UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 4, 1984

Docket No.: STN 50-601

Mr. E. P. Rahe, Jr., Manager
Nuclear Safety Department
Water Reactor Division
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Rahe:

Subject: RESAR-SP/90 Preliminary Design Approval (PDA) Application: Review of Module 2, "Regulatory Conformance"

We have completed our review of RESAR-SP/90 PDA Module 2 and have developed comments and feedback for your use in the ongoing PDA review of RESAR-SP/90.

Due to the nature of Module 2, "Regulatory Conformance," the staff is unable at this time to make detailed conclusions regarding the regulatory acceptability of the Westinghouse (W) Advanced Pressurized Water Reactor (APWR) design. Staff conclusions as to the regulatory acceptability of the APWR design must ultimately await submittal of the completed integrated application. Therefore, a separate safety evaluation report (SER) will not be issued for this module; rather, the applicable regulatory issues will be addressed in the staff review of individual modules and in the staff review of the integrated application (including Appendix C to the SER).

We offer the following additional comments:

1) We note that this module, if updated and revised to reflect the comments noted herein, would address the major regulatory requirements and guidelines. Further, the updated module should facilitate the staff's assessment of the compliance of the WAPWR design with current and future licensing requirements.

   Enclosed for your information and guidance is a draft evaluation based on the information available in Module 2 related to Unresolved Safety Issues. The Enclosure provides guidance on the information that will have to be provided prior to issuance of a final SER on the completed application.

2) It appears that the draft version of NUREG-0933 was used in developing Module 2. Please note that this document and a supplement have now been published in final form. We recommend that your submittal be revised to incorporate the latest published version of this document.

3) Although final action has not yet been taken on the proposed NRC policy statement concerning severe reactor accidents, we expect that RESAR-SP/90 will contain such information as will be necessary to address all issues contained in the approved severe accident policy statement. We believe that Module 2 is the appropriate place to address each of these issues by referencing other modules which address the issue in detail.

This completes our review of PDA Module 2.

Sincerely,

Original Signed By:

Dennis M. Crutchfield, Assistant Director
for Safety Assessment
Division of Licensing

Enclosure:
As stated

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3) Although final action has not yet been taken on the proposed NRC policy statement concerning severe reactor accidents, we expect that RESAR-SP/90 will contain such information as will be necessary to address all issues contained in the approved severe accident policy statement. We believe that Module 2 is the appropriate place to address each of these issues by referencing other modules which address the issue in detail.

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for Safety Assessment
Division of Licensing

Enclosure:
As stated

UNITED STATES
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WASHINGTON, D. C. 20555

Draft Safety Evaluation Report

RESAR-SP/90 PDA Module 2

APPENDIX C

NUCLEAR REGULATORY COMMISSION (NRC)

C.1 Introduction

The NRC staff evaluates the safety requirements used in its reviews against new information as it becomes available. Information related to the safety of nuclear power plants comes 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. After the accident at TMI the Office for Analysis and Evaluation of Operational Data was established to provide a systematic and continuing review of operating experience. 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. This assessment includes consideration of the generic implications of the issue.

In some cases, immediate action is taken to assure safety, e.g., 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 prior to making licensing decisions. In most cases, however, the initial assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. If the issue applies to several or a class of plants the issue is evaluated further as a "generic safety issue." This evaluation considers the safety significance of the issues, the cost to implement any changes in plant design or operation and other significant and relevant factors to establish a priority ranking of the issue. Based on this ranking resolution of the issue is scheduled for near term resolution, deferred until resources become available or dropped from further consideration.

These issues with the highest priority ranking are reviewed to determine whether they should be designated as "unresolved safety issues" (NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," dated January 1, 1978). However, as discussed above, such issues are considered on a generic basis only after the staff has made an initial determination that the safety significance of the issue does not prohibit continued operation or require licensing actions while the longer term generic review is underway.

These longer term generic studies were the subject of a Decision by the Atomic Safety and Licensing Appeal Board of the Nuclear Regulatory Commission. The Decision was issued on November 23, 1977 (ALAB-444) in connection with the Appeal Board's consideration of the Gulf States Utility Company application for the River Bend Station, Unit Nos. 1 and 2. These issues were also considered in the operating license proceeding Virginia Electric and Power Company (North Anna Nuclear Power Station, Unit Nos. 1 and 2), ALAB-491, NRC 245 (1978). A further discussion of these issues is contained in a decision by the Atomic Safety and Licensing Appeal Board in connection with its considerations of the Pacific Gas and Electric Company operating license application for the Diablo Canyon Nuclear Power Plant, Units 1 and 2 (ALAB-728, issued May 18, 1983). In the ALAB-728 Decision, the Board stated with regard to an operating license proceeding that: "it would be helpful to us if the staff would include in an SER supplement an explanation of the unresolved safety issues affecting the facility under review and the reasons the facility could nonetheless safely operate pending resolution of those issues." This appendix is provided in response to the Board's request.

C.2 Unresolved Safety Issues

In a related matter, as a result of Congressional action on the Nuclear Regulatory Commission budget for Fiscal Year 1978, the Energy Reorganization Act of 1974 was amended (PL 95-209) on December 13, 1977 to include, among other things, a new Section 210 as follows:

UNRESOLVED SAFETY ISSUES PLAN

> SEC. 210. The Commission shall develop a plan providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the

> Congress on or before January 1, 1978 and progress reports shall be included in the annual report of the Commission thereafter.

