PSNH PUBLIC SERVICE
Company of New Hampshire

SEABROOK STATION
Engineering Office:
1671 Worcester Road
Framingham, Massachusetts 01701
(617) - 872 - 8100

February 12, 1982

SBN-212
T.F. B 7.1.7



United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Frank Miraglia, Chief
Licensing Branch #3
Division of Licensing

References:
(a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) USNRC Letter, dated September 30, 1981, "Acceptance Review for Operating Licenses for Seabrook Station, Units 1 and 2," D. G. Eisenhut to W. C. Tallman
(c) PSNH Letter, dated November 27, 1981, "Response to Acceptance Review Requests for Additional Information," J. DeVincentis to D. G. Eisenhut

Subject: Implementation of TMI Action Plan Requirements of NUREG-0737

Dear Sir:

In Reference (b), it was stated that, "...the Seabrook FSAR addresses the requirements contained in NUREG-0737." In addition, RAI 100.2 requested that PSNH, "...identify the FSAR section where details of each applicable TMI Action Item (NUREG-0737) can be found."

Reference (c) indicated that, "Amendment 44 will include a new FSAR Section 1.9 which will provide a statement of our compliance to each applicable item of NUREG-0737," and, "...will also provide a reference to additional locations in the FSAR (if any) where the item is addressed in greater detail."

Based on conversations with Mr. Louis Wheeler (Project Manager), it was mutually agreed that FSAR Section 1.9 would function as a bare reference section only. This letter serves to provide the initial discussion of our compliance with each applicable item of NUREG-0737 (attached) Amendment 45 will include the bare reference section (1.9) and incorporate additional information into the FSAR where appropriate.

Very truly yours,

John DeVincentis

J. DeVincentis
Project Manager

B001
s
1/1

Attachment

Task I.A.1.1 Shift Technical Adviser (NUREG-0737)

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 multi-unit site, if qualified to perform the advisory 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 STA's that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

Response

In the FSAR Chapter 13 Review Meeting with NRC Region I representatives (1/4/82 - 1/6/82), the PSNH position on this item was discussed. This will be formally transmitted to the NRC in the near future.

Task I.A.1.2 Shift Supervisor Administrative Duties (NUREG-0660)

Position

The objective is to increase the shift supervisor's attention to his command function by minimizing ancillary responsibilities. NRR has required that all operating plant licensees review the administrative duties of the shift supervisor. The review should be performed by the senior officer at each utility who is responsible for plant operations. Administrative functions that detract from, or are subordinate to, the management responsibility for assuring the safe operation of the plant are to be delegated to other operations personnel not on duty in the control room. The same requirement will be imposed by the licensing review staff on all operating license applicants.

Response

In the FSAR Chapter 13 Review Meeting with NRC Region I representatives (1/4/82 - 1/6/82), the PSNH position on this item was discussed. This will be formally transmitted to the NRC in the near future.

I.A.1.3 Shift Manning (NUREG-0737)

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 rule-making). Overtime restrictions were also included in July 31, 1980 letter.

Response

Shift manning requirements and overtime restrictions for Seabrook are defined in Technical Specification 6.2, Table 6.2-1.

Task I.A.2.1. Immediate Upgrading of Reactor Operator and Senior Reactor Operator Training and Qualifications (NUREG-0737)

Position

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

Response

All applicants for SRO licenses subsequent to initial criticality will have a minimum of one years experience as a licensed operator. Cold applicants for SRO licenses will receive the training and simulator experience identified in FSAR Section 13.2.

Task I.A.2.3 Administration of Training Programs (NUREG-0737)

Position

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

Response

A fully qualified simulator staff, with previously SRO licensed instructors, will implement and conduct the simulator training program. Permanent and guest classroom instructors not previously SRO licensed will demonstrate the knowledge in that subject required of a SRO. Such knowledge will be documented by the completion of an instructor qualification program for the subject area.

Task I.A.3.1 Revise Scope and Criteria for Licensing Examinations - Simulator Exams (Item 3) (NUREG-0737)

Position

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

Response

Seabrook has a site specific simulator maintained current with Unit 1 design as per ANSI/ANS 3.5-1978. The simulator will be made available for the NRC licensing examinations.

Task I.B.1.2 Independent Safety Engineering Group (NUREG-0737)

Position

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

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

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

Response

In the FSAR Chapter 13 Review Meeting with NRC Region I representatives (1/4/82 - 1/6/82), the PSNH position on this item was discussed. This will be formally transmitted to the NRC in the near future.

Task I.C.1 Guidance for the Evaluation & Development of Procedures for Transients and Accidents (NUREG-0737)

Position

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

Response

This task requires analyses, revised operating procedure guidelines, revised operating procedures, and operator training related to:

- Small-break loss-of-coolant accidents (LOCAs)
- Inadequate core cooling
- Transients and accidents

A summary of Westinghouse activities (in support of the Westinghouse Owners Group) related to these requirements is presented below:

- o Small Break LOCAs - Westinghouse has performed and submitted comprehensive analyses to the NRC covering a large spectrum of small-break LOCAs. The results of these analyses (Reference 1) demonstrated that the models and methods used by Westinghouse to evaluate the safety of the design are conservative, and yet, acceptably realistic.

- o Inadequate Core Cooling - Westinghouse has performed and submitted a series of comprehensive analyses to the NRC covering the subject of inadequate core cooling (References 2 and 3). In addition to providing the times required to attain voiding in the core, the results of this effort gave an indication of the substantial failures of safety equipment that were necessary, thus providing added confidence in the "defense in depth" approach used in nuclear plants.

- o Transients and Accidents - Westinghouse has performed and submitted an analysis to the NRC covering the major design basis accident scenarios (Reference 4). This analysis contains event trees for the accidents and an evaluation of the coverage of the event sequences in the Westinghouse emergency operating instructions.

In light of the TMI incident; results from the above mentioned small-break LOCA analyses, inadequate core cooling analyses, and transient and accident analyses; discussions with the NRC (and subsequent NRC reviews); and inputs from utilities, Westinghouse has reviewed and revised the generic Westinghouse operating procedure guidelines.

The NRC has indicated that additional efforts are necessary in the areas of inadequate core cooling, transient and accidents, and associated emergency procedure guidelines. Owners Group letter OG-61 includes a plan of action to close out Task I.C.1. The action plan detailed specific steps to be taken by the Westinghouse Owners Group and Westinghouse jointly, which are required to completely address the NRC concerns documented in Task I.C.1. The Owners Group and Westinghouse are continuing to pursue implementation of this action plan based on NRC agreement (in principle).

