DEC 2 7 1973

8

V. A. Moore, Assistant Director
for Light Water Reactors, Group 2
Directorate of Licensing

ALVIN W. VOGTLE NUCLEAR PLANTS 1, 2, 3 & 4, DOCKET NOS. 50-424/425/426/427

Plant Name: Alvin W. Vogtle
Licensing Stage: CP
Docket Nos.: 50-424/425/426/427
Responsible Branch and Project Manager: LWR 2-2, L. Crocker
Requested Completion Date: 12/21/73
Description of Response: Safety Evaluation Report
Review Status: Complete - SER Input Only (see below)

The information submitted by the applicant in the PSAR and in amendments through #13 has been reviewed and evaluated by the Mechanical Engineering Branch. Appropriate sections of the Safety Evaluation are enclosed.

While the report as written indicates acceptance of each of the SER sections the information presented to date is incomplete or deficient in several areas. It is suggested that the issues be further discussed with the applicant as soon as practicable to seek resolution.

DKT #
50-424

Issue	SER Section Effected
1. Stress limits and design loading combinations for all plant operating conditions for ASME Class 1, 2, 3 components; design analysis methods for faulted condition of ASME Class 1 components	3.9.2.1 5.2.1.2 5.2.1.5 5.2.1.7
2. Seismic Qualification of Equipment	3.10
3. Pipe Break	3.6
4. Reactor Internals	3.9.1.3

Discussion of Issues

1. With regard to the design analysis methods for the faulted condition for Class 1 components Questions 5.13 and 5.14 remain unanswered. Methods proposed in RESAR have not been accepted as being equivalent to Appendix F of Section III of the Code. A commitment to use Appendix F would be acceptable. The issue of stress limits and loading combinations originally

MEMO

8503080322 841009
PDR FOIA
BELL84-664 PDR

OFFICE ▸						
SURNAME ▸						
DATE ▸						

posed in Question 3.37 remained incomplete at the Q2 milestone and information was again asked in 3.49.1. The response provided is still inadequate and should be revised to resolve the situations which follow. The applicability of RESAR-3, particularly where differences exist between RESAR-3 and the PSAR is not clear. Class 2 & 3 vessels are not covered in the PSAR but are covered in RESAR, Class 2 & 3 pumps and valves are covered in both places with differences in stress limits and loading combinations. The response to Question 3.49.1 indicates that the answer which should address all components appears in Section 3.7.2.1.1.6, yet this section covers only piping. Page 3.7-38 of the PSAR indicates stress limits and load combinations are under study and are candidates for revision. Load combinations appearing on Pages 3.7-37 and 3.9-1h of the PSAR for the upset and faulted conditions and in table 3.9-1 of RESAR are inconsistent, and both versions are less conservative than Regulatory Guide 1.48 and are not acceptable. To clarify this matter it is suggested that the applicant prepare a table of the load combinations and stress limits he is using for each of the plant operating conditions. These should be given for each of the twelve classes of components as categorized in Part C of Regulatory Guide 1.48.

2. Question 3.44 response does not provide a definitive commitment with regard to equipment seismic qualification. Our position was contained in the question and remains unchanged. We will require the use of criteria consistent with the AEC staff position or the draft IEEE Standard 344, 1973 in lieu of IEEE-344, 1971.

3. Section 3.6.5.3.2.1 of the PSAR does not provide the restraint design criteria for locations outside containment. This should be provided. Such information is provided in Section 3.6.5.1(A) of the PSAR for inside containment and is an acceptable value. The one intermediate break location criterion of 3.6.2.1.1(D) and 3.6.2.2(C)(3) of the PSAR, is acceptable only if justified that the effects of pipe break designed to this criteria are equivalent to the criteria contained in Sections C(1)(d) or C(2)(d) of Regulatory Guide 1.46. Reference to WCAP-8082 in Section 3.6 on page 3 of the PSAR is insufficient since the scope is limited to the determination of break locations for identically designed Westinghouse reactor coolant loops. The following additional information should be provided: (a) Impingement effects due to all pipe breaks including those in the reactor coolant loop, (b) movement of supports, forcing functions and dynamic analysis used for pipe whip restraint design, including the reactor coolant loop, (c) since WCAP-8082 does not include branch lines connected to the reactor coolant loop or any other ASME

OFFICE						
SURNAME						
DATE						

Class 1 systems, provide for their protection, (d) Table A-5 of WCAP-8082 is not acceptable. Appendix F of Section III should be used for faulted condition stress limits.