The Joint Explanatory Statement of the House-Senate Conference Committee for the Fiscal Year 1978 Appropriations Bill (Bill S. 1131) provided the following additional information regarding the Committee's deliberations on this portion of the bill:

> SECTION 3 - UNRESOLVED SAFETY ISSUES
>
> The House amendment required development of a plan to resolve generic safety issues. The conferees agreed to a requirement that the plan be submitted to the Congress on or before January 1, 1978. The conferees also expressed the intent that this plan should identify and describe those safety issues, relating to nuclear power reactors, which are unresolved on the date of enactment. It should set forth: (1) Commission actions taken directly or indirectly to develop and implement corrective measures; (2) further actions planned concerning such measures; and (3) timetables and cost estimates of such actions. The Commission should indicate the priority it has assigned to each issue, and the basis on which priorities have been assigned.

In response to the reporting requirements of the new Section 210, the NRC staff submitted NUREG-0410 to Congress on January 1, 1978. This NUREG describes the NRC generic issues program. The NRC program was already in place when PL 95-209 was enacted and is of considerably broader scope than the unresolved safety issues plan required by Section 210. In the letter transmitting NUREG-0410 to the Congress on December 30, 1977, the Commission indicated: "the progress reports, which are required by Section 210 to be included in future NRC annual reports, may be more useful to Congress if they focus on the specific Section 210 safety items."

It is the NRC's view that the intent of Section 210 was to ensure that plans were developed and implemented on issues with potentially significant public safety implications. In 1978, the NRC undertook a review of more than 130 generic issues addressed in the NRC program to determine which issues fit this description and qualify as unresolved safety issues for reporting to the Congress. The NRC review included the development of proposals by the NRC staff and review and final approval by the NRC Commissioners.

The review is described in a report, NUREG-0510, "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants - A Report to Congress," dated January 1979. The report provides the following definition of an unresolved safety issue:

> An Unresolved Safety Issue is 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 that involves conditions not likely to be acceptable over the lifetime of the plants it affects.

Further, the report indicates that in applying this definition, matters that pose "important questions concerning the adequacy of existing safety requirements" were judged to be those for which resolution is necessary to (1) compensate for a possible major reduction in the degree of protection of the public health and safety, or (2) provide a potentially significant decrease in the risk to the public health and safety. Quite simply, an unresolved safety issue is potentially significant from a public safety standpoint and its resolution is likely to result in NRC action on the affected plants.

All of the issues addressed in the NRC program were systematically evaluated against this definition as described in NUREG-0510. As a result, 17 unresolved safety issues addressed by 22 tasks in the NRC program were identified.

An in-depth and systematic review of generic safety concerns identified between January 1979 and March 1981 was performed by the staff to determine if any of these issues should be designated as Unresolved Safety Issues. The candidate issues originated from concerns identified in NUREG-0660, "NRC Action Plan as a Result of the TMI-2 Accident"; from ACRS recommendations; from abnormal occurrence reports; and from other operating experience. The staff's proposed list was reviewed and commented on by the ACRS, the Office of Analysis and Evaluation of Operational Data (AEOD), and the Office of Policy Evaluation. The ACRS and AEOD also proposed that several additional Unresolved Safety Issues be considered by the Commission. The Commission considered the above information and approved the four Unresolved Safety Issues A-45 through A-48. A description of the review process for candidate issues, together with a list of the issues considered, is presented in NUREG-0705, dated March 1981. An expanded discussion of each of the new Unresolved Safety Issues is also in

NUREG-0705. In addition to the four issues identified above, in December 1981 the Commission approved another issue, A-49, Pressurized Thermal Shock, as an Unresolved Safety Issue.

The issues are listed below. The number(s) of the generic task(s) (for example, A-1) in the NRC program addressing each issue is indicated in parentheses following the title.

Unresolved Safety Issues (Applicable Task Nos.)

(1) Waterhammer (A-1)
(2) Asymmetric blowdown loads on the reactor coolant system (A-2)
(3) Pressurized water reactor steam generator tube integrity (A-3, A-4, A-5)
(4) BWR Mark I and Mark II pressure suppression containments (A-6, A-7, A-8, A-39)
(5) Anticipated transients without scram (A-9)
(6) BWR nozzle cracking (A-10)
(7) Reactor vessel materials toughness (A-11)
(8) Fracture toughness of steam generator and reactor coolant pump supports (A-12)
(9) Systems interaction in nuclear power plants (A-17)
(10) Environmental qualification of safety-related electrical equipment (A-24)
(11) Reactor vessel pressure transient protection (A-26)
(12) Residual heat removal requirements (A-31)
(13) Control of heavy loads near spent fuel (A-36)
(14) Seismic design criteria (A-40)
(15) Pipe cracks at boiling water reactors (A-42)
(16) Containment emergency sump reliability (A-43)
(17) Station blackout (A-44)
(18) Shutdown decay heat removal requirements (A-45)
(19) Seismic qualification of equipment in operating plants (A-46)
(20) Safety implications of control systems (A-47)
(21) Hydrogen control measures and effects of hydrogen burns on safety equipment (A-48)
(22) Pressurized thermal shock (A-49)

Nine of the 27 tasks identified with the unresolved safety issues are not applicable to RESAR-SP/90. Six of these nine tasks (A-6, A-7, A-8, A-10, A-39, and A-42) are applicable only to boiling water reactors. Task A-4 and A-5 address steam generator tube problems in Combustion Engineering and Babcock and Wilcox plants. A-46 deals with seismic qualification of equipment in operating plants and does not apply to RESAR-SP/90. RESAR-SP/90 will be required to be designed on the basis of current seismic design criteria, and commitments for seismic equipment qualification should be in accordance with the latest codes and standards (see Sections 3.9.2 and 3.10 of this SER). With regard to the remaining tasks that are applicable to this facility, the NRC staff has issued NUREG reports providing its proposed resolution of nine of these issues (Table 1). Each of these will be addressed in this Safety Evaluation Report in a future supplement. The table below lists those issues and the section of this SER in which they will be discussed.