The generic guidelines developed by the WOG will be considerd as appropriate in the development of Seabrook plant specific operating procedures.

Task I.C.2 Shift and Relief Turnover Procedures (NUREG-0660)

Licensees are to revise plant procedures for shift and relief turnover to ensure that each oncoming shift is made aware of critical plant status information and system availability.

Response

Shift and relief turnover procedures will be developed and implemented three months prior to fuel load. These procedures will ensure the oncoming shift is aware of critical plant status information and system availability prior to assuming duty.

Task I.C.3 Shift Supervision Responsibilities (NUREG-0660)

Position

Licensees are to revise plant procedures to assure that duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined.

Response

Plant procedures which detail the duties, responsibilities and authority of the Unit Shift Supervisor and Control Room Operators will be developed and implemented three months prior to fuel load.

Task I.C.4 Control Room Access (NUREG-0660)

Position

Licensees are to revise procedures to assure that instructions covering the authority and responsibilities of the person in charge of access, and clear lines of authority and responsibility in the control room in the event of an emergency, are established.

Response

Procedures detailing the authority and responsibilities of the Unit Shift Supervisor as the person in charge of control room access will be developed and implemented three months prior to fuel load. Procedures will include the identification of clear lines of authority and responsibilities in the control room in the event of an emergency.

Task I.C.5 Procedures for Feedback of Operating Experience to Plant Staff (NUREG-0737)

Position

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

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

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

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

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

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

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

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

Response

A management system will be developed and implemented three months prior to fuel load to perform the following functions:

(1) Review operating experience information orignating both within and outside the facility;

(2) Promptly supply information pertinent to plant safety, including proposed procedural changes and plant modifications, to operators and other appropriate plant personnel; and,

(3) Assure that such information is incorporated into training and requalification programs.

Task I.C.6 Guidance on Procedures for Verifying Correct Performance of Operating Activities (NUREG-0737)

Position

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

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

Response

The Seabrook control board contains a safety systems status inoperable panel which is automatically actuated when major components of a safety system are placed in an abnormal mode or position. In addition, the control room operators will manually activate a safety systems inoperable light as part of the tagging procedure if maintenance or testing of a safety system requires equipment, valves, or switches, which would render the system inoperable, to be placed in an abnormal configuration. Therefore, the control room operators will be able to assess the status of all safety systems by observing this panel.

Task I.C.7 NSSS Vendor Review of Procedures (NUREG-0660)

Position

Operating license applicants are required to obtain reactor vendor review of their low-power, power-ascension, and emergency procedures as a further verification of the adequacy of the procedures.

Response

In meeting the requirements of Item I.C.1, the Westinghouse Owners Group has committed to submit a complete program of revised generic operation guidelines. The generic guidelines developed by the Westinghouse Owners Group will be used in developing Seabrook plant specific emergency operating procedures.

Therefore, as a result of the Westinghouse participation in this effort, no separate NSSS review of emergency operation procedures is deemed necessary.

NSSS review of power-ascension procedures will be accomplished through Joint Test Group subcommittee review of these procedures.

Task I.C.8 Pilot Monitoring of Selected Emergency Procedures for Near Term Operating License Applicants (NUREG-0660)

Position

Licensees will be required to correct any deficiencies identified before full power operation.

Response

Any deficiencies resulting from monitoring of selected emergency procedures will be corrected prior to full power operations.

Task I.D.1 Control Room Design Reviews (NUREG-0737)

Position

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

Response

PSNH will conduct a control room design review to identify design deficiencies. The results of that review will be evaluated, to identify those deficiencies that warrant modification. A preliminary assessment report will be made to the NRC by May 1982. This report will provide an outline of the design review methodology.

Task I.D.2 Plant Safety Parameter Display Console (NUREG-0737)

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 access plant safety status.

Response

Seabrook Station will be equipped with a Safety Parameter Display System. We are presently investigating the commercial offerings available in this area vis-a-vis the existing NRC guidance (NUREG-0696) and compatability with the existing computer system.

The NRC will be kept appraised of our progress in selecting a system.

Task I.G.1 Training Requirements (NUREG-0660)

Position

Licensees will (1) define training plan prior to loading fuel and (2) conduct training prior to full-power operation.

Response

A set of low-power tests to be performed will be identified three months prior to fuel load. However, since Seabrook has a site specific simulator which is maintained current with Unit 1 design as per ANSI/ANS 3.5 - 1979, each operating crew will perform the designated low-power tests on the simulator. Therefore, only the crew on-shift need perform the low-power testing on the actual plant.

Task II.B.1 Reactor Coolant System Vents (NUREG-0737)

Position

Each applicant and licensee shall install reactor coolant system (RCS) and reactor vessel head high point vents, remotely operated from the control room. Although the purpose of the system is to vent non-condensible gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents 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 the 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.

Response

(1) Refer to FSAR Section 5.2.6, Reactor Coolant Vent System.

(2) Procedures for the use of the Reactor Coolant Vent System will be developed three months prior to fuel load.

Task II.B.2 Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-Accident Operations (NUREG-0737)

Position

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

Response

A design review of plant shielding and qualification of equipment is in progress. The impact of the above releases of radioisotopes is being assessed.

Time-integrated radiation doses from contained post-accident sources have been established for all areas outside the containment containing safety-related equipment. The effect of a 50% cesium release is under investigation.

Maximum dose rates have been calculated for most areas outside containment including all vital areas requiring occupancy in the critical period immediately following an accident. The acceptance criterion for the dose received in locations requiring continuous occupancy is 15 mrem/hr. averaged over the first 30 days. For locations requiring infrequent access the maximum acceptable dose will be 5 rem per task.

The task of establishing post accident radiation levels in accordance with NUREG-0737 will be completed by May 1, 1982.

Task II.B.3 Post-Accident Sampling Capability (NUREG-0737)

Position

A design and operational review of the reactor coolant and containment atmosphere sampling line systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18-3/4 rem to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safety 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 radio-nuclides that are indicators of the degree of core damage. Such radio-nuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and non-volatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

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

Response

The shielding and operation of the reactor coolant and containment atmosphere sampling systems has been designed to provide the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure in excess of the limits delineated for this requirement. A post-accident sampling panel has been designed to NUREG-0737. However, additional requirements presented in Regulatory Guide 1.97 are presently being reviewed. Resolution of these additional requirements will be completed by July 1, 1982.

Procedures to obtain post-accident samples and the radiological and chemical analyses will be developed three months prior to fuel load.

