Consumers Power Company

General Offices: 212 West Michigan Avenue, Jackson, Michigan 49201 • (517) 788-0550

July 9, 1981

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U.S. NUCLEAR REGULATORY COMMISSION

Director, Nuclear Reactor Regulation
Att Mr Dennis M Crutchfield, Chief
Operating Reactors Branch No 5
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-155 - LICENSE DPR-6 - BIG ROCK POINT
PLANT - UPDATE OF RESPONSE TO NUREG-0737,
CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

The NRC letter dated October 31, 1980, also identified as NUREG-0737, "Clarification of TMI Action Plan Requirements," incorporated into one document all TMI related items approved for implementation by the Commission at that time. Subsequent NRC letters, eg NUREG-0696 and Generic Letter 81-10, provided additional explanation or clarification of specific NUREG-0737 items. This total document package provides items to be addressed by Licensees to improve safety at power reactors as a result of lessons learned from the accident at Three Mile Island, Unit 2.

Consumers Power Company's initial response to NUREG-0737 was provided by our letter dated December 19, 1980. That response provided Consumers Power Company's intended actions with respect to the items of NUREG-0737 and superseded previous submittals on equivalent items from documents such as NUREG-0578, 0626, and 0660, except where specific references were made to previous submittals.

Since the December 19, 1980 submittal, Consumers Power Company has submitted numerous letters addressing in greater detail our responses to various NUREG-0737 items. Most notably, the Big Rock Point Plant Probabilistic Risk Assessment (PRA) was submitted for NRC review by our letter dated March 31, 1981. Because of the diversity and magnitude of the correspondence subsequent to our December 19, 1980 letter, Consumers Power Company has decided to consolidate this information into a formal update of our NUREG-0737 response.

Consumers Power Company's update of our response to NUREG-0737 is provided by this letter and its enclosure, entitled, "Consumers Power Company's NUREG-0737 Response - Big Rock Point Nuclear Plant - July 1981 Update." The information contained in this submittal does not supersede our initial response of December 19, 1980. Rather, it provides the status and schedule, as of July 1, 1981, of Consumers Power Company's continuing efforts to address various NUREG-0737 items. Therefore, the format and pagination of this update is such that the updated response to a particular item can be inserted

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Mr Dennis M Crutchfield, Chief
Big Rock Point Plant
July 9, 1981

into the December 19, 1980 submittal, "Consumers Power Company's NUREG-0737 Response - Big Rock Point Nuclear Plant" behind the corresponding item response.

This update is intended to provide Consumers Power Company's latest estimates of how and when various commitments will be performed, and the information contained herein should be valid for the period between July 1 and December 31, 1981. Nevertheless, this update represents Consumers Power Company's intended actions and best estimate schedules; unforeseen problems may require modification of these actions and schedules as evaluation, design and procurement progress. Such modification to Consumers Power Company's intended actions, where significant, will be formally submitted.

Please note the following changes included in this update concerning commitment dates given in Consumers Power Company's PRA submittal dated March 31, 1981:

1. Regarding the basis for continued deferral of BRP wide-range level instrumentation, the results of the operator action event tree analysis will be completed by July 31, 1981 rather than July 15, 1981. The proposed plan of action (if any), which will be based on the operator action event tree analysis, will be submitted by September 1, 1981 instead of July 15, 1981.

2. Regarding the basis for continued deferral of BRP isolation of emergency condenser on high radiation modification, the selection of a method to reduce the probability of false indication of tube rupture due to shine from the tube bundles will not be completed by July 15, 1981. A specific date for the completion of this effort cannot be provided at this time. Also, collection of information that will provide additional assurance that fatigue is not a significant contributor to the likelihood of emergency condenser tube bundle failure will be completed by December 31, 1981 rather than July 15, 1981.

David P Hoffman

David P Hoffman
Nuclear Licensing Administrator

CC Director, Region III, USNRC
NRC Resident Inspector-Palisades Plant

CONSUMERS POWER COMPANY
Big Rock Point Plant

NUREG-0737, Clarification of TMI Action Plan Requirements
Update of our December 19, 1980 Response to NRC letter,
dated October 31, 1980

Docket No 50-155
License No DPR-6

At the request of the Commission and pursuant to the Atomic Energy Act of 1954, and the Energy Reorganization Act of 1974, as amended, and the Commission's Rules and Regulations thereunder, Consumers Power Company submits an update of our December 19, 1980 response to NRC letter dated October 31, 1980 (NUREG-0737 - "Clarification of TMI Action Plan Requirements"). Consumers Power Company's update is dated July 9, 1981.

CONSUMERS POWER COMPANY

By R B DeWitt 7/9

R B DeWitt, Vice President
Nuclear Operations

Sworn and subscribed to before me this 9th day of July 1981

Helen I. Dempski

Helen I Dempski, Notary Public
Jackson County, Michigan
My commission expires December 14, 1981

CONSUMERS POWER COMPANY'S
NUREG-0737 RESPONSE

BIG ROCK POINT NUCLEAR PLANT

JULY 1981 UPDATE

I.A.1.1 SHIFT TECHNICAL ADVISOR

NRC POSITION

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

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

LICENSEE ACTION

Consumers Power Company submittal dated February 16, 1981 (updated July 1, 1981 per NRC request), provided our request for Technical Specifications changes related to Administrative Controls in which STA requirements were included.

Consumers Power Company letter, dated February 18, 1981, provided a summary of the STA Long-Term Training Program.

DEVIATIONS FROM AND BASIS FOR

Recommendations

In response to NRC Generic Letter 81-10, Consumers Power Company letter, dated June 22, 1981, explained that the inclusion of a Shift Technical Advisor among the on-shift complement of seven people is provisional. The decision to maintain this position will be made upon completion of a PRA-based assessment of emergency staffing levels.

Schedule

An assessment of the role of Shift Technical Advisor, before and during an accident event, and the resulting determination of whether an on-duty STA is necessary, will be submitted by March 31, 1982.

REFERENCES

1. Letter from D G Eisenhut, NRC, to All Licensees of Operating Plants and Holders of Construction Permits (Generic Letter 81-10), dated February 18, 1981

2. Letter from G C Withrow, CP Co, to D M Crutchfield, NRC, dated February 18, 1981

3. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated February 16, 1981

4. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated June 22, 1981

NRC POSITION

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

LICENSEE ACTION

Based on a PRA study, past operating experience at Big Rock Point, and the plant simplicity, the on-shift personnel needs at the onset of an emergency condition were determined. As a result, the shift manning level was defined as follows in Consumers Power Company letter dated June 22, 1981 and corrected by our letter dated July 7, 1981:

1 Shift Supervisor (SRO) - Notification/Communication*

2 Control Room Operators (RO)

2 Auxiliary Operators (AO)

1 Shift Technical Advisor**

1 HP - Rad/Chem Technician***

7 Minimum Total

Clarification Notes:

*Notification/Communication is the site emergency director's responsibility. Initially, the Shift Supervisor (SS) is the designated Site Emergency Director. In this capacity, the SS may delegate the Notification/Communication function to a shift member who is not immediately involved in mitigating the emergency event in progress. Once the Plant Technical Engineer arrives, he assumes all further responsibilities for Notification/Communication. Please note that this description of responsibilities is in accordance with the Big Rock Point Emergency Plan and is a change from the description of shift manning responsibilities delineated in Consumers Power Company letters dated June 22, 1981 and July 3, 1981.

