QUESTION 112.1

The information presented in Subsection 3.6.2.1.1 concerning pipe break criteria for piping between containment isolation valves is not completely acceptable. To justify a break exclusion region in piping systems penetrating primary containment, the criteria presented in Branch Technical Position APCSB 3-1, Paragraph B.2d should be specified in addition to the information presented. It is the staff's position that one hundred percent volumetric examination of all process piping welds in this region be performed during each inspection interval.

RESPONSE:

This information is contained in revised Subsections 3.6.2.1.1 and 5.2.4.7.

QUESTION 112.2

Provide sketches showing the locations of the resulting postulated pipe ruptures, including identification of longitudinal, and circumferential breaks, structural barriers, if any, restraint locations, and the constrained direction in each restraint.

Also provide a summary of the data developed to select postulated break locations including, for each point, the calculated stress intensity, the calculated cumulative usage factor, and the calculated primary plus secondary stress range as delineated in SRP Section 3.6.2, Paragraph III.I.b.

RESPONSE:

This information is contained in revised Subsection 3.6.2.1.1 and Figure 3.6-17.

QUESTION 112.3

Provide a summary comparison of the results obtained from the use of each program in Subsection 3.9.1.2 of the FSAR with the results derived from a similar recognized and approved program or results from test problems.

RESPONSE:

Subsection 3.9.1.2 has been revised and Appendix 3.9A has been added to provide the requested information.

QUESTION 112.4

The information presented in Subsections 3.9.2.2b and 3.10 of the FSAR concerning seismic qualification of Category I mechanical and electrical equipment may not be completely acceptable. Criteria which is acceptable to the staff and is currently being implemented on all plants is outlined in the Regulatory Guide 1.100. Provide a comparison of your program with the criteria of the above mentioned Regulatory Guide.

RESPONSE:

The implementation paragraph of this regulatory guide states that the requirements of the position statements will only be applied to plants that received construction permits after November 16, 1976. The Construction Permit for Susquehanna SES was issued in November 1973 and therefore the guidelines of this regulatory guide have not been utilized in the design of this nuclear power station.

Seismic qualification of the NSSS safety related electric equipment has been conducted in accordance with the IEEE Standard 344-1971. Section 3.10a describes the complete qualification methods and procedures that have been utilized.

For an explanation of seismic qualification criteria for non-NSSS equipment, see the following subsections:

a) Mechanical Components - revised Subsection 3.9.2.2b and new Table 3.9-18;

b) Instrumentation - revised Section 3.10b;

c) Electrical Components - revised Subsection 3.10c.2.2 and revised Table 3.10c-1.

QUESTION 112.5

Explain in detail how the loads discussed in Subsections 3.9.2.2a, 3.9.2.2b.2, 3.9.3.1 and Table 3.9-2 of the FSAR are combined for various plant conditions. Table 3.9-2(d) of the FSAR shows that for Emergency and Faulted conditions, the peak loads are combined using the square root of the sum of the squares method. It is the staff's position that these peak loads be combined by absolute sum unless acceptable justification is provided for an alternative method of combination. Provide such justification, or alternatively evaluate the effect of combining responses to dynamic loads by absolute sum on components and supports.

RESPONSE:

For the method of Load Combination for emergency and faulted plant conditions for non-NSSS ASME III Class 1, 2, and 3 components, see Table 3.9-6 and revised Subsections 3.9.3.1.19 and 3.9.2.2b.2.

The combination of loads discussed in Subsections 3.9.2.2a (NSSS components) and 3.9.3 is detailed in Table 3.9-2. Table 3.9-2 is the major part of this section and presents the loading combinations, analytical methods (by reference or example) and also the calculated stress or other design values for the most critical areas in the design of each component. These calculated values are also compared to the applicable code allowables. The combining of two or more peak dynamic loads by the SRSS method in Table 3.9-2(d) is a generic issue and has been recently addressed and resolved by GE and Mark II owners group (see NUREG 0484). The generic resolution applies to Susquehanna SES.

