3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.1.1 Design Transients

3.9.1.1.1 Load Combinations

The load combinations considered in the original design of Ginna Station were (1) normal + design earthquake, (2) normal + maximum potential earthquake, and (3) normal + pipe rupture loads. "Normal," "Upset," "Emergency," and "Faulted" terminology was not used in the original safety evaluation of Ginna Station.

3.9.1.1.2 Cyclic Loads

3.9.1.1.2.1 Thermal and Pressure Cyclic Loads

The various components in the reactor coolant system were designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes. These cyclic loads are introduced by normal unit load transients, reactor trip, and startup and shutdown operation (see Section 5.1.5). The number of thermal and loading cycles used for design purposes is shown in Table 5.1-4.

3.9.1.1.2.2 Pressurizer Surge Line

NRC Bulletin 88-11 requested licensees to take certain actions to monitor thermal stratification in the pressurizer surge line because recent measurements indicate that top-to-bottom temperature in the surge line can reach 250°F to 300°F in certain modes of operation, particularly during heatup and cooldown. Surge line temperature stratification causes bending of the pipe and possible reduction of fatigue life. RG&E joined the Westinghouse Owners Group in a program to perform a generic evaluation of surge line stratification in Westinghouse PWRs. Temporary thermocouples were installed on the pressurizer surge line and four temporary displacement transducers were installed on the surge line to monitor movement during heatup, cooldown, and other temperature stratification conditions. The data was continuously monitored by a data logging computer installed in the Multiplexer (MUX) room for the duration of the test, which commenced in June 1989 and was completed during the 1990 MODE 6 (Refueling) outage when the instrumentation was removed.

The generic evaluation of surge line stratification in Westinghouse PWRs was reported in Westinghouse Owners Group report, WCAP 12639, submitted to the NRC in June 1990. Westinghouse performed a plantspecific analysis of the Ginna pressurizer surge line to demonstrate compliance with NRC Bulletin 88-11, and the results were reported in WCAP 12928 (*Reference 1*). The results indicated that the surge line meets the stress limits and usage factor requirements, and the pressurizer surge nozzle meets the code stress allowables under thermal stratification loading and fatigue usage requirements of ASME Section III, 1986 edition. By *Reference 20*, the NRC found the RG&E response to Bulletin 88-11 to be acceptable.

3.9.1.1.2.3 Unisolable Connections to the Reactor Coolant System

NRC Bulletin 88-08 requested licensees to review systems connected to the reactor coolant system piping to determine whether unisolable sections of piping connected to the reactor coolant system can be subjected to stresses from temperature stratification or temperature oscillations that could be induced by leaking valves and that were not evaluated in the design analysis of the piping. The Bulletin requested that

a. For any unisolable sections of piping connected to the reactor coolant system that may have been subjected to excessive thermal stresses, licensees nondestructively examine the welds, heat-affected zones, and high stress locations, including geometric discontinuities in that piping, to provide assurance that there are no existing flaws.

b. Licensees plan and implement a program to provide continuing assurance that unisolable sections of all piping connected to the reactor coolant system will not be subjected to combined cyclic and static thermal and other stresses that could cause fatigue during the remaining life of the unit. This assurance may be provided by

    1. Redesigning and modifying these sections of piping to withstand combined stresses caused by various loads including temporal and spatial distributions of temperature resulting from leakage across valve seats.
    2. Instrumenting this piping to detect adverse temperature distributions and establishing appropriate limits on temperature distributions.
    3. Means for ensuring that pressure upstream from block valves that might leak is monitored and does not exceed reactor coolant system pressure.

RG&E determined that there were three unisolable sections of piping connected to the reactor coolant system that had the potential for thermal cycling. These sections are as follows:

aa. Charging system to loop B hot leg between check valve 393 and the reactor coolant system nozzle.

bb. Alternate charging system to loop A cold leg between check valve 383A and the reactor coolant system nozzle.

cc. Auxiliary pressurizer spray system between check valve 297 and the 3-in. tee, which connects the auxiliary pressurizer spray to the main pressurizer spray line.

Examinations were performed at the most susceptible locations, as recommended by Westinghouse, on each of the three unisolable pipe sections. All examination results were acceptable.

A program to provide assurance that the identified unisolable sections of piping attached to the reactor coolant system do not fail, due to thermally initiated or advanced fatigue, was initiated. This assurance was provided, in part, by instrumenting the affected piping to detect adverse temperature conditions and by nondestructive examinations during MODE 6 (Refueling) outages. Temporary thermocouples were installed on the affected piping during the 1989 MODE 6 (Refueling) outage. The data was monitored by a data logging computer installed in the MUX room for that purpose.

The temperature monitoring was continued until the 1991 refueling outage when the instrumentation was removed. The data was analyzed and it was determined that adverse temperature conditions did not exist. Based on the results of the temperature monitoring, nondestructive examinations, and engineering analysis, the program was restructured to provide continued assurance based on periodic nondestructive examinations during MODE 6 (Refueling) outages. By *Reference 21*, the NRC reported that the staff had determined that the RG&E response to Bulletin 88-08 met the requirements.

3.9.1.1.3 Transient Hydraulic Loads

Transient hydraulic loads were considered in the dynamic analysis of the pressurizer safety and relief valve discharge lines (*References 2* and *22*) (see Section 3.9.2.1.4).

3.9.1.1.4 Operating-Basis Earthquake

The mechanical systems and components in the original design of Ginna Station were designed for the operating-basis earthquake using the response spectra developed by Housner and characterized by a peak ground acceleration of 0.08g at 0.5% damping. The operating-basis earthquake was not considered during the Systematic Evaluation Program (SEP) reevaluation (see Section 3.7).

3.9.1.1.5 Safe Shutdown Earthquake

The mechanical systems and components in the original Ginna design were reviewed for a safe shutdown earthquake of 0.2g peak ground acceleration. The response spectra developed by Housner were used for this purpose. For the SEP review, the seismic input motion was typically defined by means of floor response spectra generated by direct method or by means of a time-history analysis. See Section 3.7 for details of how the floor response spectra were developed.

3.9.1.1.6 Secondary System Fluid Flow Instability (Water Hammer)

Secondary system flow instability (water hammer) was considered in the dynamic analysis of the main and auxiliary feedwater piping (*Reference 3*) presented in Section 3.9.2.1.6. It was determined that the primary cause for water hammer was the recovery of the feed ring while feedwater flows were above a threshold flow. This threshold flow was determined to be approximately 200 gpm. Design of the feed ring piping, installation of J-tubes in the feed ring and operating procedures minimize the possibility of water hammer.

3.9.1.1.7 Loss-of-Coolant Accident

The forces exerted on reactor internals and core, following a loss-of-coolant accident, were originally computed by employing the BLODWN-1 digital computer program developed for the space-time-dependent analysis of multiloop PWR plants (see Section 3.9.2.3). Additional analysis of the blowdown effects was performed during the resolution of the unresolved safety issue A-2, Asymmetric Blowdown Loads, discussed in Section 3.9.2.4.

3.9.1.2 Computer Programs Used in Analysis

The following computer programs were used in the dynamic and static analyses of the Seismic Category I systems and components:

ITCHVALVE	Used to perform the transient hydraulic analysis of the pressurizer safety and relief line analysis.
FORFUN	Used to calculate unbalanced forces for each straight segment of pipe from the pressurizer to the relief tank.
WESTDYN	A special purpose program designed for the static and dynamic solution of redundant piping systems with arbitrary loads and boundary conditions.
FIXFM and FIXFM3	Computer programs which determine the time-history response of three-dimensional structures excited by an internal forcing function.
WESTDYN-2 and WESTDYN2	A slightly modified version of WESTDYN program, this program accepts the time-history displacements from FIXFM (or FIXFM3) and calculates the time-history internal forces in the pipe elements.
ADLPIPE	Was used in the original pipe stress analysis of Ginna Station. The verification of this piping analysis program developed by Arthur D. Little, Inc., was provided to the NRC in a memorandum dated April 19, 1979.
M003	A Gilbert/Commonwealth computer program for piping stress analysis. It consists of the Southern Service Company thermal stress program and the IBM scientific subroutine for eigenvalue problems. M003 has been verified against PIPDYN II.
PIPDYN II	(Gilbert/Commonwealth version) - A piping analysis computer program developed by Franklin Institute Research Laboratory. It has been verified against ASME Sample Problem No. 1 in the ASME publication, Pressure Vessel and Piping: 1972 Computer Programs Verification, and ANSYS and PIPESD.
DYNAFLEX	A piping analysis computer program developed by Auton Computing Corporation. It has been verified against ADLPIPE and PIPESD.
PIPESD	A piping analysis computer program developed by URS/John A. Bloom and Associates. It has been verified against ANSYS, ADLPIPE, PIPDYN, and SAP IV.
NUPIPE	A piping analysis computer program developed by Nuclear Services Corporation. It has been verified against ADLPIPE and ASME Benchmark Problem No. 5 in the ASME publication, Pressure Vessel and Piping: 1972 Computer Programs Verification.
PIPSAN	A Westinghouse piping support analysis code.
PS+CAEPIPE	Ginna in house piping anaylsis code.
PD STRUDL	Structural finite element code used @ Ginna.

3.9.1.3 **Experimental Stress Analysis**

3.9.1.3.1 Plastic Model Analysis

During the original design of Ginna Station the mode shapes and frequencies of the primary coolant loop piping system were determined experimentally using model analysis (*Reference 4*).

A plastic model was employed to perform this analysis. Since the reactor pressure vessel, the steam generator, the reactor coolant pump, and their supports are integral to the analysis of the primary loop, they were included in both the plastic model and the mathematical model. The plastic model output of mode shapes and frequencies was coupled with the Housner 0.2g response spectra and used as input to a three-dimensional mathematical model of the primary coolant loop. A computer solution to yield stresses, deflections, support reactions, and equipment nozzle reactions was obtained.

3.9.1.3.2 Plastic Model Details

The model, shown in Figure 3.9-2, was built with a geometric ratio of 0.25 in. equals 1 ft. The plastic model material used was ABS plastic extrusion grade for piping and plexiglas for support structures and equipment. The reactor pressure vessel, steam generator, and reactor coolant pump were represented by hollow circular plastic cylinders filled with lead shot positioned with cotton spacers to properly represent the mass and center of gravity locations of these three pieces of equipment. They were supported by modeled plastic supports.

For a steel beam of identical geometry the natural frequency of the cantilever is 114 Hz. Therefore,

f(steel)/ f(plastic) = 2.78

The ratio of the natural frequency of the model to the prototype was determined by

$$\frac{\omega_{model}}{\omega_{prototype}} = \frac{L_p}{L_m}\sqrt{\frac{E_M}{E_p}\cdot\frac{p}{m}}$$

(Equation 3.9-1)

where L_p/L_m = geometric factor and

$$\sqrt{\frac{E_M}{E_p}\cdot\frac{p}{m}} = \text{material factor.}$$

(Equation 3.9-2)

Therefore

ω(model)/ ω (prototype) = 48 / 2.78 = 17.2

3.9.1.3.3 Plastic Model Test Arrangement

Three separate tests were conducted in order to examine the response of the model to a sinusoidal input at various levels. A vertical test and horizontal tests in two perpendicular directions were conducted.

In the horizontal tests, the model was flexibly suspended from a framed supporting structure. One end of the base plate of the model was then secured to the MB vibrator. The arrangement was such that the rigid body rocking modes frequencies were much lower than the frequencies of interest in the piping system. The sizable moment introduced by not driving through the dynamic center of gravity of the system was therefore not a problem. It was possible to conduct the tests in the intended linear direction without very much cross talk or rocking motion.

There was a slight distortion in the geometric scaling of the connecting piping because of available model materials. This geometric relationship is as follows:

Location	Actual Pipe Size		Assumed Pipe Size		Model Pipe Size	
	I.D.	O.D.	I.D.	O.D.	I.D.	O.D.
Cold leg	27.5	32.3	30	36	5/8	3/4
Crossover	31.0	36.8	30	36	5/8	3/4
Hot leg	29.0	34.0	30	36	5/8	3/4
All dimensions are in inches.						

To determine the properties of the plastic, a rectangular sample was separately measured and dynamically tested. The sample was clamped as a cantilever beam to the vibrator and the frequency noted.

The dynamic modulus of elasticity was then calculated. Physical characteristics are as follows:

Sample size = 0.25 x 10 x 1 in.

Volume = 2.5 in.^{3}

Weight = 0.1 lb

Density = 0.04 lb/in.^{3}

For a cantilever beam 8.5 in. long, the test natural frequency was 41 Hz.

Using the equation

$$f_n = \frac{0.56}{l^2}\sqrt{\frac{gEI}{\rho A}} = 0.56\frac{h}{l^2}\sqrt{\frac{EI}{\rho A}}$$

(Equation 3.9-3)

Then the dynamic modulus is

E (plastic) = 547,000 psi

The vertical test was conducted with the model mounted directly to the exciter plate of the vibrator. Since the geometry of the model permitted driving through the center of gravity of the system, rocking excitation was again minimized.

Resonant frequencies and mode shapes were noted by sweeping the model frequency span of 17 to 172 Hz and noting the modal response of the model by use of a strobotac light.

3.9.2 DYNAMIC TESTING AND ANALYSIS

3.9.2.1 Piping Systems

3.9.2.1.1 General

All safety-related and non-safety-related piping systems were originally designed and fabricated to the requirements of USAS B31.1, Power Piping Code. Since the original construction, repairs and/or modifications have been made that have been designed and fabricated to later codes, including ASME Section III. Reanalysis of critical safety-related piping 2-1/2 in. and larger was performed under the Seismic Upgrade Program, which was reviewed by the NRC under SEP Topic III-6 (see Section 3.9.2.1.8). This program updated the piping analysis basis to criteria consistent with the ANSI B31.1 Code, including Summer 1973 Addenda, with some amendments. This code edition remains as the current analysis basis for modifica- tions performed on safety-related piping. Non-safety-related piping is designed and fabricated in accordance with the appropriate current edition of ANSI B31.1.

The loads and load combinations considered in the original design of Ginna Station are given in Table 3.9-1.

The original Ginna Station design did not utilize dynamic computer analyses for seismic qualification of Seismic Category I piping. Seismic Category I piping was divided into three groups, reactor coolant system piping, piping 2-1/2 in. nominal size and larger and piping 2-in. nominal size and smaller. The reactor coolant system piping was seismically qualified using a combination of model testing and analysis. Seismic Category I piping, 2-1/2 in. nominal pipe size and larger, was seismically qualified using equivalent static analyses. Seismic Category I piping, 2-in. nominal pipe size and smaller, was seismically qualified using support spacing tables. Dynamic analysis of sections of the A residual heat removal and B main steam piping were performed solely to verify the equivalent static analysis method. In addition, an onsite inspection of Seismic Category I piping was performed which resulted in the installation of additional supports.

In general, modifications or additions to piping systems at Ginna Station since initial operation have been seismically qualified using dynamic analyses. Some small piping has been seismically qualified utilizing equivalent static analysis or spacing table techniques.

3.9.2.1.2 Seismic Category I Piping, 2-1/2 Inch Nominal Size and Larger

3.9.2.1.2.1 Static Analysis

This group of Seismic Category I pipes was originally analyzed (*Reference 4*) by dividing each pipe run into lumped masses. The number of masses lumped between any two supports was based upon the spacing interval and increased with the length of the spacing interval. Every mass was given an acceleration equal to the maximum response from the response curve with 0.5% of critical damping, i.e., 0.8g for 0.2g ground acceleration. Each piping system, with its supports, was modeled as a three-dimensional frame and the loads given by the mass times the acceleration were applied at each lumped mass along three directions, two horizontal and one vertical, separately. The moments and torque for each of the three loading directions were then obtained by stiffness analysis. The stresses were calculated at critical points in the piping and its supports for each loading direction. The stresses in the piping were found by using the USAS B31.1 formula

$$S = \left[\frac{Mx^2 + My^2 + Mz^2}{z^2}\right]^{1/2}$$

(Equation 3.9-4)

where

S =	stress
Mx, My, Mz =	moments about the two horizontal directions and the vertical direction
Z =	section modulus

At each point the stresses obtained for the two horizontal earthquakes were compared and the one giving the larger value was then combined with the stress obtained for the vertical loading by direct addition. The maximum stresses imposed by the normal loads plus the loads associated with the larger of the two earthquakes (0.8g) were below 1.2S, where S is taken from the power piping code, USAS B31.1.1.0-1967, Paragraph 119.6.4. If the combination of normal loads and no-loss-of-function earthquake loads is considered as a faulted condition, the allowable membrane and bending stresses could be chosen to be the stresses corresponding to 20% and 40% of the material uniform strain at temperature, respectively. This would give more than a factor of 2 margin between the allowable and the maximum actual stresses.

3.9.2.1.2.2 Dynamic Analysis

In order to increase the confidence in the adequacy of the seismic design of this group of Seismic Category I piping, two pipe runs were selected and analyzed employing modal and response spectra methods. These pipe runs were (1) the residual heat removal system line from the reactor coolant system loop A to the containment penetration, and (2) the main steam line from steam generator B to the containment penetration.

Dynamic analyses were also performed for sections of the above pipe runs and the charging line as a result of IE Bulletin 79-07. These analyses were based on the as-built piping system isometrics and support information.

The defined piping/support systems which were analyzed were evaluated incorporating three-dimensional static and dynamic models which included the effects of the supports, valves, and equipment. The static and dynamic analysis employed the displacement method, lumped parameters, and stiffness matrix formulation and assumed that all components and piping behaved in a linear elastic manner. The response spectra modal analysis technique was used to analyze the piping. The 0.5% Housner ground response spectrum was employed with zero period acceleration values of 0.08g and 0.2g for the operating-basis earthquake and safe shutdown earthquake, respectively. The stress intensification factors due to welds were included in the reanalysis.

3.9.2.1.2.3 Residual Heat Removal System Line From Reactor Coolant System Loop A to Containment

Original dynamic analysis

In the original dynamic analysis the residual heat removal system line was "mathematically" located at the elevation of the steam line on the containment. The reason for this was to investigate the effect of response spectrum distortion, as a function of location and elevation, on the pipe loading and associated stresses.

This pipe run with a 10-in. nominal diameter was selected because it was judged typical of a large portion of Seismic Category I piping with a diameter ranging from 6 in. to 14 in.

Idealized lumped mass models were developed and analyzed dynamically. The analysis was made by assigning three translational and three rotational degrees of freedom to each lumped mass point with each mass point representing a geometrically proportional amount of the total system mass. Elastic characteristics of the system included the translational and rotational stiffnesses. The rotational elastic characteristics were carried into the reduced stiffness matrix that was inverted and formed with the mass matrix, the dynamic matrix.

Following normal mode theory, the natural frequencies, mode shapes, and participation factors were computed to yield the dynamic system characteristics. These characteristics were then combined with the appropriate shock spectra to yield the D'Alembert reverse effective forces on the system for each mode. The modal forces were then used to compute the stresses per mode. The stresses were summed on a root mean square basis for final comparison to code allowable stresses. More than 70 modes were analyzed for their response to earthquake excitation. The Housner 0.5% critical damping ground response spectrum normalized to 0.2g was used. This spectrum was considered adequate because of the location of this pipe run low in the containment.

For the location of maximum stress, the stress values were calculated at three points on the pipe cross-section: the bottom, one side 90 degrees away, and half way between these two.

First the stresses due to the two bending moments and one torsional moment on the pipe were calculated. Then for each of the three points, the root mean square of the stresses acting at the point for the significant modes (first three) was calculated. To this was added the dead weight stress, and then the result was multiplied by the stress intensification factor, as the location of maximum stress was the end of an elbow. The pressure stress was added to this result in order to obtain the total additive longitudinal stress. The total maximum stress was calculated, considering the torsional shear stress and using the formula for maximum principal stresses.

The maximum principal stresses were close to the 1.2S values. They were well below the values corresponding to 20% or 40% of uniform strain. It was concluded that the residual heat removal system line located in the containment at the steam line elevation is not overstressed.

IE Bulletin 79-07 Reanalysis

For the IE Bulletin 79-07 reanalysis, the line analyzed was the residual heat removal system line from the anchor near reactor coolant loop A to the containment penetration.

Table 3.9-2 is a comparison of stress results for the original model, and the model reflecting as-built conditions. The reanalysis considered both as-built conditions and support stiffness. The stress results reported were obtained using B31.1-1973 Summer Addenda, Formula 12. Stress allowables given are based on the stress limits given in Table 3.9-1. The line was found to be seismically qualified.

3.9.2.1.2.4 Steam Line From Steam Generator B to Containment

Original Dynamic Analysis

A dynamic modal analysis was originally run on the steam line of loop B. The ground response spectrum was modified to factor in building effects. It was found that the previous static analysis of the steam line that used the peak of the response curve for 0.5% critical damping gave a very conservative estimate of inertially induced stresses. In order to account for the relative support movements, a separate stress analysis was run on the piping system. This analysis indicated a stress of 8500 psi, which was combined with the maximum thermal stress in the steam line of 11,000 psi. These combined secondary stresses are below the allowable stress of 20,600 psi.

IE Bulletin 79-07 Reanalysis

For the IE Bulletin 79-07 reanalysis, the line analyzed extended from steam generator 1B to the containment penetration. Seismic results were originally reported in *Reference 4*. A seismic reanalysis of this line was performed using the Westinghouse proprietary computer code WESTDYN.

The WESTDYN dynamic model reflected the as-built conditions as well as the actual support stiffness. The main steam line analyzed was coupled to a reactor coolant loop B model. In Table 3.9-3 is a comparison of stress results from the reanalysis reflecting as-built conditions, support stiffness, and the allowable stresses. The stress results reported were obtained using

B31.1-1973 Summer Addenda, Formula 12. Stress allowables given are based on the stress limits given in Table 3.9-1. The line was found to be qualified seismically.

3.9.2.1.2.5 Charging Line

IE Bulletin 79-07 Reanalysis

For the IE Bulletin 79-07 reanalysis, the lines analyzed extended from charging pumps 1, 2, and 3 to the charging pump discharge filter; and included the 2- and 3-in. discharge lines from the filter and the 3-in. bypass. A seismic analysis was originally performed of this line by the M. W. Kellogg Company. A seismic reanalysis of this line was performed using the Westinghouse proprietary computer code WESTDYN.

The WESTDYN dynamic model reflected the as-built conditions as well as the actual support stiffness. Table 3.9-4 is a comparison of stress results from the reanalysis reflecting as-built conditions, support stiffness, and the allowable stresses. The stress results reported were obtained using B31.1-1973 Summer Addenda, Formula 12. Stress allowables given were based on the stress limits given in Table 3.9-1. The line was found to be seismically qualified.

3.9.2.1.3 Seismic Category I Piping, 2-Inch Nominal Size and Under, Original Design

The pipes falling in this category were field erected (*Reference 4*). The large majority of these pipes has lateral and vertical support spacing selected in accordance with that suggested by USAS B31.1 for vertical supports. The piping so supported can be considered rigid with respect to the buildings in which they are housed. The pipes are subjected to the building acceleration only at the points of support without any further appreciable amplification. Conservative calculations show that the largest building amplification of ground acceleration is about 4. This gives inertial loads of 0.8g.