**The inclusion of a Shift Technical Advisor among the on-shift complement of seven people is provisional. The decision to maintain this position will be made upon completion of an assessment of emergency staffing levels.

***Presently, an on-shift HP Technician is not provided. Current practice involves "call up" of required personnel who can augment the on-shift staff within 60 minutes. Consumers Power Company intends to provide an on-shift HP Rad/Chem Technician by November 1, 1982. The basis for providing only one technician is that the Health Physics and Chemistry Technician functions have been traditionally combined and this is reflected in our Site Emergency Plan Implementing Procedures. It should be noted that the actions required of this technician for emergency situations are quite limited in the first few hours of Emergency Plan implementation.

DEVIATIONS FROM AND BASIS FOR

Recommendations

Contrary to the statement made in response to Item I.A.1.3 in Consumers Power Company's submittal dated December 19, 1981, a second on-shift SRO is not included in the shift complement described above. The basis for providing only one SRO on-shift involves the plant simplicity, past operating experience, and the PRA study which did not identify any need for a second on-shift SRO at the plant. Further, the cost-benefit analysis, using criteria derived from the decision rules published in NUREG-0739, showed that the maximum expenditure which can be justified to completely eliminate the residual risk from Big Rock Point is approximately $33,500/year. This figure includes the costs associated with implementation of design changes recommended by Consumers Power Company and on-site occupational exposure. The estimated cost of adding a second SRO on-shift is $240,000/year. Since it is clear that the addition of a second SRO on-shift would not eliminate all residual risk, such action is not considered to be cost effective based on NUREG-0739 criteria.

The basis for providing only one technician to handle both HP and rad/chem duties is that the health physics and chemistry technician functions have been traditionally combined, as reflected in Consumers Power Company's Site Emergency Plan Implementing Procedures. Furthermore, the actions required of this technician for emergency situations are quite limited in the first few hours of Emergency Plan implementation.

Schedule

The addition of an HP Rad/Chem Technician to the on-shift staff will be accomplished by November 1, 1982. Therefore, Consumers Power Company will not meet the July 1, 1982 date, given in NRC Generic Letter 81-10, by which all staffing deficiencies must be removed.

REFERENCES

1. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated June 22, 1981

2. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated December 19, 1980

3. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated March 31, 1981

4. Letter from D G Eisenhut, NRC, to All licensees of Operating Plants and Holders of Construction Permits (Generic Letter 81-10), dated February 18, 1981

5. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated July 7, 1981

I.D.1 CONTROL ROOM DESIGN REVIEWS

NRC POSITION

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

LICENSEE ACTION

Although the final NRC criteria and recommendations concerning Control Room design have not yet been issued (NUREG-0700), Consumers Power Company, with the assistance of a consultant, will begin to evaluate the design of the Big Rock Point Control Room in 1982. A mock-up of the BRP control room will be constructed and an environmental review will be started as part of this effort. In addition, "walk-throughs" of operating and emergency procedures will be conducted in order to identify areas where improvements can be made.

A final report providing the results of the efforts described above will be submitted Since the 1982 refueling outage is scheduled for January 1982, Control Room design modifications recommended by this review, if any, will most likely be implemented during the 1983 refueling outage.

DEVIATIONS FROM AND BASIS FOR

Recommendations

None Expected.

Schedule

None Expected.

REFERENCES

NUREG-0700 (To Be Issued)

I.D.2 PLANT SAFETY PARAMETER DISPLAY CONSOLE

NRC POSITION

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

LICENSEE ACTION

This requirement will be evaluated as past of the Control Room design review described in the LICENSEE ACTION section for Item I.D.1. Consumers Power Company's final recommendation concerning the Safety Parameter Display System (SPDS) will be included in the BRP Control Room design review report, scheduled to be submitted in 1982 (see LICENSEE ACTION for Item I.D.1). Nevertheless, because of the small size of the Big Rock Point Control Room, the simplicity of system designs, and the ready availability of key parameter indications to the operators, a separate SPDS may be neither necessary nor desirable.

DEVIATIONS FROM AND BASIS FOR

Recommendations

To be determined.

Schedule

To be determined.

REFERENCES

None.

II.B.1 REACTOR COOLANT SYSTEM VENTS

NRC POSITION

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

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

1. Submit a description of the design, location, size and power supply for the vent system along with results of analyses for Loss of Coolant Accidents initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10 CFR 50.46.

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

LICENSEE ACTION

The design of the reactor coolant system vents has been modified to include drain and flush connections. Installation of these modified vents is continuing and should be completed by the end of the 1982 refueling outage, scheduled to begin January 1982.

DEVIATIONS FROM AND BASIS FOR

Recommendations

None.

Schedule

An updated design description, test procedures, operating procedures, including supporting analysis, and appropriate Technical Specifications change requests, will be provided by March 1, 1982, rather than the July 1, 1981 date specified in Consumers Power Company submittal of December 19, 1980. Installation of the modified reactor coolant system vents will also be completed by March 1, 1982.

REFERENCES

None.

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

NRC POSITION

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

LICENSEE ACTION

Issues involving design review of plant shielding were addressed in Attachment 2 of Consumers Power Company PRA submittal letter dated March 31, 1981. This response also included the following list of actions to be completed as part of the Continuing Risk Management Program:

1. Evaluate potential problems associated with site ingress/egress following an accident involving serious core damage, and propose solutions which are appropriate to deal with this issue.

2. Evaluate the potential need for additional local shielding as part of the detailed design of modifications which are being considered for implementation at Big Rock Point.

3. In the course of refining emergency operating procedures to be consistent with the results of the PRA, consider in much greater detail the possible need for additional operator information or plant design modifications to deal with potential habitability problems arising from high radiation fields. This particular action will be accomplished by the development and analysis of operator action event trees discussed in detail in Consumers Power Company letter dated June 22, 1981.

In addition, dose calculations reported in the PRA results are presently being reviewed for accuracy and completeness. The results of recently completed plant and plant-site walk-downs, new approaches to calculating inhalation doses, and the effects of airborne contamination on activities outside the

II.B.2 DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POST-ACCIDENT OPERATIONS (Contd)

plant buildings are being considered for incorporation into the PRA analysis. Based on the conclusions of these investigations and reviews, additional shielding requirements, if any, will be identified.

DEVIATIONS FROM AND BASIS FOR

Recommendations

None.

Schedule

While shielding review carried out as part of the Continuing Risk Management Program will be an ongoing effort, the results of the actions and evaluations described above will be completed and submitted in September 1981. An implementation schedule will be developed, if necessary, after the scope of the proposed modifications is finalized.