The remaining issues applicable to this facility are

- A-3 Westinghouse Steam Generator Tube Integrity
- A-17 Systems interaction in nuclear power plants
- A-40 Seismic design criteria
- A-43 Containment emergency sump reliability
- A-44 Station blackout
- A-45 Shutdown decay heat removal requirements
- A-47 Safety implications of control systems
- A-48 Hydrogen control measures and effects of hydrogen burns on safety equipment
- A-49 Pressurized thermal shock

Table 1 Unresolved Safety Issues applicable to RESAR-SP/90 addressed in this report

Task No.	NUREG Report and Title	SER Section
A-1	NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants"	5.4, 6.3, 9.2

A-2	NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems"	3.9.2.3
A-9	NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," Vol 4	15.3.8
A-11	NUREG-0744, "Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue," Vols I and II, Revision 1.	5.3
A-12	NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports," Revision 1.	--
A-24	NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Revision 1.	3.11
A-26	NUREG-0224, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors" and RSB BTP 5-2	5.2.2
A-31	SRP 5.4.7 and BTP 5-1, "Residual Heat Removal Systems," incorporate requirements of USI A-31.	5.4.3
A-36	NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"	9.1.4

Task Action Plans for Unresolved Safety Issues for which a staff NUREG Report presenting the technical findings on the issue has been issued, are included in the Appendix to the report. Task Action Plans for Unresolved Safety Issues for which no staff NUREG has been issued and for which work is continuing are presented in NUREG-0649, Revision 1 "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants."

Each task action plan provides a description of the problem; the staff's approach to its resolution; a general discussion of the bases upon which continued plant licensing or operation can proceed pending completion of the task; the technical organizations involved in the task and estimates of the manpower required; a description of the interactions with other NRC offices, the ACRS, and outside organizations; estimates of funding required for contractor-supplied technical assistance; prospective dates for completing the task; and a description of potential problems that could alter the planned approach or schedule.

In addition to the Task Action Plans, the staff issues the "Unresolved Safety Issues Summary, Aqua Book" (NUREG-0606) on a quarterly basis; this report provides current schedule information for each of the Unresolved Safety Issues. It also includes information relative to the implementation status of each Unresolved Safety Issue for which technical resolution is complete.

The staff has reviewed the Unresolved Safety Issues listed above as they relate to RESAR-SP/90. Discussion of each of these issues, including references to related discussions in the Safety Evaluation Report, is in Section C.3. Based on its review, the staff concludes for the reasons set forth in Section C.3

that there is reasonable assurance that plants referencing RESAR-SP/90 can be operated before the ultimate resolution of these generic issues without endangering the health and safety of the public.

C.3 Discussions of USIs as they Relate to RESAR SP/90

This section provides the NRC staff's evaluation of RESAR SP/90 for each of the applicable Unresolved Safety Issues. This includes the staff's bases for licensing before ultimate resolution of these issues.

A-3 Steam Generator Tube Integrity

The primary concern is the capability of steam generator tubes to maintain their integrity during normal operation and postulated accident conditions.

Westinghouse steam generators have experienced tube degradation in several forms. These are wastage, intergranular attack, stress corrosion cracking and denting. Each of these form of degradation are discussed below, and specific measures to prevent their occurrence will be considered:

(1) Wastage is characterized by general loss of metal from the tube wall due to a chemical corrosive reaction. Wastage has occurred only in steam generators which used sodium phosphate as a chemical additive. The RESAR SP/90 steam generators can use a water treatment consisting of hydrazine and ammonium hydroxide (this is called all volatile treatment or AVT). Wastage has not been observed in steam generators using all volatile chemistry control.

(2) Intergranular attack is a chemical reaction wherein the grain boundaries of the Inconel 600 tubes are attacked by chemical solutions. Significant intergranular attack has occurred only in steam generators which have an open crevice in the tube to tubesheet area. In steam generators where there is no open crevice in the tubesheet area, intergranular attack should be eliminated.

(3) Stress corrosion cracking (SCC) refers to intergranular cracking of stressed tubes, without reference to a causative chemical agent. This term is used either to encompass a number of known SCC mechanisms or when the chemical causing the corrosion is not known. SCC resistance of steam generator tubes has been improved by a special thermal treatment.

Primary side SCC has also occurred in a number of Westinghouse steam generators in the narrow radius U-bend area of the tubes in the bundle interior. The inconel tubing of the inner 8 rows can receive a stress relief heat treatment that has demonstrated improved resistance to primary side stress corrosion cracking.

(4) Denting is the most serious degradation problem encountered in Westinghouse steam generators. Denting is caused by rapid corrosion of the tube support plates at the holes where the tubes pass through the support plates. The tube support plates can be manufactured from ferritic stainless steel material, which has been shown in laboratory tests to be corrosion resistent to the operating environment. The tube support plates can be designed and manufactured with broached holes rather than drilled holes. The broached hole design promotes high velocity flow along the tubes thereby sweeping impurities away from the support plate locations.

Pending completion of Task A-3, measures can be taken to minimize the steam generator tube problems encountered. Further, the inservice inspection and Technical Specification requirements will assure that the applicants and the NRC staff are alerted to tube degradation should it occur. Appropriate actions such as tube plugging, increased and more frequent inspections and power derating could be taken if necessary. Since the improvements that will result from Task A-3 are expected to be procedural, i.e., improved inspection of the steam generators, they can be implemented by the applicant after operation of this facility begins, if necessary.

