3.0 POST-TMI REQUIREMENTS AND RECOMMENDATIONS

Shortly after the initial recovery phases following the March 28, 1979 incident at TMI-2, various task forces and investigating groups were set up (both inside and outside of the NRC) to make recommendations for plant design and operating changes to ensure that a TMI-2 type event or similar event does not happen again. The requirements and recommendations from these task forces and investigating groups were consolidated and documented in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident." This NUREG does not specifically address requirements for new plant designs, since at that time the NRC directed its technical review resources to assuring the safety of operating power reactors rather than the issuance of new licenses or permits.

In mid-1980 the NRC staff initiated a program for Commission approval of a course of action that would lead to the establishment of TMI-2 related requirements for pending construction permit applications. This program led to the issuance of NUREG-0718, Revision 2, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," which specifies those NRC Action Plan (NUREG-0660) items that are required to be implemented or committed to by a pending applicant prior to receiving a construction permit or a license to manufacture. In addition, the NRC has issued a revision to 10CFR 50.34, "Contents of Applications; Technical Information," that essentially incorporates the post-TMI requirements of NUREG-0718 into their regulations.

This revision to 10CFR 50.34 (which is referred to as the CP/ML Rule) is written such that it is applicable to construction permit and manufacturing license applications pending at the effective date of the rule (i.e., February 16, 1982). However, the "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation," (48FR16014, April 13, 1983) indicates that the requirements of the CP/ML Rule are also applicable to new construction permit applications or reactivations. Therefore, applicable post-TMI requirements of NUREG-0718/10CFR 50.34 and certain additional potential requirements from the NRC Action Plan (NUREG-0660) are being addressed in the WAPWR design as indicated in the following sections.

3.1 NUREG-0718/10CFR 50.34 (CP/ML RULE)

The following are the licensing requirements and WAPWR design responses for each NUREG-0718/10CFR 50.34 item that impacts or potentially impacts the WAPWR design.

1. Plant/Site Specific Probabilistic Risk Assessment

   10CFR 50.34(f)(1)(i)

   "Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant."

   Discussion

   Refer to Section 3.2 which has been devoted to the inter-related issues of probabilistic risk assessment, safety goal, and severe accidents.

2. Auxiliary Feedwater System Evaluation

   10CFR 50.34(f)(1)(ii)

   "Perform an evaluation of the proposed auxiliary feedwater system (AFWS), to include: (A) a simplified AFWS reliability analysis using event-tree and fault-tree logic techniques, (B) a design review of AFWS, and (C) an evaluation of AFWS flow design bases and criteria."

   Discussion

   A conventional AFWS functions, in conjunction with a seismic Category I water source, as an emergency system for the removal of heat from the primary system when the main feedwater system is not available. It also plays an important role in mitigating the effects of some design basis

events (e.g., main feedwater line breaks and some small break loss-of-coolant accidents). Existing AFWS designs hold the plant at hot standby, or cool down the primary system to temperature and pressure levels at which the low pressure residual heat removal system can operate. The AFWS can also be used during normal plant startup and shutdown conditions. AFWS designs usually consist of a combination of steam turbine-driven and electric motor-driven pumps.

The WAPWR design is somewhat different than a conventional two electric motor-driven and one steam turbine-driven AFWS design.

The WAPWR design includes an emergency feedwater system (EFWS) and a startup feedwater system (SFWS). The EFWS is a safety system utilizing four pumps; two electric motor-driven and two steam turbine-driven. The EFWS functions similarly to a conventional AFWS except that during normal plant startup/shutdown and hot standby the SFWS is utilized. The EFWS is designed for such events as main steam line breaks, main feedwater line breaks, steam generator tube ruptures, loss-of-coolant accidents, loss of all AC power, and any other event in which the main and startup feedwater systems are not available. The SFWS is a control grade system utilizing one motor-driven pump and provides feedwater during normal plant startup/shutdown and hot standby. The SFWS is also started automatically during reactor trips and other anticipated transients.

The purpose of requirement (A) above is to: (1) assess the reliability of the AFWS design under various loss of feedwater transient conditions, with particular emphasis being given to determining potential failures that could result from human errors, common causes, single point vulnerabilities, and test and maintenance outages, and (2) incorporate design provisions and/or procedural actions as necessary to improve the AFWS reliability relative to the NRC generic AFWS reliabilities published in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants."

A quantitative fault-tree unavailability analysis of the secondary side safeguards systems (SSSS) design for the WAPWR has been performed. The WAPWR design was shown to have a reliability higher than the systems evaluated by the NRC and documented in NUREG-0611.

The purpose of requirement (B) above is to: (1) assess the level of compliance of the AFWS design to the NRC acceptance criteria documented in Standard Review Plan 10.4.9, "Auxiliary Feedwater Systems," and (2) where deviations are identified, modify the AFWS design as necessary to comply with the NRC acceptance criteria or justify the deviations.

The purpose of requirement (C) above is to assure that the design bases and criteria for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are defined and documented.

WAPWR Response

In regard to requirement (A), when the final details of the SSSS design for the WAPWR are established, the reliability reanalysis (discussed above) will be performed. The final SSSS reliability analysis will be submitted to the NRC as part of the licensing process for the WAPWR design.

In regard to requirement (B), Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 10.4.9 acceptance criteria during the licensing process for the WAPWR design.

In regard to requirement (C), Westinghouse will completely document an evaluation of the SSSS flow design bases and criteria during the licensing process for the WAPWR design.

3. Reactor Coolant Pump Seals

10CFR 50.34(f)(1)(iii)

"Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break LOCA with subsequent reactor coolant pump seal damage."

Discussion

Within the design bases of current Westinghouse plant designs, the scenario postulated in this regulation does not present a problem. During normal operation, seal injection from the chemical and volume control system is provided to cool the reactor coolant pump seals and the component cooling water system provides flow to the thermal barrier heat exchanger to limit the heat transfer from the reactor coolant to the reactor coolant pump internals. In the event of a loss of offsite power the reactor coolant pump motor is de-energized, the diesel generators are automatically started, and component cooling water to the thermal barrier heat exchanger and/or seal injection flow is automatically restored within seconds. Either of these cooling supplies is adequate to provide seal cooling and prevent seal failure due to a loss of offsite power.

In addition to the normal seal cooling provided in conventional designs, the WAPWR design includes upgraded seal injection capability which provides an alternate source of seal injection water to the reactor coolant pumps during situations involving the loss of both normal seal injection and thermal barrier cooling. Such situations are beyond the postulated loss of offsite power of the above regulation and involve multiple failures/operator errors or common mode failures. For the WAPWR design, the addition of the upgraded seal injection capability provides added assurance of maintaining seal injection cooling.

WAPWR Response

In relation to this regulation, normal reactor coolant pump seal injection for the WAPWR design is adequate and no additional evaluations will be performed.

4. Automatic PORV Isolation System

10CFR 50.34(f)(1)(iv)

"Perform an analysis of the probability of a small-break LOCA caused by a stuck-open PORV. If this probability is a significant contributor to the probability of small-break LOCA's from all causes, provide a description and evaluation of the effect on small-break LOCA probability of an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened."

Discussion

General Design Criterion 14, "Reactor Coolant Pressure Boundary," of Appendix A to 10CFR Part 50 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. Historically, the application of this criterion has emphasized the integrity of passive components in the reactor coolant system, such as the reactor vessel and the piping, however, this criterion also applies to the valves that provide isolation for the system.

The primary purpose of pressurizer relief and safety valves is that they operate in conjunction with the reactivity control system to limit system overpressure during anticipated operational transients or accidents. The pressurizer relief valves are not part of ASME Code requirements for overpressure protection and, therefore, they can be and are isolatable with remote-operated block valves. The consequence of the failure of the pressurizer relief valves to close is the loss of coolant and depressurization

of the reactor system. This consequence can be mitigated if the remote-operated block valves are closed either automatically or by operator action.

The purpose of this requirement is to evaluate (using probabilistic techniques) the benefit of incorporating an automatic pressurizer PORV isolation system.

Westinghouse (in support of the Westinghouse Owners Group) has performed an evaluation of the benefit of incorporating an automatic pressurizer PORV isolation system for conventional plant designs. This evaluation (which is documented in WCAP-9804, "Probabilistic Analysis and Operational Data in Response to NUREG-0737 Item II.K.3.2 for Westinghouse NSSS Plants") concluded that such a system should not be required. This conclusion was primarily based on the reduction of the already small PORV LOCA probability due to implementation of changes to plant designs subsequent to the TMI-2 event. These changes include both modifications which make PORV challenges less likely and changes in hardware, procedures, and training which provide assurance that the function of PORV isolation will be reliably performed by operator action. As further justification of this conclusion, failure to isolate stuck-open PORVs has been analyzed and the results predict no core uncovery.

The WAPWR also contains several design features which will minimize challenges to the PORVs. First, the charging pumps are independent of the safety injection system and second, the sizing of the pressurizer is such that the PORVs will not open even under a full load rejection.

WAPWR Response

In regard to the WAPWR design, Westinghouse is further evaluating the benefits of an automatic low pressure closing feature for the pressurizer block valves. This feature is being considered in the overall design in accordance with safety-grade cold shutdown and overpressure protection requirements. Inclusion or exclusion of this feature will be completely documented and justified during the licensing process for the WAPWR design.

5. Hydrogen Control Systems Evaluation

10CFR 50.34(f)(1)(xii)

"Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f)(2)(ix) of this section (50.34). As a minimum include consideration of a hydrogen ignition and post-accident inerting system. The evaluation shall include: (A) a comparison of costs and benefits of the alternative systems considered, (B) for the selected system, analyses and test data to verify compliance with the requirements of (f)(2)(ix) of this section (50.34), and (C) for the selected system, preliminary design descriptions of equipment, function, and layout."

Discussion

Refer to item 14 of this section for a discussion of this requirement in conjunction with the requirements of 10CFR 50.34(f)(2)(ix).

6. Simulator Capability

10CFR 50.34(f)(2)(i)

"Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCA's."

Discussion

Beyond the above regulation, 10CFR Part 55, Appendix A, "Requalification Programs for Licensed Operators of Production and Utilization Facilities," permits and encourages the use of simulators for operator training. This is due to the undesirability of imposing additional challenges to the plants protective features that would result if the actual plant is used for training operators to respond to accidents.

The purpose of this NRC requirement is to: (A) require simulator capability, and (B) ensure that the proposed simulator capability for training of operators is performed on a simulator that correctly models the actual plant specific control room design and has the capability to accurately simulate a small-break LOCA.

In addition, the NRC has issued Regulatory Guide 1.149, "Nuclear Power Plant Simulators for Use in Operator Training," which basically endorses ANSI/ANS 3.5-1981, "Nuclear Power Plant Simulators for Use in Operator Training," and describes a method acceptable to the NRC staff for specifying the functional requirements of a nuclear power plant simulator to be used for operator training.

WAPWR Response

This requirement does not impact the WAPWR design. Simulator capability is the responsibility of each utility utilizing the WAPWR design.

7. Plant Procedures

10CFR 50.34(f)(2)(ii)

"Establish a program to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts."

Discussion

The area of operating procedures has received great attention as a result of the TMI-2 event. This attention stems from certain opinions that the severity of the TMI-2 event might have been significantly reduced if the

operating procedures were better written (human engineered and supported by appropriate analyses) and if the operators were better trained in the use of the procedures.

