Isa

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

OCT 3 1 1984

Docket Nos. 50-412

MEMORANDUM FOR: George Knighton, Chief
Licensing Branch No. 2
Division of Licensing

FROM: Karl Kniel, Chief
Generic Issues Branch
Division of Safety Technology

SUBJECT: SER INPUT - BEAVER VALLEY POWER STATION, UNIT 2

PLANT NAME: Beaver Valley, Unit 2
DOCKET NUMBER: 50-412
LICENSING STAGE: OL
LICENSING BRANCH AND PROJECT MANAGER: LB#2, L. Lazo
DST BRANCH INVOLVED: Generic Issues Branch
DESCRIPTION OF REVIEW: Unresolved Safety Issues
REVIEW STATUS: Complete
REQUESTED COMPLETION DATE: October 26, 1984

The Generic Issues Branch, DST, input to the Beaver Valley Power Station-Unit 2 (BVPS-2) Safety Evaluation Report is enclosed. This appendix to the SER addresses the status of Unresolved Safety Issues pertaining to the facility, and is in response to the ALAB-444 decision on this subject. That decision suggested that the staff include an explanation of the unresolved safety issues in the SER and the reasons the facility could operate safely pending resolution of those issues.

This appendix references NUREG reports providing the proposed generic resolution of nine of the Unresolved Safety Issues. The BVPS-2 SER sections discussing the plant-specific information for the generic programs are not available at this time. However, the "standard" SER sections were assumed and should be verified by the Project Manager. The Project Manager should also assure that the plant-specific implementation of resolved USIs is addressed in the body of the SER.

As stated in Section C.3 of the Appendix C, we conclude that BVPS-2 can be operated prior to the ultimate resolution of all USI's applicable to this facility. However, we recognize that there are open items or confirmatory

C/44

8411120221 XA

OCT 3 1 1984

items identified by the line branch for the systems related to USIs A-43, and A-47. Since GIB is generally not responsible for reviewing plant-specific information, we listed A-43, and A-47 as a confirmatory item rather than open items. Thus, the conclusion for these two USIs states that the closure of these issues is subject to the satisfactory resolution of the open items by the line branch.

It should be noted that no direct reference is made in this appendix to NUREG-0800, since the Standard Review Plan does not directly address USIs. However, for many open issues, this appendix refers to functional branch reviews referenced in the SER for the licensing basis. These functional branch reviews are addressed by NUREG-0800.

Karl Kniel

Karl Kniel, Chief
Generic Issues Branch
Division of Safety Technology

Enclosures:
Hope Creek Appendix C-USIs

cc: w/o enclosures
F. Schroeder
T. Novak
N. Anderson
R. Silver

w/enclosures
T. Su
P. Norian
L. Lazo

APPENDIX C
NUCLEAR REGULATORY COMMISSION (NRC)
UNRESOLVED SAFETY ISSUES

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)

Ten of the 27 tasks identified with the unresolved safety issues are not applicable to Beaver Valley Unit 2, and six of these ten tasks (A-6, A-7, A-8, A-10, A-39, and A-42) are peculiar to boiling water reactors. Tasks 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 Beaver Valley Unit 2. The staff's evaluation of Beaver Valley Unit 2 seismic equipment qualification is reported in Sections 3.9.2 and 3.10 of this SER. Also, Tasks A-48 is related to PWR plants with ice-condenser containments or BWR with pressure suppression type containments. With regard to the remaining tasks that are applicable to this facility, the NRC staff has issued NUREG reports providing its proposed resolution of eight of these issues (Table 1). Each of these has been addressed in this Safety Evaluation Report or will be addressed in a future supplement. Table 1 lists those issues and the section of this SER in which they are 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-49 Pressurized Thermal Shock

Task Action Plans for Unresolved Safety Issues for which no staff NUREG report 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 Beaver Valley Unit 2. 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 Beaver Valley Unit 2 can be operated before the ultimate resolution of these generic issues without endangering the health and safety of the public. Tasks A-43, and A-47 are accepted subject to the resolution of those confirmatory items identified in Section C.3. The resolution of these items will be reported in a supplement to the SER.

C.3 Discussions of USIs as they Relate to Beaver Valley Unit 2

This section provides the NRC staff's evaluation of Beaver Valley Unit 2 for each of the applicable Unresolved Safety Issues. This includes the staff's bases for licensing before ultimate resolution of these issues.

A-3 Westinghouse 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.