The staff has not yet completed its review of the RESAR SP/90 steam generators. Pending completion of a satisfactory review of the steam generator design, we have concluded that the plants referencing RESAR SP/90 can be operated prior to final resolution of this generic issue without undue risk to the health and safety of the public.

Task A-17 Systems Interactions in Nuclear Power Plants

The staff's systems interaction program was initiated in May 1978 with the definition of Unresolved Safety Issue A-17 (Systems Interactions in Nuclear Power Plants). The concern arises because the design, analysis, and installation of systems are frequently the responsibility of teams of engineers with functional specialties such as civil, electrical, mechanical, or nuclear. Experience at operating plants has led to questions of whether the work of these functional specialists is sufficiently integrated to enable them to minimize adverse interactions among systems. Some adverse events that occurred in the past might have been prevented if the teams had ensured the necessary independence of safety systems under all conditions of operation. A typical plant is evaluated against current licensing requirements that are founded on the principle of defense-in-depth. Adherence to this principle and conformance to the regulations (e.g., General Design Criteria) results in requirements such as physical separation and independence of redundant safety systems as well as protection against hazards such as high-energy line ruptures, missiles, high winds, flooding, seismic events, and fires. These design provisions are subject to review against the Standard Review Plan (NUREG-0800), which requires interdisciplinary reviews of safety-grade equipment and addresses different types of potential systems interactions. Also, the quality assurance program that is followed during the design, construction, and operational phases for each plant contributes to the prevention of introducing adverse systems interactions.

The NRC staff's current review procedures assign primary responsibility for review of various technical areas to specific organizational units and secondary responsibility to other units where there is a functional interface. Designers follow somewhat similar procedures and provide the analyses of systems and interface reviews. Task A-17 is investigating methods that could identify adverse systems interactions that were not uncovered by current review procedures.

The development of systematic ways to identify, rank, and evaluate systems interactions could further reduce the likelihood of inter-system failures resulting in the loss of plant safety functions. A comprehensive program may use analytical methods, visual inspections, and experience feedback, for hidden dependencies.

Westinghouse has acknowledged the area of systems interactions as a major design consideration for RESAR SP/90 and committed to address it in a number of ways. First, Westinghouse states that the 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 RESAR-SP/90 layout will provide improved physical separation between safeguard trains A and B as well as between the safeguard trains and the control systems.

Second, Westinghouse states that their intent 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 RESAR SP/90 plant layout, safety system design, and equipment selection is to avoid unacceptable systems interactions.

Third, Westinghouse committed to perform a comprehensive systems interactions analysis as part of the RESAR SP/90 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 RESAR-SP/90 design.

Based on the foregoing discussion, the staff concludes that there is reasonable assurance that plants referencing RESAR-SP/90 can be operated pending ultimate resolution of this generic issue, without endangering the health and safety of the public.

Task A-40 Seismic Design Criteria - Short-Term Program

NRC regulations require that nuclear power plant 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 the NRC regulations and in regulatory guides issued by the Commission. Safety analysis reports from each license applicant are reviewed in accordance with the review and acceptance criteria in the Standard Review Plan.

The intent of Task A-40 is to provide changes in Standard Review Plan seismic design requirements that reflect the current state-of-the-art in seismic design and provide more realistic estimates of the response of structures, components and equipment subjected to seismic loadings. Task A-40 is, in effect, a compendium of tasks to support re-evaluation of operating plants and provide revisions to appropriate sections of the Standard Review Plan to bring them more in line with the state-of-the-art in seismic design and analysis.

Safety-related structures, systems, and components for RESAR-SP/90 plants will be designed to withstand the effects of earthquakes in accordance with current NRC regulations, regulatory guides, and the Standard Review Plan, as discussed in Sections 3.7, 3.9, and 3.10 of the SRP. Specifically, the five subjects identified in the NRC's issue description for Task A-40, i.e., magnitude of earthquakes (safe shutdown earthquake (SSE)), free-field motion (SSE), soil-structures interactions, motion of plant equipment, and load combination are discussed therein. Design of structures for protection against natural phenomena such as earthquake is described in SRP Section 3.8. Should the resolution of USI A-40 indicate that a change is needed in these licensing requirements, license applicants, including plants referencing RESAR SP/90 will be re-evaluated on a case-by-case basis.

Accordingly, the staff concludes that there is reasonable assurance that plants referencing RESAR SP/90 can be constructed and operated before ultimate resolution of this generic issue without endangering the health and safety of the public.

Task A-43 <u>Containment Emergency Sump Reliability</u>

Following a postulated loss-of-coolant accident, water would be collected in the containment emergency sump for use in the long-term recirculation mode, thus maintaining core cooling. This water could also be circulated through the containment spray cooling system for removal of heat and fission products within containment. The principal safety concern is loss of the ability to draw water from the containment emergency sump under post-LOCA conditions--thus leading to the degradation of, or disability of, the long-term recirculation safety train and impairment of decay heat removal.

Two major concerns have been postulated: (1) adverse hydraulic conditions in the sump (e.g., air ingestion, break flow effects, vortex formation, etc.) thereby leading to loss of residual heat removal pumping, and (2) severe sump screen blockages resulting from LOCA-generated insulation debris, which could cause loss of net positive suction head requirements.

