Docket No. STN 50-437

OFFSHORE POWER SYSTEMS

OFFSHORE POWER SYSTEMS RESPONSES TO POST-TMI NRC REQUIREMENTS

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A. **RESPONSES TO IE BULLETIN No. 79-06A, (including Revision 1)**

Bulletin Item

1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

   1a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both auxiliary feedwater trains at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to the eventual core damage; (3) that potential exists, under certain accident or transient conditions, to have a water level in the pressurizer simultaneously with the reactor vessel not full of water; and (4) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

Offshore Power Systems Response

Offshore Power Systems formed a TMI Task Team to study the accident at Three Mile Island. The above accident review has been completed by the OPS task team. The task team will continue to evaluate future TMI developments.

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1b. Operational personnel should be instructed to: (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section 7a.); and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available.

Offshore Power Systems Response

This item applies to the plant owner and not to the activities conducted by Offshore Power Systems under a License to Manufacture Floating Nuclear Plants. Ample time exists to implement the stated requirements before an FNP owner will require an Operating

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License. OPS will support the development of the owner's Emergency Operating Procedures by issuing Emergency Operating Instructions for the FNP.

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1c. All licensed operators and plant management and supervisors with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

Offshore Power Systems Response

This item applies to the plant owner and not to the activities conducted by Offshore Power Systems under a License to Manufacture Floating Nuclear Plants. Ample time exists to implement the stated requirements before an FNP owner will require an Operating License.

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2. Review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

   a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

   b. Operator action required to prevent the formation of such voids.

   c. Operator action required to enhance core cooling in the event such voids are formed. (e.g., remote venting).

Offshore Power Systems Response

This item is addressed by recommendations 2.1.3.b and 2.1.9 of NUREG-0578. See Section B (following) for the OPS response.

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3. For your facilities that use pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection into the reactor coolant system, trip the low pressurizer level setpoint bistables such that, when the pressurizer pressure reaches the low setpoint, safety injection would be initiated regardless of the pressurizer level. The pressurizer level bistables may be returned to their normal operating positions during the pressurizer channel functional surveillance tests. In addition, instruct operators to manually initiate safety injection when the pressurizer pressure indication reaches the actuation setpoint whether or not the level indication has dropped to the actuation setpoint.

Offshore Power Systems Response

This Item is not applicable to the Floating Nuclear Plant. As stated in Section 6.3.2.2.1 of the Plant Design Report, the injection mode of emergency core cooling is initiated by the safety injection signal ("S" Signal). This signal is actuated by any one of the following:

1. Low Pressurizer Pressure
2. High Containment Pressure
3. High Differential Pressure Between Any Two Steam Lines
4. High Steam Line Flow Coincident With Either Low T_{AVG} or Low Steam Line Pressure
5. Manual Actuation

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4. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to permit containment isolation whether manual or automatic, of all lines whose isolation does not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

Offshore Power Systems Response

This item is addressed in recommendation 2.1.4 of NUREG-0578. See Section B (following) for the OPS response.

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5. For facilities for which the auxiliary feedwater system is not automatically initiated, prepare and implement immediately procedures which require the stationing of an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate adequate auxiliary feedwater to the steam generator(s) for those transients or accidents the consequences of which can be limited by such action.

Offshore Power Systems Response

This item is not applicable to the Floating Nuclear Plant. See the OPS response to recommendation 2.1.7.a of NUREG-0578 (Section B, following) for a discussion on automatic initiation of the Auxiliary Feedwater System for the FNP.

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6. For your facilities, prepare and implement immediately procedures which:

   6a. Identify those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature or pressure indication) which plant operators may utilize to determine that pressurizer power operated relief valve(s) are open.

Offshore Power Systems Response

This item is addressed by recommendation 2.1.3a of NUREG-0578. See section B(following) for the OPS response.

Bulletin Item

6b. Direct the plant operators to manually close the power operated relief block valve(s) when reactor coolant system pressure is reduced to below the set point for normal automatic closure of the power operated relief valve(s) and the valve(s) remain stuck open.

Offshore Power Systems Response

This item applies to the plant owner and not to the activities conducted by Offshore Power Systems under a License to Manufacture Floating Nuclear Plants. Ample time exists to implement the stated requirements before an FNP owner will require an Operating License. OPS will support the development of the owner's Emergency Operating Procedures by issuing Emergency Operating Instructions for the FNP.

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7. Review the action directed by the operating procedures and training instructions to ensure that:

   a. Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe plant conditions. For example, if continued operation of engineered safety features would threaten reactor vessel integrity then the HPI should be secured (as noted in 7b (2) below).

   b. Operating procedures currently, or are revised to, specify that if the high pressure injection (HPI) system has been automatically actuated because of low pressure condition, it must remain in operation until either:

      (1) Both low pressure injection (LPI) pumps are in operation and flowing for 20 minutes or longer; at a rate which would assure stable plant behavior; or

      (2) The HPI system has been in operation for 20 minutes, and all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If 50 degrees subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated. The degree of subcooling beyond 50 degrees F and the length of time HPI is in operation shall be limited by the pressure/temperature considerations for the vessel integrity.

   c. Item 7c has been superceded by a long term action in IE Bulletin 79-06C, as follows:

      Propose and submit a design which will assure automatic tripping of the operating RCPs under all circumstances in which this action may be needed.

   d. Operators are provided additional information and instructions to not rely upon pressurizer level indication alone, but to also examine pressurizer pressure and other plant parameter indications in evaluating plant conditions, e.g., water, inventory in the reactor primary system.

Offshore Power Systems Response

This item is addressed in the Offshore Power Systems response to recommendation 2.1.9 of NUREG-0578. See section B (following).

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8. Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

Offshore Power Systems Response

To the extent that this item may apply to the manufacturing license application, it is duplicated in recommendation 5 of NUREG-0585. See section D (following) for the OPS response. Note that primary responsibility for this item is with the plant owner.

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9. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

   In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

   a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

   b. Whether such systems are isolated by the containment isolation signal.

   c. Indicate the basis on which continued operability of the features is assured.

Offshore Power Systems Response

The portion of this item dealing with (1) identification of systems which are designed (at least in part) for transfer of radioactive fluids outside containment and (2) isolation of these systems upon activation of the containment isolation signal, is addressed in recommendation 2.1.4 of NUREG-0578. See Section B (following) for the OPS response.

With the exception of the Containment Ventilation System (VCC and VCD Subsystems), interlocks with high radiation levels are not provided; rather, other parameters which more reliably detect accident situations and provide for automatic initiation of containment isolation are provided (as discussed in the response to recommendation 2.1.4 in Section B). Containment isolation design also precludes automatic opening of containment isolation valves subsequent to resetting the appropriate engineered safety features signal, thereby preventing inadvertent transfer of radioactive liquids and gases.

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The Phase "A" signal automatically isolates the Containment Ventilation System. In addition, the containment air particulate and gas radiation monitors automatically transfer the Containment Ventilation System from an external air supply mode to a recirculation mode, if the system is not already isolated.

Operability of the above features is accomplished by periodic testing in accordance with Technical Specification requirements.

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10. Review and modify as necessary your maintenance and test procedures to ensure that they require:

    a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

    b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

    c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

Offshore Power Systems Response

This item applies to the plant owner and not to the activities conducted by Offshore Power Systems under a License to Manufacture Floating Nuclear Plants. Ample time exists to implement the stated requirements before an FNP owner will require an Operating License.

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11. Review your prompt reporting procedures for NRC notification to assure that NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time an open continuous communication channel shall be established and maintained with NRC.

Offshore Power Systems Response

This item applies to the plant owner and not to the activities conducted by Offshore Power Systems under a License to Manufacture Floating Nuclear Plants. Ample time exists to implement the stated requirements before an FNP owner will require an Operating License.

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12. Review operating modes and procedures to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

Offshore Power Systems Response

Available modes for removing hydrogen from the Reactor Coolant System are the following:

1. Hydrogen can be stripped from the reactor coolant to the pressurizer vapor space by pressurizer spray operation if the reactor coolant pump is operating.

2. Hydrogen in the pressurizer vapor space can be vented by power operated relief valves to the pressurizer relief tank.

3. Hydrogen can be removed from the Reactor Coolant System by the letdown line and stripped in the volume control tank where it enters the waste gas system. Waste gas system storage consists of 8 tanks of 600 FT^3 each.

4. Hydrogen could be released to the pressurizer and/or letdown line by controlled depressurization of the Reactor Coolant System.

5. In the event of a LOCA, hydrogen would vent with the steam to the containment.

Available modes of removing hydrogen from the containment are the following:

1. Two hydrogen recombiners, each capable of processing up to a maximum of 200 SCFM air with a 100% free hydrogen removal efficiency, can be utilized to recombine containment hydrogen with available oxygen. Hydrogen is distributed uniformly throughout the containment by automatic operation of the Air Return and Hydrogen Skimmer System.

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2. Containment atmosphere can be discharged to the annulus by operation of the Post-Accident Containment Venting System. The system provides a controlled and filtered containment purge capability by releasing air at a maximum rate of 50 SCFM to the annulus.

In addition to the above design features, which are currently incorporated in the FNP, a pressure vessel head vent system will be developed to remove hydrogen or other gases from the reactor vessel head via remote manual operation from the Control Room. This system will discharge into the pressurizer relief tank in the containment (see Response to 2.1.5, Section B following).

