6.0 REGULATIONS AND REGULATORY GUIDANCE

6.1 NRC REGULATIONS

The Code of Federal Regulations (CFR) is a codification of the general and permanent rules published in the Federal Register by the Executive departments and agencies of the Federal Government. Title 10, "Energy," of the CFR is composed of four chapters. The first chapter, Parts 0 through 199, contains the regulations of the Nuclear Regulatory Commission (Chapter I). The other chapters contain the regulations of the Department of Energy (Chapters II, III, and X).

In simple terms, Chapter I of Title 10 of the CFR (referred to as "10CFR") provides those regulations that must be met in the design, constru on, and operation of all nuclear power plants.

An indepth discussion of each regulation of 10CFR is not appropriate for this section of this report. During the licensing process for the WAPWR design Westinghouse will document compliance with all applicable NRC regulations.

Since the NRC (as a result of its own instance, on the recommendation of another agency of the United States, or on the petition of any other interested person) is continuously in the process of issuing, amending, or repealing the regulations of 10CFR, the following sections have been included to discuss major new rules and proposed rules/rulemakings in relation to the WAPWR design.

6.1.1 New Rules

The following discussions pertain to applicable major new or revised 10CFR regulations promulgated over the past few years as they relate to the WAPWR design.

1. Emergency Planning (45 FR 55402 dated August 19, 1980 and 46 FR 63031 dated December 30, 1981) (10CFR50.54S2)

Discussion

This final rule upgrades the Commission's regulations in order to assure that adequate protective measures can and will be taken in the event of a radiological emergency.

Most of these regulations deal with a utility's organization, procedures, and training associated with emergency planning. However, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," of 10CFR Part 50 does include certain design-related requirements that are applicable to the overall WAPWR design. Specifically there are requirements for:

- o An onsite technical support center and a near-site emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency.
- o At least one onsite and one offsite communications system; each system shall have a backup power source.
- o Onsite facilities for decontamination and emergency first aid treatment.

Requirements are repeated in 10CFR50.33(g), 50.34(f)2xxv, 50.54(g) and Appendix E IV E8.

WAPWR Response

Emergency response facilities (including the onsite technical support center) are discussed in Section 3.1 (item 25).

Communications systems and other onsite emergency facilities (i.e., decontamination and first aid) will be considered in the WAPWR design consistent with the scope definition for the Nuclear Power Block which will be completely documented with the NRC during the licensing process for the WAPWR design.

2. Fire Protection Program for Operating Nuclear Power Plants (45 FR 76602 dated November 19, 1980 and 46 FR 44734 dated September 8, 1981) (10CFR50.48 and Appendix R)

Discussion

The Commission has revised its regulations to require certain provisions for fire protection in operating nuclear power plants. This final rule is in two parts:

- 10CFR 50.48, "Fire Protection."
- Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," to 10CFR Part 50.

These new regulations are written specifically for existing operating plants. Current NRC fire protection requirements for new plants are documented in Branch Technical Position CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." This Branch Technical Position actually incorporates the requirements of 10CFR 50.48 and Appendix R to 10CFR Part 50.

The NRC has indicated that they may initiate a future rulemaking to codify the fire protection requirements for new plant designs (refer to Section 6.1.2.3, item 3). It is anticipated that a rulemaking on new plant designs would result in regulations essentially requiring compliance with the current Branch Technical Position CMEB 9.5-1.

The WAPWR design will incorporate several features which should provide improved fire protection. For example, the WAPWR plant layout provides improved physical separation between safeguards trains A and B as well as between the safeguards trains and the control systems.

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Branch Technical Position CMEB 9.5-1 acceptance criteria during the licensing process for the WAPWR design.

3. Interim Requirements Related to Hydrogen Control (46 FR 58485 dated December 2, 1981) (10CFR50.44C1)

Discussion

This final rule requires:

- o Inerted containment atmospheres for boiling water reactors with either Mark I or Mark II type containments.
- o Hydrogen recombiner capability (either an internal recombiner or the capability to install an external recombiner) for plants that rely upon a purge/repressurization system as the primary means for controlling combustible gases following a loss-of-coolant accident.
- o High point vents for all plants.

Hydrogen recombiner capability and high point vents for the WAPWR design are discussed in Section 3.1 (items 34 and 11, respectively).

4. Licensing Requirements for Pending Construction Permit and Manufacturing License Applications (47 FR 2286 dated January 15, 1982 and 47 FR 4497 dated February 1, 1982) (10CFR50.34(f))

Discussion

This final rule imposes new safety requirements on pending construction permit and manufacturing license applications. The requirements stem from the Commission's ongoing effort to apply the lessons learned from the accident at TMI-2 to nuclear power plant licensing.

This final rule, 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 NRC policy statement on severe accidents and related views on nuclear reactor regulation (48FR16013 dated April 13, 1983) indicates that the requirements of the CP/ML Rule are also applicable to new construction permit applications or reactivations. Therefore, the CP/ML Rule has been treated as being applicable to the WAPWR design and is discussed in detail in Section 3.1.

5. Guidance for Implementation of the Standard Review Plant Rule (48FR23807 dated May 27, 1983 and 47FR11651 dated March 18, 1982). Effective June 27, 1983. (10CFR50.34g)

Discussion

An evaluation of conformance with the Standard Review Plan, NUREG-0800, is required as part of the Safety Analysis Report in accordance with 10CFR50.34g. The evaluation consists of:

- A review of the proposed design against applicable SRP acceptance criteria.

- Identification of only the differences preferably in tabular format in Section 1.8 of the SAR.

- o Discussion of the difference in the applicable SAR section and the reasons for the acceptable compliance.

- o A listing of Regulatory Guide compliance as called for in the standard format is no longer required.

WAPWR Response

Licensing submittals for the WAPWR design will include identification of and justification for any deviations from the SRP acceptance criteria.

6. Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants (48FR2729 dated January 21, 1983) Effective February 22, 1983 (10CFR50.49)

Discussion

This rule codifies the environmental qualification methods and criteria for electrical equipment important to safety. It is intended to clarify different interpretations (among national standards, regulatory guides and NRC publications) with legal force.

The scope of the final rule covers equipment important to safety commonly referred to as "safety-related" (essentially "Class 1E" equipment defined in IEEE 323-1974), and nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent the satisfactory accomplishment of required safety functions. Also covered in the scope of the final rule is certain postaccident monitoring equipment specified as "Category 1 and 2," in Revision 2 of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

The rule requires that each holder of an operating license provide a list of electric equipment important to safety within the scope of this rule previously qualified based on testing, analysis, or a combination thereof, and a list of equipment that has not been qualified.

Proposed Revision 1 to Regulatory Guide 1.89, which has been issued for public comment, describes methods acceptable to the NRC for meeting the provisions of this rule and includes a list of typical equipment covered by it.

The general requirements for seismic and dynamic qualification are not included within the scope of this rule. Further guidance is provided in Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants," (Revision 1) and NUREG-0800, "Standard Review Plan" SRP 3.10.

WAPWR Response

The impact of this rule on the WAPWR design is discussed in Section 4.0, item 14.

7. Fracture Toughness Requirements for Light-Water Nuclear Power Reactors (48FR24008 dated May 27, 1983 and 45FR75536 dated November 14, 1980). Effective July 26, 1983 (10CFR50.12, 50 55a, 50.60, 10CFR50 Appendices G and H)

Discussion

This rule amends parts of 10CFR50 and particularly Appendices G and H to clarify requirements for reactor vessel fracture toughness, material surveillance programs and achieve consistency with National Standards. A significant technical change imposes fracture control at discontinuities.

Ductile properties of the head flange region can be the low temperature pressure limiting condition for both the BWR and PWR and not necessarily the irradiated beltline region as on older PWR vessels.

- o When not critical and pressure exceeds 20% of the preservice hydrostatic test pressure, the temperature of closure flange regions must exceed the reference temperature by 120°F for normal operation and by 90°F for pressure and leak tests unless the beltline region can be shown as more limiting (older vessels).

- o When critical the vessel temperature must not be lower than 40°F above the minimum permissible temperature.

- o Beltline material must have Charpy upper-shelf energy of no less than 75 ft-lbs initially and maintain 50 ft-lbs through life.

- o No material surveillance program is required if peak neutron fluence does not exceed $10^{17} n/cm^2$ to the vessel. ASTM E 185-73, -79 and -82 are approved but the 1982 version applies for reporting after July 20, 1983. Reports must be submitted within one year. An integrated report program among a set of similar vessels can be approved.

WAPWR Response

The impact of this new rule on the WAPWR design is fully addressed in Section 4.0, items 11, 12, and 27.

6.1.2 Proposed Rules/Rulemakings

Included in the overall category of proposed rules/rulemakings are:

- o Proposed rules published in the Federal Register for which a final rule has not yet been issued.
- o Advance notices of proposed rulemaking published in the Federal Register for which neither a proposed nor final rule has been issued.
- o Unpublished rules or proposed rules that have not yet been published in the Federal Register for which the NRC is considering future action.

The following sections discuss major proposed rules/rulemakings from each of these categories in relation to the WAPWR design.

6.1.2.1 Proposed Rules

1. Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation (48FR16013 dated April 13, 1983; 45FR65474 dated October 2, 1980)

   Discussion

   The concept of generic rulemaking on severe accidents was introduced in TMI Action Plan, Task II.B.8. A 1980 policy statement proposed that severe accident rulemaking for standard plants and other classes of plants be done in parallel. This 1983 policy statement proposes that severe accident rulemaking be focused on the new proposed standard plants. Resolution to the severe accident concerns would be required or accomplished during the course of standard plant reviews. See Section 3.2 for a discussion of this proposed rule and its impact on the WAPWR design.

2. Technical Specifications for Nuclear Reactors (47FR13369 dated March 30, 1983; 45FR45916 dated July 8, 1980)

Discussion

The Standard Technical Specifications (STS) provide a model for plant-specific technical specifications that are incorporated as an appendix to the Operating License (OL). It is a collection of penultimate setpoints, supportive warning limits, long-term effects, administrative and punitive items. There has been a substantial growth of items and requirement details in the STS. The increasing volume lessens the likelihood that the operators attention will be focused on items of immediate importance. Also there is an increasing paperwork burden to formally effect OL changes for the update items. The licensee must make a formal request to the NRC, the NRC must issue a public notice, review any comment reaction and issue a revision.

The proposed rule would categorize the specifications into sets. Some sets will remain direct conditions for the operator and the OL but other supplemental sets would be established for safety support functional areas but not necessarily direct conditions of the operating license.

Operational Specifications - These cover all operating modes and are part of the OL. The following sets are proposed.

- o Safety Limits - These would be limits on important process variables that if exceeded would require NRC approval for restart.

- o Limiting Safety System Setting - These would be settings for automatic safety systems. If the safety device malfunctions requiring shutdown, an NRC review for appropriate action is mandatory.

- o Operating Limits and Conditions - These would be limiting conditions associated with reactivity control, cooling and release of fission products. If not met, a safe shutdown condition must be achieved.

- o Check and Test Requirements - These would be the checks and tests performed by operators during or prior to operations to confirm processes and equipment status.

- o Operational Staffing and Reporting Requirements - These would be immediate administrative controls and would be part of the OL. The set would include crew composition, responsibility, and reporting.

Principal Design Feature Specifications - These would be physical characteristics that cannot be changed without NRC approval. Although these will not be part of Appendix A to the OL, they will be a separate part of the OL.

Supplemental Specifications - These would preserve safety analyses assumptions not included in the technical specifications. They would not be a part of the OL. Changes could be made, within bounds, without prior NRC approval. There would be three subcategories.

- o Control Provisions - These relate to operating state and standby status of systems and equipment that mitigate the consequences of fire, flood, earthquake, etc.

- o Monitoring Provisions - These would be the long-term surveillance items.

- o Adminstrative Provisions - These would be the recordkeeping, review, audit and reporting items.

The proposed rule does not include backfit except by licensee requests.

WAPWR Response

The Technical Specifications for the WAPWR will be established utilizing a probabilistic risk basis in conjunction with good engineering practice and judgement. The requirements of this proposed rule will be considered in the development of the WAPWR Technical Specifications.

3. Physical Protection of Plants and Materials; Access Controls to Nuclear Power Plant Vital Areas (10CFR15937 dated March 12, 1980)

Discussion

The NRC is proposing to revise 10CFR73.55 to clarify the original intention of the rule. Access to a vital area must correlate to the need to enter the area. Personnel will display a visible code corresponding to security plan designation for the vital area. Procedures and equipment are to assure that only authorized personnel have unescorted access to vital areas. Several IE Bulletins and Notices since 1979 have developed a complex of lessons learned on access control. Specific new requirements are as follows.

- o Personnel access list for each vital area shall be established.
- o Areas and duration of access are to be commensurate with task to be performed.
- o Access lists are to be updated monthly.
- o Emergency access lists and procedures to cope with emergency conditions for each vital area are required.
- o Alarms, control equipment, communication equipment and barriers are to be listed and maintained.

WAPWR Response

Plant physical protection plans (including access controls to nuclear power plant vital areas) are the responsibility of each utility using the WAPWR design. However, see Section 5.2, item 29 for the WAPWR design response for the reduction of vulnerability to industrial sabotage.

4. Interim Requirements Related to Hydrogen Control (46 FR 62281 dated December 23, 1981 and 47 FR 8203 dated February 25, 1982)

Discussion

The Commission is considering amending its regulations to further improve hydrogen control capability during and following an accident in those light-water reactor facilities issued construction permits prior to March 29, 1979. The amendments would require improved hydrogen control systems for boiling water reactors with Mark III type containments and for pressurized water reactors with ice condenser type containments. All light-water plants not relying upon an inerted atmosphere for hydrogen control would be required to show that certain important safety systems must be able to function during and following hydrogen burning.

This proposed rule does not apply to the WAPWR design. Similar requirements for hydrogen control for new plant designs are discussed in Section 3.1 (items 5 and 14).

5. Codes and Standards for Nuclear Power Plants (47 FR 15801 dated April 13, 1982) (10CFR50.55a)

Discussion

The NRC proposed a revision to 10CFR50.55a, "Codes and Standards" that would have expanded the current version from application of the ASME Code, Section III, Class 1 requirements to the application of Class 2 and 3 requirements as well. In order to invoke the ASME Code requirements, the proposed rule changes, borrowing heavily from Regulatory Guide 1.26, defined the systems or components to which three levels of requirements (i.e., Classes 1, 2, and 3) would the applicable using a process called "classification." The included classifications, although supposedly limited to equipment scoped by the ASME Code, nevertheless differed somewhat from the national standards created by the American Nuclear Society (ANS), both past and present.

