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EISENHUT,D.G. Division of Licensing

SUBJECT: Forwards responses to NUREG-0737 re post TMI requirements. Util will meet most implementation dates specified in NUREG-0737.Other dates cannot be committed to w/o evaluating full impact of proposed mods.

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WISCONSIN PUBLIC SERVICE CORPORATION

P.O. Box 1200, Green Bay, Wisconsin 54305

January 5, 1981

Mr. D. G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Plant
Post TMI Requirements; NUREG 0737

In your letter of October 31, 1980, you transmitted NUREG 0737 and requested that licensees respond within 45 days, furnishing confirmation that the implementation dates in enclosure 1 of NUREG 0737 will be met. In subsequent conversations with our project manager, it was agreed that our response to your letter could be included with the information requested to be submitted by January 1, 1981. It was also agreed that the January 1, 1981, letter could be delayed until January 5, 1981.

The attachment to this letter is WPSC's response to NUREG 0737. In most cases, WPSC will meet the implementation dates indicated in enclosure 1 of NUREG 0737. However, there are certain implementation dates that WPSC cannot commit to at this time, due to vendor schedules which are beyond our control, or to the unavailability of equipment which will meet the requirements. Those cases are specified in the attachment.

Certain items in NUREG 0737 have not had implementation dates established and this represents a certain amount of uncertainty in the requirements for these items. For this reason, WPSC cannot commit to the requirements for Control Room design review and Plant Safety Parameter Display at this time. WPSC will continue to monitor the staff's progress concerning these issues. In the case of the emergency support facilities, WPSC has already expended a considerable amount of money in good faith to meet requirements that were promulgated in NUREG 0578 shortly after the accident at TMI. We urge the staff to maintain flexibility in the finalized requirements concerning these buildings, so as not to cause undue hardship on utilities who have proceeded with implementation.

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Enclosure 1 of NUREG 0737 includes implementation dates for certain revisions to technical specifications. We have submitted our response to the staff's request for technical specification revisions in a letter dated December 23, 1980, from Mr. E. R. Mathews to D. G. Eisenhut.

In previous correspondence we stated that we are committed to the safe operation of the Kewaunee Plant. We believe it is prudent to proceed with modifications to the plant or its operation in a safe, orderly manner, only after first evaluating in full the effects of such modifications. Action by the staff leads us to believe the staff realizes this too, as is evidenced by the revised implementation schedule indicated by enclosure 1 of NUREG 0737. It is for this reason that we cannot commit to certain implementation dates without first evaluating the full impact of the proposed modifications.

Very truly yours,

E. R. Mathews

E. R. Mathews, Vice President
Power Supply & Engineering

snf

Attach.

cc - Mr. Robert Nelson, NRC Resident Inspector
RR #1, Box 999, Kewaunee, WI 54216

Subscribed and Sworn to
Before Me This 5th Day
of January 1981

Jerome O Lewz

Notary Public, State of Wisconsin

My Commission Expires
2-6-83

Response to NUREG 0737

I.A.1.1 Shift Technical Advisor

A. STA Duties: Accident Assessment

The STA's primary concern is the safety of the plant. He is to aide the Shift Supervisor in diagnosing off-normal events and to provide him advice. The STA is available, in the control room, within 10 minutes of being summoned.

B. STA Duties: Operational Experience Assessment

Operational Experience Assessment is performed in several stages. An on site engineering group reporting directly to the Plant Manager coordinates incoming information for operational experience assessment, and on a priority basis assigns selected items to an STA for evaluation. The evaluation is disseminated to other STA's, corporate and plant personnel as applicable. STA's are assigned from various corporate and plant departments providing valuable experience from different backgrounds and disciplines for operational experience assessment.

Routine assignments include but are not limited to:

1. Engineering evaluation of the operating history of the plant
2. Licensee Event Reports from plants of similar design
3. Significant Events distributed through industry sources
4. Engineering evaluation of the adequacy of maintenance, testing, and operating procedures, quality assurance, or other areas where problems exist or performance is not up to quality standards.

C. STA Qualification

Personnel assigned as STA's shall have the following minimum requirements:

1. Basic college level technical education or equivalent in mathematics, reactor physics, chemistry and materials, thermodynamics, fluid mechanics and heat transfer, electrical and reactor control theory.

   The qualifications of each candidate for STA shall be evaluated on a credit hour/contact hour basis for each of the above subjects in comparison to the requirements of section 6.1 of INPO's, "Nuclear Power Plant Shift Technical Advisor--Recommendations for Position Description, Qualifications, Education and Training", Rev. 0, April 30, 1980. Where lacking, specific training and education will be provided to meet these standards.