Simple beam calculations performed for the three pipe sizes falling in this category (i.e., 2 in., 1 in., and 3/4 in.) and for the typical schedules adopted for these pipes (i.e., Schedules 10, 40, 80, and 160 for stainless steel pipes and Schedules 40, 80, and 160 for carbon steel pipes) indicated that the stress levels were significantly lower than the allowable values.

3.9.2.1.4 Pressurizer Safety and Relief Valve Discharge Piping

3.9.2.1.4.1 1972 Analysis

In response to a request from the NRC for additional information in 1972 (*Reference 5*), dynamic analyses were performed for the pressurizer safety valve discharge piping.

The pressurizer safety valve piping system is a closed system and no sustained reaction force from a free discharging jet of fluid exists. Transient hydraulic loads can be imposed at various points of the piping system from the time a safety relief line begins to open until steady flow is completely developed. Calculations were performed (*Reference 22*) to provide a time-history of such loads acting on each straight leg of pipe from the safety valve downstream to the relief tank header. The FLASH IV digital computer program was employed in performing these calculations. Frictional losses were included for the piping and the associated elbows. The time-history hydraulic forces were determined based on several loop seal temperatures.

The natural frequencies and mode shapes of the system were solved using program WEST-DYN. The calculated loop seal temperature for Ginna Station with a 3-in.-thick insulated water loop was 330°F. The hydraulic forces assuming a 300°F water temperature were applied to the structural dynamic model at each change in flow direction throughout the system. This constituted a truly impulsive dynamic analysis with simultaneous contributions from all the dynamic modes of the system.

The piping systems for PCV 434 and PCV 435, were represented by lumped mass models as shown in Figures 3.9-3 and 3.9-4. The time-history analysis was performed by the mode superposition method using computer programs WESTDYN, FIXFM, and WESTDYN-2. The stresses from the deadweight, pressure, seismic, and transient hydraulic load analyses were calculated separately. It was conservatively assumed that the maximum stress around the pipe circumference occurs at the same point for all load cases considered. These stresses were added absolutely and compared with the code allowable stress limit of 1.2 x S_a, where S_a = stress allowable. A review of the analysis showed that the stress levels in the pressurizer safety valve Class 1 and Class 2 piping systems were within the allowable design requirements of USAS B31.1.

3.9.2.1.4.2 NUREG 0737, Item II.D.1 Analysis

Under NUREG 0737, Item II.D.1, it was requested that the functionability and structural integrity of the as-built pressurizer safety and relief valve discharge piping system be demonstrated on a plant-specific basis. In response to the NRC request Westinghouse performed (*Reference 2*) an analysis of the pressurizer safety and relief valve discharge piping system. Additional information was supplied in *References 23*, *24*, and *25*.

A water seal is maintained upstream of the pressurizer safety valves. The water slug, driven by high pressure steam upon actuation of the valves, generates severe hydraulic shock loads on the piping and supports. The pressurizer safety valves and Pressurizer Power Operated Relief Valves (PORV) are provided with a reflective insulation system that adds pressurizer radiant heat to the loop seal piping. This maintains the safety valve water seals at elevated temperatures such that the loop seal contents exiting the valve nozzles are converted to steam, which reduces the loads on the piping and supports.

NUREG 0737, Item II.D.1, required testing to qualify the reactor coolant system and safety valves under effected operating conditions and transients. When the pressurizer pressure reaches the safety valve set pressure of 2500 psia and the valve opens, the high-pressure steam in the pressurizer forces the water in the water loop seal through the valve and down the piping system to the pressurizer relief tank. Additionally, when the relief valve set pressure of 2350 psia is reached and the valve opens, high-pressure steam is discharged to the downstream piping.

The computer code ITCHVALVE was used to perform the transient hydraulic analysis for the system (*Reference 2*). One-dimensional fluid flow calculations applying both the implicit and explicit characteristic methods were performed. The piping network was input as a series of single pipes, generally joined together at one or more places by two- or three-way junctions.

Each of the single pipes included associated friction factors, angles of elevation, and flow areas.

Unbalanced forces were calculated for each straight segment of pipe from the pressurizer to the relief tank using program FORFUN. The time-histories of these forces were used for the subsequent structural analysis of the pressurizer safety and relief lines.

The safety and relief lines were modeled statically and dynamically. The mathematical model used for dynamic analyses was modified for the valve thrust analysis to represent the safety and relief valve discharge. The time-history hydraulic forces determined by FORFUN were applied to the piping system lump mass points. The dynamic solution for the valve thrust was obtained by using a modified predictor-corrector-integration technique and normal mode theory.

The piping between the pressurizer nozzles and the pressurizer relief tank was analyzed according to the requirements of the appropriate equations of the ANSI B31.1-1973 Code through the 1973 addenda. The allowable stresses for use with the equations were determined in accordance with the requirements of the ANSI Code. The load combinations and acceptance criteria defined in Tables 3.9-5, 3.9-6, and 3.9-7 were used in the analysis.

The piping stress analysis considered all pertinent loadings that result from thermal expansion, pressure, weight, earthquake, and transient hydraulic effects.

The transfer matrix method and stiffness matrix method were used to obtain a piping deflection solution. All static and dynamic analyses were performed using the WESTDYN computer program. It was determined that the operability and structural integrity of the system were ensured for all applicable loadings and load combinations including all pertinent safety and relief valve discharge cases.

3.9.2.1.5 Main Steam Header Dynamic Load Factor Analysis

In response to a request from the NRC for additional information in 1972 (*Reference 5*), dynamic analysis was performed for the main steam header.

In the original design of Ginna Station, the main steam header (case 2) was analyzed for the internal loads generated by the safety valve during the relieving process by modeling the system as a single degree of freedom system and using a conservative dynamic load factor of 2.0 to account for the impact effects of the safety relief valve reaction. The magnitude of the thrust was based on the combined effects of static pressure at the safety valve discharge system and the momentum of the flowing steam. This analysis indicated that, for the Ginna Station main steam header, the maximum upper bound load factors were 1.15 and 1.50 for a single and multiple valve discharge, respectively. In calculating the dynamic load factor, the analysis accounted for the contri-butions to the piping response given by all the significant vibrational modes of the structure for a single valve and multiple valve discharge. The report concluded that the valve/header design was conservative based on a calculation of the actual dynamic upper bound values of the dynamic load factor. The effects of multiple safety valve discharges should be considered since the analysis showed a possible 30% increase in load factor due to actuation of a second valve. The actual load factor achieved in the system was expected to be significantly lower than the upper bound values predicted since damping reduced the maximum contribution from each mode; and for multiple valve discharge the time between valve discharges had to be exactly equal to a period of one of the primary modes for the maximum response to occur.

3.9.2.1.5.1 Extended Power Uprate Considerations

Additional analysis was developed in support of *Reference 31* to consider potential hydraulic transients that may be developed as a result of the Ginna Extended Power Uprate.

3.9.2.1.6 Secondary System Water Hammer

3.9.2.1.6.1 Analysis

In response to an NRC request regarding secondary system fluid flow instabilities (water hammer), RG&E performed an analysis of the potential for occurrence and potential consequences of water hammer at Ginna Station (*Reference 3*). Analyses of the main feedwater piping were performed for postulated water hammer utilizing a dynamic forcing function. These analyses assumed that a steam-water slugging process was initiated at the steam generators, that the steam generator level was being recovered utilizing auxiliary feedwater, and that the main feedwater check valves were closed. The analyses were based on the piping configuration and supports installed at Ginna Station at the time of analyses.

An examination was made of the normal, abnormal, and accident transients which could result in a steam generator water level below the feed ring long enough for it to drain; and which would result in feedwater flow being initiated in order to recover level. It was determined that the following operating occurrences could cause these conditions:

a. Load changes when the steam generator level was under manual control.
b. Intermittent manual operation of auxiliary feedwater pumps to maintain steam generator level during MODE 3 (Hot Shutdown).
c. Loss of main feedwater.

The main feedwater piping at Ginna Station consists of two lines, A and B, which run from the control valve station in the turbine building to the steam generators.

The auxiliary feedwater piping at Ginna Station consists of six lines: two from the motor-driven auxiliary feedwater pumps (MDAFW) 1A and 1B, two from the turbine-driven auxiliary feedwater pump (TDAFW), and two from the standby auxiliary feedwater pumps (SAFW).

The forcing function used for the analyses is shown in Figure 3.9-1. The forcing function is a time-dependent mathematical quantity representative of the energy released by water hammer in the feedwater piping connected to PWR steam generators. The forcing function provides a time-history of the pressure in the piping system which results from the acoustic shock wave generated by a steam-water slug. The forcing function shown in Figure 3.9-1 was modified for the specific piping configuration at Ginna.

This forcing function was derived by Westinghouse from measurements of pressure and displacement observed during a water hammer test at the Tihange site in Belgium. Calculations performed by Westinghouse employing this forcing function for the Tihange feedwater piping resulted in displacements in fair agreement with those observed. Westinghouse considered the forcing function as preliminary and it was still under development at the time the analyses were performed.

The loading combinations and stress criteria used in evaluating the results of the analyses were based on the original construction code, ANSI B31.1, Power Piping. These criteria were that the sum of the longitudinal stresses due to pressure, weight, and water hammer would not exceed 1.2 times the allowable stress in the hot condition, S_h.

3.9.2.1.6.2 Evaluation Results

Evaluation of the stresses obtained in the analyses showed that inside the containment there were several locations on the A main feedwater piping and several locations on the B main feedwater piping which exceeded the stress criteria. Outside the containment there were no locations on the A main feedwater piping and several locations on the B main feedwater piping which exceeded the stress criteria. Analyses were not performed for the auxiliary feedwater piping systems for a postulated water hammer from the steam generators.

3.9.2.1.6.3 Corrective Actions

Various administrative controls, steam generator mechanical modifications, and piping support modifications were evaluated to determine their effectiveness in either preventing the occurrence of water hammer, or reducing its consequences should it occur. In evaluating these changes, the effect of other changes that were being made to the plant and the overall reliability and integrity of the steam generators were also considered.

It was determined that the best alternative available for precluding water hammer was installation of J-shaped discharge tubes on top of the feed rings and plugging of the bottom holes in the rings to provide for top discharge of water rather than bottom discharge. See Section 10.3.2.2.

In 1996, Ginna Station replaced the steam generators. The replacement steam generators incorporated many of the guidelines from NRC Branch Technical Position ASB-10-2, “Design Guidelines for Avoiding Water Hammers in Steam Generators,” to minimize the potential and consequence of waterhammer in the feedwater system. Specifically, the BWI replacement steam generators are designed to minimize the potential for a steam pocket forming in the feed header using top discharge J-tubes in the feed ring, internals which maximize secondary water inventory above the feed ring, and an all-welded thermal sleeve/internal feed header assembly that eliminates the possibility of steam leakage into the feed ring through sleeve/header mechanical joints. The BWI design is also less prone to serious consequences from a steam pocket forming because of the feed header gooseneck which tends to retard rapid condensation and water-slug acceleration better than a horizontal header run would.

3.9.2.1.6.4 Extended Power Uprate Considerations

Additional analysis was developed in support of Reference 31 to consider potential hydraulic transients that may be developed as a result of the Ginna Extended Power Uprate.

3.9.2.1.7 Velan Swing Check Valves

In response to IE Bulletin 79-04, RG&E analyzed the effect of changes in weights previously assumed for swing check valves manufactured by Velan Engineering Corporation. There is one 6-in. Velan swing check valve installed in both low head safety injection system lines and four 3-in. valves installed in the high head safety injection system lines. The initial installation assumed a weight of 225 lb for the 6-in. valves and 60 lb for the 3-in. valves. The correct weights were 450 and 95 lb, respectively.

In order to investigate the effect of valve weight differences, Westinghouse performed seismic analyses on some representative configurations of the safety injection system and studied the effect of increasing valve weight by 100% on the pipe stresses and support loads of the line.

An operating-basis earthquake seismic analysis was performed for each case. It was a two-dimensional response spectrum analysis considering each horizontal direction separately, combined with the vertical direction. It was determined from the analysis that the increase in valve weight did not result in unacceptable pipe stress for the lines investigated.

3.9.2.1.8 Seismic Piping Upgrade Program

As a result of SEP preliminary seismic review of Ginna (SEP Topic III-6), the NRC IE Bulletin 79-14, and other NRC seismic requirements, RG&E initiated a seismic piping upgrade program described in Section 3.7.3.7. In order to conservatively respond to the SEP seismic review and possible future NRC seismic requirements, a set of analysis procedures and criteria that conform with current NRC review criteria were used for the piping analysis. These are discussed in Section 3.7.3.7. The loading combinations and associated stress limits used for the piping systems that are part of the seismic upgrading program are given in Table 3.9-8. Pipe rupture loads were not considered; as such, the stress limits used for the safe shutdown earthquake condition did not correspond to the faulted condition, as they could be for the safe shutdown earthquake evaluation, but to the emergency condition stress limits. The piping stresses were calculated using the formulas given in ANSI B31.1-1973, 1973 Summer Addenda. Thermal stresses were evaluated per ANSI B31.1-1973, Summer 1973 Addenda requirements.

The maximum loads that the main feedwater piping and steam line piping were permitted to transmit to the steam generator nozzles are given in Table 3.9-9.

The allowable loads for the seal injection and component cooling system nozzles on the reactor coolant pump and motor are listed in Table 3.9-10.

Two pipe lines from the upgraded piping systems were selected and analyzed independently by the NRC to verify the adequacy of the as-built design and confirm the upgrade analysis results. The pipe lines selected were portions of residual heat removal and safety injection system piping. Audit analyses, which incorporated current ASME Code and Regulatory Guide criteria and used the floor response spectra as input motion, were performed for each portion of the piping system selected. The results from these analyses were compared to ASME Code requirements for Class 2 piping systems at the appropriate service conditions. This comparison provided the bases for assessing the structural adequacy of the piping under the postulated seismic loading condition. Assumptions made for the analysis, methodology employed and analysis results are found in *Reference 6*. The results from the confirmatory analysis showed that the sampled piping systems are capable of withstanding the postulated safe shutdown earthquake seismic input.

Structural members within the various buildings at Ginna Station were analyzed and were modified as required to accept new or recalculated pipe support loads from the seismic piping upgrade program and to transfer these loads into the main structural framing.

Pipe supports were analyzed as discussed in Section 3.9.3.3.

3.9.2.2 Safety-Related Mechanical Equipment

Mechanical equipment was originally seismically qualified by a combination of test and analysis. The methods of analysis used in the original analyses and during the SEP reevaluation are described briefly in Section 3.7.3. The results of the analysis are presented in this section.

3.9.2.2.1 Original Seismic Input and Behavior Criteria

For Seismic Category I mechanical equipment, all components and systems originally classified as Class I were designed in accordance with the criteria described in Section 3.7.1.1. All components of the reactor coolant system and associated systems were designed to the standards of the applicable ASME or USAS Codes. The loading combinations and behavior criteria not otherwise defined by the USAS and ASME Codes in use at the time of the original design, which were employed by Westinghouse in the design of the components of these systems, i.e., vessels, piping, supports, vessel internals and other applicable components, are given in Table 3.9-1. Table 3.9-1 also indicates the stress limits which were used in the design of the equipment for the various loading combinations. In addition, the supports for the reactor coolant system were designed to limit the stresses in the pipes and vessels to the stress limits given in Table 3.9-1.

Heat exchangers were designed in accordance with the criteria set forth in Section 3.7.1.1. The peak of the 0.5% critical damping response spectra corresponding to the 0.2g maximum potential earthquake was selected as the seismic design load. Stress limits were set equivalent to those of the pressure vessel codes and the structural steel standards of AISC.

The design of pumps (casing and shafting) was based not on stress criteria, but on deflection limits. For the case where efficiency was of minimum importance, deflection at the stuffing box controlled the design. For the case where efficiency was of importance, deflection of the shaft at the impeller wear rings controlled the design. In either case, the natural frequency (identical to critical speed) was approximately 20 Hz and 30 Hz for 1800 rpm and 3600 rpm machines, respectively, for flexible shafting. In reality, the stuffing boxes served as an additional bearing and the natural frequency was above that corresponding to the operating speed.

For stiff shafting, the fundamental frequency was above that corresponding to the operating speed (30 Hz and 60 Hz). Both the pump casings and the motor casings were extremely stiff when evaluated as simply supported beams with uniform load distribution. A typical natural frequency for a casing with a length-to-diameter ratio of 3 and a diameter of 36 in. was 100 Hz.

The combined pump-motor unit is mounted on a common bedplate which is grouted into the foundation. The stiffness of the foundation mass and the rigid bolting eliminated possible relative movement between the pump and motor under operating loads as the couping between the motor and pump was designed only to accommodate geometric misalignment.

The analysis of tanks was performed in the manner set forth in TID 7024, taking into account the possible dynamic effects resulting from the sloshing of the water. The techniques are set forth in Chapters 5 and 6 of TID 7024.

Shell stresses and support stresses are limited to those permitted in the pressure vessel codes and the structural steel standards of AISC.

Electric motor-operated valves were verified to be capable of sustaining a 1g shock load without interruption of circuitry or loss of function. This was verified up to 20 Hz.

3.9.2.2.2 Current Seismic Input

Current seismic input requirements for determining the seismic design adequacy of mechanical equipment are normally based on in-structure (floor) response spectra for the elevations at which the equipment is supported. The floor spectra used in the SEP reassessment, which are based on Regulatory Guide 1.60 spectra, are shown in Figures 3.7-12 through 3.7-28.

For mechanical equipment, a composite 7% equipment damping was used in the evaluation for the 0.2g safe shutdown earthquake.

3.9.2.2.3 Systematic Evaluation Program

Seismic Category I components that are designed to remain leaktight or retain structural integrity in the event of a safe shutdown earthquake are typically designed to the ASME Section III Code (ASME III), Class 1, 2, or 3 stress limits for Service Condition D. The stress limits for supports for ASME leaktight components are limited as shown in Appendix F or Appendix XVII to ASME III (1977).

When qualified by analysis, active ASME III components that must perform a mechanical motion to accomplish their safety functions typically must meet ASME III Class 1, 2, or 3 stress limits for Service Condition B. Supports for these components are also typically restricted to Service Condition B limits to ensure elastic low deformation behavior.

For other passive and active equipment, which are not designed to ASME III requirements, and for which the design, material, fabrication, and examination requirements are typically less rigorous than ASME III requirements, the allowable stresses for passive components are limited to yield values and to normal working stress (typically 0.5 to 0.67 yield) for active components.

The current behavior criteria used in various equipment and distribution systems for Ginna passive components are given in Table 3.9-11.

Experience in the design of such pressure retaining components as vessels, pumps, and valves to the ASME III requirements, at 0.2g zero period ground acceleration, indicates that stresses induced by earthquakes seldom exceed 10% of the dead weight and pressure-induced stresses in the component body (*Reference 7*). Therefore, design adequacy of such equipment is seldom dictated by seismic design considerations.

Seismically induced stresses in nonpressurized mechanical equipment and component supports may be significant in determining design adequacy.

3.9.2.2.4 Systematic Evaluation Program Reevaluation of Selected Mechanical Components for Design Adequacy

The Systematic Evaluation Program (SEP) Seismic Review Team selected mechanical and electrical components representative of items installed in the reactor coolant system and safe shutdown systems for review in order to develop conclusions as to the overall seismic design adequacy of Seismic Category I equipment installed at Ginna Station. The electrical equipment is listed in Table 3.10-2 and discussed in Section 3.10.2.1. The mechanical equipment is listed in Table 3.9-12 and the seismic analysis of these components is described in the following sections.

3.9.2.2.4.1 Essential Service Water (SW) Pumps

The essential service water (SW) pump and motor units are oriented vertically in the screen house and supported at elevation 253.5 ft. The intake portion of the pumps extend down from the discharge head and pump base a distance of approximately 36.5 ft, including the clip-on type basket strainer installed on the suction end bell.

The previous seismic analysis was performed for equivalent static loads of 0.32g acting simultaneously in one horizontal and the vertical direction.

The pump-motor units are located at grade; therefore, the seismic input used in SEP reevaluation was essentially the Regulatory Guide 1.60 ground response spectrum for 7% of critical damping. The pumps were evaluated for an inertial acceleration value considering peak response of 0.52g horizontal acceleration and 0.35g vertical acceleration. Overturning tensile and shear stresses in the pump base anchor bolts were determined as were stresses at the attachment of the intake column pipe to the discharge head.

Because the intake portion of the pumps are oriented vertically as cantilever beams, the dynamic characteristic of the intake suction pipes were determined. The intake suction pipes were found to have a fundamental frequency of 1.6 Hz based on a weight distribution that includes water in the shaft. Because of this natural frequency, the spectral acceleration used was the peak of Figure 3.7-4, 0.52g.

It was determined that a brace needed to be installed on the intake column pipes. With the brace, the stresses at the bolts would be 15,700 psi in tension and 7000 psi in shear, which would yield a minimum factor of safety in shear of 2.29 for ASME Condition D stress limits for an assumed A307 bolt material. Also, the stresses calculated at the flange connecting the discharge head to the intake column pipes were well within allowable stresses. This modification was performed in 1984.

3.9.2.2.4.2 Component Cooling Heat Exchanger

The component cooling heat exchanger is a horizontal heat exchanger located in the auxiliary building and supported by two saddles at elevation 281.5 ft. One saddle is slotted in the longitudinal direction to permit thermal expansion. During the SEP reevaluation the previous analysis was reviewed and independent evaluation of the dynamic response characteristics of the heat exchanger and its saddle support system using the response spectra for 7% damping shown in Figure 3.7-21 was performed. The review indicated that the system was relatively rigid and had no response frequencies below 33 Hz. Thus, safe shutdown earthquake input horizontal seismic accelerations in the orthogonal directions used were 0.36g and 0.60g. The seismic stresses induced in the tubes and shell were determined, combined with other applicable loads, and compared to code allowables. The safety factor determined for the heat exchanger tube is 33.9 and that for the shell is 11.0.

Both the component cooling heat exchanger and the component cooling surge tank are supported by a complex structural steel framework. Evaluation of the fundamental frequencies of both the heat exchanger and the surge tank did not consider any flexibility of the structural steel support framing. It was assumed that the dynamic characteristics of this structural steel framing were included in the response spectra.

The anchor bolt stresses were also determined. The analysis established a factor of safety with respect to ASME Code-allowable stress limits of 1.41 for the anchor bolts. Therefore, it was concluded that the component cooling heat exchanger will withstand a 0.2g safe shutdown earthquake without loss of structural integrity.

