the two feed headers into the steam generators. Each feed header feeds one steam generator. A single flow control valve in each feed header regulates flow. The EFW system was the central design issue of the TMI-1 Restart proceedings and the capabilities and limitations of the system are well documented. (See NUREG-0680 and Supplements, Atomic Safety & Licensing Board Partial Initial Decision dated December 14, 1981, and Atomic Safety & Licensing Appeal Board decision ALAB-729 dated May 26, 1983.) The Licensing Board decision, as modified by the Appeal Board, is that the system will be able to support safe plant operation upon completion of certain short-term modifications and procedural revisions. Additional modifications, the so called long-term modifications, are necessary to bring the EFW system into conformance with the criteria of NUREG-0737, Items II.E.1.1, and II.E.1.2, and otherwise environmentally and seismically qualify the system and make it single failure proof.

The TMI-1 EFW motor-driven pumps are safety-grade with the exception of an outstanding environmental qualification question which is being addressed under the generic program. The turbine-driven EFW pump is not safety-grade due, at a minimum, to a lack of seismic qualification. At the time of restart, the TMI-1 EFW pumps will automatically start on loss of main feedwater and on loss of all four reactor coolant pumps, independent of the control grade integrated control system (ICS). However, the ICS does control the single flow control valve in the EFW supply to each of the two steam generators. There will be a remote manual capability to open the EFW flow control valves from the control room should the ICS fail. There will also be redundant safety-grade indication of flow to each steam generator. Other pre-start modifications and system capabilities are described in NUREG-0680 and other hearing related documents.

The principal long-term modifications to be completed at the first refueling after restart are to provide: safety-grade automatic EFW control, automatic initiation on low steam generator level, redundant flow control and block valves, and safety-grade condensate storage tank level indications. Certain additional modifications may be necessary to achieve environmental and seismic qualification. The upgraded EFW system would meet the staff unavailability acceptance criterion for feedwater transients when implemented.

Design/Installation:

Short-term modifications: Complete.

Long-term modifications: Design and engineering in progress.

Installation scheduled for first refueling after restart, or December 1984 if the plant is shutdown for all of 1984. The flow control valves, which have a one-year procurement lead time, were ordered in June 1983. The instrumentation cabinets, which have a 46 week lead time, had not been ordered at the time of the site visit (January 1984).

Operability Tests:

Short-term modifications: Complete, except for test of auto-initiation during trip from power on loss of main feedwater. Some additional testing of the flow indication system may be necessary because of observed oscillations at low flow.

Long-term modifications: First refueling after restart.

Procedures:

Short-term modifications: Complete.

Long-term modifications: First refueling after restart.

Training:

Short-term modifications: Complete.

Long-term modifications: First refueling after restart.

Technical Specifications:

Short-term modifications: Complete. Issued under amendment 78, dated October 20, 1982.

Long-term modifications: Submittal pending.

NRC Review Status:

Short-term modifications: Complete, except for (1) the issue of EFW flow indication oscillations at low flow, and (2) the acceptability of the GPU Nuclear test intended to demonstrate that adequate EFW flow can be provided with the recirculation lines open. These reviews will be complete prior to restart.

Long-term modifications: In progress.

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

NUREG-0737 Position

In the long term, all auxiliary feedwater systems shall have an automatic initiation capability. The automatic signals and circuits shall meet safety grade requirements. In addition, safety grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room. These safety grade indicators must be powered from the emergency buses. B&W plants must provide as a minimum two auxiliary feedwater flowrate indicators for each steam generator.

The following describes how each B&W licensee met these guidelines:

ANO-1

Description:

A detailed description of the ANO-1 AFW system is found under Item II.E.1.1. Basically ANO-1 has a control grade automatic initiation system for their AFW and it will be upgraded to safety grade when the EFIC system is installed. EFIC stands for Emergency Feed Initiation and Control system, a control system which goes beyond our requirements. AFW flow indication is safety grade already.

Design/Installation:

The design of all upgrades is almost complete with detailed review in progress. Installation of the safety grade EFIC will be completed by the end of 1984.

Operability Tests:

The licensee has indicated that the startup tests for EFIC will be a major effort.

Procedures:

The procedures for the EFIC system will be completed and implemented by the end of 1984.

Training:

Training will be completed on the EFIC system by the end of 1984.

Technical Specifications:

Technical Specification changes will be submitted in June 1984.

NRC Review Status:

NRC pre-implementation review for the control grade modification is complete. An SER has been issued on the safety grade modifications to EFIC.

Crystal River Unit 3

Description:

The licensee installed control grade AFW flow indication and automatic AFW initiation to satisfy the short-term (Category A) TMI Action Plan Items. The present system has redundant flow control paths in each AFW line to the Steam Generators. The licensee has committed to install a significantly upgraded AFW control system referred to as the Emergency Feedwater Initiation and Control, or EFIC, system designed by Babcock and Wilcox. This system will utilize steam generator level transmitters and solenoid-actuated control valves controlled by a logic system and auxiliary relays. The system will also include safety-grade AFW flow instrumentation, modulated AFW flow controlled by steam generator level, main feedwater overfill protection, and improvements to the steam/feed line rupture logic matrix. A pre-implementation design review by the NRC has been completed to verify that the new system will meet the requirements of NUREG-0737, Item II.B.1.2. The features of the EFIC system go beyond the minimum requirements of NUREG-0737.

Design/Installation:

The overall design of the EFIC system has been submitted to the NRC for review and is described in Babcock and Wilcox Technical Document No. 15-1121209, Revision 4, dated August 30, 1982. Although prefabrication work and detailed design refinements are presently in progress, installation of the EFIC system will not begin until March 1985, which is the start of the next scheduled refueling outage at Crystal River Unit 3.

Operability Tests:

The EFIC system and auxiliary feedwater system as a whole will be tested at the conclusion of the 1985 refueling outage, presently scheduled to start in March 1985 and last for approximately 20 weeks.

Procedures:

To be developed in conjunction with the EFIC modifications.

Training:

To be performed in conjunction with the EFIC modification.

Technical Specifications:

To be submitted by the licensee prior to the March 1985 refueling outage and issued as part of the reload amendment.

NRC Review Status:

The NRC issued a Safety Evaluation on September 27, 1982 which documented the acceptability of the proposed design in meeting NUREG-0737, Item II.E.1.2. A subsequent letter to the licensee, dated November 15, 1982, the staff requested additional detailed electrical schematic diagrams which will be used to re-examine the final design of the EFIC system. As stated in NUREG-0737, a post-implementation review will also be performed following installation of this system.

Davis Besse, Unit No. 1

Description

Davis Besse has an original installed AFW System which provides safety grade actuation and level control. The system was designed with three modes of control available to the operator. These are "manual", "auto essential", and "ICS (Integrated Control System) control". The operator selects the desired control mode by a control switch on the main control board. Because the staff required the AFWS control to be independent of the ICS, the licensee placed a stop on the control switch during the May 1979 shutdown to prevent the switch from being placed in the ICS mode. The AFW is automatically initiated from a Steam and Feedwater Line Rupture Control System (SFRCS). The SFRCS starts the AFWS in the event of a main steam line or feedwater line rupture, loss of both main feedwater pumps, or the loss of all four RC pumps.

The operators normally observe steam generators levels to determine flow to the steam generators from the AFWS. However, as a result of the NUREG-0737 requirements, the licensee installed one safety grade AFW flow indication to each steam generator and one control grade AFW flow indication to each steam generator. In view of the existence of redundant flow indication and the diverse means to access the status of the AFWS, the staff found that the NUREG-0737 requirements were met.

Design/Installation

The safety grade initiation and control are of the original design which was licensed for the plant. The licensee installed redundant AFW flow indication to each steam generator to satisfy the NUREG-0737 requirements. One channel to each steam generator is safety grade. The other channel to each steam generator is control grade, but will be included in the plant technical specifications.

Operability Tests

The AFWS initiation and control was tested for operability during start up testing for the operating license. The AFW flow indication was tested following installation. This item is complete.

Procedures

All associated procedures have been developed and implemented.

Training

Training is complete.

Technical Specifications

The technical specifications related to the initiation and control have been in place since issuance the license. The licensee was requested to submit proposed technical specifications and has submitted proposed technical specifications relating to the flow indication. Staff review is in progress.

NRC Review Status

The NRC staff review is complete for review of the proposed technical specification as noted above. Schedule for completion is estimated for June 1984.

Oconee Units 1, 2 and 3

Description:

An adequate description of the Oconee AFW system is found in Item II.E.1.1. Each units AFW system has an automatic initiation system independent of the integrated control system which initiates flow on a loss of main feed pumps. A safety grade flow indication is obtained from a flow orifice for each steam generator, each orifice having two transmitters for redundancy.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

The licensee has stated that NRC inspection report 83-36 closed this item for all three units per license amendments 110 and 107.

NRC Review Status:

Completed with the resident inspectors review of the licensees modifications. The SERs issued June 1981 and April 1982 concluded that the Oconee Units met the requirements of II.E.1.2.

Rancho Seco Nuclear Generating Stations

Description:

A detailed description of the Rancho Seco AFW system is found under section II.E.1.1 Auxiliary Feedwater System Evaluation. As indicated previously (Section II.E.1.1), the Rancho Seco licensee considers that its AFW system presently has safety grade automatic initiation features and safety grade flow indication which satisfies II.E.1.2. However, the licensee is installing an upgraded control system (EFIC) which goes beyond our requirements for II.E.1.2. EFIC will be installed at the next refueling outage (1986). The NRC staff agrees that once EFIC is installed the requirements of II.E.1.2 will be fully satisfied but has some questions about the present system fully satisfying these requirements.

Design/Installation:

Complete.

Procedures:

Complete.

Operator Training:

Complete.

Technical Specifications:

Complete.

NRR Review Status:

Complete for the proposed EFIC system. After EFIC is installed, the staff will review the final drawings to ensure that the staff conclusions on II.E.1.2 are still valid.

Three Mile Island, Unit 1

Description:

A detailed description of the TMI-1 AFW initiation and flow indication systems and their upgrades are found under Item II.E.1.1. The basic licensee conclusion is that the short term modifications are complete and the long term modifications will be done at the next refueling outage.

Design/Installation:

Short-term modifications: Complete.

Long-term modifications: Design and engineering in progress. Installation scheduled for first refueling outage after restart.

Operability Tests:

Short-term modifications: Complete, except for test of auto-initiation during trip from power on loss of main feedwater. Some additional testing of the flow indication system may be necessary because of observed oscillations at low flow.

Procedures:

Short-term modifications: Complete.

Long-term modifications: Next refueling outage.

Training:

Short-term modifications: Complete.

Long-term modifications: Next refueling outage.

Technical Specifications:

Short-term modifications: Complete.

Long-term modifications: Next refueling outage.

NRC Review Status:

Short-term modifications: Complete, except for (1) the issue of EFW flow indication oscillations at low flow, and (2) the acceptability of the GPU Nuclear test intended to demonstrate that adequate EFW flow can be provided with the recirculation lines open. These reviews will be complete prior to restart.

Long-term modifications: In progress.

II.E.4.1.2 DEDICATED HYDROGEN PENETRATIONS

NUREG-0737 Position

Plants using external recombiners or purge systems for post-accident 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 following describes how each licensee met the guidelines:

ANO 1

Description:

The licensee has a designated dedicated containment penetration to which a hydrogen recombiner could be attached in an emergency. In addition, the licensee has a designated area where the recombiner could be set in an emergency. The licensee has an agreement with another utility to obtain a recombiner in an emergency within an adequate time frame.

Design/Installation:

The penetration is installed.

Operability Tests:

Other than normal technical specifications testing of containment penetrations, there is no operability requirement on this penetration.

Procedures:

Not applicable

Training:

Not applicable

Technical Specifications:

No special changes needed

NRR Review:

Complete

Crystal River

Description:

The licensee has a designated dedicated containment penetration to which a hydrogen recombiner could be attached in an emergency. The licensee has a designated area to physically place a recombiner. The licensee has evaluated the availability of hydrogen recombiners from either a vendor or another utility and has determined that they can obtain a hydrogen recombiner within an adequate time frame in an emergency.

Design/Installation:

The penetration is installed.

Operability Tests:

Not applicable

Procedures:

Not applicable

Training:

Not applicable

Techncial Specifications:

No special changes needed.

NRR Review:

Complete

Davis Besse

Description:

The licensee has a designated dedicated containment penetration to which a hydrogen recombiner could be attached in an emergency. In addition, the licensee has a designated area where the recombiner will be set in an emergency. The licensee has evaluated the availability of hydrogen recombiners from either a vendor or another utility and has determined that they can obtain a hydrogen recombiner within an adequate time frame in an emergency.

Design/Installation:

The penetration is installed.

Operability Tests:

Not applicable

Procedures:

Not applicable

Training:

Not applicable

Technical Specifications:

No special changes needed.

NRR Review:

Complete.

Oconee 1, 2, and 3

Description:

The licensee has designated dedicated containment penetrations to which a hydrogen recombiner could be attached in an emergency. In addition, the licensee has designated areas where the recombiners will be set in an emergency. The licensee owns two hydrogen recombiners and one is in storage at the Oconee site.

Design/Installation:

The penetration is installed.

Operability Tests:

Not applicable

Procedures:

Not applicable

Training:

Not applicable

Technical Specifications:

No special changes needed.

NRR Review:

Complete.

Rancho Seco

Description:

The licensee has a designated dedicated containment penetration to which a hydrogen recombiner could be attached in an emergency. The licensee has designated a general area to set a recombiner. The licensee has evaluated the availability of hydrogen recombiners from either a vendor or another utility and has determined that they can obtain a hydrogen recombiner within an adequate time frame in an emergency.

Design/Installation:

The penetration is installed.

Operability Tests:

Not applicable

Procedures:

Not applicable

Training:

Not applicable

Techncial Specifications:

No special changes needed.

NRR Review:

Complete.

TMI-1

Description:

The licensee has designated a dedicated containment penetration to which hydrogen recombiners are attached. The licensee has physically installed two hydrogen recombiners (one came from TMI-2)

Design/Installation:

The penetration is installed.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Techncial Specifications:

No changes proposed.

NRR Review:

Complete.

II.F.1 ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION

II.F.1.1 NOBLE GAS EFFLUENT MONITOR

NUREG-0737 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 microcuries per 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 microcuries per 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.