In addition to the above it should be noted that the NRC recommends a rulemaking regarding design features that would mitigate a severe core damage or a core melt accident. As a result, further design measures for the control of hydrogen may be suggested. OPS will consider and implement appropriate requirements in the FNP design once the rulemaking is completed and the requirements defined (see Recommendation 10 of Section D, following).

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13. Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the above items and identify design changes necessary in order to effect long term resolutions of these items.

Offshore Power Systems Response

The normal process of technical specification development during review of the final FNP design will certainly include careful consideration of lessons learned at TMI and any design changes required will be included in the FNP.

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B. RESPONSES TO TMI-2 LESSONS LEARNED TASK FORCE STATUS REPORT AND SHORT-TERM RECOMMENDATIONS, NUREG-0578, AS MODIFIED BY D.B. VASSALLO LETTERS DATED 10/10/79 AND 11/9/79.

Recommendation 2.1.1: Emergency Power Supply Requirements for the Pressurizer Heaters, Power-Operated Relief and Block Valves, and Pressurizer Level Indicators in PWR's

Statement of NRC 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 offsite power, the following positions shall be implemented:

Pressurizer Heater Power Supply

1. The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite 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 efficient capacity for the connection of the pressurizer heaters.

3. The time required to accomplish the connection of the preselected 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.

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

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

2. Motive and control components associated with the PORV block valves shall be capable of being supplied from either the

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offsite power source or the emergency power sources when the offsite power is not available.

3. Motive and control power connections to the emergency buses for the PORVs 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. These buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

Offshore Power Systems Response

1. Pressurizer heaters

The total pressurizer heater capacity for the FNP is 1800 KW. Four separate backup heater groups (346 KW each) are supplied directly from 4 independent and redundant safety class 480 V switchgear buses. Each bus is supplied from its respective standby diesel generator following a loss of offsite power. The control group (416 KW) is supplied from a non-safety class 480 V bus which could be supplied from a diesel-generator bus within several minutes following a loss of offsite power, in the unlikely event that this should become necessary.

Each independent backup group is large enough to maintain natural circulation in the hot standby condition.

The Class 1E circuit breakers supplying each of the backup groups are tripped open on either a safety injection (SI) or loss of offsite power actuation signal.

The heaters can be manually loaded onto the bus from the main control board after SI is reset and loads required in the initial stages of the incident are no longer required. Sufficient diesel generator capacity is provided to supply the

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minimum required number of heaters in the time required. Diesel generator instrumentation is provided to prevent overloading a diesel generator with these heater loads.

OPS will provide the owner with the necessary procedures for energizing the pressurizer heaters, including procedures that might be required for load shedding.

2. Power Operated Relief Valves (PORV's)

Each PORV is supplied with operating air from a separate Safety Class-3 air system which is available following a loss of offsite power. Each PORV pilot solenoid is supplied from independent and redundant 125V DC sources, which are also available following a loss of offsite power. The PORV's are controlled from the main control board. Both PORV's fail closed on loss of motive or control power.

3. PORV Block Valves

The PORV block valves are supplied from motor control centers which are readily energized from a corresponding standby diesel generator following a loss of offsite power. The PORV block valves are controlled from the main control board. Thus the PORV block valves can also be operated following a loss of offsite power.

4. Pressurizer Level Indication Channels

All of the pressurizer level indication channels are derived (and isolated) from their respective protection channels. The instrument loop power supplies for these protection channels (including the isolated outputs) are supplied from their respective Class 1E Instrument buses. Thus level indication is available following a loss of offsite power.

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Recommendation 2.1.2: Performance Testing for BWR and PWR Relief and Safety Valves

Statement of 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. The licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualificatiion shall include qualification of associated control circuitry piping and supports as well as the valves themselves.

Offshore Power Systems Response

OPS considers that the integrity and functionability of Reactor Coolant System relief and safety valves should be verified through a combined industry effort rather than an individual vendor effort. This should circumvent the need for redundant testing of specific valve types. The Westinghouse Owners' Group has provided input via MPR Associates to the EPRI program for valve testing. This input includes valve descriptions and technical parameters, valve actuation transient characterizations, and qualification program recommendations. It is expected that the Westinghouse information will be included in the EPRI program to be submitted to the NRC on or about 1/1/80. Reactor Coolant System relief and safety valves which have been qualified under a testing program will be used in the FNP.

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Recommendation 2.1.3.a: Direct Indication of Power-Operated Relief Valve and Safety Valve Position for PWR's and BWR's

Statement of NRC Position

Reactor 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.

Offshore Power Systems Response

Positive indication of pressurizer relief valve position is currently provided in the FNP design. Such indication is accomplished in the following manner:

1. Each PORV has indication lights on the control board which are activated by stem-actuated limit switches. In addition, a position disagreement light/alarm prominently displays a failure of the PORV to achieve the last position commanded.

2. The temperature downstream of the PORVs and safety valves is displayed on the control board and high temperature alarms are provided.

3. The pressurizer relief tank has temperature, pressure and fluid level indication and alarms on the main control board.

4. High pressurizer pressure alarms in the Control Room.

OPS is presently evaluating methods to provide safety valve position indication. During FNP final design, safety valve position indication, meeting the requirements of this recommendation, will be provided.

Recommendation 2.1.3.b: Instrumentation for Detection of Inadequate Core Cooling for PWRs and BWRs

Statement of NRC Position:

1. Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation" (see Section 2.1.9 of this appendix).

   In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that is not to be used exclusive of other related plant parameters.

2. Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. 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.

Offshore Power Systems Response

The portion of this recommendation dealing with procedures using "existing" instrumentation does not apply to the FNP. Procedures will be developed for the instrumentation provided in the final design.

A primary coolant saturation meter will be installed and will provide on-line indication of coolant subcooled conditions. In addition, the FNP will include instrumentation necessary to provide an unambiguous indication of inadequate core cooling. OPS proposes to evaluate options presently being developed by the Westinghouse Owners' Group before deciding on the specific means of saturation and core cooling monitoring.

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Westinghouse has performed initial analyses to specify the instruments available and guidelines for detection of inadequate core cooling. The core exit thermocouples were identified as the appropriate instruments for determining inadequate core cooling. Utilities with operating plants are using the analyses in developing their emergency procedures and retraining their operators prior to January 1, 1980 as required by NRC. Westinghouse has also been authorized by the Owners' Group to perform additional, more detailed analyses of inadequate core cooling for completion during the first quarter of 1980.

Recommendation 2.1.4: Containment Isolation Provisions for PWR's and BWR's

Statement of NRC Position

1. All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

2. All plants shall give careful reconsideration to the definitior of essential and non-essential systems, shall identify each system determined to be essential, shall identify each system determined to be non-essential, shall describe the basis for selection of each essential system, shall modify their containment isolation designs accordingly, and shall 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.

Offshore Power Systems Response

The current Floating Nuclear Plant containment isolation design satisfies all of the provisions of the NRC recommendations as follows:

1. Phase A isolation (T signal) results in the isolation of all non-essential systems penetrating the containment with the exception of component cooling water lines to the reactor coolant pumps and the lower compartment coolers which are closed by Phase B isolation (P signal).

   Phase A isolation provides for diversity in parameters sensed as well as being automatically actuated any time a safety injection signal (S signal) is initiated. Phase A isolation is initiated from the following process variables:

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(a) High steam flow coincident with low steam line pressure or lo-lo T_{AVG}.

(b) High steam line differential pressure

(c) Low pressurizer pressure

(d) High containment pressure

(e) Manual initiation

Phase B isolation is initiated from hi-hi containment pressure or manually. Although it is not automatically generated by diverse means, the P signal can only be generated after the T signal, which is diverse, has been initiated. In addition to initiating Phase B isolation, the P signal also is used to initiate containment spray.

2. Offshore Power Systems has given careful consideration to the systems penetrating the containment which are required to mitigate the consequences of a loss of coolant accident, or any accident calling for containment isolation. The systems which are required to operate following the accidents are as follows:

- Safety Injection System
- Residual Heat Removal System (supply lines to cold legs)
- Containment Spray System (including recirculation sump lines)
- Upper Head Injection System
- Auxiliary Feedwater System

The above systems are required to supply cooling and/or make up fluid to the Reactor Coolant System, the containment, and the Main Steam System. These systems, or parts of these systems required for post-accident cooling, do not receive any containment isolation signal.

The following systems are not essential to mitigate the consequences of a design basis loss of coolant accident but are considered desirable in assisting in plant recovery from accidents of lower magnitude than a design basis accident. They are not part of Phase A isolation, but instead are isolated by the P signal (Phase B isolation).

- Component Cooling Water System (supply and return lines to RCP thermal barrier cooling)
- Component Cooling Water System (cooling water flow to the lower compartment fan coolers)

The systems determined to be non-essential are isolated by the T signal (Phase A). They are as follows:

- Chemical and Volume Control System
- Post-Accident Sampling System
- Radiation Monitoring System (containment air sample lines)
- Nuclear Sampling System
- Containment Ventilation System
- Post-Accident Containment Ventilation System
- Liquid Waste Treatment System
- Service Air System
- Instrument Air System
- Emergency Air System
- Ice Condenser Refrigeration System
- Non-Essential Service Water System
- Reboiler Condensate Return System
- Reboiler Steam Distribution System
- Fire Protection Water Spray System
- Safety Injection System (test lines)
- Upper Head Injection System (test lines)
- Containment Purge Supply and Exhaust System

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3. All non-essential lines are properly isolated following the initiation of a containment isolation signal. In addition to the systems which are listed as being subject to Phase A isolation, other non-essential systems or lines which penetrate containment have normally closed manual isolation valves, subject to administrative control.