3.9.2.2.4.3 Component Cooling Surge Tank

The component cooling surge tank is a horizontal component located in the auxiliary building and supported by two saddles at elevation 281.5 ft. For the SEP reevaluation the previous analysis was reviewed. In addition, independent evaluation of the structural characteristics of the surge tank and its support system using the response spectra for 7% damping shown in Figure 3.7-23 was performed. In the transverse (east-west) direction, the tank-support system was found to be rigid. However, it was determined that it was not completely anchored against sliding. As a result, the tank saddle supports were modified to provide restraint in the longitudinal direction.

The seismic forces in the transverse (east-west) direction developed from a 0.75g in-structural spectral acceleration were applied to the surge tank and the resulting tank, saddle, and anchor bolt stresses were determined. Factors of safety for the tank, saddle, and anchor bolts--loaded seismically in the transverse and vertical directions--were 125.5, 57.7, and 5.08, respectively.

3.9.2.2.4.4 Diesel-Generator Air Tanks

The diesel-generator air tanks are oriented vertically in the diesel-generator building and supported at grade elevation in a rock-supported structure.

The seismic input used for the SEP reevaluation was the Regulatory Guide 1.60 ground response spectrum for 7% of critical damping (Figure 3.7-4). The previous analysis to seismically qualify the tanks used a 0.2g safe shutdown earthquake ground response spectrum. The tanks are supported by a skirt structure and the combined tank-support system was found to have a fundamental frequency of 33 Hz. Therefore, the input acceleration used was 0.2g. The maximum calculated stress in the anchor bolts was approximately 0.28 ksi in shear, which yields a safety factor of 61.3 for A307 bolt material. The minimum safety factors in the tank body and skirt support were 4.43 and 3968, respectively.

3.9.2.2.4.5 Boric Acid Storage Tank

The boric acid storage tank is a column-supported tank. The tank, its support legs, and its anchors were reviewed to determine seismic design adequacy. The tank, which is supported at elevation 271 ft, was evaluated using the in-structure response spectra shown in Figure 3.7-24. The dynamic analysis considered the effective impulsive and convective response of the contained fluid. The fundamental response frequencies for the tank were calculated to be 17.2 Hz for tank-support system bending and shear deformation under impulsive loading (7% damping) and 0.56 Hz under convective loading (0.5% damping). The analysis established minimum factors of safety of approximately 41.7 for membrane stress in the tank, 6.20 for compressive stresses in the tank legs, and 4.65 for compressive stresses in the anchor bolts.

3.9.2.2.4.6 Refueling Water Storage Tank (RWST)

The refueling water storage tank (RWST) is a vertical vessel that is 81 ft high to the top of the cylindrical portion and 26.5 ft in diameter. The anchorage consists of thirty, 2.5-in. diameter A36 bolts. The tank was originally qualified according to TID 7024 assuming a safe shutdown earthquake ground acceleration of 0.2g (without vertical amplification) and assuming that it was supported at the ground floor (elevation 236 ft) of the auxiliary building.

In 1983, RG&E investigated the ability of the refueling water storage tank (RWST) to withstand dead weight and seismic forces (*Reference 8*). Analysis loads consisted of the dead weight of the tank and contents, and seismic loads in two horizontal and the vertical directions. The seismic loads were defined by the site specific ground response spectrum for R. E. Ginna as specified by Regulatory Guide 1.60. The full spectrum was used for the horizontal analysis. Two thirds of the full spectrum was used for the vertical analysis.

The dynamic response analysis followed the requirements of NUREG/CR-1161. Analysis of the convective (sloshing) horizontal response was performed using the conventional "rigid tank" assumptions. Tank flexibility and fluid-structure interaction was incorporated in the analysis of the impulsive (non-sloshing) horizontal response. Tank flexibility was incorporated in the vertical response analysis. A damping level of 0.5% was used for the convective horizontal response analysis. A 7% damping was used for the impulsive horizontal and vertical response analysis.

The acceptance criteria considered the following principal points:

a. Anchorage Stresses: These include the stresses in the bolts, brackets, and bracket welds. Allowables were calculated per ASME Section III, Subarticle NF 3300.

b. Tank Wall Material Stress: The axial, hoop, and shear stresses developed in the tank wall were compared to material allowables per ASME Section III, Subarticle NC 3800.

c. Tank Wall Buckling: The axial, hoop, and shear stresses developed in the tank wall were compared to experimentally derived buckling criteria.

The results of the analysis indicated that no modifications to the refueling water storage tank (RWST) were required and that the tank was capable of withstanding dead weight loads in combination with the (SEP) site specific postulated seismic event.

In 1992, RG&E responded to Generic Letter 87-02, Supplement 1 and Generic Letter 88-20, Supplement 4 (SQUG and seismic events issues). As part of this response, RG&E stated that a review of the RWST would be performed for response spectra based on a peak ground acceleration of 0.2g and a Regulatory Guide 1.60 shape.

As a result of subsequent seismic analysis, modifications were determined to be required. The modifications consisted of 16 equally spaced vertical stiffeners, a welded steel support skirt extending 360° around the tank at the operating floor of the auxiliary building, and a large number of 3" diameter pins set through the skirt and into the concrete floor. As a result of these modifications which were completed in 1996, the RWST is capable of resisting the higher seismic input loads associated with 0.2g peak ground acceleration.

3.9.2.2.4.7 Motor-Operated Valves

During the SEP reevaluation, calculations performed on randomly selected motor-operated valves (2-in., 3-in., and 4-in. diameter) in the Ginna plant demonstrated that stress levels were in excess of the guideline value of 10% stress levels of ASME III, Class 2, Condition B for active valves and Condition D when pressure boundary integrity was required.

It was recommended that RG&E evaluate the seismic stresses induced by motoroperated valves in supporting pipe that is 4 in. in diameter and smaller and show that stresses resulting from motor operator eccentricity are less than 10% of the service Condition B code-allowable stresses. Rochester Gas and Electric explicitly modeled motor-operated valves in the as-built installation as part of the Seismic Piping Upgrade Program and either found the stresses to be acceptable or modified the supports. The Seismic Piping Upgrade Program is discussed in Sections 3.7.3.7 and 3.9.2.1.8.

Additionally, in accordance with the motor-operated valve program, as described in the Ginna Station Motor-Operated Valve Qualification Program Plan, the impact of design basis seismic events is evaluated and identified for susceptible components of each motor-operated valve under the requirements of NRC Generic Letter 89-10. (See Section 5.4.9.3.)

3.9.2.2.4.8 Steam Generators

In 1975, a generic stress report was written which contained updated analyses of most areas of the steam generator that are subject to external loads, i.e., primary nozzles, feedwater nozzle, steam nozzle, and lower support pads. The updated stress report also contained an analysis of the tubes, swirl vanes, and feedwater ring. Calculated stress intensities were compared with the ASME III design condition allowable levels for an operating-basis earthquake and the emergency condition allowable levels for a safe shutdown earthquake.

A detailed seismic analysis was not performed during the SEP reevaluation, but a comparison of the seismic input used in the original design of Ginna Station with that determined from the in-structure response spectra was used as a criterion for qualification.

Since the fundamental frequency of the steam generator was found to be below 10 Hz, the peak acceleration in both the north-south and east-west directions is 0.60g (see Figures 3.7-15 through 3.7-18) and the square root of the sum of the squares value for two horizontal components is 0.85g. Since the original horizontal response spectra used for the design of the steam generator had a minimum spectral acceleration of 2.0g for the safe shutdown earthquake condition, the seismic stresses resulting from use of the Ginna reassessment response spectra would be less than the stress values from the original analysis. The steam generator components were determined adequate by the 1975 analysis.

In 1996, the steam generators were replaced. Seismic evaluation of the primary and secondary side pressure boundaries demonstrate that these components satisfy ASME III Class 1 design requirements for Service Levels A, B, C and D.

3.9.2.2.4.9 Reactor Coolant Pumps

In the original design of Ginna Station, a static seismic load stress analysis was performed for the pumps. The safe shutdown earthquake analysis used 0.8g horizontally and 0.54g vertically. The stresses and deformations resulting from these loads were then combined with the dead weight and other normal operating loads to determine the total stresses in the motor, support stand cylinder, flange welds, support stand bolts, and main flange bolts. This analysis also contained evaluations of the pump support feet, primary nozzles, and casing for seismic plus normal operating loads. The stresses calculated in these analyses were compared with ASME III allowables.

A detailed seismic analysis was not performed for the SEP reevaluation. Instead, a comparison of the input acceleration with that used in the earlier analysis was used to check the adequacy of the reactor coolant pump.

For the SEP reevaluation, in-structure response spectra for the reactor coolant pump given in Figures 3.7-19 and 3.7-20 were used. For the peak spectral acceleration of 0.55g for both the north-south and east-west directions, the square root of the sum of the squares value was 0.78g, and the ratio of this value to the original design value of 0.8g was 0.97. The pump input acceleration was less than that considered in the 1968 analysis and therefore the pumps were considered adequate based on the original generic analysis.

3.9.2.2.4.10 Pressurizer

The pressurizer is a vertical cylindrical vessel with a skirt type support attached to the lower head. The lower part of the skirt terminates in a bolting flange where 24 1.5-in. bolts secure the vessel to its foundation. In 1969, a generic seismic analysis of the pressurizer shell, support skirt, support skirt flange, and pressurizer support bolts was performed. The weight of the largest pressurizer (1800 ft^3) was used instead of the actual operating weight of the Ginna pressurizer (800 ft^3). In the safe shutdown earthquake evaluation, accelerations were applied statically at the center of gravity of the 1800 ft^3 model: 0.48g in the horizontal direction and 0.32g in the vertical direction. ASME III upset condition allowable levels were used for safe shutdown earthquake load cases.

In 1973, a more detailed evaluation was performed of the pressurizer skirt and shell (*Reference 9*). For that evaluation the loads applied to the skirt were equivalent to 10 times the operating-basis earthquake loads and 14 times the safe shutdown earthquake loads used in the 1969 evaluation. The results contained the primary membrane and bending stresses.

The pressurizer heaters were qualified generically for the 51 Series Pressurizer (*Reference 9*). The heaters in the 800-ft^3pressurizer are shorter than those qualified but are otherwise identical. The qualification procedure used an equivalent static load of 37.5g for the safe shutdown earthquake condition. The fundamental frequency of the heater rods was found to be greater than 33 Hz.

The in-structure response spectra were used in the SEP reevaluation of the pressurizer as shown in Figure 3.7-12. Since the fundamental frequency of the pressurizer may be as low as 3 Hz, peak spectral accelerations were used: 0.55g for the north-south direction and 0.60g for the east-west direction. The square root of the sum of the squares value is 0.81g, and the ratio of this value to the original design value of 0.48g is 1.7. Based on the primary stress resultants of the 1973 analysis, the seismic input of 0.81g is well within the design limits presented in *Reference 9*.

3.9.2.2.4.11 Control Rod Drive Mechanism

The response spectra for the SEP reevaluation of the control rod drive mechanisms are given in Figures 3.7-13 and 3.7-14. Assuming the fundamental frequency of the drive mechanism as less than 12.5 Hz, the peak spectral acceleration in both the north-south and east-west directions was 0.60g and the square root of the sum of the squares value was 0.85g and this square root of the sum of the squares value is greater than the design value of 0.8g used in the original analysis. As noted in the NRC safety evaluation report on SEP Topic III-6 (*Reference 10*) the Westinghouse analysis was found to have utilized correct loadings and that the stresses are well within acceptable levels.

3.9.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

Sections 3.9.2.3.1 through 3.9.2.3.5 reflect information resulting from the original analyses of the Ginna Station reactor vessel internals under dynamic loading conditions. It is preserved here for historical information. In anticipation of Extended Power Uprate (EPU), the dynamic response of the internals was reanalyzed (*Reference 31*). This reanalysis incorporated leak-before-break technology as allowed by 1972 General Design Criteria GDC-4. Consequently, double-ended RCS breaks could be removed from the design basis for the reactor vessel internals (*Reference 32*). This reanalysis is discussed further in Section 3.9.2.3.6.

3.9.2.3.1 Design Criteria

3.9.2.3.1.1 General

The criteria for acceptability is that the core should be coolable and intact following a pipe rupture up to and including a double-ended rupture of the reactor coolant system. This implies that core cooling and adequate core shutdown must be ensured. Consequently, the limitations established on the internals are concerned principally with the maximum allowable deflections and/or stability of the parts.

3.9.2.3.1.2 Critical Internals

Upper Barrel

The upper barrel deformation has the following limits. To ensure reactor trip and to avoid disturbing the rod cluster control assembly guide structure, the barrel should not interfere with any guide tubes. This condition requires a stability check to assure that the barrel will not buckle under the accident loads. The minimum distance between guide tube and barrel is 10 in. This figure is adopted as the limit beyond which proper function can no longer be guaranteed. An allowable deflection of 5 in. has been selected.

Rod Cluster Control Assembly Guide Tubes

The rod cluster control assembly guide tubes in the upper core support package has the following allowable limits. The maximum horizontal transient deflection as a beam shall not exceed 1 in. over the length of the guide tube. The no loss of function limit is 1.5 in. Tests on guide tubes show that when the transverse deflection of the guide tube becomes significant, the cross section of the rod cluster control assembly guide tube changes. A maximum allowable transient transverse deflection of 1.0 in. has been established for the blow-down accident. Beam deflections above these limits produce cross section changes with increasing delay in scram time until the control rod will not scram due to interference between the rods and the guide. With a maximum transient transverse deflection of 1.5 in., the cross section distortion will not exceed 0.072 in. after load removal. This cross section distortion allows control rod insertion. For a maximum transient transverse deflection of 1.0 in., a cross section distortion not in excess of 0.035 in. is anticipated.

Fuel Assemblies

The limitations for this case are related to the stability of the thimbles at the upper end. During the accident, the fuel assembly will have a vertical displacement and could impact the upper and lower packages subjecting the components to dynamic stresses.

The upper end of the thimbles shall not experience stresses above the buckling compressive stresses because any buckling of the upper end of the thimbles will distort the guide lines and could affect the fall of the control rod.

Upper Package

The maximum allowable local deformation of the upper core plate where a guide tube is located is 0.100 in. This deformation will cause the plate to contact the guide tube since the clearance between plate and guide tube is 0.100 in. This limit will prevent the guide tubes from being put in compression. In order to maintain the straightness of the guide tube a maximum allowable total deflection of 1 in. for the upper support plate and deep beam has been established. The corresponding no loss of function deflection is above 2 in.

3.9.2.3.1.3 Allowable Stress Criteria

The allowable stress criteria fall into two categories dependent upon the nature of the stress state: membrane or bending. A direct state of stress (membrane) has a uniform stress distribution over the cross section. The allowable (maximum) membrane or direct stress is taken to be equal to the stress corresponding to 0.2 of the uniform material strain or the yield strength, whichever is higher. For unirradiated 304 stainless steel at operating temperature the stress corresponding to 20% of the uniform strain is:

(S_m) allowable = 39,500 psi

For irradiated materials, the limit stress is higher.

For a bending state of stress, the strain is linearly distributed over a cross-section. The average strain value is, therefore, one half of the outer fiber strain where the stress is a maximum. Thus, by requiring the average strain to satisfy an allowable criterion similar to that for the direct state of stress, the outer fiber strain may be 0.4 times the uniform strain. The maximum allowable outer fiber bending stress is then taken to be equal to the stress corresponding to 40% of the uniform strain or the yield strength, whichever is higher. For unirradiated 304 stainless steel at operating temperature, we obtain from the stress strain curve:

(S_b) allowable = 50,000 psi

For combinations of membrane and bending stresses, the maximum allowable stress is taken to be equal to the stress corresponding to the maximum outer fiber strain not in excess of 40% uniform strain and average strain not in excess of 20% uniform strain.

3.9.2.3.2 Blowdown and Force Analysis

3.9.2.3.2.1 Computer Program

The MULTIFLEX computer code (*References 11*, *12*) calculates the thermal-hydraulic transient within the RCS and considers subcooled, transition, and early two-phase (saturated) blowdown regimes. The code employs the method of characteristics to solve the conservation laws, assuming one-dimensional flow and a homogeneous liquid-vapor mixture. The RCS is divided into subregions in which each subregion is regarded as an equivalent pipe. A complex network of these equivalent pipes is used to represent the entire primary RCS.

The following operating conditions were considered in establishing the limiting temperatures and pressures for the Ginna Station LOCA hydraulic forces analyses:

- Initial RCS conditions associated with a minimum thermal design flow of 85,100 gpm per loop.
- Uprated core power of 1811 MWt (analyzed NSSS power of 1817 MWt).

- A nominal RCS hot full power (HFP) T_{AVG} range of 564.6°F to 576.0°F. This provides an RCS Tcold range of 528.3°F to 540.2°F.
- An RCS temperature uncertainty of ±4°F.
- A feedwater temperature range of 390.0°F to 435.0°F.
- A nominal RCS pressure of 2250 psia.
- A pressurizer pressure uncertainty of ±60 psi.

Based on these conditions, the LOCA forces were generated at a minimum Tcold of 524.3°F, including uncertainty, and a pressurizer pressure of 2310 psia, including uncertainty.

The hydraulic forcing functions that occur as a result of a postulated LOCA are calculated assuming a limiting break location and break area. The limiting break location and area vary with the RCS component under consideration, but historically the limiting postulated breaks are a limited displacement reactor pressure vessel (RPV) inlet/outlet nozzle break or a double-ended guillotine (DEG) reactor coolant pump (RCP)/steam generator (SG) inlet/outlet nozzle break. General Design Criterion 4 (GDC-4) allows main coolant piping breaks to be "excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping." This exemption is generally referred to as leak-before-break (LBB).

Furthermore, Constellation Generation Group had requested Westinghouse to exempt all the 10-inch piping connections to the RCS from the dynamic analysis of pipe break loads. Therefore, the next limiting RCS break sizes less than 10-inch diameter are the smaller auxiliary (or branch) lines connected to the RCS. The smaller branch line breaks analyzed for hydraulic forces are the 3-inch pressurizer spray line in the cold leg, the 4-inch upper plenum injection nozzle on the vessel, and the 2-inch safety injection line connection to the hot leg. The 4-inch pressurizer safety valve line on top of the pressurizer was not considered for the Forces analysis because the Forces analysis tracks the acoustic wave propagating through the subcooled fluid of the RCS, while the break for the safety valve line would occur in the voided region of the pressurizer. It would, therefore, be non-limiting as compared to breaks modeled in either the cold or hot legs of the RCS.

The only exception to the use of auxiliary line breaks for structural qualification is the modeling of a limited displacement double-ended guillotine reactor vessel outlet nozzle (RVON) break to demonstrate control rod insertion following a LOCA.

3.9.2.3.2.2 Blowdown Model

The MULTIFLEX computer code calculates the thermal-hydraulic transient within the RCS and considers subcooled, transition, and early two-phase (saturated) blowdown regimes. The code employs the method of characteristics to solve the conservation laws, assuming one-dimensional flow and a homogeneous liquid-vapor mixture. The RCS is divided into subregions in which each subregion is regarded as an equivalent pipe. A complex network of these equivalent pipes is used to represent the entire primary RCS.

The reanalysis performed in support of the Extended Power Uprate has made use of the MULTIFLEX computer code. MULTIFLEX is an extension of the BLODWN-2 computer code and includes mechanical structure models and their interactions with the thermal-hydraulic system. Both versions of the MULTIFLEX code share a common hydraulic modeling scheme, with differences being confined to a more realistic downcomer hydraulic network and a more realistic core barrel structural model that accounts for non-linear boundary conditions and vessel motion. Generally, this improved modeling results in lower, more realistic, but still conservative hydraulic forces on the core barrel. The NRC staff has accepted (*Reference 13*) the use of MULTIFLEX 3.0 for calculating the hydraulic forces on reactor vessel internals (*Reference 14*).

A coupled fluid-structure interaction is incorporated into the MULTIFLEX code by accounting for the deflection of the constraining boundaries, which are represented by separate spring-mass oscillator systems. For the reactor vessel/internals analysis, the reactor core barrel is modeled as an equivalent beam with the structural properties of the core barrel in a plane parallel to the broken inlet nozzle. Mass and stiffness matrices that are obtained from an independent modal analysis of the reactor core barrel are applied in the equations of structural vibration at each of the mass point locations. Horizontal forces are then calculated by applying the spatial pressure variation to the wall area at each of the elevations representative of the mass points of the beam model. The resultant core barrel motion is then translated into an equivalent change in flow area in each downcomer annulus flow channel. At every time increment, the code iterates between the hydraulic and structural subroutines of the program at each location confined by a flexible wall. For the reactor pressure vessel and specific vessel internal components, the MULTIFLEX code generates the LOCA pressure transient that is input to the LATFORC and FORCE2 post-processing codes (*Reference 11*). These codes, in turn, are used to calculate the actual forces on the various components.

3.9.2.3.2.3 LATFORC MODEL

The LATFORC computer code employs the field pressures generated by MULTIFLEX code, together with vessel geometric information (component radial and axial lengths), to determine the horizontal forces on the vessel wall and core barrel. The LATFORC code represents the downcomer region with a model that is consistent with the model used in the MULTIFLEX blowdown calculations. The downcomer annulus is subdivided into cylindrical segments, formed by dividing this region into circumferential and axial zones. The results of the MULTIFLEX/LATFORC analysis of the horizontal forces are calculated for the initial 500 msec of the blowdown transient and are stored in a computer file. These forcing functions, combined with vertical LOCA hydraulic forces, seismic, thermal, and flow-induced vibration loads, are used by the cognizant structural groups to determine the resultant mechanical loads on the reactor pressure vessel and vessel internals.

3.9.2.3.2.4 FORCE2 MODEL

The FORCE2 computer code calculates the hydraulic forces that the RCS coolant exerts on the vessel internals in the vertical direction. The FORCE2 code uses a detailed geometric description of the vessel components and the transient pressures, mass velocities, and densities computed by the MULTIFLEX code. The analytical basis for the derivation of the mathematical equations employed in the FORCE2 code is the one-dimensional conservation of linear momentum. Note that the computed vertical forces do not include body forces on the vessel internals, such as deadweight or buoyancy. When the vertical forces on the reactor pressure vessel internals are calculated, pressure differential forces, flow stagnation forces, unrecoverable orifice losses, and friction losses on the individual components are considered. These force components are then summed together, depending upon the significance of each, to yield the total vertical force acting on a given component. The results of the MULTIFLEX/FORCE2 analysis of the vertical forces are calculated for the initial 500 msec of the blowdown transient and are stored in a computer file. These forcing functions, combined with horizontal LOCA hydraulic forces, seismic, thermal, and flow-induced vibration loads, were used in the structural evaluations to determine the resultant mechanical loads on the vessel and vessel internals.