The following describes how each licensee met the guidelines:

ANO-1

Description:

The monitoring for this item is performed using a system called the "Super Particulate Iodine and Noble Gas System" (SPINGS) made by Eberline. There are four SPINGS dedicated to ANO-1 and one to the PASS building which is shared between ANO-1 and ANO-2. These are in-line monitors located at selected gaseous effluent streams. The gaseous monitor has three ranges covering $1\text{-}1\times10^{-7}$ microcuries per cc to 1.3×10^{5} microcuries per cc. Currently, the readouts from the SPINGS are recorded in the control room. With installation of the Gaseous Effluent Radiation Monitoring System (GERMS) capability, the plant computer will receive and process all the radiation and meterological data to calculate site doses.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

To be submitted by March 19, 1984.

NRC Review Status:

A post implementation review will be found at a later date.

Crystal River Unit 3:

Description:

Mid-range and high-range noble gas radiation monitors meeting the range requirements have been installed to augment the existing low-range monitors which sample the Reactor Building and Auxiliary Building environments. The equipment includes valving and logic for control of sample flow to each detector based on activity levels and subsequent purging to protect the instruments from over-range exposures. The system also provides for monitoring with an automatic isotopic measurement system, for "grab-samples" for off-site analyses, and for monitoring steam generator safety relief valve discharge radioactivity. The control panel for this system is adjacent to the control panel for the post-accident sampling system.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Proposed technical specifications were submitted by the licensee in June 1983. The technical specifications will be issued in a license amendment scheduled for April 1984.

NRC Review Status:

NRC will conduct a post-implementation review (not yet scheduled).

Davis Besse, Unit No. 1

Description

The licensee has installed a redundant inline monitoring system which samples the gaseous effluents in the vent stack. The monitors are mounted on skids located on the turbine room floor. The skids contain the noble gas monitors and the iodine particulate monitors. The range of the noble gas monitors are from normal conditions to 10^5 microcuries per cc (Xe-133).

Design/Installation

Complete.

Operability Tests

Complete.

Procedures

Complete.

Training

Complete.

Technical Specifications

Proposed Technical Specifications were requested by Generic Letter 83-37 dated November 1, 1983. Also, proposed technical specifications were requested by letter dated January 28, 1982. At the meeting with the licensee of February 3, 1984, the licensee indicated that since this system was replacing an existing system, the current technical specifications cover the system. In reply to Generic Letter 83-37 the licensee stated that the technical specifications for these monitors were submitted as part of the Radiological Effluent Technical Specifications.

NRC Review Status

The NRC review is complete and a letter dated January 28, 1982 closed this item. The staff will review the technical specification issue when reviewing the licensee's response to Generic Letter 83-37.

Oconee Units 1, 2 and 3

Description:

One noble gas monitor, made by Victoreen, is provided per unit to measure the activity routed through the stack. The instrument is a normal area monitor modified to withstand high radiation levels. The range of the instrument is 0.1 to 10^7 microcuries per cc.

Design/Installation:

Complete.

Operability Tests:

The instruments have been found to perform well when new but appear to develop a noise problem due to moisture in the cable after some months of service. The licensee feels that the latest measures to correct the problem will be effective.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Proposed tech specs will be submitted by April 2, 1984.

NRC Review Status:

A post implementation review will be conducted at a later date.

Rancho Seco

Description:

Three General Atomic Wide Range Gas Monitors were installed to monitor both noble gases and iodine and particulates in the auxiliary building vent, the new radwaste area vent, and the containment building atmosphere. The range for noble gas measurement is 10^{-7} to 10^{+5} microcuries per cc.

In addition, two monitors have been mounted on the main steam lines to assess releases from the main steam relief valves and the atmosphere dump valves. The readout for the monitors is a digital based system with computer driven CRT and keyboard in the control room and a strip chart recorder in the Technical Support Center (TSC).

The licensee has not installed a high range noble gas monitor on the containment purge line because the licensee is required by Technical Specification and commitment to keep the purge and vent valves closed with their breakers deenergized.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training

Complete.

Technical Specifications:

The licensee submitted proposed Technical Specifications by letter dated January 23, 1984.

NRC Review Status:

NRC review of the licensee position with regard to the purge line monitor is complete and the SER was issued March 16, 1984. Region V inspection of the monitoring system was initiated and is continuing.

Three Mile Island Unit 1

Description:

Post-accident noble gas monitoring modifications for TMI-1 involved extending the monitoring ranges for the reactor building purge exhaust, auxiliary and fuel handling building exhaust, condenser off-gas exhaust and main steam line radiation monitors. Ranges have been increased to 10^5 microcuries per cc for the reactor building purge exhaust and condenser off-gas, and to 10^3 microcuries per cc for the auxiliary and fuel handling building exhaust and main steam lines. GPU Nuclear encountered problems with the new monitoring equipment as originally provided by the manufacturer. The problems related to the physics, shielding design, maintainability, sensitivity and range requirements, calibration and electronics of the monitors. GPU Nuclear resolved the problems by contracting with a laboratory to examine, modify as required, and calibrate the monitors. Additionally, studies indicated that the auxiliary and fuel handling buildings and reactor building monitors, if left in their original locations, would experience high post-accident background radiation levels. This led to the construction of a new dedicated building for these monitors.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Request submitted and under review by NRR.

NRC Review Status:

Complete.

II.F.1.2 Iodine/Particulate Sampling

NUREG-0737 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 absorption on charcoal or other media, followed by onsite laboratory analysis. The equipment shall have the capability to determine quantative releases of radioiodines and particulates for dose calculation and assessment. The shielding envelope shall permit a level of 10^2 microcuries per cc of gaseous radioactive and particulates to be deposited on a sampling media; assuming an average energy (e) of 0.5 MeV and a 30 minute sampling time.

The following describes how each licensee met the guidelines:

ANO-1

Description:

The SPINGS system described in Item II.F.1.1 is also used to measure iodine/particulate by passing the gaseous steam through a filter and measuring the radioactivity collected. The sensitivity of the system has been indicated to be 10^{-3} microcuries per cc. At higher levels of activity, the filter would be removed for measurement to be performed in the laboratory. The licensee has indicated that the sampling time will be adjusted to obtain a suitable sample to meet the requirements of II.F.1.2.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

To be submitted by March 19, 1984.

NRC Review Status:

A post implementation review will be performed at a later date.

Crystal River Unit 3

Description:

The system installed to collect samples for iodine and particulate radioactivities in the Reactor Building and Auxiliary Building is used in conjunction with the new mid- and high-range noble gas sample system described under Item II.F.1.1 for Crystal River Unit 3. The system utilizes Kaman Particulate Samplers with particulate and iodine grab sample cartidges.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Proposed technical specifications were submitted by the licensee in June 1983. The technical specifications will be issued in a license amendment scheduled for April 1984.

NRC Review Status:

NRC will conduct a post-implementation review.

Davis Besse, Unit No. 1

Description

The licensee has installed a redundant inline monitoring system which samples the gaseous effluents in the vent stack. The monitors are mounted on skids located on the turbine room floor. The skids contain the noble gas monitors and the iodine particulate monitors. The capability of the iodine particulate monitors encompass concentrations to 10^{2} microcuries per cc, 30 min. sampling time, and an average gamma energy at 0.5 MeV.

Design/Installation

Complete.

Operability Tests

Complete.

Procedures

Complete.

Training

Complete

Technical Specifications

Proposed Technical Specifications were requested by Generic Letter 83-37 dated November 1, 1983. Also, proposed technical specifications were requested by letter dated January 28, 1982. At the meeting with the licensee of February 3, 1984, the licensee indicated that since this system was replacing an existing system, the current technical specifications cover the system. In response to Generic Letter 83-37, the licensee has stated that technical specifications for these monitors were included in their Radiological Effluent Technical Specifications.

NRC Review Status

The NRC review is complete and a letter dated January 28, 1982 closed this item. The staff will review the technical specification issue when reviewing the licensee's response to Generic Letter 83-37.

Oconee Units 1, 2 and 3

Description:

An iodine and particulate monitor is provided for each unit at the stack through which all potentially radioactive gaseous effluents are vented. A sampling line draws a sample, passes it through filter and returns the gas to the vent. The iodine/particulate monitoring is performed by removing the filter cartridge and measuring the radioactivity trapped in it. The sensitivity of the system is such that it can measure 10^{-11} microcuries per cc and saturates at 10^{-8} microcuries per cc. If the vented gases contain extremely high levels of radioactivity, a radiation detector held at the sampling line is expected to provide an estimate of the release. However the correlations for this type of reading begin at 3000 microcuries per cc. The licensee has not provided a basis for determing that the design requirements in NUREG-0737 Table II.F.1-2 (10^2 microcuries per cc with 30 minute sampling time) will be met.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Proposed Technical Specifications will be submitted April 2, 1984.

NRC Review Status:

NRC approved the system for Oconee Units 1, 2 and 3 on April 7, 1980. A post implementation review will be conducted at a later date.

Rancho Seco

Description:

The iodine and particulate samplers consists of three parallel path iodine and particulate collection filters, one of which is remotely operable, that are part of the General Atomic Wide Range Gas Monitors described in Item II.F.1.1. The filters are periodically removed from the monitor and are analyzed in the chemical laboratory for particulate and iodine content. The samples can handle up to 10^{+5} microcuries per cc.

Design/Installation:

Complete.

Procedures:

Complete.

Technical Specifications:

Licensee has not proposed any technical specifications.

NRC Review Status:

Same as item II.F.1.1.

Three Mile Island Unit 1

Description:

GPU Nuclear has installed three high-range radioiodine and particulate sampling systems which can handle concentration of 10^2 microcurie per cc to monitor condenser off-gas exhaust reactor building purge exhaust and auxiliary and full handling building exhaust. The monitor automatically actuates upon alarm of the noble gas monitor, and can also be manually actuated from the control room. Each monitor has three trains. One train provides continuous sampling. A second train for samples four seconds in every 40 seconds. A third train samples for four seconds in every 400 seconds. The sampling units were prototype units and thus required substantial field modification to enhance performance and reliability. The systems were designed to meet the requirements of NUREG-0737.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Request submitted.

NRC Review Status:

Complete.

II.F.1.3 CONTAINMENT HIGH RANGE MONITOR

NUREG-0737 Position

Radiation-level monitors with a maximum range of 10^8 rad/hr (beta and gamma) or alternatively /R/hr to 10^7 R/hr (gamma only) shall be installed inside containment. 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.

The following describes how each licensee met the guidelines:

ANO-1

Description:

ANO-1 has two General Atomic high range radiation monitors with a range of 1 to 10^7 rads/hr for gamma radiation. The detectors are mounted in the upper part of the containment between the secondary shield wall and the containment. The readout is provided in the control room. The calibration is done electronically.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

To be submitted by March 19, 1984.

NRC Review Status:

A post implementation review will be performed in future.

Crystal River Unit 3

Description:

Crystal River Unit 3 has installed two General Atomic Gamma Detection System instruments with a range of 0-1x10^8 rads/hr to satisfy the requirements of NUREG-0737, Item II.F.1.3. The detectors are located in the Reactor Building at the top of the "D ring" shield structure. Readout modules are located in the main control room.

Design/Installation:

Complete.

Operability Tests:

Complete for both detectors; however, electronic problems were found on one electronics package for RMG-30 and this unit was returned to the vendor for troubleshooting. The unit has been returned and re-installation is expected in May 1984.

Procedures:

Completed.

Training:

Completed.

Technical Specifications:

Proposed technical specifications were submitted by the licensee in June 1983 and will be incorporated in a license amendment to be issued in April 1984.

NRC Review Status:

NRC will conduct a post-implementation review.

Davis Besse, Unit No. 1

Description

The licensees has installed redundant high range radiation monitors which will measure to a maximum range of 10^8 rad/hr. The monitors are located in the containment in a manner to provide assessment of the area radiation conditions inside containment.

Design/Installation

Design and installation is complete.

Operability Tests

Testing is complete.

Procedures

Complete.

Training

Complete.

Technical Specifications

This monitor replaces previously installed equipment. The licensee has stated in response to Generic Letter 83-37 that existing technical specifications cover this monitor.

NRC Review Status

The NRC review is complete and a letter dated January 11, 1982 closed this item. The staff will review the revised technical specification when reviewing the licensees response to Generic Letter 83-37 dated November 1, 1983.

Oconee Units 1, 2 and 3

Description:

The Oconee units each have two Victoreen high range radiation monitors, with a range of 1 to 10^7 rads/hr for gamma radiation. The detectors are located mid way up containment on either side of the building. The readout is provided in the control room. The calibration is done with a source as well as electronically.

Design/Installation:

Complete.

Operability Tests:

A problem has been noted in Units 1 and 3 due to signal leakage through the cables. It has been corrected in Unit 2. Units 1 and 3 will be repaired at the next refueling outage.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Proposed tech specs will be submitted by April 2, 1984.

NRC Review Status:

A post implementation review will be performed in future.

Rancho Seco

Description:

Rancho Seco has installed two General Atomic Model RD-23 high range radiation detectors with range of 0 to 10^7 R/Hr for gamma at about the 60 ft. level in the reactor building. The readouts for the detectors is the same digital based radiation monitoring system described in Item II.F.1.1.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training

Complete.

Technical Specifications:

Proposed Technical Specification were submitted by letter dated January 23, 1984.

NRC Review Status:

NRR only reviewed exemption requests to the requirements. Rancho Seco did not request an exemption. The regions will conduct a post implementation inspection.

Three Mile Island Unit 1

Description:

GPU Nuclear has installed two safety-grade ion chamber radiation monitors inside containment with a range up to 10^7 R/hr for gamma only. One monitor is on each "D" ring. The calibration units for the monitors had not been designed at the time of purchase, and the means of calibration is not yet established. The system is designed to meet the requirements of NUREG-0737.

Design/Installation:

Complete.

Operability Tests:

Complete, except for calibration means which is not yet established.

Procedures:

Complete, except for calibration means which is not yet established.

Training:

Complete, except for calibration means which is not yet established.

Technical Specifications:

Submitted.

NRC Review Status:

Complete.

II.F.1.4 CONTAINMENT PRESSURE

NUREG-0737 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 for steel, and -5 psig for all containments.

The following describes how each licensee met the guidelines:

ANO-1

Description:

The licensee has installed two redundant, qualified pressure transmitters, Model 1153A by Rosemont to meet this requirement. The range is -5 psig to 210 psia.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

To be completed by March 19, 1984.

NRC Review Status:

NRC has indicated approval of the ANO-1 system in June 1983.

Crystal River Unit 3

Description:

Crystal River Unit 3 has installed a Reactor Building (RB) pressure monitoring system which monitors and records pressure in the range 0-280 psia. RB design pressure is 70 psia. Indication is provided in the main control room and a recorder-indicator for each channel is located in the EFIC room, with phone communication to the control room. This parameter may also be selected on the plant process computer.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

The licensee submitted proposed technical specifications in June 1983, which will be issued in a license amendment in April 1984.