4. Containment isolation reset logic requires deliberate and specific operator action before an isolated line can be re-opened. The following control features are provided for containment isolation valves:

   a. The containment isolation signals override all other automatic control signals.

   b. The valves will remaining in the closed position if the initiating signal is reset.

   c. Each valve can be opened or closed manually after the appropriate containment isolation signals are reset.

   d. Any valves that are normally operated in an automatic mode (for non-safety functions) are also automatically transferred to manual mode by the isolation signal. This precludes automatic opening of containment isolation valves subsequent to reset of the initiating isolation signal.

Recommendation 2.1.5.a: Dedicated Penetrations for External Recombiners or Post-Accident Purge Systems

Statement of NRC Position

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment isolation 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 Part 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

Offshore Power Systems Response

This recommendation does not apply to the Floating Nuclear Plant, because the combustible gas control systems are not external to containment.

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Recommendation 2.1.5.b: Inerting BWR Containments

Statement of NRC Position

It shall be required that the Vermont Yankee and Hatch 2 Mark I BWR containments be inserted in a manner similar to other operating BWR plants. Inerting shall also be required for near term OL licensing of Mark I and Mark II BWRs.

Offshore Power Systems Response

This recommendation does not apply to the Floating Nuclear Plant which uses a pressurized water reactor.

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Recommendation 2.1.5.c: Capability to Install Hydrogen Recombiner at Each Light Water Reactor Plant

Statement of NRC Position

The majority opinion of the Lessons-Learned Task Force is the following, "...it is the conclusion of the majority of the Lessons Learned Task Force that provisions for the post-accident installation of recombiners should not be required as a short-term action. Such consideration should be part of the long-term reconsideration of the design basis for combustible gas control systems.

Offshore Power Systems Response

The present FNP design includes recombiners permanently installed within containment. The subject of a combustible gas control design basis is addressed in NUREG-0585 (see recommendation 10 of Section D, following).

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Recommendation 2.1.6.a: Integrity of Systems Outside Containment Likely to Contain Radioactive Materials (Engineered Safety Systems and Auxiliary Systems) for PWRs and BWRs

Statement of NRC Position

Applicants and licensees shall immediately 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 a frequency not to exceed refueling cycle intervals.

Offshore Power Systems Response

Most systems which interface with reactor coolant, either directly or indirectly, will be isolated for accidents which release significant radioactivity to the coolant. The exceptions [1] are:

1. Residual Heat Removal System (RHR)
2. Safety Injection System (SIS)
3. Containment Spray System (CSS)

(1) There is a potential indirect interface through safeguards area sumps between the RCS and the liquid waste treatment system (See the response to recommendation 2.1.6.b).

For these and other potentially radioactive systems, numerous design features are incorporated in the FNP which minimize the potential for leakage. These include careful component selection, proper orientation of valve stems on normally closed valves, and use of valve backseats and/or piped valve leakoffs as appropriate. Failure analyses and reliability evaluations of safety class systems serve to identify potential leakage paths during the design stage.

As a second line of defense, piping configurations, ventilation systems, floor drains, etc. are designed to minimize the effects of leakage should it occur despite all precautions. This is discussed further in Section 11.6 of the PDR, "Radioactive System Layout, Operation, Maintenance, and Design Considerations." The designs of the FNP safeguards areas and pipe chases connecting them with containment are important features in limiting the spread of contamination as described in the Response to Item 2.1.6.b.

Development and implementation of periodic leak testing programs are the responsibility of the plant owner. None-the-less OPS will carefully review testing programs as they are established by operating utilities and results of generic studies such as those performed by the Westinghouse Owners' Group. The FNP design will be modified as necessary to accommodate upgraded leakage testing requirements so identified.

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Recommendation 2.1.6.b: Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used In Post Accident Operations

Statement of NRC Position

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 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.

Offshore Power Systems Response

Post-accident release of radioactivity, as described in Regulatory Guide 1.4, has been used to derive source terms for the current design of the FNP shielding around fluid and ventilation systems that may contain highly radioactive fluids or gases as a result of accidents. The existing design includes provision for access to RHR equipment for maintenance following such an accident since long term post-accident operation of this equipment must be assured. Following is a more detailed summary discussion of the current FNP post-accident design basis and design features. As part of the detailed design, a comprehensive design review will be conducted to insure that systems which may contain highly radioactive fluids or gases following an accident meet the provisions of this recommendation.

Most of the systems which normally interface with the Reactor Coolant System (either directly or indirectly) will be isolated from the

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Reactor Coolant System following an accident in which significant quantities of radioactivity are released. Release of radioactivity is considered potentially significant if concentrations in the reactor coolant are greater than those associated with 1% failed fuel under normal operating conditions. Those systems which will be isolated from the reactor coolant are the following:

1. Gaseous Waste Treatment System (WTG)
2. Sampling System (SSR)
3. Chemical and Volume Control System (CVC)
4. Boron Recycle System (BRS)

   and

5. Liquid Waste Treatment System (WTL)[1]

The only systems interfacing with reactor coolant which are not isolated are:

1. Residual Heat Removal System (RHR)
2. Safety Injection System (SIS)

   and

3. Containment Spray System (CSS).

These three systems (piping and components) are located within four, separate, shielded safeguards compartments in the FNP design.

Shielding thickness for spaces in which these systems are located were calculated employing a source derived from Regulatory Guide 1.4. The source term included 50% of the core equilibrium halogen inventory and 1% of all other fission products uniformily mixed in the containment sump water inventory. Noble gases were not included in the fluid sources used for design of shielding for these spaces. The sources employed are documented in Table 12.1.4 of the PDR.

(1) Potential indirect interface through safeguards area sumps.

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The dose rate criterion for shielding of these systems in safeguards compartments is that the dose in occupied areas outside the shield walls not exceed 3 Rem for an 8 hour exposure begining at 24 hours after an accident. Access to these spaces at earlier times is not expected to be necessary.

This dose criterion (< 3 Rem for an 8 hour exposure one day after the accident) is the post-accident shield design criterion for all post-accident work locations on the plant except for the control room and emergency relocation area. For the control room and emergency relocation area, the criterion used for shield design is that of General Design Criterion 19 which is that the dose to personnel inside those spaces be less than 5 Rem for the duration of the accident. Source terms for analysis of the control room and emergency relocation area are based on Regulatory Guide 1.4 radioactivity release assumptions.

The RHR, SIS and CSS system components within each safeguards compartment are located in a subcompartment which is isolated from the rest of the safeguards compartment during normal operation. Ventilation is provided by a sealed system such that neither supply nor exhaust air lines communicate the subcompartment to the surrounding space. In the event of an accident resulting in containment isolation, subcompartment exhaust is lined up to the Annulus Filtration System (AFS). The AFS maintains the subcompartment at a negative pressure, thus assuring that any airborne radioactivity released within the subcompartment is exhausted to the annulus, where it passes through charcoal and HEPA filters before release to the environment. Because of this unique design, liquid leaks from the SIS, RHR or CSS systems will not result in release of airborne radioactivity within the surrounding spaces in a safeguards compartment.

Special consideration will be given during final design to post-accident handling of fluids leaking from pumps in the RHR-SIS-CSS subcompartments. In the event of a large leak, recirculation flow

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from the containment sump to the affected subcompartment can be terminated by closing the appropriate sump isolation valve. These valves are motor operated with the motor outside the shield wall. The operator is connected to the valve via a reach rod. Manual valve wheels are also provided at the operator so that the valve may be closed even in the event of motor operator failure.

The FNP has been designed so that post-accident maintenance may be performed on either of the two RHR pumps by draining and flushing the RHR equipment. Drain and flush operations can be performed via reach rod operated valves located outside the shield walls of the RHR pump rooms. Airborne activity released to the RHR subcompartment would be swept out by the annulus ventilation system which maintains a negative pressure in the room. Additionally, the design basis for equipment important to safety includes a requirement for satisfactory operation following post-accident radiation exposure. The integrated exposure to safety equipment, which is calculated using the source term identified above, is a part of the equipment specification.

To summarize, the existing design philosophy for controlling radioactive water and airborne activity following an accident involving core damage is to isolate all systems which could remove radioactive water or air from either the containment or the Reactor Coolant System. Systems outside the containment which are needed following an accident for core cooling or containment atmosphere cooling are located within shielded subcompartments, which are part of each separate safeguards compartment. These subcompartments are maintained at a negative pressure and are connected to the annulus following an accident. Source terms specified in Regulatory Guide 1.4 were used for design of shielding for post-accident work locations near systems which could potentially contain highly radioactive water.

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Recommendation 2.1.7.a: Automatic Initiation of the Auxiliary Feedwater System for PWRs

Statement of NRC 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, the following requirements shall be implemented in the short term:

1. The design shall provide for the automatic initiation of the auxiliary feedwater system.

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

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

4. The initiating signals and circuits shall be powered from the emergency buses.

5. Manual capability to initiate the auxiliary feedwater system 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 a-c motor-driven pumps and valves in the auxiliary feedwater systems shall be included in the automatic actuation (simultaneous and/or sequential) of the loads to 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.