REFERENCES

1. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated March 31, 1981

2. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated June 22, 1981

II.B.3 POST-ACCIDENT SAMPLING CAPABILITY

NRC POSITION

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

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

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

LICENSEE ACTION

Consumers Power Company's PRA submittal letter, dated March 31, 1981, provided a detailed justification for continued deferral of the post-accident sampling capability requirement. To summarize, Consumers Power Company concluded that given the results of the PRA study and the BRP plant features, the operators and emergency director have the necessary information to be able to:

1. Determine the extent of core damage,
2. Take the appropriate action for accident recovery, and
3. Obtain the information needed to accomplish 1. and 2. above without exposing operators to doses in excess of the maximum values given in NUREG-0737 Item II.B.3.

DEVIATIONS FROM AND BASIS FOR

Recommendations

Consumers Power Company believes that the present capabilities of the plant coupled with the site emergency plan procedures meet the intent of the NUREG-0737 requirements. Therefore, Consumers Power Company has determined that there is no need for the specific post-accident sampling capability prescribed in Item II.B.3.

Schedule

None

REFERENCES

Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated March 31, 1981.

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

NRC POSITION

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

LICENSEE ACTION

The steam drum safety valve test program described in Consumers Power Company's response to NUREG-0737 for the Big Rock Point Plant, submitted December 19, 1980, has been modified in the manner described below.

Consumers Power Company has decided to conduct a safety valve test program in cooperation with Dairyland Power Cooperative, which operates the LaCrosse BWR (LACBWR), since the BRP safety valves and the LACBWR safety valves are nearly identical. A single LACBWR safety valve was selected as the test valve since the problems associated with decontaminating and shipping a contaminated valve preclude the testing of more than one valve. The LACBWR safety valve set point is 1405 psia compared to the BRP safety valve set point of ~ 1565 psia. Nevertheless, the results of the test will be valid for the BRP safety valve since Consumers Power Company has determined that the valve response is unaffected by the slightly higher pressure.

The safety valve test was conducted by Wyle Laboratories, at their Huntsville, Alabama facility, on June 25, 1981. Test conditions and measured variables are given on Table II.D.1-1. Tests with pressure ramp rates up to ~ 30 psig/sec were performed in order to envelope expected ATWS conditions and maximize the dynamic forces experienced by the safety valve, as described in Consumers Power Company's response to NUREG-0737 dated December 19, 1980.

A review was conducted of accidents and anticipated operational transients listed in Regulatory Guide 1.70, Rev 2, to determine expected safety and relief valve operating conditions. The results of this review are summarized in Table II.D.1-2. Based on this review, it was concluded that testing of a steam drum safety valve with steam at high pressure and full flow was necessary to verify valve operability (ie, the ability to open and reclose under expected flow conditions), as required by NUREG-0737, Item II.D.1. This conclusion is consistent with the findings of the Big Rock Point PRA that stuck-open steam drum safety valves contribute significantly to the overall risk.

Flow of liquid through an open safety valve was determined to be less likely than flow of steam. Using methods consistent with those used in the PRA study, the frequency of liquid flow through an operating safety valve was found to be less than 2×10^{-6}/year (see Attachment A). This low frequency is due to the availability of multiple alarms to warn the operator of high steam drum water level and the various means available to an operator for terminating or controlling feedwater flow to the steam drum.

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

The tests conducted at Wyle Laboratories were witnessed by technical representatives from Wyle Laboratories, Dairyland Power Cooperative, Consumers Power Company, and Crosby Valve Company. The preliminary results of the full-flow tests indicate the valves operated satisfactorily. Minor adjustments were made to valve nozzle ring and guide ring settings to obtain optimum performance. A final summary report providing the test results, confirmation that the results are valid for the BRP safety valves, and recommendations concerning the safety valves, if any, will be submitted by October 1, 1981.

In addition to the steam drum safety valves, Big Rock Point utilizes pilot-operated-relief valves as part of its emergency core cooling system. Specifically, these valves are part of the Reactor Depressurization System (RDS) and are not used for overpressure protection. Justification for not testing these valves is provided in Attachment B, which is a description of BRP safety and relief valve systems.

DEVIATIONS FROM AND BASIS FOR

Recommendations

None.

Schedule

A report providing the results of the safety valve tests, confirmation that these results are indeed valid for the BRP safety valves, and recommendations concerning the safety valves, if any, will be submitted by October 1, 1981.

REFERENCES

1. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated December 19, 1980.

2. Letter from D P Hoffman, CP Co to D M Crutchfield, NRC, dated March 31, 1981 (PRA submittal).

TABLE II.D.1-1

SAFETY VALVE TEST CONDITIONS

Test	Initial Conditions				Transient Conditions			
	Fluid	Temperature	Set Point Pressure	Pressure Ramp Rate	Peak Pressure	Reseat Pressure	Transient Description	Back Pressure
1	Steam	587°F	1405 psia	30 psi/sec	1545 psia max	1363 psia	ATWS or Isolation with Emergency Condenser Failure	0-2 psig
2	Steam	587°F	1405 psia	1 psi/sec	1432 psia max	1363 psia	Isolation with Emergency Condenser Makeup Failure	0-2 psig
3	Steam	587°F	1405 psia	16 psi/sec	1545 psia max	1363 psia	LACBWR MSIV Closure with 3 of 4 Scram Failures Plus Emergency Condenser Failure	0-2 psig

Measured Variables: Nozzle Ring & Guide Ring Settings
Valve Stem Lift
Steam Flow, lbm/hr
Valve Body Temperatures, °F
Set Point Pressure, psia
Peak Pressure, psia
Reset Pressure, psia
Ramp Rate, psi/sec

TABLE II.D.1-2

SAFETY VALVE OPERATION DURING ACCIDENTS AND ANTICIPATED OPERATIONAL TRANSIENTS

Event	Frequency of Safety Valve Operation	Fluid Conditions
1. Increase in Heat Removal by the Secondary System		
1.1 Decrease in Feedwater Temperature	(1)	Steam @ 1600 psia
1.2 Increase in Feedwater Flow	< 2×10^{-6} (2)	Water @ 1600 psia
1.3 Increase in Steam Flow	(1) (3)	Steam @ 1600 psia
2. Decrease in Heat Removal by the Secondary System		
2.1 Loss of External Load	(1)	Steam @ 1600 psia
2.2 Turbine Trip	(1)	Steam @ 1600 psia
2.3 Loss of Condensor Vacuum	(1)	Steam @ 1600 psia
2.4 Closure of Main Steam Isolation Valve	(1)	Steam @ 1600 psia
2.5 Steam Pressure Regulator Failure (Closed)	(1)	Steam @ 1600 psia
2.6 Loss of Normal Feedwater Flow	(1)	Steam @ 1600 psia
2.7 Loss of AC Power	(1)	Steam @ 1600 psia
3. Decrease in Reactor Coolant System Flow Rate		
3.1 Single and Multiple Reactor Coolant Pump Trip	(1)	Steam @ 1600 psia
3.2 Reactor Coolant Pump Shaft Seizure	(1)	Steam @ 1600 psia
3.3 Reactor Coolant Pump Shaft Break	(1)	Steam @ 1600 psia
4. Reactivity and Power Distribution Anomalies		
4.1 Control Rod Withdrawal	(1)	Steam @ 1600 psia
4.2 Control Rod Misoperation	(1)	Steam @ 1600 psia
4.3 Start-Up of an Inactive Recirculating Loop at Incorrect Temperature	(1)	Steam @ 1600 psia
4.4 Spectrum of Rod Drop Accidents	(1)	Steam @ 1600 psia
5. Decrease in Reactor Coolant Inventory		
5.1 Spectrum of Steam Breaks Outside Containment	(1) (3)	Steam @ 1600 psia
5.2 Spectrum of Loss of Coolant Accidents Inside Containment	(1) (4)	Steam @ 1600 psia
6. Anticipated Transients Without Scram		
6.1 Loss of Feedwater	(1)	Steam @ 1600 psia
6.2 Loss of AC Power	(1)	Steam @ 1600 psia
6.3 Turbine Trip Without Bypass	(1)	Steam @ 1600 psia

