4.0 UNRESOLVED SAFETY ISSUES

The NRC continuously evaluates the safety requirements used in their reviews against new information as it becomes available. Information related to the safety of nuclear power plants can come from a variety of sources including experience from operating reactors; research results; NRC staff and Advisory Committee on Reactor Safeguards (ACRS) safety reviews; and vendor, architect engineer, and utility design reviews. Each time a new concern or safety issue is identified from one or more of these sources, the need for immediate action to ensure safe operation is assessed by the NRC.

Is some cases, immediate NRC action is taken to ensure safety (such as the derating of boiling water reactors as a result of the channel box wear problems in 1975). In other cases, interim measures, such as modifications to operating procedures, may be sufficient to allow further study of the issue before NRC licensing decisions are made. However, in most cases the initial NRC assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. In any event, further study by the NRC may be deemed appropriate before judgments are made as to whether existing requirements should be modified to address the issue for new plants or if backfitting is appropriate for the long-term operation of plants already under construction or in operation.

These issues are called "generic safety issues" or "unresolved safety issues" and they do have a potential impact on all plant designs including the WAPWR design. NRC "generic safety issues" are discussed in Section 5.0. This section is devoted to the discussion of "unresolved safety issues".

The NRC defines an Unresolved Safety Issue as "a matter affecting a number of nuclear power plants that poses important questions concerning the adequacy of existing safety requirements for which a final resolution has not yet been developed and that involves conditions not likely to be acceptable over the lifetime of the plant it affects."

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Each year the NRC reviews their task action plans and generic issues to define a current set of Unresolved Safety Issues which is reported to Congress. These annual reports usually identify those Unresolved Safety Issues that were technically resolved from the previous annual report.

The purpose of this section is to assess each Unresolved Safety Issue relative to its impact, or potential impact, on the WAPWR design.

The following current list of Unresolved Safety Issues has been obtained from the NRC Generic Issues Branch:

- o Water Hammer (A-1)
- o Asymmetric Blowdown Loads on the Reactor Primary Coolant Systems (A-2)*
- o Westinghouse Steam Generator Tube Integrity (A-3)
- o Combustion Engineering Steam Generator Tube Integrity (A-4)
- o Babcock and Wilcox Steam Generator Tube Integrity (A-5)
- o Mark I Short Term Program (A-6)*
- o Mark I Long Term Program (A-7)*
- o Mark II Containment Pool Dynamic Loads (A-8)*
- o Anticipated Transients Without Scram (A-9)*
- o BWR Feedwater Nozzle Cracking (A-10)*

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* NRC technical resolution for each of these Unresolved Safety Issues has been issued.

- o Reactor Vessel Materials Toughness (A-11)*
- o Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports (A-12)
- o Systems Interactions in Nuclear Power Plants (A-17)
- o Qualification of Class 1E Safety-Related Equipment (A-24)*
- o Reactor Vessel Pressure Transient Protection (A-26)*
- o Residual Heat Removal Requirements (A-31)*
- o Control of Heavy Loads Near Spent Fuel (A-36)*
- o Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits for BWR Containments (A-39)*
- o Seismic Design Criteria Short Term Program (A-40)
- o Pipe Cracks in Boiling Water Reactors (A-42)*
- o Containment Emergency Sump Performance (A-43)
- o Station Blackout (A-44)
- o Shutdown Decay Heat Removal Requirements (A-45)
- o Seismic Qualification of Equipment in Operating Plants (A-46)
- o Safety Implications of Control Systems (A-47)
- o Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment (A-48)

o Pressurized Thermal Shock (A-49)

1. Issue A-1: Water Hammer

Discussion

Water hammer events are intense pressure pulses in fluid systems caused by any one of a number of mechanisms and system conditions.

Total elimination of water hammer occurrence is not feasible, due to the possible coexistence of steam, water, and voids in various nuclear plant systems. Experience shows that design inadequacies and operator or maintenance-related actions have contributed about equally to initiating water hammer occurrences.

Since 1969, approximately 150 water hammer events have been reported through the NRC's Licensee Event Reports (LERs). Damage has been limited principally to pipe support systems. Approximately half of these events have occurred either in the preoperational phase or the first year of commercial operation. This suggests a learning period exists in which design deficiencies are corrected and operating errors are reduced.

Water hammer frequency peaked in the mid-1970's, at a time when the rate of introducing new plants into commercial operation was the highest.

The major conclusions reached are that the frequency and severity of water hammer occurrence can be and to some extent have been significantly reduced through design features such as the use of "J" tubes on the feedwater ring, keep-full systems, vacuum breakers, void detection systems and improved venting procedures, proper design of feedwater valves and control systems and increased operator awareness and training. The water hammer issue is less significant then had originally been thought.

The most common cause of water hammer events is line voiding. Other significant causes include steam condensation and feedwater control valve

instability. Although these are the generic causes, many of the events have resulted from both design and operational deficiencies.

No water hammer incidents have resulted in the loss of containment integrity or the release of radioactivity outside the plant. The frequency and severity of events in PWR systems are low, with the exception of steam generator water hammer and feedwater-control valve-induced water hammers.

NUREG-0918, "Prevention and Mitigation of Steam Generator Water Hammer Events in PWR Plants", November 1982, presents plans for mitigation of SG water hammer.

NUREG-0927, "Evaluation of Water Hammer Experience in Nuclear Power Plants" (May 1983 for comment) collects and qualifies reported cases to date.

The NUREG provides guidance for avoiding water hammer. In addition, the NRC has added statements to the following SRPs for future plants that direct attention to the prevention of water hammer. SRP 5.4.6, Rev. 3; 5.4.7, Rev. 3; 6.3 Rev. 2; 9.2.1 Rev. 3; 9.2.2 Rev. 2; 10.3 Rev. 3 and 10.4.7 and Rev. 3.

Other than on-going modifications of specific vendor designs there will be no generic backfit. Operator training feedback is accomplished in accordance with TMI action plan I.C.5. Risk analysis indicates that water hammer is not a significant contributor to overall risk.

Following the implementation of design features and testing contained in BTP ASB 10-2, "Design Guidelines for Avoiding Water Hammers in Steam Generators" (SRP 10.4.7), the frequency of steam generator water hammer in top feedring design steam generators has been essentially eliminated. Additional review of water potential for bottom feed (preheat) steam generators is in process.

Recommendations include:

- o Prevent or delay water draining from the feedring following a drop in steam generator water level by means such as J-tubes.

- o Minimize the volume of feedwater piping external to the steam generator which could pocket steam using the shortest possible (less than 7 feet) horizontal run of inlet piping to the steam generator feedring.

- o Perform tests acceptable to the NRC to verify that unacceptable feedwater line water hammer will not occur using the plant operating procedures for normal and emergency restoration of steam generator water level following loss of normal feedwater and possible draining of the feedring. Provide the procedures for these tests for approval before conducting the tests.

With regard to the protection against other potential water hammer events currently provided in plants, piping design codes require consideration of impact loads. Approaches used at the design stage include:

- o Increasing valve closure times
- o Laying out piping to preclude water slugs in steam lines and vapor formation in water lines
- o Using snubbers and pipe hangers
- o Using vents and drains

In addition, the NRC requires that an applicant conduct a preoperational vibration dynamic effects test program in accordance with Section III of the ASME Code for all ASME Class 1 and Class 2 piping systems and piping restraints during startup and initial operation. These tests are intended to provide adequate assurance that the piping and piping restraints have been designed to withstand dynamic effects due to valve closures, pump trips, and other operating modes associated with the design operational transients.

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

Special valve designs are being evaluated for the WAPWR design to reduce the probability of water hammer events in feedwater systems due to rapid motion of check valves. Also, feedwater piping configurations will be designed to minimize the potential for water hammer.

The current NRC criteria related to water hammer do not require any additional measures beyond those already implemented in current Westinghouse designs. However, additional design features have been incorporated into or are being evaluated for inclusion in the WAPWR design, as summarized above.

WAPWR Response

- Westinghouse will completely document and justify any deviations from the acceptance criteria in the above mentioned standard review plan sections during the licensing process for the WAPWR design.

- Appropriate testing will be performed to verify that unacceptable feedwater line water hammer will not occur.

- o During the design and licensing process, an analysis will be performed to demonstrate the inherent capability of the WAPWR design to preclude water hammer.

- o New design features (as discussed above) will be incorporated into the WAPWR design to further eliminate the probability of water hammer events.