Offshore Power Systems Response

In the current FNP design, as discussed in Section 10.4.6.7.4 of the Plant Design Report, the four motor driven auxiliary feedwater pumps automatically start on lo-lo level in any steam generator, loss of main feed pump, safety injection signal, or loss of offsite AC power. The turbine driven pump starts automatically on lo-lo level in any two steam generators or loss of offsite power. Automatic initiation signals and circuits for the Auxiliary Feedwater System are Class 1E

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and can be tested on-line. Manual capability of initiation of the Auxiliary Feedwater System is provided in such a manner that no single failure will result in loss of the system function. No single failure of the automatic initiation circuitry will prevent manual initiation of the Auxiliary Feedwater System from the Control Room.

Recommendation 2.1.7.b: Auxiliary Feedwater Flow Indication to Steam Generators for PWRs

Statement of NRC Position

Consistent with satisfying the requirements set forth in GDC 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 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 Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

Offshore Power Systems Response

Auxiliary feedwater flow channels, with an accuracy of the order of ±10%, will be Class 1E and displayed on the main control board. Each channel of flow instrumentation is powered from its respective Class 1E instrument power supply.

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Recommendation 2.1.8.a: Improved Post-Accident Sampling Capability

Statement of NRC Position

A design and operational review of the reactor coolant and containment atmosphere sampling system 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 Rems 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 (less than 2 hours) certain radioisotopes 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 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.

Offshore Power Systems Response

During the final design of the FNP, Offshore Power Systems will perform design and operational reviews of the Reactor Coolant and Containment Post-Accident Sampling Systems to determine their capability to meet post-TMI requirements.

Westinghouse is working with the Owners' Group to develop procedures to obtain and prepare samples for analysis. The effort will result in

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recommendations regarding the application of automatic or in-line analyzers, and alternative manual analysis procedures.

As described in 9.3.2.1 of the Plant Design Report (PDR), the NSSS Sampling System provides means to obtain representative liquid and gas samples from various fluid systems for chemical and radiochemical laboratory analysis. The sampling system is designed for manual and intermittent operation for conditions ranging from full power to cold shutdown. Access to the containment building is not required for sampling. The sampling system is not required currently to function during an emergency. In the event of an accident, all lines of the system penetrating containment are isolated by the T Signal. Once the T Signal is removed, the isolation valves can be remote-manually opened.

As described in Section 9.3.2.2 of the PDR, the Containment Post-Accident Sampling System currently is designed to provide representative samples of the containment post-accident atmosphere within 24 hours after the accident. The sampling system is an engineered safety feature. Lines penetrating the containment are closed by isolation valves at all times unless remote-manually opened to take a sample.

These current capabilities will be upgraded to conform to the results of the design and operational reviews and to incorporate inputs from Owners' Group Activities.

1677 304

Recommendation 2.1.8.b: Increased Range of Radiation Monitors

Statement of NRC Position

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident," which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near-term.

1. 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 to be necessary to cover the ranges of interest.

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

    b. Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (ALARA) concentrations to a maximum of 10^5 μ Ci/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.

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

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

Offshore Power Systems Response

Noble gas effluent monitors will be provided on the Floating Nuclear Plant at potential release points. These monitors will comply with this recommendation.

A method for monitoring radioiodine effluents will be determined as part of the plant final design.

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The current FNP design for the redundant containment area monitors specifies a range of 10^{-1} to 10^{7} Rad/Hr. In order to comply with Recommendation 2.1.8.b, this range will be changed to 10^{0} to 10^{8} Rad/hr. It should be noted that these detectors for the FNP design are mounted on the outer surface of the steel containment but may be considered as "In-containment" relative to compliance with this recommendation. The attenuation of the steel shell will be factored into the calibration of the monitors. Mounting the detectors outside the steel containment serves two safety related purposes: 1) the need for containment cable penetrations is eliminated, and, 2) the monitors will experience less severe postulated accident environmental conditions, (i.e., temperature, humidity, and pressure).

Recommendation 2.1.8.c:Improved In-Plant Iodine Instrumentation

Statement of NRC 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.

Offshore Power Systems Response

Sampling methods, counting equipment and other laboratory analytical equipment will be specified and procured by the plant owner. Offshore Power Systems will provide space, including space for counting rooms and laboratories, where analytical determination of radioiodine concentrations can be performed. The location and design of these spaces are such as to permit personnel occupancy for times required to perform necessary analysis following accident conditions including those specified in NRC position 2.1.8.a. Shielding will be provided to insure a low background in the counting room. Ventilation with clean air at a pressure higher than surrounding spaces will be provided for the counting room to minimize background airborne contamination in this region. Capability for purging of entrapped noble gases from charcoal samples using either clean air or nitrogen will be provided in the laboratory area. Residual noble gases will be routed to and vented from the plant stack.

A commercially available method for discriminating between residual noble gases and radioiodine absorbed on the charcoal filters in the atmospheric sampling devices is counting of the charcoal filters with a gamma ray spectrometer. OPS will recommend to the plant owner that such equipment be procured for analysis of the charcoal filters used for sampling of areas within the facility. OPS will also recommend to the utility owner that portable sampling devices be procured and available for sampling of occupied spaces within the facility for radioiodine following accidents.

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Recommendation 2.1.9: Analysis of Design and Off-Normal Transients and Accidents

Statement of NRC Position

Analyses, procedures, and training addressing the following are required:

1. Small break loss-of-coolant accidents;
2. Inadequate core cooling; and
3. Transients and accidents.

Some analysis requirements for small breaks have already been specified by the Bulletins and Orders Task Force. These should be completed. In addition, pretest calculations of some of the Loss of Fluid Test (LOFT) small break tests (scheduled to start in September 1979) shall be performed as means to verify the analyses performed in support of the small break emergency procedures and in support of an eventual long term verification of compliance with Appendix K of 10 CFR Part 50.

In the analysis of inadequate core cooling, the following conditions shall be analyzed using realistic (best-estimate) methods:

1. Low reactor coolant system inventory (two examples will be required - LOCA with forced flow, LOCA without forced flow).

2. Loss of natural circulation (due to loss of heat sink).

These calculations shall include the period of time during which inadequate core cooling is approached as well as the period of time during which inadequate core cooling exists. The calculations shall be carried out in real time far enough that all important phenomena and instrument indications are included. Each case should then be repeated taking credit for correct operator action. These additional cases will provide the basis for developing appropriate emergency procedures. These calculations should also provide the analytical basis for the design of any additional instrumentation needed to provide operators with an unambiguous indication of vessel water level and core cooling adequacy (see Section 2.1.3.b in this appendix).

The analyses of transients and accidents shall include the design basis events specified in Section 15 of each FSAR. The analyses shall include a single active failure for each system called upon to function for a particular event. Consequential failures shall also be considered. Failures of the operators to perform required control manipulations shall be given consideration for permutations of the analyses. Operator actions that could cause the complete loss of function of a safety system shall also be considered. At present, these analyses need not address passive failures or multiple system failures in the short term. In the recent analysis of small break LOCAs, complete loss of auxiliary feedwater was considered. The complete loss of auxiliary feedwater may be added to the failures

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being considered in the analysis of transients and accidents if it is concluded that more is needed in operator training beyond the short-term actions to upgrade auxiliary feedwater system reliability. Similarly, in the long term, multiple failures and passive failures may be considered depending in part on staff review of the results of the short-term analyses.

The transient and accident analyses shall include event tree analyses, which are supplemented by computer calculations for those cases in which the system response to operator actions is unclear or these calculations could be used to provide important quantitative information not available from an event tree. For example, failure to initiate high-pressure injection could lead to core uncovery for some transients, and a computer calculation could provide information on the amount of time available for corrective action. Reactor simulators may provide some information in defining the event trees and would be useful in studying the information available to the operators. The transient and accident analyses are to be performed for the purpose of identifying appropriate and inappropriate operator actions relating to important safety considerations such as natural circulation, prevention of core uncovery, and prevention of more serious accidents.

The information derived from the preceding analyses shall be included in the plant emergency procedures and operator training. It is expected that analyses performed by the NSSS vendors will be put in the form of emergency procedure guidelines and that the changes in the procedures will be implemented by each licensee or applicant.

In addition to the analyses performed by the reactor vendors, analyses of selected transients should be performed by the NRC Office of Research, using the best available computer codes, to provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons together with comparisons to data, including LOFT small break test data, will constitute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures.

Offshore Power Systems Response

The objective of this recommendation is to improve the performance of reactor operators during transient and accident conditions. Offshore Power Systems is maintaining cognizance of the work being performed by Westinghouse, through the Westinghouse Owners' Group, which is pertinent to the Floating Nuclear Plant. These activities are described below:

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1. Small break LOCA analyses have been performed, documented in WCAP-9600/9601, "Report on Small Break Accidents for Westinghouse NSSS Systems," and submitted to the NRC (Bulletins and Orders Task Force) on June 29, 1979. This report presents a comprehensive study of Westinghouse system response to small break LOCAs. The Bulletins and Orders Task Force issued one set of questions on WCAP-9600/9601 on August 13, 1979. These questions were responded to by Westinghouse in September 1979.

   Included in the Westinghouse scope of work authorized by the Owners' Group is a complete review and rewriting of the Westinghouse generic emergency operating instructions. Preliminary sets of emergency operating instructions E-0 (Immediate Actions and Diagnostics) and E-1 (Loss of Reactor Coolant) were included in WCAP-9600/9601. These preliminary instructions have undergone NRC and Owners' Group review and were subsequently revised and finalized in November 1979. These finalized instructions can be utilized by utilities in developing emergency procedures and training programs as required by the NRC.