2. An understanding of the details of the design, functions, arrangement and operation of Kewaunee Plant systems.

   The qualifications of each candidate for STA shall be evaluated in comparison to the requirements of a licensed reactor operator in these areas. Where lacking, a candidate will be given training which at least meets the guidance provided in sections 6.2, 6.4, 6.6 and 6.8 of the previously referenced INPO document.

3. An understanding of transient and accident response including multiple equipment failures and operator errors.

   Each candidate shall receive transient and accident response training equivalent to or greater than that specified in section 6.7 of the previously referenced INPO document.

Furthermore, the evaluation of the selection and assignment of an STA shall be made following the guidance of Section 5 of the INPO document.

D. STA Training Program

In order to meet the requirements of having a trained STA on duty by January 1, 1981 candidates selected must meet the requirements of item C defined above. Senior staff engineers met these requirements through education, experience, previous or current reactor operator license/certification and participating in Simulator Training and Transient and Accident Assessment training. Junior personnel, (less than 5 years nuclear experience) met these requirements through background education and a rigorous training program based on a senior reactor operators training program (INPO guidance was not issued at the commencement of the program). The basic outline for the STA training program is:

Academic Training (10 weeks, 3 day/week)
Test Reactor Training (2 weeks)
Kewaunee Plant Systems Training (13 weeks, 3 day/week)
PWR Simulator Training (3 weeks)
Accident Assessment Training (3 weeks, 3 day/week)
Administrative Controls and Procedures (3 days)
Final Exam (RO, SRO, Oral 3 days)

Future STA training will be based on candidates needs and may not be identical to the program just completed for junior staff personnel.

E. STA Retraining

STA retraining and upgrading of personnel qualificatons will be three phased:

1. Periodic lectures/meetings held to keep STA's current on modification of systems, equipment, and procedures.

2. Annual review of transient and accident analyses including multiple failures, and the review of selected industry events that could have led to more serious incidents.

3. Simulator training.

The total contact hours will be at least 80 hours with no less than 20 hours in each phase of retraining.

F. Long Term STA Program

The STA program is seen by WPSC as an interim program to be eventually replaced by a permanent assignment. At this time, the requirements for a permanent resolution to this program are admittedly undetermined, as noted in item I.A.1.1 of NUREG 0737. Therefore, WPSC cannot commit to a long term phase-out of the STA program, or, for that matter, to a permanent STA program.

WPSC favors the eventual phase out of the STA program. We feel that an appropriate resolution to this item would be to upgrade the operating shift personnel by providing training in those areas where prior education and experience are not sufficient. WPSC encourages the staff to consider previous nuclear related experience, including the nuclear navy, as an acceptable alternative to a rigid college degree requirement. This would enable utilities to utilize operating personnel (specifically, the shift supervisor or assistant shift supervisor in WPSC's case) as STA's, after appropriate training in accident assessment and other necessary areas.

Item 1.A.1.2 Shift Supervisor Responsibilities

This item has been completed. No documentation has been requested at this time.

1.A.1.3 Shift Manning

WPSC is implementing programs to meet the requirements concerning overtime and minimum shift manning, as specified in IE Circular 80-02 and D. G. Eisenhut's letter of July 31, 1981 with the following exceptions.

Overtime

It has always been WPSC policy to limit overtime to reasonable levels for all employees. At certain times, however, it becomes difficult to do so due to abnormal occurrences. Specifically, the Kewaunee plant typically has only one extended outage per year. During that outage, a significant amount of work (maintenance, testing, and modification) is completed in a very short time, as it is the nature of this business to minimize outage time. Typically, these outages are on the order of 6 to 8 weeks and it is not uncommon for certain personnel to work most, if not all, of these days. WPSC notes that extended periods of shutdown for refueling, maintenance, or major plant modifications was excluded from the overtime policy in NUREG 0737. Nevertheless, we felt it important to clearly state our position on this matter.

Secondly, we are confused by this statement on page I.A.1.3-2 of NUREG 0737 concerning reactor operator overtime:

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

It is not clear if this statement is intended to include any four hour period of time or only that time beyond the operator's normal shift. WPSC feels that the latter interpretation was intended, furthermore, we believe this is only of concern for extended periods of time, and not for temporary circumstances beyond our control.