This proposed regulatory change touched off an extensive dialogue between the Nuclear Power Plant Standards Committee (NUPPSCO) of the ANS, the consensus body that developed the latest ANS version of classification, and the NRC. The dialogue included an extensive letter of comment criticizing the classification features of the proposed rule changes, a large meeting of industry representatives with the NRC staff that succeeded in reaching some accommodation, and an ad hoc working group meeting with representatives from both sides in which NRC comments on the ANS classification system resulted in classification modifications.

A verbal agreement was reached that the NRC would continue to pursue its main objective, for revising 10CFR50.55a to require the application of the ASME Code, Section III requirements for Code Classes 2 and 3. Rather than specify how to decide what ASME-scoped equipment fall into these categories, the regulation would reference Regulatory Guide 1.26 for guidance, and the NRC would revise RG 1.26 to reference the standards containing the latest ANS classification system, i.e., ANSI/ANS-51.1-1983 (for PWR plants) and ANSI/ANS-52.1-1983 (for BWR plants). As of this writing, the agreement remains intact, but no progress has been made on either the rewriting of the regulation or of RG 1.26. The referenced standards got final ANSI approval on April 29, 1983.

WAPWR Response

As a result of the NRC Generic Letter 83-28, an emphasis has been placed on the need to classify equipment, extending all the way to the parts level. The new standard, ANSI/ANS-51.1-1983, provides a means for doing this. To respond to this new emphasis, Westinghouse currently has a task force evaluating all the inputs relative to component classification and definition. This evaluation will be complete in early 1984. The WAPWR design will incorporate this Westinghouse position relative to this issue.

For discussions of other NRC considerations relative to equipment classification see item 2 of Section 6.1.2.3.

6.1.2.2 Advance Notices of Proposed Rulemaking

1. Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Plants (43 FR 57157 dated December 6, 1978) (10CFR50.46)

Discussion

This advanced notice of proposed rulemaking seeks comment on several questions concerning the acceptance criteria for emergency core cooling systems. The Commission is considering changing certain technical and nontechnical requirements within the existing rule (10CFR 50.46). These changes are intended to provide improvements to the rule which would eliminate previous difficulties encountered in applying the rule and improve licensing evaluations in light of present knowledge, while preserving a level of conservatism consistent with that knowledge.

WAPWR Response

Westinghouse emergency core cooling analyses for the WAPWR design (in accordance with the criteria of 10CFR 50.46) will be performed using the latest Westinghouse 10CFR Part 50, Appendix K models approved by the NRC.

2. Modification of the Policy and Regulatory Practice Governing the Siting of Nuclear Power Reactors (45 FR 50350 dated July 29, 1980)

Discussion

This advanced notice of proposed rulemaking seeks comment on a proposal that would replace the existing reactor site criteria applicable to the licensing of nuclear power reactors with demographic and other siting criteria. The proposed rule would also establish siting requirements that are independent of design differences between nuclear power plants. A separate rulemaking would establish the minimum engineered safety features for new plants.

Included in the advanced notice of proposed rulemaking was a request for comment on many of the siting recommendations contained in NUREG-0625, "Report of the Siting Policy Task Force."

Subsequently the NRC has indicated a redirection in their siting rulemaking efforts from that published in this advanced notice.

WAPWR Response

Plant siting will be dealt with in a generic enveloping manner for the WAPWR design and completely documented during the licensing process.

3. Safety Goal Development Program (48FR10772 dated March 14, 1983 and 47FR7023 dated February 17, 1982)

Discussion

The Commission has decided to adopt qualitative safety goals supported by design objectives for a two year evaluation period.

- o The qualitative goals are 1) no significant additional risk to individuals and 2) risks to society should be comparable or less than that of competing (non-nuclear) electricity generating plants.

- o The quantitative design objectives are 1) the prompt fatality risk to the individual in the vicinity of a nuclear power plant should not exceed 0.1% of the sum of prompt fatality risks from other accidents, and 2) the risk for cancer fatalities should not exceed 0.1% of the sum of cancer fatalities risks from other causes.

- o The cost guideline for safety improvements is $1000 per person-rem averted. The cost guideline does not replace backfit regulation, 10CFR50.109.

- o The likelihood large-scale core melt should be less than one in ten thousand per year.

A staff evaluation plan addresses ways to use the safety goals during the 2 year trial period such as evaluating priorities among the safety issues and evaluating the incorporation of risk assessment techniques in future regulations.

WAPWR Response

A discussion of safety goals and the impact on the WAPWR is discussed in Section 3.2.

6.1.2.3 Unpublished Rules

1. Laboratory Accreditation Program (46FR57206, dated November 20, 1981)

Discussion

Although the NRC issued the above advance notice the Commission terminated further action by not publishing the proposed rule. The proposed rule would have required equipment qualification testing in laboratories accredited by IEEE procedures.

The NRC decided not to qualify the qualifiers and the proposed rule was withdrawn. IEEE is dealing with the NRC for reimbursement of their "on-good-faith" effort in their preparations for the anticipated program.

The rule for qualification of electrical equipment is discussed in item 6 of section 6.1.1.

WAPWR Response

Westinghouse testing facilities are aware of this potential requirement and will apply for accreditation should the need arise.

2. Applicability of Appendix B to Appendix A (10CFR Part 50)

Discussion

The NRC originally intended to issue a proposed rulemaking to extend the requirement of quality assurance beyond the traditional application of 10CFR50, Appendix B to safety-related equipment early in 1984. The genesis was TMI Action Plan item I.F.1, expressing need for an "Expand QA List." The idea is to apply quality assurance of lesser stringency than that applied to safety-related equipment to equipment considered to have some importance to safety, even though not safety-related. To do this, the NRC would define a category of "important-to-safety" that is not safety-related and had plans to issue a generic letter that would require the utilities to explain how they now classify equipment.

The Committee to Review Generic Requirements (CRGR) turned back a staff proposal to officially give status to the definition of "important-to-safety." This occurred after the ASLBs for both the Three Mile restart and the Shoreham operating license gave recognition to this category. This CRGR action caused a delay in NRC staff rulemaking plans and some NRC confusion about how to deal with the problem.

Meanwhile, EG&G of Idaho made an aborted effort to clarify NRC intents, but had its contract with NRC canceled after its first report of the subject ran into heavy industry criticism. However, NRC has a long-term contract with Sandia to study the problem and to make reports in three phases. The first-phase report, entitled "Logical Framework for Identifying Equipment Important to Safety in Nuclear Power Plants," NUREG/CR 2631, has been issued.

WAPWR Response

For discussion of additional NRC actions on equipment classification see item 5 section 6.1.2.1.

For new plant designs the CP/ML Rule (which is discussed in detail in Section 3.1) already includes a regulation that is very similar (in basic intent) to that proposed above. This existing regulation and a corresponding response for the WAPWR design is provided in Section 3.1 (item 30).

3. Fire Protection for Future Plants (10CFR50.48 and 10CFR50 Appendix R)

Discussion

The requirements for fire protection for plants licensed prior to January 1, 1979 are discussed in section 6.1.1 item 2. New plants are to meet

Branch Technical Position CMEB 9.5-1 of NUREG-0800 and Regulatory Guide 1.120. The Commission has approved the preparation of a fire protection rule for new plants but the NRC staff will delay rulemaking to incorporate the thrust of exemption decisions on the existing rule for operating plants. An improved technical bases may result.

The WAPWR design will incorporate several features which should provide improved fire protection. For example, the WAPWR plant layout provides improved physical separation between safeguards trains A and B as well as between the safeguards trains and the control systems.

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Branch Technical Position CMEB 9.5-1 acceptance criteria during the licensing process for the WAPWR design.

4. Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors

Discussion

The rule when published would update 10FR50 Appendix J on containment leak testing. ANSI/ANS 56.8 on leak testing procedures would reference and replace most of the procedures description currently contained in Appendix J. In practice the local and overall leakage results are reported as a value after correction. The NRC would like to implement more reporting of the as-found leak conditions so that credit can be established for corrections of local leaks. The NRC believes this can avoid the need to repeat the overall leak tests. ANSI/ANS 56.8 would be referenced for the majority of procedures and the revised Appendix J will contain the criteria.

The notice for a proposed rule may be issued during 1984.

WAPWR Response

Westinghouse will follow regulatory activities related to this issue and factor into the WAPWR design any NRC recommendations, as appropriate.

6.2 REGULATORY GUIDES

NRC regulatory guides are issued to describe and make available to the public methods acceptable to the NRC staff for implementing specific parts of the Commissions's regulations (10CFR Chapter I), to delineate techniques used by the NRC staff in evaluating specific problems or postulated accidents, or to provide guidance to applicants. Regulatory guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in regulatory guides are acceptable to the NRC if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

Regulatory guides are issued by the NRC in the following 10 broad divisions:

- o Division 1 - Power Reactors
- o Division 2 - Research and Test Reactors
- o Division 3 - Fuels and Materials Facilities
- o Division 4 - Environmental and Siting
- o Division 5 - Materials and Plant Protection
- o Division 6 - Products
- o Division 7 - Transportation
- o Division 8 - Occupational Health
- o Division 9 - Antitrust and Financial Review
- o Division 10 - General

Division 2, 3, 6, 7, 9, and 10 regulatory guides do not apply to Westinghouse in relation to the WAPWR design. The following sections provide a summary discussion of NRC Division 1, 4, 5, and 8 regulatory guides applicable to the WAPWR design.

Westinghouse will document the detailed level of conformance with applicable NRC regulatory guides during the licensing process for the WAPWR design. Any deviations from the NRC regulatory positions will be either justified or eliminated.

6.2.1 Division 1 Regulatory Guides - Power Reactors

Currently there are approximately 160 Division 1 regulatory guides that have been issued by the NRC for implementation or for comment.

As mentioned above, Westinghouse will document the detailed level of conformance with applicable regulatory guides during the licensing process for the WAPWR design. Table 6-1 provides a listing of current NRC Division 1 regulatory guides that have been matrixed to indicate applicability to structures, systems, and components; administrative programs (including training, quality assurance, procedures, etc.); analytical areas; siting; etc.

6.2.2 Division 4 Regulatory Guides - Environmental and Siting

Most of the issued and proposed NRC Division 4 regulatory guides deal with environmental monitoring procedures/programs and environmental reports for nuclear plants and other facilities. These areas are not, in general, applicable to Westinghouse in relation to the WAPWR design. However, one Division 4 regulatory guide (i.e., Regulatory Guide 4.7) does merit discussion in relation to the WAPWR design.

Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations," provides guidelines for identifying suitable candidate sites for nuclear power stations. The guidance of this regulatory guide will be considered as appropriate in the establishment of the WAPWR site envelope.

6.2.3 Division 5 Regulatory Guides - Materials and Plant Protection

Most of the issued and proposed NRC Division 5 regulatory guides deal with special nuclear material control/accounting procedures and physical security for nuclear plants and other facilities. These areas are not, in general, applicable to Westinghouse in relation to the WAPWR design. However, one Division 5 regulatory guide (i.e., Regulatory Guide 5.1) does merit discussion in relation to the WAPWR design.

Regulatory Guide 5.1, "Serial Numbering of Fuel Assemblies for Light-Water-Cooled Nuclear Power Reactors," describes a method acceptable to the NRC staff for numbering identification of fuel assemblies. Basically, Regulatory Guide 5.1 endorses ANSI N18.3-1972, "Fuel Assembly Identification." Westinghouse methods for fuel assembly identification for the WAPWR design will be in accordance with ANS 57.8-1978, "Fuel Assembly Identification" (Revision 1 to N18.3-1972).

6.2.4 Division 8 Regulatory Guides - Occupational Health

Most of the issued and proposed NRC Division 8 regulatory guides deal with occupational radiation monitoring programs for nuclear plants and other facilities. These areas are not, in general, applicable to Westinghouse in relation to the WAPWR design. However, three Division 8 regulatory guides (i.e., Regulatory Guides 8.8, 8.10, and 8.19) do merit discussion in relation to the WAPWR design.

1. Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Stations will be As Low As is Reasonably Achievable (ALARA)"

   Basically, Regulatory Guide 8.8 provides NRC guidance for meeting the requirements of 10CFR Part 20, "Standards for Protection Against Radiation," which state that every reasonable effort should be made to maintain radiation exposures ALARA. Regulatory Guide 8.8 includes guidance in the following areas for maintaining radiation exposures ALARA:

   - Overall program (e.g., policy, organization, and training)
   - Facility and equipment design features
   - Radiation protection program
   - Radiation protection facilities, instrumentation, and equipment

Regulatory Guide 8.8 is written primarily for utility applicants and licensees. However, Westinghouse has responsibilities related to the effort of maintaining occupational radiation exposures ALARA and, therefore, Westinghouse has established policy, design, and operational considerations that will be applied in the WAPWR design in accordance with Regulatory Guide 8.8. These considerations will be discussed in detail during the licensing process for the WAPWR design.

2. Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable."

   Basically, Regulatory Guide 8.10 is very similar to Regulatory Guide 8.8 (discussed in item 1 above) in that they both deal with the concept of maintaining occupational radiation exposures ALARA in accordance with 10CFR Part 20. The main difference between the two guides is that Regulatory Guide 8.8 provides guidance on what information relevant to ALARA should be included in licensing submittals and Regulatory Guide 8.10 describes an operating philosophy that the NRC staff believes should be followed to keep occupational radiation exposures ALARA.

   As mentioned in item 1 above, Westinghouse has established policy, design, and operational considerations that will be applied in the WAPWR design for ensuring ALARA occupational radiation exposures. These considerations will be discussed during the licensing process for the WAPWR design.

3. Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates."

   Basically, Regulatory Guide 8.19 describes a method acceptable to the NRC staff for performing an assessment of collective occupational radiation dose as part of the ongoing design review process involved in designing a light water cooled power reactor so that occupational radiation exposures will be ALARA. Regulatory Guide 8.19 includes guidance for estimating occupational radiation exposures (principally during the design stage) as a result of:

- Reactor operations and surveillance
- Routine maintenance
- Waste processing
- Refueling
- Inservice inspection
- Special maintei ance

Occupational radiation exposure estimates in accordance with Regulatory Guide 8.19 will be completely documented during the licensing process for WAPWR design.

6.3 STANDARD REVIEW PLAN

The NRC Standard Review Plan (SRP) was prepared for the guidance of NRC staff reviewers in performing safety reviews of applications to construct or operate nuclear power plants. The current version of the SRP is documented in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition."