In addition, the requirement for increased steam generator tube inspections and repairs have resulted in significant increases in occupational exposures to workers. Corrosion resulting in steam generator tube wall thinning (wastage) has been observed in several Westinghouse plants. However, plants operating exclusively with an all-volatile secondary water treatment process have not experienced this form of degradation. Another observed corrosion-related phenomenon has been buildup of support plate corrosion products in the annulus between the tubes and support plates. This buildup may eventually cause a diametric reduction of the tubes, called "denting" and deformation of the tube support plates. This phenomenon may lead to other problems, including stress corrosion cracking, leaks at the tube/support plate intersections, and U-bond section cracking of tubes which were highly stressed because of support plate deformation.

Several measures such as a secondary water chemistry control and monitoring program, that the applicant has employed to minimize the onset of steam generator tube problems, include the following:

- Beaver Valley Unit 2 will operate on all-volatile secondary water treatment process.

- Inservice inspection requirements meet the requirements of Regulatory Guide 1.83, Revision 1.

- Steam generator blowdown and sludge lancing will be used to control sludge if it forms.

Duquesne Light Company (DLP) is a member of EPRI, the Steam Generator Owners Group (SGOG) and the Westinghouse Owners Group (WOG). DLP representatives regularly participate in the meetings to maintain awareness of industry experience and obtain updates on research in progress. This information will be considered as appropriate. Pending completion of Task A-3, the measures taken at this facility should minimize the steam generator tube problems encountered. Further, the inservice inspection and Technical Specification requirements will assure that the applicant 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.

Based on the foregoing, we have concluded that Beaver Valley Unit 2 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.

The applicant has not described a comprehensive program that separately evaluates all structures, systems, and components important to safety for adverse systems interactions. However, the plant has been 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 various 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 the potential safety significance of adverse systems interactions and possible methods that could identify adverse systems interactions that were not uncovered by current review procedures. After the resolution of A-17, the staff will determine whether the applicant must perform further evaluations for adverse systems interactions.

Based on the foregoing discussion, the staff concludes that there is reasonable assurance that Beaver Valley Unit 2 can be operated safety before ultimate resolution of this generic issue without undue risk to 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. However, there are a number of plants with construction permits and operating licenses issued before the NRC's current regulations and regulatory guidance were in place. For this reason, reviews of the seismic design of various plants are being undertaken again to ensure that these plants do not present an undue risk to 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. In addition, some revisions to sections of the Standard Review Plan and regulatory guides to bring them more in line with the state of the art will result.

Safety-related structures, systems, and components for Beaver Valley Unit 2 are 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 FSAR. 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. The design of structures for protection against natural phenomena such as earthquakes is described in FSAR Section 3.8. Should the resolution of USI A-40 indicate that a change is needed in these licensing requirements, all operating reactors, including Beaver Valley Unit 2 will be re-evaluated on a case-by-case basis.

Accordingly, the staff concludes that there is reasonable assurance that Beaver Valley Unit 2 can be operated before ultimate resolution of this generic issue without undue risk to 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 head 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 have been reported in NUREG-0897 (For Comment). The destruction of plant insulation by the LOCA jet is viewed as a potential safety concern relative to screen blockage which could lead to loss of NPSH. The evaluation of debris blockage is a plant-specific concern resulting from plant design differences and types of insulation employed. Air ingestion is a function of water elevation and sump suction intake Froude number.

The BVP-2 sump design differs significantly from those types of design tested under the USI A-43 program. One-third scale model testing of the BVP-2 sump, conducted at the Alden Research Laboratory, resulted in modification of the original sump design and incorporation of horizontal gratings (which suppress vortex formation and air ingestion) to meet acceptable operation requirements at the minimum acceptable water level, assuming 50% blockage. In addition, these model tests revealed recirculation water velocities on the order of 1.0 to 1.7 ft/sec which are sufficiently high to transport LOCA generated debris to the sump screens. Consequently the staff has requested that the applicant provide a debris generation and transport analysis to justify the 50% sump blockage assumption.

Based on the above, and subject to the satisfactory resolution of such items identified in Section 6.2.2, there is reasonable assurance that Beaver Valley Unit 2 can be operated before ultimate resolution of this generic issue without undue risk to the health and safety of the public.

Task A-44 <u>Station Blackout</u>

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 was not a design-basis event for the Beaver Valley Unit 2 facility. Nonetheless, a combination of design, operating, and testing requirements has been imposed to ensure that this unit 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 the loss of all sources of offsite power with the potential to supply BVPS-2. As described in Section 8.2.1.1 of this SER, BVPS-2 has at least five independent and physically separate offsite power supplies. The staff's review and basis for acceptance of the design, inspection, and testing provisions for the offsite power system are described in Section 8.2 of this SER.