Often due to sickness, weather, or other unusual circumstances, we cannot predict if an operator will be required to work more than eight hours continuously. Because of this WPSC cannot commit to this recommendation, as current staffing does not include an extra operator on shift who would be available for temporary relief. As noted below, we have implemented a program to increase our shift staff. When that program is complete (i.e.; an additional SRO on each shift), WPSC intends to utilize the additional operator for short-term relief of the control room operator when extenuating circumstances arise and reactor operator relief is unavailable. In this way, the guidance that the reactor operator's period at the control board should not exceed four hours (after having worked eight hours) can be met. WPSC is also taking steps to provide more operators on shift than the minimum requirements promulgated in D. G. Eisenhut's July 31, 1980 letter. Since this is a long term program with several contingencies involved, a firm commitment regarding it cannot be made at this time.

Nevertheless, when completed, the program will provide additional manpower for relief of the control room operators.

Shift Manning

WPSC has implemented a training program to provide an additional SRO on each shift by July 1, 1982. Basically this program will train new personnel to replace existing RO's freeing them for SRO training. A sufficient number of new personnel have been hired to replaced the existing RO's; however it should be noted that the success of our program is contingent upon the successful reactor operator training of new personnel, followed by the successful SRO training for the existing RO's. We are concerned at the very short time frame available for this training. Since it will take 12-15 months to train the group of replacement RO's, there is only a short amount of time available for formalized SRO training. Any setbacks in qualification of personnel will force a later date for full implementation.

1.A.2.1 Immediate Upgrading of RO and SRO Training and Qualifications

Sub-item 1. SRO Experience

No documentation has been requested at this time.

Sub-item 2. SRO's be RO's one year

No documentation has been requested at this time.

Sub-item 3. Three month training on shift

No documentation has been requested at this time.

Sub-item 4. Modify training

The Kewaunee training program currently includes or will be revised to include the requirements specified in Mr. Denton's letter of March 28, 1980. The March 28, 1980 letter requires training programs to include:

1. Heat transfer, fluid flow, and thermodynamics.
2. Training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged.
3. Increased emphasis on reactor and plant transients.

We have implemented changes which will direct proper emphasis to these areas. Work on item 2 is currently being performed with the cooperation of various vendors and is not finished yet. When this work is complete WPSC will incorporate the necessary training into our training program. Since that work is not complete and since the inclusion of the other two items above did not result in a substantive change to our training program, it is premature to submit our program for NRR review at this time.

Sub-item 5. Facility Certification

No documentation has been required at this time.

I.A.2.3 Administration of Training Programs

No documentation has been requested at this time. The Kewaunee training instructors who teach systems, integrated responses, and transient courses in the reactor operator training or requalification training programs are qualified senior reactor operators and participate in a requalification program.

I.A.3.1 Revise Scope and Criteria for Licensing Exams

No documentation has been requested at this time.

I.B.1.2 Independent Safety Engineering Group

This item is applicable only for applicants for an operating license. Therefore, it does not apply to KNPP and no documentation is required.

I.C.1 Short Term Accident and Procedures Review

1. SB LOCA

The Kewaunee emergency procedures have been reviewed, and in certain instances revised using the Westinghouse Emergency Operating Instruction Guidelines, which were developed by the Westinghouse Owners Group, as a basis for comparison. No further documentation is required.

2. Inadequate core cooling, and

3. Transients and Accidents

The Westinghouse Owners Group is scheduled to submit by January 1, 1981 a detailed description of their program to comply with these requirements. When the Westinghouse Owner's Group work is complete, WPSC will utilize the results of that effort to revise the KNPP emergency operating procedures, as is deemed necessary.

I.C.2 Shift and Relief Turnover Procedures

Although no further information is requested at this time, it has come to our attention through the NRC Region III office that there is a discrepancy between our position and the staff's "Evaluation of Licensee's Compliance with Category "A" Items of NRC Recommendations Resulting from TMI-2 Lessons Learned." This letter serves to clarify our position.

Our position concerning this item (item 2.2.1.C of NUREG 0578) is clearly stated in the December 31, 1979 letter from E. R. Mathews to H. R. Denton. While procedures in the form of ACD's are in place concerning shift turnover, we do not require the shift supervisors or the control room operators to sign a shift turnover checklist. We are of the opinion that shift turnover can be accomplished in a professional, adequate manner without the aid of a checklist; requiring a checklist would be another administrative burden on the operating crew, without an increase in safety.

I.C.3 Shift Supervisor Responsibility

No further documentation has been requested at this time.

I.C.4 Control Room Access

No further documentation has been requested at this time.