3.9.2.3.3 Fuel Assembly Thimbles

When the core moves vertically it can impact the upper and lower core plates, which subjects the thimbles to compressive impact stresses. These stresses were obtained from the maximum dynamic impact forces on the fuel assemblies. The maximum impact load applied to the thimbles by the fuel elements was 2,132 lbs. The maximum axial stress was 11,660 psi. Buckling stresses result from the impact load of the fuel assembly onto the lower core plate. This load is distributed through the grids to the thimbles as drag force proportional to the drag force available at each grid. The largest fraction of the load is reacted at the bottom grid because the bottom grid is the highest force grid. The spans that would be considered in this event are the lowest spans. However this design has the tube-in-tube dashpost in those spans, which reinforces them. Therefore the critical span becomes the span where the dashpot tube ends, which has a buckling stress of 4,248 psi and an allowable buckling stress of 7,551 psi (for ZIRLO™ with a yield stress of 18,520 psi at operating temperature). Therefore the distortion will not exceed the allowable limits, and it is concluded that the capability of the control rod insertion is maintained.

3.9.2.3.4 Dynamic System Analysis of Reactor Internals Under Loss-of-Coolant Accident (LOCA)

The response of reactor internals components due to an excitation produced by complete severance of a branch line pipe is analyzed. Assuming a pipe break occurs in a very short period of time of 1 msec, the rapid drop of pressure at the break produces a disturbance which propagates along the primary loop and excites the internal structures.

The LOCA breaks considered for the Ginna Station consist of breaks located at the 3-inch pressurizer spray scoop break and the 4-inch upper plenum injection (UPI) break. The LOCA hydraulic forcing functions (horizontal and vertical forces) that were used in the analyses were generated using MULTIFLEX 3.0 computer code described by Takeuchi, et al (WCAP-9735, Rev. 1, "Multiflex 3.0-A FORTRAN IV Computer Program for Analyzing Thermal-Hydraulic-Structural System Dynamics (III) Advanced Beam Model."

3.9.2.3.4.1 Mathematical Model of the Reactor Pressure Vessel (RPV) System

The mathematical model of the RPV system is a three-dimensional, non-linear finite element model which represents dynamic characteristics of the reactor vessel/internals/fuel in the six geometric degrees of freedom. The RPV system model was developed using the WECAN computer code (Westinghouse Electric Computer Analysis). The WECAN finite element model consists of three concentric structural sub-models connected by non-linear impact elements and stiffness matrices. The first sub-model represents the reactor vessel shell and associated components. The reactor vessel is restrained by reactor vessel supports and by the attached primary coolant piping. The reactor vessel support system is represented by stiffness matrices.

The second sub-model represents the reactor core barrel assembly (core barrel and thermal shield), lower support plate, tie plates, and secondary core support components. This sub-model is physically located inside the first, and is connected to it by a stiffness matrix at the internals support ledge. Core barrel to vessel shell impact is represented by non-linear elements at the core barrel flange, core barrel nozzle, and lower radial support locations.

The third and innermost sub-model represents the upper support plate, guide tubes, support columns, upper and lower core plates, and the fuel. This sub-model includes the specific properties of the Westinghouse 14x14 422 V+ Fuel. The third sub-model is connected to the first and second by stiffness matrices and non-linear elements.

The WECAN computer code, which is used to determine the response of the reactor vessel and its internals, is a general purpose finite element code. In the finite element approach, the structure is divided into a finite number of members or elements. The inertia and stiffness matrices, as well as the force array, are first calculated for each element in the local coordinates. Employing appropriate transformation, the element global matrices and arrays are then computed. Finally, the global element matrices and arrays are assembled into the global structural matrices and arrays, and used for dynamic solution of the differential equation of motion for the structure:

$$[M]\{\ddot{U}\} + [D]\{u'\} + [K]\{U\} = \{F\}$$

where $[M]$ = Global Inertia Matrix

$[D]$ = Global Damping Matrix

$[K]$ = Global Stiffness Matrix

$\{\ddot{U}\}$ = Acceleration Array

$\{u'\}$ = Velocity Array

$\{U\}$ = Displacement Array

$\{F\}$ = Force Array, including impact, thrust forces hydraulic forces, constraints and weight

(Equation 1)

WECAN solves Equation 1 using the non-linear modal superposition theory. An initial computer run is made to calculate the eigenvalues (frequencies) and eigenvectors (mode shapes) for the mathematical model. This information is stored, and is used in a subsequent computer run which solves Equation 1. The first time step performs a static solution of Equation 1 to determine the initial displacements of the structure due to deadweight and normal operating hydraulic forces. After the initial time step, WECAN calculates the dynamic solution of Equation 1. Time history nodal displacements and impact forces are stored for post-processing.

The following typical discrete elements from the WECAN finite element library are used to represent the reactor vessel and internals components:

- Three-dimensional elastic pipe
- Three-dimensional mass with rotary inertia
- Three-dimensional beam
- Three-dimensional linear spring
- Concentric impact element
- Linear impact element
- 6x6 stiffness matrix
- 18 Card stiffness matrix
- 18 Card mass matrix
- Three-dimensional friction element

3.9.2.3.4.2 Analytical Methods

The RPV system finite element model, as described above, was used to perform the LOCA analysis. Following a postulated LOCA pipe rupture, forces are imposed on the reactor vessel and its internals. These forces result from the release of the pressurized primary system coolant. The release of pressurized coolant results in traveling depressurization waves in the primary system. These depressurization waves are characterized by a wavefront with low pressure on one side and high pressure on the other. The wavefront translates and reflects throughout the primary system until the system is completely depressurized. The rapid depressurization results in transient hydraulic loads on the mechanical equipment of the system.

The LOCA loads applied to the reactor pressure vessel system consist of (a) reactor internal hydraulic loads (vertical and horizontal), and (b) reactor coolant loop mechanical loads. All the loads are calculated individually and combined in a time-history manner.

3.9.2.3.4.3 RPV Internal Hydraulic Loads

Depressurization waves propagate from the postulated break location into the reactor vessel through either a hot leg or a cold leg nozzle.

After a postulated break in the cold leg, the depressurization path for waves entering the reactor vessel is through the nozzle into the region between the core barrel and reactor vessel. This region is called the down-comer annulus. The initial waves propagate up, around, and down the down-comer annulus, then up through the region circumferentially enclosed by the core barrel; that is, the fuel region.

The region of the down-comer annulus close to the break depressurizes rapidly but, because of the restricted flow areas and finite wave speed (approximately 3,000 feet per second), the opposite side of the core barrel remains at a high pressure. This results in a net horizontal force on the core barrel and reactor pressure vessel. As the depressurization wave propagates around the downcomer annulus and up through the core, the barrel differential pressure reduces, and similarly, the resulting hydraulic forces drop.

In the case of a postulated break in the hot leg, the waves follow a dissimilar depressurization path, passing through the outlet nozzle and directly into the upper internals region, depressurizing the core and entering the down-comer annulus from the bottom exit of the core barrel. Thus, after a break in the hot leg, the down-comer annulus would be depressurized with very little difference in pressure across the outside diameter of the core barrel.

A hot leg break produces less horizontal force because the depressurization wave travels directly to the inside of the core barrel (so that the down-comer annulus is not directly involved), and internal differential pressures are not as large as for a cold leg break. Since the differential pressure is less for a hot leg break, the horizontal force applied to the core barrel is less for a hot leg break than for a cold leg break. For breaks in both the hot leg and cold leg, the depressurization waves would continue to propagate by reflection and translation through the reactor vessel and loops.

The MULTIFLEX computer code described by Takeuchi calculates the hydraulic transients within the entire primary coolant system. It considers subcooled, transition, and two-phase (saturated) blowdown regimes. The MULTIFLEX program employs the method of characteristics to solve the conservation laws, and assumes one-dimensionality of flow and homogeneity of the liquid-vapor mixture.

The MULTIFLEX code considers a coupled fluid-structure interaction by accounting for the deflection of constraining boundaries, which are represented by separate spring-mass oscillator systems. A beam model of the core support barrel has been developed from the structural properties of the core barrel; in this model, the cylindrical barrel is vertically divided into various segments and the pressure, as well as the wall motions, is projected onto the plane parallel to the broken inlet nozzle. Horizontally, the barrel is divided into 10 segments; each segment consists of 3 separate walls. The spatial pressure variation at each time step is transformed into 10 horizontal forces, which act on the 10 mass points of the beam model. Each flexible wall is bounded on either side by a hydraulic flow path. The motion of the flexible walls is determined by solving the global equations of motion for the masses representing the forced vibration of an undamped beam.

3.9.2.3.4.4 Reactor Coolant Loop Mechanical Loads

The reactor coolant loop mechanical loads are applied to the RPV nozzles by the primary coolant loop piping. The loop mechanical loads result from the release of normal operating forces present in the pipe prior to the separation as well as transient hydraulic forces in the reactor coolant system. The magnitudes of the loop release forces are determined by performing a reactor coolant loop analysis for normal operating loads (pressure, thermal, and deadweight). The loads existing in the pipe at the postulated break location are calculated and are "released" at the initiation of the LOCA transient by application of the loads to the broken piping ends. These forces are applied with a ramp time of 1 msec because of the assumed instantaneous break opening time. For breaks in the branch lines, the force applied at the reactor vessel would be insignificant. The restraints on the main coolant piping would eliminate any force to the reactor vessel caused by a break in the branch line.

3.9.2.3.4.5 Results of the Analysis

The severity of a postulated break in a reactor vessel is related to three factors: the distance from the reactor vessel to the break location, the break opening area, and the break opening time. The nature of the decompression following a LOCA, as controlled by the internals structural configuration previously discussed, results in larger reactor internal hydraulic forces for pipe breaks in the cold leg than in the hot leg (for breaks of similar area and distance from the RPV). Pipe breaks farther away from the reactor vessel are less severe because the pressure wave attenuates as it propagates toward the reactor vessel. The LOCA hydraulic and mechanical loads described in the previous sections were applied to the WECAN model of the reactor pressure vessel system.

The results of LOCA analysis include time history displacements and non-linear impact forces for all major components. The time history displacements of upper core plate, lower core plate and core barrel at the upper core plate elevation are provided as input for the reactor core evaluations. The impact forces calculated at the vessel-internals interfaces are used to evaluate the structural integrity of the reactor vessel and its internals. Using appropriate post-processors, component linear forces are also calculated.

3.9.2.3.5 Transverse Guide Tube Excitation by Blowdown Forces

3.9.2.3.5.1 General

Since the dynamic loads on the guide tubes are more severe for a loss-of-coolant accident caused by a hot-leg rupture than for a cold-leg rupture, only the hot-leg blowdown accident was analyzed. The guide tubes closest to the ruptured outlet leg are subject to the greatest blowdown forces, with the forces decreasing on guide tubes located at greater distances from the ruptured nozzle.

From a hydraulic analysis of the fluid forces acting on the guide tubes nearest the outlet nozzles during MODES 1 and 2, the net force due to a linearly distributed drag force was found to be $F = 1/2\ C_D A V^2 = 357$ lb. The outlet flow velocity during MODES 1 and 2 was $V_{normal} = 48$ fps.

As a result of the 1 msec hot-leg rupture, the outlet mass flux ($m = \gamma V$) was found to increase from 2060 lb/ft ^{2}-sec for MODES 1 and 2 to 8060 lb/ft ^{2}-sec.

The drag force on the guide tube nearest the ruptured nozzle was found by a ratio of the blowdown outlet velocity

$V_{BLOWDOWN} = 8060 / 42.7 = 188.8$ fps

to the normal outlet velocity of 48 fps when squaring this ratio to determine the blowdown force

$F_{BLOWDOWN} = (188.8/48)^2 \times 357 = 5523 \text{ lb} = W$

3.9.2.3.5.2 Response of Guide Tube

A detailed structural analysis of the guide tubes was performed in order to establish the equivalent cross-section properties and elastic end support conditions. The model was verified by an experimental test using a concentrated force applied at the transition plate. The experimental results also produced a load deflection curve into the plastic range for the guide tubes as well as determining deflection criteria to ensure rod cluster control insertion.

The analytical model was used to establish a correlation between the net hydraulic loading for the linearly distributed drag force and a concentrated force applied at the transition plate requiring the deflection of the transition plate to be the same for both loadings. It was found

$F_c = 0.59W = 3259$ lb

The natural frequency of the guide tube was determined experimentally to be 43 Hz which corresponds to a period of T = 23.3 msec. While the hydraulic drag forces on the guide tube were applied over a finite time interval, it was conservatively assumed that the dynamic amplification factor is 2.0 resulting from an impulse loading in the form of a step function. The value of 2.0 was conservative also by virtue of the fact that if yielding occurred the amplification factor was less than 2.0 which is valid for elastic deflections. Thus the maximum dynamic equivalent concentrated force was

$F_{Max} = 2.0 (3259) = 6520$ lb

From the experimental load deflection curve, the maximum permanent guide tube deflection was calculated to be 0.31 in., which corresponds to a maximum deflection of 0.75 in. during the transient.

<u>Conclusions</u>

From the experimental study of rod cluster control insertion as a function of guide tube deflection it was concluded that, under the most severe postulated blowdown accident, rod cluster control insertion was ensured and there would be no loss of function of the rod cluster control guide tubes.

3.9.2.3.5.3 Description of Stress Location

The stress values given in Tables 3.9-15 and 3.9-16 are based upon the maximum force experienced during the blowdown excitation. The maximum stresses for various components in general do not occur simultaneously. A description of the location of the various stresses are as follows:

a. Upper core plate - Bending stresses caused by local deformation of upper core plate between upper support columns.

b. Upper support column - Direct stress in columns due to axial load. Stress calculated for minimum cross-sectional area.

c. Fuel assembly top nozzle - Bending stress in the ligaments of the adaptor plate maximum stress occurs in the section adjacent to the side plate of the top nozzle.

d. Barrel flange - The maximum stress occurs at the transition region between the barrel flange and the upper core barrel. The stresses are both axial and bending.

e. Lower support structure - Maximum bending stress at the center hole. Radius equal 8 in.

f. Core barrel - Axial (direct) stresses located in the reduced cross-sectional area between upper and lower core barrel.

g. Lower core plate - Bending stresses caused by local deformation of lower core plate between shroud tubes.

h. Fuel assembly bottom nozzle - Maximum bending stress occurs in the bars of the bottom nozzle in the section adjacent to the side plates.

3.9.2.3.6 Reevaluation of the Dynamic Response of Reactor Internals for Extended Power Uprate (EPU)

The reactor vessel internals are designed to withstand forces due to structure deadweight, preload of fuel assemblies, control rod assembly dynamic loads, vibratory loads and earthquake accelerations. Changes in the reactor coolant system (RCS) operating conditions as a result of Extended Power Uprate (EPU) result in changes to the boundary conditions (loads and temperatures) experienced by the reactor vessel internals. Therefore, a systematic evaluation of the impact of these changes on the short and long term performance of these components was performed (*Reference 31*). This analysis included eight specific tasks described below.

3.9.2.3.6.1 Reactor Pressure Vessel System Thermal-Hydraulic Analysis

Due to the change in primary side conditions, a reactor pressure vessel system thermal hydraulic analysis was performed. The hydraulic forces were used in the assessment of the structural integrity of the reactor internals, core clamping loads generated by the internals hold down spring, and the stresses in the reactor vessel closure studs.

3.9.2.3.6.2 Bypass Flow Analysis

Bypass flow is the total amount of reactor coolant flow bypassing the core region. The driving force for the bypass flow paths is dependent upon the magnitude of the pressure drop in the reactor core. Since variations in the size of some of the bypass flow paths, such as outlet nozzles and the core cavity region, occur during manufacture, plant specific as-built dimensions were used in order to demonstrate that the bypass flow limits are not violated. Therefore, an analysis was performed to determine actual, best estimate core bypass flow to ensure that the design bypass flow limit for the plant is not exceeded.

3.9.2.3.6.3 Thermal Analysis of the Baffle/Barrel Region

A baffle-barrel region temperature analysis was used to determine the temperature distribution in the baffle plates and in the core barrel. This data was used to evaluate the loadings on the baffle-former bolts, barrel-former bolts and the baffle to baffle edge bolts.

Changes in design transients and in the internal heat generation rates due to gamma heating will affect the relative growth of the barrel and baffle and resulting bolt loads, former plate temperatures, and the skin and bending stresses of all components for which gamma heating is significant. An evaluation was performed to provide thermal data for the structural evaluations of all components that are affected by the changes in the RCS conditions due to the Extended Power Uprate (EPU).

3.9.2.3.6.4 Pressure Drop Across the Baffle Plate Analyses

The hydraulic analysis determines the axial variation in pressure difference across the baffle plates and therefore provides the baffle plate and baffle-barrel region threaded fastener (bolts) pressure loading. This analysis addresses the effects of uncertainties in the relevant hydraulic loss coefficients for the fuel and for the reactor internals. Finally, this information was used as input to the evaluation of the momentum flux of the baffle jets.

3.9.2.3.6.5 Flow Induced Vibration

An assessment of the impact of the new RCS conditions due to Extended Power Uprate (EPU) on flow induced vibration on the reactor internals was performed. This work showed that the vibrational amplitudes of the reactor internals due to the new primary side conditions remain small and have no adverse affect on component structural integrity.

3.9.2.3.6.6 Reactor Internals Structural Integrity

Structural analyses and evaluations were performed to demonstrate that the short and long term structural integrity of the various components of the reactor internals were not adversely impacted by the change in operating conditions. These evaluations addressed changes in hydraulic lift forces as well as changes in component temperature distribution during steady state and transient conditions. In addition, both stress limits and fatigue criteria were addressed.

3.9.2.3.6.7 Control Rod Performance

The effect of the changes in the primary side conditions on the control rod drop times was evaluated.

3.9.2.3.6.8 Vessel/Internals/Fuel/Control Rod Response During Loca Conditions

Detailed time-history analyses were performed to recalculate system interface loads and fuel assembly grid impact loads. Since leak-before-break has been applied to the RCS (*Reference 32*), the limiting breaks considered were an accumulator line break and a pressurizer surge line break. A plant specific dynamic analysis model of the reactor vessel/internals/vessel supports/fuel system was developed using the WECAN code. The reactor pressure vessel model includes the effects of gaps between the reactor internals, fuel and reactor vessel and the non-linear modal superposition method of solution to minimize computing costs. This model was used to develop structural input (beam data) for the Multiflex code. The resulting hydraulic forces were used as input to the time history LOCA structural analysis. Once the time history analyses were performed, stress analysis was performed to determine if stresses and deflections in the Core Support Structures are within the allowable limits for the faulted condition.

3.9.2.3.6.9 Summary of Conclusions

Evaluations have been performed to assess the effect of the Extended Power Uprate (EPU) RCS conditions on the reactor pressure vessel/internals system at Ginna Station. These evaluations used the revised transients along with the consideration of leak-before-break postulated conditions.

The major conclusions reached based on the work described in this report are:

1. The vessel pressure drops, bypass flows and hydraulic lift forces are not significantly affected by the new RCS conditions due to proposed Extended Power Uprate (EPU) program.
2. The design core bypass flow value for Ginna Station is unchanged.
3. Acceptable control rod drop times will be achieved. The current Technical Specification limit of 1.8 seconds remains acceptable.
4. The structural integrity of the reactor internals is maintained with the new RCS conditions.

3.9.2.4 Asymmetric Loss-of-Coolant Accident Loading Analysis

The capability of the reactor vessel internal structures to maintain their functional integrity in the event of a major loss-of-coolant accident was evaluated during the resolution of the Unresolved Safety Issue A-2, Asymmetric Loading. Analysis performed for limited size breaks reported in WCAP 9748 (*Reference 18*), showed that the appropriate systems and components will maintain their functional capability to ensure a safe plant shutdown with a coolable core geometry. The systems and components examined were the reactor vessel assembly including internals, fuel, control rod drive mechanisms, vessel and component supports, reactor coolant loop piping, and attached emergency core cooling piping.

3.9.2.5 Seismic Evaluation of Reactor Vessel Internals

<u>3.9.2.5.1 Analysis Procedure</u>

These structures were analyzed assuming that the operating basis earthquake and the safe shutdown earthquake (0.20g) have equal horizontal and vertical components. Dynamic methods of analysis were used according to the following, with the core and the reactor internals being analyzed as part of a complex reactor structure because of the interconnection of their masses and stiffness.

The general procedure for the dynamic analysis can be summarized as follows:

A. The reactor structure from the ground to the core was reduced to a continuous structural network consisting of elements with variable stiffness, mass distribution, and cross section; concentrated masses, intermediate supports, and local releases (i.e., connections, as between fuel assemblies and core plates that are assumed to be hinges).

B. The canless fuel assembly mechanical design used in the core is composed of fuel rods arranged in a square array, with spring-clip grids locating and holding the fuel rods in the precise array required. Effective stiffness and natural frequency values were determined to establish the response of a fuel assembly to a dynamic excitation. An important characteristic of these structures is that they present a very high internal damping produced by the slippage of the rods on the finger grids. The fact that their own frequency is relatively low with respect to the supporting structure ensured that a resonance phenomenon with the support will not occur. This condition was confirmed by the dynamic analysis.

C. The lower natural transverse frequencies and normal modes were obtained for this complex structure taking into account shear deformations and using numerical methods.

D. The maximum response of the structure under horizontal earthquake excitation was obtained from the superposition of the normal modes responses (with the conservative assumption that all the modes were in phase and that all the peaks occur simultaneously) and using response curves normalized for 0.08g and 0.20g maximum ground accelerations using 1% damping.

E. After obtaining the maximum possible response under earthquake excitation, the stress values at the critical structure points were computed.

F. For the vertical earthquakes the same general method was employed but using an equivalent one degree of freedom system.

3.9.2.5.2 Analysis Results

Stresses and deflections of reactor internals and core were determined using the method explained above. The vertical and horizontal components of the ground accelerations were considered separately. The stress distribution for each case was calculated after obtaining the maximum response of the structure. These stresses were then combined with stresses of other origin (pressure stresses, thermal stresses, etc.) to obtain maximum stresses which must be within the limits given by the allowable stress criteria. The maximum stresses were, therefore, conservatively determined on whichever combination of simultaneous conditions yield the highest stress condition.

The maximum deflections under seismic accelerations were computed and combined with deflections from other loadings. These deflections were sufficiently small to permit normal operation and do not necessarily coincide in time with maximum stresses.

Stresses of earthquake origin were considered as primary stresses. For the reactor internals the primary membrane stresses induced by earthquake loadings (0.08g and 0.20g maximum ground accelerations) combined with induced primary membrane stresses from other loading conditions, as described above, remained within the design stress intensity values established by the ASME Boiler and Pressure Vessel Code, Section III. Primary bending and secondary stresses which included thermal stresses were also limited following the criteria and methods prescribed by the ASME Code, Section III.

For the fuel assemblies, stress levels are such that the fuel assembly functional integrity is maintained under the action of the imposed loads including seismic effects.

Tables 3.9-17 through 3.9-19 summarize the primary principal stress results at various elevations in the reactor. Table 3.9-20 presents the maximum primary stress intensities. These values are seen to be considerably below the allowable value of 24,000 psi. Table 3.9-21 summarizes the primary plus secondary principal stress results at various elevations in the reactor. Table 3.9-22 presents the maximum primary plus secondary stress intensities. These values are seen to be considerably below the allowable value of 48,000 psi.