Task II.B.4 Training for Mitigating Core Damage (NUREG-0737)

Position

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

Response

A training program to teach the use of equipment and systems to mitigate accidents involving core damage will be developed prior to fuel load and be completed prior to full-power operations. Operating personnel from the Station Manager through the operations chain to the licensed operators will receive training equivalent to that identified in Enclosure 3 to H. R. Denton's March 28, 1980 letter. Portions of the training will also be administered to supervisors and technicians in the Instrumentation and Control, Health Physics, and Chemistry departments commensurate with their responsibilities.

Task II.D.1 Performance Testing of Boiling Water Reactor and Pressurized Water Reactor Relief and Safety Valves (NUREG-0737)

Position

Pressurized-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.

Response

By letter dated July 1, 1981, R. C. Youngdahl (Consumers Power) transmitted the Interim Data Report for the EPRI PWR Safety and Relief Valve Test Program. This report summarizes the test data collected to date on relief and safety valves. Our Seabrook Station units each have two Garrett Model No. 3750014 relief valves and three Crosby Model No. DS-C-56964 safety valves. Relief and safety valves representative of the above valves are being tested in the EPRI program. Seabrook will submit evaluations and other plant specific data on a schedule consistant with the R. C. Youngdahl letter of December 15, 1980, and modified on July 1, 1981.

II.D.3 Direct Indication of Relief- and Safety-Valve Position (NUREG-0737)

Position

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

Response

Refer to FSAR Section 5.2.2.8

Task II.E.1.1 Auxiliary Feedwater System Evaluation (NUREG-0737)

Position

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

(1) Perform a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss-of-main-feedwater transient

conditions. Particular emphasis is given to determining potential failures that could result from human errors, common causes, single-point vulnerabilities, and test and maintenance outages;

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

(3) Re-evaluate the AFW system flowrate design bases and criteria.

Response

(1) An Emergency Feedwater System reliability analysis using event-tree and fault-tree logic techniques to determine the potential for EFW system failure under various loss-of-main-feedwater transient conditions is being developed.

In an NRC letter dated October 30, 1981 from F. J. Miraglia to PSNH concerning Auxiliary (Emergency) Feedwater System Reliability, it was stated that: "It is the NRC Staff's position that the applicant provide three AFW pumps, each capable of providing at least the minimum flow necessary to the steam generators for decay heat removal during a loss of off-site power. At least two of these pumps, and their associated trains, must be safety grade.

Your letter went on to acknowledge: "The Seabrook FSAR shows a two-pump safety grade 'emergency feedwater system' and also a non-safety grade 'startup feed pump' in parallel with the two emergency feedwater pumps."

Your letter further concluded: "The latter pump does not appear to be powered by the emergency busses. Thus, it does not appear that the present Seabrook design will meet this position regarding the availability of the pumps on loss of off-site power."

In response by PSNH letter dated December 4, 1981, additional information was provided to the NRC Staff showing that the "startup feed pump" could, in fact, be powered by an emergency bus and indicated where in the FSAR that information could be found.

Based on this, it is our position that the Seabrook design for the Emergency Feedwater System meets the reliability goals established by the NRC Staff.

(2) A review of the emergency feedwater system design is provided in FSAR Section 6.8.1.

(3) PSNH will reevaluate the Emergency Feedwater System flowrate design bases and criteria and provide results to the NRC by June 1, 1982.

Task II.E.1.2 Auxiliary Feedwater System Automatic Initiation and Flow Indication (NUREG-0737)

Part I: Auxiliary Feedwater System Automatic Initiation

Position

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

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

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

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

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

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

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

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

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

Response

Seabrook Station refers to its supplementary feedwater system as an emergency feedwater system instead of an auxiliary feedwater system (see FSAR Section 6.8). For the responses to the above seven (7) items, refer to FSAR Section 6.8.

Part 2: Auxiliary Feedwater System Flowrate Indication

Position

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

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

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

Response

(1) Refer to FSAR Section 6.8.5 and Table 7.5-1, item 9.

(2) Refer to FSAR Sections 6.8.5 and 7.3.

Task II.E.3.1 Emergency Power Supply for Pressurizer Heaters (NUREG-0737)

Position

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

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

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

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

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

Response

The following responses address the above four positions in the order shown:

(1) It has been determined that one bank of pressurizer heaters (350 kw) is necessary to establish and maintain natural circulation at hot standby conditions. One bank of heaters can be supplied from Train A emergency power supply and another bank can be supplied from Train B emergency power supply. The bank of heaters, which is fed from the Train B emergency power supply is automatically shed from the emergency power source upon the occurrence of a safety injection actuation signal.

(2) Procedures and training will be developed to make the operator aware of when and how the required pressurizer heaters are connected to the emergency bus consistent with the schedule for procedure development presented in FSAR Chapter 13.

(3) One bank of pressurizer heaters (350 kW) can be manually connected after two (2) minutes to the Train A emergency power supply (diesel generator) following the loss of offsite power. Similarly, another bank can be connected to the Train B emergency power supply. The connection can be accomplished from the control room.

(4) The breakers connecting these banks of heaters to the emergency power supplies are Class 1E.

Task II.E.4.1 Dedicated Hydrogen Penetrations (NUREG-0737)

Position

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

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

Response

Seabrook Station utilizes two separate and redundant Westinghouse thermal hydrogen recombiners located on the 25'-0" elevation inside the containment building; thus, no pipe penetrations are required.

The back-up purge system consists of two separate and redundant pipe-line/isolation valve/penetration systems sized for a normal 2% of containment volume/day (38.1 cfm) flow and a maximum of 1,000 cfm.

Refer to FSAR Subsections 6.2.5.1.i and 6.2.5.2.d.

Task II.E.4.2 Containment Isolation Dependability (NUREG-0737)

Position

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

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

(3) All non-essential systems shall be automatically isolated by the containment isolation signal.

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

(5) The containment setpoint pressure that initiates containment isolation for non-essential 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 III.3.f, during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days. (A copy of the Staff Interim Position is enclosed as Attachment 1.)

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

Response

(1) Diverse parameters are used wherever possible for developing isolation signals. The types of containment isolation signals used are: main steam line isolation signal, safety injection signal, reactor trip signal coincident with a low reactor coolant Tavg signal, steam generator hi-hi level signal, control switch, containment spray actuation signal, containment ventilation isolation signal, high containment radiation signal, phase A and phase B containment isolation signals, and a refueling water storage tank lo-lo level signal coincident with a safety injection signal. Table 6.2-83 lists all containment isolation valves and their corresponding containment isolation signal(s).

