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SUBJECT: Forwards response to Unresolved Safety Issues A-1,A-5,A-11, A-12,A-17,A-40,A-43,A-44,A-45,A-46,A-47,A-48 & A-49. Responses support position that plant can be operated safely until resolution of open items.

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TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

May 5, 1983

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Chief
Licensing Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Ms. Adensam:

In the Matter of the Application of	)	Docket No. 50-438
Tennessee Valley Authority	)	50-439

Your letter dated November 3, 1982 to H. G. Parris requested TVA to address 13 "Unresolved Safety Issues" pertaining to Bellefonte Nuclear Plant units 1 and 2. The enclosed information supports TVA's position that Bellefonte Nuclear Plant can be operated safely until final resolution of these generic issues.

If you have any questions concerning this matter, please get in touch with W. T. Watters at FTS 858-2691.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills

L. M. Mills, Manager
Nuclear Licensing

Sworn to and subscribed before me
this 5th day of May 1983

Paulette H. White

Notary Public
My Commission Expires 9-5-84

Enclosure

cc: U.S. Nuclear Regulatory Commission
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B001

ENCLOSURE
BELLEFONTE NUCLEAR PLANT UNITS 1 AND 2
UNRESOLVED SAFETY ISSUES

A-1 - Water Bammer

Problem Description

Since 1971 there have been over 200 incidents involving water hammers in BWRs and PWRs reported. The water hammers (or steam hammers) have involved steam generator feedrings and piping, the decay heat removal systems, emergency core cooling systems, and containment spray, service water, feedwater and steam lines. The incidents have been attributed to such causes as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Most of the damage reported has been relatively minor, involving pipe hangers and restraints; however, there have been several incidents which have resulted in piping and valve damage.

Bellefonte Response

The systems most frequently affected by water hammer effects are the feedwater systems. A potential auxiliary feedwater header problem was identified from information received from other utilities with B&W supplied once-through-steam-generators (OTSGs). Beader deformation in these plants was attributed to rapid steam condensation in the header following initiation of the cold auxiliary feedwater. Work has begun to replace the low-internal auxiliary feedwater header with a high-external header. No other problems have been identified to date regarding steam generator water hammer for Bellefonte. TVA has concluded that, subject to confirmation during the preoperational test program, the feedwater system and steam generator design for Bellefonte units 1 and 2 with respect to this potential steam generator water hammer concern is acceptable.

Protection against other potential water hammer events is provided by piping design codes which require consideration of impact loads. Approaches used at the design stage include: (1) increasing valve closure times, (2) piping layout to preclude water slugs in steam lines and vapor formation in water lines, (3) use of snubbers and pipe hangers, and (4) use of vents and drains. TVA will conduct a preoperational vibration dynamic effects test program in accordance with Section III of the ASME code for all ASME Class 1 and 2 piping systems and piping restraints during startup and initial operation. These tests will provide adequate assurance that the piping and piping restraints have been designed to withstand dynamic effects due to valve closures, pump trips and other operating modes associated with the design operational transients.

In the unlikely event that a large pipe break did result from a severe water hammer event, core cooling is assured by the emergency core cooling systems described in Section 6.3 of the FSAR and protection against the dynamic effects of such pipe breaks inside and outside of containment is provided as described in Section 3.6 of the FSAR.

Resolution of USI A-1 may identify some potentially significant water hammer scenarios that have not explicitly been accounted for in the design and operation of nuclear power plants, including the Bellefonte units. However, USI A-1 has not as yet identified the need for requiring any additional measures beyond those already required in the short term.

TVA has concluded that Bellefonte units 1 and 2 can be operated until there is an ultimate resolution of this generic issue without undue risk to the health and safety of the public.

Problem Description

Pressurized water reactor steam generator tube integrity can be degraded by corrosion induced wastage, cracking, reduction in tube diameter (denting) and vibration induced fatigue cracks. The primary concern is the capability of degraded tubes to maintain their integrity during normal operation and under accident conditions (LOCA or a main steam line break) with adequate safety margins.

Westinghouse and Combustion Engineering steam generator tubes have suffered degradation due to wastage and stress corrosion cracking. Both types of degradation have been decreased by conversion from phosphate to an all-volatile secondary water treatment. Degradation due to denting which leads to primary side stress corrosion cracking continues to be a problem.

B&W supplied once-through steam generators (OTSGs) were relatively free of trouble prior to the first tube leak incident at Oconee Unit 3 in July 1976. Since then, all three Oconee uuits have experienced tube leak incidents. The leaks at the Oconee units are the result of cracks of unknown origins propagated in the circumferential direction by flow induced vibration and have been limited to tubes located adjacent to the open tube inspection lane.

A second form of degradation characterized as an erosion-cavitation phenomena has been observed at Oconee and other B&W units.

Bellefonte Response

This issue deals with the capability of the steam generator tubes to maintain their integrity during normal operation and postulated accident conditions. Specific measures, such as steam generator design features and a secondary-water chemistry control and monitoring program, which TVA will use minimize the onset of steam generator tube problems are described in Chapter 10 of the FSAR. In addition, Section 5.4.2 discusses the inservice inspectiou requirements regarding steam generator tube integrity. TVA has and will continue to incorporate multifrequency inspection techniques in the implementation of steam generator inservice inspection requirements. The Technical Specifications will include requirements for actions to be taken in the event that steam generator tube leakage occurs during plant operation.