3.9.3 COMPONENT SUPPORTS AND CORE SUPPORT STRUCTURES

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

The loadings and design transients used are the same as those used for the piping, equipment, and component analyses given in Section 3.9.1. The bases for the original design of Ginna Station are as follows:

> All piping, components, and supporting structures of the reactor coolant system were designed as Seismic Category I equipment, i.e., they are capable of withstanding:
>
> 1. Within code allowable, working stresses for the design seismic ground acceleration.
> 2. The maximum potential seismic ground acceleration acting in the horizontal and vertical direction simultaneously with no loss function.

The loadings, load combinations, and stress limits used in the original design and during the Systematic Evaluation Program (SEP) reevaluation are given in Table 3.9-1 and Table 3.9-11, respectively.

3.9.3.2 Component Supports

The reactor coolant system components and supports were designed as Seismic Category I.

3.9.3.2.1 Reactor Vessel

The vessel is supported on six individual pedestals. Each pedestal rests upon plates which are in turn supported upon the circular concrete primary shield wall.

The reactor vessel has six supports comprising four support pads located one on the bottom of each of the primary nozzles and two gusset support pads. One of the reactor inlet nozzles is centered approximately 2 degrees counterclockwise from the 90-degree axis and the other is centered approximately 2 degrees counterclockwise from the 270-degree axis.

Each support bears on a support shoe, which is fastened to the support structure. The support shoe is a structural member that transmits the support loads to the supporting structure. The support shoe is designed to restrain vertical, lateral, and rotational movement of the reactor vessel, but allows for thermal growth by permitting radial sliding at each support, on bearing plates.

3.9.3.2.2 Steam Generators

Each steam generator is supported on a structural system consisting of four vertical support columns and two (upper and lower) support systems. The vertical columns, which are pin-connected to the steam generator support feet, serve as vertical restraint for operating weights, pipe rupture, and seismic considerations while permitting movement in the horizontal plane. The support systems, by using a combination of stops, guides, and snubbers, prevent rotation and excessive movement of the steam generator in any horizontal plane.

The lower support system consists of an arrangement of structural steel shapes in combination with steel plates that are in a horizontal plane. The system is designed to restrain excessive horizontal movement of the steam generator and also to accommodate thermal growth. The upper support system consists of three sets of rigid struts and one set of hydraulic snubbers (see Figure 3.9-6a). The snubbers function under tension or compression loads while the struts are compression only elements. The struts were installed so that there are minimal gaps between the strut and the corresponding support element on the steam generator. The steam generator support structures were originally designed for loads resulting from ruptures of the main steam piping and primary coolant piping. These loads exceeded the seismic loads. The upper support rings were constrained by eight hydraulic snubbers, a pair in each of the four lateral directions.

Generic Letter 87-11 eliminated the requirement to consider the dynamic effects of arbitrary intermediate pipe ruptures and removed the postulated main steam line rupture in the first horizontal run of main steam line as the controlling design load for the steam generator upper lateral support system. RG&E applied the leak-before-break theory to remove the primary coolant line rupture as the next highest design load for the support system. The removal of these two controlling loads permitted the replacement of six of the hydraulic snubbers for each steam generator with the rigid bumpers in the upper support system. The new support system was evaluated for the load combinations and allowable stress limits defined in Table 3.9-23.

3.9.3.2.3 Reactor Coolant Pumps

Each reactor coolant pump is supported by a structural system consisting of three vertical columns and a system of stops. The vertical columns are bolted to the pump support feet and permit movement in the horizontal plane to accommodate reactor coolant pipe expansion. Horizontal restraint is accomplished by a combination of tie rods and stops which limit horizontal movement for pipe rupture and seismic effects.

Support structures of the steam generators and reactor coolant pump components were designed for loads resulting from ruptures of the primary coolant piping and main steam piping. Equivalent static seismic forces equal to the component weight, accelerated by the peak response of the applicable seismic response spectra, applied through the component center of gravity, were evaluated against the corresponding pipe rupture loads. For both the steam generators and reactor coolant pumps, the resulting seismic forces were smaller than the pipe rupture loads; therefore, supports were designed for pipe rupture loads.

3.9.3.2.4 Pressurizer

The pressurizer is supported on a heavy concrete slab spanning between the concrete shield walls for the steam generator compartment. The pressurizer is a bottom skirt supported vessel.

3.9.3.2.5 Reactor Coolant Piping

The reactor coolant piping layout is designed on the basis of providing floating supports for the steam generator and reactor coolant pump in order to permit the thermal expansion from the fixed or anchored reactor vessel. A comprehensive thermal analysis was performed to ensure that stresses induced by linear thermal expansion are within code limits.

3.9.3.3 Pipe Supports

3.9.3.3.1 Original Analysis

The pipe stress analysis performed during the original design of Ginna Station also gave the pipe support reactions. The results of the analysis indicated that the margin between the ultimate support capacity and the support reactions for 0.2g ground acceleration was sufficient to handle building amplification.

For the Seismic Category I piping 2 in. nominal size and under, the support reactions were well below the capacity of the supports (*Reference 4*). For pipes falling in this category, the minimum hanger rod diameter was found to be 1/2 in. for outdoor installations and 3/8 in. for indoor installations. The 3/8-in. rods had an ultimate capacity of the order of 3700 lb. The horizontal supports had an ultimate capacity, in shear, of the order of 1100 lb. For the heaviest pipe in this category, the support reactions were of the order of 100 lb, i.e., well below the ultimate capacity of the supports.

A few pipe runs had lateral support spacing two to three times that suggested by USAS B31.1 for vertical supports. The support reactions for the heaviest pipe of this category were of the order of 200 lb and well within the ultimate capacity of the supports.

3.9.3.3.2 IE Bulletin Reanalysis

Subsequent to the original design of the Ginna Station piping, several dynamic analyses of the piping system were performed that included the later developed loading requirements and regulatory changes. The analyses performed for the residual heat removal loop, the main steam line loop, safety injection system piping, and charging line in response to IE Bulletin 79-07 are described in Section 3.9.2.1. The pipe support reactions calculated from these analyses using as-built conditions and the design loads for the residual heat removal loop, main steam line loop, and charging line are given in Table 3.9-24 through Table 3.9-26. Results indicate the adequacy of these pipe supports.

3.9.3.3.3 Seismic Piping Upgrade Program

3.9.3.3.3.1 Applicable Supports

Supports for Seismic Category I piping systems listed in Section 3.7.3.7.1 were included in the Seismic Piping Upgrade Program.

3.9.3.3.3.2 Load Combinations and Stress Limits

The piping system supports were evaluated for the following piping system imposed loads and support inertial effects:

a. Normal condition: deadweight and maximum operating thermal.

b. Design condition: deadweight, maximum operating thermal, and operating-basis earthquake.

c. Safe shutdown earthquake condition: deadweight, normal operating thermal, and safe shutdown earthquake.

The loading combinations and associated stress limits are given in Table 3.9-27. The allowable stress criteria were in accordance with Subsection NF of the ASME Section III Code, 1974. Faulted condition stress allowables from Appendix F of the ASME Section III Code and Regulatory Guide 1.124 were used to analyze the supports for the safe shutdown earthquake condition. The variance in allowable criteria between the piping and supports will not cause over-or under-designs to occur, as the satisfaction of the operating-basis earthquake condition to the working stress limits will in all cases be most stringent. The component support embedments were evaluated using current analytical techniques in accordance with the anchor bolt manufacturer's Technical Information and ACI-349, Appendix B. The expansion anchorages must meet the requirements set forth in IE Bulletin 79-02.

3.9.3.3.3.3 Structural Requirements

For anchors that separate SeismicCategory I piping systems from nonseismic piping, the loads from the Seismic Category I side were doubled. The effects of friction on supports was considered for pipes having thermal movements greater than 0.1 in. The value of was 0.35 and was used conservatively to increase support loads but not reduce loads.

The stiffness of the supports was considered in the piping system models. The local subsystem stiffness of all piping and equipment supports was determined considering the pipe or equipment supports along with the structural steel and/or concrete effect. The localized subsystem stiffness of all piping and equipment supported by reinforced-concrete members (including concrete pedestals) was considered when significant. The stiffness was based on the face of concrete interface.

Rigid supports were modeled in accordance with the following criteria:

Nominal Pipe Size (in.)	K_{min} Rigid (lb/in.)	K_{min} Rigid (in.-lb/rad)
≤2	1×10^5	1×10^7
2-1/2 to 4	5×10^5	5×10^7
≥6	1×10^6	1×10^8

Use of the above guidelines eliminates excessive support stiffness calculation effort, while yielding satisfactory support displacement results (i.e., thermal deflections <0.02 in., rotations <0.0002 radians).

"Common pipe supports" refer to those supports to which two or more pipes are attached in such a way that significant coupling occurs between the pipes. When all attached pipes are the same size and the distances to adjacent supports are similar, the local subsystem stiffness is based on the deflections resulting from an equal load acting at all support points. When different size pipes are attached, or if the distances to adjacent supports are not similar, a stiffness matrix relating the forces and displacements at the points of attachments to one another was provided to the piping analyst for use in uncoupling the piping systems.

Hydraulic seismic supports (snubbers) generally lock up at an excitation frequency of approximately 1 Hz, with a piping displacement of 0.05 in. Mechanical snubbers activate in a frequency range of 1 to 6 Hz with a similar piping displacement of 0.05 in. As piping system frequencies seldom exist below this range, seismic supports were modeled as active during all seismic events.

Supports were considered active statically in any given direction provided the support gap in that direction does not exceed 0.125 in. This 0.125 in. tolerance is essentially construction variance, which does not alter the designed function of the support. Supports with gaps greater than 0.125 in. were incorporated as follows. System analysis first assumed that the support was not active; piping displacements resulting from this run were then used to ascertain the validity of this assumption. If incorrect, reanalysis incorporated an active support statically.

The inertial effects of the supports own mass was considered. The additional inertial loads were determined based on a review of the support flexibility, support mass, and applicable response spectra.

All supports were analyzed and modified if necessary to be in compliance with IE Bulletin 79-02 criteria. Any existing support with anchor bolts subject to tension loads and which were previously only subject to compression or shear loads were inspected or tested to confirm installation adequacy.

The effects of new support loads generated by the piping reanalysis upon the existing structures were evaluated.

Piping supports were modeled as described in Section 3.7.3.7.10.

3.9.3.3.4 Base Plate Flexibility

In general, calculation of anchor bolt loads for pipe supports at Ginna Station assumed rigid base plates. This included both the shell type concrete expansion anchor bolts used in the original plant design and the wedge type which were generally used for plant modifications.

In order to assess the significance of rigid versus flexible plate assumptions, a representative sample of typical pipe support base plates were reanalyzed. The reanalysis was performed assuming both the base plate and bolts as elastic and using separate procedures for moment and axial loadings.

It was not possible to reanalyze, using flexible plate assumptions, the base plates on all pipe supports in the testing and replacement program prior to initiation. Therefore, a representative sample of 10 typical pipe support base plates has been analyzed, using rigid plate assumptions, for both existing and replacement designs. The results of these analyses are shown in Table 3.9-28. In all cases, bolt capacity has been increased in the replacement designs. In two cases, additional analyses, using flexible plate assumptions, were performed. These analyses showed minimum factors of safety of 5.00 and 5.35, respectively, for the replacement designs. The design factor of safety for the wedge type anchor bolts used in the replacement designs was 4.00. Therefore, it was determined that the design bolt capacities provide sufficient margins of safety to account for any load increases due to flexibility.

In general, pipe supports at Ginna Station with base plates using concrete expansion anchor bolts are of similar design. They are typical of the type used in Seismic Category I systems throughout the plant.

The capacity of concrete expansion anchor bolts to withstand cyclic loads (seismic as well as high cyclic operating loads) were evaluated in fast flux test facility tests. The test results indicated that

A. The expansion anchors successfully withstood two million cycles of long-term fatigue loading at a maximum intensity of 0.2 of the static ultimate capacity. When the maximum load intensity was steadily increased beyond that value and cycled for 2000 times at each load step, the observed failure load was about the same as the static ultimate capacity.

B. The dynamic load capacities of the expansion anchors under simulated seismic loading were about the same as the corresponding static ultimate capacities.

Based on the above data, it could be concluded that the design requirements for preloaded concrete expansion anchor bolts under cyclic loads are the same as for the static loads.

3.9.3.3.5 Snubbers

3.9.3.3.5.1 Design Loads

The mechanical and hydraulic suppressors (snubbers) installed on Seismic Category I piping systems and the steam generators at Ginna Station were designed to restrain seismic loads. Hydraulic snubbers installed on pressurizer safety valve discharge piping were designed to restrain hydraulic loads resulting from safety valve discharges. The loads which the snubbers had to meet were calculated by seismic or thermal hydraulic analysis, as appropriate. Standard available snubbers were purchased with rated loads greater than or equal to the calculated loads. A review of the various snubbers installed on these systems and components showed that they were capable of functioning with loads at least 1.33 times their rated loads and were structurally designed for loads at least 2.0 times their rated loads.

The hydraulic snubbers were designed to operate with an internal fluid pressure of 3000 psi and to limit fluid pressure to 4000 psi by means of a spring-loaded relief valve(*Reference 4*). When the compressive load exceeded 14.7 kips and 28 kips for the 11 kips and 21 kips snubbers, respectively, the spring-loaded relief valves opened. If this load was sustained, the snubber would eventually get solid. The mechanical ultimate capability was about four times the design capacity, i.e., 84 kips and 44 kips for 21 kips and 11 kips snubbers, respectively.

Therefore, the seismic loads associated with 0.2g ground acceleration were found not to cause mechanical failure of these snubbers. The only potential effect could be some movement of the snubber rod because of temporary loss of fluid. However, because of the dynamic nature of the seismic loads and the inherent flexibility of the supported pipes, the potential limited snubber movement would not induce stresses in the feedwater and steam lines above tolerable limits.

A review was made of the capability of the various snubbers to lock up upon application of their design loads. Since the basic seismic analysis method utilized at the time Ginna Station was designed was a static, lumped mass approach, specific dynamic requirements were not established by the seismic analysis. However, a conservative analysis of the minimum velocities that could be experienced during a seismic event, based on a frequency of 33 Hz and a ground acceleration of 0.08g, gives a result of approximately 60 in./minute. Hydraulic snubbers installed at Ginna Station are capable of locking up with velocities no greater than 10 in./minute.

3.9.3.3.5.2 Surveillance Program

A surveillance program for snubbers has been instituted at Ginna Station. The current requirements for inspection and functional testing of snubbers are included in Interface Procedure IP-IIT-5, Snubber Inspection and Testing Program.

3.9.4 CONTROL ROD DRIVE SYSTEMS

3.9.4.1 Description

3.9.4.1.1 General

The control rod drive mechanisms are used for withdrawal and insertion of the control rods into the reactor core and to provide sufficient holding power for stationary support. Fast total insertion (reactor trip) is obtained by simply removing the electrical power allowing the rods to fall by gravity.

The complete drive mechanism, shown in Figures 3.9-7 and 3.9-8, consists of the internal (latch) assembly, the pressure vessel, the operating coil stack, the drive shaft assembly, and the position indicator coil stack.

Each assembly is an independent unit which can be dismantled or assembled separately. Each drive is threaded into an adaptor on top of the reactor pressure vessel and is connected to the control rod (directly below) by means of a grooved drive shaft. The upper section of the drive shaft is suspended from the working components of the drive mechanism. The drive shaft and control rod remain connected during reactor operation, including tripping of the rods.

Main coolant fills the pressure containing parts of the drive mechanism. All working components and the shaft are immersed in the main coolant.

Three magnetic coils, which form a removable electrical unit and surround the rod drive pressure housing, induce magnetic flux through the housing wall to operate the working components. They move two sets of latches which lift or lower the grooved drive shaft.

The three operating coils are sequenced by solid-state switches for the control rod drive assemblies. The sequencing of the magnets produces step motion over the 144 in. of normal control rod travel.

The mechanism develops a lifting force approximately two times the static lifting load. Therefore, extra lift capacity is available for overcoming mechanical friction between the moving and the stationary parts. Gravity provides the drive force for rod insertion and the weight of the whole rod assembly is available to overcome any resistance.

A multiconductor cable connects the mechanism operating coils to the 125-V dc power supply. The power supply includes the necessary switchgear to provide power to each coil in the proper sequence.

In 1996, the NRC issued NRC Bulletin 96-01 (*Reference 26*) to alert licensees to problems encountered during events in which control rods failed to completely insert upon the scram signal and to have licensees assess control rod operability at their facilities. RG&E's response to IEB 96-01 (*References 27 through 30*) addressed training performed in relation to the issues, operability determinations made, justification for not performing rod drop testing and gathering recoil data at the end of Cycle 25, and future plans, and transmitted core map information and control rod drag testing results. In addition, RG&E stated that based on a review of the rod drag testing data, both Westinghouse and RG&E concluded that there was no concern for rod cluster control assembly insertion anomalies at burnups tested for Ginna.

3.9.4.1.2 Latch Assembly

The latch assembly contains the working components which withdraw and insert the drive shaft and attached control rod. It is located within the pressure housing and consists of the pole pieces for three electromagnets. They actuate two sets of latches which engage the grooved section of the drive shaft.

The upper set of latches move up or down to raise of lower the drive rod by 5/8 in. The lower set of latches have 1/32-in. axial movement to shift the weight of the control rod from the upper to the lower latches.

3.9.4.1.3 Pressure Vessel

The pressure vessel consists of the pressure housing and rod travel housing. The pressure housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel. It provides space for the drive shaft during its upward movement as the control rod is withdrawn from the core.

3.9.4.1.4 Operating Coil Stack

The operating coil stack is an independent unit which is installed on the drive mechanism by sliding it over the outside of the pressure housing. It rests on a pressure housing flange without any mechanical attachment and is removed and installed while the reactor is pressurized.

The operating coils (A, B, and C) are made of round copper wire which is insulated with a double layer of filament-type glass yarn.

3.9.4.1.5 Drive Shaft Assembly

The main function of the drive shaft is to connect the control rod to the mechanism latches. Grooves for engagement and lifting by the latches are located throughout the 144 in. of control rod travel. The grooves are spaced 5/8 in. apart to coincide with the mechanism step length and have 45 degree angle sides.

The drive shaft is attached to the control rod by the coupling. The coupling has two flexible arms which engage the grooves in the spider assembly. A 1/4-in. diameter disconnect rod runs down the inside of the drive shaft. It utilizes a locking button at its lower end to lock the coupling and control rod.

During plant operation, the drive shaft assembly remains connected to the control rod at all times. It can be attached and removed from the control rod only when the reactor vessel head is removed.

3.9.4.1.6 Position Indicator Coil Stack

The position indicator coil stack slides over the rod travel housing section of the pressure vessel. It detects drive rod position by means of discrete cylindrically wound coils that are spaced at 7.5 in. (12 step) intervals along the rod travel (144 in.).

3.9.4.2 Design Loads, Stress Limits, and Allowable Deformation

The mechanisms are designed to operate in water at 650°F and 2485 psig. The temperature at the mechanism head adaptor will be much less than 650°F because it is located in a region where there is limited flow of water from the reactor core, while the pressure is the same as in the reactor pressure vessel.

The design operating temperature of the coils is 232°C. Coil temperature can be determined by resistance measurement. Forced air cooling along the outside of the coil stack maintains a coil temperature of approximately 200°C.

3.9.4.3 Control Rod Drive Mechanism Housing Mechanical Failure Evaluation

An evaluation of the possibility of damage to adjacent control rod drive mechanism housings in the event of a circumferential or longitudinal failure of a rod housing located on the vessel head is presented.

3.9.4.3.1 Housing Description

The control rod drive mechanism schematic is shown in Figure 3.9-8. The operating coil stack assembly of this mechanism has a 10.8 in. by 10.8 in. cross section and a 39.55 in. length. The position indicator coil stack assembly (not shown in the figure) is located above the operating coil stack assembly. It surrounds the rod travel housing over nearly its entire length.

The rod travel housing outside diameter is 3.8 in. and the position indicator coil stack assembly inside and outside diameters are approximately 4 in. and 7 in., respectively. This assembly consists of a 1/8-in. thick stainless steel tube on which are mounted 20 coils. The coils are mounted at 12 step (7.5 inch) intervals along the tube. This assembly is held together by two end plates (the top end plate is square), an outer sleeve, and four axial tie rods.

3.9.4.3.2 Effects of Rod Travel Housing Longitudinal Failures

Should a longitudinal failure of the rod travel housing occur, the region of the stainless steel tube opposite the break would be stressed by the reactor coolant pressure of 2250 psia. The most probable leakage path would be provided by the radial deformation of the position indicator coil assembly, resulting in the growth of the axial flow passages between the rod travel housing and the stainless steel tube. A radial free water jet is not expected to occur because of the small clearance between the stainless steel tube and the rod travel housing, and the considerable resistance of the combination of the stainless steel tube and the position indicator coils to internal pressure. Calculations based on the mechanical properties of stainless steel and copper at reactor operating temperature show that an internal pressure of at least 4000 psia would be necessary for the combination of the stainless steel tube and the coils to rupture.

Therefore, the combination of stainless steel tube and copper coils stack is more than adequate to prevent formation of a radial jet following a control rod housing split which ensures the integrity of the adjacent rod housings.

3.9.4.3.3 Effect of Rod Travel Housing Circumferential Failures

If circumferential failure of a rod travel housing should occur, the broken-off section of the housing would be ejected vertically because the driving force is vertical and the position indicator coil stack assembly and the drive shaft would tend to guide the broken-off piece upwards during its travel. Travel is limited to less than 2 ft by the missile shield, thereby limiting the projectile acceleration. When the projectile reaches the missile shield, it would partially penetrate the shield and dissipate its kinetic energy. The water jet from the break would push the broken-off piece against the missile shield.

If the broken-off piece were short enough to clear the break when fully ejected, it could rebound after impact with the missile shield. The top end plates of the position indicator coil stack assemblies and the coil stacks would prevent the broken piece from directly hitting the rod travel housing of a second drive mechanism. Even if a direct hit by the rebounding piece were to occur, the low kinetic energy of the rebounding projectile would not be expected to cause significant damage.

3.9.4.3.4 Summary

The considerations given above lead to the conclusion that failure of a control rod housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings that would increase the severity of the initial accident.

3.9.5 REACTOR PRESSURE VESSEL INTERNALS

3.9.5.1 Design Arrangements

The reactor pressure vessel internals are shown in Figures 3.9-9 and 3.9-10. The internals, consisting of the upper and lower core support structure, are designed to support, align, and guide the core components, direct the coolant flow to and from the core components, and to support and guide the in-core instrumentation.

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and thermal shield), the upper core support structure, and the in-core instrumentation support structure.