The evaluation of such safety concerns has been carried out, and the technical findings are reported in NUREG-0897. The result has been a recommended revision to RG 1.82 that reflects these findings. The destruction of plant insulation by the LOCA jet is viewed as a potential safety concern relative to screen blockage. The evaluation of debris blockage is a plant-specific requirement resulting from design differences and types of insulation employed. Air ingestion and vortex formation are not as serious as previously hypothesized. NUREG-0897 and NUREG-0869 (which includes the proposed RG 1.82, Revision 1) and SRP 6.2.2, were issued for public comment in May 1983. The requirements that may result from A-43 are expected to be primarily procedural, i.e., an assessment of sump blockage following a posulated LOCA. Plant modifications, if necessary, can be implemented after operation of the facility begins.

With respect to RESAR-SP/90 plant, the applicant has not provided detailed information regarding the sump design. The staff will evaluate the sump design to ensure that the RESAR-SP/90 plants meet the NRC requirements.

Based on the above, the staff has concluded that, subject to the completion of a satisfactory review of the sump design, there is reasonable assurance that plants referencing the RESAR-SP/90 design can be operated before the ultimate resolution of this generic issue without undue risk to the health and safety of the public.

Task A-44 Station Blackout

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 these requirements. Each electrical division for safety systems includes two offsite alternating current (ac) power connections, a standby emergency diesel generator alternating current power supply, and direct current sources.

Task A-44 involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all alternating current power, that is, a loss of both the offsite and the emergency diesel generator alternating current power supplies. This issue arose because of operating experience regarding the reliability of alternating current power supplies. A number of operating plants have experienced a total loss of offsite electrical power, and more occurrences are expected in the future. In almost every one of these loss-of-offsite power events, the onsite emergency alternating current power supplies were available immediately to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power supplies has been unavailable. In a few cases there has been a complete loss of ac power, but during these events, ac power was restored in a short time without serious consequences. In addition, there have been numerous instances of emergency diesel generators failing to start and run in operating plants during periodic surveillance tests.

A loss of all ac power is not required currently by NRC to be a design-basis event for the RESAR SP/90 facility. Nonetheless, a combination of design, operating, and testing requirements is required to ensure that RESAR SP/90 plants will have substantial resistance to a loss of all alternating current and that, even if a loss of all ac power should occur, there is reasonable assurance the core will be cooled. These design, operating, and testing requirements are discussed below.

A loss of offsite ac power involves a loss of both the preferred and backup sources of offsite power. The staff's review methods for the design, inspection, and testing provisions for the offsite power system are described in Section 8.2 of the Standard Review Plan (SRP).

If offsite ac power is lost, the diesel generators and their associated distribution systems will deliver emergency power to safety-related equipment. The staff's review methods for the design, testing, surveillance, and maintenance provisions for the onsite emergency diesels are described in Sections 8.3 and 9.6 of the SRP. Staff requirements include preoperational testing to ensure the reliability of the installed diesel generators is in accordance with the provisions of Regulatory Guide 1.108. In addition, an applicant referencing RESAR SP/90 would be required to implement a program for enhancing diesel generator reliability to ensure the long-term reliability of the diesel generators. This program resulted from recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Generator Reliability."

If both offsite and onsite ac power are lost, cooling water can still be provided to the steam generator by the auxiliary feedwater system employing a steam turbine driven pump that does not rely on ac power for operation. In addition, the RESAR SP/90 design includes an ac-independent seal injection system to provide cooling for the reactor coolant pump seals and primary system charging capability in the event of a total loss of ac power. This feature will enhance the ability of the RESAR SP/90 plants to withstand a station blackout. However, the applicant has not provided detailed information for the staff's evaluation. When the information is made available to the staff, the results of the evaluation will be reported in a supplement to the SER.

In addition to the above, the Commission has determined that some interim measures should be taken at all plants to accommodate a station blackout pending resolution of the issue. Consequently, the NRC requested (Generic Letter 81-04, dated February 25, 1981) a review of plant operation to determine the applicant's capability to mitigate a station blackout event and properly implement, as necessary, emergency procedures and training programs for station blackout events. Westinghouse will develop loss of all ac power emergency response guidelines for use by utilities utilizing the RESAR SP/90 design.

Based on the above considerations, the staff concludes that subject to the satisfactory evaluation by the staff regarding the systems described above, there is reasonable assurance that plants referencing RESAR SP/90 can be operated before the ultimate resolution of this generic issue without endangering the health and safety of the public.

Task A-45 Shutdown Decay Heat Removal Requirements

Under normal operating conditions, power generated within a reactor is removed as steam to produce electricity through 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 the reactor is shut down, other measures must be available to remove decay heat from the reactor to ensure that high temperatures and pressures do not develop that could jeopardize the reactor and the reactor coolant system. It is evident, therefore, that all light-water reactors (LWRs) share two common decay-heat-removal functional requirements: (1) to provide a means of transferring decay heat from the reactor coolant system to an ultimate heat sink, and (2) 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.

The TMI-2 accident demonstrated how a relatively common fault, with which the operator should have been able to cope easily, could escalate into a potentially hazardous situation, with severe financial losses to the utility, as a result of difficulties arising in the decay heat removal (DHR) process.

Other circumstances, of a more unusual nature (e.g., damage to systems by external events such as floods or earthquakes, or by sabotage), which could make removal of the decay heat difficult, can also be foreseen.

The question arises, therefore, whether current licensing design requirements are adequate to ensure that LWRs do not pose unacceptable risk as a result of a failure to remove shutdown decay heat, and whether, at a cost commensurate with the increase in safety that could be achieved, improvements could be made in the effectiveness of shutdown decay heat removal in one or more transient or accident situations. Resolution of this question is considered to be of sufficient importance to merit raising it to the status of an unresolved safety issue.

To some extent, the effectiveness of the DHR systems is linked to that of the onsite and offsite electrical supplies; the performance and reliability of those supplies is being considered in A-44, Station Blackout. Consequently, the scope of work required in relation to the decay heat removal systems is complementary to Task A-44 above.