NRC Review Status:

A Safety Evaluation was issued by the NRC on January 18, 1984 concluding that the requirements of NUREG-0737 Item II.F.1.4 have been met. A post-implementation review is planned.

Davis Besse, Unit No. 1

Description:

The CPMS indicator and recorder are separate devices with a range of 0-280 psia. The CPMS indicator has a system accuracy of 1.0% of full scale and a response time of 5 seconds. The recorders are computer data points on disk files which can be displayed on the CRTs in the control room and the Technical Support Center. The displays would be running approximately one minute behind real time accident conditions. The CPMS recording has a system accuracy of 0.6% of full scale and a system response time of 0.8 seconds.

Design/Installation:

The CPMS is complete.

Operability Tests:

Testing was completed prior to declaring the system operational.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Submitted.

NRC Review Status:

The NRC review is complete and a Safety Evaluation Report was issued by letter dated May 24, 1983. The staff will review the proposed technical specification submittal with the review of the licensee's response to Generic Letter 83-37.

Oconee Units 1, 2 and 3

Description:

Each unit has redundant containment pressure monitors having a range of -5 psig to 175 psig. There are indicator and recorder readouts in the control room on one of two channels and only indicator readout on the second channel. The signal is also fed to the plant computer with a scan frequency of once every 10 seconds.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Proposed tech specs will be submitted by April 2, 1984.

NRC Review Status:

The NRC safety evaluation dated September 1, 1983 closed out this issue with the exception of environmental qualification, which will be performed later.

Rancho Seco

Description:

Rancho Seco has installed a redundant -5 to 180 psig containment pressure monitoring system consisting of two safety grade pressure transmitters with associated valves, piping and electrical wiring. The containment pressure is currently logged on a trend recorder and on the Safety Parameter Display System (SPDS). The trend recorders will be replaced by the Interim Data Aquisition and Display System (IDADS) when it is operational. All equipment for the monitoring system up to and including the isolation cabinets are class 1E (safety grade).

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training

Complete.

Technical Specifications:

The licensee submitted proposed technical specifications by letter dated January 23, 1984.

NRC Review Status:

NRC review was completed and the SER was issued by letter dated July 28, 1982.

Three Mile Island Unit 1

Description:

GPU Nuclear has installed a containment pressure monitoring system consisting of redundant channels, each with a narrow range (-5 psig to +5 psig) and wide range (0-175 psig) transmitters. A dual channel strip chart recorder provides continuous control room display of containment pressure and provides a permanent record. Input is also provided to the plant computer. The system is designed to meet the requirements of NUREG-0737.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Request submitted and under review by NRR.

NRC Review Status:

In progress.

II.F.1.5 CONTAINMENT WATER LEVEL

NUREG-0737 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.

The following describes how each licensee met the guidelines:

ANO-1

Description:

ANO-1 uses water level monitors of the float type made by Gems-Delavel. There is one channel provided for the sump over the whole depth of 53 inches and two channels for the reactor building, from the reactor building floor to a height of 144 inches. The monitored volumetric capacity is estimated to be greater than 600,000 gallons. Readout for each instrument is available in the control room. The instruments are safety grade and environmentally qualified.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

To be completed by March 19, 1984.

NRC Review Status:

NRC has indicated approval of the ANO-1 system in June 1983.

Crystal River Unit 3

Description:

A system utilizing four float-type level transmitters has been installed at Crystal River Unit 3 to monitor and record Reactor Building (RB) water level. One set of 2 instruments measures RB sump level up to the top of the sump. The other pair of transmitters measures RB floor water level up to 10 feet above floor level. Level indicators are provided on the "ES" section of the main control board and indicator/recorders are located in the EFIC room with phone communications to the main control room. A RB level of approximately 7.5 feet above floor level cooresponds to a volume of 600,000 gallons, the volume referred to in Regulatory Guide 1.97, Revision 2.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

The licensee submitted proposed technical specifications in June 1983, which will be issued in a license amendment in April 1984.

NRC Review Status:

A Safety Evaluation was issued by the NRC on January 18, 1984 concluding that the requirements of NUREG-0737 Item II.F.1.5 have been met. A post-implementation review is planned.

Davis Besse, Unit No. 1

Description

The licensee has installed a narrow-range (0-48 inches) CWLMS in the sump and a wide range (0-55Ft) CWLMS in the containment, both of which have indication in the control room and recording as computer data points on disk files which can be displayed on the CRTs in the control room and Technical Support Center. The narrow-range indicator has an accuracy of 1.4% of full scale and the narrow-range recorder has an accuracy of 1.2% of full scale. The wide-range indicator and recorder both have an accuracy of 7.8% of full scale.

Design/Installation

The CWLMS is complete.

Operability Tests

Testing was completed prior to declaring the system operational.

Procedures

Complete.

Training

Complete.

Technical Specifications

Submitted.

NRC Review Status

The NRC review is complete and a Safety Evaluation Report was issued by letter dated May 24, 1983. The staff will review the proposed technical specification submittal with the review of the response to Generic Letter 83-37.

Oconee Units 1, 2 and 3

Description:

Each unit has the capability to monitor three separate water levels: (1) the containment level, (2) the normal sump level, and (3) the emergency sump level. The normal and emergency sump are separate collection areas. For all three levels, one of two redundant channels has both indicator and recorder readouts in the control room, and the second channel has indicator readout alone. The ranges of the instruments are as follows: containment, 0 to 15 feet from the containment floor; normal sump, 0 to 2 feet; and emergency sump, 0 to 3 feet with reference to sump floor level.

Design/Installation:

Complete.

Operability Tests:

It was found after installation that the narrow range monitors were not qualified for submerged operation. Properly qualified replacements have been procured and will be installed in Unit 3 in March 1984. Procurement and replacements for Units 1 and 2 will be consistent with each upcoming refueling outage (approximately January 1985 for Unit 1 and January 1986 for Unit 2).

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Proposed tech specs will be submitted by April 2, 1984.

NRC Review Status:

Temporary operation with instruments not fully qualified has been approved on September 1, 1983. A post implementation review will be performed at a later date.

Rancho Seco

Description:

The licensee installed two wide range flood level sensors (0 to 10 ft. above the sump floor level) in the containment which are connected to two level transmitters installed outside the containment. The water level indication is provided through a trend recorder and the SPDS. The IDADS, when operational, will replace the trend recorder. In addition, the licensee added level sensors inside the emergency sump with two level switches to provide an alarm in the control room through a temporary computer based system diagnostic unit. The alarm will be put on IDAD eventually. The alarm actuates when the level is at 3 ft. and is decreasing (to warn of potential loss of pump suction).

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training

Complete.

Technical Specifications:

The licensee submitted proposed technical specifications by letter dated January 23, 1984.

NRC Review Status:

NRC review was completed and the SER was issued by letter dated July 28, 1982.

Three Mile Island Unit 1

Description:

GPU Nuclear has installed two sets of redundant containment water level monitors in the TMI-1 reactor building with control room readout. One set has a range of 0-90 inches above the reactor building sump, and the second set has a range of 0-90 inches above the reactor building floor. The top of the second set of monitors is 10 feet above the sump floor. The system is designed to meet the requirements of NUREG-0737.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Request submitted.

NRC Review Status:

In progress.

II.F.1.6 CONTAINMENT HYDROGEN

NUREG-0737 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.

The following describes how each licensee met the guidelines:

ANO-1

Description:

ANO-1 uses hydrogen monitors made by Concept Delphi with a range of 0-10%. They are installed in the auxiliary building and draw samples off the hydrogen purge lines. Readouts are provided in the control room.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Proposed tech specs will be submitted by March 19, 1984.

NRC Review Status:

NRC issued an SER in June 1983 approving the licensee's system. A post implementation review will be performed at a later date.

Crystal River Unit 3

Description:

Two independent trains of Teledyne Model 225 CM thermal conductivity hydrogen gas monitors have been installed at Crystal River Unit 3 to monitor hydrogen gas concentrations in the Reactor Building (RB) dome area and recirculation duct. The system can measure concentrations of up to 30% hydrogen (NUREG-0737 only requires measurement up to 10% hydrogen). The readout and recorder-indicator devices are located in the EFIC room, with phone communications to the main control room.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Operating procedure complete, licensee is preparing emergency procedure for system actuation.

Training:

Complete.

Technical Specifications:

The licensee submitted proposed technical specifications in June 1983, which will be issued in a license amendment in April 1984.

NRC Review Status:

A Safety Evaluation was issued by the NRC on January 18, 1984 concluding that the requirements of NUREG-0737 Item II.F.1.6 have been met. A post-implementation review is planned.

Davis Besse, Unit No. 1

Description

The CHMS indicator and recorder are separate devices which have a range of 0-10% of Hydrogen concentration.These instruments replaced 0-5% range instruments previously installed. The CHMS indicator has a system accuracy of 5.4% of full scale. The recorders are computer data points on disk files which can be displayed on the CRTs in the control room and the Technical Support Center. The displays would be running approximately one minute behind real time accident conditions. The CHMS recording has a system accuracy of 5.0% of full scale.

Design/Installation

The CHMS is complete.

Operability Tests

Testing was completed prior to declaring the system operational.

Procedures

Complete.

Training

Complete.

Technical Specifications

Submitted.

NRC Review Status

The NRC review is complete and a Safety Evaluation Report was issued by letter dated May 24, 1983. The staff will review the proposed technical specification issue with the review of the licensee's response to Generic Letter 83-37.

Oconee Units 1, 2 and 3

Description:

The licensee has installed five hydrogen sample ports in containment for each of two redundant hydrogen monitors. The hydrogen monitors are made by Comsip with a range of 0-10%. They are of the thermal conductivity type. Indicator and recorder readouts are provided in the control room. The plant computer also receives the readings.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Proposed tech specs will be submitted April 2, 1984.

NRC Review Status:

NRC has approved the system at ONS on September 1, 1983. A post implementation review will be performed later.

Rancho Seco

Description:

Two trains of hydrogen monitoring equipment with a range of 0 to 10% hydrogen concentration are installed at Rancho Seco. The system alarms at 3% hydrogen concentration and can be read out in the control room on a trend recorder and the SPDS. The trend recorder will be replaced by the IDADS when it is operational.

Design/Installation:

Complete.

Operability Tests:

Complete.

Training

Complete.

Technical Specifications:

The licensee submitted proposed technical specifications by letter dated January 23, 1984.

NRC Review Status:

NRC review was completed and the SER was issued by letter dated July 28, 1982.

Three Mile Island Unit 1

Description:

GPU Nuclear has installed safety-grade containment hydrogen monitors, with a range of 0-10% hydrogen concentration, that sample the dome of the reactor building and readout in the control room. Containment hydrogen concentration is also input to the plant computer. After installation, the vendor submitted a 10 CFR 21 Notification that the analyzer catalyst had would not survive a loss-of-coolant-accident environment. Thus, it became necessary for GPU Nuclear to replace the catalyst bed. The system was designed to meet the requirements of NUREG-0737.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Request submitted and under review by NRR.

NRC Review Status:

In progress.

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

NUREG-0737 Position

Licensees were to propose additional instruments to supplement existing instrumentation to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). Basically, the final instrumentation fell into three categories:

(a) Instrumentation for determining the margin to saturation conditions for the given situation;

(b) Thermocouples located at the core exits 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.

(c) An unambiguous indication of inadequate core cooling to include tracking of reactor vessel water inventory indication.

During the Commission review of ICC instrumentation requirements (SECY-82-407), it was concluded that Dp measurement systems for B&W reactors should have a minimum range extending from the top of the candy cane to the low point in the hot leg and from the vessel upper head to the bottom of the hot leg.

An order was issued in December 1982 which directed the B&W licensees to propose a system for reactor vessel inventory tracking. Dates for specific submittals were included in the order. Plant specific reviews by the staff of the inventory tracking system conceptual designs are complete.

The following describes how each B&W licensee met these guidelines:

ANO-1

Description:

The subcooling margin monitor includes two channels of measurement and indication. Each channel receives a pressure signal from a different reactor coolant loop. The pressure input is a safety-grade wide-range (0-2500 psig) signal. Each channel calculator also receives two wide-range (120-920^{o}F) RTD temperature inputs, one from each reactor coolant hot leg loop. The channel calculator selects the highest temperature input. The margin to saturation is recorded in the control room. Backup capability exists in the form of other pressure and temperature indications and steam tables provided in the control room. In addition to the sub-cooling margin monitor, the instrumentation will consist of the hot leg inventory measurement, reactor vessel head inventory monitor and in-core monitors; each of these systems will use the gamma thermometer as the sensing device. The hot leg inventory measurement will use a vertical run of small diameter pipe, the upper end connected to the existing high point vent and the lower end connected to the cold leg drain line.

The reactor vessel head inventory monitor will use two gamma thermometer probes, each covering the upper head and plenum regions of the reactor vessel extending down to about one foot above the top of the fuel. Each probe will contain multiple sensors distributed axially along its length. The SPDS computer will be used to display the inventory information. The in-core monitoring is currently done by 32 core-exit thermocouples feeding into the plant computer. Each output can be trended independently or within a group. The upgraded system will use four probes of gamma thermometers, one for each quadrant, each containing eight axially distributed sensors.

The licensee has stated that the installation of the Inadequate Core Cooling instrumentation is scheduled for the refueling outage during March-June 1986.

Design/Installation:

Design and procurement: March 1, 1985.

Installation: June 1986.

Operability Tests:

The licensee has contracted with TEC to perform the initial tests on the gamma thermometers, with completion expected in April 1984. The final operational tests are expected in March-June 1986.

Procedures:

Existing procedures cover subcooling margin and use of core exit thermocouples. Other procedures to cover the upgrades will be completed by June 1986.

Training:

The current training program covers the subcooling margin monitor and the core exit thermocouples. Other training to cover upgrades will be completed by June 1986.

Technical Specifications:

Technical specifications have been implemented on the subcooling monitor. Other technical specification changes to cover modifications are due in March-June 1986.

NRC Review Status:

NRC has reviewed the AP&L conceptual designs using the gamma thermometers. The pre-implementation review will continue as test data and other information becomes available. The review is expected to be a continuous process, concluding with the post-implementation review after installation.

Crystal River Unit 3

Description:

Crystal River Unit 3 presently has two subcooling margin monitors (SMM's) and 52 core exit thermocouples (CET's) which can be used to detect inadequate core cooling. Both SMM's and their 12 associated CET's are powered from Class 1E power supplies. The licensee has committed to installing an instrumentation system which fully meets NUREG-0737 Item II.F.2 during the next scheduled refueling outage. The proposed system would: upgrade the core exit thermocouples to provide at least four qualified CET's in each core quadrant; add differential pressure instrumentation to monitor coolant inventory in the full height of the two hot legs and in the reactor vessel head area when reactor coolant pumps are not operating; and add reactor coolant pump power and inlet temperature instrumentation to track coolant inventory when the pumps are running. This combination of instrumentation is similar to that proposed for other B&W plants.