Question 112.6

Recent analyses have shown that certain reactor system components and their supports may be subjected to previously underestimated asymmetric loads under the conditions that result from the postulation of ruptures of the reactor coolant piping at various locations. It is therefore necessary to reassess the capability of these reactor system components to assure that the calculated dynamic asymmetric loads resulting from these postulated pipe ruptures will be within the bounds necessary to provide high assurance that the reactor can be brought safely to a cold shutdown condition. For the purpose of this request for additional information, the reactor system components that require reassessment shall include:

(1) Reactor Pressure Vessel
(2) Fuel Assemblies, including Grid Structures
(3) Control Rod Drives
(4) ECCS Piping that is attached to the Primary Coolant Piping
(5) Primary Coolant Piping
(6) Reactor Vessel and Pump Supports
(7) Reactor Internals
(8) Biological Shield Wall and Neutron Shield Tank (where applicable)
(9) Pump Compartment Wall

The following information should be included in the FSAR about the effects of postulated asymmetric LOCA loads on the above-mentioned reactor system components and the reactor cavity structure.

(1) Provide arrangement drawings of the reactor vessel, and pump support systems to show the geometry of all principal elements and materials of construction.

(2) If a plant-specific analysis will not be submitted for your plant, provide supporting information to demonstrate that the generic plant analysis under consideration adequately bounds the postulated accidents at your facility. Include a comparison of the geometric, structural, mechanical and thermal hydraulic similarities between your facility and the case analyzed. Discuss the effects of any differences.

(3) Consider all postulated breaks in the reactor coolant piping system, including the following locations:

(a) Steam line nozzles to piping thermal ends.
(b) Feedwater nozzle to piping terminal ends.

(c) Recirculation inlet and outlet nozzles to recirculation piping terminal ends.

Provide an assessment of the effects of asymmetric pressure differentials[1] on these systems/components in combinations with all external loadings including safe shutdown earthquake loads, asymmetric cavity pressurization for both the reactor vessel, and recirculation pump which might result from the required postulate. This assessment may utilize the following mechanistic effects as applicable:

(a) limited displacement break areas
(b) fluid-structure interaction
(c) actual time-dependent forcing function
(d) reactor support stiffness
(e) break opening times.

(4) If the results of the assessment required by Item 3 above indicate loads leading to inelastic action in these systems or displacement exceeding previous design limits provide an evaluation of the inelastic behavior (including strain hardening) of the material used in the system design and the effect of the load transmitted to the backup structures to which these systems are attached.

(5) For all analysis performed, include the method of analysis, the structural and hydraulic computer codes employed, drawings of the models employed and comparisons of the calculated to allowable stresses and strains or deflections with a basis for the allowable values.

(6) Provide an estimate of the total amount of permanent deformation sustained by the fuel spacer grids. Include a description of the impact testing that was performed in support of your estimate. Address the effects of operating temperatures, secondary impacts, and irradiated material properties (strength and ductility) on the amount of predicted deformation. Demonstrate that the fuel will remain coolable for all predicted geometries.

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[1] Blowdown jet forces at the location of the rupture (reaction forces), differential pressures in the annular region between the vessel and the shield, and transient differential pressures across the core barrel within the reactor vessel.

(7) Demonstrate that active components will perform their safety function when subjected to the postulated loads resulting from a pipe break in the reactor coolant system.

(8) Demonstrate functionability of any essential piping where service level B limits are exceeded.

RESPONSE:

In accordance with NRC letter from Mr. Olan D. Parr (NRC) to Mr. N. W. Curtis (PP&L) dated February 8, 1979, Question 110.32 replaces Question 112.6, therefore it is no longer necessary to respond to Question 112.6.

QUESTION 112.7

Most of the operating BWR plants have reported finding radial cracks on the reactor vessel feedwater nozzle and the CRD return line. Describe what design modifications will be made to eliminate this problem. In addition, provide a description of the analyses that will be performed to demonstrate the adequacy of the reactor vessel feedwater nozzle and CRD return line to withstand the imposed service condition without the cracking experienced in the operating plants.