(2) Phase A containment isolation, whose function is to prevent fission product release, isolates all lines not essential to reactor protection (Refer to FSAR Section 7.3.1.1.a). Phase B containment isolation isolates the containment following a loss of reactor coolant accident or a steam or feedwater line break within containment. Together, they isolate all but engineered safety feature lines penetrating the containment (See Figure 6.2-94, for Isolation Valves Diagrams).

(3) Per Section 6.2.4.2c, Valve Actuation Signals: "Automatically-tripped isolation valves are actuated to the closed position by one of two separate containment isolation signals. The first of these signals ("T" signal) is derived in conjunction with automatic safety injection actuation or high containtment pressure, and trips the majority of the automatic isolation valves. These are valves in the non-essential process lines which do not increase the potential for damage to in-containment equipment when isolated. This is defined as "phase A" isolation, and the valves are designated by the letter "T" in the isolation diagrams, Figure 6.2-94. The second, or "phase B," containment isolation signal ("P" signal) is derived from Hi-3 containment pressure and/or actuation of the containment spray system, and trips the automatic isolation valves in the other process lines (which do not include safety injection lines) penetrating the containment. These isolation valves are designated by the letter "P" in the isolation diagrams".

(4) Per Section 6.2.4.2c, Valve Actuation Signals: "All valves that receive a containment isolation signal cannot be reopened until

the isolation signal is reset and manual action is taken to reopen the valve".

(5) Phase A containment isolation ("T" signal) isolates all non-essential process lines on receipt of a safety injection signal. This isolation signal is assumed to be generated when the containment pressure reaches a maximum of 7.4 psig, which includes a drift variation from the nominal value of 5.0 psig. The low set point value, 2.6 psig, which accounts for drift below nominal, is the minimum compatible with normal operating conditions, i.e., 0.5 psig normal to 1.5 psig maximum. See Section 6.2.1.

(6) The containment isolation purge supply air valves, COP-V1 and COP-V2, as well as the containment isolation purge exhaust air valves, COP-V3 and COP-V4, are ANSI Safety Class 2, Seismic Category I valves. They are redundant valves in series and are provided with ANSI Safety Class 2, seismic Category I, penetration piping between them. The valves are required to be shut immediately following a containment ventilation isolation or containment high radiation signal. Since the valves may be open during normal plant operation, start-up, and hot standby, these valves will be periodically tested to insure valve and valve actuator performance. The applicable General Design Criteria, valve position, closure time, etc., are given in Table 6.2-83. A full description of containment isolation valves is given in Section 6.2.4.

(7) Per Table 6.2-83, the containment isolation purge supply air valves, COP-V1 and COP-V2, as well as the containment isolation purge exhaust air valves, COP-V3 and COP-V4, close on a high radiation signal as well as on a containment ventilation isolation signal (CVIS).

Task II.F.1 <u>Additional Accident-Monitoring Instrumentation</u>

Task II.F.1, Attachment 1 <u>Noble Gas Effluent Monitor (NUREG-0737)</u>

<u>Position</u>

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 range of interest.

(1) Noble gas effluent monitors with an upper range capability of $10^5$ Ci/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 $10^5$ Ci/cc (X-133). Multiple monitors are considered to be necessary to cover the range of interest. The range capacity of individual monitors should overlap by a factor of ten.

Response

Refer to FSAR Section 12.3.4.2.b.2.(e) and FSAR Table 12.3-14.

Task II.F.1, Attachment 2 Sampling & Analysis of Plant Effluents (NUREG-0737)

Position

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

Response

Seabrook Station will have equipment to collect and analyze representative samples of radioactive iodines and particulates in station gaseous effluents during and following an accident.

The NRC will be kept appraised of our progress in selecting equipment.

Task II.F.1, Attachment 3 Containment High-Range Radiation Monitor (NUREG-0737)

Position

In containment radiation-level monitors with a maximum range of $10^8$ rad/hr shall be installed. A minimum of two (2) such monitors that are physically separated shall be provided. Monitors shall be developed and qualified to function in an accident environment.

Response

Refer to FSAR Table 7.5-1, item 16. (Range: $10^0$ to $10^7$ R/hr, gamma only)

Task II.F.1, Attachment 4 Containment Pressure Monitor (NUREG-0737)

Position

A continuous indication of containment pressure shall be provided in the control room of each operating reactor. Measurement and indication capability shall include three (3) times the design pressure of the containment for concrete, four (4) times the design pressure for steel, and -5 psig for all containments.

Response

Refer to FSAR Table 7.5-1, item 4

Task II.F.1, Attachment 5 Containment Water Level Monitor (NUREG-0737)

Position

A continuous indication of containment pressure shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWR's and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided for PWR's and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity. For BWR's, 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.

Response

Refer to FSAR Table 7.5-1, item 10, for containment sump water level.

Refer to FSAR Table 7.5.1, item 11, for containment building water level.

Task II.F.1, Attachment 6 Containment Hydrogen Monitor (NUREG-0737)

Position

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

Response

Refer to FSAR Section 6.2.5 and Table 7.5-1, item 17.

Task II.F.2 Instrumentation for Detection of Inadequate Core Cooling (NUREG-0737)

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.

Response

PSNH is in the process of selecting post-accident monitoring instrumentation vis-a-vis the guidance of Regulatory Guide 1.97 (Rev. 2).

Consistent with other submittals which we have made to the NRC (dated November 27, 1981), we plan to submit a report which addresses Seabrook post-accident monitoring instrumentation (includes inadequate core cooling instrumentation) by April, 1982.

Task II.G.1 Emergency Power for Pressurizer Equipment (NUREG-0737)

Position

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

Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators -

(1) Motive and control components of the power-operated relief valves (PORV's) shall be capable of being supplied from either the off-site power source or the emergency power source when the off-site power is not available.

(2) Motive and control components associated with the PORV block valves shall be capable of being supplied from either the off-site power source or the emergency power source when the off-site power is not available.

(3) Motive and control power connections to the emergency buses for the PORV's and their associated block valves shall be through

devices that have been qualified in accordance with safety-grade requirements.

(4) The pressurizer level indication instrument channels shall be powered from the vital instrument buses. The buses shall have the capability of being supplied from either the off-site power source or the emergency power source when off-site power is not available.

Response

(1) The motive and control circuits for the Power-Operated Relief Valves (PORV's) are supplied by the safety-related dc batteries. One valve is supplied by the A Train batteries, while the other is supplied by the B Train batteries.

(2) The motive and control power for all components associated with the PORV block valves is provided from redundant safety-related motor control centers which are supplied from the diesel generators within ten seconds following loss of the off-site power supply.

