5.0 GENERIC SAFETY ISSUES

As mentioned in Section 4.0, the NRC continuously evaluates the safety requirements used in its reviews against new information as it becomes available. In 1978 the NRC published NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants." This NUREG identified over 130 specific generic safety issues and assigned each issue to one of four categories.

- Category A

  "Those generic technical activities judged by the staff to warrant priority attention in terms of manpower and/or funds to attain early resolution. These matters include those the resolution of which could: (A) provide a significant increase in assurance of the health and safety of the public, or (B) have a significant impact upon the reactor licensing process."

- Category B

  "Those generic technical activities judged by the staff to be important in assuring the continued health and safety of the public but for which early resolution is not required or for which the staff perceives a lesser safety, safeguards, or environmental significance than Category A matters."

- Category C

  "Those generic technical activities judged by the staff to have little direct or immediate safety, safeguards, or environmental significance, but which could lead to improved staff understanding of particular technical issues or refinements in the licensing process."

o Category D

"Those proposed generic technical activities judged by the staff not to warrant the expenditure of manpower or funds because little or no importance to the safety, environmental, or safeguards aspects of nuclear reactors or to improving the licensing process can be attributed to the activity."

Since the issuance of NUREG-0410, certain generic safety issues have been resolved with the issuance of regulatory criteria or guidance, and new generic safety issues have been identified. The major sources of identification of new generic safety issues since 1978 are licensee event reports, ACRS reports, Inspection and Enforcement bulletins, circulars, and information notices. For example, NUREG-0572, "Review of Licensee Event Reports (1976-1978)," identifies certain new generic safety issues resulting from an ACRS review of licensee event reports.

The following sections provide a discussion of each of the current NRC Category A, B, C, and D generic issues and new "uncategorized" generic issues as they relate to the WAPWR design.

The NRC has assigned priorities (i.e., high, medium, low, or drop) to each generic safety issue. Current NRC priorities are documented in draft NUREG-0933, "A Prioritization of Generic Safety Issues." It should be realized that NRC priorities are heavily influenced by impacts on operating plants and a different priority assignment could have resulted if this were not the case. In other words, just because an issue is assigned a low priority does not mean that it will not be considered in the WAPWR design. Nevertheless, the NRC priority rankings were considered, as appropriate, in the development of the licensing response for the WAPWR design corresponding to each generic safety issue.

5.1 CATEGORY A ISSUES

Most safety issues identified in Category A are referred to today as Unresolved Safety Issues which are discussed in detail in Section 4.0. However, the NRC assignment of an issue to Category A does not necessarily mean that the issue is safety significant, and accordingly, all Category A issues do not involve Unresolved Safety Issues.

The following discussions pertain to current Category A issues in relation to the WAPWR design. NRC discussions and descriptions of these issues are contained in NUREG-0371, "Task Action Plans for Generic Activities (Category A)."

1. Issue A-1: Water Hammer

   Discussion

   This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

2. Issue A-2: Asymmetric Blowdown Loads on Reactor Primary Coolant Systems

   Discussion

   This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

3. Issue A-3: Westinghouse Steam Generator Tube Integrity

   Discussion

   This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

4. Issue A-4: Combustion Engineering Steam Generator Tube Integrity

Discussion

This issue is not applicable to Westinghouse steam generator designs.

5. Issue A-5: Babcock and Wilcox Steam Generator Tube Integrity

Discussion

This issue is not applicable to Westinghouse steam generator designs.

6. Issue A-6: Mark I Short Term Program

Discussion

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

7. Issue A-7: Mark I Long Term Program

Discussion

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

8. Issue A-8: Mark II Containment Pool Dynamic Loads

Discussion

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

9. Issue A-9: Anticipated Transients Without Scram

<u>Discussion</u>

This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

10. Issue A-10: BWR Feedwater Nozzle Cracking

<u>Discussion</u>

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

11. Issue A-11: Reactor Vessel Materials Toughness

<u>Discussion</u>

This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

12. Issue A-12: Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports

<u>Discussion</u>

This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

13. Issue A-13: Snubber Operability Assurance

<u>Discussion</u>

Snubbers are utilized primarily as seismic and pipe whip restraints at nuclear power plants. Their safety function is to operate as rigid supports for restraining the motion of attached systems or components under rapidly applied load conditions such as earthquakes, pipe breaks, and severe hydraulic transients.

Operating experience reports have shown that a substantial number of snubbers have leaked hydraulic fluid and the rejection rate from functional testing and inspection has been high. This lead to an NRC and ACRS concern regarding the effect of snubber malfunctions on plant safety.

The NRC considers this issue as being technically resolved for pressurized water reactors with the issuance of:

- o Standard Technical Specification 3/4.7.9, "Snubbers."
- o Standard Review Plan 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures."
- o Draft Regulatory Guide and Value/Impact Statement, Task SC-708-4, "Qualification and Acceptance Tests for Snubbers Used in Systems Important to Safety."

The following is a brief summary of the NRC criteria contained in the three sources for technical resolution of this generic issue:

- o Standard Technical Spec ication 3/4.7.9

  All safety-related* snubbers must be listed in the technical specifications. Safety-related snubbers must be visually inspection for operability at certain intervals depending on the number of inoperable snubbers found in the prior inspection. In addition, safety-related snubber types must be functionally tested at least once per 18 months during shutdown.

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* The Standard Technical Specifications currently use the term "safety-related". Indications are that the NRC really means "important to safety".

o Standard Review Plan 3.9.3

NRC acceptance criteria are provided for:

(A) Structural analysis and systems evaluation (interaction of snubbers with the systems and components to which they are attached).

(B) Characterization of mechanical properties (spring rates used in analytical models).

(C) Design specifications.

(D) Installation and operability verification.

(E) Use of additional snubbers as a result of unanticipated piping vibration or interference problems during construction.

(F) Inspection and testing.

(G) Classification and identification (safety analysis report documentation).

o Draft Regulatory Guide, Task SC-708-4.

(A) Functional specifications (in accordance with Appendix A of the guide) should be prepared for each snubber model and should be used as the basis to determine the acceptability of test results.

(B) Snubbers should be constructed according to Subsection NF of Section III of the ASME Code.

(C) Materials used that are exempted from Subsection NF should be compatible with other materials of contruction and the working environment.

(D) Snubbers should be qualified (in accordance with Appendix B of the guide).

(E) A completed snubber unit should be accepted from the production line only if it has successfully passed all the testing described in Appendix C of the guide.

(F) The quality assurance requirements of Appendix B to 10CFR Part 50 apply.

WAPWR Response

Westinghouse will completely document and justify any deviations from the above mentioned NRC snubber acceptance criteria during the licensing process for the WAPWR design.

14. Issue A-14: Flaw Detection

Discussion

The failure probability of a reactor pressure vessel is considered to be sufficiently low to exclude it from consideration as a design basis accident. The rationale for this low probability relies heavily on the maintenance of rigorous manufacturing and quality control standards, adherence to conservatively derived operating limits and effective, regularly repeated inservice inspection. The inspection method must be sufficiently sensitive to assure that all flaws approaching the severity levels used as a basis for establishing the margin against fracture during normal operating and transient conditions will be reliably detected particularly in the later stages of plant life, where reduction in fracture toughness of the vessel materials may occur.

Similarly, the integrity of the entire primary pressure boundary and of important safety system components must be assured throughout the plant lifetime. General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," requires that the design reflect consideration of uncertainties in determining the size of flaws and General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," requires that the reactor coolant pressure boundary be designed so as to permit periodic inservice inspection.

Flaw detection methods and procedures specified in the present inservice inspection rules (i.e., Section XI of the ASME Code) leave uncertainties concerning the smallest size defect which can be reliably detected by non-destructive testing in various parts of the pressure boundary. Similarly, significant uncertainties are known to be associated with dimensional characterization of identified defects. The ability to detect and adequately size flaws is essential in assuring continued integrity of the reactor coolant pressure boundary and in assessing the margin against failure under various plant conditions throughout the full life of the plant.

The purpose of Generic Task A-14 is for the NRC to assess the capability of current and new advances in flaw detection methods and recommend improvements in equipment, methods, and requirements for inclusion in industry and regulatory standards, codes, and guides. A major part of the NRC effort on this issue is being carried out under an Office of Nuclear Regulatory Research Program on Nondestructive Examination as documented in Section 2.5 of NUREG-0961, "Long-Range Research Plan (FY 1984-FY 1988)". This task has resulted in the issuance of Regulatory 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Pressure and Inservice Examination" and the preparation of piping inspection provisions which are being incorporated into Section XI of the ASME Code.

WAPWR Response

The NRC has determined that flaw detection is not a safety issue by itself and should be dropped from the list of generic safety issues as a separate issue. It should, however, be assessed in the resolution of applicable specific safety issues (i.e., Unresolved Safety Issues A-3, A-4, A-5, A-12, and A-49; and Uncategorized Safety Issues 15 and 29).

As mentioned above, the end result of this generic task is anticipated to be revised inspection requirements and flaw detection techniques. By itself this generic task will not result in any hardware impact on the WAPWR design. Westinghouse will completely document and justify any deviations from the NRC Regulatory Guide 1.150 positions during the licensing process for the WAPWR design.

15. Issue A-15: Primary Coolant System Decontamination and Steam Generator Chemical Cleaning

Discussion

Operation of a light water reactor results in slow corrosion of the interior metal surfaces of the primary coolant system. The resulting corrosion products circulate through the reactor core and are activated by neutron flux from the fissioning reactor fuel. While some of these activated corrosion prodcuts are removed by the reactor's water chemistry system, a small amount if continually deposited or plated out on the primary coolant system's internal surfaces. Once activated corrosion products are deposited or plated out, they are not removed by the reactor water cleanup system and continue to accumulate.

The presence of this accumulation of highly radioactive corrosion products adhering to the interior surfaces of the primary coolant system has, in some cases, prevented licensees from carrying out some of the less important inservice inspections required by their technical specifications.

Because of the safety significance of the systems and components being inspected, the NRC believes an approach should be developed to permit these inspections while at the same time minimizing personnel radiation exposures. Several methods of decontamination to reduce radioactivity levels in the primary system are available to the nuclear industry for application in operating reactors. These include chemical decontamination, electropolishing, mechanical and hydraulic decontamination. For example, NUREG/CR-1915, "Decontamination Processes for Restorative Operations and as a Precursor to Decomissioning: A Literature Review," and similar documents are intended to give sufficient information to allow reasonable selections for decontamination processes for any given reactor.

This generic task involves an NRC review of existing an ongoing decontamination technology with the purpose of providing guidance to the NRC staff and industry relating to acceptable methods of decontamination of reactor primary coolant systems.

The NRC considers this issue resolved with the issuance of NUREG/CR-2963, "Planning Guidance for Nuclear Power Plant Decontamination Operations." This NUREG provides generic guidance for planning, implementing, and monitoring restorative decontamination.

WAPWR Response

One of the design objectives for the WAPWR is to minimize exposures to individuals associated with operation and maintenance through such methods as material selection, chemistry control, plant layout, high purification capability, plating of manways and other sealing surfaces, etc.

Westinghouse recommends that should decontamination of systems or components be necessary for whatever reason, methods acceptable to the NRC and compatible with the particular system or component being decontaminated should be utilized by a WAPWR licensee.

16. Issue A-16: Steam Effects on BWR Core Spray Distribution

Discussion

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

17. Issue A-17: Systems Interactions in Nuclear Power Plants

Discussion

This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

18. Issue A-18: Pipe Rupture Design Criteria

Discussion

Current criteria for postulating pipe breaks and specifying the protection therefrom have been developed over a long period of time. Accordingly, these criteria lack consistency when applied inside and outside the containment and are subject to misinterpretation in certain areas. In addition, the NRC believes the effect on normal operation of piping design requirements for postulated accidents needs to be further considered. The purpose of this generic task is to develop consistent pipe rupture criteria, evaluate the break exclusion region of piping in containment penetration areas, and develop composite design requirements of piping systems for abnormal events and normal operation.

The following are the specific NRC discussions of each of these concerns:

- o Current design criteria for the postulation of pipe breaks and protection therefrom have been developed over a period of time and lack consistency when applied inside and outside containment. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside

Containment," is based on the concept of a limited number of design basis breaks and Standard Review Plan 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," combines limited design basis breaks for mechanistic protection and unlimited breaks for nonmechanistic protection. The NRC believes that their current efforts toward documentation of the rationale and engineering justification for the existing pipe break criteria should continue.

- An evaluation of the pipe break exclusion concept in the containment penetration area of both pressurized water reactor and boiling water reactor plants is required. The need for and extent of break exclusion regions, criteria for the use of guard pipes, and adequacy of design requirements for piping systems in break exclusion regions are topics for which improved NRC guidance will be developed.

- The development of postulated pipe rupture criteria and the trend towards more conservative seismic criteria have placed increased emphasis on piping system design to withstand these dynamic events, but have also resulted in systems which are significantly more rigid. These more rigidly designed systems in the newer plants have resulted in calculated stresses for normal operation which, although still within code limits, are significantly higher than in earlier plants. In addition, dynamic event devices, such as snubbers and pipe whip restraints, which have been added in increased numbers have the potential for deleterious interaction with the piping system during its normal operation. A balance in piping system design for both normal and abnormal situations should be achieved to assure that consideration is given to the effects that abnormal situation design criteria have on normal operation.

Currently, Westinghouse utilizes the criteria developed in WCAP-8172/8082, "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop," as the basis for postulating pipe breaks in the reactor coolant system. However, Westinghouse has performed extensive analyses and testing on reactor coolant system piping and weld metal to demonstrate that it is unrealistic to postulate double-ended pipe breaks in the reactor coolant system. Westinghouse has also performed a probabilistic study which determined the probability of a double-ended pipe rupture resulting from a safe shutdown earthquake to be on the order of 10-12. In research studies funded by the NRC, Lawrence Livermore Laboratories has reached similar conclusions based on studies of the Zion plant.

For piping other than reactor coolant system piping, Westinghouse has generally postulated pipe breaks in accordance with Branch Technical Position MEB 3 1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment." This NRC position essentially requires pipe breaks to be postulated at specific locations based on stress and other criteria. From a design, analysis, and operating perspective, there would be significant advantages to reducing the number of required postulated pipe break locations. Consequently, it would be of benefit to develop alternative pipe break criteria similar to that developed for reactor coolant system piping. For stainless steel piping, the analyses and testing performed for the reactor coolant system piping could be expanded to have application to all stainless steel piping in the plant. For piping other than stainless steel, an approach for justifying alternative pipe break criteria (similar to that used for the reactor coolant system piping) would have to be developed. Because of the magnitude and variety of piping outside the reactor coolant system, development of alternative pipe break criteria would be an extensive program.

In regard to postulating pipe breaks in containment penetrations, current NRC acceptance criteria include the use of guard pipe and the definition of break exclusion areas. If appropriately justified, containment penetrations can be identified as break exclusion areas.

In regard to piping systems designs to withstand dynamic events, current piping design and analysis is generally governed by upset or faulted condition transients. However, for the reactor coolant system, Westinghouse has performed inplant testing to demonstrate the acceptability of the system design for normal operating conditions. The relaxation of the pipe break criteria discussed above would add to the capability of the reactor coolant system to withstand normal operating loads. For piping outside the reactor coolant system, the development of new and relaxed pipe break criteria would permit the system design to more easily accommodate normal operating loads.

WAPWR Response

Westinghouse currently plans to develop and apply the relaxed pipe break criteria (as discussed above) to the WAPWR design. These criteria and their justifications will be documented during the licensing process for the WAPWR design.

19. Issue A-19: Digital Computer Protection Systems

Discussion

Some reactor protection systems which initiate control rod insertion now incorporate digital computers. The purpose of this task is for the NRC to standardize and document the acceptance criteria and methodology for the safety review of digital computer protection systems, and thereby improve the guidance available to NRC staff reviewers.

Digital systems when proposed by applicants are currently being reviewed by the NRC on a case-by-case basis, which is adequate.

WAPWR Response

This task is not directed toward affecting the level of safety, but toward improving the efficiency of NRC licensing reviews and, therefore, has no hardware impact on the WAPWR design.

20. Issue A-20: Impacts of the Coal Fuel Cycle

Discussion

Compliance with the National Environmental Policy Act (NEPA) requires that alternatives to a proposed Federal action be considered and that required alternatives be balanced against the base case in terms of associated environmental impacts.

A coal fired plant is currently the only realistic alternative to a nuclear power plant. Present treatment of the coal alternative is aimed essentially at economics and public health impacts. It is relatively incomplete in other areas of impact. This task is intended to provide a comprehensive summary which evaluates the environmental effects of the coal fuel cycle in a form directly comparable to that for the uranium fuel cycle. In the absence of such a generic treatment of the effects of using coal for generating electric power, it is necessary for the NRC staff to develop an analysis *de novo* for each licensing action, to present this individual analysis in detail in the environmental impact statement, and to defend it throughout the hearing process. This repetitive NRC staff effort could be avoided by preparing a generic statement suitable to support rulemaking proceedings. After the rulemaking procedure, such a statement would avoid repetitive NRC staff effort in individual cases.

The Commission has amended 10CFR Part 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection," to provide that need for power and alternative energy source issues will not be considered in operating license proceedings (i.e., applicants environmental reports, NRC environmental impact statements, and hearings). Consideration is still, however, required in construction permit proceedings.

WAPWR Response

This task is associated with an environmental proceedings issue that is not applicable to Westinghouse in relation to the WAPWR design.

21. Issue A-21: Main Steam Line Break Inside Containment Evaluation of Environmental Conditions for Equipment Qualification

Discussion

Safety-related equipment inside containment of a nuclear power plant is qualified for the most severe accident conditions under which it is expected to function. In a pressurized water reactor, this has for older generation plants been assumed to be the pressure and temperature that would accompany a loss-of-coolant accident resulting from the failure of the largest pipe in the reactor primary system. However, for most plant designs, calculations indicate that the failure of a main steam line inside containment results in a temperature that is higher than the temperature calculated for a loss-of-coolant accident and, therefore, possibly higher than the temperature for which the safety-related equipment is qualified. The purpose of this task is for the NRC to recommend acceptable methods of calculating environmental conditions that would result from a steam line failure within the containment for the purpose of qualifying safety-related equipment.

Although calculations indicated that the temperature within the containment following a steam line break are significantly higher than that following a loss-of-coolant accident, the duration of the high temperature was calculated to be short. Because of the relatively low heat transfer rate in superheated steam and the heat capacity of the affected safety-related equipment, the equipment itself would not be expected to exceed the temperature for which it was qualified as a result of this short duration peak in the temperature of the containment atmosphere. Therefore, the NRC believes that although this task may result in an improved basis for determining the environmental conditions for equipment qualification, it does not involve a major reduction in the degree of protection to the health and safety of the public.

WAPWR Response

This issue and its ultimate resolution is really aimed at operating plant licensees that did not qualify safety-related equipment to steam line break conditions. Current NRC criteria imposed on recent plant designs requires that all safety-related mechanical and electrical equipment shall be shown capable of performing their design safety functions under all normal, abnormal, accident, and post-accident environments. This criteria includes a postulated steam line break inside containment and the WAPWR design will meet this criteria.

Current Westinghouse generic environmental qualification programs are in accordance with this NRC criteria and the WAPWR design is not expected to be impacted by this issue. Westinghouse will demonstrate that the WAPWR design is enveloped by the generic qualification programs as discussed in Section 4.0, item 14 (Unresolved Safety Issue A-24, "Qualification of Class 1E Safety-Related Equipment"). The appropriate environment for equipment qualification will be determined as part of the mass and energy/ containment response analysis to be done as part of the normal design process.

22. Issue A-22: PWR Main Steam Line Break - Core, Reactor Vessel, and Containment Building Response

Discussion

Several aspects of the main steam line break analyses for pressurized water reactors as provided by licensees and applicants have been questioned by the NRC. This task involves evaluating these questions or concerns to confirm or modify the present NRC staff position on these analyses.

The first concern involves the current reliance on the operation of nonsafety-grade equipment as a backup for assumed single active failure in safety-grade equipment following a main steam line break. This task is intended to evaluate plant response to the operation or nonoperability of various non-safety-grade systems and components, and develop a reliability assessment of such equipment.

The majority of the components in the secondary system are essential to plant operation or availability, and are in a state of continuous or frequent operation. The considerable experience gained from both fossil and nuclear plant operation has demonstrated the high reliability of such components.

Awareness of this reliability level led to the current NRC staff position of permitting credit in accident analyses for selected non-safety-grade equipment as backup to safety-grade equipment.

This task effort is likely to confirm the reliability of this equipment and thus support the present NRC staff position.

An additional concern involves the mechanical response of the pressure vessel following a main steam line break. This task will consider safety systems and operator actions required to maintain acceptable pressure vessel stress levels and achieve long-term cooling. This potential safety problem related to reactor vessel integrity does not become important until the vessel has been subjected to extended neutron irradiation during plant operation. The irradiation effect is to reduce the allowable stress at reduced temperatures late in the life of the vessel.

When considering the sequence of conditions following a main steam line break, the primary system is first depressurized by overcooling through the secondary system. The reduction in primary system pressure causes a reactor trip and actuation of the emergency core cooling system (ECCS). Pressure reductions in the primary system are accompanied by temperature decrease with shrinkage of the liquid volume. Actuation of the ECCS replenishes the volume of liquid. Unless terminated or controlled by the operator, the ECCS could eventually refill and repressurize the primary system to the safety valve set point. This task involves evaluating the timing requirements for operator actions, the nature of the actions, and the likelihood of accomplishment and thus confirm that the operator actions necessary to maintain pressure vessel integrity can be reliably accomplished.

In regard to the concern dealing with reliance on non-safety-grade equipment following a main steam line break, Westinghouse analyses have traditionally taken credit for closing of the turbine stop and control valves and closing of the main feedwater control valves. The turbine valves are of high quality but they are not seismically qualified and are, therefore, classified as non-safety-grade. The turbine valves are assumed to close for the large double-ended steam line break. In this case, all steam generators blow down through the break until closure of the main steam isolation valves on a steam line isolation signal, at which time the blowdown from the intact steam generators is terminated. A reactor trip signal initiates a turbine trip thereby closing the turbine stop and control valves. The turbine valves provide backup protection for failure of a main steam isolation valve. If the turbine valves are assumed not to close because they are not safety-grade, the failure of a main steam isolation valve (taken as a single failure) in a loop other than the faulted loop would result in multiple steam generator blowdowns. The faulted loop would blow down through the break and the loop with the failed main steam isolation valve would blow down through the turbine.