The principal purpose of the SRP is to assure the quality and uniformity of NRC staff reviews and to present a well-defined base from which to evaluate proposed changes in the scope and requirements of reviews. It is also a purpose of the SRP to make information about regulatory matters widely available and to improve communication and understanding of the NRC staff review process by interested members of the public and the nuclear power industry.

The NRC safety review is primarily based on the information provided by an applicant in a Safety Analysis Report (SAR). Section 50.34, "Contents of Applications, Technical Information," of 10CFR Part 50 of the Commission's regulations requires that each application for a construction permit for a nuclear facility shall include a Preliminary Safety Analysis Report (PSAR) and that each application for a license to operate such a facility shall include a Final Safety Analysis Report (FSAR). The SAR must be sufficiently detailed to permit the NRC staff to determine whether the plant can be built and operated without undue risk to the health and safety of the public. Prior to submission of the SAR, an applicant should have designed and analyzed the plant in sufficient detail to conclude that it can be built and operated safely. The SAR is the principal document in which the applicant provides the information needed to understand the basis upon which this conclusion has been reached.

Section 50.34 specifies, in general terms, the information to be supplied in a SAR. The specific information required by the NRC staff for an evaluation of an application is identified in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition."

The SRP sections are keyed to the Standard Format, and the SRP sections are numbered according to the section numbers in the Standard Format. Review plans have not been prepared by the NRC for SAR sections that consist of background or design data which are included for information or for use in the review of other SAR sections.

In accordance with the recent Commission regulations of 10CFR 50.34, Westinghouse will completely document and justify any deviations from the SRP acceptance criteria during the licensing process for the WAPWR design.

6.4 NRC GENERIC LETTERS

In 1981 the NRC began to issue "Generic Letters" to some or all licensees of operating plants, applicants for operating licenses, and holders of construction permits. The purpose of these letters varies; some state licensing requirements and effectivity dates, others merely request comments and suggested directions for proposed rulemakings, and some set NRC administrative policy. For those which state requirements and make specific requests for information, compliance is required.

The following discussions pertain to NRC generic letters issued to date in relation to the WAPWR design.

1. Generic Letter 81-01 (May 4, 1981): Qualification of Inspection, Examination, and Testing and Audit Personnel

   Discussion

   This letter was sent to all licensees of operating plants, applicants for operating licenses, and holders of construction permits and requires all licensees of operating plants and holders of construction permits to document within 90 days, commitments to meet certain regulatory positions given in Regulatory Guide 1.58, "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel," and Regulatory Guide 1.146, "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants," or descriptions of alternative methods for complying with 10CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," in these areas.

   WAPWR Response

   The WAPWR design and testing process will follow established NRC approved Westinghouse quality assurance plans to demonstrate compliance with 10CFR Part 50, Appendix B.

2. Generic Letter 81-02 (January 27, 1981): Analysis, Conclusions, and Recommendations Concerning Operator Licensing

Discussion

This letter was sent to all licensees of operating plants, applicants for operating licenses, and holders of construction permits and requests the industry to review and comment on a NRC consultant's report, NUREG/CR-1750, "Analysis, Conclusions, and Recommendations Concerning Operator Licensing." The NRC staff had the study performed in order to assist them in revising current criteria and developing additional criteria for control room operator training and licensing. NUREG/CR-1750 concludes that a training program on a plant specific simulator is superior to one which is not an exact duplicate of the trainee's control room.

WAPWR Response

Operator training is the responsibility of each utility utilizing the WAPWR design.

3. Generic Letter 81-03 (February 26, 1981): Implementation of NUREG-0313, Rev. 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping (Generic Task A-42)"

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

4. Generic Letter 81-04 (February 25, 1981): Emergency Procedures and Training for Station Blackout Events

Discussion

This letter was sent to all licensees of operating plants and applicants for operating licenses as a result of an Atomic Safety and Licensing Appeal Board decision (ALAB-603) that station blackout (i.e., loss of all offsite and onsite AC power) be considered a design basis event for St. Lucie Unit No. 2. It requested within 90 days a documented assessment of current or planned facility emergency procedures and training to mitigate a station blackout event.

This letter emphasizes procedures and training relative to a station blackout event.

WAPWR Response

This issue of station blackout is being generically investigated by the NRC under Unresolved Safety Issue A-44 (refer to Section 4.0, item 22).

5. Generic Letter 81-05 (January 19, 1981): Information Regarding the Program for Environmental Qualification of Safety-Related Electrical Equipment

Discussion

This letter was sent to all licensees of operating plants, applicants for operating licenses, and holders of construction permits and clarifies certain requirements relative to environmental qualification of electrical equipment. Specifically, environmental qualification information is required for electrical equipment necessary to achieve and maintain a cold shutdown. For plants licensed for a hot "safe shutdown", information is required describing one path to achieve a cold shutdown.

WAPWR Response

The issue of environmental qualification of electrical equipment is being generically investigated by the NRC under Unresolved Safety Issue A-24 (refer to Section 4.0, item 14).

6. Generic Letter 81-06 (February 26, 1981): Periodic Updating of Final Safety Analysis Reports (FSARs)

Discussion

This letter was sent to all construction permit holders and applicants for operating licenses to acquaint them with the recently issued regulations of 10CFR Part 50.71(e) requiring holders of operating licenses to update their FSAR within 24 months of receipt of their operating license or July 22, 1982, whichever is later. Thereafter, it must be updated at least annually.

WAPWR Response

This letter involves an ongoing administrative activity that has no impact on the WAPWR design.

7. Generic Letter 81-07 (February 3, 1981): Control of Heavy Loads

Discussion

This letter was sent to licensees of operating plants, applicants for operating licenses, and holders of construction permits and enclosed information inadvertently left out of an earlier letter (dated December 22, 1980) on the subject of control of heavy loads. The December 22, 1980 letter included NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and guidelines for complying with this NUREG. Also, interim actions were defined until full compliance with the NUREG is demonstrated.

WAPWR Response

The issue of control of heavy loads is being generically investigated by the NRC under Unresolved Safety Issue A-36 (refer to Section 4.0, item 17).

8. Generic Letter 81-08 (January 29, 1981): ODYN Code

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

9. Generic Letter 81-09 (January 23, 1981): BWR Scram Discharge System

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

10. Generic Letter 81-10 (February 18, 1981): Post-TMI Requirements for the Emergency Operations Facility

Discussion

This letter was sent to all licensees of operating plants and holders of construction permits and establishes the NRC's position with respect to location and habitability requirements for the emergency operations facility and staffing levels for emergency situations. It further requires that licensees respond within 45 days and indicate whether or not the requirements will be implemented in accordance with the schedule established.

WAPWR Response

The impact of this issue on the WAPWR design is assessed in Section 3.1 (item 25).

11. Generic Letter 81-11 (February 20, 1981): NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking" (Corrections)

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

12. Generic Letter 81-12 (February 20, 1981): Fire Protection Rule

Discussion

This letter was sent to all power reactor licensees with plants licensed prior to January 1, 1979. It specifies the information the NRC requires from each plant to demonstrate compliance with certain portions of 10CFR 50.48, "Fire Protection," and 10CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979."

WAPWR Response

The impact of 10CFR 50.48 and Appendix R to 10CFR Part 50 on the WAPWR design is addressed in Section 6.1.1 (item 2).

13. Generic Letter 81-13 (May 28, 1981): SER for Correlation for 8 x 8 R Fuel Reload Application Per the Appendix D Submittals of the General Electric Topical Report, NEDE-24011-P-A, dated February 28, 1979 and December 14, 1979

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

14. Generic Letter 81-14 (February 10, 1981): Seismic Qualification of Auxiliary Feedwater Systems

Discussion

This letter was sent to all operating pressurized water reactor licensees and requests specific information to determine the extent to which auxiliary feedwater systems in operating plants are seismically qualified.

WAPWR Response

Current Westinghouse plant designs include auxiliary feedwater systems that are seismic Category I and, as such, meet the requirements set forth in this letter. The WAPWR design will include the secondary side safeguards system discussed in Section 3.1 (item 2). Appropriate portions of this system will be seismic Category I.

15. Generic Letter 81-15 (March 10, 1981): Environmental Qualification of Class IE Electrical Equipment; Clarification of Staff's Handling of Proprietary Information

Discussion

This letter was sent to all licensees of operating plants, applicants for operating licenses, and holders of construction permits and merely states that "summary type" information alone is not adequate to establish environmental qualification of equipment.

WAPWR Response

This letter is administrative in nature and has no impact on the WAPWR design.

16. Generic Letter 81-16 (July 1, 1981): Steam Generator Overfill

Discussion

This letter was sent to all licensees of operating plants and holders of construction permits and enclosed a copy of a report entitled, "AEOD Observations and Recommendations Concerning the Problem of Steam Generator Overfill and Combined Primary and Secondary Side Blowdown."

WAPWR Response

Refer to the discussion of Generic Letter 81-28 (item 29 below) for a complete assessment of the impact of this issue on the WAPWR design.

17. Generic Letter 81-17 (March 5, 1981): Functional Criteria for Emergency Response Facilities

Discussion

This letter was sent to licensees of operating plants and holders of construction permits and enclosed a copy of NUREG-0696, "Functional Criteria for Emergency Response Facilities." This NUREG offers a method acceptable to the NRC for meeting the requirements of the regulations for emergency response facilities, and the NRC intends to use the criteria therein to evaluate conceptual design submittals for adequacy.

WAPWR Response

The issue of emergency response facilities is discussed in Section 3.1 (item 25) in relation to the WAPWR design.

18. Generic Letter 81-18 (March 30, 1981): BWR Scram Discharge System; Clarification of Diverse Instrumentation Requirement

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

19. Generic Letter 81-19 (April 20, 1981): Thermal Shock to Reactor Pressure Vessels

Discussion

This letter was sent to all licensees of operating pressurized water nuclear power plants. It summarized a meeting held March 31, 1981 between the NRC and the PWR Owners Group to discuss the effects of potential thermal shock to reactor pressure vessels by overcooling transients and the potential consequences of subsequent repressurization at relatively low temperature.

WAPWR Response

Since issuance of this letter this issue has been identified as Unresolved Safety Issue A-49, "Pressurized Thermal Shock" (refer to Section 4.0, item 27).

20. Generic Letter 81-20 (April 10, 1981): Safety Concerns Associated With Pipe Breaks in the BWR Scram System

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

21. Generic Letter 81-21 (May 5, 1981): Natural Circulation Cooldown

Discussion

This letter was sent to all licensees of operating pressurized water nuclear power reactors and applicants for operating licenses (except for St. Lucie, Unit No. 1) and summarizes an event at St. Lucie, Unit No. 1 during which the plant was forced to cooldown on natural circulation as a result of a component cooling water malfunction. During the cooldown process, abnormally rapid increases in pressurizer level were observed. Subsequent analyses have confirmed that these abnormal level increases were produced by flashing of liquid in the upper head of the reactor vessel forcing water out of the vessel and into the pressurizer. Subsequently, the NRC identified two areas of concern:

- o The unacceptability of vessel voiding during anticipated cooldown conditions (e.g., natural circulation due to loss of offsite power, loss of pumps, etc.).

- o Failure of the operator to have prior knowledge and training for this event.

The NRC required each licensee to review current plant operations and implement procedures and training to avoid (if possible), recognize, and properly react to reactor vessel voiding during natural circulation cooldown.

WAPWR Response

An evaluation of natural circulation cooldown will be performed as part of the design and licensing process for the WAPWR. In addition, appropriate natural circulation emergency response guidelines will be developed for the WAPWR design.

22. Generic Letter 81-22 (May 5, 1981): Engineering Evaluation of the H. B. Robinson Reactor Coolant System Leak on January 29, 1981

Discussion

This letter was sent to all licensees of operating plants and holders of construction permits and identified concerns relative to the H. B. Robinson event in which approximately 6000 gallons of reactor coolant water were lost from two separate leaks in the letdown train of the chemical and volume control system (CVCS). The following areas are currently under NRC consideration for further action:

- o Whether a requirement should be placed upon operating plants to establish a procedure for identification and recovery from a spurious safety injection actuation (if such a procedure is not already in place).

- o Whether criteria for terminating safety injection should include provisions for isolating charging since charging flow would be considered high pressure safety injection for very small breaks.

- o Whether there is a need for a direct reactor trip on a safety injection actuation at other Westinghouse plants which do not have a direct trip.

- o Whether operation of the isolation valves in the CVCS at H. B. Robinson is causing the system to be operated in a manner which is contrary to its design bases.

WAPWR Response

The impact of this event on the WAPWR design is encompassed by the response given to Uncategorized Issue 8, "Inadvertent Actuation of Safety Injection in PWRs," in Section 5.5 (item 8).

23. Generic Letter 81-23 (June 4, 1981): Institute of Nuclear Power Operations (INPO) Evaluation Reports

Discussion

This letter was sent to all plants with an operating license or a construction permit and merely requested that the ACRS and NRC staff management be placed on distribution for INPO plant specific evaluation reports.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

24. Generic Letter 81-23A (July 6, 1981): INPO Evaluation Reports

Discussion

This letter was sent to all licensees of operating plants and holders of construction permits and clarifies administrative details associated with distribution of INPO evaluation reports (as initially discussed in Generic Letter 81-23, item 23 above).

WAPWR Response

This letter is administrative in nature and has impact on Westinghouse in relation to the WAPWR design.

25. Generic Letter 81-24 (June 15, 1981): Multi-Plant Issue B-56, Control Rods Fail to Fully Insert

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

26. Generic Letter 81-25 (June 15, 1981): Change in Implementing Schedule for Submission and Evaluation of Upgraded Emergency Plans

Discussion

This letter was sent to holders of construction permits and changed the implementation schedule to require upgraded emergency plans (TMI Action Plan Item III.A.2, "Emergency Preparedness," in Enclosure 2 of NUREG-0737) from prior to "fuel load" to prior to "full power" authorization.

WAPWR Response

This letter is administrative in nature and has no impact on the WAPWR design.

27. Generic Letter 81-26 (July 14, 1981): Licensing Requirements for Pending Construction Permit and Manufacturing License Application

Discussion

This letter was sent to all applicants with pending construction permits and manufacturing license applications and establishes the NRC position that all applicants for a construction permit or manufacturing license must demonstrate conformance with the proposed amendment to 10CFR Part 50 entitled, "Licensing Requirements for Pending Construction Permit and Manufacturing License Applications." This proposed regulation lists TMI-related actions that the NRC believes are needed for pending construction permit and manufacturing license applications.

