ENCLOSURE 1

MECHANICAL ENGINEERING BRANCH

DIVISION OF TECHNICAL REVIEW

DAVIS BESSE NUCLEAR POWER PLANT UNIT 1

SAFETY EVALUATION REPORT

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

3.6.1 Inside and Outside Containment
3.6.2

The criteria to be used in the design of piping systems in the Davis Besse Nuclear Power Station, Unit No. 1 are consistent with NRC Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment" for piping inside containment and the A. Giambusso letter of December 15, 1972 to the Toledo Edison Company for piping outside of containment.

These provisions for protection against the dynamic effects associated with pipe ruptures and the resulting discharging coolant provide acceptable 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 (SSE) 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) Structures, systems and components important to safety will be appropriately protected.

The analytical methods and procedures that were used to establish restraint locations and the pipe/restraint dynamic interaction are based on classical methods of engineering analysis that have been shown through practice to produce acceptable results. The pipe whip restraints are designed to withstand the resultant loadings in accordance with acceptable criteria that allow a maximum of twenty percent of the uniform ultimate strain for restraints that function by energy absorbtion under essentially uniaxial loading.

On the basis of our review, we have concluded that the criteria that will be used for the identification, design and analysis of piping systems where postulated breaks may occur constitute an acceptable design basis in meeting the applicable requirements of NRC General Design Criteria Nos. 1, 2, 4, 14 & 15 and are consistent with the staff position for plants under review for an operating license.

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

3.9.1.1 Piping Preoperational Vibration Test Program

To assure that the vibration of piping systems is within acceptable levels, the applicant will conduct a piping preoperational vibration dynamic effects test program on ANSI B31.7 Class I and ASME Class 1, 2 and Class 3 safety related piping systems and piping restraints during startup and initial operating conditions.

The content of the program is acceptable and it is expected that the tests 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 and will demonstrate structural integrity and design adequacy concerning preoperational piping dynamic effects test programs.

Compliance with this test program constitutes an acceptable basis, in partial fulfillment of the requirement of Criterion 15 of the NRC General Design Criteria.

3.9.1.2 Analysis and Tests of Mechanical Equipment

To seismically qualify all Category I mechanical equipment, the applicant utilized a program which employs acceptable dynamic testing and analysis techniques to confirm the adequacy of mechanical equipment including their supports to function during and after an earthquake of magnitude up to and including the SSE. Subjecting the equipment and supports to these dynamic test 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 the seismic event.

These criteria are an acceptable basis for satisfying in part the requirements of NRC General Design Criteria 2 and 14.

3.9.1.3 Preoperational Vibration Assurance Program for Reactor Internals

The applicant has designated the Oconee 1 unit flow-induced viration test program as the prototype for Davis Besse 1. The applicant will perform an uninstrumented preoperational test on the reactor internals with subsequent visual inspection to confirm the design adequacy of the reactor internals to flow induced biration. This procedure is consistent with Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing" relative to non-prototype units.

The preoperational vibration assurance program as planned 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 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.

Satisfactory completion of the preoperational vibration assurance program constitutes an acceptable basis for demonstrating design adequacy of the reactor internals in partially fulfilling the requirements of NRC General Design Criteria 1 and 4 and in conforming with the provisions of Regulatory Guide 1.20.

3.9.1.5 Analysis Methods for Design Basis Accident (LOCA) Loadings

The applicant has performed a dynamic system analysis of the reactor internals and of the broken and unbroken piping loops, however, in view of the recent information presented to NRC for the North Anna units, we are requesting that the Davis Besse applicant present further information to positively determine that the reactor vessel and its supports can adequately withstand an instantaneous cold leg break at the reactor nozzle. This information will be evaluated when received and a determination made of its acceptability in a supplement to this Safety Evaluation.

3.9.2 ASME Code Class 2 and 3 Components

All Seismic Category I 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 Operating Basis Earthquake, and the Safe Shutdown Earthquake within design limits which are comparable to those outlined in NRC Regulatory Guide 1.48, "Design Limits and Loading Conditions." The specified design basis combinations of loadings 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) other upset, emergency or faulted plant transients 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 constitute an acceptable basis for design in satisfying NRC General Design Criteria 1, 2 and 4 and are consistent with recent Regulatory positions.

The applicant has conducted component test programs, supplemented by analytical predictive methods which provide adequate assurance that the capability of ASME Code Class 2 and 3 active pumps and valves (a) to withstand the imposed loads associated with Normal, Upset, Emergency and Faulted plant conditions without loss of structural integrity and (b) to perform the "active" function, is confirmed under combinations of conditions comparable to those expected when a safe plant shutdown is to be effected, or the consequences of an accident are to be mitigated.

We have concluded that the design and analytical procedures used by the applicant provides reasonable assurance of pump and valve operability.

The criteria used in developing the design and mounting of the safety and relief valves of ASME Code 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 Code Class 2 Systems are consistent with Regulatory Guide 1.67, "Installation of Over Pressure Protection Devices" and constitute an acceptable design basis in meeting the applicable requirements of NRC General Design Criteria Nos. 1, 2 and 4.

3.10 Seismic Qualification of Category I Instrumentation and Electrical Equipment

Instrumentation and electrical components required to perform a safety function are designed to meet Category I design criteria. Seismic requirements established by the seismic system analysis have been incorporated into equipment specifications to assure that the equipment purchased or designed meets seismic requirements equal to or in excess of the requirements for Category I components, either by appropriate analysis or by qualification testing.