For many operating plants, the main feedwater control valves are not seismically qualified and are, therefore, classified as non-safety-grade. These valves provide backup feedwater isolation to the main feedwater isolation valves during a steam line break event. If the feedwater isolation valve in the faulted loop is taken as the worst single failure and the feedwater control valves do not close as assumed, more feedwater would be supplied to the faulted steam generator which would increase the mass and energy release to containment.

Neither of the two aforementioned scenarios are currently analyzed as part of plant safety analyses. In the past, Westinghouse has justified taking credit for closure of the turbine valves and the main feedwater control valves on the basis of the high reliability of these valves.

Improvements in the response of operators are being made as specified in TMI Action Plan Item I.C.1, "Procedures for Transients and Accidents." Licensees have been asked to perform analyses that consider the occurrences of multiple failures, consequential failures, and operator errors which, if unmitigated, could lead to inadequate core cooling. In addition, these analyses are being carried out far enough in time to assure that all relevant thermal/hydraulic/neutronic phenomena are identified, and to address possible failures and operator errors during the long-term cooling phase. These analyses are expected to serve as the bases for Emergency Procedure Guidelines for Transients and Accidents including MSLB. These emergency procedure guidelines will be used as a basis for the development of plant-specific emergency procedures.

The NRC has concluded that this issue need not be continued as a separate generic issue. In addition, the improvements in procedures as a result of Item I.C.1 of NUREG-0737 will considerably reduce the risk and therefore, the first of the two concerns associated with this issue (the failure of containment following an MSLB) is of such low safety significance that it need not be considered further. The second concern (overcooling) will be thoroughly addressed by Unresolved Safety Issue A-49, "Pressurized Thermal Shock" (Secton 4.0, Item 27).

WAPWR Response

For the WAPWR design, the turbine valves and main feedwater control valves will be high quality valves with a high degree of reliability. Much of the circuitry associated with these valves will be safety/grade. Consideration is also being given to fully qualifying the valves.

23. Issue A-23: Containment Leak Testing

Discussion

One of the requirements of all operating licenses for water-cooled power reactors is that the primary reactor containment meet the leakage test requirements of Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10CFR Part 50. These requirements provide for preoperational and periodic verification by tests of the leak-tight integrity of the primary reactor containment, and systems and components which penetrate containment, and establish the acceptance criteria for such tests. The NRC staff and reactor licensees have experienced some difficulties in implementing Appendix J since it inception. The purpose of this task is for the NRC to revise this appendix so as to clarify existing requirements, and resolve conflicting and impractical requirements. Such clarification is being done now on a case-by-case basis as part of the NRC staff review process.

Current NRC acceptance criteria for containment leakage testing in accordance with 10CFR Part 50, Appendix J, is provided in Standard Review Plan 6.2.6, "Containment Leakage Testing."

The NRC has stated that revising Appendix J and issuing a Regulatory Guide with acceptable containment leakage testing methods have a low potential for reducing risk. However, considering the work accomplished thus far, they recommend that the containment leakage task be completed as a Regulatory Impact Issue on the basis of reducing the compliance burden on licensees and the paperwork burden on the NRC. They further stated that emphasis should be placed on eliminating the ambiguities in the present regulation without imposing more strinent leakage testing requirements since they do not appear to be effective in reducing risk.

WAPWR Response

As mentioned above, Appendix J to 10CFR Part 50 establishes requirements for containment leakage testing and by itself (either in its current form or some revised form after a rulemaking process) does not impact the actual WAPWR design beyond ensuring that adequate provisions and capabilities for testing are provided in accordance with General Design Criteria 52, 53, and 54 of Appendix A to 10CFR Part 50. Containment leak testing capabilities will be provided in the WAPWR design in accordance with Appendix J and Standard Review Plan 6.2.6. Any deviations from the NRC acceptance criteria will be justified during the licensing process for the WAPWR design.

24. Issue A-24: Qualification of Class 1E Safety Related Equipment

Discussion

This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

25. Issue A-25: Non-Safety Loads on Class 1E Power Sources

Discussion

Class 1E power sources are part of the onsite emergency power system and provide the electric power for the equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, containment and reactor heat removal or are otherwise essential in preventing a signficant release of radioactive material to the environment. Past regulatory practice has allowed the connection of non-safety loads in addition to the required safety loads to Class 1E power sources by imposing some restrictions. The purpose of this task is for the NRC to determine whether or not the reliability of the Class 1E power sources is signficantly afffected by the sharing of safety and non-safety loads.

The NRC considers this issue as technically resolved with the issuance of Revision 2 to Regulatory Guide 1.75, "Physical Independence of Electric Systems." This regulatory guide basically endorses IEEE Standard 384-1974, "IEEE Trial-Use Standard Criteria for Separation of Class 1E Equipment and Circuits", (also designated ANSI N41.14), and still permits Class 1E power sources to share safety and non-safety loads with certain restrictions.

A specific NRC concern related to this issue is discussed in Section 6.5 (item 18).

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Regulatory Guide 1.75 (and IEEE Standard 384-1974) positions during the licensing process for the WAPWR design.

26. Issue A-26: Reactor Vessel Pressure Transient Protection

Discussion

This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

27. Issue A-27: Reload Applications

Discussion

In mid-1975 the NRC provided licensees of operating reactor facilities a preliminary copy of a staff paper, "Guidance for Proposed License Amendments Relating to Refueling," and a "Refueling Information Request Form." The purpose was to provide guidance, although preliminary, to licensees as to the information the NRC considered to be essential for the conduct of its review of core reload submittals.

The purpose of this generic task is for the NRC to: (A) update the preliminary guidance issued to licensees in mid 1975 to assure conformance with the latest NRC technical positions that relate to core reloads, and (B) prepare formal review procedures to assure prompt and uniform review of licensee reload submittals.

The NRC considers this issue as technically resolved with the issuance of a draft regulatory guide (currently identified as Task SC-521-4), "LWR Core Reloads; Guidance on Applications for Amendments to Operating Licenses and on Refueling and Startup Tests."

WAPWR Response

This issue and its resolution deal with informational requirements necessary for the NRC reviews of reload applications and has no impact on the WAPWR design.

28. Issue A-28: Increase in Spent Fuel Pool Storage Capacity

Discussion

With the present "no-reprocessing" posture throughout the nuclear power industry, a considerable increase in onsite spent fuel storage will be required in order to permit continued operation of many nuclear power plants. The NRC considers this issue resolved with the issuance of a letter (dated April 14, 1978) from B. Grimes (USNRC) to all power reactor licensees, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications."

WAPWR Response

As stated above, the generic task actually deals with expanding the spent fuel storage capability at existing operating plants. In itself this issue does not impact the WAPWR design. However, due to the lack of sufficient "away-from-reactor" spent fuel storage capability, the WAPWR design will include spent fuel storage capability for approximately 5 cores. Possibly through the use of high density racks or spent fuel consolidation, this capability may be increased to approximately 7 to 10 cores.

29. Issue A-29: Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage

Discussion

Extensive efforts and resources are expended in designing nuclear power plants to minimize the risk to the public health and safety from equipment or system malfunction or failure. However, reduction of the vulnerability of reactors to industrial sabotage is currently treated as a plant physical security function and not as a plant design requirement. Although present reactor designs do provide a great deal of inherent protection against industrial sabotage, extensive physical security measures are still required to provide an acceptable level of protection. An alternate approach would be to more fully consider reactor vulnerabilities to sabotage along with economy, operability, reliability, maintainability, and safety during the preliminary design phase. Since emphasis is being placed on standardizing plants, it is especially important to consider measures which could reduce the vulnerability of reactors to sabotage. Of course, any design features to enhance physical protection must be consistent with present and future system safety requirements.

The objective of this task is for the NRC to identify and evaluate possible plant design variations which could improve the inherent sabotage resistance of nuclear power plants. Should this program identify promising design alternatives, the NRC has indicated that appropriate changes in regulations will be developed for future plants. The NRC has issued a draft task plan to investigate this issue. Their scheduled completion date for publishing conclusions relative to this issue is September, 1984.

For current plants high assurance of protection against industrial sabotage is achieved by the physical security measures required by 10CFR 73.55, "Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage." For example, 10CFR 73.55 includes requirements that:

- o Vital equipment shall be located only within vital areas, which in turn, shall be located within a protected area such that access to vital equipment requires passage through at least two physical barriers of sufficient strength.

- o Walls, doors, ceiling, floor, and any windows in the walls and in the doors of the reactor control room shall be bullet-resisting.

More recently the subject of sabotage protection has been highly visible with the NRC and ACRS in relation to new plant designs. The "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation" indicates that the NRC expects applicants for design approvals (to be used beyond 1985) to address sabotage in their designs. The NRC is looking for design considerations that will inhibit sabotage and that do not increase the risk of nuclear accidents from other causes.

The WAPWR design will incorporate several features which should provide improved protection against industrial sabotage. These features include safeguards fluid system designs with reduced or eliminated interconnections, reduced or eliminated normal operation functions, improved redundancy and diversity, and improved plant layout. Also, 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. This layout allows improved control of access to vital areas and also allows free access to most normally operating equipment.

WAPWR Response

The WAPWR design will be in accordance with the provisions of 10CFR 73.55 (e.g., physical barriers).

In addition, Westinghouse will perform a sabotage assessment for the WAPWR design using risk models. Basically this assessment will compare the WAPWR design with another design in relation to inherent sabotage protection and the capability of the design to handle or recover from a successful act of sabotage. It is intended that this information be provided to utilities utilizing the WAPWR design for appropriate consideration in their physical protection plans.

30. Issue A-30: Adequacy of Safety-Related DC Power Supplies

Discussion

This generic task originated from a letter to the ACRS from one of its consultants that questioned the reliability of DC power supplies at nuclear power stations. The specific concern expressed was as follows:

> "While a nuclear power plant is operating, one of two redundant DC power supply systems fails causing a reactor scram and subsequently causing loss of all offsite power. At this point, safe shutdown of the plant requires that the residual heat from the decay of radioactivity be removed from the reactor. Control of valve position and pumps needed to remove residual heat after plant shutdown depends on availability of the DC power supply. If all remaining sources of DC power were lost, continued cooling of the reactor core cannot be assured."

The NRC view is that the simultaneous and independent failure of redundant DC power supplies is so unlikely as to be incredible and that their failure from a common event is judged to be low enough in likelihood that adequate protection of the public health presently exists, but that additional technical studies to be provided as part of this task should and will be performed to add confidence to this judgment. This view stems from the following: (A) the postulated scenario is highly unlikely, (B) the period of vulnerability to the above cited single failure of the redundant DC power supply is limited (i.e., both the DC power supply failure initiating the scenario, and the second failure of the remaining source of DC power must occur within 30 seconds to defeat starting of the redundant diesel and acceptance of critical loads), and (C) the degree of vulnerability is mitigated substantially by the availability of alternative measures for restoration of power or for removal of decay heat and of sufficient time (at least 1 hour) for operator implementation of these alternative measures.

A more detailed discussion of the design of DC power supply systems and of the NRC view on the postulated accident scenario described above is provided in NUREG-0305, "Technical Report on DC Power Supplies in Nuclear Power Plants."

The NRC considers this issue as technically resolved with the issuance of NUREG-0666, "A Probabilistic Safety Analysis of DC Power Supply Requirements for Nuclear Power Plants." This NUREG concluded that minimum requirements for DC power systems should be augmented with the following:

- o Assure that design and operational features of the DC power supplies used for shutdown cooling do not compromise division independence. This includes eliminating use of a bus tie breaker, if provided, and revising test and maintenance activities with the potential for human error causing more than one DC division to be unavailable. Specific administrative control and procedures should be provided where the human factor is involved.

- o Assure that test and maintenance activities required for battery operability also include preventive maintenance on bus connections, procedures to demonstrate DC power availability from the battery to the bus, and administrative controls to reduce the likelihood of battery damage during testing, maintenance, and charging.

- Stagger test and maintenance activities and crews to the extent practicable. This should include weekly pilot cell observations, preventive maintenance on batteries and bus connections, battery discharge and load tests, battery charger maintenance, and off line battery charging.

- Assure that plant design and operational features are such that following the loss of one DC power supply or bus: (A) redundant capability is maintained for providing shutdown cooling in the hot standby condition, (B) reactor coolant system integrity and isolation capability are maintained, and (C) operating procedures, instrumentation, and control functions are adequate to initiate and maintain shutdown cooling in the hot standby condition. In essence, reactor core cooling capability should be maintained following the loss of any one DC power supply or bus and a single independent failure in any other system required for shutdown cooling.

WAPWR Response

Westinghouse will consider the NUREG-0666 DC power supply requirements during the licensing process for the WAPWR design.

31. Issue A-31: Residual Heat Removal Requirements

Discussion

This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

32. Issue A-32: Missile Effects

Discussion

This issue addresses three types of missiles for which impact effects on nuclear power plant structures, systems, and components important to safety must be evaluated. These missiles are also addressed in Issue A-38, "Tornado Missiles;" Issue A-37, "Turbine Missiles;" and for the most energetic accident-induced missile, Issue B-68, "PWR Pump Overspeed During LOCA." Refer to the discussion of these issues.

33. Issue A-33: NEPA Reviews of Accident Risks

Discussion

In 1971 the AEC determined that, consistent with the National Environmental Policy Act, the environmental assessments of requests for construction permits and operating licenses should include consideration of the possible impacts from accidents. An Annex to 10CFR Part 50, Appendix C, was proposed which provided guidance to applicants in this regard. Basically this Annex proposed to specify a set of standardized accident assumptions to be used in environmental reports submitted by applicants for construction permits or operating licenses. It also included a system for classifying accidents according to a graded scale of severity and probability of occurrence. Nine classes of accidents were defined, ranging from trivial to very serious. It directed that "for each class, except classes 1 and 9, the environmental consequences shall be evaluated as indicated." Class 1 events were not to be considered because of their trivial consequences and class 9 events were not to be considered because of their low probability.

The purpose of this generic task was for the NRC to conduct limited additional analyses and prepare a summary survey document which could be used as a standard reference regarding accident risks in the context of the NRC environmental reviews. This same document was intended to serve as the principal basis for a decision regarding finalizing the proposed Annex to 10CFR Part 50, Appendix D.

On June 13, 1980 the Commission published in the Federal Register a statement of interim policy regarding accident considerations. This statement withdrew the proposed Annex to Appendix D of 10CFR Part 50 and suspended the rulemaking proceedings associated with it. It also put forth the Commissions's interim policy that: ". . .Environmental Impact Statements shall include considerations of the site-specific accident sequences that lead to releases of radiation and/or radioactive materials, including sequences that can result in inadequate cooling of reactor fuel and to melting of the reactor core. In this regard, attention shall be given both to the probability of occurrence of such releases and to the environmental consequences of such releases."

This interim policy is considered by the NRC as the technical resolution to this issue.

WAPWR Response

This issue and its resolution is associated with evaluating accidents in the context of environmental reviews of nuclear power plants and accordingly, would be addressed by each utility using the WAPWR design as part of its environmental impact statement, and as such it is not applicable to Westinghouse in relation to the WAPWR design.

34. Issue A-34: Instruments for Monitoring Radiation and Process Variables During Accidents

Discussion

The purpose of this task was for the NRC to develop criteria and guidelines to be used by applicants, licensees, and NRC staff reviewers to support implementation of Regulatory Guide 1.97, Revision 1, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident."

The NRC considers this issue as technically resolved with the issuance of Revision 2 to Regulatory Guide 1.97.

WAPWR Response

Regulatory Guide 1.97, Revision 2, in relation to the WAPWR design is discussed in Section 3.1 (item 23)

35. Issue A-35: Adequacy of Offsite Power Systems

Discussion

The NRC requires that electric power for safety systems be comprised of two redundant and independent divisions, each capable of providing the necessary plant protection functions during all normal operating conditions and following various design basis accidents. Each division includes an offsite AC power connection (the preferred power source), a standby emergency diesel generator AC power supply (capable of powering essential safety systems should the offsite source be lost), and DC power sources.

Events at several plants involving the loss or degradation of the offsite power system or involving its connection to the emergency onsite power system have indicated that a reassessment of current NRC requirements was appropriate. This task was undertaken by the NRC to perform such an assessment and to determine the need, if any, for upgrading the offsite power sources and/or their interfaces with the onsite power system at nuclear power stations. The issue in relation to the Millstone Unit 2 event is discussed in Section 6.5 (item 20).

The NRC considers this issue as technically resolved with the issuance of the Standard Review Plan 8.3.1, "A-C Power Systems (Onsite)," acceptance criteria.

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 8.3.1 acceptance criteria during the licensing process for the WAPWR design.

36. Issue A-36: Control of Heavy Loads Near Spent Fuel

Discussion

This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

37. Issue A-37: Turbine Missiles

Discussion

Protection of essential systems from turbine missiles is required by the NRC staff unless the combined generation, strike and damage probability is very small. For most new plants, adequate protection against turbine missiles is provided by favorable turbine placement and orientation and adherence to the guidelines of Regulatory Guide 1.115, Revision 1, "Protection Against Low-Trajectory Turbine Missiles." For plants that have safety-related structures, systems, and components that are potentially susceptible to turbine missile strikes because of unfavorable turbine placement for example, a more detailed evaluation of turbine missile protection is required. Currently, each such plant is reviewed on a case-by-case basis to assure that the probability of unacceptable damage is acceptable or, if not, that appropriate measures are taken to reduce this probability.

The purpose of this generic task is for the NRC to assess the methods currently used to estimate the probability of damage to essential systems used in these case-by-case reviews, to quantify the effect of steps that can be taken by applicants to reduce the damage probability, and to recommend means of assuring that the probability of unacceptable damage is sufficiently small. Although this task is intended to provide a more uniform review by providing better guidance to NRC reviewers and applicants, the currently used case-by-case methods are sufficiently conservative to assure adequate protection of the public health and safety.

More recently, it has been identified that for Westinghouse designed turbines, at normal operating speed the more likely failure mechanism would lead to an increase in the historically observed frequency of disc cracks, the imposition of a periodic ultrasonic inspection should leave the historically observed failure/missile frequency unchanged.

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Regulatory Guide 1.115 positions during the licensing process for the WAPWR design.

38. Issue A-38: Tornado Missiles

Discussion

General Design Criteria 2 and 4 of 10CFR Part 50, Appendix A, require in part that structures, systems, and components important to safety be designed to be able to withstand the effects of tornado missiles. A missile generated by a tornado may be energetic enough to cause damage to improperly protected systems or components. This damage may ultimately result in the release of radioactivity to the environment. This design requirement imposed new demands on the practice of structural engineering, that is, for other types of facilities, tornadoes have always been considered too rare an event to be included in the design basis. Consequently, no body of design practice existed and design criteria for tornado resistance had to be developed. The first NRC requirements were published in Standard Review Plan 3.5.1.4, "Missiles Generated by Natural Phenomena," in 1975 and revised in 1976.

Since 1976, Standard Review Plans 3.3.2, "Tornado Loadings," and 3.5.1.4, "Missiles Generated by Natural Phenomena," have been revised and Regulatory Guides 1.76, "Design Basis Tornado for Nuclear Power Plants," and 1.117, "Tornado Design Classification," have been issued and/or revised. In these documents the NRC details specific design acceptance criteria to meet the requirements of General Design Criteria 2 and 4 and recommends methods of satisfying the acceptance criteria.

The purpose of this task is not for the NRC to investigate new possibilities to increase plant safety but to refine the spectrum of possible tornado missiles. The NRC's judgment was that postulated missile velocities, size, and orientation used in the plant safety analysis are more conservative than tornado damage histories would warrant.

The end product of this generic issue was to be a set of design basis missiles that does not impose unnecessary design requirements on plant construction and for which a sound technical basis exists.

WAPWR Response

Current NRC regulations and regulatory guidance will be utilized in the WAPWR design in relation to tornado missiles. Westinghouse will completely document and justify any deviations from the NRC acceptance criteria during the licensing process for the WAPWR design.

39. Issue A-39: Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits for BWR Containments

Discussion

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

40. Issue A-40: Seismic Design Criteria - Short Term Program

Discussion

This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

41. Issue A-41: Seismic Design Criteria - Long Term Program

Discussion

This issue involves long term research programs on seismic design. In this regard, the NRC has established a Seismic Safety Margins Research Program which is basically intended to quantify how much seismic margin is available for various components in current operating plant designs. This quantification is intended to be used to develop probability models that could assess the impact of seismic events much larger than the current safe shutdown earthquake design basis.

WAPWR Response

The Westinghouse practice of generic seismic level qualification has, in general, resulted in additional seismic safety margins in Westinghouse equipment. Westinghouse anticipates no hardware impact on the WAPWR design as a result of this issue.

42. Issue A-42: Pipe Cracks in Boiling Water Reactors

Discussion

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

43. Issue A-43: Containment Emergency Sump Performance

Discussion

This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

44. Issue A-44: Station Blackout

Discussion

This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

45. Issue A-45: Shutdown Decay Heat Removal Requirements

Discussion

This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

46. Issue A-46: Seismic Qualification of Equipment in Operating Plants

Discussion

This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

47. Issue A-47: Safety Implications of Control Systems

Discussion

This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

48. Issue A-48: Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Discussion

This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

49. Issue A-49: Pressurized Thermal Shock

Discussion

This issue is identified as an Unresolved Safety Issue. Refer to Section 4.0 for a discussion of Unresolved Safety Issues.

5.2 CATEGORY B ISSUES

The following discussions pertain to current Category B issues in relation to the WAPWR design. NRC discussions and descriptions of these issues are contained in NUREG-0471, "Generic Task Problem Descriptions (Category B, C, and D Tasks)" and NUREG-0933, "A Prioritization of Generic Safety Issues."

1. Issue B-1: Environmental Technical Specifications

   Discussion

   Current NRC regulations and practice require that certain operating requirements, Technical Specifications, be made part of each operating license. The nonradiological portion of Appendix B to the operating license traditionally derives from information in the Final Environmental Statement and other relevant sources. Based on several years of NRC experience with facility licensing and a better understanding of Environmental Protection Agency and NRC responsibilities in the area of water quality regulation, it is believed that the development of Standardized Environmental Technical Specifications (SETS) is appropriate. SETS are intended to result in more efficient use of NRC and applicant resources and more uniform requirements and performance standards for licensees. The NRC intends that this task results in the development of Standardized Environmental Technical Specifications to be published as a NUREG report or as part of Regulatory Guide 4.8, "Environmental Technical Specifications for Nuclear Power Plants." SETS are being prepared on a case-by-case basis. The NRC considers this issue resolved.