Design/Installation:

Design of the instrumentation to be added or upgraded is approximately 50% complete. Installation is scheduled for completion by August 1985.

Operability Tests:

August 1985.

Procedures:

August 1985.

Training:

August 1985.

Technical Specifications:

To be submitted by licensee by March 1985.

NRC Review Status:

The licensee submitted the required design information for the proposed system in July 1983. The NRC has completed review of the conceptual design and had advised the licensee to proceed with final design and procurement by letter dated September 6, 1983. Additional information and design details have been provided by the licensee at the staff's request and are being reviewed. Following installation of the new system, the NRC will perform a post-implementation review as discussed in NUREG-0737.

Davis Besse, Unit No. 1

Description

The Davis Besse Instrumentation for Detection of Inadequate Core Cooling will consist of sub-cooling monitors, core exit thermocouples, and a reactor inventory tracking system. The sub-cooling monitors consist of a two channel Tsat monitor system which is Class 1E. Normal input into the Tsat system is the RCS hot leg temperature. In place of the RCS hot leg temperature, the Tsat system can select input from the incore thermocouples. The pressure is monitored from the input to the Safety Feature Actuation System. The input from the Tsat system and pressure is displayed on Tsat/Psat meters in the control room and recorded in the Technical Support Center computers.

The plant currently has installed 52 control grade core exit thermocouples. The core exit thermocouple temperature indication system is designed using 8 of the existing 52 incore thermocouples per channel (a total of 16 thermocouples). The signal cables, once outside the containment, have been modified to make this part of the Class 1E system. Seismic qualification of the terminal connector located inside containment appear to be a pacing schedular problem.

The licensee has proposed a reactor coolant inventory tracking system consisting of a hot leg level measurement system (HLLMS), reactor coolant pump monitoring program (RCPMP), and reactor vessel head to hot leg vent line. The HLLMS will have one differential pressure transmitter per reactor coolant hot leg. The HLLMS will monitor hot leg water inventory when the reactor coolant pumps are not running.

The licensee proposes to install a continuous vent line extending from the top of the RV head to an existing penetrating near the top of the steam generator on the hot leg loop to which the pressurizer is attached. Steam and noncondensable gases will thus be vented from the RV head to a high point in the system, and then vented from the candy cane through an existing vent valve at the top of the candy cane. Therefore, a hot leg inventory tracking system will also indicate core level. A thermocouple will be installed in the vent line to monitor the coolant temperature so that the operators can maintain subcooling in the vent line during a controlled depressurization and cooldown. When the reactor coolant pumps are running, the RCPMP will utilize existing plant computer data points for reactor coolant system pressure and temperature and reactor coolant pump motor power as input parameters to a computer program, which will calculate the void fractions that each pump sees in the RCS.

Design/Installation

The safety grade sub-cooling margin monitors are operating. The licensee will need to upgrade the core exit thermocouple to safety grade. A study is underway for final qualification of the core exit thermocouples. The licensee has installed the software for RCPMP in the 1983 refueling outage, and has committed to install the HLLMS in the following refueling outage, and RV head to hot leg vent line during the two refueling outages following the 1983 outage.

Operability Tests

The acceptability of the proposed vent design will be confirmed with experimental data from B&W Integral System Test Program. The inventory tracking system will require functional test and calibration prior to implementation.

Procedures

Procedures are complete for the sub-cooling margin monitors. Procedures for other systems are undergoing development.

Training

Training for the sub-cooling margin monitors are complete. Other training is pending completion the development program.

Technical Specifications

Technical Specifications for the sub-cooling monitors are in place. Proposed technical specifications for other systems have not been considered yet.

NRC Review Status

The NRC staff provided an evaluation of the proposed Davis Besse systems and requested additional information by letter dated September 6, 1983. By letter dated October 7, 1983, the licensee provided a schedule for submitting the requested information that extended into mid-March 1984.

Oconee Units 1, 2 and 3

Description:

The subcooling margin monitor installed at Oconee consists of a computer processed, continuously displayed system which receives buffered inputs from the monitored process variables. The licensee has stated that the system meets II.F.2 requirements although parts of the system beyond the inputs are not safety grade. Three separate subcooling margins are displayed in the control room, one for within the reactor vessel and one for each hot leg. However, redundant SMM displays are required by NUREG-0737 Item II.F.2. In case of computer failure, monitoring of saturation conditions is based on control room readouts of primary system instrumentation and the use of available steam tables. The current core exit thermocouples provide 52 temperature inputs to the operational aid computer. These are not safety grade or environmentally qualified. The upgrade program would make 24 of them safety grade and environmentally qualified. In the upgraded system, the primary display would be with the operational aid computer and the backup display would only contain the 24 safety grade readings. These would be divided into two trains with 3 readings per quadrant per train.

The full range inventory monitor would use a Dp cell reading between the high point vent and the decay heat removal drop line. The reactor vessel inventory monitor would use the reading between the control rod drive mechanism and the decay heat drop line. The licensee has stated that the instrument is likely to be of the Westinghouse variety, although the decision on the vendor has not yet been made.

The schedule for completion of the upgrades is being finalized and will then be transmitted to NRC. Tentatively, the schedule for the three units is likely to be as follows:

Unit 1: Completion by mid-1986.

Unit 2: Completion by late-1986.

Unit 3: Completion by late-1985 to subsequent refueling outage, depending on vendor supply timing.

Design/Installation:

Consistent with above schedule.

Procedures:

The timing of the procedures will be consistent with the earliest unit to achieve completion per the above schedule.

Training:

The timing of training will be consistent with the earliest unit to achieve completion per the above schedule.

NRC Review Status:

NRC SER was issued in November 1983; this corresponds to the pre-implementation review. Post-implementation review will be performed after installation.

Rancho Seco

Description:

The Rancho Seco instrumentation for detection of inadequate core cooling will consist of subcooling margin monitors, core exit thermocouples and a reactor inventory tracking system. There are currently installed and operational two types of monitors for providing subcooling margin information. One monitor is a redundant fully qualified micro-processor based hardware system with safety grade sensor input. The other monitor is a software program within the control grade plant digital process computer.

The plant currently has installed 52 control grade core exit thermocouples. At the next refueling outage (scheduled for November 1984) the licensee plans to replace 16 of these thermocouples with 16 fully qualified thermocouples. The licensee indicated that seismic qualification of the connectors may be a problem.

The licensee stated in their April 15, 1983 submittal that their preferred reactor inventory tracking system will consist of a Heated Junction Thermocouple System (HJTCS) or a hot leg differential pressure system. In the HJTCS a probe installed in the center CRD position which will detect reactor vessel inventory between one foot above the core to the top of the reactor vessel. For the hot leg differential pressure system, the differential pressure is measured between a connection to the high point vents in the hot leg and the decay heat line. This will measure water inventory between the top and the bottom of the hot leg. The licensee subsequently decided to install the hot leg differential pressure portion of the reactor coolant inventory tracking system but is still investigating alternate means for providing reactor vessel inventory.

Design/Installation:

The subcooled margin monitors are installed. The safety grade core outlet thermocouples will be installed the next refueling outage (schedule for November 1984). The hot leg differential pressure portion of the reactor coolant inventory system will be installed the next refueling outage (schedule for November 1984). Final design of the selected reactor vessel head inventory portion of the system will be submitted to the NRC by June 30, 1984. Installation of this portion of the system is scheduled to be completed during the 1986 refueling outage.

Operability Tests:

The subcooled margin monitors are operational.

Procedures:

Complete for the subcool margin monitors.

Training:

Complete for the subcool margin meters.

Technical Specifications:

No proposed Technical Specifications have been submitted.

NRC Review Status

NRC review on this item is in progress.

Three Mile Island Unit 1

Description

The TMI-1 Instrumentation for detection of inadequate core cooling when complete, will consist of subcooling margin monitors, core exit thermocouples, and a reactor coolant inventory trending system and wide range coolant pressure, pressurizer pressure and level. Full hot leg differential pressure coverage is provided with the upper tap on the top of the hot leg and the lower tap on the decay heat removal drop line. Full reactor vessel coverage is provided with the upper tap on the top of the vessel and the lower tap using the same decay heat removal drop line tap. Portions of the system are presently installed. The remainder of the system will be installed at the first refueling outage after restart. The reactor coolant inventory trending subsystem will operate in two modes. If the reactor coolant pumps are running, the trending subsystem will measure void fraction from a pump power algorithm. These measurements will be performed every six minutes (pre-reactor trip) or every 30 seconds (post-reactor trip). If the reactor coolant pumps are not running, the subsystem will measure differential pressure to determine hot leg inventory and also reactor vessel inventory. The differential pressure transmitter signals will be temperature compensated. At the time of restart, the redundant saturation margin monitors will be safety-grade except for control grade input signals, and non-seismically qualified digital indicators. Additionally, the existing backup incore thermocouple display system will meet the requirements of Item II.F.2 except for seismic and environmental qualification. Full conformance with Item II.F.2 will be achieved for the monitors and the display system at the first refueling following restart which is also when the trending subsystem will be installed.

Design/Installation:

Saturation Margin Monitor and Backup Incore Thermocouple Display: Complete for restart, upgrade to safety-grade at first refueling following restart.

Reactor Coolant Inventory Trending Subsystem: Design complete April 1984, decay heat dropline tap complete, remaining construction complete Fall 1985 (estimated refueling outage date if the plant is allowed to restart in July 1984).

Operability Tests:

Complete for the restart items noted; remaining testing upon completion of installation (outage required).

Procedures:

Complete for the restart items noted; remaining procedures July 1984.

Training:

Complete for the restart items noted; remainder in Fall 1985.

Technical Specifications:

Specifications incorporated for saturation margin monitors. Remaining submittals pending.

NRC Review Status:

Review complete for the restart items noted except for question on seismic capability of backup incore thermocouple display system. That review will be complete by restart. Conceptual design of trending subsystem has also been approved. Final design package scheduled for June 1984 submittal.

III.A.1.2 UPGRADE EMERGENCY SUPPORT FACILITIES

NUREG-0737 Supplement 1 Position

Licensees will establish three emergency response facilities, the Operational Support Center (OSC), the Technical Support Center (TSC), and Emergency Operations Facility (EOF).

The OSC will be the onsite area separate from the control room where predesignated operations support personnel will assemble. A predesignated licensee official shall be responsible for coordinating and assigning the personnel to tasks designated by control room, TSC and EOF personnel.

The TSC is the onsite technical support center for emergency response. It is staffed by predesignated technical, engineering, senior management, other licensee personnel, and five pre-designated NRC personnel. The TSC will perform EOF functions for the Alert Emergency class, for the the Site Area Emergency class and General Emergency class until the EOF is functional.

The EOF provides for management of overall licensee emergency response, coordination of radiological and environmental assessment, development of recommendations for public protective actions, and coordination of emergency response activities with Federal, State and local agencies. The EOF shall be a licensee controlled and operated facility staffed by pre-designated emergency personnel. Facilities shall be provided in the EOF for the acquisition, display and evaluation of radiological and meteorological data and containment conditions necessary to determine protective measures.

Implementation of Regulatory Guide 1.97 will affect major data inputs to the emergency support facilities. However, these facilities are to be functional as soon as possible and upgraded in accordance with a projected schedule for Regulatory Guide 1.97.

The following describes how each licensee is meeting these guidelines:

ANO-1

Description:

All emergency response facilities for ANO-1 are completed and functional with respect to the physical facilities and most of the data handling capabilities. The ANO Administration building serves as the Operational Support Center. The Technical Support Center (TSC) is located on the third floor of the ANO Administration building, which is located within the plant protected area and adjacent to the turbine building. The Emergency Operations Facility (EOF) is located 0.65 miles from the plant. The EOF is divided into two distinct functional areas, emergency response and media. The emergency response portion was designed to the same radiological habitability requirements as the control rooms, considering its location with respect to the plant. The EOF has both HEPA and charcoal filtration available. The licensee has stated that the EOF does not have to meet the protection factor requirement because construction was started prior to February 18, 1981. The licensee has proposed use of the AP&L Russellville Local Office, approximately seven miles from the site, as the backup EOF. This request is under consideration by the Commission.

At present, the SPDS terminal and the Gaseous Effluent Radiation Monitoring System (GERMS) terminal are available in the TSC. However, the GERMS is not fully operational yet. The EOF has the same capabilities as the TSC. Ultimately, GERMS will have the capability to acquire real time effluent release data, combine these with real time meteorological data (acquired from the ANO meteorological tower) and produce a graphic display of offsite dose assessment. Various primary and backup means of communications exist between the control room, TSC, OSC, NRC and state and local authorities. The computerized dose assessment portion of GERMS is scheduled for operation by June 30, 1984. The capabilities of the EOF are expected to be fully operational, including implementation of R.G. 1.97 provisions, by November 15, 1985.

Design/Installation:

Complete, with the exception of the R.G. 1.97 evaluation and the GERMS capability.

Operability Tests:

Complete, with the exception of the R.G. 1.97 evaluation and the GERMS capability. The available facilities have been employed in the last two emergency response exercises.

Procedures:

Complete with existing parameters.

Training:

Complete with existing parameters.

Technical Specifications:

None required.

NRC Review Status:

The EOF and backup EOF exemption requests are in process. The post implementation review will be performed by an audit team after all functions are installed.

Crystal River Unit 3

Description:

All emergency response facilities for Crystal River Unit 3 are completed and functional insofar as the physical facilities and most of the required data collection capabilities. The Operational Support Center (OSC) is a designated area adjacent to the main control room. The Technical Support Center (TSC) is located in the Administrative building which is near (less than 100 meters) the Crystal River Unit 3 Auxiliary

Building. The Emergency Operations Facility (EOF) is a new structure located 10.4 miles southeast of the Crystal River plant and on the southern outskirts of the town of Crystal River. The TSC and EOF have been provided with terminals for the plant RECALL data acquisition and storage system with appropriate display and printout devices. A new meteorological tower and associated equipment has been installed and provides readout of wind speed and direction in the main control room. However, this indication has not yet been provided in the TSC and EOF due to unanticipated delays in delivery of an Emergency Data Acquisition System (EDAS). This system is expected to be delivered and installed by June 1984. Future minor modifications to the Crystal River emergency response facilities may be necessary with the implementation of Regulatory Guide 1.97 provisions. However, no specific changes have been identified at this time. In addition, the SPDS will have readout capabilities in the TSC.