4. A vibration test program will be required for Vogtle or a plant prototypical of Vogtle.

Original signed by L. Shao for
R. R. Maccary

R. R. Maccary, Assistant Director
for Engineering
Directorate of Licensing

cc w/encl:
S. H. Hanauer, DRTA
J. M. Hendrie, L
K. Kniel, L
J. P. Knight, L
L. P. Crocker, L
R. J. Bosnak, L

cc w/o encl:
A. Giambusso, L
W. G. McDonald, L

Docket Files 50-424/425/426/427
L, Reading File
L:MEB File

OFFICE	L:MEB	L:MEB	L:MEB AD/E			
SURNAME	RJBosnak:jm	JPKnight	RRMaccary			
DATE	12/27/73	12/27/73	12/27/73			

MECHANICAL ENGINEERING BRANCH

DIRECTORATE OF LICENSING

ALVIN W. VOGTLE NUCLEAR PLANTS

UNITS 1, 2, 3, 4

DOCKET NOS. 50-424/425/426/427

SAFETY EVALUATION REPORT

3.0 Design of Structures, Components, Equipment and Systems

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

The design of piping restraints as applied to the reactor coolant pressure boundary and to other systems of piping and components important to safety within containment provides adequate protection for the containment structure, the unaffected reactor coolant system components, and those other systems important to safety which are either interconnected, or in close proximity to, the reactor coolant system or any other system of piping in which postulated pipe failures are assumed to occur within containment.

These provisions for protection against the dynamic effects associated with pipe ruptures and the resulting discharging fluid provide adequate assurance that, in the event of the occurrence of the combined loadings imposed by an earthquake of the magnitude specified for the Safe Shutdown Earthquake and a concurrent single pipe break of the largest pipe at one of the design basis break locations, the following conditions and safety functions will be accommodated and assured:

(1) the magnitude of the design basis loss-of-coolant accident can not be aggravated by potentially multiple failures of piping,

(2) the reactor emergency core cooling systems can be expected to perform their intended function,

(3) the containment structure's leak-tight integrity can be expected to be maintained in order to contain any radioactive materials released from the discharging coolant into the containment atmosphere.

The system which were considered, the locations and types of piping breaks which might occur, and the protection measures against pipe whip provided are consistent with Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment." The method of analysis used in the referenced topical reports adequately accounts for the dynamic loadings that are associated with the pipe rupture postulated, and will provide adequate assurance that the containment structure, unaffected system components, and those systems important to safety which are in close proximity to the systems in which postulated pipe failures are assumed to occur, will be protected.

The criteria used for the identification, design, and analysis of piping systems where postulated breaks may occur constitute an acceptable design basis in meeting in part the requirements of AEC General Design Criteria 1, 2, 4, 14, 15, 31 and 32.

In response to a request from the Regulatory staff, the applicant in amendment #9 to the PSAR has agreed to analyze the consequences of postulated pipe failures outside of the containment structure for the

FSAR. The applicant has further agreed to utilize criteria consistent with those contained in a Regulatory staff letter of July 12, 1973 addressed to utilities and architect engineers, in his analysis dealing with pipe whip, jet impingement, reaction forces and the environmental conditions which could result from pipe failure.

Note to Project Manager:
(Consistent with the division of responsibility for review of pipe whip outside containment the MEB has reviewed those aspects of the applicant's criteria that relate to the selection of design basis break locations and the analytical techniques to be employed to determine the resulting loads on structures, systems and components. Further review responsibility rests with the Auxiliary & Power Conversion Systems Branch and the Electrical, Instrumentation and Control Systems Branch.)