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

Discussion

This issue concerns asymmetric loadings which could act on the reactor's primary system as the result of a postulated double-ended rupture of the piping in the primary coolant system. The magnitude of these loads is potentially large enough to damage the supports of the reactor vessel, the reactor internals, and other primary components of the system. Therefore, the NRC initiated a generic study to gain a better understanding of these loads and to develop criteria for an evaluation of the response of the primary systems in pressurized water reactors to these loads.

The NRC has completed its investigation and issued NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems." This report provides an acceptable basis for performing plant analyses for asymmetric loss-of-coolant accident loads. Guidelines and criteria are given for the evaluation of the loading transients, structural components, and fuel assemblies.

Westinghouse has addressed the subject of asymmetric loss-of-coolant accident loads in the design and analysis of plants currently under NRC review. These analyses have demonstrated compliance with NUREG-0609.

As discussed in Section 5.1 (item 18), Westinghouse has performed extensive material testing and fracture mechanics evaluations to demonstrate that pipe breaks need not be postulated in the reactor coolant system. Both the NRC and ACRS have endorsed this concept and the methodology developed to support it.

WAPWR Response

During the licensing process for the WAPWR, Westinghouse intends to apply a revised pipe break criteria to all WAPWR high energy fluid systems in order to reduce/eliminate the need to postulated pipe breaks. These criteria do not require pipe breaks to be postulated in high energy fluid system piping unless some mechanism (e.g., corrosion, water hammer) exists which could result in a pipe break. The bases for this position are the material properties of the piping and the methodology previously developed for reactor coolant system piping.

With the elimination of pipe breaks Westinghouse intends to eliminate the structural effects considered in the piping structural evaluation. Some of these structural effects include blowdown loads and jets from previously postulated pipe breaks; pipe whip restraints on piping; and pressurization effects from previously postulated pipe breaks. In addition analyses will be performed to demonstrate that the structural integrity of the reactor vessel supports, reactor internals, and other primary system components are maintained within acceptable limits.

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

Discussion

The NRC combined into one unresolved safety issue the steam generator tube integrity concerns for the three steam generator suppliers. The primary concern is the capability of steam generator tubes to maintain their integrity during normal operation and postulated accident conditions. Tube degradation has been observed in many steam generators resulting from chemistry related corrosion and tube vibration.

An original report (NUREG-0844) was scheduled for release for public comment in November 1981. The report was delayed when the Ginna tube rupture event occured in January 1982 causing the NRC to review the whole issue again.

The NRC formed a task force under the Division of Licensing to prepare its proposed requirements regarding steam generator tube integrity. These requirements address new concerns resulting from the Ginna tube failure (such as loose parts in the secondary system and plant response to steam generator tube failure) and also corrosion related failure mechanisms. The recommendations prepared by the staff under USI A-3, 4, 5 were primarily concerned with corrosion mechanisms such as wastage and denting. Consequently, as discussed with the Commission on June 30, 1982, the requirements from the USI program are incorporated in the overall set of requirements being developed to address tube failures.

The NRC issued NUREG-0909 "NRC Report on the January 25th, 1982 Steam Generator Tube Rupture at the R. E. Ginna Nuclear Power Plant," April 1982, via Generic Letters 82-07 and 82-11 "Safety Evaluation Report Related to the Restart of the R. E. Ginna Nuclear Power Plant," May 1982 (see Section 6.4, items 48 and 52). As a result of the Ginna event, the NRC issued a set of proposed steam generator generic requirements in July 1982 which were aimed at improving tube integrity. The NRC requested additional information from utilities on tube integrity via Generic Letter 82-22 (see Section 6.4, Item 63) and provided via Generic Letter 82-32 (see Section 6.4, Item 73) a value-impact analysis on the proposed steam generator requirements that was performed under NRC contract by Science Application's, Inc.

"Draft Resolution of Unresolved Safety Issues A-3, A-4 and A-5 Regarding Steam Generator Tube Integrity." NUREG-0844 was rewritten to reflect lessons learned by the Ginna event and is being reviewed prior to issuance. A number of the proposed requirements in addition to those contained in the Generic Letter are expected to be issued under 10CFR50.54f to all PWR plants in CFR, regulatory guide or SRP form.

The steam generator design for the WAPWR will consider features to minimize tube integrity problems, such as:

(a,c)

Regulatory Guide 1.83, Revision 1, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," establishes guidelines for developing a program for inservice inspection of steam generator tubing. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," describes a method acceptable to the NRC for establishing the limiting safe conditions for tube degradation.

As currently defined, the NRC activity related to this issue is expected to result in the refinement of NRC requirements for inservice inspection of steam generator tubes (e.g., mandatory inspections, secondary water quality control, leakage limits, safety injection signal reset action, etc.). The WAPWR steam generator design is not expected to be impacted by this issue.

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Regulatory Guide 1.83 and 1.121 positions during the licensing process for the WAPWR design. In addition, Westinghouse will closely follow the above mentioned regulatory requirements in this area. Appropriate design features (such as those listed above) will be included in the WAPWR steam generator design to minimize tube integrity problems.

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

Discussion

This issue was combined with issue A-3 by the NRC, but is not applicable to Westinghouse steam generator designs (see Item 3 above).

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

Discussion

This issue was combined with issue A-3 by the NRC, but is not applicable to Westinghouse steam generator designs (see Item 3 above).

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

Discussion

Nuclear plants have safety and control systems to limit the consequences of temporary abnormal operating conditions or "anticipated transients". Some deviations from normal operating conditions may be minor; others, occurring less frequently, may impose significant demands on plant equipment. In some anticipated transients, rapidly shutting down the nuclear reaction (initiating a "scram"), and thus rapidly reducing the generation of heat in the reactor core, is an important safety measure. A potentially severe "anticipated transient" where the reactor shutdown system does not "scram" as desired, is an "anticipated transient without

scram", or ATWS. This issue has been discussed throughout the nuclear industry for a number of years. Historically, the NRC staff has excluded very low probability events from the design basis. At issue in the ATWS discussions is whether or not the probability of an ATWS event is sufficiently low to warrant the continuance of the current NRC staff practice with regard to ATWS, (i.e., continued exclusion from the design basis for nuclear power plants because of its low probability).

There have been numerous NRC publications on the ATWS issue (e.g., WASH-1270, NUREG-0460, and SECY-80-409). Salient positions from all of these documents have been incorporated into a recently proposed rule, "Standards for the Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants," (46FR57521 dated November 24, 1981). In this proposed rule the Commission is considering three alternatives, as shown below, for amending its regulations to require improvements in the design and operation of nuclear power plants to reduce the likelihood of failure of the reactor protection system to shutdown the reactor following anticipated transients and to mitigate the consequences of ATWS events.

	Staff	Hendrie	Utility Group
Analyses required	Meet acceptable performance criteria on ATWS accident.	Incorporate a reliability program.	None proposed, based on a PRA performed.
Diverse scram system	Probably required for plants after 1969.		Westinghouse plants exempted.
Other requirements	Scram discharge volume fix for BWRs, features: - probably SLCS for BWRs - AMSAC for BWRs	Scram discharge volume fix for BWRs, features: - SLCS for BWRs AMSAC for BWRs	Scram discharge fix, autostart mitigating features: - AMSAC for BWRs

PRA - Probabilistic risk assessment
SLCS - Standby liquid control system
AMSAC ATWS mitigating system actuation circuitry.

For more than 10 years the NRC and the industry have developed information on ATWS including the design of proposed mitigating features, probability nd consequences analyses. The first documented ATWS event occurred at Sa. n I on February 25, 1983 and plant records indicated that a similar event occurred four days before. The event consequences were mild but the impact on the subjective consideration of ATWS was great. An existing NRC task force on ATWS expanded its investigation to include an in-depth evaluation of the event and to make recommendations on the regulatory position. The failure involved two independent undervoltage trip breakers with common cause implications due to the design sensitivity to lack of proper maintenance practices. NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant," April 1983, summarizes current status and includes the following draft rule recommendations:

1. Each pressurized water reactor must have a system, diverse and independent from the reactor protection system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip.

2. Each pressurized water reactor manufactured by Combustion Engineering or by Babcock and Wilsox must have a diverse scram system. This scram system must be independent from the reactor protection system.

3. Each boiling water reactor must have an alternate rod injection (ARI) system that includes separate sensors and logic and redundant scram air header exhaust valves.

4. Each boiling water reactor must have a standby liquid control system (SLCS). The flow capacity and boron content of the system must be at least equivalent in control capacity to 86 gpm of a 13 percent sodium pentaborate solution. This requirement can be satisfied by modifying existing SLCS piping to allow the use of two existing pumps simultaneously, by modifying the SLCS storage tank and piping header system to allow the use of high-concentration or isotopically enriched boron, or by other means reviewed and accepted by NRC.