Design/Installation:

Complete, except for installation of the EDAS equipment. Scheduled for June 1984.

Operability Tests:

Complete, except for EDAS.

Procedures:

Complete, except for EDAS.

Training:

Complete, except for EDAS.

Technical Specifications:

None required.

NRC Review Status:

The licensee was advised by letter dated August 12, 1983 that the habitability and location of the EOF meets the NRC requirements and that overall acceptability of all emergency response facilities in support of Crystal River Unit 3 would be determined during a post-implementation appraisal at a future date. A schedule for this appraisal has not yet been established.

Davis Besse, Unit No. 1:

Description:

The Emergency Support Facilities for Davis Besse consists of (1) an Operations Support Center (OSC) for plant personnel to assemble for assignment to duties to support emergency operations, (2) an on site

Technical Support Center (TSC) for accident assessment, (3) an Emergency Control Center (ECC) for off site radiological assessment and coordination, (4) a Site Emergency Operation Center (SEOC) to provide the licensee an interface area for governmental organization, (5) a Toledo Edison Conference Room for corporate management personnel, (6) a Public Relations Office/Briefing Area for press gatherings and releases, (7) a Radiological Testing Laboratory for analyses of low level environmental samples, and (8) an Emergency Support Center (an approved backup emergency operation center) located 21 miles from the plant at the corporate offices in Toledo, Ohio. The OSC is located at the plant. The TSC, ECC, SEOC, Public Relations/Briefing Area, Toledo Edison Conference Room, and the Radiological Lab are located on the first floor of the Administration Building which is located at the edge of the site outside perimeter fencing. These areas are all designed to meet or exceed the habitability requirements of the control room. All facilities have a variety of communication systems including 4 dedicated telephone loops, lines through the Ohio Bell Telephone System, lines through the General Telephone System, and radio communication. The SPDS and Meteorological data system is operational in the TSC. The licensee considers this item of the NUREG-0737 requirements to be complete.

Design/Installation:

The licensee designed and constructed a dual purpose building to provide office space of plant administration purposes and the Emergency Support Facilities. The facilities are complete and operational.

Operability Tests:

The Emergency Support Facilities have been tested during the last two emergency drills. The Region III reports concerning these drills and the facilities have been favorable.

Procedures:

An emergency plan and procedures have been in place. The licensee considers the procedures to be completed.

Training:

Training has been completed.

NRC Review Status:

The NRC review is in progress.

Oconee Units 1, 2 and 3

Description:

The licensee has stated that all emergency response facilities at ONS are complete with the exception of incorporating the SPDS and the R.G. 1.97 evaluation.

The Operational Support Center (OSC) is located in a laboratory area close to the control room of Unit 3. The Technical Support Center (TSC) is located adjacent to the control room for Units 1 and 2. The TSC contains sufficient space to accomodate the necessary personnel. At present, it has capability to access about 50 preselected parameters from the plant computer. The selected parameters include items such as hot leg temperature, cold leg temperature, flux levels, pump status, as well as containment systems and environmental parameters. Eventually an SPDS terminal would be located in the TSC. Primary and backup communications links are also available in the TSC. These include the normal commercial lines as well as the interplant lines, microwave and radio communication with the Emergency Operations Facilities (EOF). Transmission of plant data to off-site locations, including the EOF, requires manual input to the extent of pulling data off the plant computer, loading it on to floppy disc every 15 minutes and performing a verification and validation of the data. It is then available to the EOF and other off-site users such as the NRC and the vendor. There is no SPDS or other real time access to plant parameter data at the EOF. The licensee has designated the Training Center, approximately 0.5 miles from the 3-unit plant, as the current EOF. The licensee considers this to be an interim facility while an exemption request is under consideration by NRC asking for approval of locating the EOF in Charlotte, N.C., approximately 125 air miles from Oconee. The interim EOF does not meet habitability requirements in case of an accident.

Design/Installation:

Finalization of the design of the emergency response facilities is awaiting disposition of the exemption request on the location of the EOF.

Operability Tests:

Complete with exception of SPDS and R.G. 1.97 provisions. The facilities have been used in emergency drills.

Procedures:

Complete with exception of SPDS and R.G. 1.97 provisions.

Training:

Complete with exception of SPDS and R.G. 1.97 provisions.

NRC Review Status:

The exemption request on the location of the EOF under review by the Commission. The post-implementation review will be performed by an audit team after all functions are installed.

Rancho Seco

Description:

The Emergency Response Facility (ERF) for Rancho Seco consist of (1) the Operations Support Center (OSC) which are designated assembly areas in either warehouse "A" or the conference room in the administration depending on wind direction; (2) an onsite Technical Support Center (TSC) located adjacent to the control room; (3) an offsite Emergency Operations Facility (EOF) located about 23 miles from the Rancho Seco site and adjacent to the licensees general offices in Sacramento; and (4) an Emergency News Center (ENC) located at the licensees training and storage building in Sacramento. All the ERF facilities are installed and are available for interim use. The facilities were exercised during the recently completed annual emergency drill.

All facilities have a variety of communications systems including dedicated telephone loops. A computer based Interim Data Aquisition and Display System (IDADS) is being installed in the TSC and EOC, and IDAD displays are being provided in both OSC locations.

Design/Installation:

The physical ERF facilities for the Emergency Response Facilities are complete except for the heating and ventilating system for the TSC. The heating and ventilating system will be completed by the end of the next refueling outage. Hardware for the IDADS system is installed but the system is not operational.

Operability Tests:

The Emergency Support Facilities were exercised during the recently completed annual emergency drill.

Procedures:

The emergency plan and procedure are in place.

Training:

Training has been completed.

NRC Review Status:

The NRC review is in progress.

Three Mile Island Unit 1

Description:

The TMI-1 emergency response facilities consist of the Technical Support Center (TSC), the Operations Support Center (OSC) and the Emergency Operations Facility (EOF). The TSC is located below the control room on the 322' elevation of the control building, near the remote shutdown panel. The OSC is located in the area of the Health Physics Access Control Point on the 306' elevation of the central building. The OSC provides an area for personnel to meet for duty assignments in support of emergency operations. Accordingly, emergency equipment and supplies located in this area include dosimeters, various survey meters, protection clothing, respirators and/or masks, air samplers, portable lanterns and/or flashlights, a copy of the Implementing Document, decontamination chemicals, self-contained breathing apparatus, and resuscitator and first aid kits.

The EOF is in a new building located outside the ten mile radius of Three Mile Island. This building also houses the Three Mile Island Environmental Center. The EOF is common to TMI-1 and TMI-2. The EOF is equipped with a variety of communications systems including a comprehensive network of dedicated telephone loops and computer facilities capable of providing plant parameter information. Additional plant information will be provided in the TSC and EOF as the safety parameter display system (SPDS) and Regulatory Guide 1.97 work is completed. All emergency response facilities are complete and fully functional and have been used for several emergency exercises.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete. The most recent exercise was a full-scale exercise conducted in November 1983.

Technical Specifications:

Not applicable.

NRC Review Status:

Habitability and location of the EOF have been found to meet the requirements of NUREG-0737, Supplement 1. Overall emergency response facility acceptability to be determined by post-implementation on-site appraisal.

III.A.2.2 METEORLOGICAL DATA

NUREG-0737 Position

Licensees have been provided with meteorlogical criteria to fulfill, in part, the standard that adequate methods, systems, and equipment for assessing on a real time basis and monitoring the meterological aspects of actual or potential offsite consequences of a radiological emergency condition exist. There are four essential elements that can be categorized into three functions: meterological measurements, diffusion assessment and incorporation into dose assessment, and communications. The assessment capabilities should provide inputs to decisions to determine the necessity for notification, sheltering, and evacuation of the general public. Strictly speaking, III.A.2.2 is only concerned with the meterological aspects of emergency response capabilities. But overall, the meterological upgrades are very closely associated with dose assessment. Therefore, this audit also included the licensees capabilities for dose assessment utilizing the data from the meterological upgrades. NRC reviews will be done in the future as part of an overall appraisal of emergency response facilities. The implementation of sections of Regulatory Guide 1.97 via the issuance of supplement 1 of NUREG-0737 may affect some of the detectors utilized for this upgrade. However, the implementation of Regulatory Guide 1.97 will be done on a separate schedule.

The following discusses how each licensee is meeting these guidelines and describes in some detail the dose assessment capabilities.

ANO-1

Description:

Currently, meteorological data are recorded in the control room and transmitted to the TSC verbally over telephone links. In future, an automatic system called Gaseous Effluent Radiation Monitoring System (GERMS) will provide the capability to make rapid and accurate offsite dose assessments. It is planned that GERMS will acquire real time effluent release data, combine that with real time meteorological data (acquired from the meteorological tower) and produce a graphic offsite dose assesmsent plot. GERMS terminals will be available in the TSC and EOF. It is scheduled to be operational by June 30, 1984.

Design/Installation:

The design is complete and installation is proceeding to meet the operability date of June 30, 1984.

Operability Tests:

Complete by June 30, 1984.

Procedures:

Complete by June 30, 1984.

Training:

Complete by June 30, 1984.

Technical Specifications:

None required.

NRC Review Status:

A post-implementation review will be performed along with an appraisal of all emergency response facilities.

Crystal River Unit 3

Description:

Installation of an automated system to collect and process meteorological data is nearly complete for Crystal River Unit 3. A central HP 1000 minicomputer, located in the Technical Support Center (TSC), will process inputs from two meteorological towers (an original tower plus a recently-installed new tower) and from all radiation monitors. Operator terminals are provided at the main control room, the TSC, and the Emergency Operations Facility (EOF). Parameters from the towers are also displayed on chart recorders in the main control room. The Emergency Data Acquisition System (EDAS) computer will not be delivered until April 1984 (the original delivery date was July 1983). Therefore, completion of this system, including installation, testing, and training, will not be completed until July 1984.

Design/Installation:

June 1984.

Operability Tests:

June 1984.

Procedures:

July 1984.

Training:

July 1984.

Technical Specifications:

None required.

NRC Review Status:

A post-implementation review will be performed along with an appraisal of all emergency response facilities.

Davis Besse, Unit No. 1

Description:

Davis Besse has a Class A Model installation which meets or exceeds all acceptance criteria of Reg. Guide 1.23. When the Noble Gas/Iodine effluent monitors software modifications are complete, the system will be capable of providing automatic calculations for plume measurements. Currently the data for making these calculations are called in by the shift supervisor. If the capability to obtain meteorological data is lost at the plant site, the data can be obtained from local weather stations, radio stations or airports. The meteorological data is provided in the control room and the Technical Support Center.

Design/Installation:

The meteorological data system has been operable since October 1, 1982. The capability for providing automatic calculations of plume measurements is scheduled for completion by March 31, 1984.

Operability Tests:

Testing was completed prior to declaring the system operational.

Procedures:

Procedures are complete, including procedures for calculation of plume measurements by hand held computer.

Training:

Training is complete.

Technical Specifications:

No proposed technical specifications have been identified yet.

NRC Review Status:

A post implementation NRC review will be performed along with an appraisal of all emergency response facilities.

Oconee Units 1, 2 and 3

Description:

The current capability provides data on wind speed, wind direction and high-low temperature difference from the meteorological tower to the control room. The data is not recorded in the computer, but is displayed on strip charts. With the computer system upgrades scheduled for the end of 1984, the meteorological data will be fed into the computer and the data handling will include trending capability. The Class A model to calculate offsite doses would be available for use in May 1984. However, the full capability of the system to combine the meteorological and radiological data would not be available until the end of 1984 due to the time necessary for computer upgrades. The backup to the meteorological tower is the data from the Greenville-Spartanburg airport.

Design/Installation:

End of 1984.

Operability Tests:

End of 1984.

Procedures:

End of 1984.

Training:

End of 1984.

NRC Review Status:

A post-implementation review will be performed along with an appraisal of all emergency response facilities.

Rancho Seco

Description:

The licensee stated that the currently installed meteorological measurement system meets the intent of Regulatory Guide 1.23. The data from the system is available in the control room but has to be manually entered into the computer in the Technial Support Center (TSC). However, the information will be available on the Integrated Data Aquisition and Display System (IDADS) in the TSC and EOF when the system is operational. The IDADS system is expected to be operational at the end of the next refueling outage. A class A model for calculating doses has been developed and is available for use.

Design/Installation:

The meteorological measurement system is completed. The IDADS system will be operational at the end of the next refueling outage (November 1984).

Operability Tests:

The meteorological measurement system is operational. The IDADS system will be completed and operational at the end of the next refueling outage.

Procedures:

The procedures for use of the Class A model to calculate doses are completed. The revised procedures to be utilized for the IDADS system will be completed prior to the end of the next refueling outage.

Training:

Training on the use of the Class A model is complete. Training on the IDADS will be complete prior to start-up from the next refueling outage.

Technical Specifications:

None required.

NRC Review Status:

A post implementation review will be performed along with an appraisal of all emergency response facilities.

Three Mile Island Unit 1

Description:

The TMI-1 emergency meteorological data program encompasses a number of systems. Upon declaration of an emergency, source term calculations and dose rate projections are performed on-site using a Tektronix 4054A computer program. Within one hour of the declaration, the TMI environmental center is manned and the responsibility for dose projections is shifted to the environmental center (which is in the same building as the emergency operations facility). From this point on, dose projections are performed using the MIDAS model. MIDAS, in addition to dose projections, can also provide integrated doses by sector, plume tracking, and other, specialized reports. The dose projections are supported by a real-time gamma-monitoring system with input from 16 fixed monitor locations at varying distances from the site and one of more mobile monitors. The system is sufficiently sensitive to measure dose in micro-REM.

With respect to meterological capabilities, TMI-1 forecast information is received from one of three services. Real-time data is collected from the TMI meteorological tower, the mobile laboratory meteorological tower, the national weather service, the Middletown Airport, and/or the Bureau of Air Quality. Compueter capabilities are provided by one of four systems.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Not required.

NRC Review Status:

A post implementation review will be performed along with an appraisal of all emergency response facilities.

III.D.3.4 CONTROL ROOM HABITABILITY

NUREG-0737 Position

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 (Criteria 19, "Control Room", of Appendix A, "General Design Criteria for Nuclear Power Plants", to 10 CFR Part 50).

The following describes how each B&W licensees met these guidelines:

ANO-1

Description:

The licensee conducted a control room habitability study and concluded that only two detectors had to be added to satisfy this requirement. The licensee has provided safety grade chlorine and toxic gas detectors. The system provides for automatic control room isolation of outside air on excessive radiation or toxic gas signals. Under isolation, the control room atmosphere is under positive pressure in the recirculation mode.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Tech spec change approved in May 1982.