Since the TMI-2 event there have been extensive industry efforts undertaken to improve emergency operating procedures and their use. For example, Westinghouse (in support of the Westinghouse Owners Group) nas: (A) reviewed and revised the generic Westinghouse Emergency Response Guidelines as a result of new small-break LOCA analyses, inadequate core cooling analyses, transient and accident analyses, discussions with the NRC (and subsequent NRC reviews), and inputs from utilities, (B) established a program for additional inputs or revisions to the generic Emergency Response Guidelines as a result of ongoing efforts, and (C) initiated a human factors test of the new Emergency Response Guidelines to determine any problem areas in an operating environment.

As one would expect, these efforts to date have been focused on current-day operating and near-term operating plants. The NRC concern that resulted in the above requirement is that programs for the continued improvement of plant operating procedures should be pursued and coordinated with other industry efforts (e.g., INPO) and other post-TMI related improvements (e.g., safety parameter display systems) in relation to new applications.

Although the generic Westinghouse Emergency Response Guidelines have undergone extensive review and revision since the TMI event and are believed to be a well defined and analytically supported basis for the development of plant specific operating procedures, the current generic guidelines are not expected to be totally applicable to the WAPWR design as a result of differences from conventional designs.

WAPWR Response

An important aspect of the WAPWR design is to allow the experienced gained in the development of the generic Emergency Response Guidelines to influence the design of specific WAPWR systems.

Specific task analyses will be performed for the WAPWR design at an early enough time in the development program to allow interaction with the design process such that any design improvements identified can be factored into the WAPWR systems.

For licensing purposes Westinghouse will outline a program for emergency response guideline development prior to receiving a preliminary design approval for the WAPWR design. Prior to issuance of a final design approval (and in a timely manner that permits verification, possible NRC review, and possible operator training) Westinghouse will develop the actual WAPWR Emergency Response Guidelines.

8. Control Room Design

10CFR 50.34(f)(2)(iii)

"Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts."

Discussion

General Design Criterion 19, "Control Room," of Appendix A to 10CFR Part 50 requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions.

For current-day operating plant licensees and applicants, this item is being implemented as a detailed review of their control room designs with the purpose of correcting weaknesses to improve the ability of control room operators to prevent accidents or cope with accidents if they occur. The NRC has issued guidance for performing control room design reviews in the form of NUREG-0700, "Guidelines for Control Room Design Reviews." NUREG-0700 is written specifically for existing control room designs and new guidance or criteria may be issued in the future for new control room designs.

Again for existing control room designs, the NRC has issued draft acceptance criteria for control room design reviews which is documented in NUREG-0801, "Evaluation Criteria for Detailed Control Room Design Review." NUREG-0801 includes guidelines for the organizational structure and personnel qualifications for performing control room design reviews as well as guidelines for the actual review process and results documentation.

This requirement, as written, simply states that the control room design must be submitted to the NRC for review prior to fabrication. Inherent with this requirement is that it must be demonstrated to the NRC that the control room design meets applicable licensing criteria (e.g., human factors engineering, new instrumentation requirements, etc.).

Key to the overall issue of ensuring a good control room design is the fact that there are numerous current-day licensing issues that impact the control room design. The NRC has indicated through NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability," that these requirements (including the safety parameter display system, Regulatory Guide 1.97 instrumentation, emergency operating procedures, etc.) should be integrated with respect to the overall enhancement of the operators ability to comprehend plant conditions and cope with potential emergencies.

WAPWR Response

The WAPWR safety analysis report will include a section or chapter describing the control room design and its conformance to applicable criteria.

The overall control room design for the WAPWR will integrate the requirements of this regulation concerning human factors principles with the requirements of the various regulations concerning control room instrumentation (e.g., the instrumentation required by items 9, 10, 21, 22, and 23 below).

9. Safety Parameter Display System

10CFR 50.34(f)(2)(iv)

"Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded."

Discussion

The purpose of the plant safety parameter display console (or safety parameter display system) is to provide a concise display of critical plant variables to control room personnel in order to assist them in rapidly and reliably determining the safety status of the plant. Although not specifically mentioned in the above regulation, the NRC is recommending that the licensee consider duplication of the safety parameter display console displays in the onsite technical support center and the near-site emergency operations facility to improve the exchange of information between these facilities and the control room and assist corporate and plant management in the decision-making process.

In general, this requirement is no different from that currently being implemented by operating plant licensees and applicants in response to NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability." For certain operating plant licensees and applicants the Westinghouse designed plant safety status display system, as described in WCAP-9725 (including Supplement 1), "Westinghouse Technical Support Complex," is being installed in the onsite technical support center and the nearsite emergency operations facility as well as the control room to satisfy this requirement.

The NRC has also issued NUREG-0696, "Functional Criteria for Emergency Response Facilities," which provides certain guidance information for the implementation of a safety parameter display system. In addition, the NRC has issued draft human factors acceptance criteria for safety parameter display systems which are documented in NUREG-0835, "Human Factors Evaluation Criteria for Safety Parameter Display Systems."

WAPWR Response

The human factors principles applied to the WAPWR control room design require that a task analysis be performed, and the output from this task analysis will determine the nature of the safety status display. Control room instrumentation for the WAPWR design will be fully integrated with other control room requirements as discussed in item 8 above.

During the licensing process for the WAPWR design, Westinghouse will demonstrate the level of conformance of the WAPWR design to the NRC guidance documented in NUREG-0737 (Supplement 1), NUREG-0696 and NUREG-0835 and/or other applicable documents.

10. Safety System Status Indication

10CFR 50.34(f)(2)(v)

"Provide for automatic indication of the bypassed and operable status of safety systems."

Discussion

10CFR 50.55a(h) requires that protection systems meet the requirements set forth in IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations." Section 4.13 of IEEE Standard 279-1971 requires that, if the protective action of some part of the protection system has been bypassed or deliberately rendered inoperative for any purpose, this fact shall be continuously indicated in the control room.

The intent of this requirement is to provide the operator with an automatic indication of the bypassed or inoperable status of systems and components that perform a function important to safety in accordance with Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." It should be noted that Regulatory Guide 1.47 does permit certain limited use of manual activation of system-level indicators.

The NRC has also issued Branch Technical Position ICSB 21, "Guidance for Application of Regulatory Guide 1.47," which provides (as its title suggests) additional NRC guidance for implemtation of Regulatory Guide 1.47.

Westinghouse has developed a bypassed and inoperable status indication system as part of the overall Westinghouse designed technical support complex. This system is described in WCAP-9725 (including Supplement 1), "Westinghouse Technical Support Complex," and is currently being installed

by certain utilities in their onsite technical support centers as well as their control rooms. The Westinghouse bypassed and inoperable status indication system provides primary status display of the systems comprising the engineered safety features and supporting displays of individual components within each system or subsystem.

WAPWR Response

A bypassed and inoperable status indication system will be included in the WAPWR control room design. The specific nature of this system will be determined from the task analysis described in item 9, above. Control room instrumentation for the WAPWR design will be integrated with other control room instrumentation requirements as discussed in item 8 above.

During the licensing process for the WAPWR design, Westinghouse will demonstrate the level of conformance of the WAPWR design to the NRC regulatory positions and acceptance criteria documented in Regulatory Guide 1.47 and Branch Technical Position ICSB 21.

11. Reactor Coolant System High Point Vents

10CFR 50.34(f)(2)(vi)

"Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity."

Discussion

10CFR 50.46(b)(5) requires that after any calculated successful initial operation of the emergency core cooling system, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by long-lived radioactivity remaining in the core. Additionally, General Design Criterion 35, "Emergency Core Cooling," of Appendix A to 10CFR Part 50 requires that a system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that: (A) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (B) clad metal-water reaction is limited to negligible amounts.

During the TMI-2 accident, a condition of low water level in the reactor vessel and inadequate core cooling existed and was not rectified for a long period of time. The resultant high core temperatures produced a metal-water reaction with the subsequent production of significant amounts of hydrogen. The collection of noncondensible gases impaired natural circulation cooling capability. Additionally, the collection of noncondensible gases limited reactor coolant pump operational capability because of coolant voids in the system occupied by the gases. Even when reactor coolant pump operation was possible, the installed plant venting system was capable of removing the noncondensible gases only through an extremely slow process.

The purpose of this requirement is to provide for the capability of reactor coolant system high point venting of noncondensible gases collected in the system in order to allow satisfactory long term core cooling.

The above 10CFR 50.34 regulation must be considered in conjunction with the recent requirements of 10CFR 50.44(c)(3)(iii). This regulation, which is part of the NRC interim requirements related to hydrogen control, also mandates the installation of high point vents.

"To provide improved operational capability to maintain adequate core cooling following an accident, . . . each light-water nuclear power reactor shall be provided with high point vents for the reactor coolant system, for the reactor vessel head, and for other systems required to maintain adequate core cooling if the accumulation of noncondensible gases would cause the loss of function of these systems. (High point vents are not required, however, for the tubes in U-tube steam generators.) The high point vents must be remotely operated from the control room. Since the vents form a part of the reactor coolant pressure boundary, the design of the vents and associated controls, instruments and power sources must conform to the requirements of Appendix A and Appendix B of this part (10CFR Part 50). In particular, the vent system shall be designed to ensure a low probability that: (A) the vents will not perform their safety functions, and (B) there would be inadvertent or irreversible actuation of a vent. Furthermore, the use of these vents during and following an accident must not aggravate the challenge to the containment or the course of the accident."

Reactor coolant system high point venting for Westinghouse designs is limited to the reactor vessel and pressurizer. This requirement is no different than that currently implemented or being implemented by operating plant licensees and applicants. In general, operating plant licensees have installed add-on reactor vessel and pressurizer venting systems. Certain operating plant applicants have incorporated design modifications, prior to the TMI-2 event, related to safety-grade cold shutdown capability that include the addition of a safety-grade reactor vessel head venting system and a safety-grade upgrade to the pressurizer venting path (i.e., power-operated relief valves and block valves). Therefore, Westinghouse plants designed with safety-grade cold shutdown capability were not impacted by these regulations.

WAPWR Response

Safety-grade cold shutdown capability, including safety-grade reactor vessel and pressurizer venting paths to the pressurizer relief tank, are incorporated in the WAPWR design.

12. Plant Shielding

10CFR 50.34(f)(2)(vii)

"Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID-14844 source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment."

Discussion

10CFR Part 20 and General Design Criteria 19, 60, and 64 of Appendix A to 10CFR Part 50 require the control of radiation exposure associated with plant operations. General Design Criterion 4, "Environmental and Missile Design Bases," requires that systems and components important to safety be designed to accommodate the environmental conditions associated with accidents.

After an accident in which significant core damage occurs, the radiation source terms may approximate those of Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors." In addition, systems that were not designed to contain large radiation sources may become highly radioactive. The resulting radiation fields may make it difficult to effectively perform accident recovery operations or may impair safety equipment. Currently, Westinghouse is participating in the NRC/nuclear industry effort to more accurately define the source terms based upon the information obtained as a result of the TMI incident.

The purpose of this requirement is to facilitate post-accident operations using systems that may contain abnormally high levels of radioactivity and to ensure that safety equipment in proximity to the resulting radiation fields is not unduly degraded.

Current NRC guidance for performing radiation and shielding design reviews is detailed in Item II.B.2 of NUREG-0737, "Clarification of TMI Action Plan Requirements." Basic in this guidance is that the reviews should identify the location of vital areas and equipment (such as the control room, onsite technical support center, sampling station and sample analysis area, containment isolation reset control area, security center, 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. These design reviews are intended to identify corrective actions (e.g., design changes) necessary to provide for adequate access to vital areas and protection of safety equipment.