I.C.5 Feedback of Operating Experience

No submittal has been requested at this time. WPSC has an operating experience feedback organization in place which provides the functions described by this item. The procedures which will be utilized by this organization have not been finalized at this time. These procedures should be complete by February 1, 1981.

I.C.6. Verify Correct Performance of Operating Activities

No documentation has been requested at this time. The Kewaunee plant has always had an extensive program to provide the assurance that operating activities are performed correctly. This program is under review, and necessary changes are being incorporated on a timely basis.

For example, manual valves which are safety related are now required to have their position verified by a second person prior to returning the affected system to service after surveillance or maintenance operation that affect the valve. Other areas that are currently under review are:

- revising procedures to assure that the on shift SRO is kept fully informed of surveillance and maintenance operations;

- requiring two people for tag-out verification except under high radiation conditions;

- assuring that the control room is informed of changes in equipment status;

- independent verification of system line-up prior to its return to service.

These reviews are expected to be completed by March 1, 1981.

I.D.1. Control Room Design Reviews

No implementation date has been set for this item. No documentation has been requested at this time.

I.D.2. Plant Safety Parameter Display Console

No implementation date has been set for this item.

II.B.1 Reactor Coolant System Vents

Information on the reactor coolant system vents has been requested to be submitted by July 1, 1981. Kewaunee is proceeding with installation of a Reactor Head and Pressurizer Vent System and expects installation to be complete during the 1981 refueling outage (April, 1981). Until further analysis is complete and operating procedures are in place the valves will be placed in a condition to minimize the potential for inadvertent actuation.

II.B.2. Plant Shielding

No documentation is requested at this time. WPSC is implementing the modifications necessary to assure access to vital areas in the auxiliary building. These modifications are scheduled to be completed prior to 7/1/82. The information for this work is currently available at the offices of Fluor Power Services, Inc., our A/E, who did the work. Since this work is continuing, the documentation package is a "living" document and as such, is under review and revision. The documentation will be formalized when the modifications are complete.

WPSC takes exception to clarification (4) of item II.B.2 (page 3-63 of NUREG 0737). This item concerns the Radiation Qualification of Safety Related Equipment. WPSC has, of our own accord, utilized the radiation study required by NUREG 0578 to determine equipment doses in the auxiliary building following a hypothetical accident. We initiated this study prior to the guidance given concerning the source terms to be used for this study. We have incorporated conservatism in our calculations by assuming all doses are pipe centerline doses and not taking credit for spatial attenuation. We feel that the guidance given in (4) (b), specifically, the assumption that the source term remains in an undiluted primary coolant, is beyond the design basis and is unnecessarily conservative.

II.B.3. Post Accident Sampling

No documentation has been requested at this time. The necessary modification is scheduled for completion prior to January 1, 1982.

II.B.4 Training for Mitigating Core Damage

We are presently developing a revised licensed operator training program as addressed in item I.A.2.1. This program will include general guidelines for mitigating core damage. Specific training for equipment installed to meet NRC requirements will be consistent with the installation of the equipment and may not meet the implementation dates of April 1, 1981 and October 1, 1981 in cases where the installation of the equipment utilized in mitigation of core damage is not complete.

II.D.1. Relief and Safety Valve Test Requirements

No specific information is requested at this time. We are participating in the EPRI safety and relief valve testing program. That program was described in a letter from R. C. Youngdahl (Consumers Power Company) to D. G. Eisenhut dated December 15, 1980. Upon completion of the program the requirements for plant specific analysis will be addressed. Compliance with the requirements to supply the documentation for this item will be contingent upon the timely completion of the EPRI program and a commitment to the dates given cannot be made at this time.

II.D.3. Valve Position Indication

No further information is required. WPSC submitted a letter on December 23, 1980, detailing its position concerning the technical specification for this item.

II.E.1.1 Auxiliary Feedwater System Evaluation

This item is in progress with an exchange of information continuing between WPSC and the staff. WPSC is currently working with Westinghouse on the AFW flowrate design basis. The necessity of modifications cannot be determined until that work is complete; therefore, we cannot commit to the implementation dates given in NUREG 0737 at this time.

The Kewaunee AFW system has proven to be very reliable. This fact, along with the significant amount of time for operator action due to the large steam generator inventory supports a less hurried implementation schedule.

II.E.1.2 Auxiliary Feedwater System Initiation and Flow

Item 1. a. Control Grade Initiation
No documentation has been requested at this time.

Item 1. b. Safety Grade Initiation.