(3) The PORV's and PORV block valves are connected to the emergency power supply by Class 1E breakers.

(4) Pressurizer level indication instrument channels are supplied from the Process Protection Cabinets. These cabinets are powered from the vital instrument buses which can be supplied either from the emergency ac supply or the Class 1E batteries.

Task II.K.1 IE Bulletins on Measures to Mitigate Small-Break LOCA's and Loss of Feedwater Accidents (NUREG-0694)

Task II.K.1.5 Review ESF Valves (NUREG-0660, Table C.1)

Position

Review all valve positions and positioning requirements and positive controls and all related test and maintenance procedures to assure proper ESF functioning.

Response

Proper ESF functioning will be verified through completion of the applicable portions of the start-up test program prior to fuel load.

Task II.K.1.10 <u>Operability Status (NUREG-0660, Table C.1)</u>

<u>Position</u>

Review and modify (as required) procedures for removing safety-related systems from service (and restoring to service) to assure operability status is known.

<u>Response</u>

Same as response to Task I.C.6.

Task II.K.1.17 <u>Trip Per Low-Level Bistables (NUREG-0694)</u>

<u>Position</u>

For Westinghouse designed reactors, trip the pressurizer low-level coincident signal bistables, so that safety injection would be initiated when the pressurizer low-pressure set point is reached regardless of the pressurizer level. See Bulletin 79-06A and Revision 1, Item 3 in Reference 11.

<u>Response</u>

FSAR Section 7.2 indicates that the reactor trip and safety injection are initialed on pressurizer low pressure. Pressurizer low-level trips are not utilized in the Seabrook design.

Task II.K.1.20 <u>Procedures and Training for Prompt Manual Reactor Trip (NUREG-0694)</u>

Not applicable to Seabrook Station.

Task II.K.1.21 <u>Automatic Safety-Grade Anticipatory Reactor Trip-B&W</u>

Not applicable to Seabrook Station.

Task II.K.1.22 <u>Auxiliary Heat Removal Systems Procedure-BWRs</u>

Not applicable to Seabrook Station.

Task II.K.1.23 Reactor Vessel Level Indication Procedures-BWRs

Not applicable to Seabrook Station.

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

Three items from Task II.K.2 have been made requirements for other pressurized water reactor designs. These are discussed below.

Task II.K.2.13 Thermal Mechanical Report - Effect of High Pressure Injection on Vessel Integrity for Small Break Loss-of-Coolant Accident with No Auxiliary Feedwater (NUREG-073)

Position

A detailed analysis shall be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.

Response

Westinghouse (in support of the Westinghouse Owners Group) is developing a method and will perform analyses for a spectrum of small loss-of-coolant accidents. The method will employ the NOTRUMP computer program to generate the thermal/hydraulic transients. The thermal transients on the reactor vessel beltline and the inlet nozzle will be analyzed based on the thermal/hydraulic data from the NOTRUMP code. The analyses are scheduled to be completed in 7/82; the Seabrook docket will reference appropriate documents submitted by the Westinghouse Owners Group to the NRC.

Task II.K.2.17 Potential for Voiding in the Reactor Coolant System During Transients

Analyze the potential for voiding in the reactor coolant system (RCS) during anticipated transients.

Response

Westinghouse (in support of the Westinghouse Owners Group) has performed a study which addresses the potential for void formation in Westinghouse designed nuclear steam supply systems during natural circulation cooldown/depressurization transients. This study has been submitted to the NRC by the Westinghouse Owners Group (Reference 5) and is applicable to the Seabrook Station.

In addition, the Westinghouse Owners Group is currently developing appropriate modifications to the Westinghouse Owners Group Reference Operating Instructions to take the results of the study into account so as to preclude void formation in the upper head region during natural circulation cooldown/depressurization transients, and to specify those conditions under which upper head voiding may occur. PSNH will consider the generic guidance developed by the Westinghouse Owners Group in the development of plant specific operating procedures.

Task II.K.2.19 Sequential Auxiliary Feedwater Flow Analysis

Provide a benchmark analysis of sequential auxiliary feedwater (AFW) flow to the steam generators following a loss of feedwater.

Response

Not applicable to Westinghouse pressurized water reactors per NRC Letter to Duquesne Light, dated June 29, 1981.

Task II.K.3 Final Recommendations of B&O Task Force (NUREG-0737)

Task II.K.3.1 Installation and Testing of Automatic Power-Operated Relief Valve Isolation System (NUREG-0737)

Position

All PWR licensees should provide a system that uses the PORV block valve to protect against a small-break loss-of-coolant accident. This system will automatically cause the block valve to close when the reactor coolant system pressure decays after the PORV has opened. Justification should be provided to assure that failure of this system would not decrease overall safety by aggravating transients and accidents.

Each licensee shall perform a confirmatory test of the automatic block valve closure system following installation.

Response

Westinghouse, as a part of the response prepared for the Westinghouse Owners Group to address item II.K.3.2, has evaluated the necessity of incorporating an automatic pressurizer power-operated relief valve isolation system. This evaluation is documented in Reference 6 and concluded that such a system should not be required.

Task II.K.3.2 Report on Overall Safety Effect of Power-Operated Relief Valve Isolation System (NUREG-0737)

Position

(1) The licensee should submit a report for staff review documenting the various actions taken to decrease the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV) and show how those actions constitute sufficient improvements in reactor safety.

(2) Safety-valve failure rates based on past history of the operating plants designed by the specific nuclear stream supply system (NSSS) vendor should be included in the report submitted in response to (1) above.

Response

As mentioned in the response to position II.K.3.1, the Westinghouse Owners Group has submitted a Westinghouse-prepared report (Reference 6) which provides a probabilistic analysis to determine the probability of a PORV LOCA, estimates the effect of the post-TMI modifications, evaluates an automatic PORV isolation concept, and provides PORV and safety valve operational data for Westinghouse plants. Because of the sensitivity analyses included in the report, the report is generic and is applicable to the Seabrook Station. The report identifies a significant reduction in the PORV LOCA probability as a result of post-TMI modifications, and the calculations compare favorably with the operational data for Westinghouse plants (included as an appendix to the report).

Task II.K.3.3 Reporting SV and PORV Challenges and Failures (NUREG-0694)

Position

Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report.

Response

Any failure of a safety or relief valve will be reported promptly to the NRC using the established "Licensee Event Report" (LER) System, and all challenges to such valves will be reported annually in accordance with Technical Specification 6.9.1.5.