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

3.9.1.1 Vibration Operational Test Program

The applicant has agreed to perform a preoperational vibration dynamic effects test program, with procedures to be submitted at the FSAR stage, to check the vibration performance of piping important to safety. The vibration due to pump trips and/or valve closures will be checked during plant preoperation and start-up testing procedures. This program will provide adequate assurance that the piping and piping restraints of the system have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and operating modes associated with the design operational transients. The tests, as planned, will develop loads similar to those experienced during reactor operation. A commitment to proceed with such a program constitutes an acceptable design basis at the PSAR stage in part fulfillment of the requirement of AEC General Design Criterion 2.

3.9.1.2 Analysis and Tests of Mechanical Equipment

The applicant has submitted procedures which use acceptable dynamic testing and analysis techniques to confirm the adequacy of non-pressure retaining mechanical components (such as ventilation equipment, diesel generators, etc.) which are Seismic Category I to function during and after an earthquake of magnitude up to and including the SSE and that equipment supports are adequately designed to withstand seismic disturbance. Subjecting the equipment and its supports to these dynamic testing and analysis procedures provides reasonable assurance that in the event of an earthquake at the site, the Seismic Category I mechanical equipment will continue to function during and after a seismic event.

Implementation of these dynamic testing and analysis procedures, constitutes an acceptable basis for satisfying the requirements of AEC General Design Criteria 2 and 14.

3.9.1.3 Preoperational Vibration Assurance Program for Reactor Internals

The preoperational vibration assurance program that the applicant has specified for the reactor internals provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions that will be comparable to those experienced during operation. The Indian Point 2 plant is designated as the prototype on which vibration behavior of the reactor internals was analyzed, tested and measured. There are however differences between the internals of the designated prototype and the Vogtle plant; Indian Point 2 has a thermal shield, Vogtle will be equipped with neutron shielding pads. Vogtle will also employ the 17 x 17 fuel assembly in place of the 15 x 15 fuel assembly used in Indian Point 2. The staff will require that a vibration test program be implemented for the Vogtle plant or for a plant which is prototypical of Vogtle and which will reach operating status prior to Vogtle. Visual inspection after hot functional testing will also be implemented to provide added confirmation of the capability of structural elements of the reactor internals to sustain the flow-induced vibrations. The combination of tests, predictive analysis and post-test inspection provide adequate assurance that the reactor internals may be expected, during their service lifetime, to withstand the flow-induced vibrations of reactor operations without loss of structural integrity. The continued integrity of the reactor internals in service is essential to assure the retention of all reactor fuel assemblies in their place as well as to permit unimpaired operation of the control rod assemblies in order to permit safe reactor operation and shutdowns.

3.9.1.5 Analysis Methods Under LOCA Loadings

To confirm the structural design adequacy of the reactor internals to withstand the combined dynamic effects of the postulated loss-of-coolant accident (LOCA) and a safe shtudown earthquake (SSE) the applicant has agreed to perform a dynamic analysis which is consistent with the Regulatory staff position and should provide adequate assurance that the combined stresses and strains in the components of the reactor coolant systems, and reactor internals will not exceed the allowable design stress and strain limits for the material of construction, and that the resulting deflections or displacements at any structural elements of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The assurance of structural integrity of the reactor internals under LOCA conditions for the postulated most adverse loading event provides added confidence that the design may be expected to withstand a spectrum of lesser pipe breaks and seismic loading events. Satisfactory implementation of this commitment constitutes an acceptable basis for satisfying in part, the requirements of AEC General Design Criteria 10.