WAPWR Response

Subsequent to the issuance of this letter, the NRC has published a new rule on this subject which is discussed in detail in Section 3.1.

28. Generic Letter 81-27 (July 9, 1981): Privacy and Proprietary Material in Emergency Plans

Discussion

This letter was sent to all licensees of operating plants, applicants for operating licenses, and holders of construction permits and indicates that submittals to the NRC dealing with the licensees and applicants radiological emergency plans and implementing procedures be carefully screened prior to submittal so as to clearly identify as proprietary utility telephone numbers which are essential in the event of an emergency so as to avoid the loading of these phones with non-essential calls if indeed there was an emergency.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

29. Generic Letter 81-28 (July 31, 1981): Steam Generator Overfill

Discussion

This letter was sent to all licensees of operating plants and holders of construction permits and endorsed a copy of a report entitled, "AEOD Observations and Recommendations Concerning the Problem of Steam Generator Overfill and Combined Primary and Secondary Side Blowdown." This letter is an exact copy of Generic Letter 81-16 (item 16 above) and the reason for reissuance is not known.

The enclosed report expresses the following concerns raised by the NRC Office for Analysis and Evaluation of Operational Data (AEOD) as a result of several steam generator overfill events at Babcock & Wilcox plants, including one which apparently resulted in some water in the steam lines:

- o The increased dead weight and potential seismic loads placed on the main steam line and its supports should this line become flooded.

- o The loads placed on main steam lines due to the potential for rapid collapse of steam voids resulting in water hammer.

- o The potential for secondary safety valves sticking open following discharge of water or two-phase flow.

- o The potential inoperability of the main steam isolation valves, main turbine stop and bypass valves, and atmospheric dump valves due to effects of water or two-phase flow.

Because of "the lack of safety-grade equipment to either prevent or mitigate steam generator overfill and the potential seriousness of the consequential event" the AEOD has recommended that the event be considered as an Unresolved Safety Issue. The AEOD has also made recommendations concerning operator training for steam generator overfill events.

In recent near-term operating license plants (e.g., SNUPPs), the current feedwater isolation provisions have been upgraded to full safety-grade. For plants such as SNUPPS with four narrow range steam generator level channels per steam generator, this was accomplished by a feedwater isolation signal logic change from "two-out-of-three hi-hi" to "two-out-of-four hi-hi".

WAPWR Response

The WAPWR design will have four narrow range steam generator level channels per steam generator, and the feedwater isolation signal logic will be "two-out-of-four hi-hi". Westinghouse will continue to follow any regulatory activities relative to this issue, and address any resulting requirements during the licensing process for the WAPWR design.

30. Generic Letter 81-29 (August 7, 1981): Simulator Examinations

Discussion

This letter was sent to all licensees of operating plants, applicants for operating licenses, and holders of construction permits and provided guidance in addition to NUREG-0737, "Clarification of TMI Action Plan Requirements," with respect to reactor operator simulator examinations.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

31. Generic Letter 81-30 (July 31, 1981): Safety Concerns Associated with Pipe Breaks in the BWR Scram System

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

32. Generic Letter 81-31

(Not Issued)

33. Generic Letter 81-32 (August 7, 1981): NUREG-0737, Item II.K.3.44 - Evaluation of Anticipated Transients Combined with Single Failure

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

34. Generic Letter 81-33

(Not Issued)

35. Generic Letter 81-34 (August 31, 1981): Safety Concerns Associated with Pipe Breaks in the BWR Scram System

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

36. Generic Letter 81-35 (August 31, 1981): Safety Concerns Associated with Pipe Breaks in BWR Scram System

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

37. Generic Letter 81-36 (September 29, 1981): Revised Schedule for Completion of TMI Action Plan Item II.D.1, Relief and Safety Valve Testing

Discussion

This letter was sent to all licensees of operating plants, applicants for operating licenses, and holders of construction permits and revised the schedule for completion of TMI Item II.D.1 related to testing programs for relief and safety valves.

WAPWR Response

This letter is administrative in nature and, as such, has no impact on the WAPWR design. However, an assessment of the impact of post-TMI Item II.D.1 on the WAPWR design is discussed in Section 3.1 (item 15).

38. Generic Letter 81-37

(Not Issued)

39. Generic Letter 81-38 (November 10, 1981): Storage of Low-Level Radioactive Wastes at Power Reactor Sites

Discussion

This letter was sent to all holders of and applicants for operating licenses and construction permits, and sets forth radiological safety guidance for onsite contingency storage capacity. It essentially states that a licensee may increase storage capacity without prior NRC approval per the provisions of 10CFR Part 50.59, "Changes, Tests and Experiments." In order to do this, the licensee must ensure:

- o There are no license conditions or technical specifications which prohibit increased storage.
- o No unreviewed safety question exists.
- o The proposed increased storage capacity does not exceed the generated waste projected for 5 years.

The guidelines given in this letter are essentially the same as those given in Appendix 11.4-A to Standard Review Plan 11.4, "Solid Waste Management Systems."

WAPWR Response

This letter establishes interim guidelines for additional storage of low-level waste, is administrative in nature, and has no impact on Westinghouse in relation to the WAPWR design.

40. Generic Letter 81-39 (November 30, 1981): NRC Volume Reduction Policy

Discussion

This letter was sent to all power reactor licensees, applicants for operating licenses, and holders of construction permits and reiterates a NRC policy statement on low-level radioactive waste volume reduction which stated: (A) the need for a volume reduction policy and (B) the need for waste generators to minimize the quantity of waste produced. It further references NUREG/CR-2206, "Volume Reduction Techniques in Low-Level Radioactive Waste Managements" as a detailed compilation of volume reduction techniques for wastes generated in fuel cycle and non-fuel cycle facilities.

WAPWR Response

This letter is concerned with the disposition of low-level waste (i.e., not nuclear fuel, per se) which is the responsibility of each utility utilizing the WAPWR design. However, it should be noted that one core design feature of the WAPWR is the ability to operate without the use of burnable poison rods in reload cores. This feature will eliminate the need to store and dispose of such rods. A second feature is the achievement of high peak assembly average discharge burnups. Both of these design objectives address the NRC's desire that "waste generators" minimize the quantity of waste produced.

41. Generic Letter 81-40 (December 16, 1981): Qualification of Reactor Operators - License Examinations

Discussion

This letter was sent to all power reactor licensees, applicants for operating licenses, NSSS vendors, reactor vendors, and architect engineers and revises some of the criteria to be used by the NRC staff in evaluating operator training and licensing.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

42. Generic Letter 82-01 (January 12, 1982): New Application Survey

Discussion

This letter was sent to all utilities, NSSS and fuel vendors, and architect engineers and requests a forecast of projected significant licensing submittals to assist the NRC in their manpower planning.

WAPWR Response

This letter is administrative in nature and has no impact on the WAPWR design.

43. Generic Letter 82-02 (February 8, 1982): Nuclear Power Plant Staff Working Hours

Discussion

This letter was sent to all licensees of operating plants, applicants for an operating license, and holders of construction permits and establishes maximum working hours for operating personnel at nuclear reactors.

WAPWR Response

This letter is administrative in nature a.d has no impact on Westinghouse in relation to the WAPWR design.

44. Generic Letter 82-03 (March 31, 1982): High Burnup MAPLHGR Limits

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

45. Generic Letter 82-04 (no date): Use of INPO See-In Program

Discussion

This letter was sent to all licensees of operating plants, applicants for operating licenses, and holders of construction permits and endorses the INPO See-In Program as an adequate mechanism for central collection and screening of all events from both U.S. and foreign nuclear plants.

WAPWR Response

This letter is administrative in nature and has no impact on the WAWPR design.

46. Generic Letter 82-05 (March 17, 1982): Post-TMI Requirements

Discussion

This letter was sent to all licensees of operating power reactors, and reiterates the schedular requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements." In addition, it requests that each licensee provide specific information regarding the status of certain items still to be implemented.

WAPWR Response

This letter is administrative in nature and has no impact on the WAPWR design. However, the impact of post-TMI issues on the WAPWR design is addressed in detail in Section 3.0.

47. Generic Letter 82-06

(Not Issued)

48. Generic Letter 82-07 (April 15, 1982): Transmittal of NUREG-0909 Relative to Ginna Tube Rupture

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

49. Generic Letter 82-08 (April 15, 1982): Transmittal of NUREG-0909 Relative to Ginna Tube Rupture

Discussion

This letter was sent to all pressurized water reactor plant licensees and applicants and transmitted NUREG-0909, "NRC Report on the January 25, 1982 Steam Generator Tube Rupture at R. E. Ginna Nuclear Power Plant," which documents the factual information relative to what transpired during the Ginna tube rupture incident.

WAPWR Response

The secondary side safeguards system will be designed to provide for safety-grade secondary cooldown capability to mitigate the consequences of such an event. Therefore, the WAPWR design will adequately address this issue.

50. Generic Letter 82-09 (April 20, 1982): Environmental Qualification of Safety-Related Electrical Equipment

Discussion

This letter was sent to all power reactor licensees, applicants for an operating license, NSSS vendors, and reactor vendors and provided clarifying information relative to the draft regulatory requirements for environmental qualification of safety-related electrical equipment given in draft Regulatory Guide 1.89 and the proposed regulations of 10CFR 50.49.

WAPWR Response

The issue of environmental qualification of safety-related electrical equipment (including Regulatory Guide 1.89 and 10CFR 50.49) is addressed in Section 4.0 (item 14).

51. Generic Letter 82-10 (May 5, 1982): Post-TMI Requirements

Discussion

This letter was sent to all licensees of operating reactors and provided an updated schedule for implementation of the requirements given in NUREG-0737, "Clarification of TMI Action Plan Requirements."

WAPWR Response

This letter is administrative in nature and has no impact on the WAPWR design.

52. Generic Letter 82-11 (June 9, 1982): Transmittal of NUREG-0916 Relative to the Restart of R. E. Ginna Nuclear Power Plant

Discussion

This letter was sent to all light water reactor plant licensees and applicants, and encloses a copy of NUREG-0916 which documents the NRC's Safety Evaluation report based on findings relative to the cause of the R. E. Ginna tube rupture event, and the repairs and modifications made to the plant.

WAPWR Response

See Item No. 49 in this section for the WAPWR response.

53. Generic Letter 82-12 (June 15, 1982): Nuclear Power Plant Staff Working Hours

Discussion

This letter was sent to all licensees of operating plants, applicants for an operating license, and holders of construction permits and encloses a copy of revised pages of NUREG-0737 that incorporates the Commission Policy on working hours for operating personnel at nuclear reactors.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

54. Generic Letter 82-13 (June 17, 1982): Reactor Operator and Senior Reactor Operator Examinations

Discussion

This letter was sent to all power reactor licensees, applicants for an operating license and holders of a construction permit and documents minutes of a meeting held between the NRC and industry representatives relative to changes being considered for reactor operator examinations.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

55. Generic Letter 82-14 (August 9, 1982): Submittal of documents to the Nuclear Regulatory Commission.

Discussion

This letter was sent to all reactor licensees, holders of construction permits and applicants and summarizes the requirements concerning quantities of documents to be submitted to the NRC.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

56. Generic Letter 82-15 (Not used)

57. Generic Letter 82-16 (September 20, 1982): NUREG-0737 Technical Specifications

Discussion

This letter was sent to all pressurized power reactor licensees and summarizes NUREG-0737 Technical Specification Requirements.

WAPWR Response

The Technical Specifications for the WAPWR will be established utilizing a probabilistic risk basis in conjunction with good engineering practice and judgement. The requirements enclosed with this Generic Letter will be considered in the development of the WAPWR Technical Specifications.

58. Generic Letter 82-17 (October 1, 1982): Inconsistency Between Requirements of 10CFR 50.54(t) and Standard Technical Specifications for Performing Audits of Emergency Preparedness programs.

Discussion

This letter was sent to all licensees and applicants for operating power reactors and holders of construction permits for power reactors, and clarifies a schedular inconsistency between 10CFR 50.54(t) and the Standard Technical Specifications regarding the frequency of emergency preparedness audits.

WAPWR Response

This letter is administrative in nature and has no impact on the WAPWR design.

59. Generic Letter 82-18 (October 12, 1982): Reactor Operator and Senior Reactor Operator Requalification Examinations

Discussion

This letter was sent to all power reactor applicants and licensees, and informs them of NRC plans for conducting requalification examinations for licensed reactor operators (at least 20% of licensed personnel per year per facility). This letter also encloses a copy of SECY-82-232, "Use of Non-Plant-Specific Simulations for Initial, Replacement, and Requalification Examinations for Licensed Reactor Operators and Senior Operators," which states that operating exams are to be conducted on plant specific simulators, or in lieu of a simulator the operating exams will be conducted on the facility.

WAPWR Response

Operator training and qualification is the responsibility of each utility utilizing the WAPWR design.

60. Generic Letter 82-19 (October 5, 1982): Submittal of Copies of Documents to NRC

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

61. Generic Letter 82-20 (October 26, 1982): Guidance for Implementing Standard Review Plan Rule

Discussion

This letter was sent to all power reactor licensees/permit holders and applicants for construction permits, and enclosed a draft copy of

NUREG-0906 for comment. This document provides interim guidance for complying with 10CFR 50.34(g), "Conformance with the Standard Review Plan" until a revision to Regulatory Guide 1.70 is issued.

WAPWR Response

Subsequent to issuance of this Generic Letter, the proposed rulemaking discussed above was issued. See Section 6.1.1, Item 5 for a discusson of this new regulation.

62. Generic Letter 82-21 (October 6, 1982): Technical Specifications for Fire Protection Audits

Discussion

This letter was sent to all licensees and applicants of nuclear power reactors, and provides a fire protection audit program which the NRC finds "responsive to overall programmatic requirements contained in 10CFR 50.48(a) and guideline positions in Branch Technical Position (BTP) 9.5-1."

Section 6.1.1, Item 2 provides an overall discussion of the impact of fire protection regulations (i.e., 10CFR 50.48 and Appendix R to 10CFR 50) on the WAPWR design.

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC BTP 9.5-1 during the licensing process for the WAPWR design.

63. Generic Letter 82-22 (October 26, 1982): Congressional Request for Information Concerning Steam Generator Tube Integrity.

Discussion

This letter was sent to all pressurized power reactor licensees, and forwards a list of questions asked by the Congressional Committee on Oversight and Investigation regarding steam generator tube integrity.

WAPWR Response

The subject of steam generator tube integrity is encompassed by Unresolved Safety Issue A-3 (Section 4.0, Item 3).