3.9.5.1.1 Lower Core Support Structure

3.9.5.1.1.1 Support Structure Assembly

The major containment and support member of the reactor internals is the lower core support structure. This support structure assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the thermal shield, the intermediate diffuser plate, and the bottom support plate which is welded to the core barrel. All the major material for this structure is type 304 stainless steel. The core support structure is supported at its upper flange from a ledge in the reactor vessel head flange and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are axial baffle and former plates which are attached to the core barrel wall and form the enclosure periphery of the assembled core. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

3.9.5.1.1.2 Lower Core Plate

The lower core plate is a 1.5-in.-thick member through which the necessary flow distributor holes for each fuel assembly are machined. Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns are placed between this plate and the bottom support plate of the core barrel in order to provide stiffness to this plate and transmit the core load to the bottom support plate. Intermediate between the support plate and lower core support plate is positioned a perforated plate to diffuse uniformly the coolant flowing into the core.

3.9.5.1.1.3 Thermal Shield

The thermal shield is a solid, relatively thick (3.56 in.) cylinder that is supported from the core barrel at both the top and bottom end.

The upper end of the shield is rigidly connected to the core barrel at six equally spaced points through mounting pads projecting from the core barrel. This connection is designed to prevent relative motion between the shield and barrel in both the radial and axial direction.

To provide for a difference in axial elongation between the shield and core barrel resulting from the temperature distribution at operation conditions, the lower connection is designed to allow axial movement between the two members but restrict the radial movement. This is accomplished by means of six flexible strap connections between the shield and barrel. These relatively thin straps are sufficiently flexible to withstand the axial displacement between the shield at core barrel but have sufficient width and cross-section area to restrict the radial motion.

A rigid connection is used at the upper end of the shield to obtain the inherent stability of suspending a heavy mass from the top and also because field and model tests have indicated that the maximum disturbing forces occur at the upper end.

Response of the thermal shield to the design dynamic loading was determined for both ring and beam mode vibration. The resulting force and moment reactions were used in determining the design requirements of the upper and lower connections.

The design dynamic loading used was considerably greater than any expected loading, based on measurements of actual pressure fluctuations during hot functional tests and also from model tests. The total stress was obtained by combining the thermal stresses, resulting from axial and radial elongation, with the anticipated dynamic stresses.

Irradiation baskets in which materials samples can be inserted and irradiated during reactor operation are attached to the thermal shield. The irradiation capsule basket supports are welded to the thermal shield. There is no extension of this support above the thermal shield as was done in the older designs. Thus, the basket has been removed from the high flow disturbance zone. The welded attachment to the shield extends the full length of the support except for small interruptions about 1 in. long. This type of attachment has an extremely high natural frequency. The specimens are held in position within the baskets by a stop at the bottom and a slotted cylindrical spring at the top which fits against a relief in the basket. The specimen does not extend through the top of the basket and thus is protected by the basket from the flow.

3.9.5.1.1.4 Coolant Flow Passages

The lower core support structure and the core barrel serve to provide passageways and control for the coolant flow. Inlet coolant flow from the vessel inlet nozzles proceeds down the annulus between the core barrel and the vessel wall, flows on both sides of the thermal shield, and then into a plenum at the bottom of the vessel. It then turns and flows up through the lower support plate, passes through the intermediate diffuser plate and then through the lower core plate. The flow holes in the diffuser plate and the lower core plate are arranged to give a very uniform entrance flow distribution to the core. After passing through the core, the coolant enters the area of the upper support structure and then flows, generally radially, to the core barrel outlet nozzles and directly through the vessel outlet nozzles.

A small amount of water also flows between the baffle plates and core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum and exits through the vessel output nozzles.

3.9.5.1.1.5 Support and Alignment Arrangements

Vertical downward loads from weight, fuel assembly preload, control rod dynamic loading, and earthquake acceleration are carried by the lower core plate, partially into the lower core plate support flange on the barrel shell and partially through the lower support columns to the bottom support plate. From there the loads are carried through the core barrel shell to the core barrel flange supported by the vessel head flange. Transverse loads from earthquake acceleration, coolant cross flow, and vibration are carried by the core barrel shell to be shared by the lower radial support to the vessel head flange. Transverse acceleration of the fuel assemblies is transmitted to the core barrel shell by direct connection of the lower core support plate to the barrel shell, by direct connection of the lower core support plate to the barrel wall, and by a radial support type connection of the upper core plate to slab-sided pins pressed into the core barrel.

The main radial support system of the core barrel is accomplished by key and keyway joints to the reactor vessel wall. At equally spaced points around the circumference, an Inconel block is welded to the vessel I.D. Another Inconel block is bolted to each of these blocks, and has a keyway geometry. Opposite each of these is a key which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. With this design, the internals are provided with a support at the furthest extremity and may be viewed as a beam fixed at the top and simply supported at the bottom.

Radial and axial expansions of the core barrel are accommodated but transverse movement of the core barrel is restricted by this design. With this system, cycle stresses in the internal structures are within the ASME Section III limits. This eliminates any possibility of failure of the core support.

<u>3.9.5.1.2 Upper Core Support Assembly</u>

The upper core support assembly consists of the top support plate, deep beam sections, and upper core plate between which are contained support columns and guide tube assemblies. The support columns establish the spacing between the top support plate, deep beam sections, and the upper core plate and are fastened at top and bottom to these plates and beams. The support columns transmit the mechanical loadings between the two plates and serve the supplementary function of supporting thermocouple guide tubes. The guide tube assemblies sheath and guide the control rod drive shafts and control rods, but provide no other mechanical function. They are fastened to the top support plate and are guided by pins in the upper core plate for proper orientation and support. Additional guidance for the control rod drive shafts is provided by the control rod shroud tube which is attached to the upper support plate and guide tube.

The upper core support assembly, which is removed as a unit during the MODE 6 (Refueling) operation, is positioned in its proper orientation with respect to the lower support structure by flat-sided pins pressed into the core barrel which in turn engage in slots in the upper core plate. At an elevation in the core barrel where the upper core plate is positioned, the flat-sided pins are located at equal angular positions. Slots are milled into the core plate at the same positions. As the upper support structure is lowered into the main internals, the slots in the plate engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support structure, the upper core support assembly, the fuel assemblies, and control rods is ensured by this system of locating pins and guidance arrangement. The upper core support assembly is restrained from any axial movements by a large circumferential spring which rests between the upper barrel flange and the upper core support assembly and is compressed by the reactor vessel head flange.

Vertical loads from weight and fuel assembly preload are transmitted through the upper core plate via the support columns to the deep beams and top support plate and then through the circumferential spring to the reactor vessel head. Transverse loads from coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the top support plate and upper core plate. The top support plate is particularly stiff to minimize deflection.

3.9.5.1.3 In-Core Instrumentation Support Structures

The in-core instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom.

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip-connected to in-line columns that are in turn fastened to the upper support plate. These port columns protrude through the head penetrations. The thermocouples are carried through these port columns and the upper support plate at positions above their readout locations. The thermocouple conduits are supported from the columns of the upper core support system. The thermocouple conduits are sealed stainless steel tubes.

In addition to the upper in-core instrumentation, there are reactor vessel bottom port columns which carry the retractable, cold-worked stainless steel flux thimbles that are pushed upward into the reactor core. Conduits extend from the bottom of the reactor vessel down through the concrete shield area and up to a thimble seal line. The minimum bend radii are about 90 in. and the trailing ends of the thimbles (at the seal line) are extracted approximately 13 ft during MODE 6 (Refueling) of the reactor in order to avoid interference within the core. The thimbles are closed at the leading ends and serve as the pressure barrier between the reactor pressurized water and the containment atmosphere.

Mechanical seals between the retractable thimbles and the conduits are provided at the seal line. During MODES 1 and 2, the retractable thimbles are stationary and move only during MODE 6 (Refueling) or for maintenance, at which time a space of approximately 13 ft above the seal line is cleared for the retraction operation.

The in-core instrumentation support structure is designed for adequate support of instrumentation during reactor operation and is rugged enough to resist damage or distortion under the conditions imposed by handling during the MODE 6 (Refueling) sequence.

The flux mapping system includes a drive and control system for inserting the in-core detectors. A portion of the drive system, which includes the fifteen-path rotary transfer devices and the isolation valves, is mounted on the movable seal cart, which is normally located above the seal table (see Section 7.7.4.2.3). The seal cart is mounted on a rail structure used to move the seal cart out of the way during refueling. The seal cart is designed and restrained to prevent the flux mapping system from collapsing onto the seal table during a seismic event and jeopardizing the seal table reactor coolant system pressure boundary. The reactor bottom-mounted instrumentation system is Seismic Category I.

3.9.5.2 Loading Conditions

The internals are designed to withstand the forces due to weight, reload of fuel assemblies, control rod dynamic loading, vibration, and earthquake acceleration. Under the loading conditions, including conservative effects of design earthquake loading, the structure satisfies stress values prescribed in ASME Section III.

The reactor internal components are designed to withstand the stresses resulting from startup, steady-state operation with any number of pumps running, and shutdown conditions. The abnormal design conditions assume blowdown effects due to an accumulator line break or pressurizer surge line break.

3.9.5.3 Design Bases

The criteria for acceptability is that the core should be coolable and intact following a pipe rupture up to and including a double-ended rupture of the reactor coolant system. This implies that core cooling and adequate core shutdown must be ensured. Consequently, the limitations established on the internals are concerned principally with the maximum allowable deflections and/or stability of the parts. The allowable stress criteria is discussed in Section 3.9.2.3.1.3.

For abnormal operation the criteria for acceptability are that the reactor be capable of safe shutdown and that the engineered safety features are able to operate as designed. The limitation established on the internals for these types of loads are also concerned principally with the maximum allowable deflections. The deflection criteria for critical structures under abnormal operation are presented in Table 3.9-29.

3.9.6 INSERVICE INSPECTION OF PUMPS AND VALVES

3.9.6.1 General

The following information defines the Inservice Pump and Valve Testing Program for the period starting January 1, 2010, through December 31, 2019. Included in this program are the quality groups A and B pumps which are provided with an emergency power source and those quality groups A, B, and C valves which are required to shut down the reactor or to mitigate the consequences of an accident and maintain the reactor in a safe shutdown condition. Quality groups A, B, and C components correspond to those defined in Regulatory Guide 1.26.

This program has been developed as required by Section 50.55a(g) of 10 CFR 50 following the guidance of the ASME OM Code-2004, "Code for Operation and Maintenance of Nuclear Power Plants." The program follows the guidance of Generic Letter 89-04 with possible exceptions approved by the NRC. The program was submitted to the NRC. The NRC has reviewed and approved the program and acted on program relief requests (*Reference 19*).

Further addenda and editions of ASME OM Code-2004 will be used for clarification of test requirements and performance.

The Inservice Pump and Valve Testing Program substantially augments but does not affect the pump and valve surveillance program required by the Technical Specifications. Technical Specifications requirements associated with pump and valve surveillance will continue to be implemented as specified. When changes to Technical Specifications create conflicts with the program, the revised Technical Specifications will provide guidance until the program is revised to incorporate the changes.

The motor-operated valve analysis and test system (MOVATS) program described in Section 5.4.9.3 supports the Inservice Pump and Valve Testing Program via Code Case OMN1.

When a valve, pump, or its control system has been replaced or repaired or has undergone maintenance that could affect its performance and prior to the time it is returned to service, it will be tested as necessary to demonstrate that the performance parameters which could have been affected by the replacement, repair, or maintenance are within acceptable limits.

Code Edition and Testing Interval

The Inservice Pump and Valve Testing Program for the period January 1, 2010, through December 31, 2019, was developed using the 2004 Edition of the ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

3.9.6.2 Inservice Testing of Pumps

The inservice pump testing program was developed in accordance with the requirements of subsection ISTB of the ASME OM Code. This program includes all quality group A and B pumps, which are provided with an emergency power source and are required to perform a specific function in shutting down the reactor or in mitigating the consequences of an accident and maintain the reactor in a safe shutdown condition.

The pumps to be tested and the test parameters and frequencies are specified in the inservice pump and valve testing program.

Testing of a pump need not be performed if that pump is declared inoperable without the testing. Consistent with the Technical Specifications, specified intervals may be extended by 25% to accommodate normal test schedules.

Records for the inservice pump testing program are developed and maintained in accordance with Subsection ISTA-9000, "Records and Reports" of the Code for Operation and Maintenance of Nuclear Power Plants.

3.9.6.3 Inservice Testing of Valves

The inservice valve testing program was developed in accordance with the requirements of subsection ISTC of the ASME OM Code. All those valves that are required to perform a specific function either to shut down the reactor to the MODE 5 (Cold Shutdown) condition or in mitigating the consequences of an accident and maintain the reactor in a safe shutdown condition are included in the program.

The inservice valve testing program requirements for category A, B, and C valves are included in the Pump and Valve Testing Program. Category D valves are not included in this testing program because there are none included in Ginna Station design.

Some exceptions and exemptions to the testing requirements of ISTC have been taken based on operational interference, placing the plant in an unsafe condition, and Technical Specifications requirements. All exceptions and exemptions are listed and explained in the Pump and Valve Testing Program.

Records for the inservice valve testing program are developed and maintained in accordance with Subsection ISTA-9000, "Records and Reports" of the Code for Operation and Maintenance of Nuclear Power Plants.

3.9.7 EXTENDED POWER UPRATE (EPU)

During the 2006 RFO, Ginna Station implemented Plant Change Request, PCR 2004-0009, "Ginna Station Extended Power Uprate (EPU) Project." Additional information to support EPU can be obtained from plant records associated with PCR 2004-0009 and *References 31* and *33*.

REFERENCES FOR SECTION 3.9

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Table 3.9-1
ORIGINAL DESIGN LOADING COMBINATIONS AND STRESS LIMITS

Loading Combinations	Vessels and Reactor Internals	Piping	Supports
Normal + design earthquake loads	$P_m \leq S_m$ $P_L + P_B \leq 1.5\ S_m$	$P_m \leq 1.2\ S$ $P_L + P_B \leq 1.2\ S$	Working stresses
Normal + maximum potential earthquake loads	$P_m \leq 1.2\ S_m$ $P_L + P_B \leq 1.2\ (1.5\ S_m)$	$P_m \leq 1.2\ S$ $P_L = P_B \leq 1.2\ (1.5\ S)$	Within yield after load redistribution
Normal + pipe rupture loads	$P_m \leq 1.2\ S_m$ $P_L + P_B \leq 1.2\ (1.5\ S_m)$	$P_m \leq 1.2\ S_m$ $P_L + P_B \leq 1.2\ (1.5\ S)$	Within yield after load redistribution

Where:

P_m = primary general membrane stress or stress intensity.

P_L = primary local membrane stress or stress intensity.

P_B = primary bending stress or stress intensity.

S_m = stress intensity value from ASME B&PV Code, Section III.

S = allowable stress from USAS B31.1 Code for Pressure Piping.

Table 3.9-2
RESIDUAL HEAT REMOVAL LOOP A STRESS SUMMARY

Description	Original[a] Design (psi)	As-Built[b] Condition (psi)	Allowable Stress (psi)
SEISMIC STRESSES			
Operating-basis earthquake			
Vertical + Z-horizontal	---	3,356	---
Vertical + X-horizontal	---	3,900	---
Safe shutdown earthquake			
Vertical + Z-horizontal	10,564	8,284	---
Vertical + X-horizontal	5,674	9,716	---
COMBINED STRESSES			
Operating-basis earthquake + pressure + deadweight	---	9,436	19,080
Safe shutdown earthquake + pressure + deadweight	16,715	15,252	28,620

a. Results obtained using WESTDYN and 1969 model which considers the supports rigid.

b. Results obtained using WESTDYN and as-built conditions considering support stiffnesses.

Table 3.9-3
MAIN STEAM LINE-LOOP B STRESS SUMMARY[a]

Description	As-Built Condition	
	Dynamic[a] Results (psi)	Allowable Stress (psi)
SEISMIC STRESSES		
Operating-basis earthquake		
Vertical + Z-horizontal	965	---
Vertical + X-horizontal	963	---
Safe shutdown earthquake		
Vertical + Z-horizontal	2,373	---
Vertical + X-horizontal	2,238	---
COMBINED STRESSES		
Operating-basis earthquake + pressure + deadweight	7,278	16,440
Safe shutdown earthquake + pressure + deadweight	8,686	24,660
NOTE: Additional evaluations to support Ginna Extended Power Uprate are available from plant records associated with PCR 2004-0009 and *Reference 31*.		

a. Stresses given are obtained using B31.1-1973 Summer Addenda, formula 12.

Table 3.9-4
CHARGING LINE STRESS SUMMARY[a]

Description	As-Built Dynamic Analysis Condition (psi)	Allowable Stress (psi)
SEISMIC STRESSES		
Operating-basis earthquake		
Vertical + Z-horizontal	150	---
Vertical + X-horizontal	245	---
Safe shutdown earthquake		
Vertical + Z-horizontal	436	---
Vertical + X-horizontal	638	---
COMBINED STRESSES		
Operating-basis earthquake + pressure + deadweight	6,941	20,580
Safe shutdown earthquake + pressure + deadweight	7,334	30,870

a. Stresses given are obtained using B31.1-1973 Summer Addenda, formula 12.

Table 3.9-5
LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR PRESSURIZER SAFETY AND RELIEF VALVE PIPING AND SUPPORTS - UPSTREAM OF VALVES

Combination	Plant/System Operating Condition	Load Combination [a]	Piping Allowable Stress Intensity
1	Normal	N	1.0 S_h
2	Upset	N + OBE + SOT_U	1.2 S_h
3	Emergency	N + SOT_E	1.8 S_h
4	Faulted	N + MS/FWPB or DBPB + SSE + SOT_F	2.4 S_h
5	Faulted	N + LOCA + SSE + SOT_F	2.4 S_h

a. Definitions of load abbreviations are in Table 3.9-7.

Table 3.9-6
LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR PRESSURIZER SAFETY AND RELIEF VALVE PIPING AND SUPPORTS - SEISMICALLY DESIGNED DOWNSTREAM PORTION

Combination	Operating Condition	Load Combination [a]	Piping Allowable Stress Intensity
1	Normal	N	1.0 S_h
2	Upset	N + SOT_U	1.2 S_h
3	Upset	N + OBE + SOT_U	1.8 S_h
4	Emergency	N + SOT_E	1.8 S_h
5	Faulted	N + MS/FWPB or DBPB + SSE + SOT_F	2.4 S_h
6	Faulted	N + LOCA + SSE + SOT_F	2.4 S_h

a. Definitions of load abbreviations are in Table 3.9-7.

Table 3.9-7
DEFINITIONS OF LOAD ABBREVIATIONS [a]

N	Sustained loads during normal plant operation
SOT	System operating transient
SOT_U	Relief valve discharge transient
SOT_E	Safety valve discharge transit
SOT_F	Maximum of SOT_U and SOT_E; or transition flow
OBE	Operating-basis earthquake
SSE	Safe shutdown earthquake
MS/FWPB	Main steam or feedwater pipe break
DBPB	Design-basis pipe break
LOCA	Loss-of-coolant accident
S_h	Basic material allowable stress at maximum (hot) temperature

a. Abbreviations used in TABLES 3.9-5 and 3.9-6.

Table 3.9-8
LOADING COMBINATIONS AND STRESS LIMITS FOR PIPING FOR SEISMIC UPGRADE PROGRAMS

	Loading Combinations	Stress Limits
DEADWEIGHT	Design Pressure + Deadweight	$P_m \le S_h$; $P_L + P_B \le S_h$
OBE SEISMIC	Design Pressure + Deadweight Design + Earthquake Loads (OBE)	$P_m \le 1.2\ S_h$; $P_L + P_B \le 1.2\ S_h$
SSE	Operating Pressure + Deadweight + Maximum Potential Earthquake Loads (SSE)	$P_m \le 1.8\ S_h$; $P_L + P_B \le 1.8\ S_h$
THERMAL	Maximum Operating Thermal + OBE Displacements	$S_E \le S_A$
	Design Pressure + Deadweight + Maximum Operating Thermal + OBE Displacements	$P_L + P_B \le (S_h + S_A)$

Where:

OBE = operating-basis earthquake

P_m = primary general membrane stress; or stress intensity

P_L = primary local membrane stress; or stress intensity

P_B = primary bending stress; or stress intensity

S_A, S_h = allowable stress from USAS B31.1 Code for pressure piping

S_E = thermal expansion stress from USAS B31.1 code for pressure piping

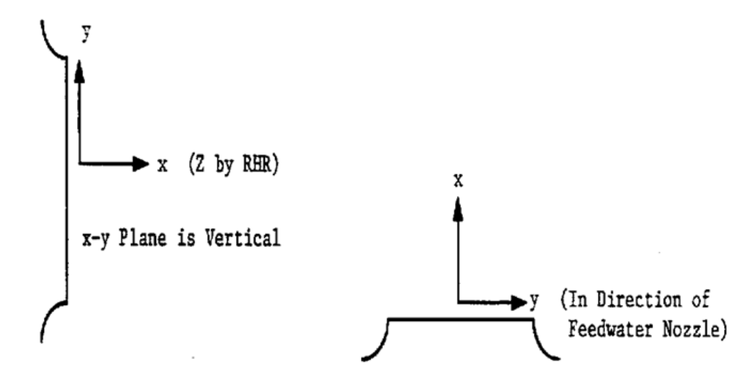
SSE = safe shutdown earthquake

Table 3.9-9
ALLOWABLE STEAM GENERATOR NOZZLE LOADS

Condition	Fx	Fy	Fz	Mx	My	Mz
FEEDWATER NOZZLE						
Thermal	15	40	40	1000	1500	1500
Weight	5	15	5	250	500	500
Seismic operating-basis earthquake	75	75	75	1500	2000	2000
Seismic design-basis earthquake	100	100	100	2000	3000	3000
STEAM NOZZLE						
Thermal	100	50	50	6000	5000	5000
Weight	20	10	10	500	500	750
Seismic operating-basis earthquake	150	150	150	5000	5000	5000
Seismic design-basis earthquake	200	200	200	7500	7500	7500

Notes:

1. All loads are ± unless indicated.
2. Units are kips and in -kips.
3. Coordinate system



Feedwater Nozzle　　Steam Nozzle

Table 3.9-10
REACTOR COOLANT PUMP AUXILIARY NOZZLE UMBRELLA LOADS

Nozzle	Condition /Load	Fx (lb)	Fy (lb)	Fz (lb)	Mx (in.-lb)	My (in.-lb)	Mz (in.-lb)
Seal injection	Thermal	350	100	300	3500	2800	2000
	Deadweight	10	-80	10	300	250	400
	Seismic OBE	250	50	225	1600	4500	2000
	Seismic SSE	800	250	350	3200	15000	4000
No. 1 seal bypass	Thermal	75	70	40	300	315	1525
	Deadweight	5	-25	1	75	50	350
	Seismic OBE	50	50	45	900	1200	900
	Seismic SSE	160	170	170	1650	2550	2000
No. 1 seal leakoff	Thermal	400	200	300	2000	2000	2000
	Deadweight	10	-80	5	300	250	400
	Seismic OBE	500	400	500	1000	5000	2000
	Seismic SSE	800	500	600	2000	8000	3500
No. 2 seal leakoff	Thermal	75	100	100	300	350	1600
	Deadweight	5	-25	5	75	75	400
	Seismic OBE	50	100	100	900	1500	1200
	Seismic SSE	160	170	170	1650	2500	2000

Nozzle	Condition /Load	Fx (lb)	Fy (lb)	Fz (lb)	Mx (in.-lb)	My (in.-lb)	Mz (in.-lb)
No. 3 seal injection	Thermal	90	45	45	290	290	180
	Dead-weight	15	35	10	90	45	180
	Seismic OBE	90	150	150	480	560	480
	Seismic SSE	180	300	300	960	1120	960
No. 3 seal leakoff	Thermal	90	45	45	290	290	180
	Dead-weight	15	35	10	90	45	180
	Seismic OBE	90	150	150	480	560	480
	Seismic SSE	180	300	300	960	1120	960
Thermal barrier component cooling water in and out	Thermal	75	200	150	3200	1300	2500
	Dead-weight	20	-75	1	5	5	150
	Seismic OBE	100	250	100	1000	1200	1200
	Seismic SSE	200	700	200	4500	3000	3600
Upper bearing oil cooler and air cooler component cooling water in and out	Thermal	100	100	100	300	300	200
	Dead-weight	5	-80	5	100	50	200
	Seismic OBE	100	300	300	500	600	500
	Seismic SSE	200	600	600	1000	1200	1000
Lower bearing oil cooler component cooling water in and out	Thermal	95	340	305	470	480	525
	Dead-weight	10	-35	10	100	125	125
	Seismic OBE	90	90	90	290	290	180
	Seismic SSE	90	90	90	290	290	180

Nozzle	Condition /Load	Fx (lb)	Fy (lb)	Fz (lb)	Mx (in.-lb)	My (in.-lb)	Mz (in.-lb)

Note:

1. Values at ± unless otherwise specified.
2. Loads on the No. 3 seal connections apply only if a No. 3 "Double Dam" seal is supplied.
3. Loads on pump nozzles are to be applied at the nozzle to shell juncture.
4. Loads on motor nozzles are to be applied at the flange end.
5. Coordinate system.
6. OBE = operating-basis earthquake.
7. SSE = safe shutdown earthquake.