3.9.2 ASME Code Class 2 and 3 Components

3.9.2.1 Design, Load Combinations and Stress Limits

All Seismic Category I pressure retaining systems, components and equipment outside of the reactor coolant pressure boundary, including active pumps and valves, are designed to sustain normal loads, anticipated transients, the one-half Safe Shutdown Earthquake, and the Safe Shutdown Earthquake within stress limits which are comparable to those outlined in Regulatory Guide 1.48, "Design Limits and Loading Combinations." The specified design basis combinations of loading as applied to the design of the safety-related ASME Code Class 2 and 3 pressure-retaining components in systems classified as Seismic Category I provide reasonable assurance that in the event (a) an earthquake should occur at the site, or (b) an upset, emergency or faulted plant transient should occur during normal plant operation, the resulting combined stresses imposed on the system components may be expected not to exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of the system components to withstand the most adverse combinations of loading events without gross loss of structural integrity. The design load combinations and associated stress and deformation limits specified for all ASME Code Class 2 and 3 components, including the active pumps and valves, constitute an acceptable basis for design in satisfying the General Design Criteria 1, 2 and 4 and are consistent with recent Regulatory positions.

3.9.2.2 Pressure Relief Devices

The maximum full discharge loads resulting from the opening of ASME Class 2 safety and relief valves are calculated by a dynamic anlaysis of the system. The maximum stress intensities and stresses resulting from these loads will be calculated in accordance with Subarticle NC-3600 of Section III of the ASME Code. In the case of safety or relief valves mounted on a common header with full discharge occurring simultaneously the additional stresses induced in the header will be combined with previously computed local and primary membrane stresses to obtain the maximum stress intensity.

The criteria used in developing the design and mounting of the safety and relief valves of ASME Class 2 systems provides adequate assurance that, under discharging conditions, the resulting stresses are expected not to exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design of the system components to withstand these loads without loss of structural integrity and impairment of the overpressure protection function.

The criteria used for the design and installation of overpressure relief devices in ASME Class 2 Systems constitute an acceptable design basis in meeting, in part, the requirements of AEC General Design Criteria 1 & 2, 4, 14 and 15.

3.9.2.4 <u>Class 2 and 3 Active Valves and Pumps Operability Assurance Program</u>

The applicant has agreed to utilize an operability assurance program, in addition to the limits on stress and deformation, to qualify active ASME Class 2 and 3 Seismic Category I pumps and valves. Such a program will include component testing, or a combination of tests and predictive analysis supplemented by seismic qualification testing of motors, operators, and component appendages to provide assurance that such components can withstand postulated seismic loads in combination with other significant loads without loss of structural integrity, and can perform the "active" function (i.e., valve closure or opening or pump operation) when a safe plant shutdown is to be effected, or the consequences of an accident are to be mitigated. A commitment to develop and utilize a component operability assurance program satisfactory to the staff constitutes an acceptable basis for implementing the requirements of General Design Criterion #1 as related to operability of ASME Code Class 2 and 3 active valves.

3.9.3 Components Not Covered by the ASME Code - Mechanical Design
4.2 of Fuel Assemblies - Mechanical Design of Control Rod Drives

The fuel and control rod assemblies and control rod drives of the Vogtle Nuclear Station Units 1, 2, 3 and 4 are comparable, except for differences introduced by substitution of the 17 x 17 fuel assemblies for the previous 15 x 15 fuel assemblies, to the assemblies and drives of a number of nuclear power plants now operable. The design criteria which have been applied are comparable to those which were found acceptable for Indian Point Station Unit 2 and additional analyses and test programs are both in process and planned for the immediate future to demonstrate both the integrity of the modified designs and the validity of applying experience with the 15 x 15 assemblies to the evaluation of the conceptually similar 17 x 17 fuel assemblies. Information from these tests substantiating the design is scheduled to be provided in part in interim design reports to be provided for the Catawba Power Plant which is also applicable to the Vogtle plants.

The design criteria being used and the successful completion and proper documentation of the results of the tests described will provide reasonable assurance that the fuel and control rod assemblies and control rod drives may be expected to withstand the imposed loads associated with normal reactor operation, anticipated operational transients, postulated accidents, and seismic events without gross loss of their structural integrity or impairment of function. Compliance with these design criteria fulfills the requirements of AEC General Design Criteria 2 and 14 as these criteria relate to fuel and control rod assemblies, and control rod drives.