An important design feature of the WAPWR Primary Safeguard System (PSS) is the reduction and modularization of the post-accident recirculation equipment located outside the primary containment building. The WAPWR PSS consists of four separate modularized subsystems which are located in four physically separate and independent safeguard component areas. This system configuration and related layout/HVAC arrangement facilitates post-accident recirculation operation/maintenance and provides a significant improvement in the access to and the separator/shielding between the PSS recirculation systems and other nonradioactive vital areas.

WAPWR Response

Radiation and shielding evaluations will be performed (and documented with the NRC during the licensing process) for those vital areas and equipment included in the overall WAPWR design. The results of these evaluations

will demonstrate adequate access to vital areas and adequate protection of safety equipment or the design will be modified. As a practical matter, provisions for adequate shielding are being considered in the early phases of establishing the WAPWR plant layout.

13. Post-Accident Sampling

10CFR 50.34(f)(2)(viii)

"Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID-14844 source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole-body or 75 rem to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g, noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations."

Discussion

Prompt sampling and analysis of reactor coolant and the containment atmosphere can provide information important to the efforts to assess and control the course of an accident. Chemical and radiological analysis of reactor coolant liquid and gas samples can provide substantial information regarding core damage and coolant characteristics. Analysis of containment atmosphere (air) samples can determine if there is any prospect of hydrogen reaction in containment, as well as provide core damage information.

Beyond the above requirement, no definitive regulations exist for obtaining and analyzing reactor coolant or containment samples following an accident. NRC guidance and acceptance criteria (e.g, Standard Review Plan 9.3.2 and Regulatory Guide 1.97) have, however, been revised since the TMI-2 event to require this capability.

Most recent plant designs include (or are being modified to include) post-accident sampling capability through the use of in-line monitoring systems. These systems usually have the capability to obtain samples from the reactor coolant system hot legs, the containment recirculation sump, and the containment atmosphere. The time required for taking and analyzing samples (in an onsite radiological and chemical analysis facility) must be 3-hours or less from the time a decision is made to sample, except for chloride which is 24-hours or less.

In addition to in-line monitoring systems, backup sampling is require to be available through grab samples. Capability of analyzing these samples must be demonstrated (established planning for analysis at offsite facilities is acceptable to the NRC).

The above regulation also requires that radiation exposures to those individuals performing sampling and analyses be limited to acceptable values. Facility, system, and shielding design must be such that personnel exposure is minimized.

WAPWR Response

The WAPWR plant design will include in-line sampling capability as well as grab sample capability. An onsite radiological and chemical analysis capability will be considered in the WAPWR design consistent with the scope definition for the Nuclear Power Block.

During the WAPWR licensing process, Westinghouse will:

- o Demonstrate compliance with all applicable requirements of NUREG-0737 (Item II.B.3) for sampling, chemical, and radionuclide analysis capability under accident conditions.

- o Demonstrate that sufficient shielding is provided to meet the requirements of General Design Criterion 19, assuming Regulatory Guide 1.4 (TID-14844) source terms.

- Demonstrate compliance with the sampling and analysis requirements of Regulatory Guide 1.97, Revision 3, in accordance with the overall WAPWR post-accident monitoring design that addresses this regulatory guide as discussed in item 23 below.

- Demonstrate that all electrically powered components associated with post-accident sampling are capable of being supplied with power and operated within 30-minutes of an accident in which there is core degradation, assuming loss of offsite power.

- Demonstrate that any valves associated with post-accident sampling which are not accessible for repair after an accident are environmentally qualified for the conditions in which they must operate.

- Provide a procedure for relating radionuclide gaseous and ionic species to estimated core damaged.

- Demonstrate the design or operational provisions to prevent high pressure carrier gas from entering the reactor coolant system from in-line gas analysis equipment.

- Demonstrate a method for verifying that reactor coolant dissolved oxygen is at < 0.1 ppm if reactor coolant chlorides are determined to be > 0.15 ppm.

- Provide information on: (A) testing frequency and type of testing to ensure long-term operability of the post-accident sampling system, and (B) recommended operator training requirements for post-accident sampling.

14. Hydrogen Control

<u>10CFR 50.34(f)(2)(ix)</u>

"Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (1)(xii) of this section (50.34) is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that:

(A) Uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel-clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

(B) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(C) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system.

(D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation."

Discussion

The accident at TMI-2 resulted in a severely damaged or degraded reactor core with a concomitant release of radioactive material to the primary coolant system and a large amount of fuel cladding metal-water reaction in the core with hydrogen generation well in excess of the amounts required to be considered for design purposes by historical Commission regulations. The accident revealed design and operational limitations that existed relative to mitigating the consequences of the accident and determining the status of the facility during and following the accident.

10CFR 50.44(c)(1) requires that it be shown that during the time period following a LOCA but prior to effective operation of the combustible gas control system either: (A) an uncontrolled hydrogen-oxygen recombination would not take place in the containment, or (B) the plant could withstand the consequences of uncontrolled hydrogen-oxygen recombination without loss of safety function. If these conditions cannot be shown, the containment is required to be provided with an inerted or an oxygen deficient atmosphere in order to provide protection against hydrogen burning and explosions.

For operating plant licensees and applicants prior to the TMI-2 event, the NRC is proposing regulations similar to those contained in 10CFR 50.34(f)(1)(xii) and 10CFR 50.34(f)(2)(ix). The major differences between the proposed rules for existing plants and the effective rules for new plants are that:

- o The uniform hydrogen concentration in the containment must not exceed 10 percent by volume during and following a degraded core accident for new plants. The proposed rules for existing plants do not impose such a limit on hydrogen concentration.

- o The amount of hydrogen to be considered for new plants is equivalent to that generated from the reaction of 100 percent (versus 75 percent for existing plants) of the fuel cladding surrounding the actual fuel region.

For new plant designs a suitable hydrogen control system will be required to meet this regulation, whereas no hydrogen control system is needed for large dry containments to meet the proposed regulations of the interim rule for existing plants.

Among the various hydrogen control systems evaluated by the industry thus far, a hydrogen ignition system appears to be the best choice. A hydrogen ignition system is relatively inexpensive, easy to test, and inadvertent actuation of the system during normal plant operation will not result in any adverse effects.

It should be noted that the WAPWR design will differ from a conventional pressurized water reactor in that there will be a significant increase in the amount of Zircaloy utilized in the design. This increase in Zircaloy is not simply related to an increase in the amount of fuel cladding present (i.e., due to the larger core) but results from a combination of other design features.

WAPWR Response

Westinghouse will perform an evaluation of alternative hydrogen control systems for the WAPWR design. The system selected for use in the WAPWR design will be in accordance with the above requirements and fully documented during the licensing process for the WAPWR design.

Westinghouse will perform all calculations and analyses considering the additional Zircaloy in the WAPWR design rather than restricting the calculations to 100 percent of the fuel cladding as required by this regulation.

15. Reactor Coolant System Valve Testing

10CFR 50.34(f)(2)(x)

"Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWR's, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transients without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed."

Discussion

General Design Criteria 14, 15, and 30 of Appendix A to 10CFR Part 50 require that the reactor coolant pressure boundary be designed, fabricated, and erected to the highest quality standards and be tested to ensure an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. These criteria also require that the design conditions of the reactor coolant pressure boundary not be exceeded during any condition of normal operation, including anticipated operational occurrences.

Proper operation of the reactor coolant system relief, safety, and block valves is necessary for conformance to these design criteria. The inability of these valves to open or close could lead to a violation of the integrity of the reactor coolant pressure boundary.

When the reactor coolant system relief and safety valves open, the flow through these valves is normally saturated steam. Some reactor coolant system transients and accidents as well as alternate core-cooling methods can result in solid-water or two-phase steam-water flow through these valves. Historical qualification requirements for these valves included only flow under saturated steam conditions.

The purpose of this regulation is to require qualification of reactor coolant system relief, safety, and block valves under expected operating conditions (including solid-water and two-phase flow conditions) and ATWS conditions.

Generic reactor coolant system valve testing (sponsored by EPRI) has been conducted in support of operating plant licensees and applicants. The EPRI program included representative testing of Westinghouse reactor coolant system valve types at representative fluid conditions including solid-water and two-phase flow conditions. The EPRI program did not, however, include specific consideration of ATWS conditions.

Operating plant licensees and applicants have submitted documentation to demonstrate applicability of the generic EPRI test results to their plant specific reactor coolant system valves, their plant specific expected fluid conditions, and their plant specific piping and support configurations.

WAPWR Response

The generic EPRI test results discussed above are expected to be directly applicable to the WAPWR design, since the latest Westinghouse pressurizer power-operated relief valves and safety valves were included in the test program.

Westinghouse will document the applicability of the generic EPRI test results to the WAPWR design (including valve designs, piping and support designs, and fluid conditions) during the licensing process for the WAPWR design. If the generic EPRI test results do not envelope the specific WAPWR design, Westinghouse will either: (A) perform additional testing, or (B) demonstrate justification for not performing additional testing possibly through additional analyses and/or evaluations.

16. Valve Position Indication

10CFR 50.34(f)(2)(xi)

"Provide direct indication of relief and safety valve position (open or closed) in the control room."

Discussion

This regulation is written in very general terms. A review of the NRC background material in relation to this regulation (i.e., NUREG-0578, NUREG-0660, NUREG-0718, and NUREG-0737) indicates that this requirement for valve position indication applies to reactor coolant system relief and safety valves.

General Design Criterion 14, "Reactor Coolant Pressure Boundary," of Appendix A to 10CFR Part 50 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. Historically, the application of this criterion has emphasized the integrity of passive components in the reactor coolant system, such as the reactor vessel and the piping, however, this criterion also applies to the valves that provide isolation for the system. Failure of relief and safety valves to close can cause events that result in small-break loss-of-coolant accidents. Unambiguous indication of the position of the valves can aid the operator to detect a failure and take proper corrective action.

The purpose of this requirement is to provide the control room operator a positive indication of valve position and, therefore, provide additional assurance that the integrity of the reactor coolant pressure boundary can be maintained or a loss of integrity directly diagnosed.

Conventional Westinghouse designs include a positive control room position indication for the presssurizer power-operated relief valves (i.e., indication lights which are activated by limit switches). Conventional safety valve designs have been upgraded to provide position indication through stem-mounted limit switches or acoustic monitoring of flow downstream of the valves.

The above regulation, as written, requires a direct valve position indication and, therefore, the option of flow indication through utilization of an acoustic monitoring system does not satisfy this requirement.

WAPWR Response

The WAPWR design will incorporate positive control room position indication for the pressurizer power-operated relief valves and safety valves.

17. Auxiliary Feedwater System Initiation and Indication

10CFR 50.34(f)(2)(xii)

"Provide automatic and manual auxiliary feedwater system initiation, and provide auxiliary feedwater system flow indication in the control room."

Discussion

In Westinghouse designs the auxiliary feedwater system (AFWS) has been treated as a safety system. It is used to remove heat from the reactor system when the main feedwater system is not available.

The need to automatically initiate the operation of the AFWS was not considered by all vendors to be essential to safety in the past, and in some plants dependence was placed on the operator to put the system in

service when required. Although this need was not emphasized, the initiation of the AFWS is automatic in conventional Westinghouse designed plants in accordance with General Design Criterion 20, "Protection System Functions," of Appendix A to 10CFR Part 50.