64. Generic Letter 82-23 (October 30, 1982): Inconsistency Between Requirements of 10CFR 73.40(d) and Standard Technical Specifications for Performing Audits of Safeguards Contingency Plans (Security Plan).

Discussion

This letter was sent to all licensees and applicants for operating power reactors and holders of construction permits for power reactors, and clarifies a discrepancy between 10CFR 73.40(d) and the Technical Specifications concerning audit frequency.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

65. Generic Letter 82-24 (November 4, 1982): Safety Relief Valve Quencher Loads: Evaluation for BWR Mark II and III Containments

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

66. Generic Letter 82-25 (November 3, 1982): Integrated IAEA Exercise for Physical Inventory at LWRs.

Discussion

This letter was sent to all power reactor licensees and requests both BWR and PWR voluntary participants for field testing new measurement equipment and inventory verification techniques developed under IAEA sponsorship.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

67. Generic Letter 82-26 (November 12, 1982): NUREG-0744 Revision 1; Pressure Vessel Material Fracture Toughness

Discussion

This letter was sent to all power reactor licensees and encloses a copy of NUREG-0744, Revision 1 which contains an elastic-plastic fracture mechanics analytical procedure acceptable to the NRC.

WAPWR Response

See Unresolved Safety Issues A-11, "Reactor Vessel Materials Toughness" (Section 4, Item 11) and A-49, "Pressurized Thermal Shock" (Section 4, Item 27) for a discussion of this concern and the WAPWR response.

68. Generic Letter 82-27 (November 15, 1982): Transmittal of NUREG-0763, "Guidelines for Confirmatory In-Plant Tests of Safety-Relief Valve Discharges for BWR Plants," and NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments".

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

69. Generic Letter 82-28 (December 10, 1982): Westinghouse Reactor Vessel Level Instrumentation System

Discussion

This letter was sent to all Westinghouse operating plant licensees, and requests commitments to having a reactor coolant system inventory tracking system.

WAPWR Response

The WAPWR design will include a reactor vessel level instrumentation system in the overall WAPWR post-accident monitoring design. See Section 3.1, Item 22 for a further discussion.

70. Generic Letter 82-29 (Not used)

71. Generic Letter 82-30 (December 28, 1982): Filings Related to 10CFR50 Production and Utilization Facilities

Discussion

This letter was sent to all licensees and applicants for operating power reactors and holders of construction permits for power reactors, and identifies proper NRC addressees for routine correspondence.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

72. Generic Letter 82-31 (Not used)

73. Generic Letter 82-32 (December 9, 1982): Potential Steam Generator Related Generic Requirements

Discussion

This letter was sent to all pressurized water reactor plant licensees, and encloses a draft of a consultant's report entitled, "Value-Impact Analysis of Recommendations Concerning Steam Generator Tube Degradations and Rupture Events" for comment.

WAPWR Response

The subject of steam generator tube integrity is encompassed by Unresolved Safety Issue A-3 (Section 4, Item 3).

74. Generic Letter 82-33 (December 17, 1982): Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability

Discussion

This letter was sent to all licensees of operating reactors, applicants for operating licenses, and holders of construction permits and encloses supplement 1 to NUREG-0737. This supplement calls out the requirements for Emergency Response Facilities (ERFs) and religates other NUREGs and Regulatory Guides that deal with ERFs and related subjects as guidance documents. Topics in the generic letter are:

- o Safety Parameter Display System (SPDS)
- o Control Room Design Review
- o Application of Regulatory Guide 1.97 to ERFs.
- o Emergency Response Procedures, and
- o ERFs

In addition, it stresses the need to integrate elements of the facilities, properly train personnel to use the facilities, and employ state of the art Human Factor principles to develop the facilities.

Since TMI the NRC issued many documents addressing ERFs. By late 1981 the documents were being interpreted as requirements and there were priority and schedule conflicts. The NRC Committee to Review Generic Requirements found the situation to be confusing and caused the requirement content of Generic Letter 82-33 to be developed and issued.

Currently the NRC has a draft NUREG-0835 on the SPDS criteria relative to human factors in preparation.

WAPWR Response

The impact of this issue on the WAPWR design is assessed in Section 3.1 (Item 25).

75. Generic Letter 82-34 (Not used).

76. Generic Letter 82-35 (Not used).

77. Generic Letter 82-36 (Not used).

78. Generic Letter 82-37 (Not used).

79. Generic Letter 82-38 (December 22, 1982): Meeting to Discuss Recent Developments for Operating Licensing Examinations

Discussion

This letter was sent to all power reactor licensees, applicants for operating licenses, NSS vendors, reactor vendors and architect engineers; and transmits minutes of a meeting regarding NRC operator and senior operator licensing examinations.

WAPWR Response

Operator training and qualification is the responsibility of each utility utilizing the WAPWR design.

80. Generic Letter 82-39 (December 22, 1982): Problems With the Submittals of 10CFR73.21 Safeguards Information for Licensing Review.

Discussion

This letter was sent to all reactor licensees, construction permit holders and applicants for construction permits; and requests that specific procedures be followed for the transmittal of safeguards information.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

81. Generic Letter 83-01 (January 11, 1983): Operator Licensing Examination Site Visit

Discussion

This letter was sent to all power reactor licensees and applicants for an operating license; and requests input regarding schedular needs for conducting reactor operator examinations so the NRC can establish its examination site visit schedule.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

82. Generic Letter 83-02 (January 10, 1983): NUREG-0737 Technical Specifications

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

83. Generic Letter 83-03 (Not issued)

84. Generic Letter 83-04 (February 1, 1983): Regional Workshops Regarding Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability

Discussion

This letter was sent to all licensees of operating reactors, applicants for operating licenses, and holders of construction permits; and schedules workshops in the area of emergency response requirements.

WAPWR Response

The impact of this issue on the WAPWR design is assessed in Section 3.1, item 25.

85. Generic Letter 83-05 (February 8, 1983): Safety Evaluation of "Emergency Procedure Guidelines, Revision 2," NEDO-24934, June 1982.

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

86. Generic Letter 83-06 (January 31, 1983): Certificates and Revised Format for Reactor Operator and Senior Reactor Operator Licenses

Discussion

This letter was sent to all power reactor licensees, applicants for operating licenses, NSS vendors, reactor vendors and architect-engineers; and indicates that certificates for reactor operators and senior reactor operators have been reformatted and are suitable for framing.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

87. Generic Letter 83-07 (February 16, 1983): The Nuclear Waste Policy Act of 1982

Discussion

This letter was sent to all power and non-power reactor licensees, applicants for operating license and holders of construction permits; and highlights the requirement given in the Nuclear Waste Policy Act of 1982 that all facilities licensed under Sections 103 and 104 of the Atomic Energy Act of 1954 have a contract with the Secretary of Energy by June 30, 1983 for the disposal of spent nuclear fuel or high level waste.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

88. Generic Letter 83-08 (February 2, 1983): Modification of Vacuum Breakers on Mark I Containments

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

89. Generic Letter 83-09 (February 1983): Review of Combustion Engineering Owner's Group Emergency Procedures Guidelines Program

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

90. Generic Letter 83-10a (February 8, 1983): Resolution of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," (CE applicants).

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

91. Generic Letter 83-10b (February 8, 1983): Resolution of TMI Action Item, II.K.3.5, "Automatic Trip of Reactor Coolant Pumps (CE Licensees).

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

92. Generic Letter 83-10c (February 8, 1983): Resolution of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" (Westinghouse applications).

Discussion

This letter was sent to all applicants with Westinghouse designed NSSS, and establishes the NRC position regarding reactor coolant pump (RCP) operation following a transient or accident event (i.e., whether or not the RCPs should be tripped following such events) This item was originally identified as a result of post-TMI licensing requirements (Section 3.3.1, Item 4 of this document). The staff concluded that for Westinghouse designed NSSSs, the need for RCP trip following a transient or accident should be determined by the applicant on a case-by-case basis.

The Westinghouse position (which applies to the WAPWR design as well) is summarized below:

- The RCPs should be tripped if indications of a small break LOCA exist.

- The RCPs should remain operational for non-LOCA transients and accidents where their operation is beneficial to accident mitigation and recovery.

- If there is doubt as to what type of transient or accident is in progress, the RCPs should be tripped.

- RCP trip can be achieved safely and reliably by the operator when required.

WAPWR Response

Westinghouse will perform LOCA analyses for the WAPWR with analysis assumptions consistent with the above Westinghouse position. During the licensing process for the WAPWR design, evaluations will be performed addressing the guidelines given in this letter (and letter 83-10d) which will result in the establishment of appropriate RCP trip setpoints, and will demonstrate that sufficient operator action time is available for manual tripping of the pumps. There will be no automatic RCP trip system in the WAPWR design.

93. Generic Letter 83-10d (February 8, 1983): Resolution of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" (Westinghouse Licensees).

Discussion

This letter was sent to all licensees with Westinghouse designed nuclear steam supply systems and establishes (in conjunction with Generic Letter 83-10c, above) the NRC position regarding reactor coolant pump (RCP) operation following a transient or accident event. This issue is discussed completely under Item 92 above.

94. Generic Letter 83-10e (February 8, 1983): Resolution of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" (B&W Licensees).

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

95. Generic Letter 83-11 (February 8, 1983): "Licensee Qualification for Performing Safety Analysis in Support of Licensing Actions."

Discussion

This letter is sent to all operating reactor licensees, and mandates that when licensees perform their own safety analyses they must submit supporting code verification, performed by the licensee, to the NRC.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

96. Generic Letter 83-12 (February 24, 1983): Issuance of NRC Form 398-Personal Qualifications Statement - Licensee

Discussion

This letter was sent to all power and non-power reactor licensees, applicants for an operating license, holders of construction permits and NSSS vendors; and issues a new form which must be completed by all applicants for operator and senior operator licenses.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

97. Generic Letter 83-13 (March 2, 1983): Clarification of Surveillance Requirements for HEPA Filters and Charcoal Adsorber Units in Standard Technical Specification (STS) on ESF Cleanup Systems.

Discussion

This letter was sent to all applicants for operating licenses and holders of construction permits for power reactors; and clarifies the Surveillance Requirements in the STS for each unit.

WAPWR Response

The WAPWR Technical Specification, including Surveillance Requirements for the various components of the ESF atmospheric cleanup system, will be established utilizing a probabilistic risk approach in conjunction with good engineering practice and judgement. The level of conformance of such equipment to the regulatory position of Regulatory Guide 1.52 will be identified during the licensing process for these systems.

98. Generic Letter 83-14 (March 7, 1983): Definition of "Key Maintenance Personnel", (Clarification of Generic Letter 82-12)

Discussion

This letter was sent to licensees of operating reactors, applicants for operating licenses, and holders of construction permits; and clarifies the Staff's definition of "key maintenance personnel" given in an earlier generic letter (82-12).

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

99. Generic Letter 83-15 (March 23, 1983): Implementation of Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations, Revision 1".

Discussion

This letter was sent to all licensees of operating power reactors and applicants for operating licenses; and incorporates an alternative method for complying with the regulatory position in Regulatory Guide 1.150 into this regulatory guide. The alternative method was recommended by an Ad Hoc Committee of the Electric Utility Industry and is aimed at increasing the reliability in detection and characterization of service induced defects in the reactor vessel welds.

For recently licensed Westinghouse plants, the Westinghouse position with respect to Regulatory Guide 1.150, Revision 1 has been found to be acceptable.

WAPWR Response

A fundamental objective of the WAPWR is to perform design and layout of equipment, systems and structures as well as the entire plant layout to facilitate inspection, and as such this item is not expected to impact the WAPWR reactor vessel design. However, during the detailed licensing process for the WAPWR, Westinghouse will fully document the level of compliance with Regulatory Guide 1.150, Revision 1.

100. Generic Letter 83-16 (March 31, 1983): Transmittal of NUREG-0977 Relative to the ATWS Events at Salem Generating Station; Unit No. 1.

Discussion

This letter was sent to all light water plant licensees and applicants, and encloses the Summary section of NUREG-0977, "NRC Fact-Finding Task Force Report on the ATWS Events at Salem Nuclear Generating Station, Unit 1, on February 22 and 25, 1983."

WAPWR Response

The issue of Westinghouse DB-50 reactor trip breakers failure to open on an automatic trip signal is fully discussed in Unresolved Safety Issue (USI) A-9, "Anticipated Transients Without Scram" (Section 4.0, Item 9 of this document).

101. Generic Letter 83-17 (April 8, 1983): Integrity of the Requalification Examinations for Renewal of Reactor Operator and Senior Reactor Operator Licenses

Discussion

This letter highlights an incident in which a licensed operator cheated on a requalification examination.

WAPWR Response

Operator training and qualification is the responsibility of each utility using the WAPWR design.

102. Generic Letter 83-18 (April 19, 1983): NRC Staff Review of the BWR Owners' (BWROG) Control Room Survey Program

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

103. Generic Letter 83-19 (May 2, 1983): New Procedures for Providing Public Notice Concerning Issuance of Amendments to Operating Licenses

Discussion

This letter was sent to all power reactor and testing facility licensees, and highlights changes in federal regulations concerning processing of operating license amendments.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

104. Generic Letter 83-20 (May 9, 1983): Integrated Scheduling for Implementation of Plant Modifications

Discussion

This letter was sent to all operating reactor licensees and holders of construction permits, and encourages efforts by utilities to establish realistic schedules for implementation of safety improvements, both utility-initiated and NRC required, at operating reactors.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

105. Generic Letter 83-21 (May 11, 1983): Clarification of Access Control Procedure for Law Enforcement Visits

Discussion

This letter was sent to all licensees of operating nuclear power plants, applicants for operating licenses and holders of construction permits, and clarifies the intent of 10CFR Part 73.55(d)(1) regarding access control procedures as they relate to bona fide law enforcement officers.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

106. Generic Letter 83-22 (June 3, 1983): Safety Evaluation of "Emergency Response Guidelines"

Discussion

This letter was sent to all operating reactor licensees, applicants for an operating license and holders of construction permits for Westinghouse pressurized water reactors; and concludes that Westinghouse Owners' Group proposed Westinghouse Emergency Response Guidelines are acceptable for implementation.

WAPWR Response

The subject of Westinghouse Emergency Response Guidelines as they relate to the WAPWR design is fully discussed in Section 3.1, Item 7 of this document.

107. Generic Letter 83-23 (July 29, 1983): Safety Evaluation of "Emergency Procedure Guidelines"

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

108. Generic Letter 83-24 (June 29, 1983): TMI Task Action Plan Item I.G.1, "Special Low Power Testing and Training," Recommendations for BWRs.