The following is a technical summary of the Bellefonte Nuclear Plant (BLNP) design information related to this issue.

> The Bellefonte OTSGs are constructed of cylindrical sections and hemispherical heads of manganese-molybdenum-nickel steel. The primary internal surfaces of the OTSG are clad with stainless steel or Ni-Cr-Fe. The overall length of the OTSG is 75 feet 5-1/4 inches from the top inlet nozzle to the bottom of the support skirt. The nominal inside diameter is 139 inches. The minimum wall thicknesses are 3-3/4 inches (thin shells) and 5-3/4 inches (thick shells). The OTSG is supported by a sliding support.

Upper Head

Located in the upper head of the OTSG are one 16-inch I.D. manway and one 6-inch I.D. inspection opening.

Vessel Shell Assembly

The outer shell of the OTSG is comprised of three sections: an upper shell, intermediate shell, and lower shell.

The upper shell assembly contains one 16-inch I.D. secondary manway.

The intermediate shell assembly contains one 1-1/2inch temperature sensing connection which houses the steam temperature protection system instrumentation (STPS).

Located in the lower shell assembly are seven 6-inch I.D. secondary inspection openings, and one 16-inch I.D. secondary manway.

The OTSG support skirt is welded to the lower tubesheet.

Shroud Assembly

The shroud assembly, consisting of the upper and lower shroud which accommodates the 16,013 tubes through which the primary coolant flows, forms the secondary feedwater annulus and steam outlet annulus. Fifty-six alignment pins are welded to the upper shroud and eight alignment pins are welded to the lower shroud to assure vessel and shroud alignment. Seventeen broached tube support plates are contained within the shroud assemblies to maintain tube positioning. Support rods connect the support plates and ensure stability.

Tubesheets

Two tubesheets restrict the primary coolant flow to the upper and lower heads and the individual tubes. The upper tubesheet is welded to upper head and upper shell, and contains one 1-1/2 inch vent and level sensing connection. The lower tubesheet is welded to the lower head and lower shell, and contains four 2-1/2-inch drain connections.

Lower Head

Located in the lower head of the OTSG are one 16-inch I.D. manway, one 6-inch I.D. inspection opening, and one 1-inch drain connection.

Tubes

Tubes are 55 feet and 9-3/8 inches long and are 0.625 inch O.D. by 0.034 inch wall thickness. The material is inconel, ASME SB-163.

Environmental Control

BLNP will operate on an all volatile secondary water treatment process. Dissolved and suspended impurities and corrosion products are removed from the system by deaeration and deep bed type mixed bed demineralizers which treat feedwater taken from the hotwell. Dissolved oxygen is removed by continuous injection of hydrazine to the feedwater. The secondary water chemistry program at BLNP will be consistent with the EPRI, Steam Generator Owners Group, PWR Secondary Water Chemistry Guidelines as Applicable to Once-Through Steam Generators. Primary system chemistry will also have provisions to minimize the potential of steam generator tube corrosion during plant operation and shutdown conditions. Additional detail is provided in Chapter 10 of the BLNP FSAR.

Preventative Measures

The feedwater chemistry program, choice of construction materials, and proposed plant operation were compared with those of Sequoyah and Watts Bar Nuclear Plants, and several potential changes and studies are being considered for Bellefonte. Condenser maintenance programs to preclude undetected condenser tube degradation will also be initiated to further protect steam generators from harmful corrodents.

USI A-5 is expected to result in improvements in the NRC's current requirements for inservice inspectin of steam generator tubes. These improvements will include a better statistical basis for inservice inspection program requirements and consideration of the cost/benefit of increased inspection. Pending completion of USI A-5, the measures taken at Bellefonte should minimize the steam generator tube problems encountered. Furthermore, the inservice inspection and Technical Specification requirements will ensure that TVA is 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. Because the improvements that will result from USI A-5 will be procedural (an improved inservice inspection program), they can be implemented after operation of Bellefonte begins, if necessary.

Based on the foregoing, TVA has concluded that Bellefonte can be operated before ultimate resolution of this generic issue without undue risk to the health and safety of the public.

A-11 - Reactor Vessel Materials Toughness

Problem Description

Because the possibility of failure of nuclear reactor pressure vessels designed to the ASME Boiler and Pressure Vessel Code is remote, the design of nuclear facilities does not provide protection against reactor vessel failure. However, as plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial safety margins.

Results from reactor vessel surveillance programs indicate that up to approximately 20 operating PWRs will have beltline materials with marginal toughness, relative to the requirements of Appendices G and H of 10 CFR Part 50, after comparatively short (approximately 10 EFPY) periods of operation. For most plants now in the licensing process, current criteria, together with the materials currently employed, are adequate to ensure suitable safety margins for reactor vessels throughout their design lives. However, a few plants under licensing review have reactor vesels that have been identified as having the potential for marginal fracture toughness within their design lives; these vessels will have to be reevaluated in the light of the new criteria for long term acceptability.