If offsite ac power is lost, two independent and physically separate diesel generators and their associated distribution systems will deliver emergency power to safety-related equipment. The staff's review of the design, testing, surveillance, and maintenance provisions for the onsite emergency diesels is described in Sections 8.3 and 9.5 of this SER. Staff requirements include preoperational testing to ensure the reliability of the installed diesel generators in accordance with the provisions of Regulatory Guide 1.108. In addition, the applicant has been 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. The auxiliary feedwater system design and operation is described in SER Section 10.4.9.

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. Appropriate review of the procedures and training programs for station blackout events will be completed before fuel load. Beaver Valley Unit 2 will utilize the Westinghouse Owner's Group Loss of All AC Power Guidelines for development of emergency operating procedures and operator training for station blackout events will be included in operator training programs.

Based on the above considerations, the staff concludes that there is reasonable assurance that Beaver Valley Unit 2 can be operated before the ultimate resolution of this generic issue without undue risk to the health and safety of the public.

Task A-45 <u>Shutdown Decay Heat Removal Requirements</u>

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 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 reguire 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 of 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.

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 is described in Section 10.4.9 of this SER.

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 desriability of, and possible design requirements for, an alternative DHR method, other than that nromally 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.

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. 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's review of the RHR system design and operation is described in Section 5.4.7 of the SER.

* 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 Beaver Valley Unit 2, the staff has concluded that there is reasonable assurance that Beaver Valley Unit 2 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 Beaver Valley Unit 2 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.

Also, the applicant has been 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 is also evaluating 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. The staff's evaluation of the applicant's response to Information Notice 79-22 and the adequacy of the qualification program are reported in Sections 7.7.2.2 and 3.11 of this SER, respectively.

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. The applicant was requested to evaluate the Beaver Valley Unit 2 control systems and identify any control systems whose malfunction could impact plant safety. The applicant has been requested to document the degree of interdependence of these identified control systems and identify the use (if any) of common power supplies and the use of common sensors or common sensor impulse lines whose failure could have potential safety significance. The status of these reviews and the staff's evaluation are discussed in Section 7.7.2.3 of the SER.

In addition, IE Bulletin 79-27 ("Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation," November 30, 1979) was issued to the applicant 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. The results of this review are in SER Section 7.5.2.5.

The subtask of this issue concerning the steam generator overfill transient in pressurized water reactors is currently under review by the staff. Pending ultimate resolution of this item, the applicant has incorporated in the Beaver Valley Unit 2 design a control grade high-level initiation trip signal to trip the main feedwater pumps, the feedwater isolation control and bypass valves and the main turbine to prevent the occurrence of overfill transients. The status of this review and the staff evaluation is discussed in Section 7.3.3.12 of the SER.

On the basis of these above considerations, and subject to the satisfactory resolution of the open items identified in Section 3.11 and items identified in Sections 7.3.3.12 and 7.7.2.3, the staff concludes that there is reasonable assurance that Beaver Valley Unit 2 can be operated before the ultimate resolution of this generic issue without undue risk to the health and safety of the public.

A-49 <u>Pressurized Thermal Shock</u>

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 accidents (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 the Beaver Valley Unit 2 reactor vessel was designed and fabricated are described in Sections 5.2 and 5.3 of the FSAR.

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 the Beaver Valley Unit 2 plant could easily meet the requirements of the proposed rule. The applicant states that the estimated RT_{NDT} values for the Beaver Valley Unit 2 vessel are expected to remain below the applicable NRC-staff-proposed screening criterion of 270°F.

On the basis of the above considerations, the staff concludes that there is reasonable assurance that the Beaver Valley Unit 2 facility can be operated before ultimate resolution of this generic issue without undue risk to the health and safety of the public.

C.5 References

---, NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," January 1978.

---, NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," Vol 4, March 1980.

---, NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Revision 1, July 1981.

---, NUREG-0606, "Office of Nuclear Reactor Regulation Unresolved Safety Issues, Aqua Book," issued quarterly.

---, NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980.

- -, NUREG-0649, Revision 1, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants," September 1984.

---, NUREG-0660, "NRC Action Plan as a Result of the TMI-2 Accident," May 1980.

---, NUREG-0705, "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plants, Special Report to Congress," March 1981.

---, NUREG-0744, "Resolution of the Task A-11 Reaetor Vessel Materials Toughness Safety Issue," Vols I and II, Revision 1, October 1982.

---, NUREG-0800, "Standard Review Plan," July 1981.

---, NUREG/CR-0660, "Enhancement of Onsite Emergency Generator Reliability," February 1979.

---, NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants," March 1984 (to be issued).

Table 1 Unresolved Safety Issues applicable to Beaver Valley Unit 2 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"	
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-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