   WAPWR Response

   This issue and its resolution is associated with site specific environmental technical specification guidance and accordingly, it is not applicable to Westinghouse in relation to the WAPWR design. Environmental technical specifications are the responsibility of each utility utilizing the WAPWR design.

2. Issue B-2: Forecasting Electricity Demand

Discussion

Originally, this issue was directed at improving the NRC's capability to forecast electricity demand for the purpose of evaluating an applicant's need for power forecasts in individual licensing cases.

As discussed in some detail in Section 5.1 (item 20), the NRC has recently revised their regulations to no longer require that the issue of "need for power" be addressed in operating license proceedings.

As a matter of policy, the Commission endorses placing substantial reliance on state assessments of need for power, energy conservation, and alternative energy source analyses to fulfill the NRC's National Environmental Policy Act responsibilities at the construction permit stage and has initiated the development of procedures for soliciting this input.

This Environmental issue has been resolved with the publication of the following documents: (A) Regulatory Guide 4.1, Rev. 2, Chapter 1 on "Purpose of the Proposed Facility and Associated Transmission," July 1976; (B) NUREG/CR-0022 on "Need for Power: Determination in the State Decision Making Process," March 1978; (C) NUREG/CR-0250 on "Regional Econometric Model for Forecasting Electricity Demand by Sector and State," September 1978; (D) Section 8 of NUREG-0555 on "Environmental Standard Review Plans for the Environmental Review of Construction Permit Applications for Nuclear Power Plants," May 1979; (E) Part III of March 1980; (F) ORNL/TM-7947 on "An Integrated System for Forecasting Electric Energy and Load for States and Utility Service Areas," May 1982; and (G) NUREG-0942 on "Conducting Need-for-Power Review for Nuclear Power Plants: Guidelines to States," draft report of December 1982.

WAPWR Response

The need for power issue is associated with an environmental proceedings task that does not affect plant safety nor impact Westinghouse in relation to the WAPWR design.

3. Issue B-3: Event Categorization

Discussion

There are several inconsistencies in event categorization between the NRC General Design Criteria, Standard Review Plan, standard format and content guide, and applicant submittals. In addition, categorization by other groups such as ANSI and ANS is not always consistent with NRC positions. In several cases, applicants have proposed that certain events be categorized as accidents (which would permit limited fuel damage) whereas the NRC categorizes them as anticipated transients. The purpose of this task is for the NRC to categorize postulated transients and accidents and define acceptance criteria for the various categories. The resulting categorizations and acceptance criteria are intended to improve the licensing process and provide possible relief from current restrictive requirements for some licensees.

Westinghouse has long considered that event categorization for application to nuclear power plants should be the responsibility of the American Nuclear Society (ANS). ANS has just completed the development of new twin standards, i.e., "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants", ANSI/ANS-51.1-1983 and "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants", ANSI/ANS-52.1-1983. They replace ANSI N18.2-1973 and ANSI/ANS-52.1-1978 having like titles. These standards include event categorizations, providing five "Plant Conditions", designed to fit with existing NRC rules and regulations, replacing the four "Conditions of Design" previously used. The extra, Plant Conditions 2, was specifically created to match, as nearly as possible, the catgory "anticipated transients" used by the

NRC. Events are categorized on the basis of best-estimate frequencies of occurrence as modified by combinations with other events, using methodology set forth in Table 3-4 of the twins standards. Other facets given for Plant Conditions: examples of each category in Table 3-3, acceptance criteria in Table 3-2, and dose criteria in Table 3-1. Requirements associated with Plant Conditions are treated in Sections 3.2, including those for application of the single-failure criterion (the overall bases), coincident occurrences, multiple failures, common-cause failures, operator actions and human errors.

WAPWR Response

Westinghouse is considering utilizing the event categorization of draft ANS-51.1 for the WAPWR design and licensing activities. That is:

- o Plant Condition 1: planned operations
- o Plant Condition 2: $F \geq 10^{-1}$
- o Plant Condition 3: $10^{-1} > F \geq 10^{-2}$
- o Plant Condition 4: $10^{-2} > F \geq 10^{-4}$
- o Plant Condition 5: $10^{-4} > F \geq 10^{-6}$

Where F is defined as the best estimate frequency of occurrence per reactor year.

4. Issue B-4: ECCS Reliability

Discussion

This issue has been superseded by Item II.E.2.1 of NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident." Refer to Section 3.3.2 (item 9) for a discussion of Item II.E.2.1 of NUREG-0660.

5. Issue B-5: Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments

Discussion

Ductility of Two-Way Slabs and Shells: This task involved NRC development of a more dependable and realistic procedure for evaluating the design adequacy of Category I reinforced concrete slabs subject to a postulated loss-of-coolant accident or high energy pipe break.

More specifically, the NRC intended to determine with sufficient accuracy the influence of biaxial membrane tension on the resistance function and the permissible ductility ratio of two-way slabs loaded in flexure and shear. Since the response of the slab to the postulated loading conditions will likely be in the nonlinear range because of the simultaneous application of the severe, time dependent pressure load and concentrated jet force, the analysis performed must encompass the nonlinear range. This NRC task investigated the following specific items:

o A summary of the existing state-of-the-art on the subject resulting from a literature search.

o The relationship between ductility of one-way slabs and two-way slabs.

o The ductility of two-way slabs under shear and flexure separately and under combined loading conditions, including the biaxial membrane tensile force.

o Recommendations relative to avoidance of shear failure that could be utilized in practical design applications.

o A comparison of solutions obtained by analytical methods with applicable tests performed on two-way slabs.

The NRC has concluded that there is sufficient information pertaining to the design of two-way slabs subjected to dynamic loads and biaxial tension to enable a reasonably accurate analysis. Thus, this portion of issue B-5 is considered resolved.

Buckling Behavior of Steel Containments: The structural design of a steel containment vessel subjected to unsymmetrical dynamic loadings may be governed by the instability of the shell. For this type of loading, the current design verification methods, analytical techniques, and the acceptance criteria may not be as comprehensive as they should be. Section III of the ASME Code does not provide detailed guidance on the treatment of buckling of steel containment vessels for such loading conditions. Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components," recommends a minimum factor of safety of 2.0 against buckling for the worst loading condition provided a detailed rigorous analysis, considering inelastic behavior, is performed. On the other hand, the 1977 Summer Addenda of the ASME Code permits three alternate methods, but requires a factor of safety between 2.0 and 3.0 against buckling depending upon the applicable service limits.

At present, the NRC has developed and is using a set of interim criteria for evaluating steel containment buckling for plants undergoing operating license review. NRC investigation into this item is continuing. This item is currently considered to be medium priority by the staff, with considerable uncertainty.

WAPWR Response

The interim criteria discussed above will be considered during the WAPWR containment design. In addition, Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 6.2.1 acceptance criteria during the licensing process for the WAPWR design.

6. Issue B-6: Loads, Load Combinations, Stress Limits

Discussion

The designer of pressure vessels and piping system components must consider (A) the individual and combined loads that will act on each component due to normal operating conditions, system transients, and postulated low probability events (accidents and natural phenomena), and (B) the stress limits to be used in evaluating structural integrity and component operability when subject to these loads.

The work effort to investigate and establish a position on dynamic response combination methodology was completed and reported in NUREG-0484, Rev. 1, "Methodology for Combining Dynamic Responses". The conclusions in this report have been incorporated into the latest version of SRP 3.9.3. In addition, work has been completed on an evaluation of the loads and load combinations for containment structures. The only work remaining is research into decoupling the LOCA and SSE events. Reports on two investigations addressing this issue have been released as NUREG/CR-2136, "Effects of Postulated Event Devices on Normal Operation of Piping Systems in Nuclear Power Plants," and NUREG/CR-2189, "Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant."

The purpose of this task is for the NRC to provide guidance on load combination methods and acceptable stress limits.

The NRC considers this issue as being technically resolved with the issuance of the latest version of Standard Review Plan 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures."

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 3.9.3 acceptance criteria during the licensing process for the WAPWR design.

7. Issue B-7: Secondary Accident Consequence Modeling

Discussion

The NRC intends to develop more reliable models and associated computer capability than currently available to the NRC for assessing the radiological consequences of accidents that could result in the release of radioactivity through secondary systems. The objective is to determine if these types of accidents could be bounded by a simple source term thus precluding plant specific analyses.

The events resulting in the most significant release of radioactivity through the secondary systems are

- Steamline break with a subsequent small-break loss-of-coolant accident resulting from failure of a partially degraded steam generator tube(s).

- Steamline break with a subsequent small-break loss-of-coolant accident (other than a steam generator tube rupture) resulting from any of the following:

  - stuck-open power-operated relief valve,
  - safety valve actuated during the primary system transient,
  - pipe whip,
  - jet impingement from the broken steamline.

This issue is encompassed by Uncategorized Issue 18, "Steamline Break with Consequential Small LOCA" (refer to Section 5.5, item 18).

8. Issue B-8: Locking Out of ECCS Power-Operated Valves

Discussion

The physical locking out of electrical sources to specific motor-operated valves required in the engineered safety functions of emergency core cooling systems has been required by the NRC, based on the assumption that a spurious electrical signal at an inopportune time could activate the valves to the adverse position (e.g., closed rather than open, or opened rather than closed). While such an event has a finite probability, another probability exists that the valves might be adversely positioned due to operator error.

This task involves a reevaluation of the NRC requirement using a systems approach, and considering such items as (A) the evaluation of the probability of a spurious signal, (B) the time required to reactivate the valve operator, (C) the status of signal lights when the circuit breaker is open, (D) can the valve be locked out in an improper position due to a faulty indicator, (E) are there other designs improving reliablity without lock-out and (G) what are the advantages and disadvantages of corrective action by an alert operator in case of incorrect positioning vis-a-vis a system with power locked out.

Historically, Westinghouse has argued (based on WCAP-9207, "Evaluation of Mispositioned ECCS Valves") that spurious movement of a motor-operated valve due to an electrical fault in the motor actuation circuitry, coincident with a loss-of-coolant accident, is an acceptably low probability event. This, coupled with valve position control room indication as well as periodic visual inspection and operability testing of the valves, should preclude the need for the physical locking out of the electrical source to these valves. Nevertheless, the NRC has been requiring physical lock-out in accordance with Branch Technical Position ICSB 18, "Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves."

It is the current NRC position that locking out ECCS power operated valves provides an acceptable approach to meet the single failure criterion required by 10CFR 50. The NRC further recommends that until a quantifiable reduction in risk to the public or a significant cost savings to industry can be ascertained, as a result of an alternate solution to the current ECCS lock-out position, this issue should be dropped from further consideration as a safety issue.

Additional NRC acceptance criteria in this area are documented in Branch Technical Position RSB 6-1, "Piping from the RWST (or BWST) and Containment Sump(s) to the Safety Injection Pumps." Specifically for new plant designs the NRC acceptance criteria are intended to eliminate the need for various schemes such as locking out power to valves to ensure no interruption of water supplies to emergency core cooling pumps.

WAPWR Response

The Integrated Safeguards System of the WAPWR consists of four independent high head ECCS pumps, core reflood tanks, accumulators, and containment spray pump subsystems. Therefore, the consequences of a spurious electrical signal causing a single valve to move to an adverse position has been greatly diminished, since it can at most, only affect one of four subsystems. However, it is still envisioned that power lockout will be employed for certain valves, but restoration of power to these valves can be accomplished from the main control room.

9. Issue B-9: Electrical Cable Penetrations of Containment

Discussion

The purpose of this task was for the NRC to reevaluate current licensing criteria for the design and qualification testing of electrical penetrations in the reactor containment in light of concerns raised by these failures. Some prototype electrical penetration failures occurred in both high- and low-voltage penetration modules at licensed facilities. It was originally postulated that the failures of the low-voltage penetration modules were due to electrical short circuits caused by collection of moisture in fissures (cracks) in the epoxy insulator-sealant. However, results of the laboratory analysis indicated that the failures were caused by heating of the conductors at the connection splices within the penetration module. The heating resulted from high contact resistance due to epoxy intrusion into an area of connector splices that were not insulated during the manufacturing process. The accumulation of carbon deposits over a period of time, resulting from the heating process, created a conductive path (short circuit) between adjacent conductors in the penetration modules.

Existing requirements in IEEE Standard 317-1976, "Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," and Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light-Water Cooled Nuclear Power Plants" provide adequate direction for the design of containment electrical cable penetrations. Thus, the NRC considers this issue technically resolved.

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC positions of Regulatory Guide 1.63 (which endorses IEEE Standard 317-1976) during the licensing process for the WAPWR design.

10. Issue B-10: Behavior of BWR Mark III Containment

Discussion

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

11. Issue B-11: Subcompartment Standard Problems

Discussion

The calculations of differential pressures that occur in containment subcompartments from a loss-of-coolant event require a complex fluid dynamic analysis to assure that the subcompartment design pressures are not exceeded. To check the various industry computer codes used for the analyses, the NRC has issued a standard problem to the reactor vendors and architect engineers so that their models and calculational methods can be evaluated. This task, now complete, involved the NRC review and evaluation of the subcompartment standard problem analyses supplied by vendors and architect engineers to determine the validity of their models. Standard Review Plan 6.2.1.2, "Subcompartment Analysis," provides current NRC acceptance criteria for containment subcompartment analyses.

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 6.2.1.2 acceptance criteria during the licensing process for the WAPWR design.

12. Issue B-12: Containment Cooling Requirements (Non-LOCA)

Discussion

The rationale for normal and postaccident containment cooling has been reviewed by the NRC to determine the adequacy of the design requirements imposed on the containment ventilation systems. By reviewing typical designs the NRC developed a basic understanding of the consequences of a loss of normal containment cooling, including the impact, if any, on the operability of safety systems and control systems. Specifically, the purpose of this task was to establish whether or not (A) the normal ventilation system is essential to achieve a safe cold shutdown, (B) a failure in the system could cause an accident, and (C) the system is required to mitigate accidents.

The NRC considers this issue as being technically resolved and their current acceptance criteria are documented in Standard Review Plan 6.2.2, "Containment Heat Removal Systems."

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 6.2.2 acceptance criteria during the licensing process for the WAPWR design.

13. Issue B-13: Marviken Test Data Evaluations

Discussion

Test data from the Marviken containment tests have been obtained by the NRC for the purpose of validating containment pressure codes currently used for performing independent calculations related to licensing reviews. The Marviken data are containment pressure responses from a full-scale blowdown using a pressure suppression type containment. This task, now complete, correlated the Marviken data and compared the results with existing computer programs.

The NRC considers this issue as being technically resolved and Standard Review Plan 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containments," provides acceptance criteria for the containment response (e.g., pressure and temperature) as a result of a postulated loss-of-coolant accident and secondary system steam and feedwater line breaks.

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 6.2.1.1.A acceptance criteria during the licensing process for the WAPWR design.

14. Issue B-14: Study of Hydrogen Mixing Capability in Containment Post-LOCA

Discussion

This issue is included as part of Unresolved Safety Issue A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment" (refer to Section 4.0, item 26).

15. Issue B-15: CONTEMPT Computer Code Maintenance

Discussion

The CONTEMPT computer code is used by the NRC staff to perform independent containment analyses. This task involves the maintenance and revision of the CONTEMPT code to accommodate new containment designs or new problem areas as they are defined.

WAPWR Response

This issue applies to an ongoing NRC administrative activity and has no impact on the WAPWR design.

16. Issue -16: Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

Discussion

This issue has been incorporated as part of Issue A-18, "Pipe Rupture Design Criteria" (refer to Section 5.1, item 18).

17. Issue B-17: Criteria for Safety-Related Operator Actions

Discussion

Current plant designs are such that reliance on the operator to take action in response to certain transients is necessary. In addition, some current pressurized water reactor designs require manual operations to accomplish the switchover from the injection mode to the recirculation mode following a loss-of-coolant accident. The realignment operations must be accomplished before the inventory in the refueling water storage tank is depleted.

The NRC plans to develop a time criterion for safety-related operator actions including a determination of whether or not automatic emergency core cooling system realignment will be required.

The ANS is currently developing a new standard (i.e., ANS-58.8, "Time Response Design Criteria for Safety-Related Operator Actions") to provide guidance in determining the time response requirements that are acceptable for relying on operator actions to mitigate the consequences of design basis events in nuclear power plants. Westinghouse endorses and is participating in the development of ANS58.8 (also referred to as ANSI N660).

This issue has been superseded by TMI Action Plan Items I.A and I.C (NUREG-0660). Following the conclusion of the safety-related operator actions criteria development efforts from TMI Action Plan item I.A.4.2, a more rigorous analysis has been suggested to reassess the value/impact associated with the adoption and implementation of specific safety-related operator actions requirements which are not currently available.

Specifically in relation to the development of the WAPWR, Westinghouse has established (as a design objective) a 30-minute time period before the operator is assumed to take any safety-related action to mitigate the consequences of most design basis events. In addition, the WAPWR design uses of an emergency water storage tank. The emergency water storage tank is inside containment and provides a continuous suction source for the high head pumps, thus eliminating the conventional realignment from the refueling water storage tank to the containment sump.

WAPWR Response

Westinghouse will completely document and justify assumed operator action times during the licensing process for the WAPWR design.

18. Issue B-18: Vortex Suppression Requirements for Containment Sumps

Discussion

This issue is included as part of Unresolved Safety Issue A-43, "Containment Emergency Sump Performance" (refer to Section 4.0, item 21).

19. Issue B-19: Thermal-Hydraulic Stability

Discussion

Demonstrating the thermal-hydraulic stability of a reactor is an essential element in the thermal-hydraulic design. Instabilities can result in fuel failures from premature departure from nucleate boiling or excessive hydraulic loads. This task involves the NRC development of the analytical methods to perform independent calculations to check vendor analyses of thermal-hydraulic stability.

Westinghouse has successfully demonstrated the inherent thermal-hydraulic stability of open-channel fuel assemblies similar in configuration to the WAPWR fuel by testing and analysis.

WAPWR Response

This issue applies to an ongoing NRC administrative activity. However, as part of the detailed design and licensing process, Westinghouse will demonstrate the thermal-hydraulic stability of the WAPWR reactor core by appropriate testing or analysis.

20. Issue B-20: Standard Problem Analysis

Discussion

Most vendors, in the conduct of internal audits of emergency core cooling performance computer codes, have discovered errors in coding and/or logic which have significant effects on the prediction results of approved models. This task involves the use of standard problems to evaluate the predictive accuracy of these complex computer codes and to detect errors to the extent that the errors affect the results of code predictions.

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.

21. Issue B-21: Core Physics

Discussion

The NRC has a variety of technical activities ongoing or planned related to core physics. For the most part these activities are directed at improving the NRC's analytical and computer capabilities for performing independent analyses related to such areas as reactor kinetics, predicting static core physics characteristics, core parameters for transient analyses, and departure from nucleate boiling. The purpose of this task is to coordinate all NRC staff reactor physics efforts into a single program with clearly defined objectives.

The NRC considers this issue as being a licensing issue and not a generic safety concern. This issue has been dropped from further NRC consideration.

WAPWR Response

This issue applies to a completed NRC administrative activity and has no impact on the WAPWR design.

22. Issue B-22: LWR Fuel

Discussion

Individual reactor fuel rods sometimes fail during normal operation, and many rods are calculated to fail during severe accidents releasing activity to the surroundings and providing a source for releases from the plant. Failures during some accidents could be severe enough to fragment the cladding and disperse fuel pellets into the coolant, but regulations require that the coolable rod-like geometry must be maintained. Behavioral characteristics, such as rod bowing and densification, also have an effect on plant-limiting conditions. Thus, fuel behavior during normal operation and postulated accidents must be predictable in order to set operating limits, to limit activity releases, and to ensure no more than acceptable degradation of the fuel system. The objective of this task is to assure that such predictions are reliable.

Standard Review Plan 4.2, "Fuel System Design," provides detailed NRC acceptance criteria for the design of fuel and core components.

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 4.2 acceptance criteria during the licensing process for the WAPWR design.

23. Issue B-23: LMFBR Fuel

Discussion

This issue involves NRC efforts related to the review of liquid metal fast breeder reactor (LMFBR) fuel designs and is not applicable to the WAPWR design.

24. Issue B-24: Seismic Qualification of Electrical and Mechanical Components

Discussion

This issue is included as part of Unresolved Safety Issue A-46, "Seismic Qualification of Equipment in Operating Plants" (refer to Section 4.0, item 24).

25. Issue B-25: Piping Benchmark Problems

Discussion

Applicants are required to provide confirmation of the adequacy of computer programs used in the structural analysis and design of piping systems and components. In the past this consisted of applicants providing (and the NRC reviewing) brief descriptions of the computer programs used and solutions to simple textbook problems. In order to better provide assurance or the reliability of these programs, this task involved the NRC development of benchmark problems (and solutions to these problems) for use in the review of applications for construction permits.

The results from this task were incorporated into Standard Review Plan 3.9.1, "Special Topics for Mechanical Components," which provides detailed acceptance criteria for demonstrating the applicability and validity of computer programs used in the structural analysis and design of piping systems and components.

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 3.9.1 acceptance criteria during the licensing process for the WAPWR design.

26. Issue B-26: Structural Integrity of Containment Penetrations

Discussion

Containment penetration assemblies provide a means to maintain the integrity of the containment pressure boundary and prevent overstressing of the penetration nozzle due to thermal stresses. A typical penetration assembly may consist of a flued head, a guard pipe, an expansion bellows and an inpingement ring. The flued head may be fabricated from a forging which may be welded into the process line or may be welded to the outer surface of the process piping. This task involves a NRC evaluation to assess the adequacy of specific containment penetration designs from the point of view of structural integrity and inservice inspection requirements.