Task II.K.3.5 Automatic Trip of Reactor Coolant Pumps During Loss-of-Coolant Accident (NUREG-0737)

Position

Tripping of the reactor coolant pumps in case of a loss-of-coolant accident (LOCA) is not an ideal solution. Licensees should consider other solutions to the small-break LOCA problem (for example, an increase in safety injection flow rate). In the meantime, until a better solution is found, the reactor coolant pumps should be tripped automatically in case of a small-break LOCA. The signals designated to initiate the pump trip are discussed in NUREG-0623.

Response

In response to IE Bulletin No. 79-06C, Westinghouse (in support of the Westinghouse Owners Group) performed an analysis of delayed reactor coolant pump (RCP) trip during small-break LOCAs. This analysis is documented in Reference 7 and is the basis for the Westinghouse position on RCP trip (i.e., automatic RCP trip is not necessary since sufficient time is available for manual tripping of the RCPs).

Westinghouse (again in support of the Westinghouse Owners Group) has performed test predictions of the LOFT Experiment L3-6. The results of these predictions are documented in References 8 and 9. The results constitute both a best estimate model prediction with the NOTRUMP computer program and an evaluation model prediction with the Westinghouse FLASH computer program using the supplied set of initial boundary assumptions.

The NRC has indicated that small-break tests at the Semiscale and LOFT facilities as well as Owners Group test predictions will aid in the final resolution of this issue. The results of the above mentioned Westinghouse analyses and predictions are in good agreement and, therefore, design modifications are not considered to be necessary.

Task II.K.3.7 Evaluation of Power-Operated Relief Valve Opening Probability During Over-Pressure Transient (NUREG-0737)

Position

Most overpressure transients should not result in the opening of the power-operated relief valve (PORV). Therefore, licensees should document that the PORV will open in less than 5% of all anticipated over-pressure transients using the revised setpoints and anticipatory trips for the range of plant conditions which might occur during a fuel cycle.

Response

Not applicable to Seabrook Station. See NUREG-0737 Applicability.

Task II.K.3.9 Proportional Integral Derivative Controller Modification Position (NUREG-0737)

The Westinghouse recommended modifications to the proportional integral derivative (PID) controller should be implemented by affected licensees.

Response

The Seabrook design includes a pressure integral derivative (PID) controller in the power-operated relief valve control circuit (see Figures 7.7-4 and 7.2-1, Sheet 11). The time derivative constant in the PID controller for the pressurizer PORV will be turned to "OFF". The appropriate plant procedure for calibrating the setpoints in this nonsafety grade system will reflect this decision.

Setting the derivative time constant to "OFF", in effect, removes the derivative action from the controller. Removal of the derivative action will decrease the likelihood of opening the pressurizer PORV since the actuation signal for the valve is then no longer sensitive to the rate of change of pressurizer pressure.

Task II.K.3.10 Proposed Anticipatory Trip Modification (NUREG-0737)

Position

The anticipatory trip modification proposed by some licensees to confine the range of use to high-power levels should not be made until it has been shown on a plant-by-plant basis that the probability of a small-break loss-of-coolant accident (LOCA) resulting from a stuck-open power-operated relief valve (PORV) is substantially unaffected by the modification.

Response

The Seabrook plant design includes the capability to undergo a 50% load reduction without requiring a reactor trip. This capability is made available through the use of a 40% steam bypass to the condenser and automatic rod control to reduce core power by 10%. Plant analysis shows that pressurizer power-operated relief valves (PORVs) will not be challenged by a 50% load reduction from full power. An evaluation of a full load reduction from 50% power also shows that PORVs will not be challenged even though the reactor is not tripped. The deletion of a direct (or anticipatory) reactor trip from turbine trip below 50% power will not cause the PORVs to be challenged. Therefore, the probability of a small-break Loss-Of-Coolant Accident (LOCA) from a stuck-open PORV is substantially unaffected by the deletion of an anticipatory reactor trip from turbine trip below 50% power.

Task II.K.3.11 Justification for Use of Certain PORVs (NUREG-0694)

Position

Demonstrate that the PORV installed in the plant has a failure rate equivalent to or less than the values for which there is an operating history.

Response

The PORVs utilized at Seabrook are a relatively new design developed by the Garrett Pneumatic Systems Division of the Garrett Corporation. Similar type valves are presently being supplied to both Combustion Engineering and Westinghouse for use in their NSSS design. At this time, there is insufficient operating data upon which to base a statistically accurate failure rate history.

However, a valve of similar design to that supplied to both Combustion Engineering and Westinghouse was tested at the Wyle Laboratories as a part of the EPRI/PWR Safety and Relief Valve Test Program. In addition to the successful functional test results, the Garrett valve operated normally, with no tendency to fail to operate, either open or closed, through at least 79 cycles.

As operating history is gained on this particular valve design, should an abnormal failure rate become apparent, appropriate corrective action will be taken.

Task II.K.3.12 Confirm Existence of Anticipatory Reactor Trip Upon Turbine Trip (NUREG-0737)

Position

Licensees with Westinghouse-designed operating plants should confirm that their plants have an anticipatory reactor trip upon turbine trip. The licensee of any plant where this trip is not present should provide a conceptual design and evaluation for the installation of this trip.

Response

The Seabrook design includes an anticipatory reactor trip upon turbine trip (refer to Figure 7.2-1, Sheet 16).

Task II.K.3.13 Separation of High-Pressure Coolant Injection and Reactor Core Isolation Cooling System Initiation Levels--Analysis and Implementation (NUREG-0737)

Position

Currently, the reactor core isolation cooling (RCIC) system and the high-pressure coolant injection (HPCI) system both initiate on the same low-water-level signal and both isolate on the same high-water-level signal. The HPCI system will restart on low water level but the RCIC system will not. The RCIC system is a low-flow system when compared to the HPCI system. The initiation levels of the HPCI and RCIC system should be separated so that the RCIC system initiates at a higher water level than the HPCI system. Further, the initiation logic of the RCIC system should be modified so that the RCIC system will restart on low water level. These changes have the potential to reduce the number of challenges to the HPCI system and could result in less stress on the vessel from cold water injection. Analyses should be performed to evaluate these changes. The analyses should be submitted to the NRC staff and changes should be implemented if justified by the analyses.

Response

Not applicable to Seabrook Station

Task II.K.3.15 Modify Break-Detection Logic to Prevent Spurious Isolation of High-Pressure Coolant Injection and Reactor Core Isolation Cooling (NUREG-0737)

Position

The high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems use differential pressure sensors on elbow taps in the steam lines to their turbine drives to detect and isolate pipe breaks in the systems. The pipe-break-detection circuitry has resulted in spurious isolation of the HPCI and RCIC systems due to the pressure spike which accompanies startup of the systems. The pipe-break-detection circuitry should be modified so that pressure spikes resulting from HPCI and RCIC system initiation will not cause inadvertent system isolation.