The initiation circuitry for the auxiliary feedwater system at Kewaunee is a safety grade system receiving signals from safety grade equipment. The initiation system meets the requirements of this item; information concerning this has been submitted to the staff in Section 6.6 of the FSAR and in the following letters:

Letter from E. R. Mathews to D. G. Eisenhut dated October 30, 1979;

Letter from E. R. Mathews to D. G. Eisenhut dated December 14, 1979;

Letter from E. R. Mathews to Steve Varga dated October 17, 1980.

Item 2. a. No Control Grade Flow Indication documentation has been requested at this time.

Item 2. b. The WPSC position regarding Technical Specifications for AFW Flow Indication was submitted to the staff in the December 23, 1980 letter from E. R. Mathews to D. G. Eisenhut.

Item 2. c. Safety Grade Flow Indication.

By July 1, 1981, the flow indication for the auxiliary feedwater system will be upgraded to a safety grade indication. In regards to that indication:

1. The transmitters will be located in the auxiliary building and thus need not be qualified for post-LOCA use in containment. They will be qualified for radiation performance to IEEE 323, 1971 and seismic performance to IEEE 344, 1975.

2. The power supply will be from a separate vital instrument bus for each of the two loops. Indication will be separated from the power supply and transmitter by an isolation amplifier.

3. Test jacks will be provided with the instrument rack for periodic test and calibration of the loops.

4. The modifications will be performed in accordance with our approved Quality Assurance program.

5. The present indicators will be utilized. These indicators give a continuous indication of auxiliary feedwater flow.

The present transmitters and power supplies will be replaced with QA type 1 components and powered from vital instrument buses. New cable will be run to insure separation of the two trains. Additional information regarding the auxiliary feedwater system will be submitted in response to the staff's request for information WPSC received on December 15, 1980. That response is scheduled to be submitted by January 25, 1981.

II.E.3.1 Emergency Power for Pressurizer Heaters

1. Upgrade Power Supply

   No documentation has been requested at this time.

2. Technical Specifications

   The WPSC position concerning these technical specifications was submitted in a letter from E. R. Mathews to D. G. Eisenhut on December 23, 1980.

II.E.4.1 Dedicated Hydrogen Penetrations

No documentation has been requested at this time.

II.E.4.2. Containment Isolation Dependability

Items 1-4 No documentation has been requested at this time.

Item 5. Containment Pressure Setpoint

Containment isolation occurs on any safety injection signal at the Kewaunee Plant. In addition to safety injection signals generated by RCS or S/G parameters, a containment pressure of 4 psig will also generate a safety injection signal, causing containment isolation. Technical Specification 3.6.C of the Kewaunee Technical Specifications limits the maximum containment pressure for normal operation to 2 psig. Based on this and the diverse actuation signals for containment isolation, we feel that the concerns of the staff have been satisfied and that no further action is necessary in this regard.

Item 6. Containment Purge Valves

WPSC has installed blocks which limit the opening of the Kewaunee containment purge and vent valves. Information regarding this was sent to the staff by letter from E. R. Mathews to A. Schwencer dated July 2, 1980. Correspondence between WPSC and the NRC is continuing on this subject, therefore, further information is not necessary in this submittal.

Item 7. Radiation Signal on Purge Valves

A high containment radiation signal will close the subject valves at Kewaunee.

Item 8. Technical Specifications
WPSC response to the staff's request for technical specifications was submitted to the staff in a letter from E. R. Mathews to D. G. Eisenhut dated December 23, 1980.

It has come to our attention through the efforts of the Region III offices of the NRC that a discrepancy exists between our commitment and the evaluation by the staff referenced in item I.C.2. The specific item is concerning normally closed isolation valves. The staff evaluation states that "normally closed isolation valves will be locked closed and administratively controlled such that at any time they are open during plant operation, a dedicated person will be assigned to close it immediately in the event of an emergency or when the operation is complete."

This commitment was not made by WPSC. Our position concerning this item is given in the following letters:

October 19, 1979 letter, E. R. Mathews to D. G. Eisenhut

December 31, 1979 letter, E. R. Mathews to H. R. Denton

II.F.1. Accident Monitoring

Item 1. Noble Gas Monitor

Item 2. Iodine/Particulate Sampling

Item 3. Containment High Range Monitor

Item 4. Containment Pressure

Item 5. Containment Water Level

Item 6. Containment Hydrogen

No documentation has been requested at this time.

WPSC is implementing these items and anticipates completions of all items prior to January 1, 1982.

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

Item 1: Subcooling Meter
No documentation has been requested at this time.

Item 2: Technical Specifications
WPSC's response to the staff's request for technical specifications was submitted in the December 23, 1980 letter from E. R. Mathews to D. G. Eisenhut.