5. Each boiling water reactor must have equipment of high reliability to automatically trip the reactor coolant recirculating pumps under conditions indicative of an ATWS.

6. Each boiling water reactor must have a scram discharge volume system of sufficient capacity to received water exhausted by a full reactor scram.

7. Each operating licensee shall implement procedures and training programs for mitigating an ATWS.

The NRC intends to issue a final rule based on the above recommendations and intends to issue a proposed rule subject to public review and comment that would require a diverse scram system on Westinghouse plants commensurate with the other PWRs as given in item 2 above. In addition, the NRC has issued Generic Letter 83-28 "Required Actions Based on Generic Implications of Salem ATWS Events." The actions related to the reactor trip system. The actions areas are noted below.

Post-Trip Review (Program Description and Procedure)

Equipment Classification and Vendor Interface (Reactor Trip System Components)

Post-Maintenance Testing (Reactor Trip System Components)

Reactor Trip System Reliability (Vendor-Related Modifications)

Reactor Trip System Reliability (Preventive Maintenance and Surveillance Program for Reactor Trip Breakers)

Reactor Trip System Reliability (Automatic Actuation of Shunt-trip Attachment for Westinghouse and B&W plant)

The schedules for implementation of these actions will be combined with all other existing plant programs. Therefore, schedules for implementation of these actions will be negotiated between the NRC and licensees.

WAPWR Response

The WAPWR will be designed with an ATWS mitigating system which generates a turbine trip and emergency feedwater start signal independent of the integrated protection system. The system will be designed to be safety-grade to the maximum extent feasible from a cost-benefit viewpoint, and will otherwise be highly reliable.

In addition, the WAPWR design will contain new features that provide additional mitigation capability for ATWS events. Specifically, the new core design will have a more negative MTC for most of the plant lifetime, relative to current designs. Also, ATWS considerations will be factored into the sizing and number of pressurizer power-operated relief valves.

A detailed safety analysis of the limiting ATWS events will be performed during the WAPWR design and licensing process to demonstrate that ATWS acceptance criteria are met.

10. Issue A-10: BWR Feedwater Nozzle Cracking

Discussion

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

11. Issue A-11: Reactor Vessel Materials Toughness

Discussion

Steel commonly used in the construction of reactor pressure vessels (RPV) exhibits fracture toughness that varies greatly with temperature. Steel has relatively high toughness at high temperatures but low toughness at low temperatures. The temperature or temperature range where the transition from high-toughness (ductile) to low-toughness (brittle) behavior occurs is commonly referred to as the ductile-brittle transition temperature. Thus the temperature-dependent fracture toughness has three more-or-less distinct zones: a lower shelf with low toughness, an intermediate transition region, and an upper shelf with high toughness.

Charpy-impact (C_v) test data in the form of specimen-fracture energy, as a function of temperature, reflect the ductile-brittle transition. The transition temperature can be identified in several ways, the simplest of which is to report the temperature at an arbitrary C_v energy level (for example, 35 ft-lb). The upper shelf energy is the energy level of the upper asymptote of the $E_{Cv} = f(T)$ curve.

The embrittling effect of neutron radiation may so change the mechanical properties that the steel in a RPV would fail to meet the toughness requirements of 10CFR50. This could result from either too large a temperature increase in the reference-transition temperature (RT_{NDT}), or too large an energy decrease in the C_v upper-energy level, or both. The magnitude of the irradiation-induced changes depends, among other things, on the chemistry and metallurgical condition of the steel. The effect of copper content can be singled out because it plays a major role in the behavior. Copper was introduced by the practice (later abandoned) of coppercoating the consumable electrode weld wire to protect it from rusting and to increase its electrical conductivity. Experiments have shown that the radiation-induced changes in both the transition temperature and the C_v, increase with copper content and the most sensitive steels include weld metals with relatively high-copper content. Because some high copper

welds exhibited relatively low initial upper shelf energy levels, it was found to be more significant with respect to violation of regulatory requirements than the corresponding transition temperature increase.

Regulatory Guide 1.99 (Rev. 1) shows conservative measures of the changes in transition temperature and upper shelf with fluence, copper and phosphorus contents are shown parametrically. The guide is updated as significant additional data from surveillance or test reactor programs become available. Conservatism was included by constructing the curves as upper bounds of property changes rather than averages.

Guidance for licensees to provide justification for continued operation is given in NUREG 744, Rev. 1 (Generic Letter 82-26). In accordance with the requirements of 10CFR50, Appendix G, all licensees should take the following course of action. The upper shelf energy at the plant-specific end of life (EOL) should be established in accordance with 10CFR50 and the ASME code. If the EOL upper shelf energy $\geq$ 50 ft-lb, the reactor pressure vessel is acceptable (other factors, detailed in 10CFR50 and in the Code, remain in force). If the EOL upper shelf energy $\leq$ 50 ft-lb, either a safety analysis should be performed to demonstrate that the vessel can operate with adequate margin or a thermal anneal could be performed to restore the material toughness. To be acceptable, the analysis must show adequate margin under normal, upset, emergency, faulted, and test conditions. The analysis may follow either the method recommended by the NRC or a method of equal or better reliability.

Appendix G to 10CFR50 essentially adopts the method of ASME Code Appendix G, with additional restrictions related to the presence of fuel or criticality. However, 10CFR50, Appendix G, extends the applicability of the design rules to operations, and fluence effects that must be considered. Because the resulting pressure and temperature limitations must be included in the plant Technical Specification, which controls plant operation, the 10CFR50 Appendix G rules apply to all operating plants.

The need to include rules for emergency and faulted condition control in the ASME Code, Section III, Appendix G, is not clear. The Section III rules are of value only to the extent that they influence the construction and it is not apparent that such rules would have that effect. Although material selection might be influenced, indications are that the current acceptance criteria are satisfactory in that they provide adequate life-time fracture resistance.

Results from reactor vessel surveillance programs indicates that as many as 20 operating PWRs will have beltline materials with marginal toughness, relative to the requirements of Appendices G and H of 10CFR50, after comparatively short (approximately 10 effective-power years) periods of operation. The specific requirement that may be voided is that of paragraph V.B, Appendix G, 10CFR50. For vessels failing to satisfy that requirement, paragraph V.C.3, Appendix G, 10CFR50, must be satisfied (along with the rest of V.C); that is, the owner must perform an analysis demonstrating the existence of adequate operational safety margins against fracture. For plants currently under licensing review, reactor vessels generally have acceptable fracture toughness. However, a few plants under licensing review have reactor vessels that have been identified as having the potential for marginal fracture toughness within their design life; these vessels will have to be reevaluated in the light of the new criteria for long-term acceptability.

Techniques for periodic surveillance of reactor vessel welds are discussed in Regulatory Guide 1.150, Rev. 1, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations."

For the WAPWR design the core baffle/reflector region has been optimized to provide increased shielding of the reactor vessel to significantly reduce neutron irradiation in the reactor vessel beltline region to less than that of the best currently operating plants. The reduced neutron irradiation will lead to increased fracture toughness of the material in the reactor vessel beltline. Residual copper content of the WAPWR reactor

vessel beltline material will be at or below that specified for the most recent Westinghouse reactor vessels. Residual copper content is a key contributor to the loss of reactor vessel material toughness in the presence of neutron irradiation.

WAPWR Response

The design features summarized above will ensure that the WAPWR reactor pressure vessel will maintain high fracture toughness properties throughout plant life, and thus, will not require additional analysis under 10CFR Part 50, Appendix G. Therefore, final resolution of this issue has no additional impact on the WAPWR design.

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

Discussion

This issue deals with the potential for lamellar tearing and low fracture toughness of the steam generator and reactor coolant pump support materials.

NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," currently in draft form, is expected to be formally issued by the NRC by the end of 1983. This NUREG categorizes operating plants relative to the adequacy of the plant's steam generator and reactor coolant pump supports with respect to fracture toughness. In general, the conclusions of NUREG-0577 are that supports for the reactor coolant pumps and steam generators in recently licensed pressurizer water reactors have adequate fracture toughness. Westinghouse believes that designing and fabricating these supports in accordance with Subsection NF of Section III of the ASME Code provides adequate assurance of acceptable fracture toughness of materials, and ensures compliance with the "for comment" version of NUREG-0577.

A new SRP that endorses Subsection NF of Section III of the ASME Code is expected to be issued shortly. With respect to lamellar tearing, the current Westinghouse design for supports does not contain the thick, heavy weldments of the type possibly susceptable to lamellar tearing.