Specifically, this NRC task involves two areas. The first (which is now considered complete) is an independently performed stress analysis by the NRC of the various penetrations produced as integral fittings and welded into the process line, or penetrations which are welded to the outside circumference of the process line. The model considered the applicable requirements of Section III, Subsections NC and NE, of the ASME Code, NRC stress criteria, any existing fabrication residual stresses, and the mechanical loadings resulting from normal plant operation, from postulated pipe breaks, and from seismic events. The second area involves a determination that the configuration and accessibility of the welds in the proposed design and the procedures proposed for performing volumetric examination will permit the inservice examination requirements of Section XI of the ASME Code to be met.

WAPWR Response

Standard Review Plans 6.2.1 through 6.2.7 specify NRC acceptance criteria for containment design. Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan acceptance criteria during the licensing process for the WAPWR design.

27. Issue B-27: Implementation and Use of Subsection NF

Discussion

Since the adoption by the ASME Code, Section III, of Subsection NF on component supports, NRC technical review has been limited to conformance of the information provided in the application and verification of a commitment by the applicants to component support design in accordance with the provisions in Subsection NF.

Certain deficiencies in the use of Subsection NF, however, have been identified by the NRC. These include:

- o The absence of definitive criteria to be used in defining the jurisdictional boundary between a load carrying building structure designed by AISC rules which do not contain inservice inspection requirements and an attached Subsection NF component support having inservice inspection requirements.

- o As the design limits for Class 1 linear type component supports presently appear in the ASME Code, the allowable stresses exceed those permitted for other ASME Code designed components. If these limits are approached repeatedly in the component support, the support could fail by fatigue.

The NRC plans to develop a Branch Technical Position that will assess deficiencies for use by the NRC in case reviews of component supports.

WAPWR Response

The design of the **W**APWR component supports (which will be documented during the licensing process for the **W**APWR design) will be performed with due consideration of the above mentioned deficiencies.

28. Issue B-28: Radionuclide/Sediment Transport Program

Discussion

As a result of Appendix I and the Liquid Pathway Generic Study (NUREG-0440), the NRC is taking a more realistic look at the effects of sediment (surface waters) and aquifer materials (groundwater) on radionuclide transport through the hydrosphere. To accomplish this objective, it is necessary that the NRC have available for its use radionuclide/sediment transport models that have been field verified. This task is intended to accomplish this objective through NRC radionuclide/sediment transport model development and verification.

The NRC considers this issue to be technically resolved with the issuance of NUREG/CR-2425, "Sediment and Radionuclide Transport in Rivers."

WAPWR Response

This item is concerned with internal NRC radionuclide/sediment transport model development and, as such, has no impact on the **W**APWR design.

29. Issue B-29: Effectiveness of Ultimate Heat Sinks

Discussion

This task involves the NRC confirmation of currently used mathematical models for prediction of ultimate heat sink performance by comparing model performance with field data and development of better guidance regarding the criteria for weather record selection to define ultimate heat sink design basis meteorology.

The NRC considers this issue to be technically resolved with the publication of three reports. NUREG-0693, "Analysis of Ultimate-Heat-Sink Cooling Ponds" and NUREG-0733, "Analysis of Ultimate Heat-Sink Spray Ponds," look at two sources of the ultimate-heat-sink (UHS) in use today, identifying methods that may be used to select the most severe combinations of controlling meteorological parameters for cooling ponds of conventional design. NUREG-0858, "Comparison Between Field Data and Ultimate Heat Sink Cooling-Pond and Spray-Pond Models" compares the results of the cooling pond and spray pond performances to the NRC model predictions in the former two reports.

WAPWR Response

The ultimate heat sink is plant specific and outside the scope of the WAPWR design. However, Westinghouse will develop interface criteria for use by applicants in establishing their ultimate heat sink design.

30. Issue B-30: Design Basis Floods and Probability

Discussion

The purpose of this task was for the NRC staff to prepare a paper for presentation to the Advisory Committee on Reactor Safeguards (ACRS) detailing the bases for design basis flood events used by the NRC staff in case reviews. Additionally, the task was to address the possible use of probability estimates for the principal flood producing events. This task has been completed and a report to the ACRS was issued in July 1977. The report presents discussion and definitions of flood events which may be used as Design Basis Floods for review of nuclear power plants. It supports continued use by the NRC staff of a deterministic approach for identifying the Design Basis Flood events in preference to possible use of a probabilistic approach. The deterministic approach identifies the upper limit of flood potential physically possible. As indicated in the report, the NRC does not feel that a probabilistic approach is appropriate for use in licensing reviews at the present time because of the lack of confidence in estimates of extreme flooding events using current techniques.

The preliminary results of the risk-based evaluation indicate that the probability of a flood-induced core meltdown accident at most sites is very low. However, ongoing research efforts aimed at developing improved methodological techniques for the probabilistic analysis of flooding are being undertaken by the NRC Office of Nuclear Regulatory Research.

Standard Review Plan 2.4.2, "Floods," provides NRC acceptance criteria to meet the hydraulic aspects of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and 10CFR Part 100, "Reactor Site Criteria." In addition, Regulatory Guide 1.29, "Seismic Design Classification," identifies the safety-related structures, systems, and components and Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," describes flood protection acceptable to the NRC to prevent the safety-related facilities from being adversely affected.

WAPWR Response

During the licensing process for the WAPWR design, Westinghouse will completely document and justify any deviations from the NRC regulatory positions and acceptance criteria of Regulatory Guides 1.29 and 1.102 and Standard Review Plan 2.4.2 for those safety-related facilities within the WAPWR scope.

31. Issue B-31: Dam Failure Model

Discussion

During licensing reviews, the need has arisen on several occasions to have a NRC model to predict the failure discharge hydrograph due to erosional failures of earthen dams. No known model presently exists for such evaluations and, accordingly, the NRC and the applicants have been forced to conservatively postulate complete and instantaneous failure of the dam. This NRC task is intended to develop an analytical model, or nomograph, to predict erosion rates and patterns of failure for an earthen enhancement for a given initiating mode (e.g., overtopping, cracking).

WAPWR Response

This item deals with internal NRC efforts related to the development of a dam failure model and, as such, has no impact on Westinghouse in relation to the WAPWR design.

32. Issue B-32: Ice Effects on Safety-Related Water Supplies

Discussion

The operating experience during some severe winters has identified physical phenomena which might adversely impact the proper operation of safety-related systems (i.e., the ultimate heat sink) and impair the ability to obtain sufficient cooling water to safely shut the plant down. Typical icing conditions (e.g., surface ice) appear less important than subsurface frazile ice as a flow blockage mechanism.

Pack ice on packed surface ice has, in the past, been assumed sufficiently porous to pass the relatively low flows necessary for ultimate heat sink operations. Frazile ice may not be as porous and may, under rare conditions, reduce the flow below acceptable levels. Also, forces produced by expanding ice sheets could damage safety-related equipment and structures and impair the ability of the ultimate heat sink to function. The purpose of this NRC task is to ensure that operating reactors have the ability to circulate warm water to the intake (or have other proces- ses) to limit ice buildup.

Current NRC acceptance criteria are provided in Standard Review Plan 2.4.7, "Ice Effects."

WAPWR Response

The ultimate heat sink is plant specific and outside the sccpe of the WAPWR design. However, Westinghouse will develop interface criteria for use by applicants in establishing their ultimate heat sink design.

33. Issue B-33: Dose Assessment Methodology

Discussion

This NRC task involves the maintenance and improvement of calculational capabilities for assessing doses to individuals from radiation and radioactive effluents from normal operation and from radioactive releases from postulated accidents. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I," provides methods acceptable to the NRC for assessing public exposure to radio- active materials and effluents.

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Regulatory Guide 1.109 positions during the licensing process for the WAPWR design.

34. Issue B-34: Occupational Radiation Exposure Reduction

Discussion

This NRC task involves the development of additional criteria and guidelines to provide an improved basis for the NRC staff to review reactor plant designs and operations to support full implementation of the regulatory requirement that radiation exposures should be maintained as low as is reasonably achievable (ALARA).

A preliminary NRC risk-based evaluation indicates that occupational radiation exposures at operating nuclear facilities are averaging roughly 400 man-rem per reactor year and have generally been increasing with time. Further, the expected value for the annual accident exposure associated with plants analyzed in WASH-1400, "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," is predicted to be approximately 250 man-rem per reactor year. Although it was recognized in the study that a meaningful comparison of the occupational exposure risks with those associated with accidents is difficult, the study concluded that reduction of occupational exposures can be very important to reducing the overall radiologically-associated risks associated with the nuclear reactor industry.

This assessment of the significance of occupational exposures in the preliminary risk-based evalution is consistent with the NRC's view of the importance of occupational radiation exposure reduction, as evidenced by the requirement to maintain such exposures ALARA. In this regard, general guidance is now available to the industry in Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As is Reasonably Achievable." This guidance has been utilized by the NRC in performing licensing reviews for a number of years. The NRC intends that Task B-34 will draw from that experience and, with the aid of supplementary studies, will develop additional criteria regarding techniques and methods to maintain occupational radiation exposures ALARA.

Although the preliminary risk-based evaluation was correct in that occupational radiaton exposures are important, current NRC requirements and review procedures assure that they will be maintained ALARA. This task may provide some improved guidance to designers and operators. Resolution of this issue will be accomplished through TMI Action Plan Item II.D.3.1, "Radiation Protection Plans."

WAPWR Response

Westinghouse has established a WAPWR design objective of maintaining occupational radiation exposures to less than the WASH-1400 value of 250 man-rem per year. This will be achieved through improvements in design which will improve plant availability and enhance inspectability and maintainability.

35. Issue B-35: Confirmation of Appendix I Models for "Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors"

Discussion

This NRC task involves the revision of models for calculating releases of radioactive materials to improve the accuracy of current NRC models for 10CFR Part 50, Appendix I, calculations.

All research programs described in the action plan have been completed except for the source term measurement program which is still underway and is due to be completed in FY1983 or FY1984 depending upon the availability of funding to support the collection of additional data from selected operating reactors. NUREG-0017, Revision 1, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from PWRs", once issued, will document the results of NRC efforts related to model enhancement.

WAPWR Response

This issue applies to an ongoing NRC administrative activity associated with their internal model development and has no impact on the WAPWR design.

36. Issue B-36: Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems

Discussion

This NRC task involved the development of revisions to current guidance and technical positions regarding engineered safety feature and normal ventilation system air filtration and adsorption units. The NRC considers this issue technically resolved with the issuance of Revision 2 to Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and Revision 1 to Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Regulatory Guide 1.52 and 1.140 positions during the licensing process for the WAPWR design.

37. Issue B-37: Chemical Discharges to Receiving Waters

Discussion

In accordance with NRC licensing responsibilities under the National Environmental Policy Act (NEPA), the NRC plans to assess the impact of discharges of chemicals to surface waters. The objective of this assessment is to afford a weighing of impacts of the proposed action and a comparison of alternative actions rather than to provide absolute protection to surface waters.

This task is intended to provide additional insight into the impact of chemical discharges and provide procedures for quantifying the magnitude of any such impacts. This improvement in NRC procedures for impact assessment should provide a clearer division between NRC responsibilities under NEPA and EPA responsibilities under the FWPCA.

There are three specific water quality effects which have been questioned and which will be studied initially by the NRC. These are

- o Environmental significance of condenser tube copper in cooling water discharges.
- o Impact of increased total dissolved solids in receiving waters.
- o Significance of chlorinated organic compounds produced during condenser chlorination.

The NRC resolution of this task is expected to be documented in a revision to Regulatory Guide 4.2, "Preparation of Environmental Reports for Nuclear Power Stations."

WAPWR Response

Plant specific environmental reports (including consideration of this issue) are the responsibility of each utility utilizing the WAPWR design.

38. Issue B-38: Reconnaissance Level Investigations

Discussion

NRC environmental information needs for licensing fall into the categories of (A) detailed site-specific investigations at a preferred site, and (B) reconnaissance level information to support alternative site assessment and selection, including early site review. The NRC intends to generate a technical report which would form the basis of an NRC position paper providing guidance to applicants concerning the need, applicability, proper utilization, scope, and content of an adequate reconnaissance level investigation. This guidance is needed because of the requirement for an applicant to demonstrate how environmental consid- erations were factored into the alternative site selection process, the emerging importance of early site reviews, and the efficiencies inherent in standardizing procedures used during the site selection process.

WAPWR Response

This issue involves NRC efforts toward providing standardized guidance to utilities for use in conducting site reconnaissance level investigations. As such, this issue is not applicable to Westinghouse in relation to the WAPWR design.

39. Issue B-39: Transmission Lines

Discussion

This task involves NRC participation in an interagency government effort to set forth practices for siting and managing transmission line corridors for the betterment of wildlife. Also, the NRC plans to participate with other agencies to develop a single environmental review process involving all transmission systems of joint concern.

WAPWR Response

Power distribution transmission lines are outside the scope of the WAPWR design.

40. Issue B-40: Effects of Power Plant Entrainment on Plankton

Discussion

The effects of entrainment on phytoplankton and zooplankton populations are often minimal and occasionally beneficial. Numerous studies of the effects of entrainment on plankton organisms, phytoplankton and zooplankton, have shown impacts to be minimal and/or not significant. Studies have also shown that even when entrainment mortality is high, the overall impacts may be minimal due to the fast reproductive and recovery time for many species (a few hours for some phytoplankters to several days for zooplankton).

In the past, utilities have undertaken exhaustive and sometimes unnecessary preoperational and operational environmental monitoring programs. In view of the above points, the NRC believes that it may be possible to reduce or eliminate studies of certain planktonic elements, perhaps on a site or regional basis. A NRC study of these matters is intended to form the basis for a NRC position on monitoring requirements of plankton and entrainment programs. If the state-of-the-art as defined in the study is adequate, perhaps intensive studies can be reduced, saving time and expense for both utilities and the NRC.

WAPWR Response

This issue is associated with site specific environmental considerations that are not relevant to Westinghouse in relation to the WAPWR design.

41. Issue B-41: Impacts on Fisheries

Discussion

This NRC task involves studies related to the impacts of power plant operations on fishery resources. Possible NRC studies to be undertaken include the following:

- Advanced modelling and field monitoring to evaluate the effects of plant operation on fishery resources.

- Sources of entrainment mortality during passage through condensor cooling systems.

- The potential for entrainment and impingement with canal cooling system intakes.

- The process of assessing and predicting potential impacts on aquatic systems from construction and operation.

WAPWR Response

This issue is associated with site specific environmental considerations that are not relevant to Westinghouse in relation to the WAWPR design.

42. Issue B-42: Socioeconomic Environmental Impacts

Discussion

As part of the cost-benefit analysis of nuclear power plant licensing applications the NRC is required to assess likely socioeconomic impacts of power plant construction and operation on local communities and the surrounding region. This task encompasses several studies to improve the NRC's ability to forecast socioeconomic impacts for preparation of environmental statements and hearing testimony. Areas to be studied include

- Nuclear power station construction labor force mobility and residential choices.

- Visual change within a region due to alternative closed cycle cooling systems and associated socioeconomic impacts.

- o Impacts of coastal and offshore nuclear generating stations on recreational and tourist behavior at adjacent coastal sites.

The NRC considers this issue to be technically resolved with the publication of NUREG/CR-2749," Socio-Economic Impacts of Nuclear Generating Stations," and NUREG/CR-2750, "Socio-Economic Impacts of Nuclear Generating Stations: Summary Report on the NRC Post-Licensing Studies."

WAPWR Response

This task is associated with a site specific environmental proceedings issue that is not applicable to Westinghouse in relation to the WAPWR design.

43. Issue B-43: Value of Aerial Photographs for Site Evaluation

Discussion

The technique of aerial photography has a long established and proven utility for earth resource inventory and evaluation. Applicants for nuclear construction permits are becoming aware of this and are making increasing use of aerial photographs in their environmental reports. The uncertainties with the methodology at present relate to (A) photo interpretation techniques and the extent to which existing regulatory guidance can be met using this method, (B) fine tuning of the interplay between aerial photography and ground truthing needed to meet licensing requirements, (C) quantification of presumed cost advantages of this method, and (D) relative information return from different films, photographic scales, and seasons of coverage. The NRC plans to examine existing regu- latory guidance and produce a list of items which might be fulfilled in whole or in part from aerial photographic information. Field tests on actual sites are planned to be carried out to determine the information return from photographs in relation to regulatory requirements and in relation to conventional ground based data collection efforts. The results are intended to give the NRC a documentary basis for accepting aerial photographic inventories and resource evaluation in environmental reports and for revising existing guidance for making environmental surveys.

Work on this issue has resulted in the publication of NUREG/CF-2861, "Image Analysis for Facility Siting: A Comparison of Low and High Altitude Image Interpretabilty for Land Use/Land Cover Mapping."

WAPWR Response

This issue is not directed toward affecting the level of safety, but toward improving the efficiency of environmental licensing reviews and therefore, has no impact on Westinghouse in relation to the WAPWR design.

44. Issue B-44: Forecasts of Generating Costs of Coal and Nuclear Plants

Discussion

In the performance of National Environmental Policy Act obligations to evaluate alternatives to the proposed action, the NRC must reach a conclusion as to the comparative costs of generating power among the feasible alternatives. While alternatives other than coal are treated in the NRC's analysis, coal represents by far the most feasible alternative and requires detailed cost comparison equivalent to those performed for nuclear. For several years, the NRC has used a computer code known as CONCEPT to obtain forecasts of plant capital costs. This task involves NRC maintainence of (and development of improvements to) the CONCEPT code so that it remains up-to-date for use in projections of power plant capital cost, front-end cost, and generating cost forecasts.

The NRC considers this issue to be technically resolved with the publication of ORNL-5470, "Concept-5 User's Manual" and ORNL/TM-6467, "A Procedure for Estimating Non-Fuel Operation and Maintenance Costs for Large Steam-Electric Power Plants."

WAPWR Response

This issue is associated with an ongoing NRC administrative activity in relation to environmental proceedings that does not affect plant safety nor impact Westinghouse in relation to the WAPWR design.

45. Issue B-45: Need for Power - Energy Conservation

Discussion

This issue is included as part of Issue B-2, "Forecasting Electricity Demand" (refer to item 2 above).

46. Issue B-46: Costs of Alternatives in Environmental Design

Discussion

Frequently, regulatory changes are made in the applicant's proposal for design and/or operation of systems or subsystems based on perceived needs to mitigate impacts on the environment. Also, differences in design and/or operation are an integral part of the NRC treatment of alternatives in Environmental Impact Statements.

The cost of such changes or alternatives, if calculated, are determined on an ad hoc basis. However, this cost is not always calculated, and many times they are not calculated on a consistent basis. The NRC believes more consistent and comprehensive analysis of the cost of var- ious design and operating modes is warranted so that there is a reason- able and documented rationale for determining such costs. Such costs would also have to include costs of redesign. Once experience is gained in this area, the NRC has indicated that consideration will be given to expanding the study to the cost of making changes because of changing safety criteria, both from a redesign standpoint as well as from a "backfit" point of view.

WAPWR Response

This issue is associated with an environmental concern that does not impact Westinghouse in relation to the WAPWR design.

47. Issue B-47: Inservice Inspection Criteria for Supports and Bolting of Class 1, 2, and 3 and MC Components

Discussion

Results from inspections of various structural components in the torus support systems of operating boiling water reactors have indicated several inconsistencies between the design drawings and the "as built" hardware, including missing support struts, out of tolerance weld dimensions, unwelded regions and unsupported columns. In addition, a limited number of separate inspections have been performed on pressurized water reactor steam generator supports. The results of these inspections revealed several cracked support bolts.

In view of the above, the NRC believes that additional investigation of boiling water reactor and pressurized water reactor component support systems should be undertaken to determine if similar deficiencies and "off design" conditions exist in operating plants. This investigation should determine the extent of support system deficiencies, and whether the deficiencies are service induced or are the result of faulty construction. Determination of the extent and nature of the deficiencies is necessary to define the possible safety significance and to provide guidance for further appropriate NRC staff action regarding inservice inspection of supports.

This task is partly covered by Unresolved Safety Issue A-12, "Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports" (refer to Section 4.0, item 12). Current NRC acceptance criteria in this area are included in Standard Review Plan 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures."

The NRC has concluded that the discrepancies between the support design drawings and the as-built support hardware, these are problems that related directly to the quality assurance program of the licensee and its contactors existing during construction and are not part of the inservice inspection program per se. Therefore, these deficiencies are failures to implement the QA program. No changes in QA criteria of requirements are indicated.

Further, with regard to the degradation of supports, the ASME Code, Section XI (1980 edition), addresses the matter of inservice inspection of component supports for Classes 1, 2, 3 and MC components (Subsection IWF) and contains the inservice requirements which appear to fully address the concerns in this issue. Moreover, the current effort under Item A-12 will result in a NUREG document in which guidance and requirements for the selection of materials and the construction of reactor coolant pump and steam generator support structures will be addressed. In addition, pre-service and inservice inspection requirements of these support structures for operating plants will also be addressed in this NUREG.

In view of the existing inservice inspection requirements for supports of Classes 1, 2, 3 and MC components and quality assurance program requirements, it appears that the concern in this issue are already being addressed and that no additional safety benefit could be expected from this issue as stated. Therefore, the NRC recommends that this item should be DROPPED from further consideration.

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 3.9.3 acceptance criteria during the licensing process for the WAPWR design.

48. Issue B-48: BWR Control Rod Drive Mechanical Failure

Discussion

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

49. Issue B-49: Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments

Discussion

General Design Criterion 53, "Provisions for Containment Testing and Inspection," requires, in part, that the reactor containment be designed to permit (A) periodic inspection of all important areas, and (B) an appropriate surveillance program. 10CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," requires a general inspection of the surfaces of the containment prior to any Type A test to uncover any evidence of structural deterioration.

Containment designs typically utilize any one of the following structural materials: steel, steel lined reinforced concrete, steel lined prestressed concrete. To date the only detailed criteria that have been developed for inservice inspection of containments relate to tendon surveillance for prestressed concrete containments. These criteria are contained in Regulatory Guides 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments," and 1.90, "Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons." These regulatory guides deal primarily with the prestressing hardware; no detailed inservice inspection criteria exist for the steel liner or other portions of the containment. Similarly, there are no criteria for inservice inspection of steel containments or steel lined reinforced concrete containments. In view of this, the NRC believes that detailed and comprehensive criteria need to be developed for performing inservice inspections of all types of containments.

In addition, the long-term corrosion problems of reinforcements and of the steel liner in contact with concrete in concrete containments, or the corrosion of the steel surface in contact with the water in boiling water reactor suppression chambers, have yet to be adequately analyzed. The NRC believes that long-term studies of these corrosion phenomena need to be undertaken to develop criteria and requirements to prevent corrosion in all types of containments.