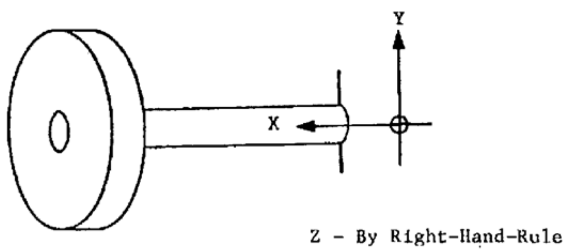


Table 3.9-11
SYSTEMATIC EVALUATION PROGRAM STRUCTURAL BEHAVIOR CRITERIA FOR DETERMINING SEISMIC DESIGN ADEQUACY

Components	Systematic Evaluation Program Criteria, Safe Shutdown Earthquake	
Vessels, pumps, and valves	$S_{m\ (all)} \leq 0.7\ S_u$ and $1.6\ S_y$	ASME III Class 1 (Table F 1322.2.1)
	$S_{m\ (all)} \leq 0.67\ S_u$ and $1.33\ S_y$	ASME III Class 2 (NC 3217)
	$\sigma_{m\ (all)} \leq 0.5\ S_u$ and $1.25\ S_y$	ASME III Class 2 (NC 3321)
	$\sigma_{m\ (all)} \leq 0.5\ S_u$ and $1.25\ S_y$	ASME III Class 3 (ND 3321)
Piping	$S_{m\ (all)} \leq 1.0\ S_u$ and $2.0\ S_y$	ASME III Class 1 (Table F 1322.2.1)
	$S_h \leq 0.6\ S_u$ and $1.5\ S_y$	ASME III Class 2 and Class 3 (NC 3611.2)
Tanks	No ASME III Class 1	
	$\sigma_{m\ (all)} \leq 0.5\ S_u$ and $1.25\ S_y$	ASME III Class 2 and Class 3 (NC 3821)
Electric equipment	$S_{\ (all)} \leq 1.0\ S_y$	
Cable trays	$S_{\ (all)} \leq 1.0\ S_y$	
ASME supports	$S_{\ (all)} \leq 1.2\ S_y$ and $0.7\ S_u$	ASME III Appendices XVII, F for Class 1, 2 and 3
Other supports	$S_{\ (all)} \leq 1.6\ S$	Normal AISC S allowable increased by 1.6 consistent with NRC Standard Review Plan, Sec. 3.8
Bolting	$S_{\ (all)} \leq 1.4\ S$	ASME Section III Appendix XVII for bolting where S is the allowable stress for design loads
NOTE:—$S_{(all)}$ = Stress Allowable.		

Table 3.9-12
MECHANICAL COMPONENTS SELECTED FOR SEP SEISMIC REVIEW

Item	Mechanical Component Description	Reason for Selection
1	Essential service water pump	This item has a long vertical unsupported intake section which was originally statically analyzed for seismic effects.
2	Component cooling heat exchanger	This item is supported on what appears to be a relatively flexible structural steel framing and by two saddles.
3	Component cooling surge tank	Same as Item 2.
4	Diesel-generator air tanks	This item is a skirt-supported vertical tank.
5	Boric acid storage tank	This item is a column-supported vertical tank.
6	Refueling water storage tank (RWST)	Evaluate anchor-bolt systems for in-structure flat-bottom tanks that are flexible.
7	Motor-operated valves	A general concern with respect to motor-operated valves, particularly for lines 4 in. or less in diameter, is that the relatively large eccentric mass of the motor will cause excessive stresses in the attached piping if the valves are not externally supported.
8	Steam generators	Items are particularly critical to ensure reactor coolant system integrity.
9	Reactor coolant pumps	Same as Item 8.
10	Pressurizer	Same as Item 8.
11	Control rod drive mechanism	Same as Item 8.
12	Reactor coolant system supports	Same as Item 8.

Table 3.9-13
MAXIMUM STRESS HOT-LEG BREAK (ORIGINAL ANALYSIS)

-	Stresses			Allowable	
Components	**Direct**	**Bending**	**Total**	**Direct**	**Total**
Core plate	0	17,800	17,800	39,500	50,000
Upper support columns	15,000	---	15,000	39,500	50,000
Top nozzle (minor)	0	24,800	24,800	39,500	50,000
Top nozzle (major)	0	20,600	20,600	39,500	50,000
Flange barrel	4,000	31,800	35,800	39,500	50,000
Lower support structure	0	7,670	7,670	39,500	50,000
Barrel	3,200	0	3,200	39,500	50,000
Fuel assembly thimbles	40,400	---	40,400	45,000	---

Table 3.9-14
MAXIMUM STRESS COLD-LEG BREAK (ORIGINAL ANALYSIS)

-	Stresses			Allowable	
Components	**Direct**	**Bending**	**Total**	**Direct**	**Total**
Upper core plate	0	4,800	4,800	39,500	50,000
Upper support column	8,700	0	8,700	39,500	50,000
Bottom nozzle (minor assembly)	0	45,200	45,200	39,500	50,000
Bottom nozzle (major assembly)	0	47,800	47,800	39,500	50,000
Flange barrel	4,000	31,800	35,800	39,500	50,000
Lower support structure	0	21,400	21,400	39,500	50,000
Barrel	11,500	0	11,500	39,500	50,000
Lower core plate	0	8,400	8,400	39,500	50,000
Fuel assembly thimbles	40,400	---	40,400	45,000	---

Table 3.9-15
MAXIMUM CORE BARREL STRESS AND DEFLECTION UNDER HOT-LEG BLOWDOWN (ORIGINAL ANALYSIS)

Rupture Time (msec)	Maximum Deflection (in.)	Allowable Deflection (in.)	Maximum Stress (psi)	Allowable Stress (psi)	Compressive Wave (psi)	Critical Pressure (psi)
1	0.031	5	14,110	39,500	450	2,612

Table 3.9-16a
MAXIMUM STRESS INTENSITIES AND DEFLECTION COLD-LEG BLOWDOWN (ORIGINAL ANALYSIS) - IN THE UPPER BARREL

Rupture Time (msec)	Maximum Stress Intensity (psi)	Allowable Stress Intensity (psi)	Maximum Membrane Stress (psi)	Allowable Membrane Stress (psi)	Maximum Deflection (mils)
1	44,500	50,000	36,750	39,500	150
5	34,500	50,000	26,750	39,500	95
20	34,500	50,000	26,750	39,500	95

Table 3.9-16b
MAXIMUM STRESS INTENSITIES AND DEFLECTION COLD-LEG BLOWDOWN (ORIGINAL ANALYSIS) - AT THE UPPER BARREL ENDS

Rupture Time (msec)	Rise Time (msec)	Peak Pressure (psi)	Maximum Upper Bending Stress (psi)	Maximum Lower Bending Stress (psi)	Allowable (psi)
1	2	750	49,800	26,850	50,000
5	4.5	650	40,370	21,755	50,000
20	4.5	650	40,370	21,755	50,000

Table 3.9-17
CORE BARREL STRESSES (ORIGINAL ANALYSIS)

-	Primary Principal Stresses		
Barrel Flange Weld	S_1 (psi) (Tangential)	S_2 (psi) (Longitudinal)	S_3 (psi) (Radial)
OUTSIDE SURFACE			
Normal operating	2159	2797	-1655
0.08g vertical earthquake	0	141	0
0.08g horizontal earthquake	0	90	0
Normal operating + 0.08g earth-quake	2159	3028	-1655
0.20g vertical earthquake	0	235	0
0.20g horizontal earthquake	0	150	0
Normal operating + 0.20g earth-quake	2159	3413	-1655
INSIDE SURFACE			
Normal operating	3378	-1825	-1618
0.08g vertical earthquake	0	14	0
0.08g horizontal earthquake	0	90	0
Normal operating + 0.08g earth-quake	3378	-1594	-1618
0.20g vertical earthquake	0	235	0
0.20g horizontal earthquake	0	150	0
Normal operating + 0.20g earth-quake	3378	-1209	-1618
Note: The values in this Table remains bounding for Extended Power Uprate (EPU).			

Table 3.9-18
CORE BARREL STRESSES (ORIGINAL ANALYSIS)

-	Primary Principal Stresses		
Barrel Middle Girth Weld	S_1 (psi) (Tangential)	S_2 (psi) (Longitudinal)	S_3 (psi) (Radial)
OUTSIDE SURFACE			
Normal operating	-5686	-9347	-2250
0.08g vertical earthquake	0	307	0
0.08g horizontal earthquake	0	235	0
Normal operating + 0.08g earth-quake	-5686	-8805	-2250
0.20g vertical earthquake	0	512	0
0.20g horizontal earthquake	0	392	0
Normal operating + 0.20g earth-quake	-5686	-7901	-2250
INSIDE SURFACE			
Normal operating	-5414	-8295	-2200
0.08g vertical earthquake	0	307	0
0.08g horizontal earthquake	0	235	0
Normal operating + 0.08g earth-quake	-5414	-7753	-2200
0.20g vertical earthquake	0	512	0
0.20g horizontal earthquake	0	392	0
Normal operating + 0.20g earth-quake	-5414	-6849	2200
Note: The values in this Table remains bounding for Extended Power Uprate (EPU).			

Table 3.9-19
CORE BARREL STRESSES (ORIGINAL ANALYSIS)

-	Primary Principal Stresses		
Barrel Lower Girth Weld	S_1 (psi) (Tangential)	S_2 (psi) (Longitudinal)	S_3 (psi) (Radial)
OUTSIDE SURFACE			
Normal operating	-4059	-6608	0
0.08g vertical earthquake	0	165	
0.08g horizontal earthquake	0	35	0
Normal operating + 0.08g earth-quake	-4059	-6408	-609
0.20g vertical earthquake	0	275	0
0.20g horizontal earthquake	0	58	0
Normal operating + 0.20g earth-quake	-4059	-6075	-609
INSIDE SURFACE			
Normal operating	1103	7962	916
0.08g vertical earthquake	0	165	0
0.08g horizontal earthquake	0	35	0
Normal operating + 0.08g earth-quake	1103	8162	916
0.20g vertical earthquake	0	275	0
0.20g horizontal earthquake	0	58	0
Normal operating + 0.20g earth-quake	1103	8495	916
Note: The values in this Table remains bounding for Extended Power Uprate (EPU).			

Table 3.9-20
CORE BARREL STRESSES (ORIGINAL ANALYSIS)

Barrel Flange Weld	**Maximum Primary Stress Intensity (psi)**
Outside Surface	
Normal operating + 0.08g earthquake	4683
Normal operating + 0.20g earthquake	5068
Inside Surface	
Normal operating + 0.08g earthquake	4996
Normal operating + 0.20g earthquake	4996
Barrel Middle Girth Weld	
Outside Surface	
Normal operating + 0.08g earthquake	6555
Normal operating + 0.20g earthquake	5651
Inside Surface	
Normal operating + 0.08g earthquake	5553
Normal operating + 0.20g earthquake	4649
Barrel Lower Girth Weld	
Outside Surface	
Normal operating + 0.08g earthquake	5799
Normal operating + 0.20g earthquake	5466
Inside Surface	
Normal operating + 0.08g earthquake	7246
Normal operating + 0.20g earthquake	7579
Note: The values in this Table remains bounding for Extended Power Uprate (EPU).	

Table 3.9-21
CORE BARREL STRESSES (ORIGINAL ANALYSIS)

	Primary Plus Secondary Principal Stresses		
	S_1 (psi) (Tangential)	S_2 (psi) (Longitudinal)	S_3 (psi) (Radial)
Barrel Flange Weld			
OUTSIDE SURFACE			
Normal operating + 0.08g earth-quake	10,289	20,135	-1,640
Normal operating + 0.20g earth-quake	10,289	20,520	-1,640
INSIDE SURFACE			
Normal operating + 0.08g earth-quake	6,298	-4,963	-1,603
Normal operating + 0.20g earth-quake	6,298	-4,578	-1,603
Barrel Middle Girth Weld			
OUTSIDE SURFACE			
Normal operating + 0.08g earth-quake	2,768	4,071	-2,261
Normal operating + 0.20g earth-quake	2,768	4,975	-2,261
INSIDE SURFACE			
Normal operating + 0.08g earth-quake	-17,206	-20,666	-2,211
Normal operating + 0.20g earth-quake	-17,206	-19,762	-2,211
Barrel Lower Girth Weld			
OUTSIDE SURFACE			
Normal operating + 0.08g earth-quake	-4,059	-6,408	-609

	Primary Plus Secondary Principal Stresses		
	S_1 (psi) (Tangential)	S_2 (psi) (Longitudinal)	S_3 (psi) (Radial)
Normal operating + 0.20g earth-quake	-4,059	-6,075	-609
INSIDE SURFACE			
Normal operating + 0.08g earth-quake	1,103	8,162	916
Normal operating + 0.20g earth-quake	1,103	8,459	916
Note: The values in this Table remains bounding for Extended Power Uprate (EPU).			

Table 3.9-22
CORE BARREL STRESSES (ORIGINAL ANALYSIS)

	Maximum Primary Plus Secondary Stress Intensity (psi)
Barrel Flange Weld	
OUTSIDE SURFACE	
Normal operating + 0.08g earthquake	21,775
Normal operating + 0.20g earthquake	22,160
INSIDE SURFACE	
Normal operating + 0.08g earthquake	11,261
Normal operating + 0.20g earthquake	10,876
Barrel Middle Girth Weld	
OUTSIDE SURFACE	
Normal operating + 0.08g earthquake	6,332
Normal operating + 0.20g earthquake	7,263
INSIDE SURFACE	
Normal operating + 0.08g earthquake	18,455
Normal operating + 0.20g earthquake	17,551
Barrel Lower Girth Weld	
OUTSIDE SURFACE	
Normal operating + 0.08g earthquake	5,799
Normal operating + 0.20g earthquake	5,466
INSIDE SURFACE	
Normal operating + 0.08g earthquake	7,246
Normal operating + 0.20g earthquake	7,579

Note: The values in this Table remains bounding for Extended Power Uprate (EPU).

Table 3.9-23a
LOAD COMBINATIONS AND ALLOWABLE STRESS LIMITS FOR PRIMARY EQUIPMENT SUPPORTS EVALUATION - FOR PLANT EVENTS

	Plant Event	Plant Operating Conditions	Service Loading Combinations[a]	Service Level Stress Limits [b]
1.	Normal operation (MODES 1 and 2)	Normal	Sustained loads	A
2.	Plant/system operating transients (SOT) + OBE	Upset	Sustained loads + SOT + OBE	B
3.	DBPB	Emergency	Sustained loads + DBPB	C
4.	SSE	Faulted	Sustained loads + SSE	D
5.	DBPB (or MS/FWPB) + SSE	Faulted	Sustained loads + (DBPB D or MS/FWPB + SSE)	

a. The pipe break loads and SSE loads are combined by the square root sum of the squares method.

b. Stress levels are defined by ASME Code, Section III, Subsection NF, 1974 edition.

Table 3.9-23b
LOAD COMBINATIONS AND ALLOWABLE STRESS LIMITS FOR PRIMARY EQUIPMENT SUPPORTS EVALUATION - DEFINITION OF LOADING CONDITIONS FOR PRIMARY EQUIPMENT SUPPORTS EVALUATION IN TABLE 3.9-23a

1.	Sustained loads	DW, deadweight +P, operating pressure +TN, normal operating thermal
2.	Transients	SOT, system operating transient
3.	Overtemperature transient	TA
4.	Operating-basis earthquake	OBE
5.	Safe shutdown earthquake	SSE
6.	Design basis pipe break / design basis accident	DBPB/DBA
	Residual heat removal line	RHR
	Accumulator line	ACC
	Pressurizer surge line	SURG
7.	Main steam line break	MS
8.	Feedwater pipe break	FW

Table 3.9-24
RESIDUAL HEAT REMOVAL LOOP A SUPPORT LOADS[a] CALCULATED FOR IE BULLETIN 79-07

Supports	Description	As-Built Conditions (lb)	Design Load (lb)
RH-34 vertical	Operating-basis earthquake		3600
	Vertical + Z-Horizontal	2820	
	Vertical + X-Horizontal	2720	
	Safe shutdown earthquake		5400
	Vertical + Z-Horizontal	3370	
	Vertical + X-Horizontal	3110	
RH-8 vertical	Operating-basis earthquake		1680
	Vertical + Z-Horizontal	1110	
	Vertical + X-Horizontal	1260	
	Safe shutdown earthquake		2520
	Vertical + Z-Horizontal	1340	
	Vertical + X-Horizontal	1680	
RH-7 vertical	Operating-basis earthquake		2160
	Vertical + Z-Horizontal	1080	
	Vertical + X-Horizontal	1090	
	Safe shutdown earthquake		3240
	Vertical + Z-Horizontal	1200	
	Vertical + X-Horizontal	1220	
RH-6 horizontal	Operating-basis earthquake		5640
	Vertical + Z-Horizontal	990	

a. Support load combination is seismic plus deadweight.

Supports	Description	As-Built Conditions (lb)	Design Load (lb)
	Vertical + X-Horizontal	860	
	Safe shutdown earthquake		8460
	Vertical + Z-Horizontal	2390	
	Vertical + X-Horizontal	2030	
RH-5 vertical	Operating-basis earthquake		2160
	Vertical + Z-Horizontal	740	
	Vertical + X-Horizontal	740	
	Safe shutdown earthquake		3240
	Vertical + Z-Horizontal	930	
	Vertical + X-Horizontal	930	
RH-4 horizon-tal	Operating-basis earthquake		3720
	Vertical + Z-Horizontal	600	
	Vertical + X-Horizontal	780	
	Safe shutdown earthquake		5580
	Vertical + Z-Horizontal	1390	
	Vertical + X-Horizontal	1850	
RH-3 vertical	Operating-basis earthquake		2160
	Vertical + Z-Horizontal	1910	
	Vertical + X-Horizontal	1880	
	Safe shutdown earthquake		3240

Supports	Description	As-Built Conditions (lb)	Design Load (lb)
	Vertical + Z-Horizontal	2250	
	Vertical + X-Horizontal	2180	
RH-2 vertical	Operating-basis earthquake		2160
	Vertical + Z-Horizontal	1600	
	Vertical + X-Horizontal	1600	
	Safe shutdown earthquake		3240
	Vertical + Z-Horizontal	1920	
	Vertical + X-Horizontal	1930	
RH-1 vertical	Operating-basis earthquake		2160
	Vertical + Z-Horizontal	1780	
	Vertical + X-Horizontal	1870	
	Safe shutdown earthquake		3240
	Vertical + Z-Horizontal	2200	
	Vertical + X-Horizontal	2420	
RH-1 horizontal	Operating-basis earthquake		3720
	Vertical + Z-Horizontal	324	
	Vertical + X-Horizontal	880	
	Safe shutdown earthquake		5580
	Vertical + Z-Horizontal	780	
	Vertical + X-Horizontal	2150	

Table 3.9-25a
MAIN STEAM LINE LOOP B SUPPORT LOADS[a] CALCULATED FOR IE BULLETIN 79-07 - SEISMIC SUPPORT

Seismic Supports	Description	As-Built Conditions (lb)	Design Load (lb)
MS-7	Operating-basis earthquake		
	Vertical + Z-Horizontal	3,040	21,000
	Vertical + X-Horizontal	6,930	21,000
	Safe shutdown earthquake		
	Vertical + Z-Horizontal	6,200	21,000
	Vertical + X-Horizontal	14,060	21,000
MS-8	Operating-basis earthquake		
	Vertical + Z-Horizontal	6,140	21,000
	Vertical + X-Horizontal	5,260	21,000
	Safe shutdown earthquake		
	Vertical + Z-Horizontal	15,350	21,000
	Vertical + X-Horizontal	13,240	21,000

a. Support load combination is seismic plus deadweight.