WAPWR Response

The WAPWR steam generator and reactor coolant pump supports will be designed and fabricated in accordance with Subsection NF of Section III of the ASME Code. Once finalized by the NRC, any new requirements of NUREG-0577 and the proposed SRP beyond ASME Code requirements will be reviewed for impact, and the level of compliance will be documented during the licensing process for the WAPWR design.

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

Discussion

Systems interactions are those events that can occur in a plant due to one or multiple systems or components acting upon one or more other systems in a manner not intended by design.

The design and analyses by the plant designers, and the subsequent review and evaluation by the NRC staff take into consideration the interdisciplinary areas of concern and account for systems interaction to a large extent. For example, national standards and regulatory criteria provide requirements that, if met, reduce the probability of adverse systems interactions. Examples (for standards) are those dealing with proper design to prevent failures of pressure boundaries, even under accident conditions (ASME Code, Section III), the single-failure criteria (ANSI ANS-51.1-1983 and -52.1-1983; ANSI/ANS-58.9-1981; IEEE Std 379-1977), protection and separation criteria (ANSI/ANS-58.3-1977; IEEE Std 384-1981) requirements for remote shutdown in case the control room must be evacuated (ANSI/ANS-58.6-1982), protection against effects of pipe ruptures (ANSI/ANS-58.2-1980), and quality-assurance requirements (ANSI/ASME NQA-1-1979 and -2-1983).

Nevertheless, there is some question regarding the interaction of various plant systems, both as to the supporting roles such systems play and as to the effect one system can have on other systems, particularly with regard to whether actions or consequences could adversely affect the presumed redundancy and independence of safety systems.

In November, 1974 the Advisory Committee on Reactor Safeguards requested that the NRC give attention to the evaluation of safety systems from a multidisciplinary point of view, in order to identify pontentially undesirable interactions between plant systems. In mid-1977 systems interaction was identified as Generic Task A-17 in the NRC program for the resolution of Generic Issues. Because adverse-systems interactions are potentially of large significance to plant safety, the NRC further identified this issue as an "Unresolved Safety Issue."

Following approval of the A-17 Task Action Plan, the NRC employed outside consultants to further develop the NRC position. In May, 1978 Sandia Laboratories was appointed as the initiating contractor and was subsequently author of the results of Phase I of the program published in January, 1980. Enquiries into TMI underlined the need for urgent NRC action and later events, such as those at Crystal River 3 and Browns Ferry 3, caused the scope of the systems-interaction concern to be broadened to include the safety implications of control systems (Generic Task A-47 was established to address this issue). As a consequence of NRC actions in the A-17 area intensified during 1980.

Additional consultants were appointed to establish a methodology for identifying potentially adverse-systems interactions; Diablo Canyon and Indian Point 3 were identified as lead points for investigating these issues and many plants in the final stages of the Operating License (OL) application during 1981 and 1982 were asked to discuss Systems Interaction before the ACRS.

At an AIF Subcommittee meeting on System Interaction held May 10, 1983, the NRC provided an update of the resolution status of A-17 within the NRC. The major effects being conducted are:

- o Search of LERs to determine areas most susceptible to system interaction.

- o Review of current regulations to determine adequacy.

- o Assess the ability of PRAs to highlight areas that are susceptible to potential interactions.

- o Initiate constructive collaboration with industry.

The NRC past efforts involved methodology and not potential solutions. The current staff activities are aimed at criteria for solutions.

A recent affidavit, filed by James H. Conran, NRC in response to a contention on the Shoreham licensing hearing indicates that there is some dissension within the staff on its approach to finding resolution of A-17. The affadavit basically states that some NRC staff members (Mr. Conran in particular), no longer think that a plant should be licensed to operate with this issue unresolved by the NRC. The NRC management has reconsidered USI A-17 in view of Mr. Conran's differing professional opinion. The staff is to update this task action plan to clearly identify the objectives and methods to objectives. The relationship with TMI Item II.C.3, PRAs, SRPs and applicability of operating plants are to be clarified.

The systems interaction concern is a major consideration being addressed in the WAPWR design. The WAPWR design incorporates several features that will reduce the probability of any adverse interactions occurring. 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.

WAPWR Response

One goal of the WAPWR plant design is to address the systems interactions issue early in the design phase. All systems interactions that have been identified in the past are being addressed by either hardware changes or analyses to show the applicable safety criteria are met. Also, a key consideration in the plant layout, safety system design, and equipment selection is to avoid any unacceptable systems interactions.

In addition to considering systems interactions in the design phase of the plant, a comprehensive systems interactions analysis will be performed as part of the WAPWR design and licensing process. A description of the systems interaction study to be performed will be documented as part of the licensing process for the WAPWR design.

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

Discussion

This NRC task is concerned with developing adequate design criteria for electrical equipment in safety systems such that it will perform its function in adverse environmental conditions as a consequence of certain postulated accidents. The NRC requires that such equipment (principally equipment associated with the emergency core cooling, containment isolation, and cleanup systems) be environmentally qualified.

Specific electrical equipment of concern during postulated accident conditions includes:

o Instrumentation needed to initiate the safety systems and provide diagnostic information to the plant operators (e.g., electrical penetrations into containment, any electrical connectors to cabling which transmit signals, and the instruments themselves).

- o Control power to motor operators for certain valves (e.g., emergency core cooling and containment isolation valves located inside containment).

- o Fan cooler motors for those plants that utilize fan coolers for containment heat removal.

NUREG-0588, Revision 1, "Interim Staff Position on Environmental Qualification of Safety-Related Equipment," establishes the methods and procedures to be used to environmentally qualify safety-related electrical equipment and supplements the requirements given in the 1971 and 1974 versions of IEEE Standard 323, "Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." This NUREG does not address in detail all areas of the qualification issue, since some areas (e.g., effects of aging, sequential versus simultaneous testing, including synergistic effects, and the potential for combustible gas and chloride formation in equipment containing organic materials) are not yet fully defined.

In addition, the NRC has codified a new regulation, 10CFR Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," January, 1983. It prescribes aging and testing for synergistic effects. Each holder of an operating license was required by May 20, 1983, to identify the electric equipment important-to-safety already qualified and submit a schedule for the qualification or replacement of the remaining electric equipment important-to-safety. The final environmental qualification of the electric equipment was required by the end of the second refueling outage occurring after March 31, 1982 or by March 31, 1985, whichever is earlier. Applicants for operating licenses were required to perform an analysis to ensure that the plant can be safely operated, pending completion of equipment qualification required by this section. The rule requires that:

- A program shall be established for qualifying electric equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal or that are otherwise essential in preventing significant release of radioactive material to the environment.

- All electric equipment covered by this rule shall be listed and this list shall be maintained in auditable form.

- The electrical equipment qualification program must include temperature and pressure, humidity, chemical effects, radiation, aging, submergence, synergistic effects, and margins.

- Electric equipment must be qualified by testing an identical item of equipment, testing a similar item of equipment (with a supporting analysis to show acceptability), experience with identical or similar equipment under similar conditions (with a supporting analysis to show acceptability), or analysis in lieu of testing (if type testing is precluded by the physical size of the equipment or by the state-of-the-art).

- A record of the qualification must be maintained in an auditable form to permit verification that each item of electric equipment is qualified for its application and meets the specified performance requirements.

Also, the NRC has issued Revision 1 to Regulatory Guide 1.89, "Environmental Qualification of Electric Equipment for Nuclear Power Plants," for comment that describes a method acceptable to the NRC staff to demonstrate compliance with the requirements of 10CFR 50.49.

WAPWR Response

Westinghouse has an ongoing environmental qualification program which has resulted in successful qualification of electrical equipment for recently licensed plants. Currently qualified equipment which is intended to be used for the WAPWR design will be reassessed relative to its position within containment, and any anticipated changes in the potential environment it will experience. Analyses will be performed to demonstrate that the current Westinghouse generic envelope is valid for the WAPWR design. In addition, a detailed WAPWR environmental qualification report will be prepared which will address all of the documentation requirements of the current rulemaking. Finally, Westinghouse will completely document and justify any deviations from the NRC Regulatory Guide 1.89 positions during the licensing process for the WAPWR design.

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

Discussion

Over the past several years, incidents known as "pressure transients" have taken place at various pressurized water reactor facilities. A pressure transient occurs when the pressure-temperature limits included in the technical specifications for the facility have been exceeded. There has been greater than 33 such events. Half of these events occurred before the plant achieved initial criticality (i.e., before initial operation of the reactor); the majority occurred during startup or shutdown operations. In all of these incidents fracture mechanics and fatigue calculations indicated that the reactor vessels were not damaged and continued operation of the vessels was acceptable. Nevertheless, the NRC concluded that appropriate regulatory actions were necessary to reduce the frequency of pressure transient events and restrict future transients to acceptable pressures. The NRC deemed that action was necessary to conserve reactor vessel safety margins over the lifetime of the vessel.