Discussion

This letter is not applicable to Westinghouse pressurized water reactor designs.

109. Generic Letter 83-25 (Not used)

110. Generic Letter 83-26 (July 5, 1983): Clarification of Surveillance Requirements for Diesel Fuel Impurity Level Tests

Discussion

This letter was sent to all applicants for operating licenses and holders of construction permits for power reactors, and clarifies Standardized Technical Specifications (STS') surveillance requirements for diesel fuel impurity level tests. In particular, it clarifies inconsistencies between Regulatory Guide 1.137, Revision 1, ANSI N195, SRPs 9.5.4 and 9.5.8, and the STS'.

WAPWR Response

The WAPWR Technical Specifications, including surveillance requirements for the various diesel generator support systems, will be established utilizing a probabilistic risk approach in conjunction with good engineering practice and judgement. During the licensing process for the WAPWR, Westinghouse will fully document the level of compliance with the acceptance criteria of SRPs 9.5.4 and 9.5.8.

111. Generic Letter 83-27 (July 6, 1983): Surveillance Intervals in Standard Technical Specifications.

Discussion

This letter was sent to all licensees and applicants for operating power reactors and holders of construction permits, and clarifies ambiguities in the Technical Specification surveillance intervals.

WAPWR Response

This letter is administrative in nature and has no impact on Westinghouse in relation to the WAPWR design.

112. Generic Letter 83-28 (July 8, 1983): Required Actions Based on Generic Implications of Salem ATWS Events

Discussion

This letter was sent to all licensees of operating reactors, applicants for an operating license and holders of construction permits; and discusses generic actions required as a result of the Salem ATWS event. As a result of the Salem ATWS, the NRC is requiring timely reporting by addressees who have applied for operating licenses as to the programs, procedures, maintenance, testing, equipment improvements, and training relative to the reactor trip systems. The NRC focuses on (a) post-trip reviews, (b) equipment classification and vendor interfaces, (c) post-maintenance testing, and (d) reactor-trip-system-reliability improvements.

This letter identifies the items of the highest priority and requires that for these, planned changes be integrated into existing plant schedules first.

The letter is for information only for those addressees who have not yet applied for operating licenses for their plants.

WAPWR Response

The issue of Westinghouse DB-50 reactor trip breakers failure to open an automatic trip signal is fully discussed in Unresolved Safety Issue (USI) A-9, "Anticipated Transients Without Scram," (Section 4.0, Item 9 of this document).

113. Generic Letter 83-29 (not used).

114. Generic Letter 83-30 (July 21, 1983): Deletion of Standard Technical Specification Surveillance Requirement 4.8.1.1.2.d.6 for Diesel Generator Testing

Discussion

This letter was sent to all holders of operating licenses, applicants for operating licenses and holders of construction permits for power reactors, and clarifies an inconsistency between the subject Standardized Technical Specifications surveillance requirements and GDC 17, Regulatory Guide 1.108, and SRPs 8.2 and 8.3.1 concerning diesel generator testing.

WAPWR Response

The WAPWR Technical Specifications, including surveillance requirements for diesel generators, will be established utilizing a probabilistic risk approach in conjunction with good engineering practice and judgement. During the licensing process for the WAPWR, Westinghouse will fully document the level of compliance with GDC 17, Regulatory Guide 1.108, and SRPs 8.2 and 8.3.1 with due consideration of the inconsistency noted in this letter.

115. Generic Letter 83-31 (September 19, 1983): Safety Evaluation of "Abnormal Transient Operating Guidelines."

Discussion

This letter was sent to all operating reactor licensees, applicants for an operating license and holders of construction permits for Babcock & Wilcox pressurized water reactors; and concludes that the "Abnormal Transient Operating Guidelines" submitted by the Babcock & Wilcox Owners Group are acceptable for Oconee Unit 3.

WAPWR Response

This letter is not applicable to Westinghouse pressurized water reactor designs.

116. Generic Letter 83-32: (Not issued)

117. Generic Letter 83-33 (October 19, 1983): NRC Positions on Certain Requirements of Appendix R to 10CFR50.

Discussion

This letter was sent to all licensees and applicants of nuclear power reactors, and encloses the guidance NRC fire protection inspection teams use in conducting inspections for evaluating conformance to 10CFR50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979."

WAPWR Response

See Section 6.1.1, item 2 for a discussion of the WAPWR design features with respect to fire protection.

118. Generic Letter 83-34: (Not issued)

119. Generic Letter 83-35 (November 2, 1983): Clarification of TMI Action Plan Item II.K.3.31

Discussion

This letter was sent to all licensees of operating reactors, applicants for operating licenses and holders of construction permits; and states that although item II.K.3.30 requires that each licensee revise its current ECCS SBLOCA model, it is acceptable to continue to use a previously approved model provided the revised model does not demonstrate that the previously approved model is nonconservative.

WAPWR Response

Westinghouse emergency core cooling performance analyses for the WAPWR design (in accordance with 10CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors") will be performed using models approved by the NRC in accordance with Appendix K, "ECCS Evaluation Models," to 10CFR Part 50.

120. Generic Letter 83-36 (November 1, 1983): NUREG-0737 Technical Specifications

Discussion

This letter was sent to all boiling water reactor licensees, and as such is not applicable to Westinghouse pressurized water reactor designs.

121. Generic Letter 83-37 (November 1, 1983): NUREG-0737 Technical Specifications

Discussion

This letter was sent to all pressurized water reactor licensees, and updates Generic Letter 82-16 (item 59 of this section) regarding implementation of certain post-TMI Technical Specification requirements.

WAPWR Response

The Technical Specification for the WAPWR will be established utilizing a probabilistic risk basis in conjunction with good engineering practice and judgement. The requirements enclosed with this Generic Letter will be considered in the development of the WAPWR Technical Specifications.

122. Generic Letter 83-38 (October 31, 1983): NUREG 0965, "NRC Inventory of Dams"

Discussion

This letter is not applicable to WAPWR nuclear power block design.

6.5 IE BULLETINS AND LICENSING ISSUES

6.5.1 IE BULLETINS

The NRC Office of Inspection and Enforcement monitor nuclear plant compliance during construction and operation by reviewing various mandatory reports, and by site inspections. Their enforcement actions must be based on established documented requirements. In addition to citations and punitive actions they issue Bulletins.

The subject of a Bulletin must be generic or potentially generic for a set of plants. A Bulletin can mandate immediate corrective action; however, most Bulletins mandate that the utility respond with a description of their status and intention on the subject.

The following IE Bulletins include only those with subjects of potential impact to the design or design interface with the Westinghouse Advanced Pressurized Water Reactor.

1. IE Bulletin 79-01, 01A, 01B, Environmental Qualification of Class 1E Equipment

   Discussion

   Licensees are required to identify and document all Class 1E electrical equipment required to function under accident conditions and submit written evidence of its ability to function. In addition they must submit a master list of all installed safety-related electrical equipment and report on documentation that shows qualification for the adverse environment. Also, they must pursue programs to establish qualification or replacement of unqualified equipment.

   WAPWR Response

   This issue and its impact on the WAPWR design is fully discussed in Section 4.0, item 14.

2. IE Bulletin 79-02, Revisions 1 and 2, Pipe-Support Base-Plate Designs Using Concrete-Expansion Anchor Bolts

Discussion

Structural failures of anchor bolts systems at the support plate interface with masonry walls resulted in an NRC mandate to:

1) Recalculate individual base-plate and bolt design loads.

2) Determine the load capabilities of concrete anchors and the factor of safety.

3) Test anchor bolt capabilities documentation if documentation does not verify an adequate safety margin.

4) Report on the verification and discrepancies.

5) Provide schedules for redesign and corrections.

WAPWR Response

See item 10 of this section for a sequel Bulletin on anchor bolts.

3. IE Bulletin 79-05, 05A, 05B, 05C, Nuclear Incident at Three Mile Island

Discussion

The accident at Three Mile Island Nuclear Power Plant, Unit 2 (TMI-2) resulted in major core damage with minor radioactive releases to the environment. Improper valve positioning prevented early auxiliary-feedwater system function. In addition, a power operated relief valve failed to close. The nature of the pressurizer instrumentation caused the operators to believe that the pressurizer water level was high adding to the maloperation and continuation of the core damage sequences.

WAPWR Response

See Section 3.0 for a complete discussion of significant post-TMI requirements and their impact on the WAPWR design.

4. IE Bulletin 79-17, Revision 1, Pipe Cracks in Stagnant Borated Water Systems at PWR Plants

Discussion

Stress corrosion cracking in austenitic piping material of PWR secondary systems resulted in small leaks that were found during routine inspections. Nondestructive examinations of selected safety related stainless steel piping systems were made to determine the generic extent of the problem. The problem did not appear to be widespread. The results emphasized the need for continued control of secondary system chemistry and the need for secondary system inspection criteria to ensure that cracking not go undetected.

WAPWR Response

The WAPWR design will utilize state-of-the-art secondary system chemistry control. In addition, appropriate inspection criteria will be established as part of the overall design process.

5. IE Bulletin 79-21, Temperature Effects on Level Instrumentation

Discussion

Increased containment temperature could cause the steam generator water level indication to be higher than the actual water level. This bulletin required that the transmitters that provide the level signal are to be qualified for the post accident (temperature) environment or moved to a location that does not experience an adverse environment.

WAPWR Response

Temperature feedback from the reference legs will be supplied to the Integrated Protection System to calculate a compensated steam generator level.

6. IE Bulletin 79-24, Frozen Lines

Discussion

Freezing of a high pressure coolant injection system recirculation lines was caused by inadequate insulation. The NRC identified other water systems that malfunctioned due to freezing and requested that protective measures be taken to ensure that safety-related process, instrument and sampling systems do not freeze during cold weather.

WAPWR Response

In the WAPWR design, the RWST has been eliminated. A large EWST inside containment is the water source for the ISS system and for refueling. All of the WAPWR ISS system, including the recirculation lines, are inside and protected against adverse weather.

7. IE Bulletin 79-27, Loss of Non-Class-1-E Instrumentation and Control Power System Bus During Operation

Discussion

Loss of power to the non-nuclear instrumentation system rendered control room indicators and recorders for the reactor coolant system and most of the secondary system inoperable. Utilities were required to review the Class 1E and non-Class 1E buses supplying safety and non-safety instrumentation and control systems that could affect the cold shutdown ability.

Utilities were also required to describe and submit schedules for modifications and prepare emergency procedures for loss of instrument power.

WAPWR Response

The power distribution system for the WAPWR will be designed such that the loss of an instrument bus will not result in the loss of secondary systems and instrumentation necessary for the safe shutdown of the plant.

8. IE Bulletin 80-04, Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition

Discussion

A deficiency was identified in original analysis of steam line break for several plants. It was determined that continued delivery of auxiliary feedwater at run-out conditions would cause the containment design pressure to be exceeded. In addition to needed curtailment of auxiliary feedwater it was determined that main feedwater curtailment for the post accident phase of steam/feed break had not been considered in some cases. Utilities were requested to review and report on containment pressure response including the impact of feedwater runout flow. Also, they were required to evaluate the operability of pumps after extended runout operation. Provide proposed corrective actions and schedules.

WAPWR Response

The WAPWR design will address this issue through improved design of the main and auxiliary feedwater systems. If additional SG inventory due to auxiliary feed or main feed design can be postulated, then the containment ME release analysis will reflect this.

9. IE Bulletin 80-06, Engineered Safety Features (ESF) Reset Controls

Discussion

Plants had been designed for the use of the Safety Injection Reset actuation that also reset many supporting functions throughout the plant. This created changes in various subsystems state of operation that might not be desirable or within the operators awareness. Utilities were to (1) review circuitry for undesirable change of state actuations, (2) provide a schedule for testing the systems for reset actuations, and (3) describe planned modifications and implementation schedule. Any reset actuations that did not retain systems or equipment in the expected emergency mode were not to change state but to be redesigned for individual operator action and hence operator awareness.

WAPWR Response

The WAPWR design precludes automatic realignment of systems affected by Safety Injection for those cases where realignment could be adverse to accident mitigation. Following Safety Injection Reset these systems will remain in the condition required for safety injection until manually repositioned by the operator. Automatic realignment of systems following safety injection reset will be allowed only where it has been demonstrated that it is necessary or beneficial to continued accident mitigation.

10. IE Bulletin 80-11, Masonry Wall Design

Discussion

Several cases of inadequate anchor supports in masonry walls and inadequate structural strength of the walls were found. The NRC issued requirements for identifying those masonry walls that support safety-related piping or equipment to (1) identify those masonry walls that are in the proximity of safety-related equipment, and (2) provide a re-evaluation of design adequacy of the anchors and masonry walls.

WAPWR Response

The concerns discussed in IE Bulletin 80-11 will be considered in the WAPWR plant design. The pipe support base plate flexibility will be accounted for in the calculation of anchor bolt loads or they will be considered as rigid with supporting justification.

Addressment of this concern will be completely documented during the licensing process for the WAPWR design.

11. IE Bulletin 80-18, Maintenance of Adequate Minimum Flow Thru Centrifugal Charging Pumps Following Secondary Side High Energy Line Rupture

Discussion

Under certain conditions the centrifugal charging pumps (CCPs) could be damaged due to lack of minimum flow before safety injection (SI) termination criteria are met. The particular circumstances that could result in damage vary somewhat from plant to plant, but involve unavailability of the pressurizer power operated relief valves (PORVs), with operation of one or more CCPs repressurizing the reactor during SI following a secondary system high energy line break. Since the SI signal automatically isolates the CCP mini-flow return line, the flow through the CCPs is determined by the individual pump characteristic head vs. flow curve the pressurizer safety valve setpoint, and the flow resistances and pressure losses in the piping and in the reactor core. Flow at or near shut-off head may not be adequate to insure pump cooling, and resulting pump damage could violate design criteria before SI termination criteria are met.

The NRC required utilities to calculate the design capability for minimum charging pump flow and implement as necessary the modifications to equipment or procedures to ensure flow.

WAPWR Response

For the WAPWR design, the scenario described above does not apply since the charging pumps are not used as safety injection pumps. Also, the safety injection miniflow valves are not automatically closed on a safety injection signal. Thus, there are individual miniflow paths provided for each safety injection pump which provide continuous miniflow for safety injection operation.