Bellefonte Response

Resistance to brittle fracture, a rapidly propagating catastrophic failure mode for a component containing flaws, is described quantitatively by a material property generally denoted as fracture toughness. Fracture toughness has different values and characteristics, depending upon the material being considered. For steels used in a nuclear reactor pressure vessel, three considerations are important; first, fracture toughness increases with increasing temperature; second, fracture toughness decreases with increasing load rates; and third, fracture toughness decreases with neutron irradiation. Prevention of reactor vessel failure depends primarily on maintaining the reactor vessel material fracture toughness at levels which will resist brittle fracture. Present reactor vessel materials offer adequate vessel fracture toughness. However, as plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and thus, the safety margins.

In recognition of these considerations, power reactors are operated within pressure restrictions imposed by the Technical Specifications in accordance with 10 CFR Part 50, Appendix G, during heatup and cooldown operations. These restrictions ensure that the reactor vessel will not be subjected to a combination of pressure and temperature that could cause brittle fracture of the vessel if there were significant flaws in the vessel materials. The effect of neutron radiation on the fracture toughness of the vessel material is accounted for in developing and revising these Technical Specification limitations.

The objective of USI A-11 is to provide an engineering method to assess the safety margin for failure prevention in nuclear reactor pressure vessels.

This method will use elastic-plastic fracture mechanics concepts in order to evaluate safety rather than the linear elastic fracture mechanics techniques which may only be applicable at low temperatures. The industry has responded to the resolution of this issue by providing assistance with data gathering, testing, and research funding to help develop the fracture mechanics concepts necessary for resolution of this issue.

The NRC has issued NUREG-0744, Revision 1 to provide guidance in performing the analyses required by 10 CFR Part 50, Appendix G, Section V.C. for reactor pressure vessels (RPVs) which fail to meet the toughness requirement during service life as a result of neutron radiation embrittlement. NUREG-0744 focuses on the RPV beltline materials because of the radiation-induced loss of upper shelf energy (USE) in that region and presents the NRC position and the proposed solution and verification tests of applicabilty. An acceptable elastic-plastic engineering method is presented. The need for this analytical method was dictated by the fact that some material (primarily weld metals) used in RPVs may have Charpy V-notch (CVN) impact test USE levels of less than 50 ft-lb before end of their design life.

When the RPV material exhibits a CVN USE level of less than 50 ft-lb, the requirements of 10 CFR Part 50 are not being met and a safety analysis must be performed to ensure continued safety operation of the reactor. Radiation-induced changes in both the transition temperature and the CVN USE increase with copper content. The most sensitive steels include weld metals with relatively high copper content (in the rauge of 0.2 to 0.5 percent by weight). Some variability in radiation-induced notch ductility changes in steels has been traced to residual copper and phophorous and high copper and nickel combinations in welds in the older steels.

Bellefonte Nuclear Plant units 1 and 2 RPV core beltline region material have a copper and phosphorus content which is at a level at which no unusual sensitivity to radiation damage is expected. Bellefonte is committed to meeting the requirements of Appendix G to 10 CFR Part 50, including the upper shelf energy level requirements, consequently no impact of USI A-11 is anticipated. It is possible, however, that the use of NUREG-0744 rules in evaluation of rector vessel adequacy may be desirable in the future, and its utility will be evaluated if the need arises.

TVA has concluded that the materials in Bellefonte reactor pressure vessel exhibit an adequate margin of safety and the plant may be safely operated without undue potential risk to the health and safety of the public.

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

Problem Description

During the course of the licensing action for North Anna Power Station Units 1 and 2, a number of questions were raised as to the potential for lamellar tearing and low fracture toughness of the steam generator and reactor coolant pump support materials for those facilities. Two different steel specifications (ASTM A36-70a and ASTM A572-70a) covered most of the material used for these supports. Toughness tests, not originally specified and not in the relevant ASTM specifications, were made on those heats for which excess material was available. The toughness of the A36 steel was found to be adequate, but the toughness of the A572 steel was relatively poor at an operating temperature of 80°F.

Since similar materials and designs have been used on other nuclear plants, the concerns regarding the supports for the North Anna facilities are applicable to other PWR plants. It was therefore necessary to reassess the fracture toughness of the steam generator and reactor coolant pump support materials for all operating PWR plants and those in CP and OL review.

Lamellar tearing may also be a problem in those support structures similar in design to North Anna. This possibility will be investigated on a generic basis.

Bellefonte Response

TVA is reviewing the materials used in all primary reactor coolant system component (reactor vessel, steam generators, reactor coolant pumps, and pressurizer) supports to determine the degree of compliance with the proposed criteria contained in NUREG-0577, 'Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports' (issued by the NRC in November 1979 for comment), as modified by the May 19 and 20, 1980, implementation letters to all power reactor licensees from D. G. Eisenhut. In addition, TVA is actively participating in the Atomic Industrial Forum (AIF) Subcommittee on Material Requirements and with the Metal's Property Council to develop an industry response to the generic issue. The subcommittee's work has resulted in the formulation of an industry position on fracture toughness of component support materials. This position is in draft form and is under review. The subcommittee's efforts will include operating temperature measurements on primary support members at a Babcock and Wilcox (B&W) NSSS plant. This information will be extrapolated to the Bellefonte supports and will be utilized in our response to USI A-12 as necessary.

In addition, the subcommittee is working to establish an industry position on the bolting concerns associated with proposed NUREG-0577, particularly relating to stress corrosion cracking potential. In this regard, TVA has been working with B&W in an effort to define the lowest permissible primary component support bolting preloads allowed by design and material properties.