The NRC staff's review of this safety issue was completed in September 1978 with the issuance of NUREG-0224, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors."

Upgraded procedural controls were implemented at operating pressurized water reactor facilities which significantly reduced the occurrence of pressure transient events. In addition, most operating plants incorporated equipment modifications involving the addition of a second lower set point on existing power-operated relief valves, the addition of new spring-loaded relief valves, or modifications to allow use of existing spring-loaded relief valves.

Branch Technical Position RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures," establishes the current NRC acceptance criteria for a low temperature overpressurization protection system.

In summary, Branch Technical Position RSB 5-2 states that:

- o A system should be designed and installed which will prevent exceeding the applicable technical specifications and 10CFR Part 50, Appendix G limits for the reactor coolant system while operating at low temperatures.

- o The system should be able to perform its function assuming any single active component failure. Analyses using appropriate calculational techniques must be provided which demonstrate that the system will provide the required pressure relief capacity assuming the most limiting single active failure. The cause for initiation of the event (e.g., operator error, component malfunction) should not be considered as the single active failure. The analyses should assume the most limiting allowable operating conditions and systems configuration at the time of the postulated cause of the overpressure event.

- o The system should be designed using IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," as guidance.

- o To assure operational readiness, the overpressure protection system should be testable.

- o The system must meet the requirements of Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and Section III of the ASME Code.

- o The overpressure protection system should be designed to function during an operating basis earthquake.

- o The overpressure protection system should not depend on the availability of offsite power to perform its function.

- o Overpressure protection systems which take credit for an active component(s) to mitigate the consequences of an overpressurization event should include additional analyses considering inadvertent system initiation/actuation or provide justification to show that existing analyses bound such an event.

- o If pressure relief is from a low pressure system, not normally connected to the primary system, the overpressure protection function should not be defeated by interlocks which would isolate the low pressure system from the primary coolant system.

Low temperature overpressure protection systems were implemented on all PWR plants. With the exception of four plants, the post implementation reviews by the NRC are complete.

WAPWR Response

The WAPWR design will include a low temperature overpressurization protection capability. Westinghouse will completely document and justify any deviations from the NRC Branch Technical Position RSB 5-2 acceptance criteria during the licensing process for the WAPWR design.

16. Issue A-31: Residual Heat Removal Requirements

Discussion

The safe shutdown of a nuclear power plant following an accident not related to a loss-of-coolant accident has been typically interpreted as achieving a "hot standby" condition (i.e., the reactor is shutdown, but system temperature and pressure are still at or near normal operating values). Considerable emphasis has been placed on the hot standby condition of a power plant in the event of an accident or abnormal occurrence. A similar emphasis has been placed on long-term cooling.

Even though it may generally be considered safe to maintain a reactor in a hot standby condition for a long time, experience shows that there have been events that required eventual cooldown and long-term cooling until the reactor coolant system was cold enough to perform inspection and repairs. For this reason the ability to transfer heat from the reactor to the environment after a shutdown is an important safety function. Therefore, the NRC believes it is essential that a power plant be able to go from hot standby to cold shutdown conditions (when this is determined to be the safest course of action) under any accident conditions.

This NRC task is concerned with establishing specific design requirements for the systems that are employed to achieve and maintain a safe shutdown including cooldown from hot standby to cold shutdown (e.g., reactor coolant system, main steam system, auxiliary feedwater system, chemical and volume control system, borated refueling water system, residual heat removal system, component cooling water system, essential service water

system, supportive heating, ventilation and air conditioning systems, emergency diesel generators, spent fuel cooling system, and supportive portions of the instrument air system).

Regulatory Guide 1.139, "Guidance for Residual Heat Removal," Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System," and Standard Review Plan 5.4.7, "Residual Heat Removal (RHR) Systems," contain regulatory positions and acceptance criteria for the system(s) used to take the reactor from normal operating conditions to cold shutdown. Specifically, the system(s) must:

- Be safety-grade
- Be single failure proof
- Function with or without offsite power
- Be capable of being operated from the control room
- Be capable of achieving cold shutdown within 36 hours

In addition, the residual heat removal pump system must meet specific isolation, pressure relief, pump protection, and testability requirements.

WAPWR Response

The WAPWR design includes safety-grade cold shutdown capability. Westinghouse will completely document and justify any deviations from the NRC Regulatory Guide 1.139, Branch Technical Position RSB 5-1, and Standard Review Plan 5.4.7 positions and acceptance criteria during the licensing process for the WAPWR design.

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

Discussion

Overhead handling systems (cranes) are used to lift heavy objects in the vicinity of spent fuel in light-water-cooled nuclear power plants. If a

heavy object (e.g., a spent fuel shipping cask or shielding block) were to fall or tip onto spent fuel in the storage pool or the reactor core and damage the fuel, there could be a release of radioactivity to the environment and a potential for radiation overexposure to inplant personnel. If many fuel assemblies are damaged, and the damaged fuel contained a large amount of undecayed fission products, radiation releases to the environment could exceed the guidelines of 10CFR Part 100, "Reactor Site Criteria."

Additionally, a heavy object could fall on safety-related equipment and prevent it from performing its intended function. If equipment from redundant shutdown paths were damaged, safe shutdown capability may be defeated.

The purpose of this task was to provide an evaluation of current NRC requirements and existing licensee design measures, operating procedures, and technical specifications associated with the movement of heavy loads near spent fuel pools inside or outside containment, and over the reactor core during refueling. The current NRC requirements and review procedures in effect at the time this issue was identified, were given in Standard Review Plans 9.1.2, "Spent Fuel Storage," 9.1.4, "Light Load Handling System (Related to Refueling)," 15.7.4, "Radiological Consequences of Fuel Handling Accidents," and 15.7.5, "Spent Fuel Cask Drop Accidents." These Standard Review Plans provide procedures for review of the spent fuel storage pool, the fuel handling system, radiological consequences of fuel handling accidents, and spent fuel cask drop accidents. Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," provides additional guidance in this area. Further, the Standard Technical Specifications, included in all new operating licenses, include a prohibition on the movement of loads over spent fuel in the storage pool that weigh more than the equivalent weight of a fuel assembly. These load restrictions have been successfully demonstrated, for recently licensed plants, as providing assurance that miscellaneous loads which have not been reviewed from the

standpoint of rigging) will not be carried over stored fuel, and in the event such loads are dropped, radioactivity release is limited and critical array does not result from rack distortion.

Although it is the NRC's view that continued operation with currently licensed facilities' designs, operating procedures, and technical specification limitations that meet the criteria listed above presents no undue risk to the health and safety of the public, the advent of increased (higher density storage configurations) and longer term storage of spent fuel assemblies in spent fuel storage pools caused the NRC to reevaluate the above requirements.

As a result of this reevaluation the NRC expanded this issue to also include the control of heavy loads over safe shutdown equipment (i.e., safety-related equipment and associated subsystems that would be required to bring the plant to cold shutdown conditions or provide continued decay heat removal following the dropping of a heavy load). The NRC has documented their technical resolution of this issue in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and issued Standard Review Plan 9.1.5, "Overhead Heavy Load Handling Systems," which includes NUREG-0612 as one of the NRC acceptance criteria. All licensees have responded to the concern and the NRC expects to complete their review in 1984.

WAPWR Response

Westinghouse will completely document and justify any deviations from the NRC Standard Review Plan 9.1.5 acceptance criteria during the licensing process for the WAPWR design.

18. 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.

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

Discussion

NRC regulations require that nuclear power structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes. Detailed requirements and guidance regarding the seismic design of nuclear plants are provided in NRC regulations and regulatory guides. However, there are a number of plants with construction permits and operating licenses issued before the NRC's current regulations and regulatory guides were in place. For this reason, reviews of the seismic design of various plants are being undertaken to ensure that these plants do not present an undue risk to the health and safety of the public. Task A-40 is, in effect, a compendium of short-term efforts to support such reevaluation efforts of the NRC staff, especially those related to older operating plants.

Current criteria require:

- o Define intensity of SSE
- o Determine free-field ground motion
- o Determine interaction with structures
- o Determine equipment motion
- o Combine seismic loads with other loads and compare with the allowable loads

Westinghouse has developed advanced seismic analysis and design techniques necessary to meet the current conservative NRC seismic design basis. These techniques have been applied to recently licensed plants and found acceptable by the NRC.