Response

Not applicable to Seabrook Station

Task II.K.3.16 Reduction of Challenges and Failures of Relief Valves -- Feasibility Study and System Modification (NUREG-0737)

Position

The record of relief-valve failures to close for all boiling-water reactors (BWRs) in the past 3 years of plant operation is approximately 30 in 73 reactor-years (0.41 failures per reactor-year). This has demonstrated that the failure of a relief valve to close would be the most likely cause of a small-break loss-of-coolant accident (LOCA). The high failure rate is the result of a high relief-valve challenge rate and a relatively high failure rate per challenge (0.16 failures per challenge). Typically, five valves are challenged in each event. This results in an equivalent failure rate per challenge of 0.03. The challenge and failure rates can be reduced in the following ways: (See list in NUREG-0737).

Response

Not applicable to Seabrook Station

Task II.K.3.17 Report on Outages of Emergency Core-Cooling Systems Licensee Report and Proposed Technical Specification Changes NUREG-0737)

Position

Several components of the emergency core-cooling (ECC) systems are permitted by technical specifications to have substantial outage times (e.g., 72 hours for one diesel-generator; 14 days for the HPCI system). In addition, there are no cumulative outage time limitations for ECC systems. Licensees should submit a report detailing outage dates and lengths of outages for all ECC systems for the last 5 years of operation. The report should also include the causes (i.e., controller failure, spurious isolation).

Response

A procedure for collecting and submitting information concerning ECC system outages will be developed and implemented three months prior to fuel load.

Task II.K.3.18 Modification of Automatic Depressurization System Logic—Feasibility for Increased Diversity for Some Event Sequences (NUREG-0737)

Position

The automatic depressurization system (ADS) actuation logic should be modified to eliminate the need for manual actuation to assure adequate core cooling. A feasibility and risk assessment study is required, to determine the optimum approach. One possible scheme that should be considered is ADS actuation on low reactor-vessel water level provided no high-pressure coolant injection (HPCI) or high-pressure coolant system (HPCS) flow exists and a low-pressure emergency core cooling (ECC) system is running. This logic would complement, not replace, the existing ADS actuation logic.

Response

Not applicable to Seabrook Station

Task II.K.3.21 Restart of Core Spray & Low-Pressure Coolant-Injection Systems (NUREG-0737)

Position

The core-spray and low-pressure, coolant-injection (LPCI) system flow may be stopped by the operator. These systems will not restart automatically on loss of water level if an initiation signal is still present. The core spray and LPCI system logic should be modified so that these systems will restart, if required, to assure adequate core cooling. Because this design modification affects several core-cooling modes under accident conditions, a preliminary design should be submitted for staff review and approval, prior to making the actual modification.

Response

Not applicable to Seabrook Station

Task II.K.3.22 Automatic Switchover of Reactor Core Isolation Cooling System Suction -- Verify Procedures & Modify Design (NUREG-0737)

Position

The reactor core isolation cooling (RCIC) system takes suction from the condensate storage tank with manual switchover to the suppression pool when the condensate storage tank level is low. This switchover should be made automatically. Until the automatic switchover is implemented, licensees should verify that clear and cogent procedures exist for the manual switchover of the RCIC system suction from the condensate storage tank to the suppression pool.

Response

Not applicable to Seabrook Station

Task II.K.3.24 Confirm Adequacy of Space Cooling for High-Pressure Coolant Injection & Reactor Core Isolation Cooling System (NUREG-0737)

Position

Long-term operation of the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems may require space cooling to maintain the pump-room temperatures within allowable limits. Licensees should verify the acceptability of the consequences of a complete loss of alternating-current power. The RCIC and HPCI systems should be designed to withstand a complete loss of off-site alternating-current power to their support systems, including coolers, for at least 2 hours.

Response

Not applicable to Seabrook Station

Task II.K.3.25 Effect of Loss of Alternating-Current Power on Pump Seals (NUREG-0737)

Position

The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating-current (ac) power for at least 2 hours. Adequacy of the seal design should be demonstrated.

Response

During normal operation, seal injection flow from the chemical and volume control system is provided to cool the RCP seals, and the component cooling water system provides flow to the thermal barrier heat exchanger to limit the heat transfer from the reactor coolant to the RCP internals. In the event of loss of offsite power, the RCP motor is deenergized and both of these cooling supplies are terminated; however, the diesel generators are automatically started and <u>either</u> seal injection flow or component cooling water to the thermal barrier heat exchanger is automatically restored within 12 and 32 seconds respectively. Either of these cooling supplies is adequate to provide seal cooling and prevent seal failure due to loss of seal cooling during a loss of offsite power for at least 2 hours.

Task II.K.3.27 Provide Common Reference Level for Vessesl Level Instrumentation (NUREG-0737)

Position

Different reference points of the various reactor vessel water level instruments may cause operator confusion. Therefore, all level instruments should be referenced to the same point. Either the bottom of the vessel or the top of the active fuel are reasonable reference points.

Response

Not applicable to Seabrook Station

Task II.K.3.28 Verify Qualification of Accumulators on Automatic Depressurization System Valves (NUREG-0737)

Position

Safety analysis reports claim that air or nitrogen accumulators for the automatic depressurization system (ADS) valves are provided with sufficient capacity to cycle the valves open five (5) times at design pressures. GE has also stated that the emergency core cooling (ECC) systems are designed to withstand a hostile environment and still perform their function for 100 days following an accident. Licensee should verify that the accumulators on the ADS valves meet these requirements, even considering normal leakage. If this cannot be demonstrated, the licensee must show that the accumulator design is still acceptable.

Response

Not applicable to Seabrook Station

Task II.K.3.30 Revised Small-Break Loss-of-Coolant Accident Methods to Show Compliance with 10 CFR Point 50, Appendix K (NUREG-0737)

Position

The analysis methods used by nuclear steam supply system (NSSS) vendors and/or fuel suppliers for small-break loss-of-coolant accident (LOCA) analysis for compliance with Appendix K to 10 CFR Part 50 should be revised, documented, and submitted for approval. The revisions should account for comparisions with experimental data, including data from the LOFT Test and Semiscale Test facilities.

Response

The present Westinghouse Small Break Evaluation Model used to analyze the Seabrook Station (refer to Section 15.6.5) is in conformance with 10CFR Part 50, Appendix K. However (as documented in Reference 10), Westinghouse has indicated that they will, nevertheless, address the specific NRC items contained in NUREG-0611 in a model change scheduled for completion by mid 1982.