TABLE II.D.1-2 SAFETY VALVE OPERATION DURING ACCIDENTS AND ANTICIPATED OPERATIONAL TRANSIENTS (Contd)

(1) With either the main or emergency condensor available as a heat sink, safety valve operation will not occur. The Big Rock Point Probabilistic Risk Assessment (PRA) estimated the frequency of safety valve operation under transient and accident conditions to be $\sim 4.5 \times 10^{-3}$/yr. Except for events involving excessive coolant addition, liquid or two-phase flow through the safety valves cannot occur at Big Rock Point.

(2) The frequency of excessive coolant addition transients resulting in liquid relief through the safety valves is calculated to be less than 2×10^{-6}/yr using methods consistent with those used in the Big Rock Point PRA.

(3) Transients involving excessive steam flow can result in liquid carry-over into the main steam line. Subsequent main steam line isolation and repressurization during such events (ie, pressure regulator failures and steam line breaks outside containment) can lead to safety valve operation while water is trapped in the main steam line. Nevertheless, the liquid will not be relieved through the safety valves since they are located on top of the primary steam drum which is well above the normal primary coolant system water level and approximately 30 feet above the elevation of the MSIV.

(4) If a break is small or can be isolated, and if both emergency condensor tube bundles are unavailable, the operator would need to decrease the ECCS flow rate in order to prevent the overfilling of the primary coolant and the consequent possibility of liquid relief due to repressurization. At the maximum possible core spray injection flow rate of ~ 600 gpm at 75 psig, it would take more than 14 minutes to completely fill the primary steam drum. The likelihood of the operator failing to reduce the ECCS flow rate coincident with a small LOCA is about two orders of magnitude less than the likelihood of an excessive feedwater event involving liquid relief through the primary safety valves. (The initiator frequencies are 10^{-3}/yr for a small LOCA and 10^{-1}/yr for feedwater controller malfunction - maximum demand.)

II.D.1 ATTACHMENT A - EXCESSIVE FEEDWATER EVENT TREE

The event considered here is any fault or failure in the main feedwater system or steam drum level control system which results in excessive feedwater flow. Like other plant transient events, this initiating event will eventually lead to reactor and turbine trip and subsequent demands on plant shutdown systems. Thus, this event was considered under the class of initiating events entitled turbine trip in the Big Rock Point Probabilistic Risk Assessment. Because the possibility exists for safety valve operation under liquid or two-phase flow conditions in this type of event (note that the BRP safety relief valves were not designed for liquid relief), the potential for a LOCA due to a stuck open relief valve is higher for this transient initiator than for other transient events.

In order to evaluate the importance of the excessive feedwater event as a possible LOCA initiator, an event tree (Figure II.D.1-1) was constructed. The event tree is very similar, and the potential consequences are nearly identical, to the event tree for small steam line break inside containment. A description of each branch follows.

Event Tree Description

Branch Point 1: At branch point 1, excessive feedwater is being delivered to the primary steam drum and the operator either manually controls feedwater or fails to do so. The operator would first become aware of the excessive feedwater addition when steam drum water level reached the high level annunciator set point (+ 4"). Failure of the high-level annunciator is backed up by a completely diverse annunciator on high-high water level (+ 16"). The high water level alarm utilizes the same level instruments as the feedwater regulating system and, therefore, may be susceptible to the same failure that caused the event. However, the high-high water level alarm is provided by four separate annunciators which utilize the steam drum level instruments associated with the Reactor Depressurization System (RDS). The RDS level instruments are independent and completely diverse from the feedwater regulating system level instruments. Failure to control feedwater at branch point 1 leads to reactor protection system operation.

Branch Point 2: Failure to control feedwater at this point will lead to reactor scram on either high neutron flux or high primary system pressure. Success at this point leads to successful core shutdown. Failure to shut down the reactor at this point, by either operator or reactor protection system action, leads to an ATWS condition, which is addressed separately in the BRP Probabilistic Risk Assessment Study.

Branch Point 3: With feedwater delivery to the steam drum continuing at a high rate, the primary system will fill until one or more of the primary safety valves opens to control pressure at 1600 psia. Consumers Power Company has estimated that between one and two of the six safety valves are required to operate in order to relieve liquid from the primary system at the same rate as the maximum delivery rate of feedwater to the steam drum. Therefore, failure of four or more of the safety valves to open will lead to vessel rupture and core damage.

Branch Point 4: With the turbine tripped, the feedwater pumps will eventually trip due to loss of hot well inventory unless the operator intentionally trips the pumps first. In either case, the safety valves will be called on to close. With the valves closed, the operator may proceed to cold shutdown using available plant systems. This sequence is included in the turbine trip event tree. Failure of any one valve to close results in a small steam line break.