3.10 Seismic Qualification of Category I Instrumentation and Electrical Equipment

Operability of the instrumentation and electrical equipment is essential to assure the capability of such equipment to initiate protective actions in the event of a safe shutdown earthquake (SSE) as necessary for the operation of engineered safety features and standby power systems. The proposed seismic qualification program which will be implemented for Seismic Category I instrumentation and electrical equipment and supports will provide assurance that such equipment may be expected to function properly and that structural integrity of the supports will be maintained during the excitation and vibratory forces imposed by the safe shutdown earthquake under the conditions of post-accident operation. The referenced IEEE Standard 344, 1971, is undergoing a major revision which will make it consistent with the requirements of the AEC staff position. We will require the use of test and input criteria consistent with the criteria now contained in the draft IEEE Standard 344, 1973. A detailed presentation concerning the results of test and analysis will be evaluated during the review of the Final Safety Analysis Report. Commitment to perform such a program constitutes an acceptable basis for satisfying staff requirements and AEC General Design Criterion 2.

5.0 Reactor Coolant System and Connected Systems

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1.2 Design of Reactor Coolant Pressure Boundary Components
5.2.1.5 Design, Load Combinations, Stress Limits
5.2.1.7

The specified design transients, design loadings and combination of loading as applied to the design of the reactor coolant pressure boundary components provide reasonable assurance that in the event (a) an earthquake should occur at the site, or (b) a system upset, emergency, or faulted transient should occur during normal plant operation, the resulting combined stresses imposed on the system components may be expected not to exceed the allowable design stresses and strain limits for the materials of construction.

Limiting the stresses and strains under such loading combinations provides an acceptable basis for the design of the system components to withstand the most adverse loading events which have been postulated to occur during the service lifetime without gross loss of the system's structural integrity. Special considerations which would require a review for potential reactor pressure vessel failure have not been identified in the staff safety review. The design load combinations and associated stress and deformation limits specified for the reactor coolant pressure boundary components, including active valves, constitute an acceptable basis for design in satisfying the General Design Criteria 1, 2 and 4, and are comparable to those in Regulatory Guide 1.48.

5.2.1.6 Reactor Coolant Pressure Boundary Component Operability Assurance Program

The applicant has identified the active components within the reactor coolant pressure boundary for which operation is required to safely shut down the plant and maintain it in a safe condition in the event of a safe shutdown earthquake or design basis accident. The applicant has agreed to utilize an operability assurance program, in addition to stress and deformation limits, to qualify active valves. Such a program will include valve testing, or a combination of tests and predictive analysis, supplemented by seismic qualification testing of valve operator systems to provide assurance that active components (1) will withstand the imposed loads associated with normal, upset, emergency and faulted plant conditions without loss of structural integrity and (2) will perform the "active" function under conditions comparable to those expected when safe plant operation or shutdown is to be effected, or the consequences of a seismic transient or of an accident are to be mitigated.

A commitment to develop and utilize a component operability assurance program satisfactory to the staff constitutes an acceptable basis for implementing the requirements of AEC General Design Criterion 1 as related to the operability of ASME Code Class 1 active valves.

5.2.2.2 Mounting of Pressure Relief Devices

The maximum full discharge loads resulting from the opening of all ASME Class 1 safety and relief valves are to be calculated by a dynamic analysis of the system. The maximum stress intensities and stresses resulting from these loads will be calculated in accordance with Subarticle NB-3600 of Section III of the ASME Code. In the case of safety or relief valves mounted on a common header and full discharge occurring simultaneously, the additional stresses induced in the header will be combined with previously computed local and primary membrane stresses to obtain the maximum stress intensity.

The criteria used in developing the design and mounting of the safety and relief valves of the reactor coolant pressure boundary provides adequate assurance that, under discharging conditions, the resulting stresses are expected not to exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design of the system components to withstand these loads without loss of structural integrity and impairment of the overpressure protection function.

The criteria used for the design and installation of overpressure relief devices in reactor coolant pressure boundary constitute an acceptable design basis in meeting the applicable requirements of AEC General Design Criteria 1 & 2, 4, 14 and 15.