NRC Review Status:

The post implementation review is completed and the NUREG-0737 item was closed out in October 1982.

Crystal River Unit 3

Description:

The licensee has installed toxic gas detectors for sulfur dioide, chlorine, and ammonia to protect control room operators from hazardous levels of these gases. Each gas is measured by two redundant detectors and an alarm condition on any one detector will sound an audible alarm in the control room and automatically isolate incoming air flow to the control room. Upgrading of the control room air intake isolation dampers and changes to control room air flow rates was also necessary to improve control room habitability in accordance with NUREG-0737, Item III.D.3.4.

Design/Installation:

Complete.

Operability Tests:

Complete.

Procedures:

Complete.

Training:

Complete.

Technical Specifications:

Submitted in June 1983 by licensee. Amendment is scheduled for issuance in April 1984.

NRC Review Status:

Review complete. Safety Evaluation issued February 17, 1982.

Davis Besse, Unit No. 1

Description:

The licensee has a control room habitability system as part of the original plant design which will provide safe habitability conditions within the control room under both normal and accident radiological and toxic gas conditions, including loss-of-coolant accidents. The occupancy of the control room can be maintained under accident conditions without personnel receiving radiation exposure in excess of 5 rem whole body or its equivalent to any part of the body, for the duration of the accident. The licensee provided a control room habitability study by letter dated May 28, 1981, which indicates that the control room personnel are protected from the effects of accidents involving all industrial, transportation, mining and military installations and from potential accidents involving hazardous materials on the site and being transported near the site. The licensee considers this item complete.

Design/Installation:

No modifications were required. The control room habitability systems were part of the original plant design.

Procedures:

All procedures related to this issue are complete, including a procedure to isolate the control room if formaldehyde odors are detected (a specific procedure identified as necessary because of this review).

Training:

Training is complete.

Technical Specifications:

There are currently Technical Specifications for chlorine. No additional proposed Technical Specifications have been identified.

NRC Review Status:

A Safety Evaluation Report was provided by letter dated June 30, 1982, which found that the control room habitability systems are acceptable. The Region has subsequently discovered a potential design deficiency with respect to the remote chlorine detectors and the licensee is considering modifying their design to correct this deficiency.

Oconee Units 1, 2 and 3

Description:

The ONS control rooms are located in the Auxiliary Building. Units 1 and 2 share a control room while Unit 3 has a separate control room. The Units 1 and 2 control room is served by two 100 percent capacity air handling units which also service associated cable and equipment rooms. The Unit 3 control room and associated cable and equipment rooms are served by six air handling units. Outside air is supplied to the control rooms after passing through a filter train consisting of pre-filters, 99.5% efficiency HEPA filters, 99% efficient charcoal filter beds and a centrifugal fan. The filter trains are inoperative normally and required operator action to start. Isolation dampers are not provided for either control room. No chlorine of toxic gas detectors are currently provided. The closest chlorine storage is 500 feet away from the Units 1 and 2 control room air intake.

The licensee had taken the position that control room habitability requirements were met at Oconee until tests in mid-1983 showed that requisite positive pressure could not be obtained in the Units 1 and 2 control room. The ONS position of full compliance has changed. The licensee is still in the engineering/design phase to determine necessary mechanical modifications necessary to obtain the required positive pressure. The licensee is also, tentatively, considering the installation of monitors. No schedule has been established yet for completing the design effort and the necessary modifications. No specific modifications to the Unit 3 control room have been identified, but some modifications may be necessary based on the review of Units 1 and 2.

Design/Installation:

Modification schedule to be determined for Units 1 and 2.

Operability Tests:

Complete for Unit 3. Subsequent to modifications for Units 1 and 2. May have to be revised for Unit 3.

Procedures:

Complete, subject to change due to modifications.

Training:

Complete, subject to change due to modifications.

Technical Specifications:

Technical specs in this area have not yet been identified.

NRC Review Status:

NRC staff has been informed in January 1984 of the modifications proposed by the licensee. Staff review is continuing.

Rancho Seco

Description:

The licensee, in a letter dated January 16, 1981, provided a comprehensive evaluation of the control room habitability for the Rancho Seco plant. The proposed modifications to the control room to meet the NUREG-0737 requirements included installing a new control room heating, ventilation and air conditioning (HVAC) train that is redundant to the existing HVAC train; and upgrading the existing HVAC train. Included in the modification are new radiation and chlorine monitors which automatically isolate the control room and initiate the emergency HVAC trains.

Design/Installation:

The control HVAC modifications are scheduled to be completed during the next refueling outage (scheduled for November 1984).

Operability Tests:

To be conducted when modifications are completed.

Procedures:

Will be modified prior to completion of the next refueling outage (schedule to start November 1984).

Training:

Training will be completed prior to start-up from the next refueling outage.

Technical Specifications:

Proposed Technical Specifications will be submitted by the licensee by April 1, 1984.

NRC Review Status:

NRR completed its review and the SER was issued by letter dated April 9, 1982.

Three Mile Island Unit 1

Description:

GPU Nuclear performed the required review and initially committed to (1) modify the control building ventilation system, principally by adding redundant, low leakage dampers, and (2) install a quick-acting toxic gas detection system. However, GPU Nuclear subsequently performed a probabilistic risk assessment (PRA) of on-site and off-site toxic gas releases which led it to conclude that no modifications were necessary. The NRC staff has advised GPU Nuclear that probabilistic arguments may be reasonable with respect to transportation sources of hazardous materials, but that the staff position regarding fixed storage facilities, especially on-site storage, is that design basis protection should be provided. Onsite hazards are chlorine and ammonium hydroxide. GPU Nuclear recently responded to the staff by providing an outline of its reevaluation program.

Design/Installation:

Fall 1985 (estimated refueling outage date if the plant is allowed to restart in July 1984).

Operability Tests:

Next refueling outage (Fall 1985).

Procedures:

Next refueling outage (Fall 1985).

Training:

Next refueling outage (Fall 1985).

Technical Specifications:

To be determined.

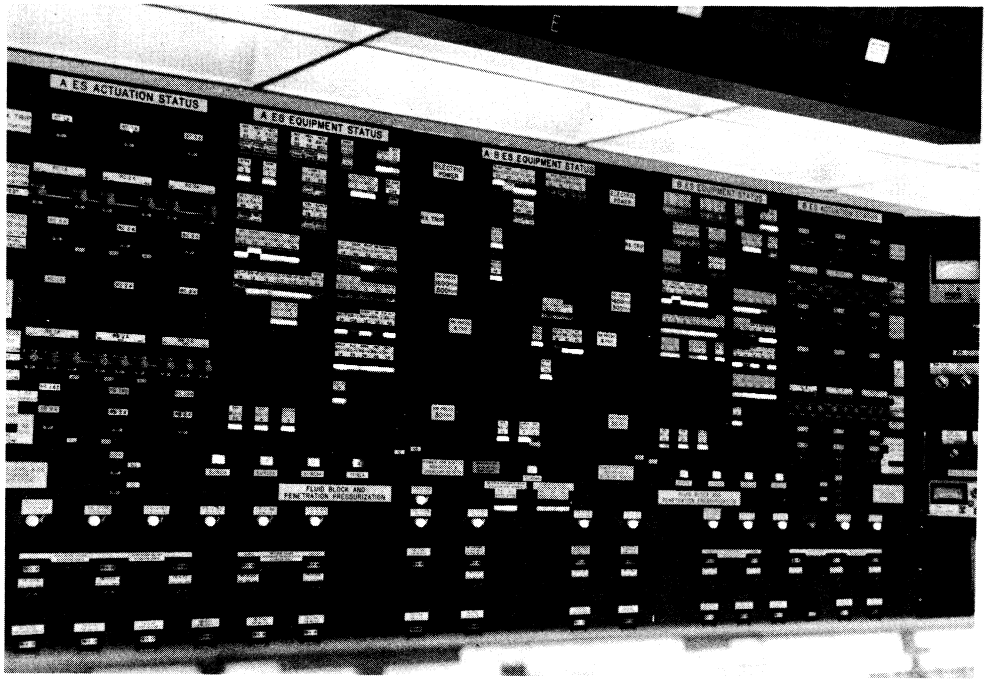
NRC Review Status:

The staff review is in progress.

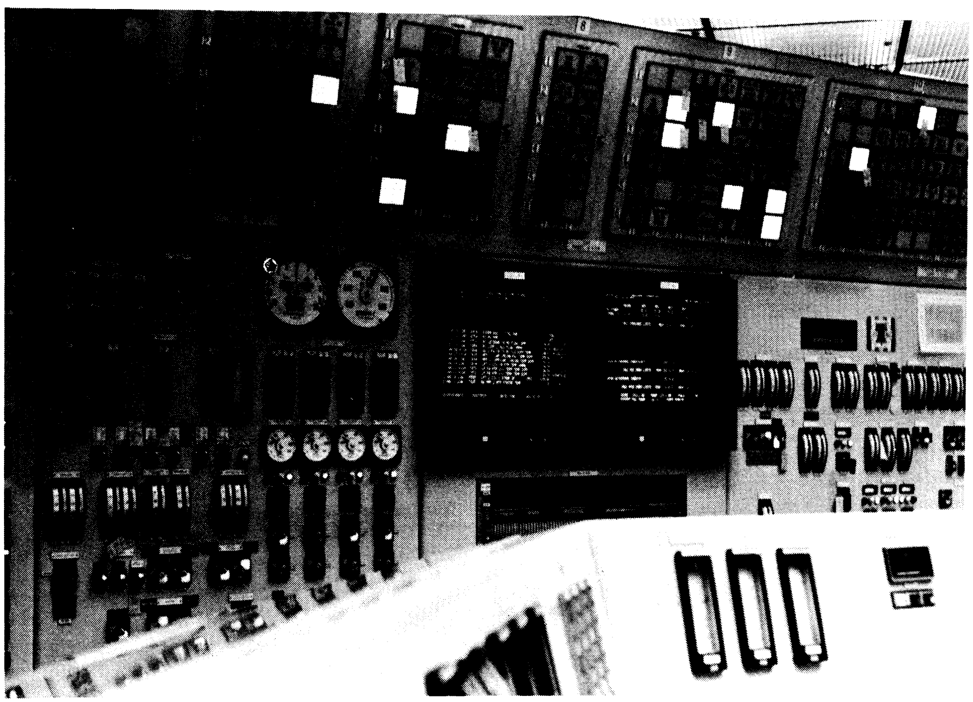
APPENDIX B

Photographs of Selected Installations

This appendix contains photographs of how licensees complied with selected NUREG-0737 requirements at the various B&W units.

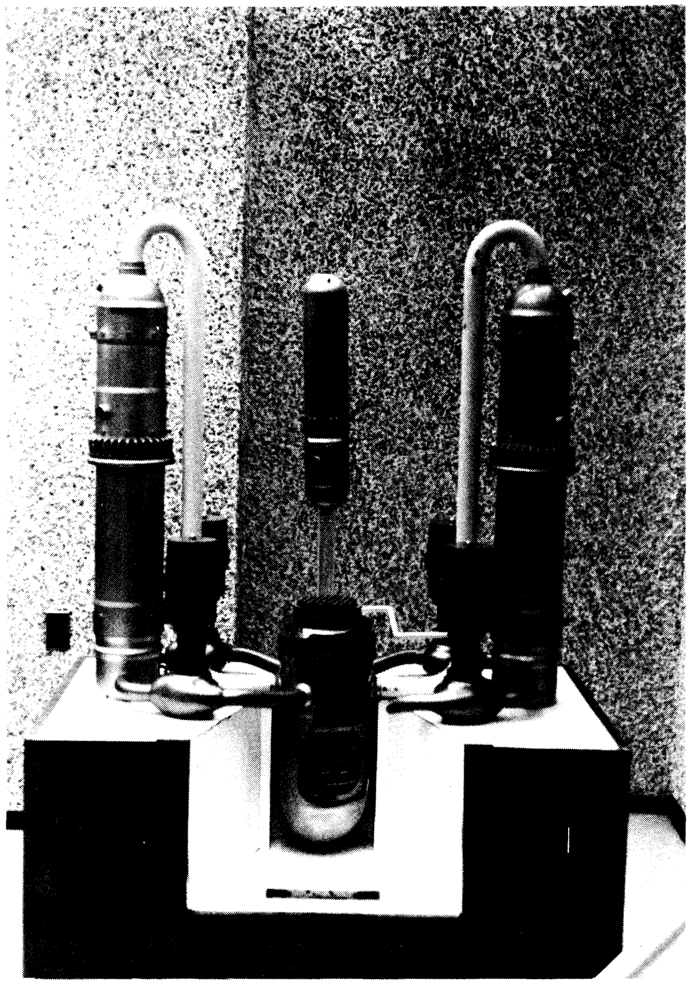


TMI-1 Control Room Panel after Human Factors Modifications are complete.



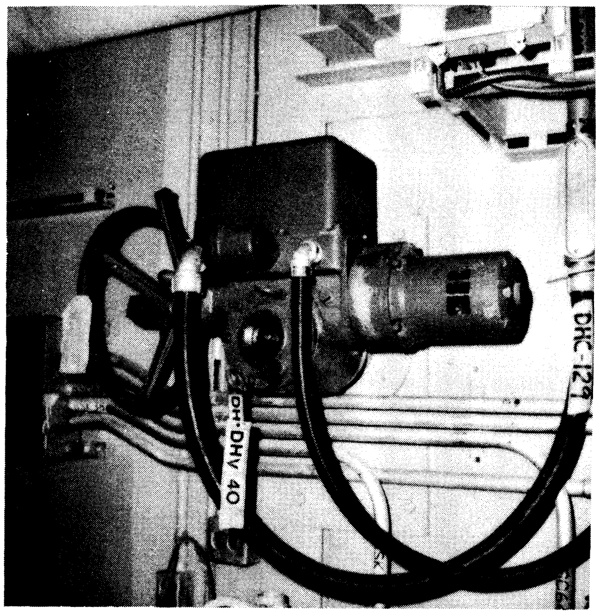
Davis Besse control room before human factors study is complete.