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Regulatory Guide 1.35 and 1.90 positions during the licensing process for the WAPWR design.

50. Issue B-50: Post Operating-Basis-Earthquake Inspection

Discussion

Section V(a)(2) of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10CFR Part 100 states that licensees will be required to shut down their plants in the event of an earthquake if vibratory ground motion exceeds that of the operating basis earthquake (OBE). Prior to restart the licensee must demonstrate to the NRC that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public. In order to determine the capability of a plant to resume operation following an OBE, an adequate inspection of the plant and site area must be performed. The requirements for this post-OBE inspection are also stated in Standard Review Plan 3.7.4, "Seismic Instrumentation." However, since neither the regulations nor Standard Review Plan 3.7.4 provide details on the extent of such inspections, this NRC task is intended to develop an acceptable inspection procedure.

WAPWR Response

Procedures for performing post-OBE inspections are the responsibility of each utility utilizing the WAPWR design.

51. Issue B-51: Assessment of Inelastic Analysis Techniques for Equipment and Components

Discussion

In the design of nuclear power plants, inelastic response of equipment and components due to severe transients from low probability events is permitted in the ASME Code, Section III, Subsection NA, Appendix F. Local inelastic response is also permitted for structures under severe impact loads due to low probability events.

This task involves NRC activities to ensure that properly qualified analysis techniques are used, and that their limitations are properly understood. Resolution of this issue will be accomplished through Unresolved Safety Issues A-40, "Seismic Design Criteria Short-Term Program," (refer to Section 4.0, item 19).

WAPWR Response

Westinghouse performs faulted conditions analyses in accordance with the requirements of Appendix F of the ASME Code. As permitted by Appendix F, Westinghouse uses inelastic analysis in very limited applications and employs the latest state-of-the-art techniques in conjunction with the Appendix F rules to perform in-elastic analysis.

52. Issue B-52: Fuel Assembly Seismic and LOCA Responses

Discussion

This issue is included as part of Unresolved Safety Issue A-2, "Asymmetric Blowdown Loads on the Reactor Primary Coolant Systems" (refer to Section 4.0, item 2).

53. Issue B-53: Load Break Switch

Discussion

Plant designs which utilize generator load circuit breakers to satisfy the requirement for an immediate access circuit stated in General Design Criterion 17, "Electric Power Systems," must be prototype tested to demonstrate functional capability.

This task involves the preparation of a NRC position to clarify and document the prototype testing requirements for generator load circuit breakers and associated circuitry used to provide an immediate access circuit. The NRC technical position has been completed and has been incorporated into a revision to Standard Review Plan 8.2, "Offsite Power System" (FR 35201 dated August 3, 1983).

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 8.2 acceptance criteria during the licensing process for the WAPWR design.

54. Issue B-54: Ice Condenser Containments

Discussion

This task involves two NRC efforts associated with the ice condenser containment concept:

- o Verification of the established design margin for ice condenser containments using the NRC CONTEMPT 4 code.

- o Reviewing the surveillance programs for ice inventory and functional performance testing at operating facilities to determine whether the surveillance frequencies should be increased or other action should be taken.

WAPWR Response

The design of the WAPWR does not include an ice condenser containment. Therefore, this item is not applicable to the WAPWR design.

55. Issue B-55: Improved Reliability of Target Rock Safety-Relief Valves

Discussion

This issue is specifically concerned with the failure of Target Rock safety relief valves in BWRs, and as such has no apparent impact on the WAPWR design.

56. Issue B-56: Diesel Reliability

Discussion

An examination of licensee event reports by the NRC on the experience with diesel generators (1969 to 1975) indicated that the emergency onsite diesel generators at operating plants have an average reliability of about 0.94 compared with the NRC's reliability goal of 0.99. The reliability of the diesel generator is strongly dependent on the interaction of the following factors: design, testing and operational requirements, operational history, inspections, maintenance, and the personnel qualifications of operators.

The lack of detail regarding the failures reported in the licensee event reports has made it difficult for the NRC to establish the causes of the reported failures. The NRC believed a comprehensive review to determine the underlying and recurring causes of the reported failures was necessary in order to enable the NRC to establish improved guidance and requirements to increase the reliability of the emergency onsite diesel generators.

A proposed set of interim backfit requirements for operating plants have been developed by the NRC and encompass elements of Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants" (which includes the NRC's goal for new plants), and recommendations in NUREG/CR-0660, "Enhancement of On-Site Emergency Diesel Generator Reliability."

The proposed program establishes a graded set of requirements based on the reliability actually exhibited by diesel generators. The proposed program adopts a diesel generator startup reliability of 0.95/demand as the minimum desired reliability and 0.9/demand as the minimum acceptable level of reliability. At or below the minimum desired level, licensees would be required to improve their diesel generator reliability and document their program for doing so. Below the minimum acceptable level, licensees would be required to improve or repair diesel generators with reliability below the minimum acceptable level and perform a requalification program to demonstrate that the causes of the failures have been corrected. The requalification program is intended to pass diesel generators only if the reliability has been increased to 0.95/demand or greater.

The proposed interim program imposes a normal surveillance period of no more than 1 month. To increase assurance that a real change in reliability will be detected quickly, an increased test frequency would be required when two or more failures have been experienced on an individual diesel generator in the last 20 months. However, the frequency of tests and the anticipated duration of the accelerated test frequency are not as restrictive as currently recommended by Regulatory Guide 1.108.

An extended out-of-service period may, in many cases, be necessary to allow sufficient time to correct the problems that are causing low reliabilities. Therefore, the proposed program will allow out-of-service periods in excess of the current 72-hour limit, when necessary, while at the same time placing a yearly limit upon the cumulative time that a plant may operate in modes 1 through 4 with one of the diesel generators of the power systems inoperable. The cumulative limit would very depending upon the reliability of the inservice diesel generator with the lowest reliability.

Diesel reliability will also be a factor in the criteria associated with the resolution of Unresolved Safety Issues, A-44 and A-45 (see Section 4.0, items 22 and 23, respectively).

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 8.3.1 acceptance criteria during the licensing process for the WAPWR design.

57. Issue B-57: Station Blackout

Discussion

This issue has been reclassified as Unresolved Safety Issue A-44, "Station Blackout" (refer to Section 4.0, item 23).

58. Issue B-58: Passive Mechanical Failures

Discussion

This NRC task involves a review of valve failure data in a more systematic manner to (A) confirm the NRC's present judgment regarding the likelihood of passive mechanical valve failures, (B) categorize these and other valve failures as to expected frequency, (C) specify acceptance criteria, and (D) determine if and how the results of this effort should be applied in licensing reviews.

WAPWR Response

The failure of passive mechanical valves will be considered in the design of the WAPWR fluid systems. That is, safety systems will be capable of withstanding a single active failure or a passive failure at any time following an initiating event. However, passive failures which are considered to have a low probability (e.g., check valve failing to open) may not be considered.

59. Issue B-59: (N-1) Loop Operation in BWRs and PWRs

Discussion

The majority of operating boiling water reactors and pressurized water reactors are designed to operate with less than full reactor coolant flow. If a reactor coolant pump in a pressurized water reactor or a recirculation pump in a boiling water reactor becomes inoperative, the flow provided by the remaining (N-1) loops is sufficient for steady-state operation at a power level less than full power. Although safety analysis reports for the licensed plants present (N-1) loop calculations showing allowable power and protective system trip set points, the NRC has disallowed this mode of operation for most plants primarily due to insufficient emergency core cooling analyses.

The purpose of this NRC task is to develop a set of acceptance criteria and review guidelines for (N-1) loop authorization requests.

WAPWR Response

Westinghouse will perform accident analyses and establish technical specification requirements for all modes of operation to be licensed for the WAPWR design.

60. Issue B-60: Loose Parts Monitoring Systems

Discussion

The presence of a loose (i.e., disengaged and/or drifting) object in the primary coolant system can be indicative of degraded reactor safety resulting from failure or weakening of a safety-related component. A loose part, whether it be from a failed or weakened component or from an item inadvertently left in the primary system during construction, refueling, or maintenance procedures, can contribute to component damage and material wear by frequent impacting with other parts in the system. A loose part can pose a serious threat of partial flow blockage with attendant departure from nucleate boiling which in turn could result in failure of fuel cladding. In addition, a loose part increases the potential for control rod jamming and for accumulation of increased levels of radioactive crud in the primary system.

The primary purpose of a loose part detection program is the early detection of loose metallic parts in the primary system. Early detection can provide the time required to avoid or mitigate safety-related damage to, or malfunction of, primary system components.

The NRC considers this issue as technically resolved with the issuance of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors."

WAPWR Response

The WAPWR design will include a loose parts monitoring system. Westinghouse will completely document and justify any deviations from the NRC Regulatory Guide 1.133 positions during the licensing process for the WAPWR design.

61. Issue B-61: Allowable ECCS Equipment Outage Periods

Discussion

Surveillance test intervals and allowable equipment outage periods in the technical specifications for safety-related systems are largely based on engineering judgment. This task involves the NRC development of analytically based criteria for use in confirming or modifying these surveillance intervals and allowable equipment outage periods.

WAPWR Response

Westinghouse will use probabilistic risk assessment, statistical assessment of reliability and availability, and the Westinghouse statistical set point methodology to specify equipment outage times and surveillance intervals for the WAPWR design. Concerning equipment outage times and surveillance intervals, the Westinghouse objective is to optimize the relationship between outage times, surveillance intervals, reliability, availability, and safety. This optimization will ensure that safety needs are satisfied while maximizing plant availability and operability.

62. Issue B-62: Reexamination of Technical Bases for Establishing SLs, LSSSs, and Reactor Protection System Trip Functions

Discussion

The methods used to establish safe operating limits for reactor cores were developed about 10 to 15 years ago. At present, safety margins are reviewed utilizing previous NRC judgments based on individual plant reviews. The NRC planned to develop a uniform NRC position for application to core performance reviews of new plants and to reloads and core modifications of operating plants. Subsequently the NRC has determined that this item does not involve a safety issue and has dropped it from further consideration.

WAPWR Response

Westinghouse will establish appropriate WAPWR technical specification requirements as an integral part of the design as discussed in Section 3.2.

63. Issue B-63: Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary

Discussion

There are several systems connected to the reactor coolant pressure boundary that have design pressures that are considerably below the reactor coolant system operating pressure. The NRC staff has required that valves forming the interface between these high and low pressure systems have sufficient redundancy to assure that the low pressure systems are not subjected to pressures which exceed their design limits.

Recently, there has been discussions relative to the adequacy of the isolation of low pressure systems that are connected to the reactor coolant pressure boundary. Past reviews have concentrated on ensuring isolation of the residual heat removal system, which is a low pressure system on almost all plants. Current reviews of license applications for new plants are based on NRC guidelines set forth in the Standard Review Plan (mainly Standard Review Plan 3.9.6, "Inservice Testing of Pumps and Valves").

This issue involves activities related to plants licensed prior to issuance of the NRC Standard Review Plan guidance. A related issue is discussed in Section 6.5 (item 8)

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 3.9.6 acceptance criteria during the licensing process for the WAPWR design.

64. Issue B-64: Decommissioning of Reactors

Discussion

10CFR 50.82, "Applications for Termination of Licenses," provides criteria by which licensees may terminate their licenses. Under this regulation, the Commission may require information from the licensee to demonstrate that the methods and procedures to be used for decontamination and for disposal of radioactive materials provide reasonable assurance that the dismantling and disposal will not be inimical to the common defense and security or to the health and safety of the public. 10CFR 50.33(f) includes the requirement that operating license applicants show that they possess or have reasonable assurance of obtaining funds necessary to cover the "estimated costs of permanently shutting the facility down and maintaining it in a safe condition."

Since 1960, about 50 research-type reactor facilities and 15 small power and test reactors have been decommissioned in accordance with the above regulations. In addition, the NRC reviews the general plans for decommissioning and financial arrangements for decommissioning as a part of its review of operating license applications. Based on acceptable findings, including this area, the NRC has issued operating licenses. As a result of the need for increased guidance to the industry in this area, the NRC has issued Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors." This regulatory guide includes methods and procedures considered acceptable by the NRC for the termination of licenses for operating reactors. However, because of the increasing interest in decommissioning, additional guidance is needed on this topic.

The studies and resultant safety acceptance criteria and guidelines for decommissioning operations developed under this task currently include consideration of occupational radiation safety. In addition, current requirements to keep occupational exposures as low as is reasonably achievable (ALARA) require that decommissioning plans proposed by licensees are reviewed within the context of ALARA regulations. While the NRC anticipates that improved guidance will be forthcoming as a result of this task, its completion is not expected to significantly reduce occupational exposures during decommissioning operations.

The NRC presently has under development new decommissioning rules to supplement the present rules. Technical evaluations have been completed and are documented in NUREG/CR-0672, "Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station", and NUREG/CR-0130, "Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station." A draft rulemaking environmental impact statement, NUREG-0586, "Draft Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," has been prepared. Proposed rule amendments for decommissioning have been prepared and are intended to assure that decommissioning of all licensed facilities will be accomplished in a safe and timely manner and that adequate license funds will be available for this purpose.

WAPWR Response

One of the design objectives of the WAPWR is to limit exposures to ALARA which will enhance any decommissioning effort. Decommissioning is, however, the responsibility of each utility utilizing the WAPWR design.

65. Issue B-65: Iodine Spiking

Discussion

The calculated radiological consequences for some postulated design basis accidents are highly dependent on the magnitude of the iodine spike postulated to occur following the transient. These calculations in turn determine the coolant activity limits allowed in the technical specifications. This NRC task is intended to develop and confirm a model for the iodine spiking phenomena. Procurement of data from operating plants and the development of a fuel release model for predicting the magnitude of the spikes will provide an understanding of this phenomenon which is not presently available. Improved knowledge of this topic will allow setting of the coolant activity limits at realistic levels. In addition, this could provide the basis for more realistic accident calculations.

WAPWR Response

Although current NRC iodine spike calculations are considered very conservative, the analyses for the WAPWR design will follow current iodine spike calculational methods, or any deviations thereto will be justified.

66. Issue B-66: Control Room Infiltration Measurements

Discussion

A key parameter affecting control room habitability under the conditions described in General Design Criterion 19, "Control Room," and Standard Review Plan 6.4, "Control Room Habitability System," is the magnitude of control room air infiltration rates. Estimates of these rates have been based on data relating to buildings that are substantially different than typical nuclear power plant control room buildings. This task involved the development of an improved data base.

The NRC considers this issue as being technically resolved and acceptance criteria have been incorporated in Standard Review Plan 6.4 "Control Room Habitability".

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 6.4 Revision 2 acceptance criteria during the licensing process for the WAPWR design.

67. Issus B-67: Effluent and Process Monitoring Instrumentation

Discussion

Monitoring of radioactivity in gaseous and liquid effluent streams from nuclear power plants is required for several purposes: (A) assessment of the adequacy of process and waste treatment systems, (B) the control of releases of radioactivity to the environment so that they do not exceed the limits of 10CFR Part 20 and 10CFR Part 50, Appendix I, and (C) the evaluation of environmental impact. This NRC task involves improving current guidance provided to applicants and reviewers in the areas of radiation monitoring for process and effluent systems and reviewing the effluent monitoring system for selected operating plants to determine their effectiveness in meeting the effluent release limits of 10CFR Parts 20 and 50.

The NRC has completed their activities related to this issue with the exception of a sub-task associated with the radiological monitoring of effluents which is encompassed by TMI Action Plan III.D.2.1. Current NRC acceptance criteria are documented in the Standard Review Plan (i.e., Sections 11.2, 11.3, 11.4, and 11.5).

WAPWR Response

Westinghouse will completely document and justify and deviations from the above mentioned NRC Standard Review Plan acceptance criteria during the licensing process for the WAPWR design.

68. Issue B-68: Pump Overspeed During a LOCA

Discussion

There is a potential for boiling water reactor recirculation pumps or pressurized water reactor main coolant pumps to overspeed during a loss-of-coolant accident, resulting in the potential for missile generation. This NRC task involves the conduct of analytical and experimental work to determine whether or not destructive overspeeds could be attained and to determine if corrective actions are necessary.

Each nuclear steam supply system vendor has supplied reports on pump overspeed which have been under review for several years by the NRC Reactor Systems Branch.

WAPWR Response

This issue has been addressed generically by Westinghouse in WCAP-8163, "Reactor Coolant Pump Integrity in LOCA." WCAP-8163 is applicable to the WAPWR design and will be referenced in appropriate licensing documents.

69. Issue B-69: ECCS Leakage Ex-containment

Discussion

In the event of a severe accident, such as a loss-of-coolant accident, or any other event which could lead to significant cladding failures, the levels of radioactivity in the coolant could be high. Such a situation would require effective control of any resultant leakage. Because of the inaccessibility of the equipment under post-LOCA conditions and the manual operations involved in aligning equipment for loop functions and isolating excessively leaking components, advanced planning of the steps involved in controlling the probable leakages for the required long-term loop configurations should be set out in emergency operating procedures. Technical specifications governing loop boundary integrity, leak detection equipment, isolation equipment, and leakage control equipment should be established, including limiting conditions for operation and surveillance requirements.

While existing equipment and procedures may permit a successful post-accident recovery operation, the current NRC Standard Review Plan does not provide an explicit basis for confirming that these objectives will be met.

This task has subsequently been superseded by TMI-2 lessons learned Item III.D.1.1 (refer to Section 3.1, item 26).

70. Issue B-70: Power Grid Frequency Degradation and Effect on Primary Coolant Pumps

Discussion

Offsite power system frequency decay, depending on the rate of decay, could provide an electrical brake on the reactor coolant pump motors that could slow the pumps faster than the assumed flywheel coastdown flow rates normally used in analyzing loss-of-flow accidents. Task A-35, "Adequacy of Offsite Power Systems," (refer to Section 5.1, item 35) was used to determine the maximum credible frequency decay rate used by the NRC in this task. The NRC considers this issue as resolved with the determination that no additional measures (beyond those documented in Standard Review Plan 8.3.1, "A-C Power Systems (Onsite)") are necessary to protect against a frequency decay event.

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 8.3.1 acceptance criteria during the licensing process for the WAPWR design.

71. Issue B-71: Incident Response

Discussion

Prior to the TMI-2 event, NRC actions taken in response to a serious incident were directed from an Incident Response Center (IRC). This NRC task dealt with ensuring an adequate response through the IRC being equipped with appropriate communications services, information handling and evaluation aids, pre-approved action guidelines, and technical and management personnel resources.

The NRC considers this issue as resolved with the implementation of post-TMI requirements for response to incidents, covered in TMI Action Plan Item III.A.3.1, "Emergency Preparedness - NRC Rule in Responding to Nuclear Emergencies."

WAPWR Response

This issue and its resolution apply to a NRC administrative activity that has no impact on the WAPWR design.

72. Issue B-72: Health Effects and Life Shortening from Uranium and Coal Fuel Cycles

Discussion

Current practice in health impact assessments is to convert radiation exposure estimates into estimates of health effects, such as cancer deaths, illness, and life-shortening. However, the models presently being used, such as those in WASH-1400, GESMO, current NRC case related testimony, and EPA assessments, all suffer from similar weaknesses. A major common weakness, which appears amendable to solution, is related to the correct treatment of competing risks among populations with life expectancies, age, and sex distributions that vary with time. Since the NRC staff is currently attempting to assess health effects in the future (e.g., Year 2000 and beyond), it is reasonable to expect significant changes in current population statistics. To make such an assessment, a demographic model is required which extrapolates the current population into the future, correctly allowing for competing risks of mortality from various causes (e.g., accidents, heart disease, and cancer). Failure to do so results, for example, in hypothetical cancer deaths for people who would statistically die from other causes. In the absence of better predictive models, it is not possible to even evaluate the uncertainty associated with the use of the current simplified methods for estimating health effects and consequent life shortening. Uncertainties in the use of current models are greatly magnified when attempting to make comparisons of health effects for the coal and nuclear fuel cycles.

Current health effects models generally are used for estimating long-term impacts. Chronic exposure may be the primary determinant of the number of deaths for a given period for a given pollutant. However, in the case of nonradiological pollutants from the coal fuel cycle, short-term fluctuations leading to acute exposures may determine the time of death and consequent life-shortening. Current evaluations of the coal fuel cycle generally fail to account for short-term mortality, disease and illness. In addition, short-term effects from chemical pollutants are generally dependent on the prior history of chronic (long-term) exposure.

Current models generally assume linear dose-response relationships even when evidence exists for real or practical thresholds, or where experimental data support a nonlinear dose response relationship.

This task involves the development of models to address these problems so that health effects (morbidity and mortality) can be assessed for both the coal and uranium fuel cycles as completely as current data permit and on a comparable basis. Resolution of this issue will be done through issue A-20, "Impacts of the Coal Fuel Cycle," (refer to Section 5.1, item 20).

WAPWR Response

This issue applies to an ongoing NRC administrative code development activity and has no impact on Westinghouse in relation to the WAPWR design.

73. Issue B-73: Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel

Discussion

This NRC task involves assessing the need for and, if necessary, development of criteria for acceptable vibration monitoring systems to provide early warning of excessive vibration inside the reactor vessel.

Current NRC acceptance criteria for a preoperational vibration test program for reactor vessel internals are provided in Standard Review Plan 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment," and Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing." Beyond acceptable reactor vessel internals preoperational vibration test programs, Westinghouse does not believe that additional vibration monitoring of the internals is necessary. Resolution of this isssue is covered through issue C-12, "Primary System Vibration Assessment," (refer to Section 5.3, Item 12).

WAPWR Response

The WAPWR program has extensive testing planned and the test results will be used to document the adequacy of the components with respect to vibration. Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 3.9.2 and Regulatory Guide 1.20 acceptance criteria and regulatory positions during the licensing process for the WAPWR design.

5.3 CATEGORY C ISSUES

The following discussions pertain to current Category C issues in relation to the WAPWR design. NRC discussions and descriptions of these issues are contained in NUREG-0471, "Generic Task Problem Descriptions (Category B, C, and D Tasks)," and NUREG-0933, "A Prioritization of Generic Safety Issues."