The overall purpose of Task A-45 is to evaluate the adequacy of current licensing design requirements to ensure that nuclear power plants do not pose an unacceptable risk because of a failure to remove shutdown decay heat. This will require the development of a comprehensive and consistent set of shutdown cooling requirements for existing and future LWRs, including the study of alternative means of shutdown decay heat removal and of diverse "dedicated" systems for this purpose.

This task will evaluate the benefit of providing alternate means of DHR that could substantially increase the plant's capability to handle a broader spectrum of transients and accidents. The study will include a number of plant-specific DHR systems evaluations and will result in recommendations regarding the desirability of, and possible design requirements for, improvements in existing systems or an alternative decay heat removal method, if the improvements or alternatives can significantly reduce the overall risk to the public in a cost-effective manner.

An integrated systems approach to the problem will be employed. Accordingly, quantitative methods will be used, where possible, to define design requirements for future plants and to measure the effectiveness and acceptability of the shutdown DHR systems in existing plants. The principal means for removing the decay heat in a PWR under normal conditions immediately following reactor shutdown is through the steam generators, using the auxiliary feedwater system. In addition to the WASH-1400 study (NUREG-75/014), later reliability studies and related experience from the Three Mile Island Unit 2 (TMI-2) accident have reaffirmed that the loss of capability to remove heat through the steam generator is a significant contributor to the probability of a core melt event. The staff's review of the auxiliary feedwater system design and operation has not yet been completed.

It should be noted, as discussed below, that the NRC required licensees to implement many improvements to the steam generator auxiliary feedwater system following the TMI-2 accident. However, the staff still believes that providing an alternative means of decay heat removal could substantially increase the plant's capability to deal with a broader spectrum of transients and accidents and potentially could, therefore, significantly reduce the overall risk to the public. Consequently, this task will investigate alternative means of decay heat removal in PWR plants, including but not limited to, using existing equipment where possible. This study will include a representative sample of plant-specific DHR system evaluations. It will result in recommendations regarding the adequacy of existing DHR requirements and the desirability of, and possible design requirements for, an alternative DHR method, other than that normally associated with the steam generator and secondary coolant system.

The auxiliary feedwater (AFW) system is a very important safety system in a PWR in terms of providing a heat sink via the steam generators to remove core decay heat. As mentioned above, the TMI-2 accident and subsequent studies have further highlighted the importance of the AFW systems. As discussed below, the NRC staff has required certain upgrading of the AFW systems for all LWRs following the TMI-2 accident. Although this task will investigate alternative means of decay heat removal, the NRC staff concludes that in general (not on a plant-specific basis) if the licensees comply with the upgrading of requirements for the AFW system, the action taken following the TMI-2 accident justifies continued operation and licensing pending completion of this task. Further discussion and the bases for this view are provided below.

° TMI-2 Accident

The accident at TMI-2 on March 28, 1979 involved a main feedwater transient coupled with a stuck-open pressurizer power-operated relief valve and a temporary failure of the auxiliary feedwater system, and subsequent operator intervention to severly reduce flow from the safety injection system. The resulting severity of the ensuing events and the potential generic aspects of the accident on other operating reactors led the NRC to initiate prompt action to: (1) ensure that other reactor licensees, particularly those with plants similar in design to TMI-2, took the necessary action to substantially reduce the likelihood for TMI-2-type events, and (2) investigate the potential generic implications of this action for other operating reactors.

The Bulletins and Orders Task Force (BOTF) was established within the NRC Office of Nuclear Reactor Regulation (NRR) in early May 1979 and completed its work on December 31, 1979. This task force was responsible for reviewing and directing the TMI-2-related staff activities associated with the NRC Office of Inspection and Enforcement (IE) Bulletins, Commission Orders, and generic evaluations of loss-of-feedwater transients and small-break loss-of-coolant accidents for all operating plants to ensure their continued safe operation. NUREG-0645, "Report of the Bulletins and Orders Task Force," summarizes the results of the work performed.

° Generic and Plant-Specific Studies

For B&W-designed operating reactors, an initial NRC staff study was completed and published in NUREG-0560, "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock & Wilcox Company." This study considered the particular design features and operational history of B&W-designed operating plants in light of the TMI-2 accident and related current licensing requirements. As a result of this study, a number of findings and recommendations resulted that are now being pursued.

Generally, the activities involving the B&W-designed reactors are reflected in the actions specified in the Commission Orders. Consequently, a number of actions have been specified regarding transient and small-break analyses, upgrading of auxiliary feedwater reliability and performance, procedures for operator action, and operator training. The results of the NRC staff review of the B&W small-break analysis are published in NUREG-0565, "Generic Evaluation of Small-Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox-Designed Operating Plants."

Similar studies have been completed for operating plants designed by Westinghouse (W), Combustion Engineering (CE), and General Electric (GE). Those studies, which also focus specifically on the predicted plant performance under different accident scenarios involving feedwater transients and small-break LOCAs, are published in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants."

Based on the review of the operating plants in light of the TMI-2 accident, the NRC staff reached the following conclusions:

(1) The continued operation of the operating plants is acceptable provided that certain actions related to the plants' design and operation and training of operators identifed in NUREG-0645 are implemented, consistent with the recommended implementation schedules.

(2) The actions taken by the licensees with operating plants in response to the IE Bulletins (including the actions specified in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors") provide added assurance for the protection of the health and safety of the public.

In addition, the BOTF independently confirmed the safety significance of those related actions recommended by other NRR task forces as discussed in NUREG-0645.