This NRC task action plan is expected to remain open at least through 1984 while the NRC is obtaining further results from its research programs. However, the NRC has stated that: "We do not expect any need for upgrading (the current seismic design criteria) from this task action plan. Any such need that does arise would not be major. Thus, we expect that modifications, if required, would not be major."

The NRC intends to issue a draft SRP on seismic design in 1983 and issue a final SRP in 1984.

WAPWR Response

As indicated above, this issue is primarily concerned with plants licensed prior to the issuance of current NRC regulations and regulatory guidance regarding seismic design, and therefore, the ultimate resolution of this issue is not expected to impact the WAPWR design.

Specifically for the WAPWR, Westinghouse will: (A) establish and document a generic seismic design envelope, (B) apply established Westinghouse seismic evaluation methodology which has been successfully applied to current plant designs and which meets current NRC regulations and regulatory guidance, where appropriate, and (c) pursue certain licensing initiatives related to the establishment of the seismic design bases.

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

Discussion

This issue is not applicable to Westinghouse pressurized water reactors.

21. Issue A-43: Containment Emergency Sump Performance

Discussion

Following a postulated loss-of-coolant accident in a pressurized water reactor, water discharged from the break would collect on the containment floor and within the containment emergency sump. Although the emergency core cooling and containment spray systems initially draw water from the refueling-water-storage tank, long-term core cooling is affected by re-alignment of these system pumps to the containment emergency sump. Thus, successful long-term recirculation depends upon the sump providing adequate, debris-free water to the recirculation pumps for extended periods of time. Moreover, the flow conditions through the sump and associated piping must not result in pressure losses or air entrainment that would inhibit proper pump operation. Without a proper sump design, long-term cooling should be significantly impaired.

The importance of the emergency sump and safety considerations associated with its design were early considerations in containment design. Net-positive-suction-head requirements, operational verification, and sump design requirements are issues that have evolved and are currently contained in the following regulatory guides:

- o Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps" (Safety Guide 1, November 1970).

- o Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors" (June 1974) (Revision 1, January 1975). (Revision 2, September 1978).

- o Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems" (June 1974) (Proposed Revision 1, May 1983).

- o Standard Review Plan 6.2.2, Revision 4 (Proposed)

Initially, NRC concerns regarding emergency sump performance were addressed through in-plant tests (per the guidelines of Regulatory Guide 1.79) with a transition to containment and sump model tests in the mid-1970's. At that time, considerable emphasis was placed on "adequate" sump hydraulic performance during model tests, and vortex formation was identified as the key determinant. The main concern was that formation of an air-core vortex would result in unacceptable levels of air ingestion and, subsequently, in severely degraded pump performance.

There was also concern about sump damage or blockage of the flow as a result of loss-of-coolant accident generated insulation debris, missiles, etc. These concerns led to the formulation of some of the guidelines set forth in Regulatory Guide 1.82 (cover plates, debris screen, 50 percent screen blockage, etc).

In 1979, as a result of continued NRC concern for safe operation of emergency sumps, the Commission designated the issue an an unresolved safety issue.

The principal NRC concern is summarized in the following question:

> "In the recirculation mode following a loss-of-coolant accident, will the pumps receive water sufficiently free of debris and air and at sufficient pressure to satisfy net positive suction head requirements so that pump performance is not impaired?"

This concern was divided into the following three areas for technical consideration by the NRC:

- o Sump Design. Sump hydraulic performance under post loss-of-coolant accident adverse conditions such as air ingestion, elevated temperatures, break and drain flow, etc.

- o Insulation Debris Effects loss-of-coolant accident generated debris arising from the break jet destroying large quantities of insulation, this insulation debris being transported to the sump screen(s), and the resulting screen blockage being sufficient to reduce net positive suction head significantly below that required to maintain adequate pumping.

- o Pump Performance. The performance capability of residual heat removal and containment spray system pumps to continue pumping when subjected to air ingestion, debris ingestion, and effects of particulates.

The NRC has issued (for comment) NUREG-0897, "Containment Emergency Sump Performance," April 1983, that includes technical findings to be used as a basis for resolving Unresolved Safety Issue A-43. NUREG-0869, "USI A-43 Resolution Positions," April 1983, also is issued for comments. It contains the proposed revision to Regulatory Guide 1.82, a draft generic letter, a value-impact, meeting minutes and references to SRP 6.2.2, revision 4. The proposed regulatory guide, draft generic letter and SRP contain the intended requirements. The NRC is changing its practice of referencing NUREGs for requirements by providing the requirements in appropriate documents (other than NUREG's). Plants that do not have an SER would be required to evaluate sump performance relative to criteria proposed in Appendix A of Regulatory Guide 1.82, Revision 1.

Complete implementation of the requirements resulting from this USI would, by generic letter, require operating and NTOL plants to review sump performance to the same criteria. New designs may not require full or scale testing if the design meets the criteria. The issue of sump blockage by loose paint is categorized as a (new) generic safety issue and subject to further processing for priority and evaluation.

The NRC expects to incorporate comments October 1983 and issue requirements in the first quarter of 1984.

Generally speaking, it is not expected that PWRs that extensively use reflective metallic insulations will encounter a debris blockage problem. Unencapsulated fibrous insulations are believed to present the principal debris problem and it is estimated that six to ten PWRs may require some type of corrective action.

WAPWR Response

An important design feature of the WAPWR is the in-containment Emergency Water Storage Tank (EWST) which has replaced the conventional outside-containment refueling water storage tank (RWST). The four WAPWR ECCS subsystems and the four WAPWR containment spray subsystems are initially aligned to draw water from the in-containment EWST therefore no reliagnment of these systems is required for long term recirculation. The in-containment EWST is an annular tank which is located below the containment floor and sized to contain sufficient borated water to fill the refueling canal during refueling operations.

Following a postulated loss of coolant accident, water discharged from the break would; (1) collect on the containment floor; (2) flood all compartments below the containment floor elevation such as the reactor vessel cavity, and (3) spill back into the EWST via several physically separated spillways located in the containment floor and outside the loop compartments. Each spillway is protected by rough screens and trash racks to prevent debris from entering the EWST. The elevation of each spillway is several inches above the containment floor, therefore, the containment floor serves as a large settling pond for the recirculation water.

Inside the EWST, there are four physically separate EWST pump pits located below the EWST floor elevation. Each sump pit is dedicated to one of the four ECCS and containment spray subsystems. Rough screens and fine screens are provided at each of the four EWST sump pits in addition to the rough screens and trash racks provided at each of the EWST spillways. The EWST therefore serves as a second settling pond for the recirculation water.

Evaluations are performed to establish the minimum post accident EWST water levels and to verify that the ECCS and containment spray pump net positive suction head (NPSH) requirements are satisfied for all normal or accident system operation. The WAPWR EWST configuration therefore meets all NPSH and sump design requirements currently specified in SRP 6.2.2g NRC Regulatory Guide 1.1, and 1.82.

The WAPWR EWST configuration in conjunction with the ECCS piping configuration also provides a unique means for performing full flow system performance verification not only during preoperational testing but a anytime during the plant life. Therefore all operational verification requirements specified in NRC Regulatory Guide 1.79 are satisfied by the WAPWR design.

22. Issue A-44: Station Blackout

Discussion

Electrical power for safety systems at nuclear power plants must be supplied by at least two redundant and independent divisions. The systems used to remove decay heat to cool the reactor core following a reactor shutdown are included among the safety systems that must meet this requirement. Each electrical division for safety systems includes an offsite AC power connection, a standby emergency diesel generator AC power supply, and DC sources.

NRC Task A-44 involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all power (that is, loss of both the offsite AC power and the emergency diesel generator AC power). This issue arose because of operating experience regarding the reliability of AC power supplies. A number of operating plants have experienced a total loss of offsite electrical power, and more occurrences are expected in the future.

During each of these loss of offsite power events, the onsite emergency AC power supply was available to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power supplies has been unavailable. In addition, there have been numerous reports of emergency diesel generators in operating plants failing to start and run during periodic surveillance tests.

Current NRC safety requirements require a minimum that diverse power drives be provided for the redundant auxiliary feedwater pumps. This is normally accomplished by utilizing an AC powered electric motor-driven pump and a redundant steam turbine-driven pump.

The NRC has established an action plan geared to the technical resolution (i.e., establishment of requirements) of this issue. This plan involves:

- o Evaluations of the expected frequency and duration of offsite (preferred) power losses at nuclear power plants.
- o Estimates of the reliability and evaluations of the dominant factors affecting the reliability of emergency AC power supplies.
- o Estimates of reliability (trade-offs) of decay heat removal, diesel generators, and direct current power.
- o Evaluations of the risks posed by station blackout accidents and assessments of the effectiveness of safety improvements in reducing those risks.