In the interim, until the fracture toughness of the support materials at Bellefonte units 1 and 2 is evaluated and corrective measures implemented (if needed), operation is justified based on the following.

Support failures from inadequate fracture toughness are not expected to occur except under the unlikely combination of:

1. The initiating event (i.e., a large pipe break) which has a low probability of occurrence.

2. The existence of nonredundant and critical support structural member(s) with low fracture toughness (many supports contain redundant members).

3. The existence of support structural members at operating temperatures low enough that the fracture toughness of the support material is reduced to a level at which brittle failure could occur if a large flaw existed.

4. The existence of a flaw of such size that the stresses imparted during the initiating event could cause the flaw to rapidly propogate, resulting in brittle failure of the member(s).

Accordingly, TVA has concluded that Bellefonte can be operated before ultimate resolution of this generic issue without endangering the health and safety of the public.

Problem Description

The design of a nuclear power plant is accomplished by groups of engineers and scientists organized into engineering disciplines and into scientific disciplines. The reviews performed by the designers include interdisciplinary reviews to assure the functional compatibility of the plant structures, systems, and components. Safety reviews and accident analyses provide further assurance that system functional requirements will be met. These reviews include failure mode analyses.

The NRC review and evaluation of safety systems is accomplished in accordance with the Standard Review Plan (SRP) which assigns primary and secondary review responsibilities to organizational units arranged by plant systems or by disciplines. Each element of the SRP is assigned to an organizational unit for primary responsibility and, where appropriate, to other units for secondary responsibilities.

Thus, the design and analyses by the plant designers, and the subsequent review and evaluation by the NRC staff take into consideration the interdisciplinary areas of concern and account for systems interaction to a large extent. Furthermore, many of our regulatory criteria are aimed at controlling the risks from systems interactions. Examples include the single failure criterion and separation criteria.

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

The problem to be resolved by this program is to identify where the present design, analysis, and review procedures may not acceptably account for potentially adverse systems interaction and to recommend the regulatory action that should be taken to rectify deficiencies in the procedures.

Bellefonte Response

Review and evaluation of safety systems is accomplished in accordance with the Standard Review Plan (SRP) which includes several types of interdisciplinary reviews of safety-grade equipment and addresses several different types of potential system interactions. Two specific sections of the SRP (sections 3.6 and 7.4) extend the reviews to include the adverse effect of nonsafety equipment (i.e., high energy lines and associated electric circuits). The evaluation of system interactions occurring from high energy line breaks, jet impingement, local flooding, and pipe whip are summarized in section 3.6. Environmental qualification of equipment is covered in section 3.11. The interaction of reactor protection and control systems are addressed in sections 7.2 and 7.4. This includes the response to IE Bulletin 79-27 and IE Notice 79-22. The evaluation of interactions between fire protection systems and safety-grade systems are covered in section 9.5. Also, the quality assurance program which is followed during the design, construction, and operational phases each contribute to the prevention of introducing adverse systems interactions.

The current NRC position, pending the conclusion of the overall plan, is that the existing regulatory requirements and procedures provide an adequate degree of public health and safety assurance. Therefore, TVA concludes that there is reasonable assurance the Bellefonte units 1 and 2 can be operated before the ultimate resolution of this generic issue without endangering the health and safety of the public.

A-40 - Seismic Design Criteria

Problem Description

The seismic design process required by current NRC criteria includes the following sequence of events:

(a) Define the magnitude or intensity of the earthquake which will produce the maximum vibratory ground motion at the site (the safe shutdown earthquake or SSE).

(b) Determine the free-field ground motion at the site that would result if the SSE occurred.

(c) Determine the motion of site structures by modifying the free-field motion to account for the interaction of the site structures with the underlying foundation soil.

(d) Determine the motion of the plant equipment supported by the site structures.

(e) Compare the seismic loads, in appropriate combination with other loads, on structures, systems, and components important to safety, with the allowable loads.

While this seismic design sequence includes many conservative factors, certain aspects of the sequence may not be conservative for all plant sites. At present, it is believed that the overall sequence is adequately conservative. The objective of this program is to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternate approaches to parts of the design sequence, to quantify the overall conservatism of the design sequence, and to modify the NRC criteria in the Standard Review Plan if changes are found to be justified. In this manner, this program will provide additional assurance that the health and safety of the public is protected, and if possible, reduce costly design conservatisms by improving (1) current seismic design requirements, (2) NRC capability to quantitatively assess the overall adequacy of seismic design for nuclear plants in general.

Bellefonte Response

USI A-40 was initiated to identify and quantify the conservatism inherent in a seismic design sequence that is consistent with current NRC criteria. The two major phases of the program are Phase 1, which deals with seismic response of structures, systems and components; and Phase 2, which is concerned with the definition of seismic input to analytical models. Phase 1 of the program is nearing completion. All contractor work has been completed and documented, and a final NUREG will be issued on this phase. Phase 2 of the program has been deferred pending the availability of NRC personnel.

The goal of this program is to revise seismic design-related sections of the Standard Review Plan (sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3) and the related Regulatory Guides (1.60, 1.92, and 1.122).