Item 3: Install level instruments

Reactor vessel level indication has been proposed as an indication of inadequate core cooling. Installed equipment which could be used for detecting inadequate core cooling at the Kewaunee Plant includes:

1. Core Exit Thermocouples
2. Hot and Cold Leg Wide Range RTD's
3. Pressurizer Pressure and Level Instruments
4. Primary Coolant Flow Instruments
5. Subcooling Monitors
6. Reactor Coolant Pump Ammeters
7. Steam Generator Level Instruments
8. Incore and Excore Neutron Detectors

We believe that this existing instrumentation is adequate to detect inadequate core cooling. We concur that a reactor vessel level device which gives an unambiguous indication of actual vessel fluid level would be useful to the operator. However, we have not been able to find any system on the market that has been tested and proven to give an unambiguous indication. Therefore, at this time we cannot commit to a specific system, but are continuing to monitor progress on several systems, including differential pressure and heated thermocouples. Until an unambiguous system has been identified we cannot commit to the implementation dates specified in NUREG 0737.

II.G.1. Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators

No documentation has been requested at this time.

II.K.1. IE Bulletins

No documentation has been requested at this time.

II.K.2. Orders on B & W Plants

Items 8, 9, 10, 11

These items are for B & W Plants only. Since Kewaunee is a Westinghouse PWR, no documentation is required.

II.K.2.13 Thermal Mechanical Report

To completely address the NRC requirements of detailed analysis of the Thermal/Mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater, a program will be completed and documented to the NRC by January 1, 1982. This program will consist of analysis for Generic Westinghouse PWR Plant groupings. It will be performed for the Westinghouse Owners Group by Westinghouse.

Following completion of this generic program, additional plant specific analyses, if required, will be provided. A schedule for the plant specific analysis will be determined based on the results of the generic analysis.

II.K.2 Orders on B & W Plants

Items 14 through 16 are for B & W Plants and do not apply to KNPP.

II.K.2. Orders on B & W Plants

Item 17. Voiding in RCS

Item 19. Benchwork Analysis of Sequential Auxiliary Feedwater Flow

No documentation has been requested at this time.

II.K.3.1 Auto PORV Isolation, and

II.K.3.2 Report on PORV failures

The Westinghouse Owners Group is in the process of developing a report (including historical valve failure rate data and documentation of actions taken since the TMI-2 event to decrease the probability of a stuck open PORV) to address the NRC concerns of item II.K.3.2. However, due to the time consuming processing of data gathering, breakdown, and evalution this report is scheduled for submittal to the NRC on March 1, 1981. As required by the NRC, this report will be used to support a decision on the necessity of incorporating an automatic PORV isolation system as specified in task action item II.K.3.1.

II.K.3.3. Reporting of SV and RV Failures and Challenges

WPSC will comply with the requirements specified in item II.K.3.3. as transmitted to us in D. G. Eisenhut's letter of May 7, 1980. Specifically,

a. the failure of a pressurizer power operated relief valve to close will be reported promptly to the NRC,

b. challenges to pressurizer power operated relief valves will be documented in the annual report,

c. the failure of a pressurizer safety valve to close will be reported promptly to the NRC,

d. challenges to the pressurizer safety valves will be documented in the annual report.

This commitment will be effective as of the date of this letter.

II.K.3.5. Auto-trip of the RCP's

The Kewaunee plant has installed a safety grade reactor coolant pump trip circuit to eliminate the concerns generated after the accident at TMI-2 concerning tripping of the reactor coolant pumps during a small break LOCA. WPSC is following the analyses being performed by Westinghouse for the Westinghouse Owners Group concerning this issue, and will take appropriate action when those analyses are complete.

II.K.3.7. Evaluation of PORV Opening Probability

This item is applicable only to B & W plants.

II.K.3.9 PID Controller

WPSC has completed the work on this item as you were informed by our August 4, 1980 letter, from E. R. Mathews to D. G. Eisenhut.

II.K.3.10 Proposed Anticipatory Trip Modifications

Reactor trip occurs on a turbine trip at any power level above approximately 10 percent of full power at the Kewaunee Plant. No further action is necessary.

II.K.3.11 Justify Use of Certain PORV

This item is not applicable to the Kewaunee Plant.

II.K.3.12 Anticipatory Trip on Turbine Trip

The Kewaunee design utilizes a reactor trip on turbine trip. No further action is required on this item.

II.K.3.13 - II.K.3.16

These items are for BWR's only. No action is required by WPSC.