1. Issue C-1: Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment

Discussion

Certain classes of instrumentation incorporate seals. When safety related components within containment must function during post-LOCA conditions, their operability is sensitive to the ingress of steam or water. If the seals should become defective as a result of personnel errors in the maintenance of such equipment, such errors could lead to the loss of effective seals and the resultant loss of equipment operability. The NRC believes that the establishment of a basis for confidence that sensitive equipment has a seal during the lifetime of the plant is needed.

The NRC considers this issue as being technically resolved with the issuance of current criteria for qualification of safety-related electrical equipment. This criteria is discussed in detail in Section 4.0, item 14.

2. Issue C-2: Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure

Discussion

Inadvertent operation of containment sprays can result in a rapid depressurization of the containment building. Where containment external design pressure may be exceeded many plants have been provided with vacuum breakers or control system interlocks to prevent the containment external design pressure from being exceeded. The depressurization of the containment is a transient behavior and can take place in a short time period. This NRC task involves the development of a code to be used for the analysis of containment pressure response (both with and without the effects of vacuum breakers or control systems) for the inadvertent spray accident.

The NRC considers this issue as being technically resolved. Standard Review Plan Section 6.2.1.1 is used in reviewing licensee analyses of containment depressurization due to inadvertent spray operation.

WAPWR Response

This NRC program to develop computer technologies is independent of Westinghouse activities in this area.

Westinghouse has developed a conservative analytical methodology to determine the containment depressurization transient following an inadvertent spray actuation. This methodology has been utilized in prior Westinghouse applications and will be utilized for the WAPWR design.

3. Issue C-3: Insulation Usage Within Containment

Discussion

This issue is included as part of Unresolved Safety Issue A-43, "Containment Emergency Sump Performance" (refer to Section 4.0, item 21).

4. Issue C-4: Statistical Methods for ECCS Analysis

Discussion

Appendix K, "ECCS Evaluation Models," to 10CFR Part 50 specifies the requirements for ECCS analysis. These requirements presently call for specified conservatisms to be applied to certain models and assumptions used in the analysis to account for data uncertainties at the time Appendix K was written. The resulting conservatism in the calculated peak clad temperature, however, has never been thoroughly compared against the uncertainty in peak clad temperature obtained from a realistically calculated (best estimate) LOCA.

In order to assess the safety margin in the Appendix K requirements, the NRC planned to equate the peak clad temperature requirement (2200°F) to an uncertainty level of a realistic calculation. This would be accomplished by analytical analyses utilizing best estimate LOCA analysis codes in which certain input parameters are simultaneously varied about their uncertainty distribution functions such that a resulting uncertainty distribution function in peak clad temperature is obtained. It would then be possible to express the conservatism of the 2200°F cladding temperature limit in terms of probability and/or standard deviations from the most probable peak clad temperature.

The statistical methods for ECCS analysis would provide a probabilistic quantification of the safety margin imposed by Appendix K ECCS safety evaluation requirements. The results of this program are intended to be used to aid the NRC in the review of changes to vendor ECCS models and in performing NRC audit calculations of ECCS performance.

Mainly as a result of the TMI-2 event and the resulting deemphasis of the large-break LOCA, the NRC has reduced the priority of this work and may not fully complete the statistical assessment.

Although not directly related to this issue, the NRC has issued an advanced notice of proposed rulemaking concerning acceptance criteria for emergency core cooling systems. This proposed rulemaking is expected to result in procedural and technical changes to the current ECCS rule (refer to Section 6.1.2.2, item 1).

5. Issue C-5: Decay Heat Update

Discussion

This NRC task involves following the work of research groups in determining best estimate decay heat data and associated uncertainties for use in LOCA calculations. The results of this task could be incorporated into future revisions of the current regulations regarding ECCS performance.

Westinghouse has been active in the ANS decay heat subcommittee (ANS-5.1) and has reviewed and concurred with their findings. Westinghouse has gone on record requesting that the Appendix K rule be more flexible to allow the impact of new experimental data including the new decay heat standards. Westinghouse will continue to press for this additional flexibility and will actively support NRC best estimate LOCA calculations which use the new decay heat standards.

The NRC considers this issue as being technically resolved. As a result of following the development of ANS 5.1, the NRC does not intend to propose rulemaking to change 10CFR 50, Appendix K.

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.

6. Issue C-6: LOCA Heat Sources

Discussion

The contributors to LOCA heat sources, along with their associated uncertainties, and the manner in which they are combined have an impact on LOCA calculations. An evaluation of the combined effect of power density, decay heat, stored energy, fission power decay, and their associated uncertainties with regard to calculations of LOCA heat sources is needed. This NRC task involves the review of vendor's data and approaches for determining LOCA heat sources and developing NRC staff positions as needed.

Current LOCA analyses use a conservative approach for handling uncertainties on power density, decay heat, stored energy, local peaking factors, and nuclear uncertainty. The maximum values of each are used in a product manner to maximize the hot rod power and stored energy. Discussions with the NRC staff indicate that the non-prescriptive portions of the LOCA heat sources (everything except decay heat) and their uncertainties may be possible candidates for statistical convolution as long as the maximum uncertainties are included in the convolution process. Westinghouse had submitted an approach to the NRC staff which attempted to utilize this statistical approach (WCAP-9180/9181, "Consideration of Uncertainties in the Specification of Core Hot Channel Factor Limits"). However, at that time (pre TMI-2) it was felt that the proposed Appendix K rulemaking process would address methods of handling these uncertainties. The rulemaking changes to Appendix K never occurred, and now the NRC appears more receptive to identification and convolution of these uncertainties.

Westinghouse plans to resubmit a document similar to WCAP-9180/9181 to document a recommended method of handling LOCA input heat power uncertainties. This input will be generic and will cover all Westinghouse plants including the WAPWR.

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.

7. Issue C-7: PWR System Piping

Discussion

Combinations of fabrication, stress and environment have resulted in instances of stress corrosion cracking of low pressure schedule 10 type 304 stainless steel piping systems. Although these systems are not part of the reactor coolant pressure boundary, they are safety related; e.g., the containment spray system. The incidence of cracking has been restricted to thin wall, low pressure, low flow systems. These cracks have occurred adjacent to the weld zones of the thin-walled piping after approximately three to five years of service and were identified by volumetric examination, by leak detection systems, or by visual inspection. In each of the cracking events that have occurred to date, the affected piping was determined to have been inadvertently exposed to corrosive environments, such as thiosulfate and chlorides.

Current licensing criteria attempts to minimize the use of sensitized piping in safety-related piping systems and place increased emphasis on the use of corrosion-resistant material in such systems. The purpose of this task is to continue to evaluate operating experience to determine if augmented inservice inspection requirements should be established to further enhance the reliability of such piping systems.

The NRC considers this issue as being technically resolved with the issuance of NUREG-0691, "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors."

WAPWR Response

Based upon operating experience, it has been concluded that current ISI requirements for thin-walled piping in PWRs are adequate. Therefore, this issue has no impact on the WAPWR design.

8. Issue C-8: Main Steam Line Leakage Control System

Discussion

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

9. Issue C-9: RHR Heat Exchanger Tube Failures

Discussion

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

10. Issue C-10: Effective Operation of Containment Sprays in a LOCA

Discussion

This NRC task is intended to respond to a concern of the ACRS about the effectiveness of various containment sprays to remove airborne radioactive materials which could be present within the containment following a LOCA. This concern has been expanded to include the possible damage to equipment located inside containment due to an inadvertent actuation of the sprays.

The NRC considers this issue as being technically resolved with the issuance of ANSI/ANS 56.5-1979, "PWR and BWR Containment Spray System Design Criteria," which is referenced in Standard Review Plan Section 6.5.2.

WAPWR Response

Westinghouse has given appropriate consideration to the criteria of ANSI/ANS 56.5-1979 in the design of the WAPWR containment spray system.

11. Issue C-11: Assessment of Failure and Reliability of Pumps and Valves

Discussion

This issue is included as part of Unresolved Safety Issue A-45, "Shutdown Decay Heat Removal Requirements" (refer to Section 4.0, item 23).

12. Issue C-12: Primary System Vibration Assessment

Discussion

Structural damage to the primary system, including the reactor pressure vessel and internals, associated piping and steam generator tubing in pressurized water reactors can be caused by vibrations of sufficient magnitude. These vibrations can be either flow-induced or the result of operation of the pumps to which primary system piping is attached. There have been a number of instances where components internal to the reactor coolant pressure boundary have come loose as the result of flow-induced vibration and been carried through the primary system by the coolant flow.

Excessive core barrel movement, caused by flow-induced vibration, may lead to many detrimental effects including damage to reactor internals and interference with control rod movement. Problems resulting from excessive core barrel movement have been encountered at Palisades and possibly other operating plants.

Structural damage due to flow-induced vibration of steam generator tubing has also been encountered. Anti-vibration bars are currently utilized to minimize tube vibration. However, fretting has occurred due to deficient design and material selection for the anti-vibration bars.

Piping systems are also susceptible to forced vibration as a result of pump vibration during operation. If a natural frequency of the connected piping is very nearly the same as the driving frequency of the pump there is then the possibility, depending on the amplitude of vibration, for fatigue failures in the system, particularly at the nozzle where the stresses will be highest.

Preoperational testing of reactor internals, piping systems and mechanical equipment is conducted during startup functional testing to assure structural and functional integrity per Standard Review Plan 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment," and Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing." However, vibration frequency shifts are possible during operation as a result of component and/or component support wear or degradation. Also, vibration effects for the longterm may not have been properly assessed during startup testing.

Inservice inspection during the life of the plant and possible visual and audible detection of vibration during plant operation may be necessary in order to arrest structural damage already incurred or, if the vibration were to continue, might occur at some future time. This vibration assessment could lead to modifications in the design of systems components or component support arrangements of system operation sequences.

Beyond acceptable primary system preoperational vibration test programs, Westinghouse does not believe that additional vibration monitoring of the primary system is necessary.

The NRC considers this issue as being technically resolved. Current guidelines in SRP 3.9.2, combined with NRC positions on loose parts monitoring in Regulatory Guide 1.133 provide sufficient basis for considering this issue to be resolved.

WAPWR Response

The WAPWR program has extensive testing planned and the test results will be used to document the adequacy of the components with respect to vibration. Westinghouse will completely document and justify any deviation from the NRC Standard Review Plan 3.9.2 Acceptance Criteria and regulatory positions of Regulatory Guides 1.20 and 1.133 acceptance criteria during the licensing process for the WAPWR design.

13. Issue C-13: Non-Random Failures

Discussion

This issue is included as part of Unresolved Safety Issue A-17, "Systems Interactions in Nuclear Power Plants" (refer to Section 4.0, item 13).

14. Issue C-14: Storm Surge Model for Coastal Sites

The NRC is required to estimate the design basis water levels for each site. For coastal and estuarine sites, the design basis water level is often caused by a storm surge, which results from the wind and pressure fields of an intense storm acting on the water.

The primary tool used by the NRC for estimating storm surge has been the "bathystrophic" model as developed by the U.S. Army Corps of Engineers, Coastal Engineering Research Center (CERC). This model is based on the bathystrophic approximation, relating sea surface slope to wind stress, bottom stress, and pressure gradient, with a correction for Corriolis force due to along-shore currents. The NRC considers this model to now be obsolete. Bigger and faster computers are now capable of solving multidimensional dynamic equations which account for many effects not included in the bathystrophic model. The multidimensional dynamic mathematical models can account for irregular shorelines, while the shape of the shoreline is not considered at all by the bathystrophic model.

True long wave dynamics are simulated by multidimensional dynamic mathematical models, but are completely neglected by the bathystrophic models. These two effects are especially important when estimating storm surges in semienclosed areas.

The purpose of this task is for the NRC to develop a replacement for the bathystrophic model so that their evaluation of storm surge reflects state-of-the-art techniques.

WAPWR Response

This issue applies to an ongoing NRC administrative activity related to code development and is not applicable to Westinghouse in relation to the WAPWR design.

15. Issue C-15: NUREG Report for Liquid Tank Failure Analysis

Discussion

Standard Review Plan 15.7.3, "Postulated Radioactive Releases Due to Liquid-Containing Tank Failures," requires an analysis of the consequences of failure of tanks containing radioactive liquids outside containment. This task involves the development of a NUREG report that is intended to describe a consistent and acceptable method for analyzing the effects of a failure of a radioactive liquid waste tank.

The current version of Standard Review Plan 15.7.3 does provide certain criteria for analyzing the effects of a failure of radioactive liquid waste tanks. These criteria include:

- o Limiting radionuclide concentrations to those specified in 10CFR Part 20, "Standards for Protection Against Radiation."

- o Assuming 0.12 percent failed fuel.

- o Assuming 80 percent volume in failed components.

- o Credits in analyses that can and cannot be taken.

WAPWR Response

Westinghouse will perform an analysis of the consequences of failure of tanks containing radioactive liquids outside containment in accordance with Standard Review Plan 15.7.3 during the licensing process for the WAPWR design.

16. Issue C-16: Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection

Discussion

Interpretations of the National Environmental Policy Act (NEPA) require that environmental impact assessments include land use impacts and alternatives in nuclear power plant licensing cases. The NRC has performed both economic and non-economic land resource assessments in compliance with these NEPA requirements. Some licensing cases have questioned the adequacy of the NRC's resource evaluative methods with respect to large land areas required for sites and cooling lakes. The primary issue concerning the NRC's assessment is that neither economic analyses nor resource assessment as currently performed provides a convincing rationale for preemption of high quality land in view of continued population pressures, predicted impending lags in world-wide agricultural food production and probable increasing international demands on the United States for exports of agricultural products.

Food and fiber production and distribution rank with energy production and utilization as vital world problems now and for the foreseeable future. These problems are inextricably linked since energy production facilities can be consumers of large land areas while energy is a prime requirement for even modest levels of agricultural production. Thus, land use is and probably will remain a key siting issue in nuclear plant licensing.

This NRC task is intended to involve the conduct of a confirmatory exploration of new energy techniques to determine their suitability for application to environmental licensing assessment under NEPA. A problem of immediate licensing concern to the NRC is the conflict in land use which occurs when power plants with large cooling lakes are sited in regions of prime agricultural land.

WAPWR Response

This task is associated with an environmental proceedings issue that is not applicable to Westinghouse in relation to the WAPWR design.

17. Issue C-17: Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes

Discussion

There are no current NRC criteria for acceptability of solidification agents. This NRC task involves the development of criteria for acceptability of radwaste solidification agents to properly implement a process control program for the packaging of diverse plant waste for shallow land burial.

The NRC considers this issue as technically resolved with the issuance of a proposed rule, "Licensing Requirements for Land Disposal of Radioactive Waste (10CFR Part 61)."

WAPWR Response

This issue and the associated proposed rule are related to requirements for land disposal of radioactive wastes which are not applicable to Westinghouse in relation to the WAPWR design.

5.4 CATEGORY D ISSUES

The following discussions pertain to current Category D issues in relation to the WAPWR design. NRC discussions and descriptions of these issues are contained in NUREG-0471, "Generic Task Problem Descriptions (Category B, C, and D Tasks)" and NUREG-0933, "A Prioritization of Generic Safety Issues."

1. Issue D-1: Advisability of a Seismic Scram

   Discussion

   The ACRS has recommended that studies be made of techniques for seismic scram and of the potential safety advantages and potential disadvantages of prompt reactor scram in the event of strong seismic motion, say more than one-half the safe shutdown earthquake. Various suitable techniques have been identified and exist, but thus far only limited studies have been reported on the pros and cons of seismic scram.

   Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," of 10CFR Part 100 requires that suitable instrumentation shall be provided so that the seismic response of nuclear power plant features important to safety can be determined promptly to permit comparison of such response with that used as the design basis. Such a comparison is needed to decide whether the plant can continue to be operated safely and to permit such timely action as may be appropriate.

   Regulatory Guide 1.12, "Instrumentation for Earthquakes," describes seismic instrumentation acceptable to the NRC staff as satisfying the above stated requirements of Appendix A to 10CFR Part 100. Regulatory Guide 1.12 requires that one triaxial response spectrum recorder capable of providing signals for immediate control room indication be provided at the containment foundation.

   These criteria and regulatory guidance do not address the need for instrumentation that would automatically shutdown a nuclear power plant when an earthquake occurs which exceeds a predetermined intensity. This issue involves considerations of the need for such instrumentation.

Westinghouse believes that the automatic shutdown of a nuclear power plant for an earthquake event with a magnitude less than or equal to the operating basis earthquake does not seem necessary. For an operating basis earthquake occurrence the structural integrity of the plant is maintained to the extent that the plant can continue to operate. Therefore, if immediate control room indication is provided in accordance with Regulatory Guide 1.12, operator action and administrative procedures for plant shutdown are sufficient for an earthquake less than or equal to the operating basis earthquake.

WAPWR Response

Westinghouse is considering incorporating a seismic scram in the WAPWR design. Inclusion or exclusion of this feature will be completely documented and justified during the licensing process for the WAPWR design.

2. Issue D-2: Emergency Core Cooling System Capability for Future Plants

Discussion

This issue is included as part of the Unresolved Safety Issue A-45, "Shutdown Decay Heat Removal Requirements" (refer to Section 4.0, item 23).

3. Issue D-3: Control Rod Drop Accident (BWRs)

Discussion

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

5.5 UNCATEGORIZED ISSUES

The NRC continuously evaluates the safety requirements used in its reviews against new information as it becomes available. Sections 5.1 through 5.4 provide a discussion of NRC generic safety issues identified and categorized by the NRC in 1978. Since that time, new generic safety issues have been identified as a result of licensee event reports, ACRS reports, and other NRC activities. Major sources of new generic safety issues are NUREG-0572, "Review of Licensee Event Reports (1976-1978)," NUREG-0933, "A Prioritization of Generic Safety Issues," and NUREG-0705, "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plants."

New generic safety issues have not been categorized by the NRC in the manner the previous safety issues were categorized (i.e., Category A, B, C, and D). The following discussions pertain to these new "uncategorized" generic safety issues in relation to the WAPWR design.

1. Issue 1: Failures in Air-Monitoring, Air-Cleaning, and Ventilating Systems

   Discussion

   This issue is identified in Appendix D of NUREG-0572 and is one of the key observations made after the ACRS requested its members and consultants to make comprehensive reviews of all licensee event reports issued during the years 1976, 1977, and 1978.

   Data collected over the 3-year period showed that 14 percent of all licensee event reports were related to failures in the air-monitoring, air-cleaning, and ventilating systems. This translates into more than 350 licensee event reports each year. Monitoring equipment failures accounted for more than one-half the system failures in boiling water reactors and more than one-third the system failures in pressurized water reactors. This disparity occurred because of the presence of more air-cleaning and ventilating systems in boiling water reactors.

The proper operation of air-monitoring, air-cleaning, and ventilating systems is important for maintaining primary containment integrity and controlling airborne particles and gaseous releases from plants.

Proper ventilating system performance is important to the operation of high pressure coolant injection and reactor core isolation cooling systems in boiling water reactors and waste gas processing systems in pressurized water reactors. Twenty-four ventilating system failures were reported during the 3-year period, the consequences of which are addressed in NUREG-0572, Appendix D, Item XIV. Other failures related to dampers in ventilating systems are discussed in NUREG-0572, Appendix D, Item XVII.

The proper performance of air-monitoring equipment is essential for the avoidance of buildup of hazardous concentrations of gases (e.g., hydrogen) and the assessment of the potential impact of environmental releases.

No technical solution to this issue has been identified in NUREG-0572. However, the NRC has indicated that further development is needed to produce more reliable monitoring systems. There is also an indication that a possible solution in improving the licensee's maintenance and testing program will result in reduced failures of air monitors and ventilation system dampers. As a result, this issue may be ultimately resolved administratively by implementation of an improved test and maintenance program on the affected systems.

WAPWR Response

The WAPWR design will consider the possibility of a technical solution to the reliability problems in the air-monitoring and ventilation systems. That solution will consider automatic surveillance systems or improvements in the design of the monitoring and ventilation system.

2. Issue 2: Failure of Protective Devices on Essential Equipment

Discussion

The ACRS identified this potential safety concern in NUREG-0572. A large number of licensee event reports have reported failure or incapacitation of essential equipment as a result of failure of fuses or other devices installed for the sole purpose of protecting that essential equipment or its services. The systems affected exist throughout the plant and include the plant control system, the plant protection system, and the engineered safety features. Particularly vulnerable are actuators that require power in order to drive motors and operate valves. The failures are not limited to overcurrent protectors but occur in equipment such as torque limiters, overspeed protectors, and other interlocks and may be caused by improper applications or adjustments as well as component failures.

Safety implications arise because the expected failure rate of essential equipment may be overly optimistic because of not accounting for failure of protective devices. Where failures result from improper selection of fuse sizes or adjustment of protective devices, there is an increased probability of common mode failure of redundant vital services.

In the past, the corrective action has been to replace the failed fuse or readjust the adjustable devices. Where disabling of such equipment could remove or substantially degrade vital services, the NRC feels that the basic criteria for protecting the equipment should be reexamined. For example, the NRC believes the rules for protection of vital equipment should perhaps be different than current standard electrical practice.

WAPWR Response

The design process for the WAPWR will investigate the above concerns, and the potential for problems will be minimized within existing practice. If necessary, consideration will also be given to the modification of "industry practice" for protection of equipment. For example, bypass of certain protective functions under accident conditions might provide a solution. Any criteria modification will be undertaken with adequate consideration given to any increased probability of damage to equipment, the resulting effect on utility financial risk, and within the risk/safety goal considerations.

3. Issue 3: Set Point Drift in Instrumentation

Discussion

This issue is identified in Appendix D of NUREG-0572 and is one of the key observations made after the ACRS requested its members and consultants to make comprehensive reviews of all licensee event reports issued during the years 1976, 1977, and 1978.

Data collected over the 3-year period showed that 10 percent of all licensee event reports were related to drift in the set points of instrumentation beyond technical specification limits. This amounted to an average of 258 licensee event reports each year. The proportion of these events that resulted in simultaneous drifts in redundant channels was not established in NUREG-0572.

An unplanned change in the set point of an instrument (set point drift) will alter the actual value of the measured parameter at which a particular action is to occur. Excessive drift in an instrument's set point beyond technical specification limits could result in the instrument not providing timely warning signals prior to or during an accident thereby failing to perform its safety function. All safety instrumentation channels are redundant but simultaneous drift of redundant instruments beyond technical specification limits could affect plant safety.