The only known NRC concern is the evaluation for the eighty-fourth percentile earthquake. TVA has generated data that are expected to resolve this concern. However, future requirements may be imposed following the release of the planned NUREG on the subject. No interim measures are to be taken and no alternate course of action will be developed until the sections of the Standard Review Plan related to seismic design are revised.

Therefore, TVA concludes that there is reasonable assurance that the Bellefonte units 1 and 2 can be operted before the ultimate resolution of this generic issue without endangering the health and safty of the public.

A-43 - Containment Emergency Sump Reliability

Problem Description

Following a Loss of Coolant Accident (LOCA) in a PWR, water flowing from the break in the primary system would collect on the floor of containment. During the injection mode, water for core cooling and containment spray is drawn from a large supply tank. When the water reaches a low level in the tank, pumps are realigned to draw from the containment. This is called the recirculation mode wherein water is drawn from the containment snmp and pumped to the primary system or containment spray headers. This program addresses the safety issue of adequate sump or suppression pool function in the recirculation mode. It is the objective of A-43 to develop improved criteria for design, testing, and evaluation which will provide better assurance that emergency sumps will function to satisfy system requirements.

The principal concerns are somewhat interrelated but are best discussed separately. One deals with the various kinds of insulation used on piping and components inside of containment. The concern being that break-initiated debris from the insulation could cause blockage of the sump or otherwise adversely affect the operation of the pnmps, spray nozzles, and valves of the safety systems.

The second concern deals with the hydraulic performance of the sump as related to the hydraulic performance to safety systems supplied therefrom. Preoperational tests have been performed on a number of plants to demonstrate operability in the recirculation mode. Adverse flow conditions have been encountered requiring design and procedural modifications to eliminate them. These conditions, air entrainment, cavitation, and vortex formation, are aggravated by blockage. If not avoided or suppressed, these conditions could result in pump failure during the long-term cooling phase following a LOCA.

Bellefonte Response

The Bellefonte containment sump design conforms to RG 1.82 except for minor deviations as discussed in FSAR Section 6.2.2.2.2.8. These deviations are locating the fine mesh screen within the sump and allowing a higher fluid velocity at the screen than specified by RG 1.82. RG 1.79 addresses, in part, the testing of the sump hydraulic performanee. The sump vortex and pressure drop regulatory concerns are addressed for Bellefonte by model testing rather than the in-plant testing presented in RG 1.79. However, based on the following discussion, TVA believes there is reasonable assurance that the sump design will perform adequately during a LOCA and that the deviations discussed above will not degrade the sump performance.

(1) The Bellefonte sump design provides more screening capability than required by RG 1.82 to prevent debris from entering the containment sump. To further protect the emergency sumps and their associated screens and emergency recirculation lines, all piping insulation used

within containment and containment penetrations is metallic reflective insulation constructed of austenitic stainlesss steel. This insulation is sectionalized, mechanically strong and rigid. Each section is held together with a combination of buckles and bands. The bands are held together by other buckles and fit into grooves at either end of each insulation section. These design features limit the quantity of insulation which will be affected by a pipe break. In addition, the weight of the reflective insulation in conjunction with the low flow rates within the containment areas because of the large and numerous passages to the sumps will tend to cause any reflective insulation to sink and remain in the vicinity of the original location even it is dislodged during a pipe break event. All other possible debris generating materials (such as paint, wiring insulation, etc.) within contaiment have been carefully selected and limited to restrict the amount of debris which can be liberated during accidents. Surface coatings are as discussed in FSAR Section 6.1.2.

(2) Under contract to TVA, Alden Research Laboratory has completed sump model tests for Bellefonte. The snmp model was constructed at a scale of 1:2.26 and was tested for a wide variety of possible flow distribution and screen blockage schemes.

NUREG/CR-2792, 'An Assessment of Residual Heat Removal and Containment Spray Pump Performance Under Air and Debris Ingesting Conditions,' states that pump performance degradation is negligible if air ingestion quantities are less than 2 percent by volume. During the Alden tests for Bellefonte, no surface vortex activity was observed. No air ingestion will occur without surface vortex activity.

In the Alden tests, a maximnm inlet loss of 0.65 feet was established for the worst case of 50 percent screen blockage. The measured loss includes the screen and sump inlet losses and the bellmouth inlet loss. The NPSH calculations performed for Hellefonte assumed an inlet loss of 2.0 feet. The test results show that this is a conservative assumption.

The Alden test report has been submitted to the NRC by letter from D. S. Kammer to E. Adensam on August 10, 1982. TVA does not plan to conduct any in-plant tests to further demonstrate the adequacy of the Bellefonte emergency sump design.

Based on the above, TVA has concluded that there is reasonable assurance that Bellefonte units 1 and 2 can be operated before the ultimate resolution of this generic issue without endangering the health and safety of the public.

A-44 - Station Blackout

Problem Description

Electric power for safety systems at nuclear power plants is supplied by two redundant and independent divisions. Each of these electrical divisions includes an offsite alternating current (A.C.) source, an onsite A.C. source (usually diesel-generators), and a direct current (D.C.) source. Appendix A to 10 CFR 50 defines a total loss of offsite power as an anticipated occurrence, and as such it is required that an independent emergency onsite power supply be provided at nuclear power plants.