RESPONSE:

The mechanisms which have caused cracking in operating BWRs are understood. A summary discussion of the previously observed problems and the solutions incorporated in the Susquehanna design is presented in the following.

A detailed evaluation of the problems of the feedwater nozzle and sparger is presented in NEDE-21821 "BWR Feedwater Nozzle/Sparger Final Report" March 1978. The solution of the feedwater nozzle and sparger cracking problems involves several elements, including material selection and processing, nozzle clad removal, and thermal sleeve and sparger redesign. The following summarizes the problems that have occurred in the nozzle and sparger and shows the solution that eliminates each problem:

Problem	Cause	Fix
Sparger arm cracks	Mechanical fatigue	Eliminate/minimize clearance between thermal sleeve and safe end.
	Thermal fatigue	Eliminate low flow stratification by use of top-mounted elbows.
Flow hole cracks	Thermal fatigue	Eliminates separation by use of converging nozzles.
Nozzle cracks	Thermal	Eliminate clad, control leakage, protect nozzle with multiple sleeves.

The sparger vibration has been attributed to a self-excitation caused by instability of leakage flow through the annular clearance between the thermal sleeve and safe end. Tests have shown that the vibration is eliminated if the clearance is reduced sufficiently or sealed. The solution which has been selected uses a two-stage piston ring seal mounted in the thermal sleeve in conjunction with an interference fit between the sleeve and safe end. This feature is also an essential part of the solution of the nozzle cracking problem, and is described later in more detail. Freedom from vibration over a range of conditions has been demonstrated by the tests reported in NEDE-21821, Section 4.

Sparger arm cracking has also been caused by thermal fatigue, both at the flow holes and adjacent to the tee connection with the thermal sleeve. In both cases, excessive cyclic thermal stresses are caused by the exposure of material in a constrained structure to an unstable boundary between cold feedwater fluid and hot reactor fluid. At low feedwater flow, the presence of exit flow holes at the midplane of the sparger allowed the sparger to be only partially filled with cold fluid. This caused a temperature gradient from the top of the sparger to the bottom, with associated bending stresses which changed directly with changes in the flow gradient. Relocation of the exit flow holes at the top of the sparger allows complete filling of the sparger with the feedwater fluid even at low flow, producing a more stable and homogeneous temperature distribution. As shown by the data reported in NEDE-21821, Section 4.3, stratification has been eliminated over the range of operating flows.

Flow hole cracks occurred partly because the surface of the hole was constrained by the in-plane stiffness of the surrounding sparger material when exposed to the exit flow to reactor coolant gradients, and partly because the gradients themselves were unstable. The instability of the gradients resulted from changing location of the operation point between the cold exit flow and the warmer boundary layer produced by heating of the sparger by reactor fluid.

The result was a high-cycle thermal stress around the edge of the hole. This condition is eliminated by the exit flow elbows which have a long enough exit throat to stabilize the flow separation.

Also, the thermal stress produced by a given gradient is much less with the exit hole in a cylindrical tube, rather than in what previously would behave locally more as a flat plate. Testing, as reported in NEDE-21821, Section 4.3, has shown that the high frequency cycling is eliminated by the new design.

In order to allow for removal of the sparger, it is necessary to provide a sealed joint between the nozzle safe end and the thermal sleeve. This seal is achieved by use of a metal piston ring backed up with a coil spring expander. Even if the piston ring seal was leaktight when initially installed, its long-term sealing ability is unknown. The effects of wear and corrosion on the mating safe end surface would eventually cause leakage to increase to the point where nozzle cracking would initiate. The rate of deterioration is unpredictable but is expected to be short relative to the life of the pressure vessel. To provide protection against seal failure resulting in nozzle cracking, the second piston ring and the added thermal sleeves have been incorporated in the new design. It has been demonstrated by test that the triple thermal sleeve arrangement prevents the leakage flow causing nozzle cracking. This is the result of the concentric sleeve arrangement channeling leakage away from the nozzle end and the fact that the second seal is exposed to very low driving pressures, making leakage past it very small.