II.K.3.17 ECC System Outages

Table 3.17 (next page) is a summary of ECC System outages at Kewaunee for the previous five years. Since the technical specifications require reporting of all conditions leading to a component in an ECCS system operating in a degraded mode permitted by the limiting conditions for operation; this data was based on a review of the Kewaunee LER's. In addition to those outages, routine surveillance testing and preventive maintenance outages of ECCS components are summarized as follows:

1) Monthly each SI and RHR pump is placed in a recirculation mode for less than 30 minutes.

2) Annually each pump has a bearing oil change and oil sample taken from oil reservoir and change performed if needed. The oil change usually takes from 3-4 hours.

3) Once every two years in conjunction with the oil change an insulation resistance test is performed. This test takes about 30 minutes.

4) During operation annual valve and breaker maintenance is performed on those valves having the required redundancy. Twelve valves are in this category and each is taken out of service for about two hours.

Additional testing and maintenance requirements are performed during the annual refueling shutdown when system operation is not required.

Table 3.17: ECC System Outages

OUTAGE DATE	DURATION	CAUSE	COMPONENT	CORRECTIVE ACTION
3-2-78	$<$ 4 hrs	Failure to open	Valve SI-350B	Operated on retest. No failure cause found.
4-4-78	$<$ 5 hrs	Valve torqued out after starting to open	Valve SI-350B	Cycled valve 3 times and operated normally. Valve was repacked during refueling outage.
5-30-79	$<$ 2 hrs	Failure to open from control room	Valve SI-302A	Manually opened. Retested satisfactory. No cause found.
3-3-80	$<$ 15 hrs	Would not open	Valve SI-351A	Adjust torque setting. Retested satisfactory.

II.K.3.18 - II.K.3.24

These items are for BWR's and are not applicable to Kewaunee.

II.K.3.25 Power On Pump Seals

At the Kewaunee Nuclear Plant the component cooling pump automatically starts following restoration of voltage to the corresponding safeguards bus in the event of a loss of all offsite power or within 40 seconds following a safety injection signal (i.e., bus loading following load shed). This configuration meets the criteria of item II.K.3.25 and no further action is required.

II.K.3.27 - II.K.3.29

These items are specifically intended for BWR's. No action on the part of Kewaunee is required.

II.K.3.30 Small Break LOCA Methods; and

II.K.3.31 Analysis to show compliance with 10CFR50.46

WPSC is supporting the Westinghouse Owners Group in the resolution of this item. We understand that a detailed outline of the scope and schedule for this effort was supplied to the staff via letter dated September 26, 1980. No further action on WPSC's part is required at this time.

II.K.3.40, 43, 44, 45 46 and 57

These items are for BWR's only. No further action is required.

III.A.1.1. Emergency Preparedness, Short Term

No further information required at this time.

III.A.1.2 Upgrade Emergency Support Facilities

Although no new information is required at this time, we offer the following descriptions of the effort and money that has been expended on our part to meet what has evolved to be tentative requirements on the Technical Support Center (TSC) and Emergency Operations Facility (EOF). We are concerned that to this date, the requirements still haven't been finalized, as is evidenced by the unspecified implementation dates on this item reported in NUREG 0737.

Technical Support Center

The Kewaunee Nuclear Power Plant onsite Technical Support Center is a new structure being located immediately adjacent to and directly north of the existing turbine and auxiliary buildings. The structure is designed and constructed to the same specifications as the original auxiliary building and will, therefore, be capable of withstanding the design basis earthquake. The south wall of the structure is adjacent to column row 9 which is also the north wall of the control room. Shielded access to the control room is provided via two flights of stairs. The technical support center is below grade (elevation 606') at elevation 586' while the control room is at elevation 626'.

Normal power to the TSC is via non-safeguard 4160 volt bus 1-4. Emergency backup power to supplied by a new 600 KW onsite technical support center diesel generator housed in the center at grade level. The instruments are powered via an inverter which is backed up by a new battery housed at grade level and the TSC emergency diesel generator.

The display and work area is approximately 48' x 31'. The north conference room will be equipped with telephones and be available for use by the NRC. Two offices and a south conference room provide additional private work space. Also housed in the seismic structure is the Radiological Analysis Laboratory through which direct access to the high level sampling facility is provided. Samples will be taken to this area for analysis and counting and the results will be transmitted to the other emergency facilities as required.

Record storage for the TSC will be provided in the records storage room located on the ground level of the TSC. The general office area located on this level is also available as additional work space.

The TSC ventilation system will consist of a single train recirculation unit, pressurization unit and HEPA Filters. Emergency power for these units will be provided by the TSC Diesel Generator.