12. IE Bulletin 80-24, Prevention of Water Damage Due to Water Leakage Inside Containment

Discussion

A flooded condition within containment resulted from a combination of 1) fan cooler service water leaks, 2) inoperable containment sump pumps, 3) lack of attention to sump level lights, 4) lack of sump level range and alarm instrumentation, 5) limited range and calibration error in atmosphere moisture measuring system, 6) hold-up tanks not uniquely dedicated to containment sump discharge, 7) local water level indicators in the fan cooler basin not calibrated, 8) no water level indicators in the reactor vessel cavity pit and, 9) the pit sump discharged to the containment floor.

The NRC required utilities to describe 1) all "open" cooling water services in containment, 2) history, and 3) isolation testing capabilities. Also, they were to implement means for the detection of water accumulation and a positive means for flow indication from the containment sumps.

WAPWR Response

WAPWR design will address IE Bulletin 80-24 in regard to detection of water accumulation in the containment. The WAPWR design with the EWST addresses major leaks (see Section 4.0, item 21).

13. IE Bulletin 81-01, Supplement 1, Failure of Gate Type Valves to Close Against Differential Pressure

Discussion

In response to a TMI action item, EPRI set up test conditions for typical power operated relief valves (PORV). Since a block valve is installed upstream of the PORVs, EPRI sponsored 2400 psi steam tests at the Marshall facility that also included some typical and available block valves. Some of the block valves failed to completely close against the dynamic flow and differential pressure conditions. In addition Westinghouse analyzed other motor operated valves for operation and closure against the new intended accident flow dynamics. It was identified that the original design standards and methodology needed to be upgraded.

Most corrections consisted of upgrading the motor operated drive train materials or torque.

The NRC issued requirements for reporting on the use or intended use of the identified valve types in safety-related systems, and for providing corrective action and schedule for implementation as well as evaluations of interim failure consequences.

WAPWR Response

See Section 3.1, item 15 for further discussion of this issue and its impact on the WAPWR design.

14. IE Bulletin 83-01, Failure of Reactor Trip Breakers to Open on Automatic Trip Signal

Discussion

The breakers in the reactor protection system failed to open automatically upon receipt of a valid trip signal. The reactor was tripped manually via shunt relays on the breakers. The failure was attributed to sticking of the undervoltage trip attachment. Similar failures that involved only one of the two series breakers had been reported.

The NRC mandated that 1) surveillance testing of the undervoltage trip function be performed, 2) review and implement a maintenance program for lubrication and testing of the trip mechanis

WAPWR Response

See Section 4.0, item 9; Section 6.1.2.3, item 6; and Section 6.4, item 112 for a complete discussion of this issue and its impact on the WAPWR design.

6.5.2 LICENSING ISSUES

As part of the ongoing NRC reviews of licensing submittals, various generic NRC concerns have been identified by the NRC and discussed in specific licensing applications. This section addresses the NRC concerns that have surfaced during various recent plant reviews and that are not covered elsewhere in this report.

1. Qualification of Control Systems

   <u>Discussion</u>

   Operating reactor licensees were informed by IE Information Notice No. 79-22, dated September 19, 1979, that certain non-safety-grade or control equipment, if subjected to the adverse environment of a high energy line break, could impact the safety analyses and the adequacy of the protection functions performed by the safety-grade equipment.

   IE Information Notice No. 79-22 described four control/protection interaction scenarios reported by Westinghouse, that could lead to consequences worse than those reported in the safety analysis reports. All four scenarios resulted from postulated control system failures caused by a high energy line break environment. The control systems affected were the main feedwater, pressurizer pressure, steam generator pressure, and rod control system.

   The NRC requested that all operating reactor licensees conduct a review to determine if similar problems would exist at each of the operating facilities. The review was to include, but not be limited to, the four scenarios identified in IE Information Notice No. 79-22. Thus, licensees were requested to perform a review to determine what, if any, design changes or operator actions would be necessary to assure that high energy line breaks will not cause control system failures to complicate the event beyond the safety analysis.

WAPWR Response

For the WAPWR, the four control/protection interaction scenarios identified in IE Information Notice No. 79-22 will be addressed by either qualifying the control systems in question to a high energy line break environment or performing a safety analysis of the postulated scenarios to demonstrate that the safety criteria are met.

Addressment of this concern will be completely documented during the licensing process for the WAPWR design.

A review of the overall WAPWR plant design for any additional control/protection interactions resulting from a high energy line break environment will be done as part of the overall systems interactions study as discussed in Section 4.0 (item 13).

2. Fracture Prevention of Containment Pressure Boundary

Discussion

General Design Criterion 51, "Fracture Prevention of Containment Pressure Boundary," of Appendix A to 10CFR Part 50, requires that under operating, maintenance, testing, and postulated accident conditions: (A) the ferritic materials of the containment pressure boundary behave in a nonbrittle manner and (B) the probability of rapidly propagating fracture is minimized.

The ferritic materials of the containment pressure boundary which are assessed by the NRC are those of components such as the free-standing containment vessel, equipment hatches, personnel airlocks, primary containment drywell heads, containment penetration sleeves, process pipes, end closure caps and flued heads, and penetrating piping systems downstream of penetration process pipes extending to and including the system isolation valves.

The acceptability of these materials within the context of General Design Criterion 51 is determined in accordance with the fracture toughness criteria identified for Class 2 materials in ASME Code, Section III.

WAPWR Response

The materials used for all the WAPWR containment pressure boundary components will be chosen to meet the criteria in the latest Addenda to the ASME Code, Section III.

3. Structures, Systems, and Components to be Protected from Externally Generated Missiles

Discussion

Safety-related structures, systems, and components are required to perform their functions for attaining and maintaining a safe shutdown condition during normal or accident conditions, mitigating the consequences of an accident or preventing the occurrence of an accident; accounting for the impact from externally generated missiles. The NRC routinely reviews all safety-related systems relative to this requirement. The NRC has questioned some designs with respect to not providing specific tornado-missile protection for the diesel generator exhaust stacks. Applicants must either add the required protection or provide an analysis of the design to show that the diesel exhaust stacks would not be rendered inoperable in the event of a tornado-missile strike. Credit can be taken for the protection from buildings surrounding the exhaust stacks.

WAPWR Response

The diesel exhaust stacks for the WAPWR design will be evaluated relative to their susceptibility to a tornado-missile and their ability to operate normally after a missile strike. If it is determined that the diesel exhaust stacks can be rendered inoperable due to tornado-missiles, additional protection will be provided. Tornado-missiles are discussed in more detail in Section 5.1 (item 38).

4. Seismic Instrumentation Program

Discussion

The NRC normally reviews the plant instrumentation program, including a comparison to Regulatory Guide 1.12, "Instrumentation for Earthquakes" (which endorses earthquake instrumentation specified in ANSI/ANS 2.2-1978), the location and description of instrumentation, control room operator notification, and the comparison of measured and predicted responses. NRC reviews of the comparison to Regulatory Guide 1.12 have determined that, in some cases, the design did not provide discrete response spectrum recorders. Rather, data from the accelerometers are fed to signal conditioning and analysis equipment and converted to response spectra. The NRC's position is that a discrete response spectrum recorder should be provided at the containment foundation. The NRC requires that the recorder be capable of providing immediate control room indication.

WAPWR Response

The WAPWR plant design will include a discrete response spectrum recorder at the containment foundation or Westinghouse will provide justification for deviating from the Regulatory Guide 1.12 position.

5. Periodic Leak Testing of Pressure Isolation Valves

Discussion

There are several safety systems connected to the reactor coolant pressure boundary that have design pressures below the rated reactor coolant system pressure. There are also some systems which are rated at full power pressure on the discharge side of pumps but have pump suction below reactor coolant system pressure. To protect these systems from reactor coolant system pressure, two or more isolation valves are placed in series to form the interface between the high pressure reactor coolant system and the low pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure system and causing an intersystem loss-of-coolant accident.

Applicants typically leak test the check valves which form the above interface in the hot and cold leg safety injection systems. These valves form the Event V configurations as described in the letter to licensees from D. Eisenhut (NRC) dated February 23, 1980. However, other pressure valve configurations exist whose failure could lead to an intersystem loss-of-coolant accident or unsafe plant operating conditions. Other subsystems of concern to the NRC are the accumulator discharge check valves, the boron injection system pressure isolation valves, and the motor-operated valves on the residual heat removal pump suction.

Applicants have recently been required to categorize pressure isolation valves for the safety injection, residual heat removal, and boron injection systems as Category A or AC. These categorizations meet the NRC requirements. Pressure isolation valves are required to be Category A or AC and to meet the appropriate valve leak rate test requirements of IWV-3420 of Section XI of the ASME Code except that the allowable leakage rate shall not exceed 1 gpm for each valve as stated in the technical specifications.

WAPWR Response

For all WAPWR safety-related systems connected to the reactor coolant pressure boundary that have design pressures or pump suction pressures below the reactor coolant system pressure, procedures will be developed to periodically test the isolation valves, which form the interface between the high pressure reactor coolant system and the lower pressure systems, to ensure they meet the required leak tight integrity.

6. Axial Growth

Discussion

The growth of fuel rods and fuel assemblies is mainly governed by the behavior of fuel pellets, Zircaloy-4 cladding, and the Zircaloy-4 guide thimble tubes.

For the Zircaloy cladding and fuel assembly components, the axial-dimensional behavior is governed by creep (due to mechanical or hydraulic loading) and irradiation growth. The critical tolerances that require controlling are: (A) the spacing between the fuel rods and the fuel assembly (shoulder gap) and (B) the spacing between the fuel assemblies and the core internals. Failure to adequately design for the former may result in fuel rod bowing, and for the latter may result in collapse of the holddown springs. With regard to inadequately designed shoulder gaps, problems have been reported (H. Schenk, IAEA report SM-178-15, October 1973; K. Kuffer and H. R. Lutz, Fifth Foratom Conf., Florence, Italy, 1973; and FSAR of R. E. Ginna Unit 1, 1972) in foreign (Obrigheim and Beznau) and domestic (Ginna) plants that have necessitated predischarge modifications to fuel assemblies.

With regard to a design basis for shoulder gap spacing, Westinghouse has stated that interference is precluded by having clearance between the fuel rod end and the top and bottom nozzles. The design clearance accommodates the differences in growth, fabrication tolerances, and the differences in thermal expansion between the fuel cladding and the thimble tubes. Westinghouse does not have specific limits on growth, but does provide a gap spacing that is equal to or greater than a percentage of the fuel rod length.

With regard to fuel assembly growth, Westinghouse has a design basis that there shall be no axial interference between the fuel assembly and upper and lower core plates caused by temperature or irradiation. As a design limit, Westinghouse provides a minimum gap (that is a fraction of the fuel assembly length) between the fuel assembly and the reactor internals.

The Westinghouse analysis of shoulder gap spacing for the fuel assembly has found that interference will not occur until achieving burnups beyond traditional values. The NRC has stated the following: "The required shoulder gap spacing has been reasonably accommodated. However, for extended burnup applications, the adequacy of the spacing should be reverified. Furthermore, because stress-free irradiation growth of zirconium bearing alloys is sensitive to texture (preferred cystallographic orientation) and retained cold work, which, in turn, are strongly dependent on the specific fabrication techniques that are employed during component production, reverification of the design shoulder gap should be performed if Westinghouse current fabrication specifications are significantly altered."

Finally, the NRC has found the Westinghouse analysis of fuel assembly growth to be acceptable. However, as stated in the above discussion on shoulder gap spacing, reverification of the fuel assembly growth should be performed if significant changes are made in the Westinghouse current fabrication techniques.

More recently, Westinghouse has submitted WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel," to the NRC which presents an evaluation of shoulder gap spacing at higher than traditional burnups.

WAPWR Response

Since the WAPWR fuel assembly design differs from previous designs, calculations will be performed to demonstrate that the design shoulder gap is adequate to account for axial growth effects. The fabrication techniques that will be used for the WAPWR fuel assembly will be similar to past practice so that the method of determining both fuel rod and fuel assembly growth remain valid.

Addressment of this concern will be completely documented during the licensing process for the WAPWR design.

7. Automatic Indication of Block of Signals Initiating Auxiliary Feedwater Following Trip of the Main Feedwater Pumps

Discussion

The signal which initiates auxiliary feedwater when the main feedwater pumps are tripped is manually blocked on normal shutdown of the main feedwater pumps. The design is such that the block is not automatically removed when the plant is returned to an operating mode where auxiliary feedwater initiation on loss of main feedwater is needed. Even though the signal to initiate auxiliary feedwater when the main feedwater pumps are tripped is considered to be an "anticipatory signal," for which no credit is taken in the safety analyses, the NRC position is that the design should include appropriate features to ensure that the block is removed when the plant is returned to an operating mode where auxiliary feedwater initiation on loss of main feedwater is needed.

Several applicants have committed to provide automatic indication of the block of the signals which initiate auxiliary feedwater or loss of both main feedwater pumps on the bypassed and inoperable status panel. Also, operating procedures have been written to limit the time during which the block can be in effect. Typically, blocking will be permitted just before shutdown of the last operating main feedwater pump and removed just after the first main feedwater pump is put into service.

WAPWR Response

The WAPWR bypassed and inoperable status indication system will provide an automatic indication of the block of any signals which initiate emergency feedwater to the steam generators on loss of both main feedwater pumps. In addition, the WAPWR emergency response guidelines will only allow the signal to be manually blocked for limited modes of operation.

8. Indicator, Alarm, and Test Features Provided for Instrumentation Used for Safety Functions

Discussion

Instrumentation for process measurements used for safety functions such as reactor trip or emergency core cooling typically are provided with the following:

- o An indicator in the control room to provide the operator information on the process variable being monitored.

- o An alarm to indicate to the operator that a specific safety function has been actuated.

- o Indicator lights or other means to inform the operator which specific instrument channel has actuated the safety function.

- o Rod positions, pump flows, or valve positions to verify that the actuated safety equipment has taken the action required for the safety function.

- o Design features to allow test of the instrument channel without interfering with normal plant operations and without lifting instrument leads or using jury rigs.

During the review of the various designs, the NRC has found that one or more of the features above were not provided for certain instrumentation used to initiate safety functions. Examples included instrumentation used to isolate essential service water to the air compressors and instrumentation used to isolate the non-safety-related portion of the component cooling water system. Applicants were asked to provide the NRC with a list of all instrument channels which perform a safety function where one or more of the features listed above were not provided. The NRC position was that the applicant should, as a minimum, provide the last four features listed above or provide a justification applicable to the specific safety function involved where any of the features were not provided.

WAPWR Response

All instrumentation for process measurements used for safety functions will be evaluated to determine which of the above features should be incorporated into the WAPWR design. Justification will be provided for any features not included.