The unlikely, but posible loss of A.C. power (that is, the loss of A.C. power from the offsite source and from the onsite source) is referred to as a station blackout. In the event of a station blackout, tbe capability to cool the reactor core would be dependent on the availability of systems which do not require A.C. power supplies, and on the ability to restore A.C. power in a timely manner. The concern is that the occurrence of a station blackout may be a relatively high probability event and that the consequences of this event may be unacceptable, for example, severe core damage may result.

Bellefonte Response

A total loss of all ac power is not a design basis event for Bellefonte units 1 and 2. Nonetheless, the combination of design, operation, and testing requirements that have been imposed on the plant will assure that these units will have substantial resistance to this event and that even if a total loss of all ac power should occur, there is reasonable assurance that the core will be cooled.

A loss of offsite ac power involves a loss of both the preferred and backup sources of offsite power. TVA'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 the FSAR. Also, a preliminary grid stability analysis was performed, portions of which may be found in the Hellefonte FSAR, Section 8.2.

If offsite ac power is lost, two independent and redundant onsite diesel generators per unit and their associated distribution systems will deliver emergency power to safety-related equipment. The design, testing, surveillance, and maintenance provisions for the onsite emergency diesels are described in Section 8.3.1 of the FSAR. TVA's requirements include testing to assure the reliability of the installed diesel generators in accordance with the provisions of RG 1.108.

Even if both offsite and onsite ac power are lost, cooling water from the condensate storage tank can still be provided to the steam generators by the auxiliary feedwater system by employing a steam turbine driven pump that does not rely on ac power for operation. The description of the auxiliary feedwater system design and operation is described in Section 10.4.9 of the Bellefonte FSAR.

In addition, as part of the Sequoyah unit 1 low power test program, TVA performed a 'blackout' test which simulated loss of all offsite and onsite ac power. The test results showed that during the blackout period, the turbine-driven auxiliary feedwater pumps delivered flow to the steam generators as required. Therefore, while the designs for the ac power and auxiliary feedwater systems at Sequoyah and Bellefonte are not identical, it is TVA's opinion that sufficient similiarities cxist in the designs such that the test results (from a system operation standpoint) could be considered applicable. Additional information on this test and other concerns (operator training, procedures, etc.) related to this issue is provided in Supplement No. 2 to the Sequoyah Safety Evaluation Report and TVA's response (L. M. Mills' June 5, 1981, letter to E. Adensam) to the NRC's generic letter No. 81-04, 'Emergency Procedures and Training for Station Hlackout Events.'

Based on the above, TVA has concluded that there is reasonable assurance that Bellefonte units 1 and 2 can be operated before the ultimate resolution of this generic issue without endangering the health and safety of the public.

A-45 - Shutdown Decay Heat Removal Requirements

Problem Description

Although many improvements to the steam generator auxiliary feedwater system were required of the reactor manufacturers by the NRC following the TMI-2 accident, the staff feels 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, USI A-45 will investigate alternative means of decay heat removal in PWR plants, including but not limited to using existing equipment where possible. This Unresolved Safety Issue will also investigate the need and possible design requirements for improving reliability of decay heat removal systems in boiling water reactors (BWRs).

The overall purpose of USI A-45 is to evaluate the adequacy of current licensing design requirements, in order to ensure that nuclear power plants do not pose an unacceptable risk due to failure to remove shut-down decay heat. The objective will be to develop 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.

Bellefonte Response

For Bellefonte Nuclear Plant (BLNP), design basis decay heat removal (DHR) is as detailed in Sections 5.4.7 (normal operation) and 6.3 (emergency operation) of the BLNP FSAR.

TVA is following developments in several generic efforts which could or do have output valuable in the resolution of USI A-45, including:

1. USI A-31, 'Residual Heat Removal Requirements'
2. USI A-44, 'Station Blackout'
3. NRC's 'Severe Accident Sequence Assessment'
4. NRC's Degraded Core Cooling Research
5. NSAC-52 - A generic report to be issued by the Nuclear Safety Analysis Center concerned with alternate decay heat removal methods. TVA will receive this report as part of EPRI participation.
6. NUREG/CR-1556, 'Study of Alternate Decay Heat Removal Concepts for Light Water Reactors - Current Systems and Proposed Options'
7. NUREG/CR-2799, 'Evaluation of Events Involving Decay Heat Removal Systems in Nuclear Power Plants'

Any output of these efforts applicable to BLNP will be considered in the resolution of USI A-45.

In addition to the applicable generic studies, TVA is involved in several plant specific efforts which will weigh heavily in the ultimate resolution of USI A-45 for BLNP. These efforts include:

1. Reliability Study - Consistent with the NRC plan to establish a minimum DHRS reliability as a basis for requiring DHRS upgrades, TVA is performing a plant specific reliability study for Bellefonte Nuclear Plant. Decay Heat Removal System reliability and contribution to the overall probability of core melt will be determined and serve as a valuable tool in evaluating DHRS acceptability.

2. Vendor Inadequate Core Cooling Study - Through the B&W Owners Group, TVA is funding and reviewing B&W development of instrumentation and procedures to detect and recover from Inadequate Core Cooling. This will comply with NUREG-0737 item II.F.2 and lead to a better understanding of decay heat removal requirements.