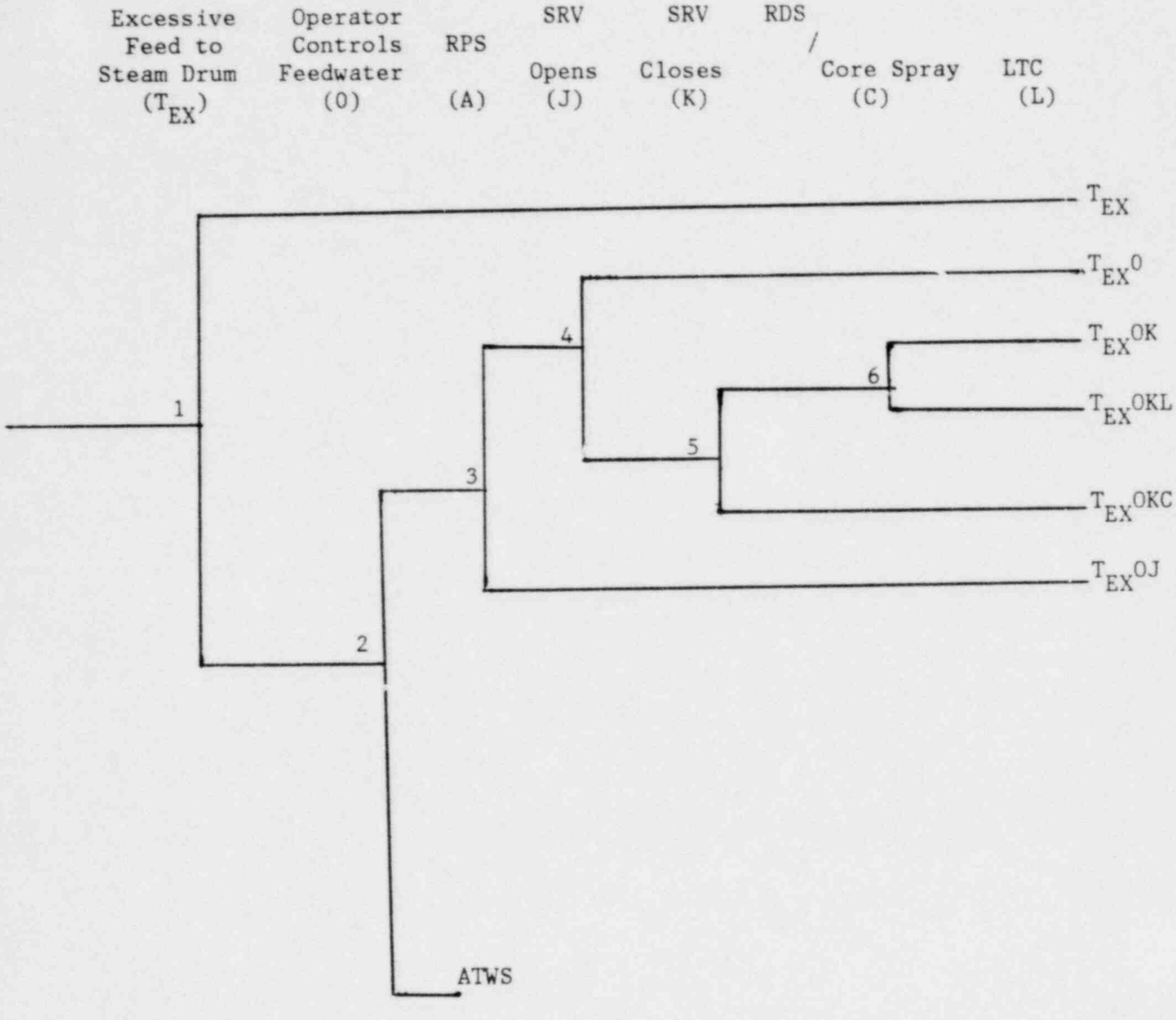
Branch Point 5: With a break in the primary coolant system boundary, it is necessary to provide adequate inventory makeup to keep the core covered. The Reactor Depressurization System (RDS) and core spray systems provide this function. The RDS causes rapid system blowdown when low level is reached in the reactor. This permits delivery of water for core cooling and inventory maintenance from the low-pressure core spray system. Failure of RDS or core spray leads to core damage.

Branch Point 6: At this point, fluid from the primary system is continually being dumped into the containment and the core is being cooled by RDS/core spray. If containment water level rises to the 587-foot elevation, the operator is instructed to switch to the recirculation mode of the core spray system to assure containment and core integrity. Failure to switch to recirculation may lead to containment structural failure due to overfilling with water.

Summary - Excessive Feedwater Event

There are no significant core damage sequences for this event as all sequence frequencies are less than 10^{-7}/yr. This is because of the low expected frequency of liquid relief through the primary safety valves during power operation (1.4×10^{-6}/yr).

FIGURE II.D.1-1



Insignificant Sequences

Sequence	Frequency of Core Damage (yr^{-1})
$T_{EX}OJ$	3×10^{-9} Feedwater controller - maximum demand 10^{-1} Operator fails to respond to high level alarm assuming two control operators and moderate dependence (O) $.14 \times 10^{-4}$ Safety valves fail to open (J) 2.2×10^{-3}
$T_{EX}OKL$	1.5×10^{-8} Safety valve fails to close (K) 1.0 Failure of Post Incident System assuming one-month mission time 1.1×10^{-2}
$T_{EX}OKC$	5.6×10^{-9} Failure of RDS/Core Spray 4×10^{-3}

II.D.1 ATTACHMENT B - SYSTEM DESCRIPTION: SAFETY AND RELIEF VALVES AT BIG ROCK POINT

Safety Relief Valves

Overpressure protection of the Big Rock Point primary coolant system is provided by six spring-operated Crosby safety valves (Model No 3-M2-6 HC-75). Each valve has a rated capacity of approximately 310,000 lbm/hr at 1600 psia. Thus, the total relief capacity of the six valves is nearly 200% of rated plant steam flow. The safety valves are only used for overpressure protection at Big Rock Point. Relief valve operability (ie, the ability to close following operation) is important so as to minimize the likelihood of a loss of coolant accident. The frequency of safety relief valve operation at Big Rock Point was estimated at 4.5×10^{-3}/yr by the BRP Probabilistic Risk Assessment for the plant as currently constructed. Core damage frequency due to a stuck open safety relief valve was estimated in the PRA at 2.9×10^{-4}/yr.

Reactor Depressurization System (RDS)

In the event of a small loss of coolant accident, reactor depressurization to permit operation of Big Rock Point's low-pressure core spray system is provided by four separate blowdown paths. Each Reactor Depressurization System (RDS) path consists of an independent pilot-operated-relief valve in series with an air-operated isolation valve. The pilot-operated valves are two-stage 6" x 10" Target Rock relief valves (Model No 73V-001) and the isolation valves are 6" gate valves. The RDS is only actuated on coincident signals of low reactor water level, low primary steam drum level (after a two-minute delay), and high fire header (core spray water source) pressure. The system may be manually actuated as well. The RDS discharges reactor steam to the reactor enclosure. The Big Rock Point reactor enclosure is a dry spherical containment building which is designed to withstand 27 psig internal pressure. Big Rock Point has no suppression pool. A detailed description of the RDS can be found in the letter from R B Sewell dated August 15, 1974.

The Reactor Depressurization System is not used for overpressure protection of the Big Rock Point primary coolant system. The RDS is only operated during situations involving significant loss of primary coolant so that the blowdown paths are only intended to pass steam. Because the system emits reactor steam to the reactor enclosure which is frequently occupied and where many control and safety-related plant systems are located, the RDS would only be used when absolutely necessary.

The upgraded Emergency Procedure Guidelines (EPG) being developed by the BWR Owners Group in response to NUREG-0737, Item I.C.1, define two emergency conditions under which it may be desirable to intentionally overfill the primary system and, thereby, force liquid through the RDS blowdown paths. The first condition involves implementation of the "alternate mode of shutdown cooling," in the event of a loss of the normal shutdown cooling system. Under the alternate mode of shutdown cooling, the operator is instructed to fill the primary system using the low-pressure injection system and open one or more relief valves to provide a flow path back to the suppression pool. By this means, cold shutdown conditions (< 212°F) can be achieved. The second condition defined by the EPG involves loss of all primary system level instrumentation. In this situation, the operator is also instructed to overfill the primary coolant system.