2. An initial analysis of inadequate core cooling utilizing the W-FLASH Computer Code has been performed and was submitted to the Owners' Group and the Bulletins and Orders Task Force on October 31, 1979. This analysis basically concludes that the core exit thermocouples can be used for detection of inadequate core cooling and contains a preliminary set of guidelines describing necessary operator actions for the detection and mitigation of inadequate core cooling. This information is being utilized by utilities in the development of emergency procedures and training programs.

   Westinghouse has also been authorized by the Owners' Group to perform additional, more detailed analyses of inadequate core cooling which are scheduled for completion during the first quarter of 1980. These analyses will utilize the NOTRUMP Computer Code and investigate a spectrum of scenarios and

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subsequent operator actions. The information available from the NOTRUMP analyses is expected to be more realistic than that obtainable from W-FLASH and should lead to a better understanding of inadequate core cooling and additional guidance to the operator.

3. The purpose of the transient and accident analyses requirement is to provide an increase in safety by improving the performance of reactor operators during transient and accident conditions. The primary concern is that the operator training and emergency operating procedures currently in use are based on the conservative SAR Chapter 15 type analyses. Chapter 15 should continue to be used for design basis analyses since these show the most limiting initial approach to both core thermal and system overpressurization safety limits. Westinghouse is performing a qualitative study for the Owners' Group to assess the information presented to the operator. The study will include an evaluation of the effects of operator actions (correct or incorrect) where information presented may cause the operator to take such actions. This study is scheduled to be completed in December 1979 and it is expected that the results of this study will be incorporated into utility operating procedures and training programs as appropriate.

   What is needed to meet the intent of this recommendation in the long-term is to determine the consequences using realistic assumptions (better estimate modeling) incorporating the effects of the following:

   i. Operator's failure to act when required.
   ii. Operator's inappropriate actions during an accident.
   iii. Additional failures.
   iv. Selected system operations (e.g., re-starting of RCPs etc.)

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The results of these analyses can be used to evaluate information available to the operator and the adequacy of existing procedures. Appropriate changes can be incorporated into the existing procedures, designs, and training programs. Development of the models to incorporate such effects is in itself a long-term effort before detailed analyses can be run. Significant interaction between industry and the NRC is required to agree on the assumptions, bases, appropriate actions (correct or incorrect) to be modeled, and best estimate boundary conditions. When completed, the analyses results using the better estimate modeling tools can enhance the current operator training programs by providing additional insight into the course of events the operator will likely encounter during a transient. A schedule for such long-term analyses has not yet been developed.

4. Westinghouse performed a LOFT L3-1 pre-test small break prediction and the results were submitted to the Owners' Group and the NRC on December 15, 1979.

Both operator training and the development of operating procedures are the responsibility of the plant owner/operator. Westinghouse and Offshore Power Systems can, at the option of the plant owner, provide substantial assistance in the areas of operator training and procedure development. Well before a prospective FNP owner will reach the Operating License stage, the accident and transient analyses cited above will be complete and the results factored into standard plant procedures and operator training programs.

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Recommendation 2.2.1.a: Shift Supervisor's Responsibilities

Statement of NRC Position

1. The highest level of corporate management of each license shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

2. Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

   a. The responsibility and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

   b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.

   c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly specified.

3. Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety.

4. The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

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Offshore Power Systems Response

This recommendation applies to the plant owner and not to the activities conducted by Offshore Power Systems under a License to Manufacture Floating Nuclear Plants. Ample time exists to implement the stated requirements before an FNP owner will require an Operating License.

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Recommendation 2.2.1.b: Shift Technical Advisor

Statement of NRC Position

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The shift technical advisor 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 shift technical advisor 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 shift technical advisors that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

Offshore Power Systems Response

This recommendation applies to the plant owner and not to the activities conducted by Offshore Power Systems under a License to Manufacture Floating Nuclear Plants. Ample time exists to implement the stated requirements before an FNP owner will require an Operating License.

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Recommendation 2.2.1.c: Shift and Relief Turnover Procedures

Statement of NRC Position

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:

   a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

   b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console (what to check and criteria for acceptable status shall be included on the checklist);

   c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

2. Checklists or logs shall be provided for completion by the offgoing and oncoming auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance of test that by themselves could degrade a system critical to the prevention and mitigation of operational transients (what to check and criteria for acceptable status shall be included on the checklist); and

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments).

Offshore Power Systems Response

This recommendation applies to the plant owner and not to the activities conducted by Offshore Power Systems under a License to Manufacture Floating Nuclear Plants. Ample time exists to implement the stated requirements before an FNP owner will require an Operating License.

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Recommendation 2.2.2.a: Control Room Access

Statement of NRC Position

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g. operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implent an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access.

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including thhose who report to stations outside of the control room.

Offshore Power Systems Response

This recommendation applies to the plant owner and not to the activities conducted by Offshore Power Systems under a License to Manufacture Floating Nuclear Plants. Ample time exists to implement the stated requirements before an FNP owner will require an Operating License.

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Recommendation 2.2.2.b: Onsite Technical Support Center

Statement of NRC Position

Each operating nuclear power plant shall maintain an onsite technical support center 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 technical support center.

Records that pertain to the as-built conditions and layout of structures, systems and components shall be stored and filed at the site and accessible to the technical support center under emergency conditions. Examples of such records include system descriptions, general arrangements drawings, piping and instrument diagrams, piping system isometrics, electrical schematics, wire and cable lists, and single line electrical diagrams. It is not the intent that all records described in ANSI N45.2.9-1974 be stored and filed at the site and accessible to the technical support center under emergency conditions; however, as stated in that standard, storage systems shall provide for accurate retrieval of all pertienet information without undue delay.

Offshore Power Systems Response

The onsite Technical Support Center (TSC) for the FNP consists of the supervisor's office and visitors area adjacent to the Control Room. This center is provided with the same degree of shielding, environmental control, missile protection and security as the Control Room. This center uses the same ventilation system as the Control Room and also utilizes the Control Room radiation monitoring equipment. Necessary communication between the TSC and both the Control Room and onsite operational support center will be provided. Offsite communications will be provided by the owner. Plant status can be readily obtained in the TSC during normal as well as emergency cperation. Necessary "as-built" documentation will be filed in the TSC or elsewhere within the shielded control building.

The TSC is directly adjacent to the Control Room and access is through a doorway directly into the Control Room. Additionally, a

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glass window in the common wall between the TSC and Control Room provides for easy observation of recovery activities. For these reasons, the instrumentation requirement for the TSC is minimized. Therefore, Offshore Power Systems is presently considering a CRT terminal to access data from the plant computer. The specific instrumentation required in the TSC will be determined during final detailed design of the FNP.

OPS believes that the FNP concept provides unique advantages regarding as-built documentation, including the following:

a. greater level of detail on drawings (dimensioning, part numbers, etc.) because of the manufacturing concept.

b. greater consistency and coordination among as-built documents, since OPS is ultimately responsible for all as-built documentation for the FNP.

c. FNP units and their documentation would be virtually identical, allowing use of other units for full-scale studies regarding recovery operations.

Recommendation 2.2.2.c: Onsite Operational Support Center

Statement of NRC Position

An area to be designated as the onsite operational support center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to reflect the existence of the center and to establish the methods and lines of communication and management.

Offshore Power Systems Response

The Emergency Relocation Area (at Elev. 100' and 109' in the control building) beneath the Control Room, is provided for this purpose. This area is designed to the same criteria for shielding, missile protection and environmental controls as the Control Room. Emergency storage facilities and communications equipment for onsite operational support are provided. The Emergency Relocation Area is safely accessible from the Control Room via a stairway which is enclosed within the shielded control building.

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Recommendation 2.2.3: Revised Limited Conditions for Operation of Nuclear Power Plants Based Upon Safety System Availability

Statement of NRC Position

All NRC nuclear power plant licensees shall provide information to define a limiting operational condition based on a threshold of complete loss of safety function. Identification of a human or operational error that prevents or could prevent the accomplishment of a safety function required by NRC regulations and analyzed in the license application shall require placement of the plant in a hot shutdown condition within 8 hours and in a cold shutdown condition within 24 hours.

The loss of operability of a safety function shall include consideration of the necessary instrumentation, controls, emergency electrical power sources, cooling or seal water, lubrication, operating procedures, maintenance procedures, test procedures and operator interface with the system, which must also be capable of performing their auxiliary or supporting functions. The limiting conditions for operation shall define the minimum safety functions for modes 1, 2, 3, 4, and 5 of operation.

The limiting conditions of operation shall require the following:

1. If the plant is critical, restore the safety function (if possible) and place the plant in a hot shutdown condition within 8 hours.

2. Within 24 hours, bring the plant to cold shutdown.

3. Determine the cause of the loss of operability of the safety function. Organizational accountability for the loss of operability of the safety system shall be established.

4. Determine corrective actions and measures to prevent recurrence of the specific loss of operability for the particular safety function and generally for any safety function.

5. Report the event within 24 hours by telephone and confirm by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office, or his designee.