(I.D.1)

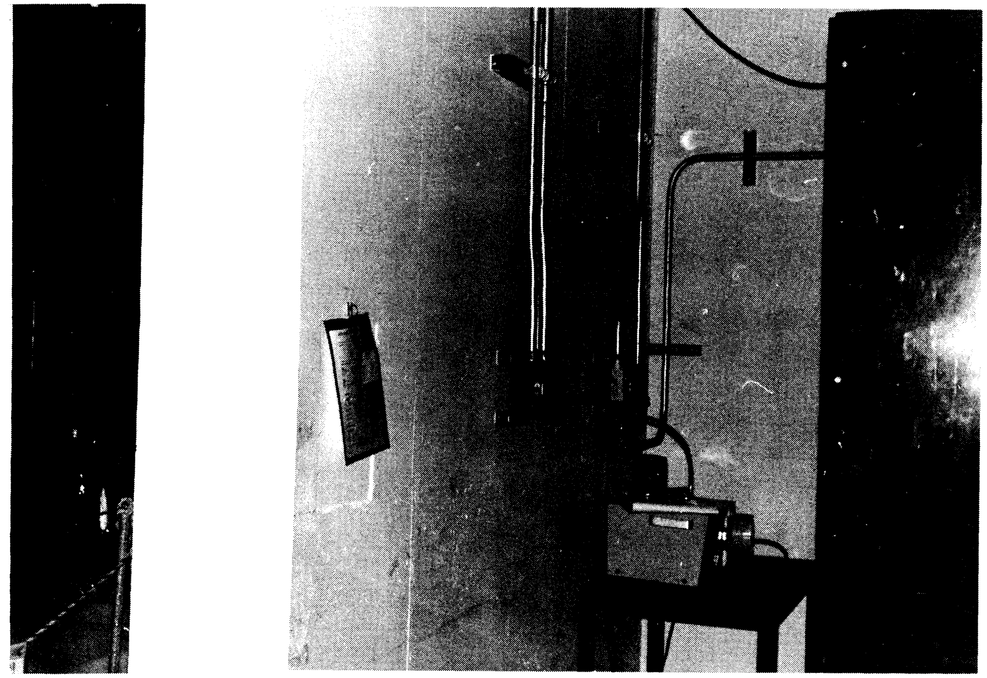


This is a model of Davis-Besse. On the other operating B&W plants, the steam generators are somewhat lower than shown here (about as low as the bottom of the reactor vessel). High point vents are installed on the hot leg of each loop and the pressurizer at all B&W plants. ANO-1, Oconee, and TMI-1 have installed reactor head vents. Rancho Seco and Crystal River claim the system design makes reactor head vents unnecessary. Davis Besse plans to install a continuous vent from the reactor head to a hand hole at the top of the steam generator on the right.

(II.B.1)



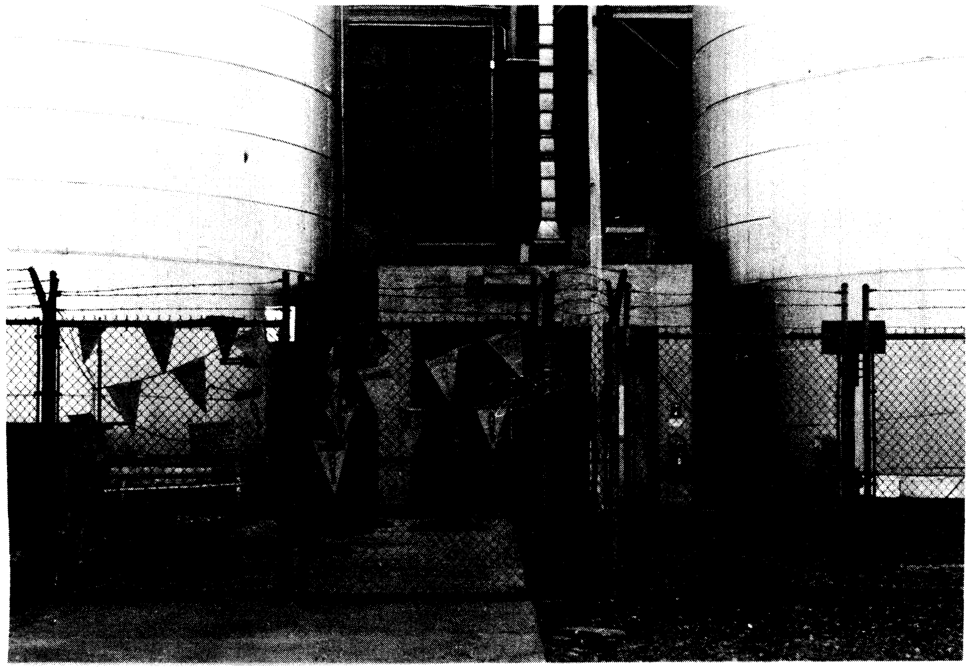
Crystal River Remote Operator for
DHR System



TMI-1 Shield Wall

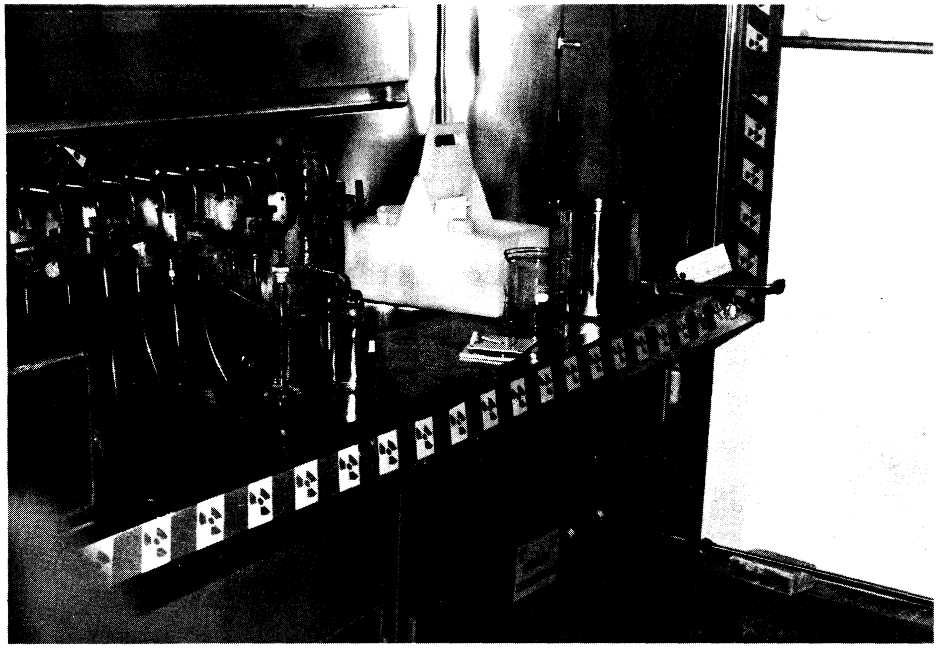
Typical physical modifications installed as a result of the II.B.2 Plant Shielding review. The remote operator shown for Crystal River actually operates a valve about ten feet on the other side of the wall where central components of the decay heat removal system are located. The shield wall installed at TMI-1 separates electrical breakers on the right from radioactive systems on the left. Under accident conditions an operator can go to the breaker panels without being over exposed.

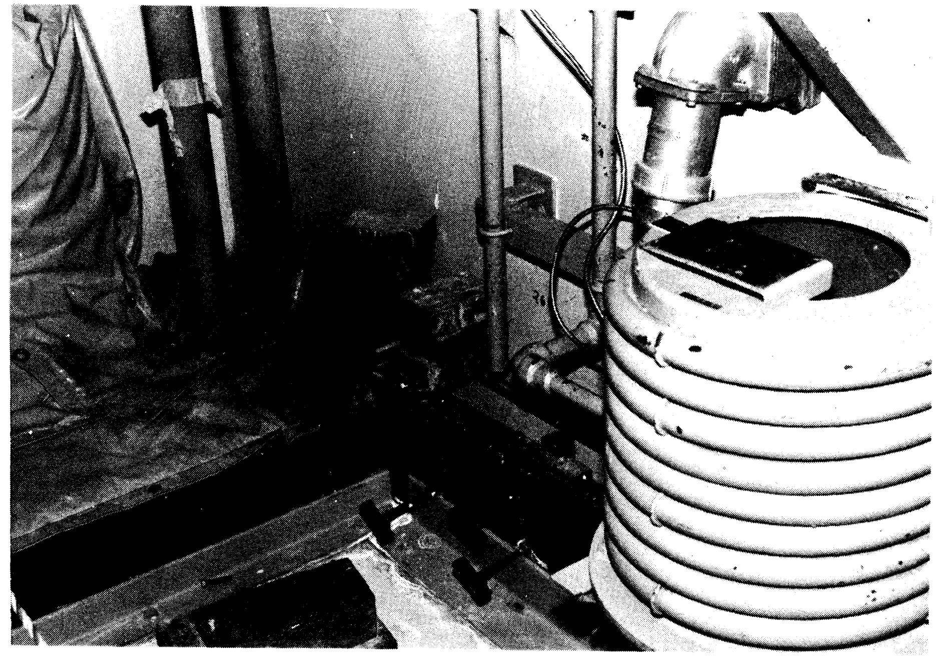
(II.B.2)

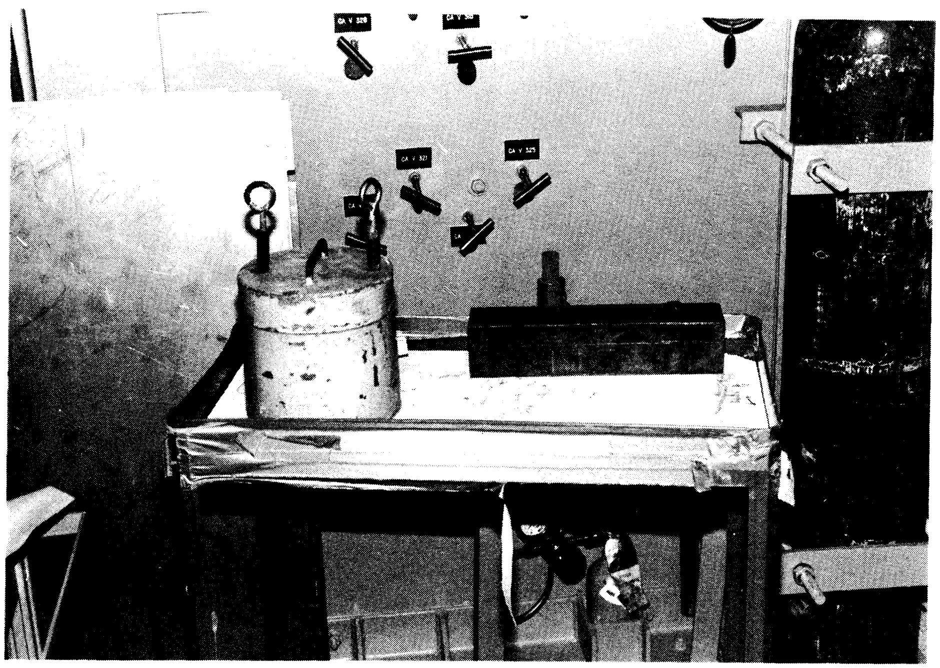


Building erected at ANO to house their normal and post accident sampling systems.

(II.B.3)







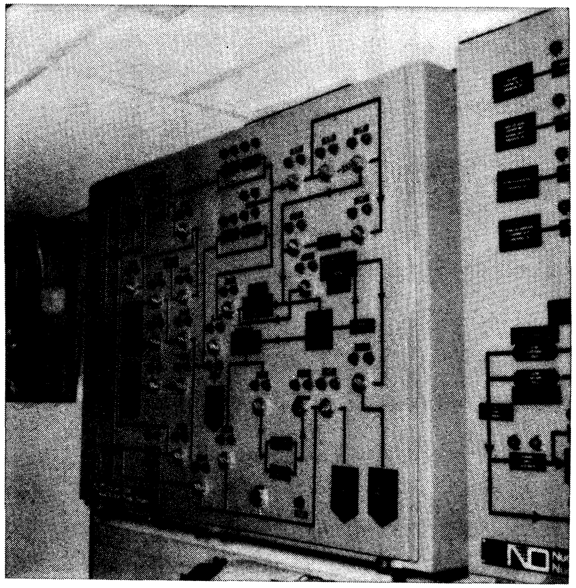
TMI-1 Post Accident Sampling System

Upper right - rerouting and shielding of normal sample lines.

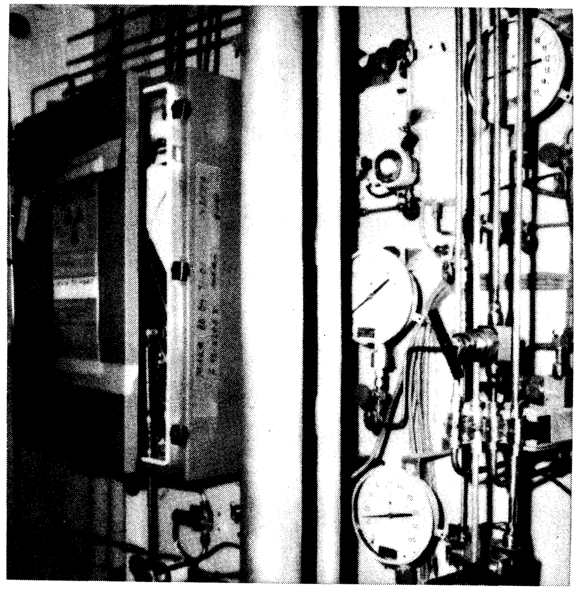
Upper left - normal and post accident sample sink.

Lower left - lead pigs for transporting liquid and gaseous samples.