A NUREG for public comment is expected to be issued by the end of 1983, and the NUREG (with its requirement documents in draft form) should be issued in 1984.

Studies of limiting conditions during which time total blackout is acceptable show such times to be as long as several days or as short as several minutes, depending on the limiting conditions that are postulated. The NRC has indicated they have been thinking in terms of 2 hours, but may require different times (presumably longer, i.e., eight or more hours) for final resolution of this issue. During this time of the total blackout, AC electrical power for instrumentation and control is assumed to be available from inverters whose power source is station battery power. By the end of this blackout interval, AC electrical power is expected to be restored from either an outside source or the emergency diesel power source.

The WAPWR has certain design features that mitigate the impact of a loss of all AC power. These include the the emergency feedwater system and the upgraded AC power independent seal injection system provided by the chemical and volume control system. In addition other areas would be impacted by a station blackout event including the batteries, communications equipment, emergency lighting, control room habitability, etc.

Loss of all AC power (station blackout) is not currently required to be a design basis. It is, however, expected that final resolution of this issue will result in requirements for plants to be designed for a loss of all AC power for some period of time.

WAPWR Response

Westinghouse has included a posutulated loss of all AC power in the design criteria for the WAPWR. [(a,c)

] In addition, Westinghouse will develop loss of all AC power emergency response guidelines for use by utilities utilizing the WAPWR design.

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

Discussion

Under normal operating conditions, power generated within a reactor is removed as steam to produce electricity via a turbine-generator. Following a reactor shutdown, a reactor produces insufficient power to operate the turbine; however, the radioactive decay of fission products continues to produce heat (so-called "decay heat"). Therefore, when reactor shutdown occurs, other measures must be available to remove decay heat from the reactor to ensure that high temperatures and pressure do not develop which could jeopardize the reactor and the reactor coolant system. It is evident, therefore, that all light water reactors share two common decay heat removal functional requirements: (A) to provide a means of transferring decay heat from the reactor coolant system to an ultimate heat sink, and (B) to maintain sufficient water inventory inside the reactor vessel to ensure adequate cooling of the reactor fuel. The reliability of a particular power plant to perform these functions depends on the frequency of initiating events that require or jeopardize decay heat removal operations, and the probability that required systems will respond to remove the decay heat.

This issue is concerned with evaluating the benefit of providing alternate means of decay heat removal which could substantially increase a plant's capability to handle a broader spectrum of transients and accidents. The NRC will perform a number of plant specific decay heat removal evaluations and establish recommendations regarding the desirability of improvements in existing systems or an alternative decay heat removal method; if the improvements or alternative can significantly reduce the overall risk to the public.

Task A-45 evolved from earlier programs that aimed more specifically at systems such as residual heat removal and auxiliary feedwater systems. Included among these were:

- o Unresolved Safety Issue A-31, "Residual Heat Removal Requirement."
- o Regulatory Guide 1.139, "Guidance for Residual Heat Removal."
- o Standard Review Plan 5.4.7, "Residual Heat Removal (RHR) Systems."
- o Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System."
- o Standard Review Plan 10.4.9, "Auxiliary Feedwater Systems."
- o Branch Technical Position ASB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants."

Regulatory Guide 1.139, Standard Review Plan 5.4.7, and Branch Technical Position RSB 5-1 are discussed in item 16 above (Unresolved Safety Issue A-31). Standard Review Plan 10.4.9 and Branch Technical Position ASB 10-1 essentially require that an auxiliary feedwater system should consist of at least two full capacity independent systems that are operated from diverse energy sources, and should be able to accommodate the single active failure of a component. The piping arrangement for each train should be designed to permit the pumps to supply feedwater to any combination of steam generators. It is also required that operating plants be capable of providing the required auxiliary feedwater flow for at least two hours from one auxiliary feedwater pump train even if both offsite and onsite AC power sources are lost.

The NRC is reviewing the subject and will prepare a report for internal approval in late 1984.

The WAPWR has several systems which have the capability to remove decay heat from the reactor core. The WAPWR secondary side safeguards system will employ an emergency feedwater system combined with a startup feedwater system as discussed in some detail in Section 3.1 (item 2).

This system provides two independent systems operated from diverse energy sources which serve to remove decay heat from the primary system via the steam generators to the secondary system. If the steam generator, EFWS and SFWS cannot remove heat then the Integrated Safeguards System (ISS) can be used. The ISS employs a residual heat removal system consisting of four residual heat removal heat exchangers, and has the capability to "feed and bleed" primary coolant as a way to borate the reactor coolant system and to remove decay heat via high head safety injection pumps in conjunction with the pressurizer power-operated relief valves.

WAPWR Response

The primary side and secondary side safeguards systems for the WAPWR design (as discussed above) provide the capability of removing decay heat from the reactor core while maintaining sufficient water inventory to ensure adequate core cooling. As such, the design of these systems is not expected to be significantly impacted by this USI.

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

Discussion

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the evolution of the commercial nuclear power industry. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and perform the intended safety functions may vary considerably. The NRC believes that the seismic qualification of equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition when it is subject to a seismic event. The objective of the NRC task program to address this issue is to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of

attempting to backfitting of the current design criteria that apply to new plants. This guidance will concern equipment required to safely shutdown the plant, as well as equipment whose function is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions.

A Seismic Qualification Utilities Group (SQUG) has collected data on nuclear and non-nuclear equipment that experienced earthquakes. Independent from the NRC a review panel is utilizing the seismic experience data for guidance in equipment qualification. The NRC is developing the selection of equipment and floor response spectra generic to LWRs.

The final program requirements will not be issued until mid-1984.

In response to this issue and other industry needs Westinghouse has developed a seismic requalification program for operating plants. This program consists of a data search to identify what type of seismic qualification (if any) is available, use of existing data to qualify equipment by similarity, qualification by analysis, testing, or a combination of analysis and testing. Current NRC acceptance criteria and regulatory guidance for new plant designs are given in Standard Review Plan 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment," and Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants," which endorse IEEE Standard 344-1975, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations."

For Westinghouse plants designed and built to the requirements of IEEE Standard 344-1975, seismic design requirements implemented by Westinghouse for equipment important to safety are consistent with the latest NRC regulatory requirements. Additionally, the NRC has found the methods used by Westinghouse acceptable on a number of recent plant applications.

WAPWR Response

As indicated above, this issue is primarily concerned with plants licensed prior to the issuance of current NRC regulatory requirements regarding seismic qualification, and therefore, the ultimate resolution of this issue is not expected to impact the WAPWR design.

Specifically for the WAPWR, Westinghouse will: (A) completely document and justify any deviations from the NRC Standard Review Plan 3.10 and Regulatory Guide 1.100 acceptance criteria and positions during the licensing process for the WAPWR design, and (B) apply established Westinghouse seismic qualification methods which have been successfully applied to current plant designs and which meet current NRC regulatory requirements.

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

Discussion

This issue concerns the potential for transients or accidents being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration. One concern is the potential for a single failure to cause simultaneous malfunction of several control features. Such an occurrence could conceivably result in a transient more severe than those transients analyzed as anticipated operational occurrences. A second concern is for a postulated accident to cause control system failures which could make the accident more severe than analyzed. Accidents could conceivably cause control system failures by creating a harsh environment in the area of the control equipment or by physically damaging the control equipment. It is generally believed by the NRC staff that such control system failures would not lead to serious events or result in conditions that safety systems cannot safely handle. Systematic evaluations have not been rigorously performed to verify this belief. The

potential for an accident that could affect a particular control system and the effects of the control system failures may differ from plant to plant. Therefore, the NRC believes it is not possible to develop generic answers to these concerns, but rather plant-specific evaluations are required. The purpose of the NRC task program to address this issue is to define generic criteria that will be used for plant specific evaluations.

The NRC initiated a long term program to evaluate control systems with particular attention to control system failures that could lead to over-cooling the reactor or overfilling the steam generator. The requirements package is scheduled for NRC internal review by CRGR in 1984 and a criteria NUREG is scheduled for 1986.

Operating plants were requested to respond to IE Bulletin 79-27 "Loss of Non-Class IE Instrumentation and Control Power System Bus During Operation". It required the utility to evaluate the effect and propose modifications for failure of IE and non-IE instrument buses, list indicators and alternate indicators and propose changes to emergency procedures for coping with the loss of instrument power.