The operator response prescribed in the generic EPG to the first condition (ie, the loss of the normal shutdown cooling system) is not considered applicable to Big Rock Point with its dry containment (ie, no suppression pool). At Big Rock Point, the preferred response to such an event would be to first continue to remove heat using either the main or emergency condenser; and second, steam through the RDS valves while providing necessary makeup with the core spray system. No advantages to implementing the alternate mode of shutdown cooling at Big Rock Point have been identified in this case.

The operator response to the second condition (ie, loss of level indication) should be to continue to provide makeup water to the primary system by all available means and to switch to core spray recirculation once the containment water level has reached the necessary level (587-feet elevation above sea level). Under this condition, liquid relief through the RDS blowdown paths may occur. Nevertheless, the integrity of the blowdown paths and operability of the valves (ie, their ability to reclose) is not important in this highly unlikely event because the core is completely covered and, thus, core cooling is assured.

Thus, Consumers Power Company's position is that testing is not required of the Big Rock Point Reactor Depressurization System relief valves under liquid conditions at low pressure, as is currently proposed by the BWR Owners Group for the BWR/2 through BWR/6 classes of plants.

II.E.4.2 CONTAINMENT ISOLATION DEPENDABILITY

NRC POSITION

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

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

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

4. The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

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

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

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

LICENSEE ACTION

The following actions concerning containment isolation dependability were taken by Consumers Power Company following our NUREG-0737 response submitted on December 19, 1980:

1. Consumers Power Company letter, dated March 23, 1981, submitted a Technical Specifications change request to lower the pressure set point for containment isolation from 1.5 psig to $\leq$ 1.0 psig.

2. Consumers Power Company letter, dated May 26, 1981, provided qualification information for the Atwood-Morril swing-check containment isolation valves.

With the submittal of these two letters, Consumers Power Company completed its response to NUREG-0737, Item II.E.4.2, requirements.

DEVIATIONS FROM AND BASIS FOR

Recommendations

None.

Schedule

None.

REFERENCES

1. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated December 19, 1980.

2. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated March 23, 1981.

3. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated May 26, 1981.

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

NRC POSITION

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

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

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

LICENSEE ACTION

Consumers Power Company will install a noble gas effluent monitor on the plant stack by the end of the 1982 refueling outage, scheduled to begin January 1982. The appropriate Technical Specifications change request will be submitted by September 15, 1981.

DEVIATIONS FROM AND BASIS FOR

Recommendations

None.

Schedule

No deviations are expected. Nevertheless, if unanticipated delays are encountered in vendor delivery of equipment, the installation date given above could be jeopardized.

REFERENCES

Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated December 19, 1980.

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

NRC POSITION

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

LICENSEE ACTION

Consumers Power Company will install the required on-line sampling and analysis capability by the end of the 1982 refueling outage, scheduled to begin January 1982. The appropriate Technical Specifications change request will be submitted by September 15, 1981.

DEVIATIONS FROM AND BASIS FOR

Recommendations

None.

Schedule

No deviations are expected. Nevertheless, if unanticipated delays are encountered in vendor delivery of equipment, the installation date given above could be jeopardized.

REFERENCES

Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated December 19, 1980

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

NRC POSITION

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

LICENSEE ACTION

The exterior containment high-range radiation monitors, described in Consumers Power Company letter dated December 19, 1980, will be installed by the end of the 1982 refueling outage, scheduled to begin January 1, 1982. Design details of the monitor are available at Consumers Power Company for inspection and review as of July 1, 1981. The appropriate Technical Specifications change request will be submitted by September 15, 1981.

DEVIATIONS FROM AND BASIS FOR

Recommendations

In addition to the deviations specified in Consumers Power Company letter dated December 19, 1980, we note one other deviation from the guidance given in NUREG-0737. In-situ calibration of the high-range radiation monitor for one decade below 10 R/h by means of a calibrated radiation source is not provided. Since a large source, on the order of 20 curies cesium-137, would have been needed to meet this requirement, Consumers Power Company decided to investigate alternative calibration methods in the interest of ALARA and personnel safety considerations.

At the request of Consumers Power Company, the radiation monitor supplier has proposed a modification to their standard area monitor field calibration kit. The modified calibration kit will have installed a 400 millicurie cesium-137 source that will enable an operator to confirm operability of the containment high-range radiation monitors in the range of one decade below 10 R/h. For all ranges above 10 R/h, calibration will be performed by electronic signal substitution. This calibration will be carried out from a display panel located in the Control Room.

Schedule

The containment high-range radiation monitor and associated equipment will be installed by the end of the 1982 refueling outage, scheduled to begin January 1982. The appropriate Technical Specifications change request will be submitted by September 15, 1981.

REFERENCES

Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated December 19, 1980.

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

NRC POSITION

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

LICENSEE ACTION

As committed to in Consumers Power Company letter dated December 19, 1980, modification of the existing containment water level monitoring systems to provide recordings of the measurements will be completed by the end of the 1982 refueling outage, scheduled to begin January 1982. A redundant continuous level measurement system will also be installed by the same date. The appropriate Technical Specifications change request will be submitted by the end of the 1982 refueling outage.

DEVIATIONS FROM AND BASIS FOR

Recommendations

None.

Schedule

No deviations are expected. Nevertheless, if unanticipated delays are encountered in vendor delivery of equipment, the installation and modification dates given above could be jeopardized.

REFERENCES

Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated December 19, 1980.

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

NRC POSITION

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

LICENSEE ACTION

Replacement of the existing narrow-range reactor vessel level instrumentation with a design of improved reliability, committed to in Consumers Power Company letter dated December 19, 1980, has been accomplished. The commitment to consider the need for additional level instrumentation, also contained in Consumers Power Company letter dated December 19, 1980, was carried out.

The results of this consideration, provided in Consumers Power Company letter dated March 31, 1981, led to the conclusion that additional level instrumentation (ie, wide-range) will not significantly improve the safety of Big Rock Point. Nevertheless, Consumers Power Company, in its letter of March 31, 1981, did commit to continue evaluating the usefulness of wide-range level instrumentation as part of the Continuing Risk Management Program. By developing and analyzing operator action event trees (described in more detail in the Consumers Power Company letter of June 22, 1981), we intend to submit our final evaluation of wide-range level instrumentation by July 31, 1981.

DEVIATION FROM AND BASIS FOR

Recommendations

None.

Schedule

If the PRA-based analysis described above indicates that wide-range level instrumentation does indeed provide useful information to an operator, Consumers Power Company will submit a proposed plan of action by September 1, 1981.

REFERENCES

1. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated December 19, 1980

2. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated March 31, 1981

3. Letter from D P Hoffman,CP Co, to D M Crutchfield, NRC, dated June 22, 1981

II.K.3.14 ISOLATION OF ISOLATION CONDENSERS ON HIGH RADIATION

NRC POSITION

Isolation condensers have radiation monitors on their vents. These monitors provide alarms in the control room but do not isolate the isolation condenser. The isolation condensers are currently isolated on a high-radiation signal in the steam line leading to the isolation condensers. The design should be modified such that the isolation condensers are automatically isolated upon receipt of a high-radiation signal at the vent rather than at the steam line. The purpose of the change is to increase the availability of the isolation condensers as heat sinks.