General Design Criterion 13, "Instrumentation and Control," sets forth the requirements for instrumentation to monitor the variables and systems, over their anticipated ranges of operation, that can affect reactor safety. Auxiliary feedwater flow indication to the steam generators is considered an important adjunct to the manual regulation of auxiliary feedwater flow to maintain the required steam generator level and Westinghouse has recommended that this indication be included in plant designs.

WAPWR Response

The secondary side safeguards function for the WAPWR design will differ from conventional designs as discussed in detail in item 2 above.

The secondary side safeguards function for the WAPWR design includes manual and automatic initiation of flow to the steam generators. Heat removal indication (i.e., flow and level) in the control room will be provided by Class 1E indicators in accordance with the above regulation and the guidance of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" (refer to item 23 below).

18. Pressurizer Heater Power Supplies

10CFR 50.34(f)(2)(xiii)

"Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available."

Discussion

Pursuant to NRC regulations in 10CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," the loss of offsite power is considered to be an anticipated operational occurrence, since it is expected to occur one or more times during the life of a nuclear plant. Following a loss of offsite power, stored and decay heat from the reactor would normally be removed by natural circulation using the steam generators as the heat sink. Natural circulation cooling of the primary system requires the use of the pressurizer to maintain a suitable overpressure on the reactor coolant system. Consistent with satisfying the basic requirements in General Design Criteria 10, 14, 15, 17, and 20 a selected number of pressurizer heaters should be supplied from the emergency power buses.

Evaluations of this item for operating plant licensees and applicants indicate that Westinghouse interface criteria in this area (i.e., minimum number of pressurizer heaters necessary to support natural circulation and the time available for connection of the emergency power source following a loss of offsite power) are conservative.

WAPWR Response

For the WAPWR design one group of pressurizer backup heaters (manually loaded within 1 hour) is sufficient to maintain natural circulation following a loss of offsite power. To ensure availability of at least one group of backup heaters upon loss of offsite power, emergency power will be provided from separate diesel generators to two groups of backup heaters.

19. Containment Isolation System

10CFR 50.34(f)(2)(xiv)

"Provide containment isolation systems that: (A) ensure all nonessential systems are isolated automatically by the containment isolation system, (B) for each non-essential penetration (except instrument lines) have two isolation barriers in series, (C) do not result in reopening of the containment isolation valves on resetting of the isolation signal, (D) utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation, and (E) include automatic closing on a high radiation signal for all systems that provide a path to the environs."

Discussion

General Design Criterion 54, "Piping Systems Penetrating Containment," of Appendix A to 10CFR Part 50 requires that piping systems penetrating primary reactor containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capability which reflect the importance to safety of isolating the piping systems. Standard Review Plan 6.2.4, "Containment Isolation System," requires that there be diversity in the parameters sensed for the initiation of containment isolation.

Some early plants (including TMI-2) provided automatic containment isolation demand on high containment pressure only. For small rates of loss of coolant, there would be little pressure increase in the containment, and automatic isolation could be delayed or possibly not occur. The loss of coolant at TMI-2, which produced a small pressure rise in the containment, was accompanied by substantial core damage and a large release of radionuclides into the containment building. Containment isolation was not achieved for some hours after the start of the event.

The purpose of this requirement is to ensure that effective containment isolation is accomplished and maintained.

WAPWR Response

In regard to requirement (A) above, careful consideration will be given to the definition and identification of essential and nonessential systems. Westinghouse will document the basis for selection of essential systems during the licensing process for the WAPWR design.

In regard to requirement (B) above, for post-accident situations each non-essential penetration (except instrument lines) will have two isolation barriers in series in accordance with the requirements of General Design Criteria 54, 55, 56, and 57, as clarified by Standard Review Plan 6.2.4. Isolation will be performed automatically (i.e, no credit will be given for operator action). Manual valves will be sealed closed, as defined by Standard Review Plan 6.2.4, to qualify as an isolation barrier. Each automatic isolation valve in a non-essential penetration will receive diverse isolation signals.

In regard to requirement (C) above, the design of control systems for automatic containment isolation valves will be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves will require deliberate operator action. Administrative provisions to close all isolation valves manually before resetting the isolation signals will not be considered an acceptable method of meeting this requirement.

Ganged reopening of containment isolation valves is not acceptable. Reopening of isolation valves will be performed on a valve-by-valve basis, or on a line-by-line basis, provided that electrical independence and other single-failure criteria continue to be satisfied.

In regard to requirement (D) above, the containment set point pressure that initiates containment isolation for non-essential penetrations will be the minimum compatible with normal operating conditions. The pressure set point selected will be far enough above the maximum expected pressure

inside containment during normal operation so that inadvertent containment isolation does not occur during normal operation from instrument drift or fluctuations due to the accuracy of the pressure sensor.

In regard to requirement (E) above, all systems that provide a path from the containment to the environs (e.g., containment purge and vent systems) will close on a safety-grade high radiation signal.

20. Containment Purging/Venting

10CFR 50.34(f)(2)(xv)

"Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions."

Discussion

While the containment purge and vent systems provide plant operational flexibility, their designs must consider the importance of minimizing the release of containment atmosphere to the environs following a postulated loss-of-coolant accident. Therefore, the NRC position is that plant designs must not rely on their use on a routine basis.

The need for purging has not always been anticipated in the design of plants, and therefore, design criteria for the containment purge system have not been fully developed. The purging experience at operating plants varies considerably from plant to plant. Some plants do not purge during reactor operation, some purge intermittently for short periods, and some purge continuously. There is similar disparity in the need for, and use of, containment vent systems at operating plants.

Containment purge systems have been used in a variety of ways; for example, to alleviate certain operational problems, such as excess air leakage into the containment from pneumatic controllers, for reducing the airborne activity within the containment to facilitate personnel access during reactor power operation, and for controlling the containment pressure, temperature, and relative humidity. Containment vent systems are typically used to relieve the initial containment pressure buildup caused by the heat load imposed on the containment atmosphere during reactor power ascension, or to periodically relieve the pressure buildup due to the operation of pneumatic controllers.

The sizing of the purge lines in most plants have been based on the need to control the containment atmosphere during refueling operations. This need has resulted in very large lines penetrating the containment (some on the order of 42 inches in diameter). Since these lines are normally the only ones provided that will permit some degree of control over the containment atmosphere to facilitate personnel access, some plants have used them for containment purging during normal plant operation. The NRC is concerned with this situation during a postulated loss-of-coolant accident, since the lines provide an open path from the containment to the environs and calculated accident doses could be significant.

Therefore, the NRC is currently requiring compliance with Branch Technical Position CSB 6-4, "Containament Purging During Normal Plant Operations." The following are included as requirements in Branch Technical Position CSB 6-4:

- o The use of large containment purge lines is restricted to cold shutdown and refueling operations (the lines must be sealed closed in all other operational modes).

- o Additional smaller purge lines (about 8 inches in diameter or smaller) can be provided for continuous purging (lines larger than 8 inches in diameter must be justified to the NRC).

WAPWR Response

The containment purging/venting capability for the WAPWR design will be such that:

- o Reliable containment isolation will be achieved under accident conditions in accordance with item 19 above.

- o Purge time will be minimized consistent with ALARA principles for occupational exposure.

During the licensing process for the WAPWR design, Westinghouse will demonstrate the level of conformance of the WAPWR design to the NRC acceptance criteria documented in Branch Technical Position CSB 6-4.

21. Specific Accident Monitoring Instrumentation

10CFR 50.34(f)(2)(xvii)

"Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples."

Discussion

General Design Criterion 13, "Instrumentation and Control," of Appendix A to 10CFR Part 50 requires instrumentation to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can effect the containment and its associated systems.

In the past, General Design Criterion 13 had been implemented based on design basis accidents analyzed in Chapter 15.0 of safety analysis reports. Based on conditions experienced at TMI-2, situations can arise which produce containment conditions beyond those postulated for the Chapter 15.0 events.

The purpose of this requirement is to ensure that capability is provided in the control room to ascertain containment conditions during the course of an accident.

WAPWR Response

The above required instrumentation is included in Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," and, as such, it will be included in the overall WAPWR post-accident monitoring design that addresses this regulatory guide. Refer to the discussion of item 23 below.

Control room instrumentation for the WAPWR design as a result of this regulation will be integrated with other control room instrumentation requirements as discussed in item 8 above.

22. Inadequate Core Cooling Instrumentation

10CFR 50.34(f)(2)(xviii)

"Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWR's, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWR's and BWR's."

Discussion

General Design Criterion 13, "Instrumentation and Control," of Appendix A to 10CFR Part 50 requires instrumentation to monitor variables for accident conditions as appropriate to assure adequate safety. In the past, General Design Criterion 13 was not interpreted to require instrumentation to directly monitor water level in the reactor vessel or the adequacy of core cooling. The conventional instrumentation available that could indicate inadequate core cooling includes core exit thermocouples, cold leg and hot leg resistance temperature detectors, in-core neutron detectors, and ex-core neutron detectors. Generally, such instrumentation is included in the reactor design to perform functions other than monitoring of core cooling or indication of vessel water level.

During the TMI-2 accident, a condition of low water level in the reactor vessel and inadequate core cooling existed and was not recognized for a long period of time. This problem was the result of a combination of factors including an insufficient range of existing instrumentation, inadequate emergency procedures, inadequate operator training, unfavorable instrument location (scattered information), and perhaps insufficient instrumentation.

The purpose of this requirement is to provide the reactor operator with instrumentation that, together with improved operating procedures and training, will enable him to readily recognize and implement actions to correct or avoid conditions of inadequate core cooling.

WAPWR Response

The above required instrumentation (reactor vessel level instrumentation system and thermocouple/core cooling monitoring system) is included in Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," and, as such, it will be included in the overall WAPWR post-accident monitoring design that addresses this regulatory guide. Refer to the discussion of item 23 below.

Control room instrumentation for the WAPWR design as a result of this regulation will be integrated with other control room instrumentation requirements as discussed in item 8 above.

23. Post-Accident Monitoring Instrumentation

10CFR 50.34(f)(2)(xix)

"Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage."

Discussion

General Design Criterion 13, "Instrumentation and Control," of Appendix A to 10CFR Part 50 requires instrumentation to monitor variables and systems over their anticipated ranges for accident conditions as appropriate to ensure adequate safety. General Design Criterion 19, "Control Room," requires that a control room be provided from which actions can be taken to maintain the nuclear power unit in a safe condition under accident conditions. In addition, General Design Criterion 64, "Monitoring Radioactivity Releases," requires means for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from postulated accidents.

The overall subject of adequate post-accident monitoring has been a concern of the NRC and the industry for many years. As a result of this initial concern which was amplified in light of the TMI-2 accident, the NRC has issued guidance in the form of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," which describes a method acceptable to the NRC staff for complying with the Commission's requirements to provide instrumentation to monitor plant variables and systems during and following an accident.

Westinghouse has developed an interpretation of the requirements necessary to meet the intent of Regulatory Guide 1.97 for a conventional operating plant applicant. The specific variables specified in this design basis interpretation of Regulatory Guide 1.97 are not entirely applicable to the WAPWR design as a result of differences from conventional designs.

WAPWR Response

Using the above mentioned Westinghouse design basis interpretation of Regulatory Guide 1.97 as a starting point, a similar document for the WAPWR will be developed and the results implemented in the design.

Westinghouse will completely document and justify any deviations from the NRC Regulatory Guide 1.97, Revision 3, positions during the licensing process for the WAPWR design.