For those instruments where set point drift is due to component failures, a possible solution is to make the necessary repair, recalibrate, and restore the instruments to service. For those instruments where the margin between the selected set point and the technical specification limit is not sufficient to allow for normal instrument inaccuracy, a possible solution is to increase the margin between the selected set point and the technical specification limit to accommodate the inherent instrument inaccuracy.

There are two considerations which will reduce any problems with the WAPWR protective system to those associated with component failure.

By using the digital integrated protection system, the WAPWR design will eliminate some of the problems associated with set point drift in the protection system. The only portion of the instrumentation which will be subject to drift will be that analog portion from the sensor through the analog to digital converter. The redundant sensor selector will identify any sensor channels that have drifted outside of tolerance and will make this information available to the operators.

For the WAPWR, the set points for the protection system will be determined using the guidance of Regulatory Guide 1.105, Revision 2, "Instrument Setpoints," (currently in draft) which references ISA S67.04, 1982 "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants." Use of this guidance will eliminate all of the problems of drift beyond the technical specification limits except those associated with component failures.

WAPWR Response

The WAPWR protection system will be designed and set points selected using the Westinghouse setpoint methodology approved by the NRC in NUREG-0717, Supplement 4, dated August 1982.

4. Issue 4: End-of-Life and Maintenance Criteria

Discussion

This issue has been addressed as part of the NRC overall equipment qualification program. Existing and proposed requirements include both end-of-life and maintenance considerations. Available material aging information coupled with actual plant operating and maintenance experience, could be factored into the process of determining the end-of-life for various components as well as determining appropriate maintenance periodicity. The failure of safety-related components can lead to loss of reactor coolant pressure boundary integrity or loss of safety functions. Such failures possibly could be reduced by using end-of-life data and improved periodic maintenance criteria.

NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," and the "Guidance for Evaluating Qualification of Class IE Electrical Equipment in Operating Reactors" require that qualification programs for electrical equipment should identify materials susceptible to aging effects and establish a schedule for periodically replacing the equipment and/or materials.

The proposed Revision 1 to Regulatory Guide 1.89, "Environmental Qualification of Electrical Equipment for Nuclear Power Plants," was drafted in February 1982 and includes a number of specific positions on the subject of equipment end-of-life and maintenance. The Regulatory Guide positions are:

- o The qualified life of the equipment (or component, as applicable) and the basis for its selection be defined and documented.
- o Qualified life should be established on the basis of the severity of the testing performed, the conservatisms employed in the extrapolation of data, the operating history, and the other methods that may reasonably be used. All assumptions should be documented.

- An ongoing program to review surveillance and maintenance records to identify age-related degradations should be established.

- A component maintenance and replacement schedule that includes consideration of aging characteristics of the installed components should be established.

- Sections 6.4 and 6.5 of IEEE 323-1974 discuss qualification by operating experience and by analysis, respectively. The adequacy of these methods should be evaluated on the basis of the quality and detail of the information available in support of the assumptions made. Operating experience and analysis based on test data may be used where testing is precluded by the physical size of the equipment or the state of the art of testing. When the analysis method is employed because of the physical size of the equipment, tests on vital components of the equipment should be provided.

The NRC is in the process of coding (refer to Section 6.1.2.3, item 5) the similar requirements for mechanical equipment.

The NRC Standard Review Plan, Section 3.11., "Environmental Qualification of Mechanical and Electrical Equipment," includes requirements for maintenance/surveillance programs for equipment located in mild environments. Specifically, it is required that "the maintenance/surveillance program data shall be reviewed periodically (not more than every 18 months) to ensure that the design qualified life has not suffered thermal or cyclic degradation resulting from the accumulated stress triggered by the abnormal environmental conditions and the normal wear due to its service condition. Engineering judgement shall be used to modify the replacement program and/or replace the equipment as deemed necessary."

WAPWR Response

The WAPWR design will provide for design improvements in maintainability and an extension of the time between maintenance periods as practicable.

In addition, Westinghouse will fully document the level of conformance with the regulatory positions of Regulatory Guide 1.89 and the acceptance criteria of SRP 3.11 during the licensing process for the WAPWR.

5. Issue 5: Design Check and Audit of Balance-of-Plant Equipment

Discussion

This issue involves a potential improvement that might be achieved by requirements for verification that the balance-of-plant "as-built" configuration satisfies the design intent. Such action could improve the reliability of balance-of-plant equipment and reduce demands on safety equipment. This issue has arisen because of failures of balance-of-plant equipment to perform as intended for many reasons and as a result, place various demands on safety systems.

The WAPWR, in moving toward a nuclear power block concept, has placed more of the plant scope within the Westinghouse sphere of direct control. By so doing, portions of the concern described here are of less importance because that portion of equipment which represents balance-of-plant is further removed from the plant safety equipment.

Regardless of this consideration, some greater capability to verify that as built conditions accurately reflect design needs will be required in the future. In a one-step licensing process, there is a strong need to certify that the plant has been built as licensed and the commitments made in the safety analysis report have been fulfilled. As part of the fulfillment of this verification, some consideration should be made to verify that balance-of-plant systems adequately support the plant and will not unnecessarily increase the challenges to elements of the nuclear power block concept.

WAPWR Response

There is no direct impact on the WAPWR design posed by this issue. However, a program will be developed to demonstrate that the plant has been built as licensed.

6. Issue 6: Separation of Control Rod from Its Drive and BWR High Rod Worth Events

Discussion

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

7. Issue 7: Failures Due to Flow-Induced Vibrations

Discussion

NUREG-0572 indicated that a large number of licensee event reports are the result of flow-induced vibrations. These vibrations occur in equipment and piping carrying single and two-phase fluid. Flow-induced vibrations are caused by vortex shedding resulting from rapid area change, buffeting due to random flow turbulence, fluid structures interaction instability, leakage excitation, steady operation of positive displacement pumps and cavitating valves. The vibrations frequently cause failure of equipment, electrical wiring or components, pumps, valves and piping systems. The three major failure mechanisms are high cycle fatigue, impact, and fretting (wear).

Vibration problems inside the reactor vessel manifest themselves as worn guide tubes, loose guide thimbles, cracked shrouds, cracked nozzles and spargers. Charging pumps have been damaged by cavitation as well as turbulent buffeting vibrations which show up as cracked casings and welds. Vibrating valve internals result (in closed and open positions) in cracked and worn valve seats as well as cracked welds. Other failures resulting from vibration include loosened bolts, broken fittings, leaking snubbers, damaged pipe hangers, broken wires, thrown switches, loosened relays, damaged printed circuit boards, loosened instrument terminals, radiation monitor failures, false instrumentation activation, and open breakers.

The problem of excessive vibration is important because it can often lead to damage of multiple components. These events are frequently precursors to more serious events inasmuch as continued occurrences can result in pipe cracks, failures of valves and snubbers, and damage to electrical and mechanical equipment. Other aspects of this problem include the effects of vibration (as well as water hammer) on engineered safety features following severe transients.

The NRC currently requires plants to perform preoperational testing of plant fluid systems to verify that no excessive vibration exists. These requirements provide a large degree of certainty that flow-induced vibration will not cause problems during the plant life.

WAPWR Response

In the WAPWR design, consideration will be given to reduction in the potential for excessive vibration. Any information which is currently available will be used to reduce the operational vibrations due to fluid flow, and consideration will be given to potential tradeoffs between stiffening and softening of piping systems. An extensive vibration test program will be performed on the WAPWR. This test program will provide indication that no vibration problems exist in the WAPWR design.

8. Issue 8: Inadvertent Actuation of Safety Injection in PWRs

Discussion

Operator errors, instrument malfunction, and reactor transients and trips have been reported as the cause of inadvertent actuation of the safety injection system. At least 40 cases of inadvertent actuation of safety injection have been identified in NUREG-0572. Approximately one-fourth of the events sampled were due to operator error. The problem is repetitive in nature; at several facilities the problem has a long history. The vast majority of events occurred in Westinghouse nuclear steam supply systems, whereas plants supplied by other vendors had few or no reported events.

Safety injection systems are required to operate during loss-of-coolant accidents and other severe transients that require borated water addition to the primary system. Inadvertent actuation of the system injects cold borated water into the reactor when it is not needed, subjecting injection nozzles to thermal stresses and requiring removal of boron from the primary system before startup. The present number of occurrences is probably not significant with respect to the effects upon the primary system; however, operator response to an inadvertent safety injection involves termination of the injection and resetting of the injection signal. This generally occurs within 1 to 8 minutes following the start of injection and follows a check of other plant status instrumentation. Repeated operator exposure to inadvertent safety injection and its termination may produce an unacceptable response in cases where the injection is required to provide core cooling water.

The WAPWR design will be less likely to experience a spurious reactor trip or an inadvertent safety injection. The protection system objectives provide for a reduced probability of spurious actuation due to the failure of any single component or system and increased margin between the low pressurizer pressure safety injection set point and the minimum pressurizer pressure reached following a reactor trip from full power. The control system and the advanced control room (ACR) will lower the probability of putting the WAPWR into a state from which a reactor trip from full power would result in a safety injection. For example, improved steam generator feedwater control will prevent steam generator related inadvertent safety injection. The ACR will also make the assessment of plant safety problems both more reliable and easier to make. Because of ACR related improvements there is a much greater certainty that the operations personnel will recognize the need for a safety injection. Additionally, the sizing of reactor coolant system components will be performed with an objective of increasing the margin between the low pressurizer pressure safety injection set point and the minimum reached following a reactor trip from full power.

If the WAPWR does experience an inadvertent safety injection, the ACR in conjunction with plant procedures will aid in the assessment of plant state. Additionally, any concerns over combined pressure and thermal stresses to injection nozzles will be reduced as the shut off head of the safety injection pumps will be such that injection will not occur following reactor trip.

WAPWR Response

The WAPWR design described above, particularly the protection system design and the sizing of reactor coolant system components, eliminate inadvertent safety injection as a problem in the WAPWR.

9. Issue 9: Reevaluation of Reactor Coolant Pump Trip Criteria

Discussion

The issue of reevaluation of reactor coolant pump trip criteria involves the potential improvement that might be achieved by establishing better criteria on when to allow the operation of reactor coolant pumps and when to trip them. It was believed that better criteria might allow the use of reactor coolant pumps to aid in recover from certain transients while still ensuring that these pumps could be tripped during small-break LOCA.

This issue was also raised as a result of post-TMI licensing requirements (Section 3.3.1, Item 4 of this document) and is fully discussed in NRC Generic Letters 83-10c and 83-10d (Section 6.4, Items 92 and 93 of this document).

WAPWR Response

See the above referenced items for a complete discussion of this item and its relation to the WAPWR.

10. Issue 10: Surveillance and Maintenance of Tranversing Incore Probe Isolation Valves and Squib Charges

Discussion

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

11. Issue 11: Turbine Disc Cracking

Discussion

This issue has been raised because of the discovery of stress corrosion cracking in the low pressure discs of Westinghouse-designed turbines.

This issue is not by itself a distinct generic issue but is part of existing Issue A-37, "Turbine Missiles." Refer to Section 5.1 (item 37) for a discussion of this issue.

12. Issue 12: BWR Jet Pump Integrity

Discussion

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

13. Issue 13: Small Break LOCA from Extended Overheating of Pressurizer Heaters

Discussion

This is an ACRS concern raised by the Subcommittee on TMI-2 Implications in October of 1979. The issue centers around the possibility of a breach in the reactor coolant system boundary caused by the failure of nonsafety interlocks between pressurizer water level and pressurizer heater power and prolonged overheating of the immersion heaters due to operator failure to detect and terminate electrical power.

One solution would be to upgrade and expand the heater power and pressurizer level interlocks, operator training, and control of instrumentation and control modifications and repair efforts in accordance with quality assurance procedures for protective systems rather than normal nonsafety plant system.

WAPWR Response

The WAPWR design will address the issue described above. The risk associated with this transient will be evaluated in the context of safety goals.

14. Issue 14: PWR Pipe Cracks

Discussion

Cracking has occurred in PWR piping systems as a result of stress corrosion, vibratory and thermal fatigue, and dynamic loading. However, to date, no cracking has been experienced in the primary system piping of PWRs. Thus far, all incidents of cracking have been detected and corrective actions taken prior to any catastrophic failures.

Cracking in PWR nonprimary system piping could lead to a lessening of the system functional capability and possibly result in situations such as degraded core cooling. Cracking in PWR primary system piping has not been experienced, and the mechanisms and environmental conditions necessary to initiate and propagate the cracking in this piping are not known to exist. Therefore, the risk associated with PWR pipe cracks is negligible for the primary system and low for the other piping systems.

The third Pipe Crack Study Group was established in 1979. The charter of the PWR Pipe Crack Study Group included (A) the causes and safety significance of pipe cracks in PWR safety-related systems, (B) the ability of current inservice inspection and leak detection techniques to detect these cracks, and (C) recommendations for both upgrading the licensing process for plants in the operating license and construction permit stages and for implementation of new criteria on operating plants. In September 1980, the PWR Pipe Crack Study Group completed its investigation of this issue and published its findings as NUREG-0691 "Report of Investigations and Evaluations of Cracking Incidents in Piping in Pressurized Water Reactors." This report provides conclusions regarding systems safety and recommends technical solutions to the issue. As a result of issuing NUREG-0691, the NRC considers this issue to be technically resolved.

WAPWR Response

The WAPWR design will follow the recommendations of NUREG-0691 in minimizing the potential for cracking in WAPWR piping systems. The WAPWR analyses will also demonstrate that the criteria of NUREG-0691 are met. Calculations as described in the NUREG-0691 will be performed to assure that safety systems, particularly safety injection, will perform acceptably under analyzed break situations.

15. Issue 15: Radiation Effects on Reactor Vessel Supports

Discussion

This issue was first identified in June 1978 when Virginia Electric and Power Company filed a notification for its North Anna plant in accordance with 10CFR Part 21, "Reporting of Defects and Noncompliance."

Reactor pressure vessel external steel support structures may become embrittled by neutron radiation to the point where their structural integrity may be impaired by virtue of reduced fracture resistance. The theory is that neutrons with less than 1 MeV of energy can induce significant damage to supports because of their relative abundance.

Additionally, compared to the reactor vessel, supports operate at low temperatures thereby making concurrent annealing during operation very small. Structural steels vary widely throughout the industry and the problem could be quite severe at some plants.

Thus, embrittlement damage to reactor vessel supports can result in their failure to adequately support the reactor vessel under large load conditions such as an earthquake or a loss-of-coolant accident.

The WAPWR design will be less prone to embrittlement of reactor vessel supports. Improvement of the core baffle/reflector region to provide increased shielding of the reactor vessel will also reduce the affect on the fracture toughness of supports.

WAPWR Response

The WAPWR design and safety analysis will demonstrate that support structures for vital equipment are adequate under design basis loading conditions. The supports for the reactor pressure vessel will be evaluated for their adequacy under appropriate loading combinations. This will incluoe a demonstration that the reactor pressure vessel steel support structures will not become embrittled by neutron irradiation to the point where their fracture resistance is reduced to a level which yields unacceptable results under design basis loads.

16. Issue 16: BWR Main Steam Isolation Valve Leakage Control Systems

Discussion

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

17. Issue 17: Loss of Offsite Power Subsequent to a LOCA

Discussion

This issue involves a potential improvement in plant safety that might have been achieved if the plant design basis included the loss of offsite power subsequent to a LOCA. This issue has not been recommended for designation as an Unresolved Safety Issue because the probability of the combined event is judged by the NRC to be very low (on the order of 10^{-6}/reactor year) and the consequences would be insignificant, because adequate core cooling would be provided by vessel inventory during the time required for diesels to start and assume load. However, the NRC and ACRS feel that there may be some safety benefit to a resolution of this concern.

Westinghouse addresses the loss of offsite power concurrent with a LOCA, but there seems to be some potential concern over a loss of offsite power at some time post-LOCA. This concern is related to item 26 below, "Diesel Generator Loading Problems Related to SIS Reset on Loss of Off-site Power."

WAPWR Response

For the WAPWR design, an evaluation will be performed and documented to demonstrate that for large and small LOCAs, the consequences of a loss of offsite power post-LOCA do not represent a safety problem.

18. Issue 18: Steamline Break with Consequential Small LOCA

Discussion

This issue can be broken down into two issues:

- Steamline break with a subsequent small LOCA resulting from failure of a partially degraded steam generator tube(s).

- Steamline break with a subsequent small LOCA (other than a steam generator tube rupture) resulting from a stuck-open power-operated relief valve or safety valve actuated during the primary system transient or resulting from pipe whip or jet impingement from the broken steam line.

In PWRs, the potential exists for steamline breaks consequently leading to a small primary system LOCA. NRC analysis has indicated that the primary pressure and the pressurizer level may change qualitatively in the same way during a combined LOCA compared to a primary break, a steamline break, or a steam generator tube rupture. For the primary temperature and secondary pressure, a combined LOCA behaves qualitatively like a steamline break. For these latter two parameters, a primary rupture or steam generator tube rupture appear clearly distinct from the behavior of a combined LOCA.

Thus, two concerns have been identified which could increase the risk associated with these issues. These are (1) the possibility of primary side LOCAs may be increased through the consideration of new initiating mechanisms, and (2) the symptoms of a combined primary/secondary blowdown may increase the possibility for operator error through misinterpretation and improper action.

The conclusion reached is that operator misinterpretation could supply the greatest contribution to the probability of a an accident.

The WAPWR design will incorporate steam generator design improvements which will reduce the problems associated with tube degradation. This will in turn decrease the probability of a steam generator tube rupture following a steam line break. Additionally, the criteria for plugging of tubes will be reviewed in order to minimize the probability of this event, and instrumentation to address Regulatory Guide 1.97, Revision 2 (refer to Section 3.1, item 23) will be reviewed to ensure capability to detect a tube rupture following a steam line break.

In a similar sense, the work being done in conjunction with the EPRI valve testing program (described in some detail in Section 3.1, item 15) will provide a greater assurance that the safety and relief valves of the WAPWR will close when required. Instrumentation (described in some detail in Section 3.1, item 16) will also be provided which will positively indicate the positions of these valves and allow appropriate procedures to be implemented to bring the plant to a safe shutdown.

WAPWR Response

Besides the work described above, which provides assurance that this issue is of little concern for the WAPWR design, procedures will be written to address these events following a steam line break.

19. Issue 19: Safety Implications of Nonsafety Instrument and Control Power Supply Bus

Discussion

The issue is included in part of Unresolved Safety Issue A-47, "Safety Implications of Control Systems" (refer to Section 4.0, item 25).

20. Issue 20: Effects of Electromagnetic Pulse on Nuclear Power Plants

Discussion

The electromagnetic pulse (EMP) from a high altitude nuclear weapon detonation will induce electrical transients in the instrumentation, control and power lines of nuclear power plants. The extent to which these EMP transients may cause critical plant electrical and electronic systems to fail or malfunction and ultimately result in damage to the reactor is being investigated. A single EMP could affect most of the nuclear power plants in the continental United States. EMP-like effects can also be simulated locally using truck-transportable land based generators. The NRC regulations (10CFR 50.13) state that license applicants are not required to provide design features or other measures for the specific purpose of protection against the effects of (A) attacks and destructive acts; including sabotage, directed against the facility by an enemy of the United States, whether a foreign government or other person, or (B) use or deployment of weapons incident to U.S. defense activities.

The present NRC investigation was initiated as a result of informal staff discussions with five Commissioners in 1979. Subsequently, Commissioner Ahearne (then Chairman) instructed the staff to plan and carry out this investigation. The objectives of the investigations are (A) to determine the vulnerability of selected safe shutdown systems of a specific nuclear plant to EMP effects due to nuclear weapon detonations and non-nuclear generators, (B) to determine how those safe shutdown systems vulnerable to EMP may best be hardened against EMP, and (C) to characterize to the extent possible the effects of EMP on nuclear plants in general based on the study of specific systems of the subject plant. The overall objective is to provide the Commission with a basis for considering the need for amending the regulations to include design requirements for the protection of nuclear power plants against effects of EMP.

A technical assistance program with Sandia National Laboratory (SNL) was initiated in August 1980 to implement the investigation. The Watts Bar plant was selected for the study. The program includes EMP coupling analysis, evaluation of failure threshold of selected safety equipment, and an onsite test program to obtain data for confirmation of the results of analyses. The preliminary conclusion is that the safe shutdown systems at Watts Bar would not be damaged by EMP. The major work remaining to be completed is the extension of these results to nuclear power plants in general, and the preparation (by Sandia) of the interim report and the draft final report. An NRC staff report is planned for late summer, 1982.

EMP concerns during the peacetime operation of nuclear power plants derive from EMP which could be produced by terrorist actions involving nuclear weapon detonations or nonnuclear generators, or which could result from accidents involving U.S. or foreign weapons systems. The determination of the probability of occurrence of these types of EMP events is not within the scope of the current EMP investigation. However, consideration of effects due to nonnuclear generators is included in the investigation.

The NRC preliminary conclusion is that significant threat does not exist from nonnuclear generators because of the difficulty of deploying and operating such equipment in the vicinity of a plant without being detected, and because the effects of this type of equipment are low level and highly localized.

The NRC considers this issue to be technically resolved with the issuance of the final report, NUREG/CR-3069, "Interaction of Electromagnetic Pulse with Commercial Nuclear Power Plant Systems," and is included in the report to the staff on EMP, SECY-82-367. The results indicate that commercial nuclear power plants are invulnerable to EMP and that there is nothing affected that impacts any systems required for safe shutdown of the plant.

WAPWR Response

Given the above resolutions, this issue has no impact on the WAPWR design.

21. Issue 21: Vibration Qualification of Equipment

Discussion

The dynamic qualification of equipment consists primarily of seismic qualification. For boiling water reactor Mark II and III plants, equipment is also qualified to withstand the hydrodynamic loads associated with discharge into the suppression pool. In addition to the suppression pool hydrodynamic loads, the NRC has become concerned that other vibrations and accident-induced dynamic loads may have a noticeable effect on the functional capability of safety-related mechanical or electrical equipment. These dynamic loads may not have been taken into consideration by the industry in their present qualification program. In the past it has been generally accepted that seismic qualification of equipment is sufficient to cover the effects of other undefined vibratory loads that may occur during the life of a plant. Information is needed to define the anticipated vibratory environment in various locations of a plant during accident conditions and to determine whether such environments exceed the design basis envelope for the installed equipment. The currently pending Mechanical Equipment Qualification Rulemaking will provide further NRC guidance on this issue (refer to Section 6.1.2.3, item 5).