6. Prepare and deliver a Special Report to the NRC's Director of Nuclear Reactor Regulation and to the Director of the appropriate regional office of the Office of Inspection and Enforcement. The report shall contain the results of steps 3 and 4, above, along with a basis for allowing the plant to return to power operation. The senior corporate executive of the licensee responsible and accountable for safe plant operation shall deliver and discuss the contents of the report in a public

meeting with the Office of Nuclear Reactor Regulation and the Office of Inspection and Enforcement at a location to be chosen by the Director of Nuclear Reactor Regulation.

7. A finding of adequacy of the licensee's Special Report by the Director of Nuclear Reactor Regulation will be required before the licensee returns the plant to power.

Offshore Power Systems Response

This recommendation applies to the plant owner, and ample time exists to implement the stated requirements before an FNP owner will require an Operating License if such requirements result from the planned rulemaking concerning safety system availability.

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C. RESPONSES TO ADDITIONAL SHORT-TERM REQUIREMENTS: D.B.VASSALLO LETTER DATED OCTOBER 10, 1979

1. NRC Requirement (Enclosure 3): Instrumentation to monitor containment conditions during the course of an accident.

Statement of NRC Position

Consistent with satisfying the requirements set forth in General Design Criterion 13 to provide the capability in the control room to ascertain containment conditions during the course of an accident, the following requirements shall be implemented:

(1) A continuous indication of containment pressure shall be provided in the control room. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and minus five psig for all containments.

(2) 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% hydorgen concentration under both positive and negative ambient pressure.

(3) 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. Also for PWRs, a wide range instrument shall be provided and 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.

The containment pressure, hydrogen concentration and wide range containment water level measurements shall meet the design and qualification provisions of Regulatory Guide 1.97, including qualification, redundancy, and testibility. The narrow range containment water level measurement instrumentation shall be qualified to meet the requirements of Regulatory Guide 1.89 and shall be capable of being periodically tested.

Offshore Power Systems Response

A. Containment Pressure:

The provisions of this recommendation are satisfied in the current FNP design, except that the range currently specified

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for containment pressure is 0-18 psig (approx. 1.15 times design pressure).

In order to adopt this part of the recommendation, two additional wide range containment pressure channels will be incorporated into the FNP. These additional channels will range from minus 5 psig to 60 psig (4 times design pressure), and will be in accordance with this recommendation. The channels will meet the design requirements of Regulatory Guide 1.97, Rev. 1.

B. Containment Hydrogen Concentration:

During FNP final design OPS will select hydrogen monitoring instrumentation which is acceptable to the NRC.

C. Containment Water Level:

As described in Section 6.2.2.7 of the PDR, the Floating Nuclear Plant design does not incorporate a containment sump as such. Instead, the containment lower compartment will collect a sufficient volume of water following the injection phase of safety injection to allow recirculation. Redundant safety grade containment water level (wide range) measurement is currently provided and displayed in the Control Room. The range of these level channels will be increased to cover an elevation equivalent to an 800,000 gallon capacity, a quantity which includes ice melt and UHI accumulator injection.

In addition, Class 1E (narrow range) level channels will be provided for the local WTL sump at the 103 foot elevation in accordance with this recommendation. These channels will also be used as part of the RCS Leak Detection System. These channels will meet the design requirements of Regulatory Guide 1.89.

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2. NRC Requirement (Enclosure 4): Installation of Remotely Operated High Point Vents in the Reactor Coolant System

Statement of NRC Position

Each applicant and licensee shall install reactor coolant system and reactor vessel head high point vents remotely operated from the control room. 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. In particular, these vents shall be safety grade, and shall satisfy the single failure criterion and the requirements of IEEE-279 in order to ensure a low probability of inadvertent actuation.

Each applicant and licensee shall provide the following information concerning the design and operation of these high point vents:

1. A description of the construction, location, size, and power supply for the vents along with results of analyses of loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should be demonstrated to be acceptable in accordance with the acceptance criteria of 10 CFR 50.46.

2. Analyses demonstrating that the direct venting of noncondensible gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment as described in 10 CFR Part 50.44, Regulatory Guide 1.7 (Rev. 1), and Standard Review Plan Section 6.2.5.

3. Procedural guidelines for the operators' use of the vents. The information available to the operator for initiating or terminating vent usage shall be discussed.

Offshore Power Systems Response

Methods of removing hydrogen from the reactor coolant system are discussed in the response to Item 12 (See Section A) including operation of a head vent system. This system will be designed to remove gases from the reactor vessel head via remote manual operations from the Control Room. The reactor vessel head venting system will discharge into the pressurizer relief tank in order to accommodate testing and potential inadvertent releases of water and steam. Additionally the current pressurizer venting capabilities will be upgraded to meet IEEE-279 requirements.

D. RESPONSES TO TMI-2 LESSONS LEARNED TASK FORCE FINAL REPORT, NUREG-0585

Recommendations 1: Personnel Qualifications and Training

1.1 Utility management involvement

1.2 Training programs

1.3 In-plant drills

1.4 Operator licensing

1.5 NRC staff coordination

1.6 Licensed operator qualifications

1.7 Licensee technical and management support

1.8 Licensing of additional operating personnel

Offshore Power Systems Response

Each of these recommendations applies either to the plant owner or to the NRC, and not to the activities conducted by Offshore Power Systems under a License to Manufacture Floating Nuclear Plants. Ample time exists to implement the stated requirements before an FNP owner will require an operating license.

Additional OPS comments are:

1.1 No further comment

1.2 OPS agrees that training of operations personnel needs to be reviewed and upgraded accordingly. However, rather than each utility performing a position task analysis for their own training programs, more benefit could be derived by the industry as a whole establishing minimum training criteria and then each utility upgrading their training programs to at least meet the minimum established standards. This could be accomplished through the Institute of Nuclear Power Operations (INPO).

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1.3 OPS agrees with the concept of in-plant drills as proposed, provided they do not require the manipulation of plant controls to the extent that plant status is affected.

1.4.1 No further comment

1.4.2 Guidelines should be established that clearly define the operator's responsibility concerning the actions he takes while operating a nuclear generating station. Further, these guidelines should outline the review process that will be used should operational errors occur and the actions that could be taken by the NRC to deal with operators commiting operational errors.

1.4.3 Assuming that INPO establishes standards for the training and capabilities of operations personnel, any program devised by the NRC to train or evaluate these personnel should be in agreement with and conform to the established standards.

1.4.4 Same as 1.4.3

1.4.5 No further comment

1.4.6 OPS agrees with the Task Force proposed alternative to Recommendation 6 of SECY 79-330E (Qualification of Reactor Operators).

1.4.7 No further comment

1.5 No further comment

1.6 OPS believes that changes in the qualifications of operations personnel should be made only after careful study and deliberation. The proposed study by INPO and the resulting criteria should be the basis for such changes or recommendations.

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1.7 OPS recommends that INPO set the minimum standards for the capabilities of the utility staff that operates nuclear plants.

1.8 OPS recommends that the subject of the licensing of additional operating personnel be in the charter of INPO.

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Recommendation 2: Staffing of Control Room

Offshore Power Systems Response

This recommendation applies to the NRC and not to the activities conducted by Offshore Power Systems under a License to Manufacture Floating Nuclear Plants. Ample time exists to implement the stated requirements before an FNP owner will require an Operating License. However, OPS agrees that the studies should be conducted to determine manning requirements. These studies should include the following areas:

1. Man-machine interface - how effectively can the plant be diagnosed and controlled by the operators.

2. Operator response time - given a set of conditions how quickly and to what extent must an operator interact with the plant.

3. Personnel qualifications - given a set of conditions what level of capability is required to mitigate the event.

Recommendation 3: Working Hours

Offshore Power Systems Response

This recommendation applies to the plant owner and not to the activities conducted by Offshore Power Systems under a License to Manufacture Floating Nuclear Plants.

Recommendation 4: Emergency Procedures

Offshore Power Systems Response

This recommendation applies mainly to the NRC and not to the activities conducted by Offshore Power Systems under a License to Manufacture Floating Nuclear Plants. Westinghouse has rewritten its generic emergency instructions which will be reviewed and approved by the Westinghouse Owners' Group and the NRC. Offshore Power Systems will monitor the process of emergency procedure review. Lessons learned from these reviews will be applied by Offshore Power Systems during the final design phase when recommended emergency instructions will be prepared for the guidance of the plant owner in preparation of emergency procedures. Ample time exists to implement the stated requirements before an FNP owner will require an Operating License.

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Recommendation 5: Verification of Correct Performance of Operating Activities

Offshore Power Systems Response

Administrative steps contained in this recommendation are within the plant owner's scope of responsibility. The balance of the recommendation deals with plant design features for automatic system status monitoring. The Floating Nuclear Plant presently includes significant provisions for status monitoring; these are outlined in the following paragraphs. Offshore Power Systems will remain abreast of continuing developments in this area, including those by INPO, and particularly those affecting Regulatory Guide 1.47. Should any new requirements arise, they can be addressed during the final design phase.

Assurance of proper operation and/or positioning of safety-related equipment (including equipment in engineered safety features supporting systems) during all operating activities is provided by:

1) Main Control Board (MCB) Display Features: include position/status indicating lights, position/status disagreement indication, availability indication, and system level bypass indication. These features meet or exceed Regulatory Guide 1.47. Some additional criteria are stated in Section 7.5.1 of the PDR. These features are as follows:

   Position/Status Indicating Lights (Backlit Pushbutton) (PIL)

   Backlit red (open) and green (closed) pushbuttons indicate actual valve position from limit switches on the valve. The pushbutton is part of the MCB module for that valve.