As was mentioned earlier, the cracking of the feedwater nozzles is a two-part process. The crack initiation mechanism as discussed above is the result of self-initiated thermal cycling. If this were the only mechanism present, the cracks would initiate, grow to a depth of approximately 0.25 inch, and arrest. This degree of cracking could be tolerated, but unfortunately there is another mechanism which supports crack growth. This mechanism is the system induced transients, primarily the startup/shutdown transients. The triple thermal sleeve arrangement also assists in this area because, even with the piston rings leaking, the heat transfer coefficient between the feedwater and the nozzle are reduced to the point where the thermal stresses in the nozzle are not high enough to cause significant crack growth. Analysis presented in NEDE-21821, Section 4.6, demonstrates this benefit and the benefit of using unclad nozzles.

The cracking of the CRD return nozzles is caused by a mechanism which is very similar to that which caused cracking in the feedwater nozzles - thermally induced fatigue.

The CRD return flow is always at a low temperature. The low flow rate is also low and as the fluid passes through the nozzle it mixes with the hot (540°F) reactor coolant. This mixing is turbulent and results in alternating hot and cold cycling on the nozzle wall. The result is high cycle fatigue which initiates cracking. This mechanism has been demonstrated by test. Tests have also demonstrated that lower-frequency thermal cycles occur in a stagnant CRD return line nozzle.

The fix for this problem is the elimination of the CRD return flow to the vessel nozzle. It has been shown that the CRD system will operate satisfactorily with the return line cut and capped. This has been demonstrated by tests at Peach Bottom, Fitzpatrick, and other operating BWR's.

Stress analyses in keeping with the requirements of the ASME Code, Section III, will be performed to demonstrate the adequacy of the reactor vessel feedwater nozzle (and sparger) and the CRD return line nozzle cap.

QUESTION 112.8

Provide the criteria used in the design of supports for all ASME Class 1, 2 and 3 active pumps and valves to assure that the supports do not deform to the extend that operability of the supported components will not be impaired.

RESPONSE:

The recirculation piping suspension system to be supplied by GE will use three types of component supports. These are hangers, struts and snubbers to support the recirculation pumps and valves.

The design of the hanger supports which carry the load caused by dead weight only has already been completed and is in accordance with the rules and regulations of ANSI Code B31.7. The scope of supply responsibility of the component supports such as struts and snubbers which carry the dynamic loads is presently in the process of being negotiated between the customer and GE. In the event GE is responsible for supplying struts and snubbers, the design will comply with the requirements of ASME Code Section III, Subsection NF. All the component supports are designed, fabricated and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment while performing its function during the various operating conditions of the plant. The design load on each of these component supports is identified as follows:

(a) Hangers

The design load on hangers is the load caused by dead weight. The hangers are calibrated to ensure that they support the design load at both their hot and cold load settings. Hangers provide a specified down travel and up travel in excess of the specified thermal movement.

(b) Struts

The design load on struts includes those loads caused by dead weight. Thermal expansion, primary seismic loads, i.e. operating basis earthquake (OBE) and safe shutdown earthquake (SSE) and system anchor displacements, etc.

(c) Snubbers

The design load on snubbers includes those loads caused by seismic forces (OBE and SSE) and system anchor displacements, etc.

The analyses that are used for the design of these component supports to ensure that all such supports will not deform to the extent that would impair the pressure-retaining integrity of the supported components under normal, upset, emergency and faulted plant conditions, can essentially be divided into three parts as given below:

(a) Piping analysis to determine design loads on component supports.