The applicant has implemented a seismic qualification program for Category I instrumentation and electrical equipment and the associated supports for that equipment to provide assurance that such equipment can be expected to function properly and that structural integrity of the supports will not be impaired during the excitation and vibratory forces imposed by the safe shutdown earthquake and the conditions of post-accident operation. The general program, as outlined, constitutes an acceptable basis for satisfying staff requirements and the applicable requirements of NRC General Design Criterion No. 2.

The applicant has referenced IEEE Standard 344, 1971 for seismic qualification of Category I electrical equipment, BAW 10003 and supplemental requirements. Conformance with such criteria has provided an acceptable method of seismic qualification.

4.2 Mechanical Design of Reactivity Control System

The design criteria applied and the tests performed on the reactivity control system of Davis Besse Unit 1 are similar to those of other recently reviewed plants which were found acceptable. The use of these criteria provide reasonable assurance that this system 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 structural integrity or impairment of function. Compliance with these design criteria forms an acceptable basis for satisfying the mechanical reliability stipulations of General Design Criterion 27.

5.2 Reactor Coolant Pressure Boundary

5.2.2 Component Design

The design loading combinations specified for RCPB components are comparable to the plant conditions currently identified as Normal, Upset, Emergency or Faulted. The design limits used by the applicant for these plant conditions are comparable to the criteria recommended in NRC Regulatory Guide 1.48. With the exception discussed in Section 3.9.1.5, use of these criteria for the design of the RCPB components provides reasonable assurance that, (1) in the event an earthquake should occur at the site, or (2) other system upset, emergency or faulted conditions should develop, the resulting combined stresses imposed on the system components will not exceed the allowable design stresses and strain limits for the materials of construction. Limiting the stresses and strains under such loading combinations provides a basis for the design of the system components for the most adverse loadings postulated to occur during the service lifetime without loss of the system's structural integrity. The load combinations and associated stress and deformation limits considered in the design of RCPB components constitute an acceptable basis for design in satisfying the related requirements of AEC General Design Criteria Nos. 1, 2 and 4.

The applicant has conducted component test programs, supplemented by analytical predictive methods which provide adequate assurance that the capability of ASME Code Class 1 active valves (a) to withstand the imposed loads associated with Normal, Upset, Emergency and Faulted plant conditions without loss of structural integrity and (b) to perform the "active" function, is confirmed under combinations of conditions comparable to those expected when a safe plant shutdown is to be effected, or the consequences of an accident are to be mitigated.

We have concluded that the design and analytical procedures used by the applicant provides reasonable assurance of valve operability.

The criteria used in developing the design and mounting of the safety and relief valves of ASME Code Class 1 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 Code Class 1 Systems are consistent with Regulatory Guide 1.67, "Installation of Over Pressure Protection Devices," and constitute an acceptable design basis in meeting the applicable requirements of NRC General Design Criteria Nos. 1, 2, 4, 14 and 15.

5.2.8 Loose Parts Monitor

Occasionally, miscellaneous items such as nuts, bolts, and other small items have become loose parts within reactor coolant systems. In addition to causing operational inconvenience, such loose parts can damage other components within the system or be an indication of undue wear or vibration. During recent reviews many applicants have either elected to institute their own program or to participate within another ongoing program, the objective of which was the development of a functional, loose parts monitoring system within a reasonable period of time. Recently, prototype loose parts monitoring systems have been developed and are presently in operation or being installed at several plants. The applicant will install a system which will monitor acoustically the top of each steam generator and the bottom of the reactor vessel.

We have concluded that the installation of such a system on the Davis Besse Unit 1 plant is an acceptable method of implementing the staff requirement.

ENCLOSURE 2

MECHANICAL ENGINEERING BRANCH

DIVISION OF TECHNICAL REVIEW

DAVIS BESSE NUCLEAR POWER PLANT UNIT 1

REQUEST FOR ADDITIONAL INFORMATION

Recent analyses have shown that reactor pressure vessel supports may be subjected to previously underestimated lateral loads under the conditions that would exist if an instantaneous double ended break is postulated in the reactor vessel cold leg pipe at the vessel nozzle. It is therefore necessary to reassess the capability of the reactor coolant system supports to limit the calculated motion of the reactor vessel during a postulated cold leg break within bounds necessary to assure a high probability that the reactor could be brought safely to a cold shutdown condition.

The following information is required for purposes of making the necessary reassessment of the reactor vessel supports for the Davis Besse Unit No. 1 plant:

1. Provide engineering drawings of the reactor support system sufficient to show the geometry of all principle elements and materials of construction.

2. Specify the detail design loads used in the original design analyses of the reactor supports giving magnitude, direction of application and the basis for each load. Also provide the calculated maximum stress in each principle element of the support system and the corresponding allowable stresses.

3. Provide the information requested in 2 above for the RV supports considering a postulated break at the cold leg nozzle. Include a summary of the analytical methods employed and specifically state the effects of short term pressure differentials across the core barrel in combination with all external loadings calculated to result from the required postulate. This analysis should consider:
   (a) limited displacement break areas where applicable
   (b) consideration of fluid structure interaction
   (c) use of actual time dependent forcing function
   (d) reactor support stiffness.

4. If the results of the analyses required by 3 above indicates loads leading to inelastic action in the reactor supports or displacements exceeding previous design limits provide an evaluation of the following:
   (a) Yield behavior (effects of possible strain energy buildup) of the material used in the reactor support design and the effect on the loads transmitted to the reactor coolant system and the back-up structures to which the reactor coolant system supports are attached.

   (b) The adequacy of the reactor coolant system piping, control rod drives, steam generator and pump supports, structures surrounding the reactor coolant system, reactor internals and ECCS piping to assure that the reactor can be safely brought to cold shutdown.