The current Westinghouse practice addresses the effect of vibratory loads on mechanical and electrical equipment. Equipment which is line mounted incorporates normal operating vibration into the equipment qualification aging sequence. The testing and analysis of line mounted equipment includes an addressment of hydrodynamic loads resulting from blowdown as well as those vibratory loads which result from earthquakes. Nonline mounted equipment is protected from blowdown loads as required and will include 5 operating basis earthquakes in the equipment qualification aging sequence. Additionally, this equipment will be subject to safe shutdown earthquake loads following equipment aging.

This issue is complete for plants undergoing licensing review (under Standard Review Plant 3.10 which requires applicants for operating licenses to address areas of vibration sensitivity as part of their seismic qualification program) and is also complete for operating plants (as part of an existing programs rulemaking in conjunction with Unresolved Safety Issue A-46).

WAPWR Response

The WAPWR will address this issue by testing and analysis as described above. There is no further impact on the WAPWR design.

22. Issue 22: Inadvertent Boron Dilution Events

Discussion

Many pressurized water reactors have no positive means of detecting boron dilution during cold shutdown. Some operations carried out during outages (e.g., steam generator decontamination) reduce the reactor coolant system volume, thus speeding up dilution. Boron dilution has taken place during such operations although, thus far, criticality has not occurred.

The fix is to install instrumentation to detect the event and stop the dilution either automatically or, if the detection is sufficiently early, by alerting the operator.

WAPWR Response

The WAPWR protection system design will consider the impact of boron dilution and the event will be factored into the design.

23. Issue 23: Reactor Coolant Pump Seal Failures

Discussion

This issue deals with an unexpectedly high rate of failures of reactor coolant pump seals in pressurized water reactors. A seal failure results in a primary coolant leak (i.e., a very small LOCA).

The results reported in WASH-1400, "Reactor Safety Study - An Assessment of Accident Risk is U.S. Commercial Nuclear Power Plants," indicated that a break in the reactor coolant pressure boundary having an equivalent diameter in the range of 0.5 to 2 inches was a significant cause of a core melt. Since the current study shows that comparable break flow rates have resulted from reactor coolant pump seal failures at a frequency about an order of magnitude greater than the pipe break frequency used in WASH-1400, the overall probability of core melt due to these small-size breaks could be dominated by events such as pump seal failures if the WASH-1400 assessment is correct. Using the current estimates of seal failures rates and WASH-1400 scenarios for core melts induced by small LOCAs, the NRC estimates a core melt frequency of approximately 10^{-4} per reactor year.

For ranking purposes, NRC is interested primarily in the frequency of seal failures which result in the release of radioactivity. Seal failure is involved in many accident sequences, which lead to a spectrum of releases.

Possible solutions to this issue include special detectors that signal high leakage, more frequent seal replacement, new seal designs, and more smoothly running pumps that take longer to mechanically degrade the seals.

This issue also is incorporated in the probability considerations of Unresolved Safety Issue A-44, "Station Blackout" (see Section 4, Item 22).

One of the objectives of the WAPWR design is to provide better protection from small LOCAs. In a general risk sense, the contributions of small LOCA to the overall risk for the WAPWR will be smaller than for WASH-1400.

The WAPWR design will directly address the problem of reactor coolant pump seal failures. Improved instrumentation and fluid systems design will improve the normal operating reliability associated with the seal injection system. Methods of improving the overall reliability of the seals will also be investigated. Finally, a large contributor to seal failure is the loss of seal injection capability. An alternate source of seal injection water will be incorporated into the chemical and volume control system as a redundant means of providing seal injection water on loss of component cooling water and normal seal injection.

WAPWR Response

Increased fluid systems reliability, the availability of an alternate source of seal injection water, and better small-break LOCA behavior will eliminate concerns over reactor coolant pump seal failures in the WAPWR design.

24. Issue 24: Automatic Emergency Core Cooling System Switch to Recirculation

Discussion

The emergency core cooling system (ECCS) operation has two different phases, the injection phase and the recirculation phase. The first phase (injection) involves initial cooling of the reactor core and replenishment of the primary coolant following a LOCA, while the second phase (recirculation) provides long-term cooling during the accident recovery period. Switchover from the injection phase to the recirculation phase includes alignment of a number of valves to the recirculation position. Switchover can be achieved by a number of manual actions, by automating these actions or by automatic realignment of certain valves and manual completion of the switchover process. This last option is referred to as the semiautomatic option. The three switchover options (manual, automatic, and semiautomatic) are vulnerable with varying degrees to human errors, hardware failures as well as common cause failures. Moreover, an automatic system designed to control the whole switchover process or a portion of it can reduce the impact of operator error in executing the switchover. However, automatic systems may be subject to spurious actuation. Spurious switchover of ECCS and containment spray pump suction to a dry containment sump can result in pump damage and possible loss of safety function resulting in potentially unacceptable safety consequences. Review of past reactor experience indicated the existence of a significant number of ECCS spurious actuations and, in particular, four ECCS spurious automatic switchover actuations occurred in 1980 at Davis-Besse Nuclear Plant, Unit 1. Subject to the limitation of certain human factors assumptions, the automatic option provides minimum risk to the public. Moreover, it is this option which is apparently current practice in newer plants. Thus, unless Issue B-17, "Criteria for Safety-Related Operator Actions," (refer to Section 5.2, item 17) yields new information which modifies the human-factors assumptions, the NRC believes this issue can be considered resolved from a generic standpoint.

In the design of current plants, the switchover described above must be performed whether manually or automatically. For the WAPWR the problem of switchover for ECCS is eliminated. The emergency water storage tank inside containment establishes a continuous circulation path for safety injection with no actions either manual or automatic.

WAPWR Response

For the WAPWR there is no impact as a result of this issue as discussed above.

25. Issue 25: Automatic Air Header Dump on BWR Scram System

Discussion

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

26. Issue 26: Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power

Discussion

In a San Onofre Unit 1 Preliminary Notification issued in September 1980, it was reported that, during testing, the licensee had identified a problem with the design of the diesel generator sequencing circuitry. This problem occurred when a safety injection signal (SIS) was blocked, in accordance with the LOCA procedure, following safety injection initiation. Under these circumstances, a subsequent loss of offsite power could not produce automatic resequencing of safety injection loads onto the diesel generator supplied buses. This problem was the same one that was raised as Technical Issue 4 in NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRR Staff," and was considered by the NRC staff to be resolved on all operating plants. However, in view of the occurrence at San Onofre, it is believed that a deficiency in the process exists. Resolution of this generic issue will allow the NRC to document a satisfactory completion to Technical Issue 4 in NUREG-0138.

With a loss of offsite power subsequent to an SIS and a LOCA occurring after system level SIS reset and proper subsequent operator action, there would be no threat to public health and safety.

NUREG-0138 states that ample time would be available to reinitiate by operator action the SIS to pick up the LOCA loads on the diesel generator.

This issue is to be included as part of uncategorized issue 17, "Loss of Offsite Power Subsequent to a LOCA", discussed previously in this section (item 17).

WAPWR Response

The WAPWR emergency response guidelines will include procedures to address this event.

27. Issue 27: Manual Versus Automated Actions

Discussion

Plant design reviews and emergency operating procedures reviews have raised questions as to whether certain safety actions have to be accomplished automatically or whether manual operator action would be acceptable. There are no generally accepted criteria for safety-related operator actions and guidelines in current use are too ill-defined to form a basis for criteria. ANS-58.8 (ANSI N660), "Time Response Design Criteria for Safety-Related Operator Actions," is intended to fill this void and to serve as a basis for future designs.

This issue is included as part of Generic Safety Issue B-17 "Criteria for Safety-Related Operator Action" (refer to Section 5.2, item 17).

28. Issue 28: Pressurized Thermal Shock

Discussion

This issue is identified as Unresolved Safety Issue A-49, "Pressurized Thermal Shock (refer to Section 4.0, item 27).

29. Issue 29: Bolting Degradation or Failure in Nuclear Power Plants

Discussion

There are numerous bolting applications in nuclear power plants. The most crucial bolting applications are those constituting an integral part of the primary pressure boundary such as closure studs and bolts on reactor vessels, reactor coolant pumps, and steam generators. Failure of these bolts or studs could result in the loss of reactor coolant and thus jeopardize the safe operation of nuclear power plants. Other bolting applications such as component support and embedded anchor bolts or studs are essential for withstanding transient loads created during abnormal or accidental conditions.

In recent years, the number of bolting related incidents reported by the licensees of operating reactors and reactors under construction has increased. A large number of the reported bolting incidents are related to primary pressure boundary applications and major component support structures. Therefore, there is increasing concern regarding the integrity of the primary pressure boundary in operating nuclear power plants and the reliability of the component support structures following a LOCA or earthquake.

There has been a total of 44 bolting incidents reported. Most of these incidents were discovered either during refueling outages or scheduled inservice inspections or maintenance/repair outages. Therefore, such reported incidents have no immediate impact on public health and safety and the bolting incidents so far have not resulted in accidents. Degradation or failure of such studs and bolts constitutes a reduction in the integrity of the primary pressure boundary. Concern is compounded by the fact that there is currently no reliable NDE method to detect the cracking or degradation of such bolts or studs resulting from the principal modes of failure which are stress corrosion, fatigue, erosion corrosion, and boric acid corrosion.

Visual examination is currently the only reliable method to discover degradation by boric acid or erosion corrosion. In almost all cases this requires disassembly of the component in order to inspect the bolts or studs. If there is not clear evidence of boric acid leakage to the surroundings, bolting degradation by boric acid corrosion can potentially be undetected until the bolts or studs completely fail. Under the present inservice inspection program, visual inspection of bolts is not a mandatory requirement and UT inspection is not required on pressure-retaining bolts or studs with diameters less than 2 inches. A major accident such as a LOCA could conceivably occur due to undetected extensive bolting failure of the primary pressure boundary.

The NRC has expended no apparent additional effort beyond defining this issue as summarized above.

WAPWR Response

The objective of the WAPWR design is to minimize the number of bolts. Westinghouse will follow this issue and consider any recommendations which result from this effort and factor them into the WAPWR design as appropriate.

30. Issue 30: Potential Generator Missiles - Generator Rotor

Discussion

Generator rotor retaining ring failures can develop missiles that inflict considerable damage; missiles which can be ejected in an axial direction. The major cause of such a failure is attributed to brittle fracture at regions of stress concentration and stress corrosion cracking induced by the environment. An extensive review has been conducted by the NRC entitled "Potential Generator Missiles - Generator Rotor Retaining Rings," dated March 16, 1982.

WAPWR Response

The turbine-generator is outside the scope of the WAPWR Nuclear Power Block design, and as such, this issue has no impact on the WAPWR design.

31. Issue 31: Natural Circulation Cooldown

Discussion

This issue has arisen as a result of an incident that occurred at an operating pressurized water reactor a few years ago. While operating at full power on 6/11/80, one of the two containment isolation valves in the component cooling water (CCW) return line from the reactor coolant pumps (RCPs) at Saint Lucie failed closed causing a simultaneous loss of component cooling water to all reactor coolant pumps.

WAPWR Response

This issue, and its impact on the WAPWR design, is fully discussed in NRC Generic Letter 81-21 (see Section 6.4, Item 21).

32. Issue 32: Flow Blockage in Essential Equipment Caused by Corbicula

Discussion

This issue deals with fouling problems in the service water system, and the assessment of the adequacy of each operating plants preventative maintenance and surveillance programs for the service water system.

This issue has not been further defined or prioritized by the NRC.

WAPWR Response

For the WAPWR design service water system design appropriate maintenance procedures will be defined.

33. Issue 33: Connecting Atmospheric Dump Valve Opening Upon Loss of Integrated Control System Power

Discussion

This issue identifies a situation in which the failure of the nonnuclear instrumentation and integrated control system (NNI/ICS) power supply coupled with a lack of position indication on the atmospheric dump valve which automatically opens to the fifty percent open position on loss of NNI/ICS power, could significantly aggravate an overcooling transient.

This issue has not been further defined or prioritized by the NRC.

WAPWR Response

For Westinghouse pressurized water reactor designs, including the WAPWR, the atmospheric dump valves do not open to the fifty percent open position upon loss of NNI/ICS power. They remain closed. Therefore, this issue is not applicable to the WAPWR.

34. Issue 34: Reactor Coolant System Leak

Discussion

This issue is a result of an incident that occurred at the H. B. Robinson plant four years ago. Following a spurious safety injection, the plant operators initiated actions to bring the plant to hot shutdown. During automatic isolation of the CVCS letdown line due to the safety injection, it is believed that the outermost isolation valves closed faster than the two open orifice isolation valves or that leakage past the orifice isolation valves, resulted in opening of the relief valve and rupturing the isolation valve bellows. Also, a pressure surge due to the isolation valve closing caused a drain line cap to blow off.

WAPWR Response

This issue, and its impact on the WAPWR design, is fully discussed in Section 6.4, Item 22.

35. Issue 35: Degradation of Internal Appurtenances in LWRs

Discussion

This issue deals with loose parts in the primary system. From time to time, loose parts have been transported through a portion of the primary side system only to become lodged in some unidentified location before causing any damage. In the event of a steamline break, the resulting pressure transient in the primary side could cause a loose part to become dislodged, travel to the steam generator and cause a small break LOCA. Internal appurtences such as flow straighteners, orifices, thermal sleeves, screens, etc. have the potential to break loose and become "loose parts" in the fluid system.

This issue has not been further defined or prioritized by the NRC.

WAPWR Response

As this issue evolves, Westinghouse will consider and factor into the WAPWR design any NRC recommendations which result, as deemed appropriate.

36. Issue 36: Loss of Service Water

Discussion

This issue is concerned with the failure of a nonsafety related component which could cause the disablement of both redundant trains of the safety related service water system. The loss of instrument air and the loss of offsite power are also being considered in conjunction with this event.

This issue has not been further defined or prioritized by the NRC.

WAPWR Response

As this issue evolves, Westinghouse will consider and factor into the WAPWR design any NRC recommendations which result, as deemed appropriate.

37. Issue 37: Steam Generator Overfill and Combined Primary and Secondary Blowdown

Discussion

This issue has not been defined or prioritized by the NRC. The issue of steam generator overfill is discussed in NRC Generic Letter 81-28. (Section 6.4, Item 29).

38. Issue 38: Potential Recirculation System Failure as a Consequence of Injection of Containment Paint Flakes or Other Fire Debris

Discussion

This issue has not been defined or prioritized by the NRC. However, this issue may be related to Section 4.0, item 21.

39. Issue 39: Potential for Unacceptable Interaction Between the Control Rod Drive System and Nonessential Control Air System

Discussion

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

40. Issue 40: Breaks in the BWR Scram System

Discussion

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

41. Issue 41: BWR Scram Discharge Volume Systems

Discussion

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

42. Issue 42: Combination Primary/Secondary LOCA

Discussion

This issue has not been defined or prioritized by the NRC.

43. Issue 43: Contamination of Instrument Air Lines

Discussion

This issue has not been defined or prioritized by the NRC.

44. Issue 44: Failure of Saltwater Cooling System

Discussion

This issue has not been defined or prioritized by the NRC.

45. Issue 45: Inoperability of Instrumentation Due to Extreme Cold Weather

Discussion

This issue involves an assessment of the measures taken to protect instrumentation from severe weather and to verify the condition and operability of heat tracing systems and other measures taken to protect plant equipment from severe weather.

WAPWR Response

As this issue evolves, Westinghouse will consider and factor into the WAPWR design any NRC recommendations which result, as deemed appropriate.

46. Issue 46: Loss of 125 Volt D.C. Bus

Discussion

This issue has not been defined or prioritized by the NRC. However, it appears to be encompassed by Section 6.5, Item 1

47. Issue 47: Loss of Offsite Power

Discussion

Although this particular issue has not been defined by the NRC, it is perhaps encompassed by Unresolved Safety Issue A-44, "Station Blackout" (Section4.0, Item 22).

48. Issue 48: LCO for Class IE Vital Instrument Buses in Operating Reactors

Discussion

This issue has not been defined or prioritized by the NRC. However, it appears to be encompassed by Section 6.5, Item 1.

49. Issue 49: Interlocks and LCOs for Redundant Class IE Tie Breakers

Discussion

This issue is concerned with providing interlocks to prevent the inadvertent closure of the single tie breaker between Class IE buses. These tie breakers provide a means to supply power to a bus from the opposite train under certain maintenance conditions, and are not required for safety.

WAPWR Response

This issue is encompassed by Section 6.5, Item 1.

50. Issue 50: Reactor Vessel Level Instrumentation in BWRs

Discussion

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

51. Issue 51: Proposed Requirements for Improving Reliability of Open Cycle Service Water System

Discussion

This issue has not been defined or categorized by the NRC.

52. Issue 52: SWS Flow Blockage by Blue Mussels

Discussion

This issue has not been defined or categorized by the NRC.

53. Issue 53: Consequences of a Postulated Flow Blockage Incident in a Boiling Water Reactor

Discussion

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

54. Issue 54: Survey of Valve Operator Related Events Occurring During 1978, 1979, and 1980

Discussion

This issue deals with internal NRC evaluation of an NRC report, "Survey of Valve Operator Related Events Occurring During 1978, 1979, and 1980." This issue has not been further defined or prioritized by the NRC.

55. Issue 55: Failure of Class IE Safety Related Switchgear Circuit Breakers to Close on Demand

Discussion

This issue has not been defined or categorized by the NRC.

56. Issue 56: An Analysis of the Abnormal Transient Operating Guidelines

Discussion

This issue has not been defined or categorized by the NRC.

57. Issue 57: Effects of Fire Protection System Actuation on Safety Related Equipment

Discussion

In its continuing review of licensee event reports (LERs) the NRC has identified actuation resulted in degrading or jeopardizing the operability of systems important to safety. In some instances the suppression system actuated properly, in response to a valid signal. In other instances there was no real need for initiation. In these latter instances, there does not appear to have been a single common causative factor. It appears that errors have been made in design (including selection of the most appropriate sensors), in installation, and in plant operating and maintenance procedures. The NRC is concerned that fire fighting systems and activities, if not properly designed and implemented, can contribute to risks to the plant and public.

General Design Criterion 3, Fire Protection, of Appendix A to 10CFR Part 50 states in part: "Fire detection and fighting systems to appropriate capacity and capability shall be provided and designed to minimize the adverse effects on structures, systems and components important to safety. Fire fighting systems shall be designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems and components." Paragraph CMEB 9.5-1 requires that a fire hazard analysis be performed to assess the probability and consequences of fires in each utilization facility. This analysis, in considering the consequences of a postulated fire, must include the effect of fire fighting activities. Such an analysis need not be complex, but should not be limited to a "paper study". The events reported indicate that a walk-down of plant equipment would have identified instances where minor modifications such as shielding equipment and sealing conduit ends would have reduced water damage from inadvertent operation of the fire protection system, without significantly reducing its effectiveness. It appears that in many instances, the hazards analysis did not adequately address system interactions between fire suppression systems and systems important to safety, particularly those necessary for safe shutdown. The overall design must accommodate both needs; that it, it must provide an effective fire protection system but not adversely affect other aspects of plant safety.

This issue has not been further defined or prioritized by the NRC.

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.

58. Issue 58: Inadvertent Containment Flooding

Discussion

This issue was raised as a result of the following incident that occurred at a PWR a few years ago.

Upon entry for repair of a nuclear instrument, it was discovered that several inches of water had accumulated on the containment floor without the operator's knowledge. The flooding event resulted from a combination of conditions: service water leaks from piping and fan coolers; inoperable containment sump pumps; two containment sump level indicators not recognized by the operators; no high water level alarm; and high moisture levels due to an error in calibration of the moisture level indicators.

The NRC stated that acceptance criteria needed to be developed that will prescribe more comprehensive requirements for leak detection provisions, operator actions, surveillance procedures, and maintenance practices than those currently in place. Review and application of these criteria to each plant must then be accomplished on an individual plant basis with a decision in each case regarding backfitting.

IE Bulletin 80-24 was issued in response to this containment flooding incident. This bulletin required that all plants with open-cooling water systems take a number of short-term actions to preclude this type of event in the interim before longer term generic actions are accomplished. The actions in the IE Bulletin are still in place pending long-term resolution of the flooding issue.

WAPWR Response

See Section 6.5.1, item 14 for a discussion of this issue and its impact on the WAPWR design.

59. Issue 59: Technical Specification Requirements for Plant Shutdown When Equipment for Safety Shutdown is Degraded or Inoperable

Discussion

This issue is concerned with equipment failure resulting in impairment of the capability to take the plant to a shutdown condition where the Technical Specifications required that the plant be shutdown in a short time period.

This issue has not been further defined or prioritized by the NRC.

60. Issue 60: Lamellar Tearing of Reactor Systems Structural Supports

Discussion

Lamellar tearing results in almost all cases from limitations in steel plate introduced during manufacture.

WAPWR Response

This issue is encompassed by Unresolved Safety Issue (USI) A-12 (Section 4.0, Item 12).

61. Issue 61: SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments

Discussion

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

62. Issue 62: Reactor System Bolting Applications

Discussion

The NRC currently provides no bolting control regulations or guides for bolting other than reactor vessel head bolting (Regulatory Guide 1.65). There have been failures of other bolting which were probably preventable. Preparation of stress corrosion limit curves for various materials in various environments is recommended to resolve this safety concern.

This issue has not been further defined or prioritized by the NRC.

WAPWR Response

This issue appears to be encompassed by Section 5.5, item 29.

63. Issue 63: Use of Equipment Not Classified as Essential to Safety in BWR Transient Analyses

Discussion

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

64. Issue 64: Identification of Protection System Instrument Sensing Lines

Discussion

This issue has not been defined or prioritized by the NRC.

65. Issue 65: Probability of Core Melt Due to Component Cooling Water System Failures

Discussion

This issue has not been defined or prioritized by the NRC.

WAPWR Response

The potential for component cooling water system failure and its contribution to core melt will be factored into the WAPWR probabilistic risk assessment.

66. Issue 66: Steam Generator Requirements

Discussion

This issue has not been defined or prioritized by the NRC.

67. Issue 67: Steam Generator Staff Actions

Discussion

This issue has not been defined or prioritized by the NRC.