Addressment of this concern will be completely documented during the licensing process for the WAPWR design.

9. Actuation of Valve Component-Level Windows on the Bypassed and Inoperable Status Panel

Discussion

Some designs for actuation of the accumulator valve component-level windows on the bypassed and inoperable status panel are such that the bypass indication is not actuated until the valve reaches the fully closed position rather than when the valve leaves the fully open position. The NRC position is that bypass indication should be actuated when a valve leaves the position required for it to accomplish its safety function.

The NRC has stated that for all valves where valve misalignment is indicated on the bypassed and inoperable status panel, the bypass indication should occur when the valve leaves the position required for it to accomplish its safety function.

WAPWR Response

All valves whose misalignment is indicated on the bypassed and inoperable status indication system will meet the above stated criteria for the WAPWR design.

10. Sequencing of Loads on the Offsite Power System

Discussion

It is the NRC's position that the offsite power system should have sufficient capacity to supply all required loads without sequencing of loads. Several designs utilize a load sequencer for connecting each load group of safety-related loads onto their associated bus regardless of the source of power to that bus. This sequencing feature minimizes the system disturbance and provides the most stable means of starting the safety loads.

This design feature is well within the bounds of the accident analyses. When safety-related loads are sequenced onto the bus, whether onto the diesel or onto the offsite network, the starting times of all loads are less than those required by the analyses.

Based on the above justification, the NRC has concluded that proposed designs have met the capacity requirements of General Design Criterion 17, "Electric Power Systems," and are acceptable. However, sequencing of loads on the offsite power system represents an additional source of unreliability, and, because the same sequencer is used for both onsite and offsite power sources, independence between sources may be compromised. Therefore, it is the NRC's position that the applicant must perform an analysis, to demonstrate: (A) that there are no credible sneak circuits or common failure modes in the sequencer design that could render both the onsite and offsite power sources unavailable and (B) that the reliability of the offsite power source has not been compromised.

WAPWR Response

Interface criteria, relative to the offsite power system, will be generated as part of the WAPWR design which will address this concern.

11. Diesel Generator Protective Trips

Discussion

A number of tripping devices are provided for each diesel generator. The majority of tripping devices are either bypassed during accident conditions or actuated by two-out-of-three logic. These devices meet Regulatory Guide 1.9, "Selection, Design, and Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plant," are acceptable, and periodic testing requirements for the two-out-of-three logic are included in the technical specifications.

However, in some plant designs, some of these protective devices are actuated by one-out-of-one logic, are not bypassed during accident conditions, and are not in compliance with Regulatory Guide 1.9. In justification of this noncompliance, applicants have shown that if one diesel generator is tripped by a protective device, the redundant load group will function as a backup. The NRC has found this justification unacceptable. The NRC is concerned that these protective devices could interfere with the successful functioning of the diesel generators when they are most needed, that is, during an accident condition. The criterion should be to provide standby power when it is needed to mitigate the effects of an accident condition rather than to protect the diesel generators from possible damage. Thus, it is the NRC's position that these protective devices be bypassed during accident conditions.

WAPWR Response

The guidelines given in Regulatory Guide 1.9 will be addressed in the design of the diesel generator tripping devices. Any tripping devices that are not either bypassed during accident conditions or actuated by two-out-of-three logic will be justified.

Addressment of this concern will be completely documented during the licensing process for the WAPWR design.

12. Non-Safety Loads Powered from the Class 1E AC Distribution System

Discussion

Present regulatory practices for operating license applications allow the connection of non-safety loads in addition to the required safety loads to the Class 1E distribution system if it can be shown that the connection of the non-safety loads will not degrade the Class 1E power system below an acceptable level.

Several plant designs permit a number of non-Class 1E loads to be connected to Class 1E power sources. The majority of these circuits are isolated by a circuit breaker that opens on a safety injection signal.

The circuits beyond the isolation device (circuit breaker) are treated as non-Class 1E circuits and, as such, are routed in non-Class 1E cable raceways. It is the NRC's concern that these non-Class 1E circuits may unnecessarily challenge and degrade redundant standby power supplies below an acceptable level on the loss of offsite power when there is no safety injection signal. Thus, the NRC's position is that the non-Class 1E circuits also be automatically disconnected (so that they will not be automatically sequenced on the loss of offsite power) or the circuits must be analyzed to demonstrate that the standby power supplies will not degrade Class 1E systems below an acceptable level.

WAPWR Response

The above concerns will be addressed in the design of the Class 1E power supply system. All non-Class 1E circuts will be automatically disconnected from Class 1E power supplies on loss of offsite power or justification will be provided to demonstrate that the Class 1E systems will not be degraded if the non-Class 1E circuts remain connected.

Addressment of this concern will be completely documented during the licensing process for the WAPWR design.

13. Battery Capacity

Discussion

Usually, initial battery capacity is 50 percent greater than required due to temperature, voltage, and specific gravity fluctuation, as well as the replacement criterion of 80 percent. Some applicants have tried to justify the use of a battery capacity which is less than the 50 percent above that required. However, the NRC has maintained that a battery size 50 percent greater than required capacity is required.

WAPWR Response

For the WAPWR design, the initial battery capacity will be chosen to provide at least a 50-percent-greater-than-required capacity.

14. Low and/or Degraded Grid Voltage Condition

Discussion

The Millstone Unit 2 low-grid-voltage occurrence brought into focus the potential common-mode failure of redundant safety-related electrical equipment that could result from a degraded-grid-voltage condition. This occurrence prompted the NRC staff to develop various positions to ensure that the requirements of General Design Criterion 17, "Electric Power Systems," will be satisfied with regard to making provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electrical power supplies. These provisions for maintaining the independence between the offsite and onsite emergency power systems are emphasized in IEEE Standard 308-1974, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," which states that preferred offsite and the standby onsite emergency power supplies shall not have a common-mode failure between them.

The following NRC positions are being used in the evaluation of electrical power designs for operating plants and construction permit and operating license applications:

o In addition to the undervoltage scheme provided to detect loss of onsite power at the Class 1E buses, a second level of undervoltage protection with time delay should also be provided to protect the Class 1E equipment; this second level of undervoltage protection shall satisfy the following criteria:

  (A) The selection of undervoltage and time delay set points shall be determined from an analysis of the voltage requirements of the Class 1E loads at all onsite system distribution levels.

  (B) Two separate time delays shall be selected for the second level of undervoltage protection based on the following conditions:

(1) The first time delay should be of a duration that establishes the existence of a sustained degraded voltage condition (that is, longer than a motor-starting transient). Following this delay, an alarm in the control room should alert the operator to the degraded condition. The subsequent occurrence of a safety injection actuation signal should immediately separate the Class 1E distribution system from the offsite power system.

(2) The second time delay should be of a limited duration so that the permanently connected Class 1E loads will not be damaged. Following this delay, if the operator has failed to restore adequate voltages, the Class 1E distribution system should be automatically separated from the offsite power system. Bases and justification must be provided in support of the actual delay chosen.

(C) The voltage sensors shall be designed to satisfy the following applicable requirements derived from IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations":

(1) Class 1E equipment shall be used and shall be physically located at and electrically connected to the Class 1E switchgear.

(2) An independent scheme shall be provided for each division of the Class 1E power system.

(3) The undervoltage protection shall include coincidence logic on a per-bus basis to preclude spurious trips of the offsite power source.

(4) The voltage sensors shall automatically initiate the disconnection of offsite power sources whenever the voltage set point and time delay limits have been exceeded.

(5) Capability for test and calibration during power operation shall be provided.

(D) The technical specifications shall include limiting conditions for operation, surveillance requirements, trip set points with minimum and maximum limits, and allowable values for the second-level voltage protection sensors and associated time delay devices.

o The NRC requires that the system design automatically prevent load shedding of the emergency buses once the onsite sources are supplying power to loads on the emergency buses. The design shall also include the capability of the load-shedding feature to be automatically reinstated if the onsite source supply breakers are tripped. The automatic bypass and reinstate feature shall be verified during the periodic testing.

In the event an adequate basis can be provided for retaining the load-shedding feature during the above transient conditions, the set point value in the technical specifications for the first and second-level of undervoltage protection (loss of offsite power) must specify a value having maximum and minimum limits. The basis for the set points and limits selected must be documented.

o The voltage levels at the safety-related buses should be optimized for the maximum and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power sources by appropriate adjustment of the voltage tap settings of the intervening transformers. The tap settings selected should be based on an analysis of the voltage at the terminals of the Class 1E loads. The analyses performed to determine minimum operating voltages should typically consider maximum unit steady-state and transient loads for events such as a unit trip, loss-of-coolant accident, startup, or shutdown, with the offsite power supply (grid) at minimum anticipated voltage and only the offsite source being considered available. Maximum voltages should be analyzed with the offsite power supply (grid) at maximum expected voltage concurrent with minimum unit loads (for example, cold shutdown, refueling). A separate set of the above analyses should be performed for each available connection to the offsite power supply.

o The analytical techniques and assumptions used in the voltage analyses must be verified by actual measurement. The verification and test should be performed before initial full power reactor operation on all sources of offsite power by:

(A) Loading the station distribution buses, including all Class 1E buses down to the 120/208 volt level, to at least 30 percent.

(B) Recordi..g the existing grid and Class 1E bus voltages and bus loading down to the 120/208 volt level at steady-state conditions and during the starting of both a large Class 1E and non-Class 1E motor (not concurrently).

(C) Using the analytical techniques and assumptions of the previous voltage analyses as well as the measured existing grid-voltage and bus-loading conditions recorded during conduct of the test, to calculate a new set of voltages for all the Class 1E buses down to the 120/208 volt level.

(D) Comparing the analytically derived voltage values against the test results.

WAPWR Response

The above NRC recommendations will be considered in the design of the WAPWR onsite power supply system.

Addressment of this concern will be completely documented during the licensing process for the WAPWR design.

15. Submerged Electrical Equipment as a Result of a Loss-of-Coolant Accident

Discussion

The NRC staff has been concerned that, after a loss-of-coolant accident, fluid from the reactor coolant system and from operation of the emergency core cooling systems may collect in the primary containment and reach a level that may cause certain electrical equipment located inside the containment to become submerged, thereby rendering it inoperable. Both safety and non-safety-related electrical equipment is of concern, because its failure may cause electrical faults that could compromise the operability of redundant Class 1E power sources or the integrity of containment electrical penetrations. The safety-related electrical equipment that may be submerged is also of concern if this equipment is required to mitigate the consequences of the accident for both the short-term and long-term emergency core cooling functions and for containment isolation.

The NRC's position is that all electrical equipment (Class 1E and non-Class 1E)--versus only that equipment required to operate--be located above the maximum possible flood level or be qualified for submerged operation, or the lack of qualification must be justified.

WAPWR Response

The above NRC position will be addressed as part of the WAPWR plant layout configuration. All electrical equipment will be either located above the maximum flood level or justification will be provided for deviating from this NRC position.

16. Compliance with Position 1 of Regulatory Guide 1.63

Discussion

General Design Criterion 50, "Containment Design Basis," of Appendix A to 10CFR Part 50 requires, in part, that the reactor containment structures, including electrical penetrations, be designed so that the containment structure and its internal compartments can accommodate, without failure, the pressure and temperature conditions resulting from any loss-of-coolant accident. Therefore, electrical penetration assemblies are designed to withstand, without the loss of mechanical integrity, the maximum available fault-current-versus-time conditions that could occur given single random failures of circuit overload protective devices, as recommended by Regulatory Guide 1.63, Revision 2, "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants."

To verify compliance in this area, the NRC has asked applicants to provide coordinated fault-current-versus-time curves for each representative type cable. For each cable, the curves must show the relationship of the fault carrying capability between the electric penetration, the primary overcurrent protective device, and the backup overcurrent protective device. Also, technical specification requirements for locking out power to equipment that is not required to be operational during normal operation, physical independence of primary and backup protective devices, and testability of protective devices must be addressed in the safety analysis report.

WAPWR Response

All electrical penetration assemblies for the WAPWR design will be designed to withstand the maximum available fault-current-versus-time conditions that could occur while accounting for the single random failure of circuit overload protective devices.

17. Integrated Safety Assessment Program (ISAP)

Discussion

The NRC's policy and planning guidance for 1983 expresses an intention that both new and existing requirements be implemented according to the importance to safety of each and according to the licensee's ability to accomplish each. To achieve realistic scheduling of plant improvements the NRC feels that the licensee must inventory and understand the NRC imposed requirements; however the NRC also assumes that the NRC must understand the licensee's inventory, plans and capability for making improvements.

Generic Letter 82-33 (see Section 6.4, item 74) was a recent NRC thrust for the licensee to submit plant specific plans that deal with the unique aspects of installing emergency response facilities and a set of five other TMI requirements.

The NRC considers the Integrated Safety Assessment Program (ISAP) as an extension of experience gained in the Systematic Evaluation Program (SEP), the Interim Reliability Evaluation Program (IREP) and other program interfaces with utilities. SEP and IREP were conducted on selected plants and the pilot effort was expected to evolve into the National Reliability Evaluation Program (NREP). The NRC now has dropped consideration of NREP and intends to integrate its objectives into ISAP. From the NRC's point of view the following regulatory areas would be factored into ISAP

- Pending licensing actions (NUREG-0748)
- Outstanding TMI actions (NUREG-0737)
- Emergency Response Requirements (Generic Letter 82-33)
- Resolved Generic Issues
- New Requirements from SEP and IREP Lessons Learned
- Pending USIs and Generic Issues
- Operating Experience (Region "Blackbook")

The NRC plans for approximately four plants in 1984 to be selected for an ISAP pilot effort with an additional six plants in 1985. The deadlines for some existing requirements would be relaxed to enable the selected utilities to make their all-issue studies. Each would be asked to make probability safety assessments (PSAs). The PSA is similar to the probability risk assessment but also includes external events. The PSAs would not be primarily for risk or consequence evaluation as with the PRAs but would be designed to evaluate performance of the plant systems. Any existing plant PRAs will be an initial advantage in the PSA effort.

Several SEP plants are volunteers for the ISAP program. The potential benefit for the utility is the program's capability of relaxing or removing requirements for the specific plant if the implementation is relatively low in terms of cost effectiveness. The NRC also offers the Integrated Lining Schedule program as a utility option. Its charter involves realistic scheduling based on priority ranking but does not provide for the ultimate removal of ineffective requirements.

WAPWR Response

This is an administrative item aimed at plant licensees regarding prioritizing, planning and scheduling the implementation of plant improvements, and as such has no impact on the WAPWR design.