The NRC asks applicants to determine the effect of:

1. Loss of power to all the control systems powered by a single power supply.

2. Failure of each instrument sensor that provides a signal to two or more control systems.

3. Failure of each sensor.

4. Break of any sensor impulse line that is used for sensors providing signals to two or more control systems.

Current plants in the licensing review process must respond to questions on control system failures on common instrument power supplies, instrument lines (impulse/header lines) and sensors shared by more tnan one channel. Westinghouse has performed a modified failure mode and effect analysis for plants in the licensing process to justify that single room component failure are bounded by the accident analysis.

Current Westinghouse Condition II analyses of transient events of moderate frequency, that could be initiated by the single failure of a control system, show that the consequences meet acceptance criteria for Condition II events.

WAPWR Response

Since the functional requirements and design specifications for the WAPWR control systems will be no less stringent than those for current plants, it is expected that an analysis similar to that performed on recently licensed plants would likewise show that the consequences of failures in control systems of the WAPWR would be bounded by FSAR type analyses. Consequently, no hardware impacts on the WAPWR control systems are anticipated. However, a control system failure study, as part of an overall systems interactions study (refer to item 13 above), will be performed and documented during the licensing process for the WAPWR design. The objectives of this study are to:

- Minimize the potential for reactor shutdown or safeguards system actuation by failures in reactor control or protection systems.

- Reduce the number of possible interactions between control and protection systems which could lead to a degraded accident condition.

- Reduce the probability and consequences of failures in control systems on plant safety and operability.

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

Discussion

Postulated reactor accidents which result in a degraded or melted core can result in generation and release to the containment of large quantities of hydrogen. The hydrogen is formed from the reaction of the zirconium fuel cladding with steam at high temperatures and/or by radiolysis of water. Experience gained from the TMI-2 accident indicates that the NRC may require more specific design provisions for handling larger hydrogen releases than currently required by regulations (particuarly for smaller, low pressure containment designs).

The purpose of the NRC task program to address this issue is to investigate means to predict the quantity and release rate of hydrogen following degraded core accidents and various means to cope with large releases to the containment such as inerting of the containment or controlled burning. The potential effects of proposed hydrogen control measures on safety including the effects of hydrogen burns on safety-related equipment will be investigated. The NRC expects to issue a generic report mid-1985.

The NRC has issued a revision to 10CFR Part 50.34, "Contents of Applications; Technical Information," which incorporates post-TMI requirements into their regulations. This revision, known as the "CP/ML Rule," encompasses the issue of hydrogen control for new plant designs.

WAPWR Response

The CP/ML Rule is discussed in detail in Section 3.1 and hydrogen control specifically in items 5 and 14.

27. Issue A-49: Pressurized Thermal Shock

Discussion

Transients and accidents can be postulated to occur in pressurized water reactors (PWRs) that result in severe overcooling (thermal shock) of the reactor vessel, concurrent with high pressure. In these pressurized thermal shock (PTS) events, rapid cooling of the reactor vessel internal surface causes a temperature distribution across the reactor vessel wall that produces a thermal stress with maximum tensile stress at the inside surface of the vessel. The magnitude of the thermal stress varies with the rate of change of temperature and is compounded by coincident pressure stresses.

PTS events are postulated to result from a variety of causes. These include system transients, some of which are initiated by instrumentation and control system malfunctions (including stuck open valves in either the primary or secondary system), and postulated accidents such as small break loss-of-coolant accidents, main steam line breaks, and feedwater line breaks.

As long as the fracture resistance of the reactor vessel material is relatively high, these events are not expected to cause vessel failure. However, the fracture resistance of the reactor vessel material decreases with the integrated exposure to fast neutrons. The rate of decrease is dependent on the chemical composition of the vessel wall and weld materials. If the fracture resistance of the vessel has been reduced sufficiently by neutron irradiation, severe PTS events could cause small flaws that might exist near the inner surface to propagate into the vessel wall. The assumed initial flaw might be enlarged into a crack through the vessel wall of sufficient extent to threaten vessel integrity and, therefore, core cooling capability.

The toughness state of reactor vessel materials can be characterized by a "reference temperature for nil ductility transition" (RT_{NDT}). As the temperature decreases, the metal gradually loses toughness over a temperature range of about 100°F. RT_{NDT} is a measure of where this toughness transition occurs. Its value depends on the material and the integrated neutron irradiation. Correlations, based on tests of irradiated specimens, have been developed to calculate the shift in RT_{NDT} as a function of neutron fluence for various material compositions. The value of RT_{NDT} at a given time in a vessel's life is used in fracture mechanics calculations to determine whether assumed pre-existing flaws could propagate as cracks when the vessel is subjected to overcooling events.

The NRC is proposing to amend its regulation (10CFR 50.61) to (1) establish a screening criterion related to the fracture resistance of PWR vessels during (PTS) events, (2) require analyses and schedule for implementation of flux reduction programs that are reasonably practicable to avoid exceeding the screening criterion, and (3) require detailed safety evaluations before plant operation beyond the screening criterion value.

The value of RT_{NDT} can be selected so that the risk from PTS events for reactor vessels with smaller RT_{NDT} values is acceptable. Higher values of RT_{NDT} might also be shown to be acceptable, but the demonstration would require detailed plant-specific evaluations and possibly modifications. A value for RT_{NDT} as a screening criterion determines the need for, and timing of, further plant-specific evaluations.

A notice of Proposed Rulemaking that (1) establishes and RT_{NDT} screening criterion, (2) require licensees to submit present and projected values of RT_{NDT}, (3) requires early analysis and implementation of such flux reduction programs as are reasonably practicable to avoid reaching the screening criterion, and (4) requires plant-specific PTS safety analyses before a plant is within three calendar years of reaching the screening criterion, including analyses of alternatives to minimize the PTS problem, should be issued the last half of 1983.

A wide spectrum of postulated overcooling events could occur. Postulated events were grouped into categories, estimates were made of their expected frequency, and stylized characterizations of the temperature and pressure time-histories were developed for each event category. Estimates are based on a generic study of Westinghouse-designed pressurized water reactor systems, and are considered also to be generally representative of PWR systems designed by Combustion Engineering. Because there are some significant differences between those designs and PWRs designed by Babcock & Wilcox that affect the characteristics and estimated frequencies of PTS events, information was also developed for the Babcock & Wilcox designs.

By combining the estimated frequencies of postulated events with the probabilistic fracture mechanics results, some estimates of the probability of vessel failure resulting from PTS events were developed. These estimates were used by the NRC to better understand the residual risks inherent in the use of the screening criterion approach for further evaluations and resolution of the PTS issue.

On the basis of these studies, the NRC staff concluded that PWR reactor pressure vessels with conservatively calculated values of RT_{NDT} less than 270°F for plate material and axial welds, and less than 300°F for circumferential welds, present an acceptably low risk of vessel failure from PTS events. These values were chosen as the screening criterion.

The RT_{NDT} of reactor vessels for some plants will remain below the screening criterion (acceptable) throughout the service life. For many other reactor vessels, fuel management programs could be instituted that would result in core configurations reducing neutron flux at critical locations, thereby slowing the increase of RT_{NDT} so that the screening criterion would not be exceeded. Further refinements in materials information, analyses of PTS event frequencies and scenarios, and plant-specific analyses of alternative measures to reduce PTS risk may provide a basis for continued operation with RT_{NDT} values in excess of the screening criterion. The preparation and review of such analyses and

determination of their acceptability will require substantial time. However, the effectiveness of flux reduction programs depend on early implementation. Practicable flux reduction programs should be implemented to maintain reactor vessel RT_{NDT} below the screening criterion, without waiting for possible plant-specific determinations for higher values. Licensees may submit additional plant-specific analyses to justify (new information, improved analyses or evaluations of alternative measures) operation with less restrictive flux reduction programs in the future.

When it is determined that even with flux reduction measures that the vessel RT_{NDT} is still projected to exceed the screening criterion, an analysis of the vessel fracture mechanics properties and including the effects on PTS risk will be required at least three years before the screening criterion would be exceeded.

Design improvements to the safeguards systems in the WAPWR will limit thermal shock to the reactor vessel during severe accidents. The primary side safeguards system (i.e., the integrated safeguards system) will inject water to the reactor coolant system at temperatures significantly higher (e.g., 80-100°F) than that at certain conventional operating plant designs during postulated loss-of-coolant accident conditions. This is due to the location of the suction water source being within the containment building where the temperature is expected to always be greater than 80°F.

In addition, the improvements in the reactor vessel material specifications and potential increased shielding (as discussed in item 11 above) will further mitigate the impact of a thermal shock event on the reactor vessel.

WAPWR Response

The design improvements discussed above will make the WAPWR less susceptible to severe pressurized overcooling events than current operating plants. Therefore, no additional impact on the WAPWR design is anticipated as a result of the final resolution of this issue.