Control room instrumentation for the WAPWR design as a result of this regulation will be integrated with other control room instrumentation requirements as discussed in item 8 above.

24. Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators

10CFR 50.34(f)(2)(xx)

"Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) level indicators are powered from vital buses, (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety, and (C) electric power is provided from emergency power sources."

Discussion

Pursuant to NRC regulations in 10CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," the loss of offsite power is considered to be an anticipated operational occurrence, since it is expected to occur one or more times during the life of a nuclear plant. Following a loss of offsite power, stored and decay heat from the reactor would normally be removed by natural circulation using the steam generators as the heat sink. Alternatively, in the event that natural circulation in the reactor coolant system is interrupted, the feed and bleed mode of reactor coolant system operation can be used to remove decay heat from the reactor. This method of decay heat removal requires the use of the emergency core cooling system and the pressurizer power-operated relief valves. Consistent with satisfying the basic requirements in General Design Criteria 10, 14, 15, 17 and 20, the pressurizer power-operated relief valves and associated block valves and level indicators must be supplied from emergency power buses.

More specific NRC guidance for implementation of this regulation is contained in Item II.G.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements."

WAPWR Response

The WAPWR design will include provisions for appropriate emergency power for pressurizer equipment. In the determination of the power supplies for the pressurizer power-operated relief valves and associated block valves, consideration will be given to cold shutdown and reactor coolant system overpressurization requirements in addition to the post-TMI requirements of this regulation.

25. Emergency Response Facilities

10CFR 50.34(f)(2)(xxv)

"Provide an onsite technical support center, an onsite operational support center, and, for construction permit applications only, a nearsite emergency operations facility."

Discussion

In addition to the above regulation, Article IV.E.8 of Appendix E, "Emergency Planning and Preparedness for Protection and Utilization Facilities," to 10CFR Part 50 requires that adequate provisions shall be made and described for emergency facilities and equipment, including a licensee onsite technical support center and a licensee near-site emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency. (Note that "effective control" must be interpreted to mean administrative control versus actual control of the plant).

As one would expect, these regulations are quite general in that they simply require emergency response facilities to be established. The NRC has, however, issued detailed guidance (e.g, functions, locations, size, structures, habitability, communications, instrumentation, etc.) for the design of emergency response facilities in the form of NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability," and NUREG-0696, "Functional Criteria for Emergency Response Facilities."

WAPWR Response

The onsite technical support center will be included in the WAPWR design. The on-site operational supprt center and the near-site emergency operations facility is the responsibility of each utility utilizing the WAPWR design.

During the licensing process for the WAPWR design, Westinghouse will demonstrate the level of conformance of the WAPWR design to the NRC guidance documented in NUREG-0737 (Supplement 1) and NUREG-0696 and/or other applicable documents.

26. Leakage Control Outside Containment

10CFR 50.34(f)(2)(xxvi)

"Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID-14844 source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency."

Discussion

10CFR Part 20 and Part 100 specify radiation limits and guidelines that must be met by licensed facilities to assure protection of public health and safety. In a power reactor, many systems that will or may handle liquids or gases containing large radioactive inventories after a serious transient or accident are located outside containment. Several of the engineered safety features and auxiliary systems located outside reactor containment will or may have to function during a serious transient or accident with large radioactive inventories in the fluids they process. The leakage from these systems, when operated, should be minimized or eliminated to prevent the release of significant amounts of radioactive materials to the environment. Historically, these systems are checked out during preoperational testing and startup testing but are not usually included in any periodic leak testing program. It is beneficial if the plant operating staff knows the leakage rates of these systems and maintains them at rates that are as low as practical.

The purpose of this regulation is to make every effort to eliminate or reduce the leakage from these systems, perform periodic tests to assure that the leakage from these systems is maintained as low as practical, and provide the plant staff with current knowledge of the system leakage rates.

WAPWR Response

The WAPWR design includes the following features which minimize the potential for leaks and/or improve leakage control and detection.

- o The amount of equipment located outside containment has been minimized.
- o The Primary Side Safeguards Equipment is segregated into four separate independent safeguard component areas.
- o Capability to test the full recirculation path is provided.

The actual leakage testing procedures will be established by each utility using the WAPWR design.

27. Inplant Monitoring

10CFR 50.34(f)(2)(xxvii)

"Provide for monitoring of inplant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions."

Discussion

10CFR Part 20, "Standards for Protection Against Radiation," provides criteria for control of exposures of individuals to radiation in restricted areas, including airborne iodir   Since iodine concentrates in the thyroid gland, airborne concentrati s must be known in order to

evaluate the potential dose to the thyroid. Historically, the concentration of iodine in atmosphere air has been determined by measuring the activity of iodine adsorbed in a carbon filter through which air has been pumped. The charcoal filter is removed from the air pump and allowed to ventilate to permit the noble gases to diffuse to the atmosphere. The filter is then counted for radioactivity content and the remaining activity is ascribed to iodine. This procedure is conservative; however, it is possible for sufficient noble gas to be adsorbed in the charcoal so that the resulting iodine determination may be unduly conservative (high). If the airborne iodine concentration is overestimated, plant personnel may be required to perform operational functions while using respiratory equipment, which sharply limits communication capability and may diminish personnel performance during an accident.

The purpose of this requirement is to improve the accuracy of measurement of airborne iodine concentrations as well as to ensure adequate inplant monitoring of vital areas.

WAPWR Response

The WAPWR design will include sufficient iodine samplers to sample all vital areas.

28. Control Room Habitability

10CFR 50.34(f)(2)(xxviii)

"Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID-14844 source term release, and make necessary design provisions to preclude such problems."

Discussion

Control room habitability deals with assuring that control room operators will be adequately protected against the effects of an accidental release of toxic and radioactive gases and that the plant can be safely operated or shutdown under design basis accident conditions (in accordance with General Design Criterion 19, "Control Room," of Appendix A to 10CFR Part 50).

For plants designed over the last 5 to 8 years, this TMI item (in general) has not presented a significant problem (beyond software documentation), since the current NRC guidance and acceptance criteria for ensuring control room habitability was available during the design and licensing of these plants.

WAPWR Response

The WAPWR design will include appropriate provisions for control room habitability in accordance with the NRC guidance provided in Standard Review Plan 6.4, "Control Room Habitability System." Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 6.4 acceptance criteria during the licensing process for the WAPWR design.

29. Industry Experiences

10CFR 50.34(f)(3)(i)

"Provide administrative procedures for evaluating operating, design and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant."

Discussion

This requirement deals with administrative procedures which by themselves do not impact any design.

Westinghouse has always recognized the need to stay appraised of operating events to meet the need for feedback of operating experiences to design, construction, and operation. Currently, this is being accomplished through informal methods of screening various media sources (e.g., INPO, Westinghouse site managers daily reports, NRC Inspection and Enforcement Bulletins, Circulars, and Information Notices) for event information. Those events or issues identified as having potential significance are routed internally to appropriate cognizant personnel for their evaluation and follow-up action as necessary.

WAPWR Response

Westinghouse is considering more formal programs for systematically following and incorporating operating and construction experiences in the WAPWR design and reliability evaluations. This subject will be fully addressed during the licensing process for the WAPWR design.

30. Quality Assurance List

10CFR 50.34(f)(3)(ii)

"Ensure that the quality assurance (QA) list required by Criterion II, Appendix B, 10CFR Part 50 includes all structures, systems and components important to safety."

Discussion

Appendix B, "Quality Assurance Criterion for Nuclear Power Plants and Fuel Reprocessing Plants," to 10CFR Part 50 establishes quality assurance

requirements for all activities affecting the design, construction, and operation of those safety-related structures, systems, and components that prevent or mitigate the consequences of postulated accidents or could cause undue risk to the health and safety of the public. Criterion II of Appendix B futher requires the identification of the structures, systems, and components to be covered by the quality assurance program.

Historically, the requirements of Appendix B have been mostly applied only to safety-related structures, systems, and components (e.g., for a conventional design this encompasses Safety Class 1, 2, and 3 structures, systems, and components). This approach has been (in general) accepted by the NRC in the past even though Appendix A, "General Design Criteria for Nuclear Power Plants," of 10CFR Part 50 requires the establishment of principle design criteria for structures, systems, and components <u>important to safety</u>; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. Structures, systems, and components important to safety can and do include equipment that has been historically classified as Non-Nuclear Safety. A Non-Nuclear Safety classification has been translated into a quality assurance program less stringent than that required by Appendix B of 10CFR Part 50.

The purpose of this regulation is to ensure that appropriate quality assurance is applied to all structures, systems, and components important to safety versus only those that are safety-related.

<u>WAPWR Response</u>

The WAPWR design, quality assurance program, and associated listings of structures, systems, and components will comply with this requirement. Westinghouse will apply applicable 10CFR Part 50, Appendix B, quality assurance criteria to structures, systems, and components important to safety.

31. Quality Assurance Program

10CFR 50.34(f)(3)(iii)

"Establish a quality assurance (QA) program based on consideration of: (A) ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (C) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (D) establishing criteria for determining QA programmatic requirements; (E) establishing qualification requirements for QA and QC personnel; (F) sizing the QA staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of "as-built" documentation; and (H) providing a QA role in design and analysis activities."

Discussion

Various TMI-2 accident investigations and inquiries identified problems relating to the quality assurance organization, authority, reporting, and inspection. This regulation is a result of NRC actions taken to improve the quality assurance program for design, construction, and operations to provide greater assurance that these activities are conducted in a manner commensurate with their importance to safety.

Westinghouse has established and is implementing a quality assurance program, approved by the NRC, that complies with 10CFR Part 50, Appendix B, "Qaulity Assurance Criterion for Nuclear Power Plants and Fuel Reprocessing Plants," and the considerations listed in the above regulation. This program currently addresses Westinghouse design and construction activities and may be revised in the future to include onsite construction activities and operations.

WAPWR Response

Westinghouse will document the quality assurance program applicable to the WAPWR program during the licensing process for the WAPWR design.

32. Dedicated Containment Penetrations

10CFR 50.34(f)(3)(iv)

"Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system."

Discussion

As discussed in more detail in Section 3.2, there are rulemaking efforts currently underway to establish policy, goals, and requirements related to accidents involving core damage greater than the present design basis. One of the design requirements being considered in these efforts is the need for a new structure for controlled filtered venting of the reactor containment structure.

The purpose of this requirement is to ensure the capability of installing such a system should it be determined necessary.

Westinghouse studies (as well as other industry studies) of conventional plant designs have indicated that filtered vented containment systems may not be cost-effective for large dry containments similar to the WAPWR containment design.

WAPWR Response

As part of the WAPWR design risk assessment (discussed in more detail in Section 3.2), Westinghouse will evaluate the potential benefits of

filtered vented containment systems. Based on these evaluations and associated cost-benefit considerations, Westinghouse will either:

- o Include one or more dedicated penetrations in the WAPWR design for potential future installation of a filtered vented containment system in accordance with this regulation, or

- o Request exemption from this regulation for the WAPWR design.

33. Containment Design

10CFR 50.34(f)(3)(v)

"Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that:

(A)(1) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 2 Subsubarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100 percent fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above, appropriate for each type of containment, will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

(2) Subarticle NE-3220, Division 1, and subarticle CC-3720, Division 2, of Section III of the July 1, 1980 ASME Boiler and Pressure Vessel Code, which are referenced in paragraph (f)(3)(v)(A)(1) and (f)(3) (v)(B)(1) of this section, were approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. . . .