   Backlit red (on) and green (off) pushbuttons indicate breaker or contactor status from appropriate auxiliary

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contacts. The pushbutton is part of the MCB module for that component (pump, fan, etc.)

These position/status signals are also inputs (through isolation devices) to the Plant Computer Systems.

Valve Position Indicating Lights (Lights Only) (PIL*)

This valve position signal is also an input to the Plant Computer Systems.

Position/Status Disagreement Light/Alarm (Backlit Pushbutton) (PDL)

A backlit alarm indication/acknowledgement pushbutton (normally extinguished) flashes in conjunction with an audible alarm if the equipment fails to achieve the last position or state commanded. In addition, the commanded position/status indicating light flashes. This backlit pushbutton is part of the MCB module for that equipment.

Both of these flashing lights are acknowledged by this pushbutton, changing the alarm indication pushbutton from flashing to steady, and the commanded PIL from flashing to extinguished. The steady alarm indication light is not extinguished until the commanded and the actual equipment state are in agreement.

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*indicates "Lights Only", see Table D-1

Availability Light/Alarm (Same Backlit Pushbutton as PDL above) (AVL)

If the equipment is removed from service (i.e., if motive power is unavailable or locked out) either deliberately or due to failure, the backlit alarm indication/acknowledgement pushbutton (the same device actuated by the PDL) flashes in conjunction with an audible alarm.

For equipment removed from service, this alarm signal is also an input (through an isolation device) to the Plant Computer System. The Plant Computer System flashes a system level display (BYP) on the MCB indicating that the appropriate system ESF train is bypassed.

System Level Bypass Indication (BYP)

An engraved backlit window, prominently displayed to the operator, is provided for each division of each major Safety Subsystem (e.g., SIS, RHR).

This window flashes whenever any of the following conditions (within the scope of the window) indicates a bypass of a protective action:

a) Motive power unavailable to an ESF actuation device (for example, an MOV, power unavailable to the reversing contactor), due to deliberate bypass or circuit failure. This condition is derived from "AVL" signal. (AVL/BYP)

b) Valve positioned so as to create a bypass of a protective action. This condition is derived from actual valve position. (PIL/BYP)

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c) Window activated manually by operator from MCB, responding to information received through administrative control. (ADM/BYP)

If a redundant division of any subsystem were concurrently placed in a bypass mode (due to any of the above inputs), the second division window would flash and an audible alarm would occur.

Acknowledgement of the first division level bypass causes the first window to change from flashing to steady, until the bypass is cleared. Acknowledgement of the second (concurrent) division level bypass silences the audible alarm, but leaves the second window flashing until one of the bypass conditions is cleared.

The plant computer systems perform the combination and sequence logic that is required to control the system level bypass indication windows. The position/status inputs to the computer that are derived from Class IE control circuits are isolated in accordance with Regulatory Guide 1.75.

The bypass indication system meets or exceeds the requirements of Regulatory Guide 1.47. Additional design criteria for the bypass indication system are provided in Section 7.5.1 of the PDR.

System Level Monitor Indication (MON)

An engraved, backlit window, prominently displayed to the operator, is provided for each division of each major safety subsystem (e.g., SIS, CSS). This window flashes, in conjunction with its corresponding PDL light(s), whenever any equipment (within the scope of the window) has failed to respond to an ESF signal.

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2) Control Circuit Design Features: In addition to these display features, circuit design features are provided to assure proper alignment of equipment. These features include assignment of control priorities to ESF signals and selection of failure modes. These control features are described below.

Control Priority Assignment (CP)

While the equipment is in service (i.e., while motive power is available to it), its control priorities are assigned such that ESF signals will always override non-ESF signals (with the exception of electrical and mechanical circuit protection features which must override ESF signals in order to prevent component damage).

Failure Mode of Actuation Device (FM)

Removal of an air operated or solenoid operated valve from service (i.e., removing motive power) will cause the valve to move to the safe position.

Administrative Control Input (Manual) to Bypass Indication System (ADM)

The system level bypass indication (BYP) can be manually input by the operator through administrative control. Computer software supplements plant administrative controls by tracking these manual inputs (together with non-manual inputs), determining the system level effects, and providing appropriate displays.

3) Owner's Administrative Controls and Procedures: The design features described above will supplement and enhance the Owner's administrative control program. The Owner's administrative control program should be the first line of defense against improper operation.

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Table D-1 illustrates the specific application of these design features to the generic types of FNP equipment that could be incorrectly operated. The table indicates which of the FNP control and display design features provide direct defense against:

a) The effects of mispositioned circuit breakers or contactors

b) The effects of mispositioned valves, or

c) Undetected mispositioning of equipment for various conditions of plant operation and for various types of equipment.

Considered in the table are:

a) The nature of the safety system bypass (deliberate vs. inadvertant)

b) The plant operating mode (periodic test, maintenance, etc.)

c) The engineered safety features systems mode (standby vs. active)

d) The type of safety equipment (circuit breaker, motor operated valve, hand operated valve, etc.)

Table D-1 does not address any FNP design features that are not relevant to safety consequences, nor is credit taken for other types of design features (e.g., process alarms) that in some cases would further enhance safety.

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TABLE D-1, SHEET 1

DESIGN FEATURES THAT VERIFY CORRECT PERFORMANCE OF OPERATING ACTIVITIES

NATURE OF BYPASS AND ESF STATUS / TYPE OF SAFETY-RELATED EQUIPMENT	CKT BRKRS. OR CONTACTORS OPERABLE FROM MCB	VALVES OPERABLE FROM MCB: ACTUATED BY ESF SIGNAL: NOT ALIGNED IN "SAFE" POSITION AT POWER — MOV's	AOV's	VALVES OPERABLE FROM MCB: NOT ACTUATED BY ESF SIGNAL: ALIGNED IN "SAFE" POSITION AT POWER — MOV's	AOV's	HAND OPERATED VALVES (NOTE 3): OPERATED PERIODICALLY AT POWER (> ONCE/YR)	OPERATED INFREQUENTLY AT POWER (≤ 1/YR)	OPERATED AT STARTUP/ SHUTDOWN ONLY	OPERATED AT REFUELING ONLY
DELIBERATE BYPASS: ESF IN STANDBY ·On-Line Periodic Test of ESF	CP	CP	CP	PIL/BYP PIL ADM/BYP	PIL/BYP PIL ADM/BYP	(Note 1) PIL*/BYP PIL* ADM/BYP	ADM/BYP		
·On-Line Non-Routine Maintenance (ESF Equipment Temporarily Removed From Service For Repair)	AVL/BYP ADM/BYP	AVL/BYP ADM/BYP	FM AVL/BYP ADM/BYP	PIL/BYP AVL/BYP PIL ADM/BYP	FM PIL/BYP AVL/BYP PIL ADM/BYP	N/A	ADM/BYP		
INADVERTENT BYPASS: ESF IN STANDBY, PLANT AT POWER ·Plant Operator Error (Note 2)	CP	CP	CP	PIL/BYP PIL	PIL/BYP PIL	(Note 1) PIL*/BYP PIL*	(Note 1)	(Note 1) PIL*/BYP PIL*	(Note 1) PIL*/BYP PIL*
·Field Operator Error: Hand Repositioning	N/A	CP	CP	PDL PIL/BYP PIL	PDL PIL/BYP PIL	(Note 1) PIL*/BYP PIL*	(Note 1)	(Note 1) PIL*/BYP PIL*	(Note 1) PIL*/BYP PIL*
·Field Operator Error: Removal from Service	AVL/BYP ADM/BYP	AVL/BYP	FM AVL/BYP PDL	PIL/BYP AVL/BYP PIL	FM PIL/BYP AVL/BYP PIL	N/A	N/A	N/A	N/A
·Loss of Act. Equip. Motive Power	UNDER-VOLTAGE ALARM AVL/BYP ADM/BYP	AVL/BYP	FM AVL/BYP PDL	AVL/BYP PIL/BYP PIL	FM AVL/BYP PIL/BYP PIL	N/A	N/A	N/A	N/A
·Loss of Act. Device Control Power	AVL/BYP ADM/BYP	AVL/BYP	FM AVL/BYP PDL	AVL/BYP PIL/BYP PIL	FM AVL/BYP PIL/BYP PIL	N/A	N/A	N/A	N/A
·Valve "Drifts" From Position	N/A	CP	CP	PDL PIL/BYP PIL	PDL PIL/BYP PIL	(Note 1) PIL*/BYP PIL*	(Note 1)	(Note 1) PIL*/BYP PIL*	(Note 1) PIL*/BYP PIL*
BYPASS: ESF ACTIVE	PDL/MON PIL	PDL/MON PIL	PDL/MON PIL	PIL/BYP PIL	PIL/BYP PIL	(Note 1) PIL*/BYP PIL*	(Note 1)	(Note 1) PIL*/BYP PIL*	(Note 1) PIL*/BYP PIL*

LEGEND:
- CP — CONTROL PRIORITY
- FM — FAILURE MODE
- PIL — POSITION INDICATION LIGHT (BACKLIT PUSHBUTTON)
- ADM/BYP — ADMINISTRATIVELY CONTROLLED LIGHT
- AVL/BYP — AVAILABILITY LIGHT
- PDL/BYP — POSITION DISAGREEMENT LIGHT
- PIL/BYP — POSITION INDICATION LIGHT

(ADM/BYP, AVL/BYP, PDL/BYP, PIL/BYP:) INPUTS TO SYSTEM LEVEL BYPASS (BYP) INDICATION

- PDL/MON — POSITION DISAGREEMENT INPUT TO SYSTEM LEVEL MONITOR (MON) INDICATION
- * — LIGHTS ONLY

See Sheet 2 for Notes

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TABLE D-1, Sheet 2

NOTE 1. Valves are locked in safe position, and are under administrative control.