° Pressurized Water Reactors

The primary method for removal of decay heat from PWRs is via the steam generators to the secondary system. This energy is transferred on the secondary side to either the main feedwater or auxiliary feedwater systems, and is rejected to either the turbine condenser or the atmosphere via the secondary coolant system safety/relief valves. Following the TMI-2 accident, the importance of the AFW was highlighted and a number of improvements were made to improve the relaibility of the AFW (NUREG-0645). It was also required that operating plants be capable of providing the required AFW flow for at least 2 hours from one AFW pump train independent of any ac power source; that is, if both offsite and onstie ac power sources are lost.

Some PWRs potentially have at least one alternate means of removing decay heat if an extended loss of feedwater is postulated. This method is known as "feed and bleed" and uses the high-pressure injection (HPI) system to add water coolant (feed) at high pressure to the primary system. The decay heat increases the system pressure and energy is removed through the power-operated relief valves (PORVs) and/or the safety valves (bleed), if necessary. It should be noted that some PWRs incorporate HPI pumps that cannot operate at full system pressure (cutoff head about 1500 psi). For those cases, the PORVs can be manually opened, thereby reducing the system pressure to within the operating range of the HPI. Limited vendor analyses have shown that the core can be adequately cooled by this means, provided that the containment pressure can be controlled to a safe level.

When the primary system is at low pressure, the long-term decay heat is removed by the residual heat removal system to achieve and maintain cold shutdown conditions. Task A-45 will also consider the adequacy of reliability and performance criteria and standards for RHR systems. The staff has not yet completed its review of the Integrated Safeguards System (ISS) design and operation. The ISS includes the RHR function and the capability to "feed and bleed" the primary coolant.

° Conclusion

In summary, because of the upgrading of current DHR systems that was required following the TMI-2 accident, the staff concludes that, in general, plants may continue to be licensed and operated before the ultimate resolution of this generic issue without endangering the health and safety of the public. However, licensee compliance with the upgrading of DHR system requirements must be examined by the staff on an individual case basis. For RESAR-SP/90 the staff is still reviewing information related to this issue as indicated above. Consequently, the staff has concluded that, subject to the satisfactory review of the AFW and RHR systems, there is reasonable assurance that plants referencing RESAR-SP/90 can be operated prior to ultimate resolution of this generic issue without undue risk to the health and safety of the public.

Task A-47 Safety Implications of Control Systems

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--such as a loss of a power supply, short circuit, open circuit, or sensor 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 that a postulated accident could cause control system failures that would 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. Although it is generally believed that such control system failures would not lead to serious events or result in conditions that safety systems could not safely handle, indepth studies have not been rigorously performed to verify this belief. The potential for an accident that would affect a particular control system, and effects of the control system failures, may differ from plant to plant. Therefore, it is not possible to develop generic answers to all these concerns; it is possible to develop generic criteria that can be used for future plant-specific reviews. The purpose of this Unresolved Safety Issue task is to verify the adequacy of existing criteria for control systems or propose additional generic criteria (if necessary) that will be used for plant-specific review.

The Westinghouse safety systems have been designed with the goal of ensuring that control system failures (either single or multiple) will not prevent automatic or manual initiation and operation of any safety system equipment required to trip the plant or to maintain the plant in a safe shutdown condition following any anticipated operational occurrence or accident. This has been accomplished by either providing independence between safety- and nonsafety-grade systems or providing isolating devices between safety- and nonsafety-grade systems. These devices preclude the propagation of nonsafety-grade system equipment faults so that operation of the safety-grade system equipment is not impaired.

A wide range of bounding transients and accidents is presently analyzed to ensure that the postulated events would be adequately mitigated by the safety systems. In addition, systematic reviews of safety systems have been performed with the goal of ensuring that the control system failures (single or multiple) will not defeat safety system action.

As part of the operating license review the applicants are requested (NRC Information Notice 79-22, "Qualification of Control Systems," September 17, 1979) (1) to review the possibility of consequential control system failures that could exacerbate the effects of high-energy line breaks (HELBs) and (2) to adopt new operator procedures, where needed, to ensure that the postulated events would be adequately mitigated. As part of the review, the staff will also evaluate the qualification program to ensure that equipment that may potentially be exposed to HELB environments has been adequately qualified or an adequate basis has been provided for not qualifying the equipment to the limiting hostile environment.

With the recent emphasis on the availability of postaccident instrumentation (Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident"), the staff's reviews evaluate the designs to ensure that control system failures will not deprive the operator of information required to maintain the plant in a safe shutdown condition after any anticipated operational occurrence or accident. During the operating license review the applicants are requested to evaluate the control systems and identify any control systems whose malfunction could impact plant safety.

In addition, IE Bulletin 79-27 ("Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation," November 30, 1979) will be issued to applicants referencing RESAR SP/90 requesting that evaluations be performed to ensure the adequacy of plant procedures for accomplishing shutdown on loss of power to any electrical bus supplying power for instruments and controls.

Current plants in the licensing review process must respond adequately to the above request. As part of the RESAR SP/90 response, Westinghouse stated that functional requirements and design specifications for the RESAR SP/90 control system will be no less significant than those for current plants and that an analysis similar to that performed on recently licensed plants would show that the consequences of failures in the control systems of the RESAR SP/90 would be bounded by the FSAR type analysis. Westinghouse also stated that a control system failure study will be performed and documented during the licensing process of the RESAR SP/90 design. We conclude that this commitment is satisfactory for this stage of the review. We will review the applicant's analysis and their response to the above mentioned concerns during the licensing process.

Based on the above, the staff concludes that, subject to the satisfactory review by the staff on the systems described above, plants referencing the RESAR SP/90 design can be operated before complete resolution of this issue without undue risk to the health and safety of the public.