(B)(1) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC-3720, Service Load Category), (2) The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting."

Discussion

The accident at TMI-2 resulted in a severely damaged or degraded reactor core with a concomitant release of radioactive material to the primary coolant system and a large amount of fuel cladding metal-water reaction in the core with hydrogen generation well in excess of the amounts required to be considered for design purposes by historical Commission regulations. The accident revealed design and operational limitations that existed relative to mitigating the consequences of the accident and determining the status of the facility during and following the accident.

This regulation, in conjunction with the hydrogen control regulations of items 5 and 14, is intended to assure that containment structural integrity is maintained during severe accident conditions.

WAPWR Response

The WAPWR containment design will be in accordance with this regulation.

34. Hydrogen Recombiners

10CFR 50.34(f)(3)(vi)

"For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere."

Discussion

In accordance with 10CFR 50.44, "Standards for Combustible Gas Control System in Light Water Cooled Power Reactors," plant designs since about late 1970 must include a combustible gas control system (such as recombiners) as the primary means for controlling combustible gases following a loss-of-coolant accident.

Certain plant designs satisfied this requirement with provisions for post-accident installation and operation of an external hydrogen recombiner for combustible gas control. For example, TMI-2 had this external recombiner capability. The design of the external recombiner hookup at TMI-2 used the 36-inch containment penetrations for the normal containment purge system by tapping 4-inch lines off the purge lines outside the containment building between the building and the outer containment isolation valves. To place the hydrogen recombiner into service required the opening of the inboard 36-inch containment isolation valve in both a

containment purge system inlet and outlet line. With this design, once the hydrogen recombiner is put into operation, containment integrity is vulnerable to a single active failure. That is, a spurious or inadvertent opening of one of the 36-inch outboard containment isolation valves would have resulted in the venting of the containment to the environment. In addition, the design of the system to include use of the large (36-inch) containment purge penetrations resulted in the operation of the recombiner beyond the design capacity of the unit.

This requirement does not apply to Westinghouse designed plants that incorporate internal hydrogen recombiners.

Since the TMI-2 event, the NRC has revised 10CFR 50.44 to also require dedicated containment penetrations for external recombiners.

WAPWR Response

The WAPWR design will include a manually actuated recombiner system which is redundant, qualified, and installed inside containment. However, the total hydrogen control system will be in accordance with item 14 above.

35. Management Plan

10CFR 50.34(f)(3)(vii)

"Provide a description of the management plan for design and construction activities, to include: (A) the organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to operation; (E) the degree of top level management oversight and technical control to

be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort."

Discussion

One of the major findings as a result of the TMI-2 accident was the need to improve staffing to oversee design and construction activities. This regulation is intended to address this finding.

The NRC has issued draft NUREG-0731, "Guidelines for Utility Management Structure and Technical Resources," which is expected to be used by utilities as guidance in meeting this regulation.

WAPWR Response

This regulation is directed at utility management organizational and administrative capabilities and is not applicable to Westinghouse in relation to the WAPWR design.

3.2 SEVERE ACCIDENT RULEMAKING AND RELATED CONSIDERATIONS

Discussion

The TMI-2 accident and the results of subsequent reviews and investigations prompted the Commission to reconsider certain aspects of its licensing policy. One of the conclusions from the post-TMI investigations was that attention should be given to the probability and consequences of severe accidents (beyond the normal design basis accidents) and that a policy statement on the acceptance level of risk to the public health and safety was needed.

Since the completion of WASH-1400, "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," development in probabilistic risk assessment techniques has led to the recognition that it is feasible to use quantitative guidelines in evaluating reactor safety. These techniques are a viable means of assessing plant risks and comparing them with the proposed safety goal discussed below.

The Commission is requiring (through the regulations of 10CFR 50.34(f)(1)(i) and their proposed rule on severe accidents) performance of a prob- abilistic risk assessment and associated reliability engineering programs for standard plant designs to be referenced by new construction permit applica- tions.

The Commission initiated efforts to develop a safety goal policy in its 1981 Federal Register Notice entitled "Development of a Safety Goal - Preliminary Policy Consideration" and subsequently published a "Proposed Policy Statement on Safety Goals for Nuclear Power Plants" (47FR7023 dated February 17, 1982). This proposed statement served as an initial step toward the ultimate goal of explicitly defining the acceptable level of risk to the public health and safety from nuclear power reactors and included several qualitative safety goals and quantitative probabilistic guidelines for severe accidents.

After several drafts and revisions as a result of comments and recommendations received by the NRC, on March 14, 1983, the Commission published a "Policy Statement on Safety Goals for the Operation of Nuclear Power Plants" (48FR10772) for a two year trial use and evaluation period. The policy statement includes the following preliminary safety goals and preliminary numerical design objectives.

- Safety Goals

  "Individual members of the public should be provided a level of protection from the consequencies of nuclear power plant operation such that individuals bear no significant additional risk to life and health."

  "Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks."

- Quantitative Design Objectives

  1. Individual and Societal Mortality Risks

     "The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed."

     The risk to the population in the use of a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks resulting from all other causes."

2. Benefit-Cost Guideline

"The benefit of an incremental reduction of societal mortality risks should be compared with the associated costs on the basis of \$1000 per person-rem averted."

3. Plant Performance Design Objective

"The likelihood of a nuclear reactor accident that results in a large-scale core melt should normally be less than one in 10,000 per year of the reactor operation."

In regard to the individual and societal mortality risk guidelines the Policy Statement defines "the vicinity" to be considered in the individual risk of prompt fatality as the area within one mile of the site boundary, and in the design objective for cancer fatality the population within 50 miles of the plant site is to be considered. The Policy Statement also notes that: the application of the benefit cost guideline should be focused principally on situations where one of the quantified safety goals is not met; and the design objective for large scale core melt is subordinate to the principal design objectives limiting individual and societal risks.

The Commission's intention is that the design objectives and benefit-cost guideline would be used by the NRC staff in conjunction with probabilistic risk assessments and would not substitute for reactor regulations in 10CFR Chapter I. Rather, individual licensing decisions would continue at present to be based principally on compliance with the Commission's regulations.

During the next two years the safety goal policy will be evaluated as to their adequacy and usefulness in the regulatory process. In this process there is to be trial application to a number of generic issues to gain hands on experience, however the safety goal is not to be a factor in their resolution. The NRC has indicated that the implementation plan will require the following for new construction permit applications and standard plant applications:

- A plant/site specific probabilistic risk assessment

- Achievement of the Design Objectives

- Further safety improvements in accordance with the Benefit-Cost Guideline

In addition, the Commission had previously published an advanced notice of proposed rulemaking concerning consideration of degraded or melted cores in safety regulations (45FR65474 dated October 2, 1980). In that notice, the Commission indicated that a long-term rulemaking effort was being initiated that would establish policy, goals, and requirements relating to core-melt accidents greater than the present design basis. The current NRC direction in this area replaces the long-term generic rulemaking effort with severe accident rulemakings designed to certify specific standard plant design applications and with regulatory decisions based on generic evaluations and decisions regarding all classes of existing plants. The "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation" (48FR16014, April 13, 1983) contains the following nine inter-related components:

- Policy Statement on Safety Goals

- Use of Probabilistic Risk Assessment in Severe Accident Decision Making

- Lessons Learned from Three Mile Island

- Standard Review Plan

- Standardization Policy

- Further Research on Severe Accidents

- o Treatment of Severe Accidents in Ongoing Licensing Proceedings
- o Present Views on Other Safety Issues and Efforts in Progress
- o Implementation Guidelines for Severe Accident Policy

These inter-related components are summarized below and cross-referenced to discussions provided in various other sections of this document as appropriate. In addition, their impact on the WAPWR design is evaluated.

1. Policy Statement on Safety Goals

<u>Discussion</u>

The NRC "Policy Statement on Safety Goals for the Operation of Nuclear Power Plants" (48FR10772), as summarized earlier in Section 3.2, has the ultimate goal of explicity defining the acceptable level of risk to the public health and safety from nuclear power reactors; and includes several qualitative safety goals and quantitative probabilistic guidelines for severe accidents.

The impact of the Commission's safety goal policy on the WAPWR design is encompassed by the "WAPWR Response" given under item 2, below

2. Use of Probabilistic Risk Assessment in Severe Accident Decision Making

<u>Discussion</u>

The NRC has concluded that the use of probabilistic risk assessment (PRA) techniques improves the "understanding of the severe accident sequences to which plants are most vulnerable and therefore of the dominant constituents of the risk posed by specific plants." The performance of a plant/site specific PRA is a regulatory requirement (see Section 3.1, item 1; 10CFR50.34(f)(1)(i)).

The NRC has further stated that "the utility of PRA can be improved if it is integrated with the design process." Therefore, the severe accident policy statement, once promulgated, will likely require "the performance of a PRA that is as complete as practical for any standardized design to be referenced in future construction permit applications."

WAPWR Response

The WAPWR development program is dedicated to using probabilistic risk assessment both in selecting component and system designs from among alternatives, and in system reliability evaluations. Westinghouse will completely document an integrated design and siting PRA for the WAPWR. For the PDA, the PRA will be "as complete as practical." The WAPWR risk assessment will:

- o Include internal and external hazards as well as other accident indicators.
- o Reflect any operational limitations and requirements, thereby establishing appropriate technical specifications as an integral part of the design.
- o Demonstrate that the WAPWR design has a sufficiently low risk that further consideration of the benefit-cost guideline (ALARA policy) is not necessary and, in that regard, establish that the WAPWR design is immune from consideration of new regulatory requirements.

In addition, Westinghouse agrees with the NRC staff that the Commission safety goal should not consider risks from sabotage. However, as discussed in Section 5.1, item 29, Westinghouse will perform a sabotage assessment for the WAPWR design using risk models which is intended to be provided to utilities utilizing the design for consideration in their physical protection plans.

3. Lessons Learned From Three Mile Island (TMI)

Discussion

The lessons learned from TMI, as summarized in NUREG-0737, "Clarification of TMI Action Plan Requirements," have been applied by the NRC to operating plants, plants in operating license review, and plants now undergoing construction permit review. These requirements, which apply to future construction permit applications as well, have been codified in 10CFR50.34(f). A complete discussion of 10CFR50.34(f), including its impact on the WAPWR design, is given in Section 3.1.

4. Standard Review Plan

Discussion

On March 19, 1982, the NRC incorporated a paragraph (g) to 10CFR50.34 that requires future applicants for operating licenses, construction permits, manufacturing licenses, and for preliminary or final design approvals for standard plants to identify and evaluate differences from the acceptance criteria of the applicable revision of the Standard Review Plan (SRP) as part of the technical information to be submitted as part of an application.

The SRP rule and its impact on the WAPWR design is fully discussed in Section 6.3.

5. Standardization Policy

Discussion

The NRC has reiterated its support for standardization of nuclear power plants. This proposed rule requires applicants for PDAs and FDAs to address the guidance contained therein. Once a standard design has been taken through rulemaking, the approval would be binding on both the NRC and the applicants "for a period of ten years unless significant new

safety information becomes available." Applicants who intend to take their standard design through a rulemaking procedure, per 10CFR Part 50, Section 7, will be given review priority by the NRC.

WAPWR Response

This "Regulatory Conformance" document, in conjunction with the supporting RESAR SP/90 PDA submittals, demonstrates the level of conformance with the requirements of the proposed Commission policy statement on severe accidents. Westinghouse fully intends to pursue the completion of a rulemaking for the WAPWR design upon receipt of an FDA, consistent with the NRC objectives for licensing standard plant designs as embodied by this proposed rule.