Task II.K.3.31 Plant-Specific Calculations to Show Compliance with 10 CFR, Part 50.46 (NUREG-0737)

Plant-specific calculations using NRC-approved models for small-break loss-of-coolant accidents (LOCAs) as described in Item II.K.3.30 to show compliance with 10 CFR 50.46 should be submitted for NRC approval by all licensees.

Response

The present Westinghouse Small Break Evaluation Model and small break LOCA analyses for Seabrook (refer to Section 15.6.5) are in conformance with 10 CFR Part 50, Appendix K and 10 CFR Part 50.46. As stated in the response to Item II.K.3.30, Westinghouse plans to submit a new Small Break Evaluation Model to the NRC for review by the latter half of 1982. If this new Westinghouse model (with subsequent NRC review and approval) yields more limiting results versus the current approved model, a re-analysis with the new model will be submitted to the NRC in accordance with the NRC schedule.

Task II.K.3.44 Evaluation of Anticipated Transients with Single Failure to Verify No Fuel Failure (NUREG-0737)

Position

For anticipated transients combined with the worst single failure and assuming proper operator actions, licensees should demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery. Transients which result from a stuck-open relief valve should be included in this category.

Response

Not applicable to Seabrook Station

Task II.K.3.45 Evaluation of Depressurization with Other than Automatic Depressurization System (NUREG-0737)

Position

Analyses to support depressurization modes other than full actuation of the automatic depressurization system (ADS) (e.g., early blowdown with one or two safety relief valves (SRVs)) should be provided. Slower depressurization would reduce the possibility of exceeding vessel integrity limits by rapid cooldown.

Response

Not applicable to Seabrook Station

Task II.K.3.46 Response to List of Concerns from ACRS Consultant (NUREG-0660)

Position

Not applicable to Seabrook Station

Task III.A.1.1 Upgrade Licensee Emergency Preparedness - Short Term (NUREG-0660)

Position

Licensees will upgrade emergency preparedness in accordance with the requirements described in the NRC "Action Plan for Promptly Improving Emergency Preparedness" (SECY 79-450), which was distributed to all licensees during regional meetings in August, 1973, and in accordance with subsequently issued criteria (NUREG-0654).

Response

Refer to the Seabrook Station Radiological Emergency Plan, which was submitted as a separate volume of the FSAR.

Task III.A.1.2 Upgrade Emergency Support Facilities

Position

Each operating nuclear power plant shall maintain an on-site Technical Support Center (TSC) separate from, and in close proximity to, the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of, and responsible for, engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans, as necessary, to incorporate the role and location of the TSC. Records that pertain to the as-built conditions and layout of structures, systems, and components shall be readily available to personnel in the TSC.

An Operational Support Center (OSC) shall be established separate from the control room and other emergency response facilities as a place where operations support personnel can assemble and report in an emergency situation to receive instructions from the station staff. Communications shall be provided between the OSC, TSC, EOF, and Control Room.

An Emergency Operations Facility (EOF) will be operated by a licensee for continued evaluation and coordination of all licensee activities related to an emergency having, or potentially having, environmntal consequences.

Response

Refer to Section 6, "Emergency Facilities and Equipment," and Section 8, "Organization," of the Seabrook Station Radiological Emergency Plan.

Task III.D.1.1 Integrity of Systems Outside Containment Likely to Contain Radioactive Material for Pressurized-Water Reactors & Boiling-Water Reactors (NUREG-0737)

Position

Applicants shall implement a program to reduce leakage from systems outside containment that would, or could, contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate leak reduction -

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

(b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction - Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Response

(1) (a) A leak reduction program, identifying all systems that could carry radioactive fluid outside of containment, will be prepared four months prior to fuel load. During pre-operational and Hot Functional testing, these systems will be visually inspected and all practical leak reduction measures will be implemented.

(b) Actual leakage rates with the systems in operation will be provided as a part of the initial Startup Test Report.

(2) An ongoing leak reduction program, including preventative maintenance to reduce leakage to as-low-as-practical levels, and periodic integrated leak tests, at intervals not to exceed each refueling, shall be prepared four months prior to fuel load.

Task III.D.3.3 Improved Inplant Iodine Instrumentation Under Accident Conditions (NUREG-0737)

Position

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

Response

Seabrook Station will provide equipment and associated procedures and training for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident. The NRC will be kept appraised of our progress in selecting equipment.

Task III.D.3.4 Control-Room Habitability Requirements (NUREG-0737)

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 safety 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).

Response

Section 6.4, "Habitability Systems" as well as Sections 12.3, 9.4 and 9.5 describe in detail the methods employed to maintain the habitability of control room during accident conditions.

## 1.9.1 References


1. "Report on Small-Break Accidents for Westinghouse Nuclear Steam Supply System" WCAP-9600 (Proprietary) June 1979, and WCAP-9601 (Non-Proprietory) June 1979.

2. "Inadequate Core Cooling Studies of Scenarios with Feedwater Available, Using the NOTRUMP Computer Code" WCAP-9753 (Proprietary) June 1980, WCAP-9754 (Non-Proprietary).

3. Letter OG-18, dated October 30, 1979, C. Reed (Chairman, Westinghouse Owners Group) to D. F. Ross (NRC).

4. "NUREG-0578 2.1.9c Transient and Accident Analysis" WCAP-9692, March 1980.

5. Letter OG-57, dated April 20, 1981, R. W. Jurgensen (Chairman, Westinghouse Owners Group) to P. S. Check (NRC).

6. Wood, D. C. and Gottshall, C.L., "Probabilistic Analysis and Operational Data in Response to NUREG-0737 Item II.K.3.2 for Westinghouse NSSS Plants," WCAP-9804, February 1981.

7. "Analysis of Delayed Reactor Coolant Pump Trip During Small Loss of Coolant Accidents for Westinghouse Nuclear Steam Supply Systems," WCAP-9584 (Proprietary) and WCAP-9585 (Non-Proprietary), August 1979.

8. Letter OG-49, dated March 3, 1981, R. W. Jurgensen (Chairman, Westinghouse Owners Group) to D. F. Ross, Jr. (NRC).

9. Letter OG-50, dated March 13, 1981, R. W. Jurgensen (Chairman, Westinghouse Owners Group) to D. F. Ross, Jr. (NRC).

10. Letter NS-TMA-2318, dated September 26, 1980, T. M. Anderson (Westinghouse) to D. G. Eisenhut (NRC).