NOTE 2. "Operator error" includes failure to recognize a valve that is left improperly positioned (for power operation) following startup.

NOTE 3. Safety-related hand operated process valves that have the capability of significantly degrading a protective action if left mispositioned are subject to the following criteria:

a) If normally operated more frequently than once per year with the plant at power, shall be locked in the safe position under administrative control. In addition, remote position indication shall be provided.

b) If normally operated at startup, shutdown and/or refueling, shall have the provisions of paragraph a).

c) If only operated for non-routine maintenance or repair (e.g., to isolate a pump or heat exchanger for repair) with the plant at power, shall be locked in the safe position under administrative control.

NOTE 4. This table includes only those control and display features that provide <u>direct</u> defense against these conditions, recognizing that others of these features might be provided for a particular component, but would be less relevant.

NOTE 5. Where more than one design feature provides defense, the most prominent one is listed first.

Recommendations 6: Evaluation of Operating Experience

6.1 Nationwide network
6.2 Providing information to operators

Offshore Power Systems Response

These recommendations apply to the NRC Staff and plant owners. However, OPS agrees with the recommendation to review operational data and apply it to the plant operation and personnel training.

Recommendations 7: Man-machine interface

7.1 Control room reviews
7.2 Plant safety status display
7.3 Disturbance analysis systems
7.4 Manual versus automatic operations
7.5 Standard control room design

Offshore Power Systems Response

These recommendations suggest initial actions by the Nuclear Regulatory Commission, possibly leading to new design requirements. Offshore Power Systems will remain abreast of the program of these activities as well as progress in the field of human engineering generally. During final design, the Control Room and control board designs will be developed in consonance with new requirements which may result from these recommendations.

Additional OPS comments are:

7.1 No further comment

7.2 No further comment

7.3 At the present time, designs for computer based diagnostic systems do not exist; however, programs have been initiated by both EPRI and DOE to perform the scoping and feasibility study for a plant wide disturbance and analysis system. Westinghouse has the responsibility for the overall project management of the EPRI program.

OPS recommends that any program to be performed by the Nuclear Regulatory Commission be developed with full recognition of the programs already initiated by EPRI and DOE.

7.4 OPS believes that a joint Nuclear Regulatory Research/nuclear industry effort is the most effective means for satisfying this recommendation. The effort that is undertaken should recognize that specification of manual and automatic actions for nuclear power plant operation is a function of understanding and predicting every situation to which the plant is subjected. Since it is not possible to predict every potential scenario, there is a need for maintaining flexibility for manual operator actions. This need should be a recognized criterion for the program.

7.5 No further comment

Recommendations 8: Reliability Assessments of Final Designs

Offshore Power Systems Response

This recommendation requires that the NRC initiate a systematic assessment of saftey systems reliability using simplified fault tree analysis techniques. It is assumed that once these analyses are complete, appropriate design requirements would be promulgated. When such design requirements are promulgated appropriate action will be taken to incorporate the requirements in the design of the FNP.

In this respect, a general overall reliability assessment of the FNP design was performed in early 1977 as part of the Liquid Pathways Generic Study. This assessment was reported in an appendix to the FNP Liquid Pathways Generic Study Topical Report, OPS Report 22A60. FNP design features such as improved testability of the interfacing check valves associated with the low pressure injection system, redundancy in the Auxilary Feedwater System design and incorporation of 4 separate diesel generator sets have produced significantly improved reliability for the FNP design when compared with the PWR plant analyzed in WASH-1400.

Reliability assessments for an ice condenser plant recently conducted by the NRC show that the loss of both long-term core cooling recirculation flow and containment spray flow as a result of failure to open the drains between the upper and lower containment compartments after refueling, is a major contributor to risks. Accordingly, OPS intends to take steps to reduce the probability of leaving the drains closed following refueling. Design features such as automatic status indication and more stringent inspection requirements each appear to offer satisfactory results.

We believe that OPS has taken the initiative in keeping abreast of applicable reliability evaluations and applying the available results to the FNP design. We shall continue to utilize this approach as other applicable reliability assessment studies are completed.

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Recommendation 9: Review of Safety Classifications and Qualifications

Offshore Power Systems Response

The recommended evaluation can be performed for the FNP following issuance of the License to Manufacture. This evaluation would be subject to review by the Staff during the final design approval phase (post-ML). Recommendation 9 specifically provides that licensing of new plants need not be delayed pending completion of the specified evaluation. This is particularly appropriate in the case of the FNP, since both the pre-ML technical review and public hearings are substantially complete.

OPS recommends, however, that in order to obtain an orderly resolution of this issue, it is essential that a lead role be identified for establishing a detailed scope for the investigation, and for defining acceptance criteria against which the results of the industry investigation can be evaluated. OPS endorses the implementation of a classification system for electrical components that recognizes varying degrees of component utility in post-accident situations.

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Recommendation 10: Design Features for Core-Damage and Core-Melt Accidents

Offshore Power Systems Response

The NRC position recommends NRC conduct rulemaking regarding design features to mitigate accidents that would result in either core-melt or severe core-damage without core-melt. As NRC is aware, Offshore Power Systems has already committed to incorporate a core ladle in the FNP design to assist in mitigating the consequences of a core melt accident. In addition, a containment vent system has been investigated and incorporation of such a system into the FNP design was shown to be feasible (should it be required by rulemaking). This system would open at an overpressure above the containment design pressure and vent into basin water beneath the platform (See Appendix F of the OPS Generic Liquid Pathways Report, OPS Report 22A60 of 6/77). The air pathways source terms and resulting dose effects would be significantly reduced as a result of fission product adsorption by the basin water.

In addition OPS concurs with the Task Force recommendation that the Commission should define a clear criterion to define the basic safety goal for nuclear power plant regulation (Recommendation 11). Definition of this safety goal must be established prior to development of any new regulatory requirements. We note that the Task Force, however, does not recommend a time table for establishing such a safety goal. Many of the other Task Force recommendations, specifically 8, 9 and 10, are dependent on the definition of a safety goal. Thus, rather than having an initial rulemaking procedure for Recommendation 10 as proposed by the Task Force, we suggest action be taken promptly on defining the safety goal for reactor regulation.

The steps to accomplish this are as follows:

1. A rulemaking should be held to identify the safety goal. The process for this would be to publish the intent of such a rulemaking and to request input from the industry in writing on the content of an initial proposed rule that defines the safety goal. The next steps would then be for the Staff to write the proposed rule, followed by a public hearing. We recommend that the Commission issue, within the next three months, a notice of intent to conduct a rulemaking to solicit comments for the safety goal. The proposed rule would then be published for public comment within one year of the notice of intent. This is the same time table as was proposed for Recommendation 10.

2. With the safety goal defined, the next step would be for the NRC to evaluate the core-damage and core-melt design features itemized under Recommendation 10 in order to determine whether there is a need for a public hearing process. With the safety goal defined, many of the questions identified in Recommendation 10 could be readily answered or eliminated when evaluated with respect to the safety goal. However, proposed design features and questions relating to core-damage accidents may require analysis to define whether there is still a need for a hearing. The results and findings of this NRC evaluation should then be published for industry comment and a rulemaking held, if necessary.

3. The third step in the process would be the determination of whether other design features to improve safety are necessary. Such a study should utilize the results of the work performed under Recommendations 8 and 9.

In summary, the OPS position is that identification of design features for core-damage and core-melt accidents, as given in the Task Force Recommendation 10, cannot be adequately made until a

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safety goal has been defined by the NRC. This should be done as early as possible using the rulemaking process, rather than the NRC suggestion of having a rulemaking for Recommendation 10. It is not logical to try to identify new design features for core-damage and core-melt accidents without having a safety goal upon which to judge the need for additional design features. With the safety goal defined, it may well be that the existing design features coupled with the siting considerations, emergency action plans, and the improved human and operational factors that are currently being developed, will meet the safety goal without the need to incorporate additional safety systems. With respect to any design requirements that may result from the proposed rulemaking, Offshore Power Systems will consider and implement appropriate requirements in the FNP design once the rulemaking has been completed and required implementation (design features and schedule) has been defined.

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Recommendation 11: Safety Goal for Reactor Regulation

Offshore Power Systems Response

No action by Offshore Power Systems is indicated until the basic safety goal for nuclear power plant regulation is defined and resulting design requirements are promulgated in the licensing process. Refer to the OPS Response to Recommendation 10.

Recommendation 12: Staff Review Objectives

Offshore Power Systems Response

This recommendation deals only with internal NRC procedures. Ample time exists for the Staff to implement this recommendation prior to the FNP final design review. We believe this recommendation should be implemented systematically without imposing longer licensing times. In fact, efforts to streamline the licensing process, such as NUREG-0292, "Nuclear Power Plant Licensing: Opportunities for Improvement," should not be abandoned because of the TMI accident.

Recommendation 13: NRR Emergency Response Team

Offshore Power Systems Response

This recommendation requires action only by the NRC Staff and will not affect Staff review of the FNP.

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