3. Emergency Procedure Guidelines - B&W is preparing Emergency Procedure Guidelines (EPG) for BLN and will comply with NUREG-0737 item I.C.1. These EPGs deal with multiple failures of design basis DHR methods and look at alternate DHR means like 'feed and bleed'.

4. Appendix R Fire Protection - TVA is evaluating compliance of BLNP with the Appendix R Fire Protection Rule. Should this evaluation determine present DHR equipment is inadeqnate, TVA will consider hardware modification or installation of a dedicated DHRS to meet applicable criteria.

5. Alternate Decay Heat Removal Methods - B&W's small break LOCA program will include an analysis of the feasibility and effectiveness of 'feed and bleed' techniques for DHR for Bellefonte. TVA funds this effort through participation in the B&W Owners Group and will demonstrate by analysis the viability of establishing 'feed and bleed' as an acceptable alternate decay heat removal method.

TVA regards USI A-45 as a highly significant effort with the potential to affect both operating and nonoperating plants. TVA believes it is important to provide:

1. an alternate DHR method, and

2. supporting realistic analyses so that the designers and operators will understand the capabilities of any alternate DHR methods used in the EPGs.

TVA will continue to work closely with NSSS vendors on related tasks and will follow industry efforts in the resolution of A-45. TVA is awaiting results of the NRC's evaluation of the adequacy of existing DHR Systems and potential need for alternate DHR methods. TVA will follow this and initiate necessary efforts to comply with any forthcoming requirements.

Based on the foregoing, TVA believes that Bellefonte can be operated before the ultimate resolution of this generic issue without undue risk to the health and safety of the public.

A-46 - Seismic Qualification of Equipment in Operating Plants

Problem Description

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the course of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and perform the intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this Unresolved Safety Issue is to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria for new plants. This guidance will concern equipment required to safely shutdown the plant, as well as equipment whose function is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions. Also, explicit guidelines will be established for use in requalifying equipment whose seismic qualification was found to be inadequate.

Bellefonte Response

USI A-46 is concerned with the seismic qualification of equipment in older operating plants. This includes equipment which was not specifically qualified to the latest accepted industry procedures, such as IEEE 344-1975 and Regulatory Guide 1.100 and which has not been subjected to a seismic qualification audit.

This item does not specifically relate to the Bellefonte safety-related equipment. Bellefonte is not an older operating plant; the safety-related equipment has been seismically qualified consistent with the principles of the current standards.

Therefore, TVA concludes that Bellefonte can be operated before resolution of this generic issue without undue risk to the health and safety of the public.

Problem Description

This issue coucerns the potential for accidents or transients 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 and would be in addition to any control system failure that may have initiated the event. Although it is generally believed that control system failures are not likely to result in loss of safety functions which would lead to serious events or result in conditions that safety systems are not able to cope with, in-depth studies have not been performed to support this belief. The potential for an accident that would affect a particular control system and the effects of the control system failures will differ from plant to plant. Therefore, it is not likely that it will be possible to develop generic answers to these concerns, but rather plant-specific reviews will be required. The purpose of this Unresolved Safety Issue is to define generic criteria that may be used for plant-specific reviews. A specific subtask of this issue will be to study the steam generator overfill transient in PWRs and the reactor overfill transient in BWRs to determine and define the need for preventive and/or mitigating design measures to accommodate this transient.

Bellefonte Response

This issue concerns the potential for accidents or transients being made more severe as a result of nonsafety grade control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration. The independent control system failure or 'single failure' is of some concern due to the potential for a failure such as 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 analyzed as anticipated operational occurrences. Another concern is for a postulated accident to cause a control system failure which could make the accident more severe than analyzed. This latter failure could also result from a single failure-induced accident. These 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.

The Bellefonte Nuclear Plant control and safety systems have been designed with the goal of ensuring that nonsafety grade control system failures will not prevent automatic or manual initiation and operation of any safety system equipment required for accident mitigation and/or to maintain the plant in a safe shutdown condition following any anticipated operation occurrence or accident. This has been accomplished by providing independence and physical separation between safety system trains and between safety and nonsafety systems. For the latter, as a minimum, isolation devices were provided.

These devices preclude the propagation of nonsafety system equipment faults to the protection systems. Also, to ensure that the operation of safety system equipment is not impaired, the single failure criterion has been applied in the plant design. Design reviews have been performed (IE Bulletin 79-27) to ensure that the loading on certain Class IE power boards maintains the separation and independence of plant systems as designed.

A systematic evaluation of the nonsafety grade control system design, as contemplated for this Unresolved Safety Issue, has not been performed to determine whether postulated accidents could cause significant control system failures which would make the accident consequences more severe than presently analyzed. However, a wide range of bounding transients and accidents is presently analyzed to assure that the postulated events such as steam generator overfill and overcooling events would be adquately mitigated by the safety systems. In addition, systematic reviews of safety systems are being performed with the goal of ensuring that control system failures (single or multiple) will not defeat safety system action. These reviews are part of an ongoing evaluation program to qualify Class IE plant equipment to function for all postulated service conditions to which it is subjected (NUREG-0588).

Based on the above, TVA has concluded that there is reasonable assurance that the Bellefonte Nuclear Plant may be operated until the ultimate resolution of this generic issue without endangering the health and safety of the public.