LICENSEE ACTION

As stated in Consumers Power Company letter dated December 19, 1980, a determination of the necessity to modify the emergency condenser was carried out as part of the PRA study.

The PRA-based conclusion concerning the emergency condenser, provided by Consumers Power Company letter dated March 31, 1981, is summarized as follows. The addition of an automatic system to isolate the emergency condenser tube bundles appears to contribute little to the reduction of risk resulting from emergency condenser tube bundle leakage. Conversely, such a system could have a potentially detrimental effect on the reliability of the emergency condenser as a heat sink. Equipment and procedures designed to deal with ruptured tubes are already in place, and monthly surveillance of the emergency condenser shell inventory is performed to detect tube degradation as it develops.

Consumers Power Company also proposed, in our letter of March 31, 1981, to 1) examine methods to reduce the probability of false indication of tube rupture due to shine from the emergency condenser tube bundles, and 2) investigate additional assurances that fatigue is not a significant contributor to the likelihood of tube bundle failure. Currently, the status of these actions is as follows:

1. A proposed design for the installation of additional shielding around the emergency condenser vent monitor is being evaluated. While extra shielding around the vent monitor would reduce the probability of false indication of tube rupture, other possible modifications are also being considered. At this time, no definite date can be given for the completion of this effort.

2. Investigation of additional assurances that fatigue is not a significant contributor to the likelihood of tube bundle failure is continuing. The results of this effort will be reported in the PRA update, scheduled to be completed by December 31, 1981.

DEVIATIONS FROM AND BASIS FOR

Recommendations

As a result of the PRA study, Consumers Power Company has determined that the risk resulting from emergency condenser tube leaks does not warrant the installation of an automatic system of the type recommended by NUREG-0737 Item II.K.3.14. Consumers Power Company is continuing its evaluation of 1) methods to reduce the probability of false indication of tube rupture due to shine from the emergency condenser tube bundles, and 2) additional assurances that fatigue is not a significant contributor to the likelihood of tube bundle failure.

Schedule

At this time, no date can be given for the implementation of a means of reducing the shine from the emergency condenser tube bundles to the vent monitor since a specific method has not yet been selected. In the case of tube bundle failure due to fatigue, the determination of whether modifications are necessary, and a corresponding implementation schedule, will be established after December 31, 1981, when the investigation into this issue is completed.

REFERENCES

1. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated December 19, 1980

2. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated March 31, 1981

II.K.3.19 INTERLOCK ON RECIRCULATION PUMP LOOPS

NRC POSITION

Interlocks should be installed on nonjet pump plants (other than Humboldt Bay) to assure that at least two recirculation loops are open for recirculation flow for modes other than cold shutdown. This is to assure that the level measurements in the downcomer region are representative of the level in the core region.

LICENSEE ACTION

Consumers Power Company letter dated December 19, 1980, stated that the determination of need for recirculation pump loop interlocks will be made at the conclusion of the PRA study. The results of the PRA study, provided by Consumers Power Company letter dated March 31, 1981, addressed this requirement by noting that isolation of the BRP recirculation loops does not isolate reactor or steam drum level instrumentation from the levels of the liquid volumes which they are monitoring.

Consumers Power Company letter of March 31, 1981 also discussed events involving the isolation of recirculation loops which could lead to core uncovery. Quantification of the event tree for isolation of the BRP recirculation loops, performed as part of the PRA, does not reveal any transient sequences which are a significant contributor to risk.

DEVIATIONS FROM AND BASIS FOR

Recommendations

On the basis of the PRA results and the BRP design, Consumers Power Company concluded that recirculation loop interlocks provide insignificant improvement to the safe operation of the Big Rock Point Plant. Therefore, no provisions are being made to install such interlocks.

Schedule

None.

REFERENCES

1. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated December 19, 1980

2. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated March 31, 1981

II.K.3.25 EFFECT OF LOSS OF ALTERNATING CURRENT POWER ON PUMP SEALS

NRC POSITION

The licensees should determine, on a plant specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating current (ac) power for at least two hours. Adequacy of the seal design should be demonstrated.

LICENSEE ACTION

As committed to in Consumers Power Company letter dated December 19, 1980, an investigation of the possibility to provide alternating current power to a Reactor Cooling Water (RCW) pump from the emergency diesel generator during non-LOCA events was carried out. Currently, an operator has the option to manually switch the RCW pump power supply to the emergency generator, in a non-LOCA event, if he deems it necessary. Consumers Power Company believes that this flexibility with regard to operator actions would be lost if RCW pumps were automatically provided with power from the emergency diesel generator.

Consumers Power Company also investigated the results of a loss of RCW pump power by participating in an analysis, performed by General Electric Company for the BWR Owners Group. This analysis concerned the effect of loss of cooling water on the life of a nonoperating pump seal. Upon evaluation, Consumers Power Company finds that the specific results of the GE analysis cannot be applied to Big Rock Point because of the differences in the high-pressure makeup systems.

Because the options described above for analyzing or alleviating a loss of RCW pump power event were unsatisfactory, Consumers Power Company has decided to conduct a test of the O-ring seal material at our own facilities. Testing of the O-ring material alone is considered to be adequate for responding to NUREG-0737, Item II.K.3.25 for the following reasons. First, seal leakage can occur in three ways:

1. Via normal controlled paths;
2. Past the Carbon A seal faces;
3. Past one or more O-rings and U-cups, all of which are made of ASTM D-735-55T CL SB-720 BE_1 E_3 elastomer.

Seal leakage via normal controlled paths is desirable because it provides lubrication to seal faces. A test to demonstrate that there is no leakage of primary coolant (580°F maximum temperature) past the Carbon A seal faces which could lead to degradation of the seal integrity is considered by Consumers Power Company to be unnecessary. This is because past experience at Big Rock Point with a loss of station power event, when power was lost for almost one hour, proved that the Carbon A seal face integrity was not degraded by exposure to primary coolant and the resulting temperature transient. Once the transient is terminated, time at temperature (580°F maximum) will also not effect the seal faces since Carbon A is a high-temperature material. In the case of components made of elastomer, such as the O-rings contained in the recirculation pump seals, exposure to primary s·stem temperatures will have some effect.

Consumers Power Company believes that the ability of the O-rings to maintain their sealing capability can be demonstrated based on discussions with experts from the pump manufacturer and the fact that the exposure of elastomers to primary coolant temperatures can result in compression set and loss of physical properties, but not necessarily an increase in porosity or breakup of the O-ring. Nevertheless, Consumers Power Comany has identified an elastomer compound for which there is a test report qualifying this material for exposure to water at 600°F for two hours, 400°F for the next two hours, and 200°F for the final two hours. This temperature history conservatively envelopes the temperature history which the BRP recirculation pump seals would experience during the postulated event. If Consumers Power Company cannot demonstrate that the elastomer compound now used in the pump seals is adequate for use at primary system temperatures, the possibility of replacing the existing O-rings with those made from the elastomer described above, or an equivalent material, will be pursued with the pump manufacturer.