6. Further Research on Severe Accidents

Discussion

The NRC is conducting research on severe accidents. The NRC does not expect its fundamental views on severe accident considerations to change substantially due to the on-going NRC research or industry sponsored research. The intent is to obtain sufficient information in about 2 years to complete decision making. The NRC research program includes the following.

- o Probabilistic risk assessment methods, including those treating external events;

- o Common-cause accident contributors;

- o System interactions, including analysis of systems transients involving core damage;

- Accident management, including guidelines for recovery from a core-damaging event;

- Phenomenological research on fuel and fission product behavior of damaged cores and containment response to severe loadings;

- Human factors;

- Applications research on behavior of existing systems and components in the severe accident environment;

- Fission product release and transport; and

- Safety-cost tradeoff analysis of changes in hardware.

- An improved methodology for probabilistic risk assessment plus a significant extension of the data base for severe accident assessment;

- Data for a better estimate of the radiological source term used to assess accident consequences;

WAPWR Response

Westinghouse is taking an active role in industry research programs related to severe accidents including participation in the Industry Degraded Core Rulemaking (IDCOR) program. The results of such programs will be continually factored into the design process for the WAPWR. In addition, Westinghouse will follow and give careful consideration to the results of the NRC study and research programs summarized above, and factor such results into the WAPWR design process as appropriate. As discussed throughout this document, the WAPWR design and licensing program is dedicated to conscientiously addressing the above listed severe accident considerations for which the NRC study and research program will investigate. For example, (1) PRA techniques are used throughout the

design process, including the selection of design features from among alternatives; (2) a systems interaction study will be performed and documented during the licensing process for the WAPWR; (3) emergency response guidelines will be developed and documented for the WAPWR; (4) human factors principles will be integrated into the overall control room design; and (5) Westinghouse intends to utilize a more realistic radiological source term in the assessment of accident consequences for the WAPWR.

7. Treatment of Severe Accidents in Ongoing Licensing Proceedings

Discussion

With respect to the impact of this severe accident policy statement on operating plants and plants under construction, the NRC has concluded: (1) individual licensing proceedings, including hearings, are not appropriate forums for examination of the Commission's regulatory requirements on accidents more severe than the design basis, (2) the requirements of 10CFR50.34(f) are sufficient for Class 9 accident review and hearings, and (3) review of non-standardized plants can proceed and be found acceptable for severe accident concerns if they meet 1) the TMI requirements, 2) Standard Review Plans and 3) achieve resolution of the Unresolved Safety Issues.

The Industry Degraded Core Rulemaking (IDCOR) program is directed at information that can be gained from operating plants, including non-nuclear plants, relative to systems, components and functions that relate to the potential for a degraded core accident, and the NRC is interested in assuring that the IDCOR program and the NRC program are coordinated and complimentary.

WAPWR Response

This section of the severe accident policy statement is primarily concerned with the impact it has on currently operating plants and plants

under construction. As stated in the "WAPWR Response" to item 6 above, Westinghouse will closely follow the NRC severe accident study and research program, and factor the results into the WAPWR design process as appropriate.

8. Present Views on Other Safety Issues and Efforts In Progress

A. Striking a Balance Between Accident Prevention and Consequence Mitigation

Discussion

As a result of the TMI accident, the NRC developed an objective (for themselves as well as the nuclear industry) to give further consideration to severe accidents beyond the design basis, and to explore means to decrease the probability as well as mitigate the consequences of such accidents. For example, there has been increased recognition that one of the most important systems in providing core-melt prevention is a reliable decay heat removal system.

WAPWR Response

A fundamental design objective of the WAPWR is to significantly decrease the probability of a core-melt compared to current plant designs. There are numerous design features and systems in the WAPWR, as discussed throughout this document, which substantially reduce the probability of and/or mitigate the consequences of a severe accident.

B. Containment Strength

Discussion

The NRC has identified the need to gain a better understanding of containment building failure characteristics and design features or

emergency actions that decrease the likelihood of containment building failure. Following the outcome of the NRC severe accident research program, the NRC will decide whether to establish performance criteria for containment systems.

In addition, the NRC is studying the need for the following additional containment features, each of which are discussed in the following paragraphs:

- o Filtered venting of containment;
- o Core-retention devices; and
- o Hydrogen control features.

<u>WAPWR Response</u>

10CFR50.34(f)(3)(v) encompasses the severe accident policy statement concern of the adequacy of the containment design for future plants. See Section 3.1, item 33 for a discussion of this issue and its impact on the WAPWR design.

C. Filtered-Vented Containment Systems

<u>Discussion</u>

The NRC has stated that for future construction permit applicants, "filtered-vented containment systems, or a variation of such systems, should be provided if these yield a cost-effective reduction in risk."

<u>WAPWR Response</u>

The subject of the need for filtered-vented containment systems is encompassed by 10CFR50.34(f)(iv). See Section 3.1, item 32.

D. Core-Retention Devices

Discussion

The NRC has stated that "studies (such as NUREG-0850) of large, dry containment buildings indicate that classical core-retention devices are probably not cost-effective in reducing the release of radioactive materials to the atmosphere. However, unique basemat designs and unique or undesirable liquid-pathway characteristics should be carefully weighed in future construction permit applications before deciding that this concept can be dismissed."

WAPWR Response

As part of the WAPWR design risk assessment, Westinghouse will evaluate the potential benefits of a core-retention device. At present, the design is proceeding assuming such a feature is not cost-benefit effective.

E. Hydrogen Control Systems

Discussion

The NRC intends to require hydrogen control systems to deal with degraded-core accidents for all dry containments, ice condensor containments, and the Mark I, II, and III containments. In addition, they stated the cost-effectiveness of combustible gas control systems for accidents proceeding with core melt and vessel melt-through and large combustible gas releases should be examined for future construction permit applications.

WAPWR Response

The subject of hydrogen control systems and their impact on the WAPWR design can be found in the following portions of this document:

Section 3.1, items 5, 14, 33 and 34
Section 3.3, item 5
Section 4.0, item 26
Section 6.1.1, item 3
Section 6.1.2.1, item 4

F. Reliable Containment Heat Removal

Discussion

The NRC is studying the need for more reliable subsystems for containment heat removal as possible alternatives to filtered venting for prevention of gradual over-pressurization failure of the containment building. In addition, the NRC again emphasizes the need to assure high reliability of decay heat removal systems. Both of these items are to be addressed by applicants for standard design approvals.

WAPWR Response

See Section 3.1, item 32 for a discussion of the WAPWR position relative to filtered-vented containments. WAPWR design features aimed at improving the reliability of decay heat removal systems are presented in Section 4.0, item 23.

G. Other Consequence Mitigation Measures

The NRC recognizes that core-melt consequence mitigation design features and procedures should be evaluated on as realistic a basis as

possible, i.e., such features are there to mitigate the consequences of extremely low probability events. Thus, the acceptance criteria for such features will be established accordingly. In addition, the NRC recognizes that such design and operational improvements for core-melt mitigation will have certain attendant risks; and that these must not be ignored.

WAPWR Response

For the WAPWR, the attendant risks associated with any such core-melt mitigation design features will be factored into the probabilistic risk assessment.

H. External Events, Human Errors, and Sabotage

Discussion

The NRC expects that applicants for standard design approvals will address in the Safety Analysis Report the relation to severe accident considerations of sabotage and external events as well as other accident initiators such as multiple human errors and design errors.

WAPWR Response

As discussed in the "WAPWR Response" to item 2 above and in Section 5.1, item 29, the WAPWR PRA will include internal and external hazards as well as other accident initiators. In addition, Westinghouse will perform a sabotage assessment for the WAPWR.

I. Siting Policy

Discussion

A modified siting rule applicable to future plants which incorporate new radioactive source term information for severe accidents is expected to be issued for comment in the near future.

WAPWR Response

See Section 6.1.2.2, item 2, for a discussion of this issue and its impact on the WAPWR design.

9. Implementation Guidelines for Severe Accident Policy

Discussion

The NRC has established the following conditions for standard designs for reference in future construction permit applications or in reactivations of previously docketed construction permit applications:

- Demonstration of compliance with current Commission regulations
- Completion of a PRA before a standard design can be taken through rulemaking
- Completion of a Staff review of safety acceptablity; the review will be based upon the Standard Review Plan (NUREG-0800).
- Consideration of all applicable Unresolved Safety Issues
- Adherence to the post-TMI requirements as set forth in the CP rule.

WAPWR Response

The purpose of this "Regulatory Conformance" document is to establish the Westinghouse position for the WAPWR design relative to NRC regulations, regulatory guidance and policy, and generic safety issues; including the five items listed above.

3.3 OTHER POST-TMI ISSUES

3.3.1 NUREG-0737

NUREG-0737, "Clarification of TMI Action Plan Requirements," contains those post-TMI requirements that have been approved for implementation by the Commission for operating plant licensees and applicants. In many cases, the specific requirements of NUREG-0737 are identical to those of NUREG-0718/10CFR 50.34 discussed in Section 3.1. However, the NRC has determined that certain of the items contained in NUREG-0737 are not applicable at the construction permit stage and are, therefore, not included as requirements in NUREG-0718/10CFR 50.34. Westinghouse believes that the NRC does not intend to imply that certain requirements imposed on operating or near-term operating plants are not applicable to a later vintage plant, but simply that certain requirements can be more appropriately addressed at the operating license stage. Therefore, the WAPWR design will include appropriate consideration of the additional requirements of NUREG-0737.

1. Pressurizer Water Level (NUREG-0737, Item II.K.1.17)

   Discussion

   This item is really applicable to certain older generation Westinghouse operating plants that utilized a low pressurizer level coincident with low pressurizer pressure logic to provide a safety injection signal. This design feature is not utilized in current-day Westinghouse designs.

   WAPWR Response

   This design feature has not been included in the WAPWR design.

2. Thermal Mechanical Report - Effect of High Pressure Injection on Vessel Integrity for Small-Break LOCA with no Auxiliary Feedwater (NUREG-0737, Item II.K.2.13)

Discussion

Refer to the discussion of Unresolved Safety Issue A-49, "Pressurized Thermal Shock," in Section 4.0 (item 27).

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

   Discussion

   Refer to the discussion of item 4 in Section 3.1.

4. Automatic Trip of Reactor Coolant Pumps During LOCA (NUREG-0737, Item II.K.3.5)

   Discussion

   For this item the NRC considered a requirement for plant designs to incorporate automatic tripping of the reactor coolant pumps in the case of a small-break LOCA. This item and its impact on the WAPWR design is fully discussed in Section 6.4, Items 92 and 93.

5. Emergency Preparedness (NUREG-0737, Item III.A.2)

   Discussion

   Refer to the discussion of 10CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," in Section 6.1.1 (item 1).

3.3.2 NUREG-0660

Certain of the post TMI issues identified in NUREG 0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," are related to on-going NRC activities and are not included in the set of post-TMI issues currently

approved by the Commission for implementation through NUREG-0718/10CFR 50.34 or NUREG-0737. Since the NRC has made it clear that future requirements may result from the activities related to these issues, some level of consideration of the potential impact of these issues will be given in the design of the WAPWR. In certain instances, the NRC program and activities are not currently well defined and appropriate consideration might be simply to be aware of the issue and possible courses of NRC action.