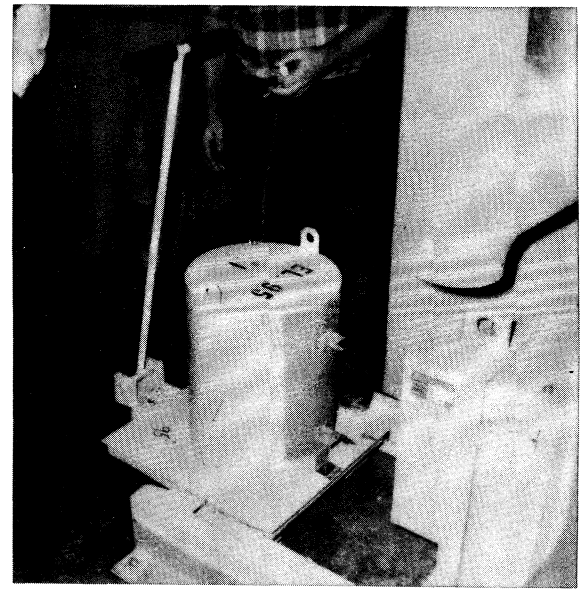
(II.B.3)



Pass Remote Controls



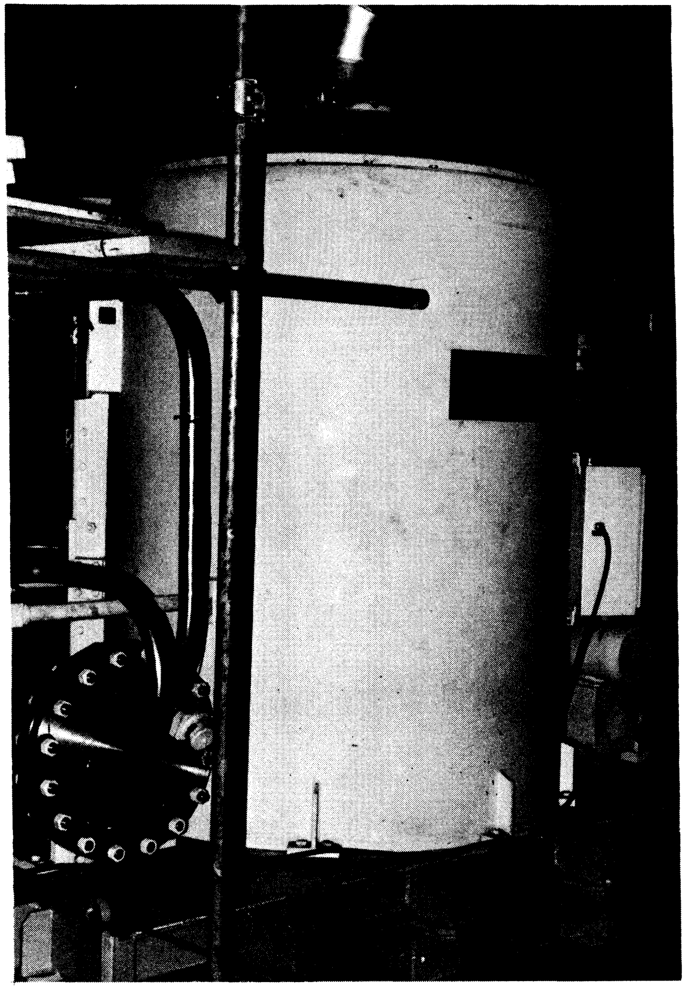
Actual Pass Sample Location
Located Behind Shielded Walls

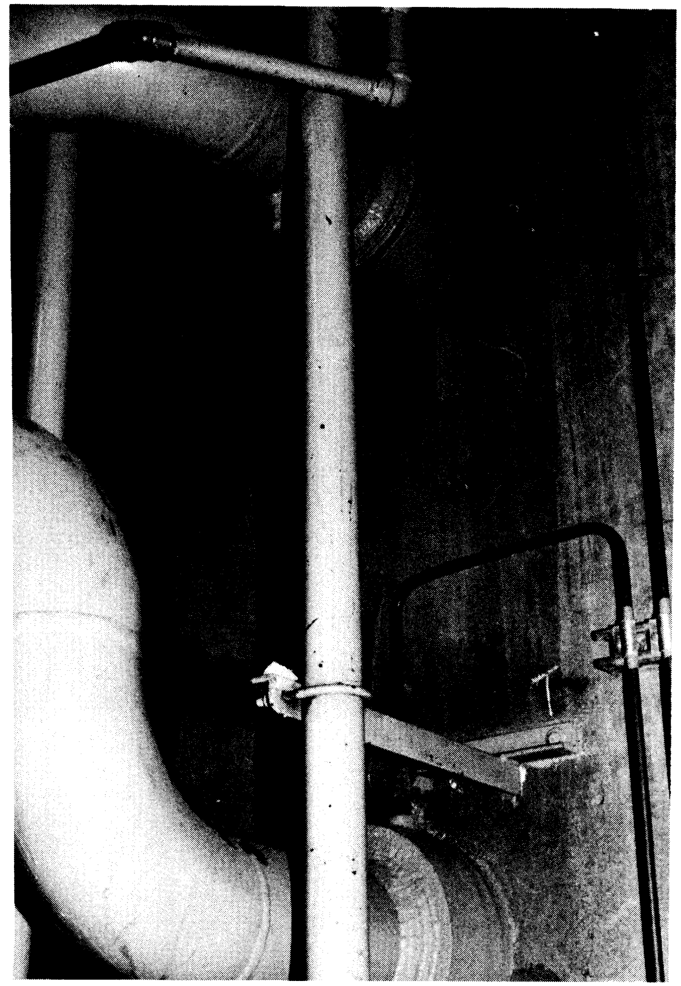


Lead Pig used for grab sampling of liquid sample. It is shown on its dolly and the two tubes on the right are where the pig automatically mates with the sampling system when pushed into place.

Liquid post accident sampling system found at Crystal River.

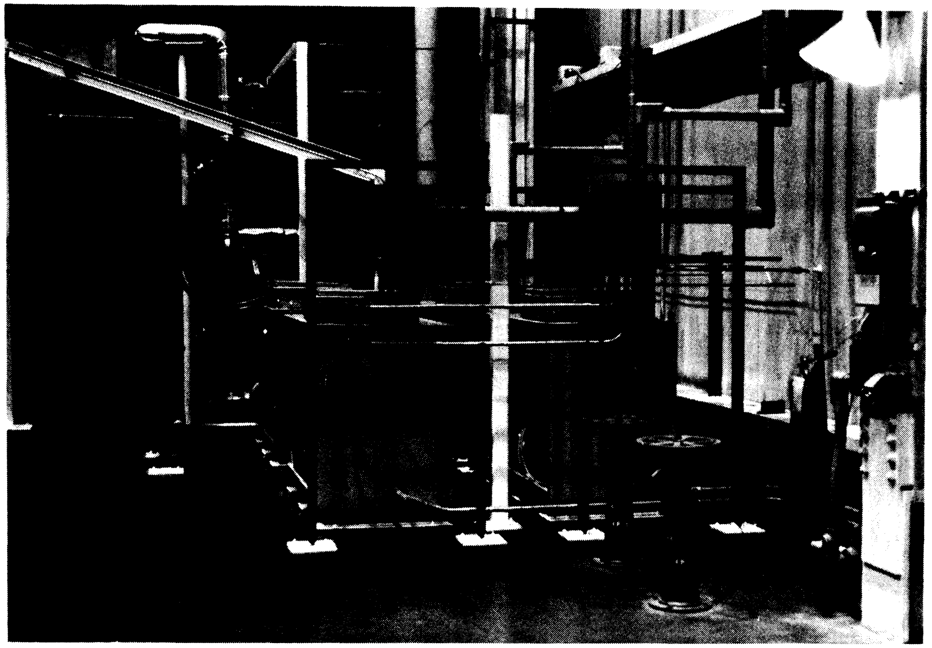
(II.B.3)

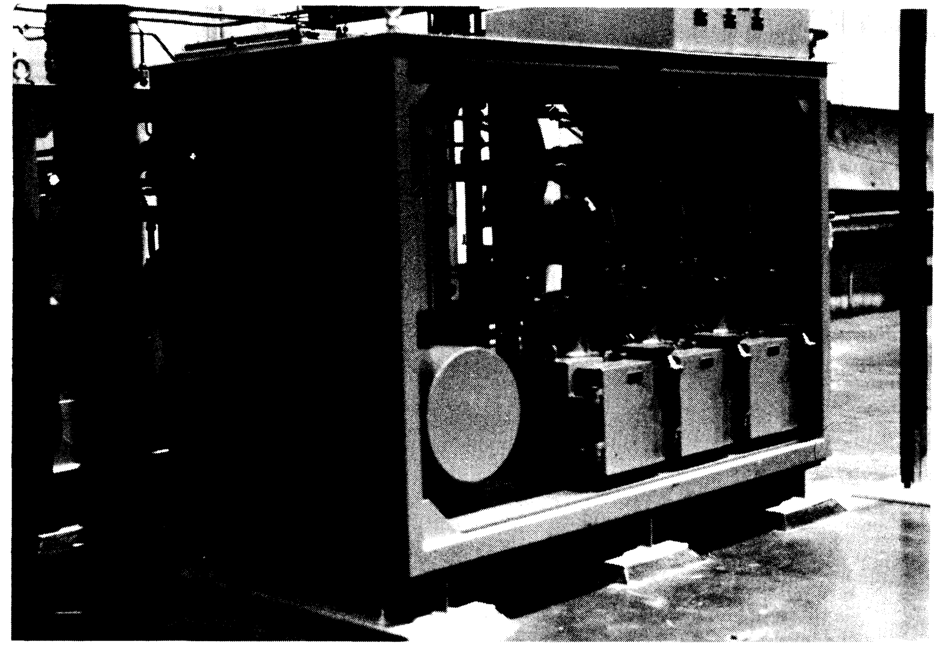




One of two hydrogen recombiners installed at TMI-1 and its dedicated penetration. Note the upper penetration is the one dedicated for the hydrogen recombiner use.

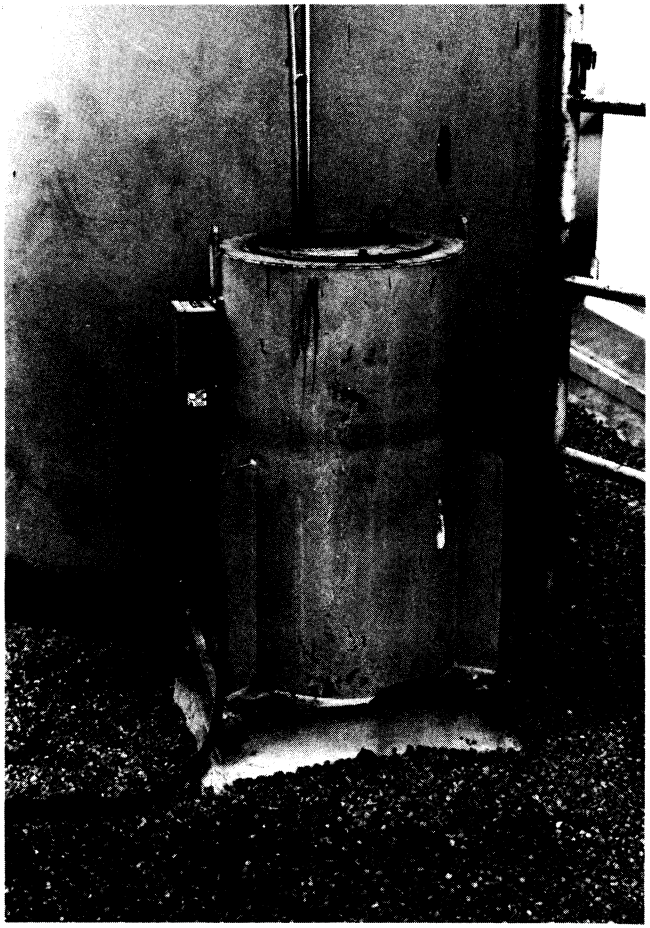
(II.E.4.1.2)





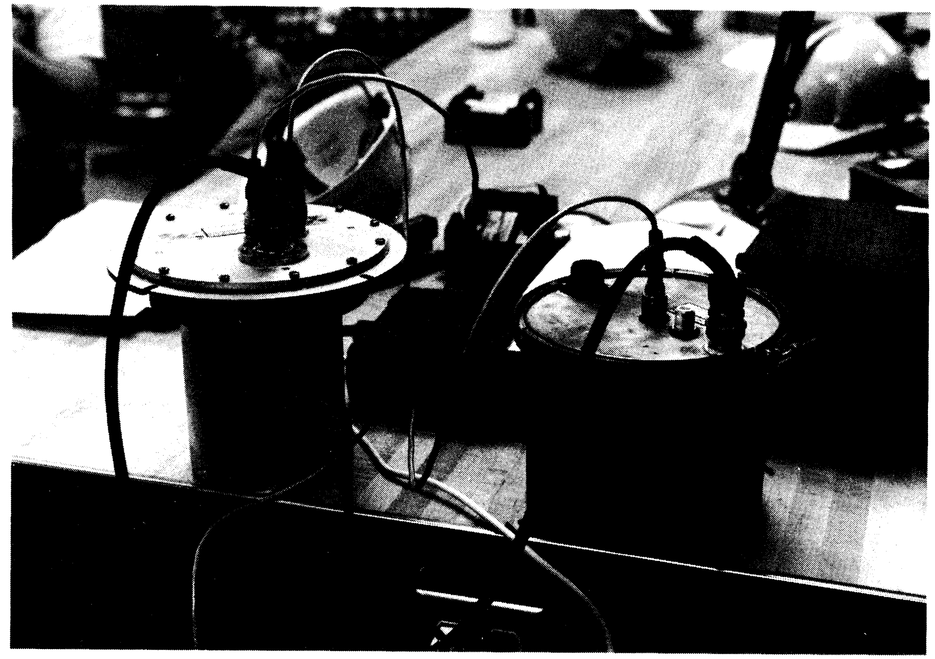
Typical arrangement of Noble Gas and Iodine Particulate Monitors found at a majority of the sites. In the picture on the right the three boxes contain the particulate filters and the circular chamber collects the gas for monitoring for Noble gases.

(II.F.1.1 and II.F.1.2)



Oconee noble gas monitor mounted outside the containment stack for unit 2. Units 1 and 3 have identical setups.

(II.F.1.1)



This is the general area radiation monitor used as the noble gas monitor at Oconee. The detector is on the left and the pre-amp on the right. Normally the pre-amp is connected directly to the detector and this setup has proven very reliable. But when used in this particular application, the pre-amp had to be encased in lead shielding to protect it from the postulated high radiation field during an accident and this requires separating it from the detector by a short distance (shorter than what is shown in the picture). Moisture getting into these connections has caused reliability problems.

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14. ABSTRACT (200 words or less)

This report provides the results of a study conducted by the U. S. Nuclear Regulatory Commission staff to compare the degree to which eight Babcock and Wilcox (B&W) designed licensed nuclear power plants have complied with the requirements in NUREG-0737, "Clarification of TMI Action Plan Requirements". The eight licensed operating plants examined are as follows: Arkansas Nuclear One Unit 1 (ANO-1), Crystal River Unit 3, Davis Besse, Oconee Units 1, 2 and 3, Rancho Seco, and Three Mile Island Unit 1 (TMI-1). The purpose of this audit was to establish the progress of the TMI-1 licensee, General Public Utilities (GPU) Nuclear Corporation, in completing the long-term requirements in NUREG-0737 relative to the other B&W licensees examined.

15a. KEY WORDS AND DOCUMENT ANALYSIS

15b. DESCRIPTORS

Babcock & Wilcox (B&W)
ANO-1
Crystal River 3
Davis Besse
Oconee 1, 2, 3
Rancho Seco
TMI-1
NUREG-0737
TMI Action Plan Requirements

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