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

Problem Description

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

This USI will investigate means to predict the quantity and release rate of hydrogen following degraded core accidents and various means to cope with large releases to the containment such as inerting of the containment or controlled burning. The potential effects of proposed hydrogen control measures on safety including the effects of hydrogen burns on safety-related equipment will be investigated.

Bellefonte Response

On December 23, 1981, the NRC issued for comment a proposed rule on hydrogen control. It would require, for all non-inerted LWRs such as Bellefonte Nuclear Plant (BLNP), that analyses of the consequences of hydrogen combustion on structural integrity and on systems necessary for shutdown be performed within two years after the effective date of the rule or the date of issuance of an operating license, whichever is later. TVA has provided written comments to NRC on this proposed rule and is awaiting issuance before beginning any analyses for BLNP. Based on the experience TVA has gained from performing similar analyses for Sequoyah, no hydrogen mitigation system is foreseen to be necessary for BLNP.

Accordingly, pending resolution of this issue and the rulemaking proceeding on hydrogen control, TVA believes that Bellefonte can be operated without undue risk to the health and safety of the public.

A-49 - Pressurized Thermal Shock

Problem Description

Neutron irradiation of reactor pressure vessel weld and plate materials decreases the fracture toughness of the materials. The fracture toughness sensitivity to radiation induced change is increased by the presence of certain materials such as copper. Decreased fracture toughness makes it more likely that, if a severe overcooling event occurs followed by or concurrent with high vessel pressure, and if a small crack is present on the vessels inner surface, that crack could grow to a size that might threaten vessel integrity.

Severe pressurized overcooling events are improbable since they require multiple failures and improper operator performance. However, certain precursor events have happened that could have potentially threatened vessel integrity if additional failures had occurred and/or if the vessel had been more highly irradiated. Therefore, the possibility of vessel failure because of a severe pressurized overcooling event cannot be ruled out.

Bellefonte Response

USI A-49 addresses the situation in which a severe system overcooling event is followed by repressurization of the reactor vessel. The mechanical response of the reactor vessel to the propagation of crack-like defects in the wall depends on the reduction of fracture toughness because of neutron flux during operation of the plant. As long as the fracture resistance of the reactor vessel material remains relatively high, an overcooling event will not cause failure. After the fracture toughness of the vessel is reduced by neutron irradiation, thermal transients could cause fairly small flaws to propagate near the inner surface of the vessel.

The vessels of concern are those which have accumulated high neutron radiation exposure and which are made of material that has a high sensitivity to neutron irradiation such as those with welds in the reactor vessel belt line region with high copper content. The base material chemistry, properties, and flaw content should not be ignored.

The reactor system overcooling events that can lead to pressurized thermal shock (PTS) result from a variety of causes including instrumentation and control system malfunctions and postulated accidents such as small break loss-of-coolant accidents (SBLOCAs), main steam line breaks (MSLBs), feedwater pipe breaks, or stuck open valves in either the primary or secondary system. Rapid cooling of the reactor vessel internal surface causes a temperature gradient across the reactor vessel wall. The temperature gradient results in thermal stresses, with a maximum tensile stress at the inside surface of the vessel and a compressive stress at the outside surface. These stresses combine with the hoop stress caused by the internal pressure in the vessel. The magnitude of the thermal stress depends on the temperature differences across the reactor vessel wall.

In order to threaten the adequacy of core cooling by PTS events, a number of contributing factors must be present. These factors are: (1) a reactor vessel flaw of correct size to propagate, (2) high copper content, (3) a relatively high level of irradiation over a period of operating time, (4) a severe overcooling transient with repressurization, and (5) a resulting crack of such size and location that the ability of the reactor vessel to maintain core cooling is affected.

NRC has proposed to establish screening criteria for vessel reference temperature RT_{NDT} , a parameter that characterizes the state of embrittlement of reactor vessels. NRC proposes that, whenever the value of RT_{NDT} for a given vessel is projected to exceed the screening criteria within the next three calendar years, the licensee would be required to submit a plant specific analysis, the scope of which has yet to be specified. NRC also proposes that a number of long-term actions be required to ameliorate the PTS problem.

The NRC and the B&W Owner's Group plan meetings in the first part of 1983 to determine the screening criteria for B&W vessels such as Bellefonte.

Radiation-iuduced changes in both the trausitiou temperature and the Charpy V-notch upper shelf energy (USE) increase with increasing copper content. The most sensitive steels include weld metals with relatively high copper content (in the range of 0.2 to 0.5 percent by weight). Some variability in radiation-induced notch ductility changes in steels has been traced to residual copper and phosphorous and high copper and nickel combinations in welds in the older steels.

Hellefonte Nuclear Plant units 1 and 2 RPV core beltline region materials have a copper and phosphorous content which is at a level at which no unusual sensitivity to radiation damage is expected.

No unresolved NRC requirements exist for Bellefonte and consequently no TVA action is required or planned. TVA will review future NRC generic recommendations on this subject resulting form the NRC and B&W Owner's Group discussions, and the commissioners' concurrence with NRC staff's recommendations to be issued during 1983. TVA will implement appropriate actions to assure reactor vessel integrity throughout plant life.

TVA has concluded that the design of Bellefonte nuclear plants is adequate and until there is a resolution of this generic issue construction of unit 1 and 2 may continue without modification and operated on completion without undue potential risk to the health and safety of the public.