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

Following a loss-of-coolant accident in a light-water reactor plant, combustible gases, principally hydrogen, may accumulate inside the primary reactor containment as a result of: (1) metal-water reaction involving the fuel element cladding; (2) the radiolytic decomposition of the water in the reactor core and the containment sump; (3) the corrosion of certain construction materials by the spray solution; and (4) any synergistic chemical, thermal, and radiolytic effects of post-accident environmental conditions on containment protective coating systems and electric cable insulation.

Because of the potential for significant hydrogen generation as the result of an accident, 10CFR 50.44, "Standards for Combustible Gas Control System in Light Water Cooled Power Reactors," and GDC 41, "Containment Atmosphere Cleanup," require that systems be provided to control hydrogen concentration in the containment atmosphere following a postulated accident to ensure that containment integrity is maintained.

10 CFR 50.44 requires that the combustible gas control system provided be capable of handling the hydrogen generated as a result of degradation of the emergency core cooling system so that the hydrogen release is five times the amount calculated in demonstrating compliance with 10 CFR 50.46 or the amount corresponding to reaction of the cladding to a depth of 0.00023 in. whichever amount is greater.

The accident at TMI-2 resulted in hydrogen generation well in excess of the amounts specified in 10 CFR 50.44. As a result, it became apparent to NRC that specific design measures are needed for handling larger hydrogen releases. The NRC also determined that a rulemaking proceeding should be undertaken to define the manner and extent to which hydrogen evolution and other effects of a degraded core need to be taken into account in plant design. An advance notice of this rulemaking proceeding on degraded core issues was published in the Federal Register on October 2, 1980.

Recognizing that a number of years may be required to complete this rulemaking proceeding, a set of short-term or interim actions relative to hydrogen control requirements were developed and implemented. With respect to the issue of hydrogen control for new plant design, the NRC issued a final rule on January 15, 1982 in the Federal Register (47FR2286), known as "The Near-Term Construction Permits and Manufacturing Licenses Rule (NTCP/ML). This rule is not limited to hydrogen issues, it addresses other issues that are an outgrowth of the TMI accident.

As stated in Section 3.1 of RESAR SP/90. The applicant commits to design the hydrogen control systems in accordance with the requirements specified in the NTCP/ML rule. Further, the applicant also commits to perform all calculations and analyses considering the additional zircaloy in the RESAR SP/90 design rather than restricting the calculations to 100 percent of the fuel cladding as required by the rule.

Based on the above, the staff concludes that, subject to the satisfactory review of the proposed hydrogen control system stated above, plants referencing the RESAR SP/90 design can be operated before ultimate resolution of this issue without undue risk to the health and safety of the public.

A-49 Pressurized Thermal Shock

The issue of pressurized thermal shock (PTS) arises because in pressurized water reactors (PWRs) transients and accidents can occur that result in severe overcooling (thermal shock) of the reactor pressure vessel, concurrent with or followed by repressurization. In these PTS events, rapid cooling of the reactor vessel internal surface results in thermal stress with a maximum tensile stress at the inside surface of the vessel. The magnitude of the thermal stress depends on the temperature profile across the reactor vessel wall as a function of time. The effects of this thermal stress are compounded by pressure stresses.

Severe reactor system overcooling events simultaneous with or followed by pressurization of the reactor vessel (PTS events) can 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 (LOCAs), main steam line breaks (MSLBs), and feedwater line breaks.

The PTS issue is a concern for PWRs only after the reactor vessel has lost its fracture toughness properties and is embrittled by neutron irradiation. The standards and regulatory requirements to which a RESAR SP/90 reactor vessel will be designed and fabricated are described in Section 4.0 of the PSAR.

As long as the fracture resistance of the reactor vessel material is relatively high, overcooling events are not expected to cause vessel failure. However, the fracture resistance of reactor vessel materials decreases with exposure to fast neutrons during the life of a nuclear power plant. The rate of decrease is dependent on the metallurgical composition of the vessel walls and welds. If the fracture resistance of the vessel has been reduced sufficiently by neutron irradiation, severe overcooling events could cause propagation of small flaws that might exist near the inner surface. 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.

For the reactor pressure vessel to fail and constitute a risk to public health and safety, a number of contributing factors must be present. These factors are (1) a reactor vessel flaw of sufficient size to initiate and propagate; (2) a level of irradiation (fluence) and material properties and composition sufficient to cause significant embrittlement (the exact fluence depends on materials present; i.e., high copper content causes embrittlement to occur more rapidly); (3) a severe overcooling transient with repressurization; and (4) the crack resulting from the propagation of initial cracks must be of such size and location that the vessel fails.

As a result of the evaluation of the PTS issue, the staff recommended to the Commission in SECY-82-465 (November 23, 1982) actions to prevent PTS events in operating reactors. The Commission accepted the staff recommendations and the staff has published Notice of Proposed Rulemaking for a rule that would establish an RT_{NDT} screening criterion (below which PTS risk is considered acceptable), require early analysis and implementation of such flux reduction programs as a reasonably practicable method to avoid reaching the screening criterion, and require plant-specific PTS safety analyses before plants are within three calendar years of reaching the screening criterion including analyses of proposed alternatives to minimize the PTS program.

Such a rule has been published for public comment (Federal Register, February 7, 1984) by the staff. We believe that a RESAR SP/90 plant could easily meet the requirements of the proposed rule.

On the basis of the above the staff concludes that there is reasonable assurance that plants referencing the RESAR SP/90 design can be operated before ultimate resolution of this generic issue and completion of the proposed rulemaking without undue risk to the health and safety of the public.