If the elastomer presently in use cannot be qualified, Consumers Power Company anticipates that the elastomer components in the lower pump seal will be changed to the qualified elastomer described above, or an equivalent material, provided the pump manufacturer's concurrence is obtained, during the 1982 refueling outage. This outage is scheduled to begin January 1982.

DEVIATIONS FROM AND BASIS FOR

Recommendations

None.

Schedule

None.

REFERENCES

1. Letter from D P Hoffman, Consumers Power Company, to D M Crutchfield, NRC, dated December 19, 1980

II.K.3.29 PERFORMANCE OF ISOLATION CONDENSERS

NRC POSITION

If natural circulation plays an important role in depressurizing the system (eg, in the use of isolation condensers), then the various modes of two-phase flow natural circulation, including noncondensibles, which may play a significant role in plant response following a small break Loss of Coolant Accident (LOCA) should be demonstrated.

LICENSEE ACTION

See the updated LICENSEE ACTION for Item II.B.1.

DEVIATIONS FROM AND BASIS FOR

Recommendations

As noted in Consumers Power Company letter dated December 19, 1980, this item is not applicable to Big Rock Point, except for the actions to be performed pursuant to Item II.B.1.

Schedule

See the updated Schedule for Item II.B.1.

REFERENCES

None.

III.A.1.2 UPGRADE EMERGENCY SUPPORT FACILITIES

NRC POSITION

Additional clarification will be provided in the near future.

LICENSEE ACTION

The requirements of NUREG-0737 Item III.A.1.2 were subsequently clarified by NUREG-0696, "Functional Criteria for Emergency Response Facilities," and Generic Letter 81-10, "Response to Post-TMI Requirements for the Emergency Operations Facility." Consumers Power Company has responded to these requirements in letters dated March 31, 1981, June 1 1981 and June 22, 1981. In order to complete all actions concerning NUREG-0737 Item III.A.1.2, Consumers Power Company, in a letter dated June 1, 1981, committed to submit the following information in September 1981:

1. Whether or not alternate shutdown panel readouts will be installed in the Control Room.

2. Review, evaluate and recalculate doses to operators and recommend additional shielding if necessary (see updated LICENSEE ACTION for Item II.B.2 for details).

DEVIATIONS FROM AND BASIS FOR

Recommendations

The significant deviation is that the interim TSC will continue to serve as the permanent TSC. Justification for deviations from the requirements of NUREG-0737 Item III.A.1.2, NUREG-0696, and Generic Letter 81-10, is described in the referenced Consumers Power Company letters.

Schedule

The final decision concerning the location of alternate shutdown panel readouts in the Control Room, and the review, evaluation, and recalculation of doses to operators will be submitted in September 1981. A schedule for the implementation of additional shielding, if any, will be developed after the scope of the proposed modifications is finalized.

REFERENCES

1. USNRC NUREG-0696, "Functional Criteria for Emergency Response Facilities," dated February 1981

2. Letter from D G Eisenhut, NRC, to All Licensees of Operating Plants and Holders of Construction Permits" (Generic Letter 81-10), dated February 18, 1981

3. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated March 31, 1981

4. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, Dated June 1, 1981

5. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, Dated June 22, 1981

6. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, Dated June 1, 1981

III.A.2 IMPROVING LICENSEE EMERGENCY PREPAREDNESS - LONG TERM

NRC POSITION

Each nuclear facility shall upgrade its emergency plans to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Specific criteria to meet this requirement are delineated in NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparation in Support of Nuclear Power Plants."

LICENSEE ACTION

The requirement to maintain primary meteorological capabilities, including a meteorological tower, and warning sirens at Big Rock Point was evaluated as part of the Probabilistic Risk Assessment. The results of the PRA, provided by Consumers Power Company letters dated March 31, 1981, do not indicate a need for immediate warning of the public within the Emergency Preparedness Zone (EPZ) for any postulated accident. The meteorological program proposed in Consumers Power Company letters of March 31, 1981, amply provides the information necessary to determine the appropriate protective actions for the public in the event of any postulated accident. This meteorological program provides National Weather Service data through an automated computer system maintained by Weather Services International. This meteorological capability, including appropriate revisions of plant procedures and personnel training, was initiated on April 1, 1981. Therefore, Consumers Power Company's position is that additional meteorological capabilities beyond those proposed in Consumers Power Company letters of March 31, 1981, are unnecessary.

Consumers Power Company is, nevertheless, continuing to investigate supplemental meteorological capabilities which might be necessary in order to assess the exposure of personnel involved in recovery and cleanup activities resulting from postualted accidents, who are ingressing or egressing the plant. These investigations, which will be completed in September 1981, will be evaluated on the basis of ALARA considerations, and additional shielding, if needed, will be recommended (see updated LICENSEE ACTION for Item II.B.2).

Consumers Power Company's response to NUREG-0654 requirements regarding shift manning is provided in our response to NUREG-0737, Item I.A.1.3 (see updated LICENSEE ACTION for Item I.A.1.3).

DEVIATIONS FROM AND BASIS FOR

Recommendations

As described above and in Consumers Power Company letters dated March 31, 1981, the Big Rock Point Plant will use National Weather Service data instead of installing primary meteorological capabilities. In addition, warning sirens will not be installed since the PRA results do not indicate a need for immediate warning of the public within the EPZ for any postulated accident. Consumers Power Company is continuing to investigate meteorological capabilities which might be needed for ingress and egress following a postulated accident.

Deviations from the shift manning requirements are described in the updated Recommendations for Item I.A.1.3.

Schedule

The results of the investigations of meteorological capabilities needed to assess the exposure of personnel ingressing and egressing the plant following a postulated accident will be submitted in September 1981. A schedule for implementing recommendations concerning additional shielding, if any, will be developed after the scope of the proposed modifications is finalized.

See the updated Schedule for Item I.A.1.3 for deviations with respect to shift manning requirements.

REFERENCES

1. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated March 31, 1981

2. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated March 31, 1981

III.D.3.4 CONTROL ROOM HABITABILITY

NRC POSITION

In accordance with Task Action Plan Item III.D.3.4 and control room habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50).

LICENSEE ACTION

The requirements of NUREG-0737 Item III.D.3.4 are addressed in Consumers Power Company's responses to NUREG-0737 III.A.1.2. Specifically, Consumers Power Company's submittal of the PRA, dated March 31, 1981, and subsequent letters dated June 1, 1981 and June 22, 1981, provided a detailed description of the LICENSEE ACTION in response to NUREG-0737 Items III.A.1.2 and III.D.3.4 (see updated LICENSEE ACTION for Item III.A.1.2). More information concerning Control Room habitability will be forthcoming as part of the PRA update report, which will be completed by December 31, 1981.

DEVIATIONS FROM AND BASIS FOR

Recommendations

See updated Recommendations for Item III.A.1.2.

Schedule

See updated Schedule for Item III.A.1.2.

REFERENCES

1. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated March 31, 1981

2. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated June 1, 1981

3. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated June 1, 1981

4. Letter from D P Hoffman, CP Co, to D M Crutchfield, NRC, dated June 22, 1981