The TSC is shielded by the North Wall of the auxiliary building, the south wall of the TSC and the work area is provided additional shielding by a 12" cement roof slab. Total integrated exposure for the 30 days following the design basis accident is 1.9 Rem.

The interim instrumentation for TSC will consist of a terminal from the existing Prodac 250 computer, a data logger which will monitor 44 points, and print out on three high speed printers, and 14 "hot pen" recorders which will activate on a reactor trip and record specific signals. WPSC in conjunction with other utilities has developed a specification for an upgraded computer which has been sent out for bids.

The communications for the onsite technical support center will consist of dedicated phone links to the Control Room, the Emergency Operations Facility, the Radiochemistry Area, the Onsite Operations Support Center, as well as the existing NRC Lines. Additionally the plant phone system, sound powered phone system and Gai Tronics paging system will be available in the TSC.

The Onsite Technical Support Center with the features described herein is scheduled to be operational by 6/1/81.

Operations Support Center

The Kewaunee Plant Assembly Room which is now utilized as the Interim Technical Support Center will be the Operations Support Center.

Dedicated communication between this center and the Control Room and the Technical Support Center will be provided.

The area is habitable within 24 hours following the design basis accident. This area will be designated as the Operations Support Center when construction is complete on the Technical Support Center. This is scheduled to be complete by 6/1/81.

Emergency Operations Facility

The emergency operations facility is a completed structure located approximately 400 feet to the northeast of the Kewaunee Containment Building outside the security fence. The structure provides working space for WPSC, NRC, State, and Local Officials.

The EOF ventilation system has recirculation capability and provisions for HEPA and Charcoal filters.

Communication exists to the Health Physics area, the onsite technical support center and the control room via dedicated phone lines. The NRC Health Physics line and Hot line have extensions in the NRC area and the WPS area of the EOF. Additionally, the plant paging system and phone system are available for use in this center.

Direct radiation at the EOF is 40 mrem/hr after eight hours, 7 mrem/hr after one day and 0.7 mrem/hr after one week following the design basis event. As shown on Figure 2.7.2 of the Kewaunee Nuclear Plant FSAR the percent of time the wind blows in the direction of the EOF is very low. Additionally we have not observed releases made from the plant to be reaching the ground elevation this close to the stacks.

Therefore, it is unlikely that this facility would require evacuation during a release, however, the meteorological conditions which would require an evacuation will be calculated and known in advance. In the event evacuation of this facility is necessary the Town of Carlton Town Hall located approximately 2.8 miles to the NNW has been leased and designated as a backup facility.

The EOF has access to all information on the plant Prodac 250 computer via remote telephone terminal.

III.A.2. Improving License Emergency Preparedness--Long Term

WPSC is upgrading its emergency plan in response to recent regulation. The upgraded plan was submitted on January 2, 1981.

III.D.1.1 Primary Coolant Outside Containment

No new information required at this time. WPSC has implemented a program to minimize leakage of systems likely to contain radioactive liquids outside of containment. In addition to that program, we are proceeding with a design change which will minimize the number of systems that will contain radioactive material post-LOCA, and will eliminate the use of the waste gas system for post accident operation. This was reported to you in the December 31, 1979 letter from E. R. Mathews to H. R. Denton.

III.D.3.3 In-plant Radiation Monitoring

WPSC has purchased the equipment necessary to provide the in-plant iodine monitoring capability required by this item. In addition to the on-site capability, WPSC has an agreement with the Point Beach Nuclear Plant which allows the use of their counting facility for iodine analysis.

III.D.3.4 Control Room Habitability Requirements

WPSC has implemented the review requested by this item but has not completed them at this time. In conjunction with the shielding review required by item 2.1.6.b of NUREG 0578, a review was undertaken by our A/E to determine radiation exposures to control room occupants after a design basis accident.

That review concluded that all elements of SRP 6.4 with respect to post-accident radiation doses will be met with the exception of the skin dose. However, that study also indicated that with the use of administrative procedures, the skin dose could be limited to the levels recommended by SRP 6.4. It should be noted that the calculated unprotected skin dose is less than 75 rem, which is permitted by the guidelines given in SRP 6.4.

Chlorine is not utilized at the Kewaunee Plant, and is not perceived to be a problem for this reason. The control room ventilation system was reviewed in accordance with IE Circular 80-03, Protection from Toxic Gas Hazards. That review did not identify any problems with the ventilation systems. Further review will be performed to determine control room ventilation system adequacy from other toxic gases.