1. Control Room Design Standard (NUREG-0660, Item I.D.4)

   Discussion

   The NRC plans to develop guidance for the designs of future control rooms. This NRC guidance is anticipated to be in the form of a regulatory guide that endorses IEEE Standards 566 and 567 concerning the design of display and control functions and the design of the control room complex, respectively. The IEEE schedule for issuance of the post-TMI versions of these standards is uncertain.

   WAPWR Response

   Westinghouse will give appropriate consideration to any available drafts of these standards in the design of the WAPWR control room.

2. Improved Control Room Instrumentation Research (NUREG-0660, Item I.D.5)

   Discussion

   This item deals with NRC Office of Nuclear Regulatory Research initiated studies aimed at developing new (longer term) instrumentation to enhance the performance of the control room operator.

The specific studies are related to:

- o Alarms and displays for improving the man-machine interface in reactor control rooms.
- o Plant status monitoring to improve the ability of reactor operators to prevent, diagnose, and properly respond to accidents.
- o On-line reactor surveillance utilizing noise diagnostic and pattern recognition techniques.
- o Disturbance analysis system feasibility and development.

WAPWR Response

These studies are in various degrees of completion; however, Westinghouse will give appropriate consideration to any available results in the design of the WAPWR control room.

3. Siting Policy Reformulation (NUREG-0660, Item II.A.1)

Discussion

Refer to the discussion of the NRC advanced notice of rulemaking concerning reactor siting criteria in Section 6.1.2.2 (item 2).

4. Research on Phenomena Associated with Core Degradation and Fuel Melting (NUREG-0660, Item II.B.5)

Discussion

The NRC Office of Nuclear Regulatory Research is conducting major research programs associated with core degradation and fuel melting. These programs are intended to support the basis for rulemaking and confirm certain

licensing decisions related to core degradation and fuel melting. Refer to the discussion of the NRC activities related to degraded or melted cores in Section 3.2.

5. Analysis of Hydrogen Control (NUREG-0660, Item II.B.7)

   Discussion

   Specific NUREG-0718/10CFR 50.34 requirements related to hydrogen control for the WAPWR design are discussed in Section 3.1 (item 14).

6. Continuation of Interim Reliability Evaluation Program (NUREG-0660, Item II.C.2)

   Discussion

   This item deals with possible future NRC requirements for operating plant licensees to perform probabilistic reliability/risk studies. Specific requirements related to probabilistic risk assessment for the WAPWR design are discussed in Section 3.2.

7. Systems Interaction (NUREG-0660, Item II.C.3)

   Discussion

   This item is actually a subpart of the overall issue of "Systems Interactions in Nuclear Power Plants" (Unresolved Safety Issue A-17). Refer to the discussion of Unresolved Safety Issue A-17 in Section 4.0 (item 13).

8. Update Standard Review Plan and Develop Regulatory Guide (NUREG-0660, Item II.E.1.3)

   Discussion

   This item deals with NRC activities related to:

- o Updating Standard Review Plan 10.4.9, "Auxiliary Feedwater System," to include TMI-2 lessons learned recommendations/ requirements.

- o Issuing a new regulatory guide on auxiliary feedwater system designs that will possibly endorse ANSI/ANS 51.10 "Auxiliary Feedwater System for Pressurized Water Reactors."

The NRC Standard Review Plan is discussed in Section 6.3. In relation to Standard Review Plan 10.4.9, the TMI-2 lessons learned recommendations/ requirements discussed in Section 3.1 (items 2 and 17) have been included in the latest version (i.e., Revision 2; July 1981).

WAPWR Response

Westinghouse has given appropriate consideration to the criteria of ANSI/ANS 51.10 in the design of the WAPWR secondary side safeguards capability.

9. Reliance on Emergency Core Cooling System (NUREG-0660, Item II.E.2.1)

Discussion

This issue involves a potential deficiency in the reliability of emergency core cooling systems (ECCS). The concern results from a higher than anticipated frequency of ECCS challenges in operating reactors, in part because of the reliance on ECCS for other than loss-of-coolant accidents. The reliability of ECCS is believed to be high, but it is not clear that it is sufficiently high to accomplish its safety function with high assurance, considering the increase in expected challenges. Further study has been recommended by the NRC to determine if this issue should be reported as an Unresolved Safety Issue. The further study would be in the form of scoping calculations related to ECCS challenges and reliability.

WAPWR Response

This issue is not applicable to the WAPWR Emergency Core Cooling System (ECCS). The WAPWR ECCS reliability has been significantly improved over the current operating reactor by utilizing four high head safety injection pumps, four core reflood tanks, and four accumulators exclusively to perform the required ECCS function. Consequently, the anticipated frequency of ECCS challenges has been essentially eliminated as compared to current operating reactors.

10. Research on Small-Break LOCAs and Anomalous Transients (NUREG-0660, Item II.E.2.2)

Discussion

The NRC is conducting research that focuses on small-breaks and transients, including experimental research at the LOFT facility, systems engineering, and materials effects programs.

Westinghouse typically follows NRC-sponsored research programs to ensure the applicability and acceptability of Westinghouse codes to accurately predict the consequences of postulated LOCAs. When appropriate, Westinghouse also provides test predictions related to specific test programs to the NRC for review and comparison purposes.

WAPWR Response

Beyond following these research programs which puts Westinghouse in a position to possibly identify any potential analytical or hardware-related problem areas, consideration of this issue in the development of the WAPWR design is not appropriate.

11. Decay Heat Removal Systems Reliability and Coordinated Study of Shutdown Heat Removal Requirements (NUREG-0660, Items II.E.3.2 and II.E.3.3)

Discussion

These items are actually subparts of the overall issue of "Shutdown Decay Heat Removal Requirements" (Unresolved Safety Issue A-45). Refer to the discussion of Unresolved Safety Issue A-45 in Section 4.0 (item 23).

12. Decay Heat Removal Alternate Concepts Research (NUREG-0660, Item II.E.3.4)

Discussion

The NRC plans to determine the technical feasibility of passive containment cooling including add-on decay heat removal systems for new plants and possible backfitting to existing plants.

WAPWR Response

Since this issue and its ultimate resolution are not sufficiently defined to permit appropriate design consideration, Westinghouse plans to follow NRC activities in relation to this issue in lieu of arbitrarily specifying requirements for the WAPWR design. Containment cooling for the WAPWR is performed by the containment spray and containment fan coolers. Passive features are not included in the design.

13. Decay Heat Removal Regulatory Guide (NUREG-0660, Item II.E.3.5)

Discussion

The NRC plans to provide improved guidance on the reliability and capability of nuclear power plant systems for removing decay heat and achieving safe shutdown conditions following transients and under post-accident

conditions. This guidance will be in the form of Revision 1 to Regulatory Guide 1.139, "Guidance for Residual Heat Removal."

WAPWR Response

Westinghouse will address Revision 1 to Regulatory Guide 1.139 in relation to the WAPWR design when it is issued by the NRC.

14. Study of Control and Protective Action Design Requirements (NUREG-0660, Item II.F.4)

Discussion

This issue involves a potential deficiency related to: (A) basing protective actions on derived variables rather than direct reading of process variables; (B) protective actions relying on coincidence of independent process variables rather than relying on either variable; and (C) lack of testing of control circuit components at expected degraded power supply conditions. The NRC believes that existing requirements already preclude these deficiencies.

WAPWR Response

Westinghouse agrees with the NRC that this issue does not present a significant safety problem. However, the WAPWR protection system will be designed to meet all applicable safety requirements.

15. Classification of Instrumentation, Control, and Electrical Equipment (NUREG-0660, Item II.F.5)

Discussion

The NRC planned to prepare a standard (in conjunction with IEEE) and a regulatory guide that endorses the standard that provides a classification

approach for determining the applicability of design criteria and design requirements for plant instrumentation, control, and electrical systems and equipment based on the level of their importance to safety. This standard, IEEE P-827, was drafted, but subsequently withdrawn by the IEEE. The industry cooperative effort planned via IEEE P-827 has been replaced by the ANS 51.1 effort.

WAPWR Response

See Section 6.1.2.1, item 5, for a discussion of Westinghouse activities relative to ANS 51.1.

16. Nuclear Data Link (NUREG-0660, Item III.A.3.4)

Discussion

This item will (when finalized) require each utility to provide equipment and interface with the NRC data acquisition system to remotely access facility data and transmit the data and display information in the NRC Operations Center.

Although not finalized (in terms of issuance for implementation), the NRC criteria for the nuclear data link are provided in NUREG-0696, "Functional Criteria for Emergency Response Facilities."

WAPWR Response

The WAPWR design for instrumentation to be incorporated as part of the onsite technical support center includes an output interface for offsite data communication. Westinghouse will give appropriate consideration to offsite data communication equipment for the WAPWR design upon finalization of NRC requirements in this area.

17. Radioactive Gas Management (NUREG-0660, Item III.D.1.2)

Discussion

The NRC plans to sponsor a future study to determine the applicability and desirability of use of available technology to minimize the release of radioactive noble gases during and following various postulated accident conditions.

At this time it is not clear if this item will have any impact on Westinghouse designs including the WAPWR design.

WAPWR Response

Since this issue and its ultimate resolution are not sufficiently defined to permit appropriate design consideration, Westinghouse plans to follow NRC activities in relation to this issue in lieu of arbitrarily specifying requirements for the WAPWR design. Presently, the design of WAPWR gaseous waste management systems is based on normal operation only.

18. Ventilation System and Radioiodine Adsorber Criteria (NUREG-0660, Item III.D.1.3)

Discussion

The NRC plans to develop future requirements for ensuring that there is adequate filtration of radioactivity in ventilation exhausts and that acceptable collection efficiencies of radioiodine adsorbers are maintained during accident conditions.

The NRC has indicated that their new requirements/recommendations will be issued as revisions to Regulatory Guides 1.52 and 1.140.

WAPWR Response

Westinghouse will address any future revisions to Regulatory Guides 1.52 and 1.140 in relation to the WAPWR design when they are issued by the NRC.

19. Radwaste System Design Features to Aid in Accident Recovery and Decontamination (NUREG-0660, Item III.D.1.4)

Discussion

The NRC plans to sponsor a future evaluation of radwaste system design features that will provide the capability to process accident-related liquids and gases and to conduct decontamination effectively.

WAPWR Response

Since this issue and its ultimate resolution are not sufficiently defined to permit appropriate design consideration, Westinghouse plans to follow NRC activities in relation to this issue in lieu of arbitrarily specifying requirements for the WAPWR design. Presently, the design of WAPWR radwaste systems is based on normal operation only.

20. Radiological Monitoring of Effluents (NUREG-0660, Item III.D.2.1)

Discussion

The NRC plans to develop future requirements for revised systems for radiological monitoring of effluents (e.g., development of atmospheric steam dump monitoring of both noble gas and radioiodine after an accident).

The NRC has indicated that their new requirements/recommendations will be issued as revisions to Regulatory Guides 1.21 and 1.97.

WAPWR Response

Westinghouse will address any future revisions to Regulatory Guides 1.21 and 1.97 in relation to the WAPWR design when they are issued by the NRC.

21. Offsite Dose Measurements (NUREG-0660, Item III.D.2.4)

Discussion

The NRC plans to sponsor a future study of the feasibility of environmental monitors capable of measuring real-time rates of exposures to noble gases and radioiodines.

WAPWR Response

This issue is not applicable to Westinghouse in relation to the WAPWR design. Environmental monitors are the responsibility of each utility utilizing the WAPWR design.