The piping analysis is performed with GE SAP4 program. SAP4 is general Structural Analysis Program for static and dynamic analysis of linear elastic complex structures. The finite element displacement method is used to solve for the displacements and loads and compute the stresses of each element of the structure. The loads resulting from thermal expansion, dead weight, primary seismic loading (OBE and SSE), and system anchor displacements are first determined individually and then combined under normal, upset, emergency and faulted plant conditions to determine the design load on the respective component support. Piping supports are then designed by the load rating method and, in general, the load combinations for the various plant operating conditions correspond to those used to design the supported pipe.

Design transient cyclic data are not applicable to piping supports as no fatigue evaluation is necessary to meet Code requirements.

(b) Selection from the vendor data.

After determining the design load by piping analysis, component supports are selected from the vendor data that indicates loads are equal to or below the load rating of the components.

(c) Analysis and/or tests to demonstrate acceptability

The vendor performs analyses and/or tests to demonstrate acceptability for his load rating data on component supports. Also, the vendor performs analyses and/or tests to demonstrate that all such component supports will not deform under faulted plant conditions to the extent that would impair the required operability of the supported components to perform a safety function for safe shutdown of the plant.

For non-NSSS supports, please see Subsection 3.9.3.4.6 for a discussion of design criteria.

QUESTION 112.9

Provide the following information in the FSAR:

(1) A tabulation of snubbers utilized in your facility as supports for safety related systems and components including:

  (a) System identification and location
  (b) Type (hydraulic, mechanical)
  (c) Fabricator and rated load capacity
  (d) Function (shock or vibration arrestor, dual purpose)

(2) A summary of the contents of the snubber design specifications.

(3) A description of snubber suppliers performance qualification tests and load tests.

(4) A summary of system and component structural analyses showing:

  (a) Structural analysis model.

  (b) Description of the characterization of snubber mechanical properties used in the structural analysis including considerations such as (i) differences in tension and compression spring rates, (ii) effect of entrapped air and temperature on fluid properties, (iii) other factors affecting snubber characteristics.

  (c) List load conditions and transients analyzed.

  (d) Maximum snubber loads, corresponding piping or component stresses.

  (e) Comparison of computed loads and stresses from (d) above with rated snubber load and component stress intensity limits.

RESPONSE:

See revised Subsection 3.9.3.4.6 for this information.

QUESTION 112.10

To aid us in our licensing review of Susquehanna, you are requested to provide the following information to us:

1. Describe those actions being taken by you to preclude the occurrence of cracking such as described in 1E Bulletin 80-07.

2. Provide a commitment to adopt whatever long-term solution is approved.

3. If you anticipate receiving an Operating License before a long-term solution is agreed upon, describe any short-term actions which you will take to prevent or detect excessive cracking.

   Provide a rationale as to why these actions are sufficient to justify plant operation until a long-term solution is found.

RESPONSE:

1. Those step(s) to be taken to preclude the occurrence of cracking of the jet pump hold-down beams such as described in 1E Bulletin 80-07 are described in PLA-670 N. W. Curtis to B. J. Youngblood, dated March 25, 1981.

2. It is anticipated that there will be alternative long term solutions developed and approved. These approved solutions will be evaluated at the conclusion of current investigative activities and a commitment made at that time.

3. If a long term solution is not agreed upon prior to receipt of an operating license, the following short term actions will be taken to prevent or detect excessive cracking:

   a. A procedure will be instituted to monitor jet pump loop flow/recirculation pump speed ratios daily. A deviation from the normal range may indicate a problem wherein individual jet pump performance data will be used to determine if a problem exists.

   b. A performance monitoring program will be established to obtain and periodically update a "normal" operation data base. Operating data will then be compared to the data base at least weekly to provide an early indication of potential problems. Calibration checks of the instrumentation used in this program will be performed at least every 18 months or more frequently should there be a tendency for significant drift over the 18 month period.

c. A review of the jet pump operability technical specifications will be performed with the objective of making them more responsive to Operating experience. The recommended Technical Specification includes daily monitoring of Recirculation Pumps Flow/Speed Ratio and Jet Pump Loop Flow/Pump Speed Ratio to detect potential problems.