Table 3.9-25b
MAIN STEAM LINE LOOP B NOZZLE LOADS CALCULATED FOR IE BULLETIN 79-07 - NOZZLE LOADS

NOZZLE LOADS	-	-	-	-	-	-
	WESTDYN Local Coordinate System					
Description	KIPS			IN-KIPS		
OBE induced load	9	2	4	300	209	514
Seismic OBE allowable loads	150	150	150	5000	5000	5000
SSE induced loads	15	5	4	649	279	1160
Seismic SSE allowable loads	200	200	200	7500	7500	7500

Table 3.9-26
CHARGING LINE SUPPORT LOADS[a] CALCULATED FOR IE BULLETIN 79-07

Supports	Description	As-Built Conditions (lb)	Design Load (lb)
S-35 vertical	Operating-basis earthquake		1,500
	Vertical + Z-Horizontal	570	
	Vertical + Z-Horizontal	580	
	Safe shutdown earthquake		2,250
	Vertical + Z-Horizontal	620	
	Vertical + Z-Horizontal	600	
S-60 vertical	Operating-basis earthquake		1,500
	Vertical + Z-Horizontal	20	
	Vertical + Z-Horizontal	20	
	Safe shutdown earthquake		2,250
	Vertical + Z-Horizontal	30	
	Vertical + Z-Horizontal	30	
S-135 vertical	Operating-basis earthquake		8,850
	Vertical + Z-Horizontal	40	
	Vertical + Z-Horizontal	40	
	Safe shutdown earthquake		12,750
	Vertical + Z-Horizontal	40	
	Vertical + Z-Horizontal	40	
S-135 axial	Operating-basis earthquake		8,500
	Vertical + Z-Horizontal	65	

Supports	Description	As-Built Conditions (lb)	Design Load (lb)
	Vertical + Z-Horizontal	65	
	Safe shutdown earthquake		12,750
	Vertical + Z-Horizontal	65	
	Vertical + Z-Horizontal	65	
S-145 vertical	Operating-basis earth-quake		1,500
	Vertical + Z-Horizontal	10	
	Vertical + Z-Horizontal	10	
	Safe shutdown earthquake		2,250
	Vertical + Z-Horizontal	20	
	Vertical + Z-Horizontal	20	
S-210 vertical	Operating-basis earth-quake		8,500
	Vertical + Z-Horizontal	50	
	Vertical + Z-Horizontal	50	
	Safe shutdown earthquake		12,750
	Vertical + Z-Horizontal	50	
	Vertical + Z-Horizontal	50	
S-210 axial	Operating-basis earth-quake		8,500
	Vertical + Z-Horizontal	65	
	Vertical + Z-Horizontal	65	
	Safe shutdown earthquake		12,750
	Vertical + Z-Horizontal	65	

Supports	Description	As-Built Conditions (lb)	Design Load (lb)
	Vertical + Z-Horizontal	65	
S-225 vertical	Operating-basis earthquake		1,500
	Vertical + Z-Horizontal	10	
	Vertical + Z-Horizontal	10	
	Safe shutdown earthquake		2,250
	Vertical + Z-Horizontal	20	
	Vertical + Z-Horizontal	10	
N 404 horizontal (2 in.)	Operating-basis earthquake		375
	Vertical + Z-Horizontal	0	
	Vertical + Z-Horizontal	10	
	Safe shutdown earthquake		562
	Vertical + Z-Horizontal	10	
	Vertical + Z-Horizontal	10	
N 404 horizontal (3 in.)	Operating-basis earthquake		375
	Vertical + Z-Horizontal	40	
	Vertical + Z-Horizontal	40	
	Safe shutdown earthquake		562
	Vertical + Z-Horizontal	50	
	Vertical + Z-Horizontal	60	
N 405 vertical (2 in.)	Operating-basis earthquake		500

Supports	Description	As-Built Conditions (lb)	Design Load (lb)
	Vertical + Z-Horizontal	90	
	Vertical + Z-Horizontal	90	
	Safe shutdown earthquake		750
	Vertical + Z-Horizontal	100	
	Vertical + Z-Horizontal	100	
N 405 horizontal (2 in.)	Operating-basis earthquake		150
	Vertical + Z-Horizontal	20	
	Vertical + Z-Horizontal	20	
	Safe shutdown earthquake		225
	Vertical + Z-Horizontal	30	
	Vertical + Z-Horizontal	30	
N 405 horizontal (3 in.)	Operating-basis earthquake		1,150
	Vertical + Z-Horizontal	210	
	Vertical + Z-Horizontal	210	
	Safe shutdown earthquake		1,725
	Vertical + Z-Horizontal	230	
	Vertical + Z-Horizontal	230	
N 405 horizontal (3 in.)	Operating-basis earthquake		400
	Vertical + Z-Horizontal	70	
	Vertical + Z-Horizontal	70	
	Safe shutdown earthquake		600
	Vertical + Z-Horizontal	80	

Supports	Description	As-Built Conditions (lb)	Design Load (lb)
	Vertical + Z-Horizontal	80	

a. Support load combination is seismic plus deadweight.

Table 3.9-27
LOADING COMBINATIONS AND STRESS LIMITS FOR SUPPORTS ON PIPING SYSTEMS

Loading Combination	**Stress Limits**
Normal	
D or (D + F + T)[a]	≤ Working Stress[b]
Upset	
D ± E or (D + F + T ± E)[a]	≤ Working Stress[b]
Faulted	
D ± E′ or (D + F + T_o ± E′)[a]	≤ Faulted Stress[c]
Deadweight and thermal are combined algebraically	
D = Deadweight T = Maximum operating thermal condition for system F = Friction load[d] E = OBE (inertia load + seismic differential support movement) E′ = SSE (inertia load + seismic differential support movement) T_o = Thermal - operating temperature	

a. For each loading condition, the greater of the two load combinations shall be used.

b. Working stress allowable per Appendix XVII of ASME Code, Section III.

c. Faulted stress allowable per Appendix XVII, Subsection NF, and Appendix F of ASME Code Section III, and Regulatory Guide 1.124. Safety Class 1 supports will be evaluated and designed in accordance with Regulatory Guide 1.124.

d. Whenever the thermal movement of the pipe causes the pipe to slide over any member of a support, friction shall be considered. The applied friction force applied to the support is the lesser of μ, W, or the force generated by displacing the support an amount equal to the pipe displacement.

   μ =0.35

   W =Normal load (excluding seismic) applied to the member on which the pipe slides.

Table 3.9-28
ANALYSIS OF TYPICAL PIPE SUPPORT BASE PLATES CALCULATED FOR IE BULLETIN 79-02

-	Existing Design				-	Replacement Design				-
-	Bolt Load		Bolt Capacity			Bolt Load		Bolt Capacity		
Support No.	Tension	Shear	Tension	Shear	Factor of Safety	Tension	Shear	Tension	Shear	Factor of Safety
ACH-106	75	0	7285	5760	97.0	75	0	14100	15195	188.0
ACH-118	241	293	7285	5760	11.9	241	293	14100	15195	27.5
SWAH-19	3161	1435	26880	26880	5.8	1452	975	14100	15195	6.0
SWAH-23	2963	1345	26880	26880	6.2	1257	897	14100	15195	6.8
SWAH-24	1972	895	26880	26880	9.4	837	597	14100	15195	10.1
SWCH-63	6	0	7285	5760	1121.0	7	0	11550	15195	1650.0
SWCH-73	18	0	7285	5760	399.0	19	0	11550	15195	608.0
SWCH-74	14	0	7285	5760	520.0	14	0	11550	15195	825.0
ACH-100	262	0	7285	5760	27.8	340	126	14100	15195	30.9
SWAH-37	499	220	7285	5760	9.4	455	250	14100	15195	20.5

Table 3.9-29
INTERNALS DEFLECTIONS UNDER ABNORMAL OPERATION

	Calculated Deflection (in.)	Allowable Limit (in.)	No loss of Function Limit (in.)
UPPER BARREL			
expansion/compression (to ensure sufficient inlet flow area / and to prevent the barrel from touching any guide tube to avoid disturbing the rod cluster control guide structure)	0.150	5	10
UPPER PACKAGE			
axial deflection (to maintain the control rod guide structure geometry)	0.005	1	2
ROD CLUSTER CONTROL GUIDE TUBE			
deflection as a beam (to be consistent with conditions under which ability to trip has been tested)	0.75	1.0	1.5
FUEL ASSEMBLY THIMBLES			
cross-section distortion (to avoid interference between the control rods and the guides)	0	0.035	0.072
Note: The values in this Table remains bounding for Extended Power Uprate (EPU).			

3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

3.10.1 SEISMIC QUALIFICATION CRITERIA

3.10.1.1 Original Criteria

At the time that Ginna Station was designed and constructed, critical electrical equipment was required by specification to be capable of withstanding the maximum seismic loads postulated for the plant site. Most components in the Class 1E electric power distribution system were designed to withstand forces due to electrical faults, which were much larger than the inertial forces due to a severe seismic event.

In the original design of Ginna Station, no in-structure response spectra were developed for the analysis of equipment. Instead, Seismic Category I items were qualified on an individual and often generic basis. Table 3.10-1 provides a list of items and the basis of seismic qualification for Ginna electrical equipment.

Seismic design requirements for Seismic Category I instrumentation and controls were originally specified in equipment specifications as follows:

A. Control room - The racks were assembled and the mounting and wiring of all components were designed such that the functions of the circuits or equipment would be performed in accordance with prescribed limits when subjected to seismic accelerations of 0.21g in the horizontal direction and in the vertical direction simultaneously. In addition, the mounting and wiring of all components were done such that simultaneous accelerations of 0.52g in the horizontal and vertical planes would not dislodge, cause relative movement, or result in any loss or change of function of circuits of equipment.

B. Containment and auxiliary building - The mounting and wiring of all components were designed such that simultaneous accelerations of 0.52g in the horizontal and vertical planes would not dislodge, cause relative movement, or result in any loss or change of function of circuits or equipment.

3.10.1.2 Current Criteria

When making modifications at Ginna Station, RG&E requires seismic qualification in accordance with the current standard when possible. When major Class 1E components that are independently anchored to Seismic Category I structures are designed and procured, it is done in accordance with the current seismic standard. This has resulted in an evaluation of seismic qualification in Ginna electrical equipment to increasingly severe standards including IEEE 344-1975.

The Systematic Evaluation Program (SEP) seismic input for determining the seismic design adequacy of mechanical and electrical equipment and distribution systems were based on in-structure (floor) response spectra for the elevations at which the equipment is supported. The floor spectra used in this reassessment, which are based on Regulatory Guide 1.60 spectra, are given in Section 3.7 (*Reference 7*). For electrical equipment, a composite 7% equipment damping was used in the evaluation for the 0.2g safe shutdown earthquake. For cable trays, the damping levels to be used in design depend greatly on the tray and support construction and the manner in which the cables are placed in the trays. Damping could be as high as 20% of critical damping. For structural evaluation, the stress criterion used was that the total stress must be less than or equal to the yield stress.

For the review of anchorage and support of safety-related electrical equipment in accordance with IE Bulletin 80-21, RG&E developed a program of inspection, analysis, testing, and modification, if necessary.

For the anchorage system of the electrical equipment, the required anchor load capacity as determined by the analysis phase, would be compared with the verified anchor load capacity for the anchor bolts associated with that component or assembly, as determined by the test and modification phase. If the verified anchor load capacity is found to be equal to or greater than the required anchor load capacity, then no modification would be required. However, if the verified anchor load capacity is found to be less than the required anchor load capacity for an electrical assembly, additional anchors would be added.

The analysis of each anchoring system to determine the minimum anchoring requirement to safely secure the equipment during a seismic event was to be performed using the following criteria and assumptions.

The static analysis described in Section 5.3 of IEEE 344-1975 was the basis for establishing shear and tensile stresses expected in the electrical equipment anchors being evaluated. Specifically, the seismic response of all floor-mounted equipment would be assumed to be the peak of the required response spectra for the equipment floor location, using damping values in accordance with Regulatory Guide 1.61, multiplied by a static coefficient of 1.5 to account for multifrequency and multimode responses. The inertial forces acting on the equipment center of mass would then be evaluated. A multianchor computer model would then be used to determine the shear and tensile stresses for all floor-mounted equipment. The stresses thus determined would establish the required anchor load capacity which would be compared to the verified anchor load capacity to establish anchor adequacy. Wall-mounted electrical equipment would be assumed to be rigid and the zero period acceleration values would be used to determine the seismic forces. The tensile and shear stresses would be calculated using the multianchor model.

3.10.2 SEISMIC QUALIFICATION OF ELECTRICAL EQUIPMENT AND INSTRUMENTATION

3.10.2.1 Introduction

The SEP Seismic Review Team selected electrical equipment representative of items installed in the reactor coolant system and safe shutdown systems at Ginna Station and evaluated them for structural integrity and electrical and mechanical functional operability. Electrical components that potentially have a high degree of seismic fragility were identified for review during a site visit by members of the team. A representative sample of components was selected for review by one of two methods:

A. Selection based on a walk-through inspection of Ginna Station by the SEP Seismic Review Team. Based on their experience, team members selected components as to the potential degree of seismic fragility for the component's category. Particular attention was paid to the component's support structure.

B. Categorization of the safe shutdown components into generic groups such as motor control centers and motors.

Rochester Gas and Electric provided seismic qualification data on the selected components from each group. Table 3.10-2 lists five components selected for review and includes the reasons for their selection. The details of the analyses and conclusions reached regarding the adequacy of these components is described in Sections 3.10.2.2 through 3.10.2.6.

3.10.2.2 Battery Racks

These racks were manufactured by Gould-National Battery Inc. The racks are seismically qualified in accordance with IEEE standard 344-1975 and RG&E site specific response spectra for floor elevation 253'-0". Rack design incorporates minimum cell spacing requirements imposed by the manufacturer.

3.10.2.3 Motor Control Centers 1L and 1M

A previous computer analysis was made of a Westinghouse type W ac motor control center which was originally tested at Wyle Laboratories in October 1972 to meet the seismic requirements recommended by IEEE Standard 344-1971. The calculations determined the acceleration levels and type of motion response that were excited in the equipment by a simultaneous horizontal and vertical sine beat type of motion input (5 cycles/beat). Subsequently, a similar dynamic analysis was made of the equipment as modified for Ginna, with attention focused on the new panelboard and distribution transformers.

The original Ginna response spectra, as specified for the safe shutdown earthquake condition, gave a total rms vector input acceleration of 0.79g calculated as 0.56 times the square root of the sum of the squares value of the following three components:

x-direction (front to rear) = 0.707 x 0.56g = 0.4g

y-direction (side to side) = 0.707 x 0.56g = 0.4g

z-direction (vertical) = 1.0 x 0.56g = 0.56g

The value of 0.56g was specified for the Ginna test. The Wyle Laboratories response spectra, on the other hand, gave a total rms vector input acceleration of 1.49g.

The response spectra at the auxiliary building platform and operating floor centers of gravity were compared to the Wyle Laboratories spectrum. Above 5 Hz, the acceleration levels throughout the equipment were greater when calculated for the 5 cycles/beat test at the 8.5 Hz fundamental natural frequency, compared to an envelope of the Ginna in-structure response spectra.

Based on review of the test results and comparison of input response spectra, as well as corresponding acceleration levels sustained in the equipment, it was concluded that the existing fragility level tests performed at Wyle Laboratories could be used to qualify the Ginna motor control centers, which have fundamental frequencies above 5 Hz.

3.10.2.4 Switchgear

The previous seismic qualification of Westinghouse type DB-50 reactor trip switchgear for Ginna was performed at the Westinghouse Astronuclear Laboratory. The reports present results of seismic simulation testing for the "low seismic" (safe shutdown earthquake peak acceleration not exceeding 0.2g) and "high seismic" (safe shutdown earthquake between 0.2g and 0.4g) classes of plants over the frequency range 1 to 35 Hz. The simulated seismic tests consisted of three elements:

A. Inputting a sine beat type acceleration to the base of the equipment being tested.

B. Monitoring the resulting accelerations at various locations in the equipment.

C. Monitoring the electrical functions of the equipment both during and after the tests to check for any loss of function.

Each sine beat of the vibration input consisted of 10 cycles of the test frequency with the amplitude of the beat (i.e., the acceleration of the vibration) increasing from a small value to the specified maximum value and returning to the initial value in sine wave fashion. The maximum required vertical input acceleration of the sine beat, as a function of test frequency for the "low seismic" plant classification, was 0.5g up to 10 Hz and reduced to a minimum value of 0.2g at 25 Hz. For horizontal excitation, the maximum required acceleration level of the sine beat was 0.8g up to 10 Hz and reduced to a minimum value of 0.2g at 25 Hz. Corresponding values for the "high seismic" plant classification were 0.93g up to 10 Hz, reducing to 0.32g at 25 Hz for vertical excitation and 1.4g up to 10 Hz, reducing to 0.5g at 25 Hz for horizontal excitation.

The applicable SEP reassessment response spectra for the switchgear were higher than both the "low seismic" and "high seismic" horizontal acceleration input curves for frequencies between 15 and 30 Hz. Based on the review of the tests performed at the Westinghouse Astronuclear Laboratory, it was concluded that the Westinghouse type DB-50 reactor trip switchgear would maintain its electrical function during a safe shutdown earthquake event. This conclusion was based on the assumption that there were no resonant frequencies in the 15 to 30 Hz range, or, if such resonances existed, that the response spectra developed from the sine beat test at the resonant frequency for 7% of critical damping enveloped the Ginna spectra (*Reference 1*).

3.10.2.5 Control Room Electrical Panels

The structural integrity of the main control board was evaluated for seismic loads for the safe shutdown earthquake as part of the SEP review (*Reference 3*). The seismic stresses were calculated using the modal response properties of the main control board determined by in-situ modal testing. A response spectrum analysis was used to calculate the seismic inertial load in each significant mode for three mutually perpendicular directions of eathquake motion. The inertial loads were then used in a static analysis to determine forces, moments, and stresses in critical elements of the seismic load path of the main control board. The results of the analysis indicated that the main control board would survive the safe shutdown earthquake. However, RG&E decided to provide some additional stiffeners and supports in order to enhance the structural integrity of the control board. These modifications were implemented in 1984.

3.10.2.6 Electrical Cable Raceways

The cable tray and conduit support anchors were installed using the manufacturers recommended procedures. As a result of SEP seismic review, a comprehensive testing and analysis program to demonstrate the seismic adequacy of electrical cable trays and conduit raceways of the type used in SEP plants was initiated by the SEP Owners Group. By letter of October 15, 1984, from R. M. Kacich, Chairman of the SEP Owners Group, to C. I. Grimes of the NRC (*Reference 4*), the SEP Owners Group responded to concerns relative to the seismic capability of cable trays as follows:

> The overall conclusion of the SEP cable tray test and evaluation program indicates that it is highly unlikely that any of the cable tray systems used in SEP plants will suffer structural collapse during a safe shutdown earthquake of the magnitude specified for eastern SEP plants. This conclusion is based on the fact that no system failures occurred in any of over 200 full-scale shake table tests of cable tray configurations selected, based on detailed plant walk-downs, as being typical of those in SEP plants. This conclusion is also supported by actual earthquake experience data from power plants and industrial facilities that have experienced strong motion earthquakes.
>
> Based on the results of the Owners Group efforts to date, it is concluded that the existing raceway systems in SEP plants possess substantial inherent seismic resistance and that the seismic qualification of raceway systems is not a significant safety issue. Therefore, no further work on this issue by the SEP owners is planned.
>
> As noted above, world-wide experience in power plants which have undergone significant earthquakes strongly supports the conclusion of the test and evaluation program. These experience data are expected to be documented as part of the ongoing efforts of the Seismic Qualification Utilities Group.

3.10.2.7 Constant Voltage Transformers

The constant voltage transformers are located in the battery rooms of the control building at elevation 253.7 ft. The constant voltage transformers are seismically qualified in accordance with IEEE Standard 344-1975 and RG&E site-specific response spectra for floor elevation 253.7 ft. Mounting requirements have been analyzed to this response spectra.

3.10.3 SEISMIC QUALIFICATION OF SUPPORTS OF ELECTRICAL EQUIPMENT AND INSTRUMENTATION

The SEP Seismic Review Team recommended that all safety-related equipment at Ginna Station be checked for adequately engineered anchorage; that is, the anchorage should be found to be adequate on the basis of analysis or tests employing design procedures (load stress and deformation limits, materials fabrication procedures, and quality acceptance) in accordance with a recognized structural design code.

Rochester Gas and Electric Corporation initiated a three-phase Seismic Action Plan (*Reference 5*) to provide assurance that the electrical equipment anchorage systems will perform their design function during the safe shutdown earthquake. Phase I consisted of inspection and preparation of as-built sketches for all safety-related electrical equipment as listed below. Anchor bolts used on this equipment were field inspected. As-built sketches were prepared showing all necessary information to perform Phase II. Phase II consisted of an analysis of each electrical equipment anchoring system, the results of which were compared to the test information. Phase III consisted of testing the anchor bolts and performing any resulting modifications required to upgrade the existing anchoring system to the criteria described in the analysis section of Phase II.

3.10.3.1 Equipment Addressed

The action plan included all Class 1E electrical systems and components. Certain Class 1E equipment installed during recent modifications in accordance with IEEE 344-1975 requirements was known to be seismically anchored and was not considered in the study.

The following electrical assemblies and/or components were evaluated by the Seismic Action Plan:

- Relay rack assemblies.
- 480-V 1E buses.
- 480-V (ac) 1E motor control center.
- 125-V (dc) 1E starters.
- Power panels.
- 1E battery racks.
- 1E battery chargers.
- Instrument racks.
- Control panels.
- Diesel-generator panels.
- Non-1E items (ancillary items).

All internally mounted components and devices weighing more than 25 lb were analyzed as separate assemblies. The results of the seismic evaluation program are described in *References 6* and *7*. The details are summarized in Section 3.10.3.2.

3.10.3.2 Raceway Anchorages

<u>3.10.3.2.1 Test Program</u>

All trays and conduit runs in the safety-related buildings had their anchorage systems inspected, tested, and, if required, reworked. No attempt was made to distinguish between Class 1E and non-1E raceways in any of the Seismic Category I structures.

Test criteria were established including the information necessary to test the anchorage of the supports making up the raceway system. Specific test procedures were prepared, consistent with the test criteria, for each category of anchorage included in the program. The categories of anchorages were

A. Expansion anchors for both conduit and tray supports in ceiling and/or wall locations.

B. Clips and unistrut hardware that rely on frictional resistance.

C. Embedded hardware such as keystone Q deck nuts, embedded unistrut, and poured-in-place anchors.

Detailed sketches of each of the embedded hardware type anchors are shown in Figures 3.10-1, 3.10-2, and 3.10-3.

The test program included all the hardware comprising the load path for each specific type of support. The bolts suspending the strut members to the ceiling or wall section were tested on a generic basis if they were the embedded hardware type and sample tested if they were shell anchors. The hardware used to attach the strut members to the anchor bolts and which rely on friction was also tested. Figure 3.10-4 shows the various generic strut support configurations in use at Ginna Station that were part of the friction bolt testing program.

3.10.3.2.2 Test Loads

In order to establish test load per bolt requirements for the shell anchors and embedded anchors, the original plant specification for cable trays was consulted. Section 4 of Specification SP-5375, (*Reference 8*), specifies the design load for the cable tray type as 100 lb/ft. This load, applied to any of the specified cable tray widths, should produce no more than 0.25 in. deflection at midspan when calculated on a simple beam basis. In addition to the tray loads, the supports were designed to carry a 200-lb person standing at any position in the tray. The design span lengths were assumed to be 8 ft. The 8-ft span lengths carry a total load of 800 lb between supports or 4000 lb for a stack of five trays. Two vertical members were assumed per support. A 2000-lb test load was used on each vertical support member to test the anchorages.

The test load for the frictional anchors was based on the manufacturer's design manual, Unistrut General Engineering Catalog No. 9 (*Reference 9*). The design torque values for various bolt sizes needed to maintain a resistance to slippage of at least 1500 lb for a 1/2-in. bolt used on P1000 strut were determined to be as follows:

1.

	Bolt Size			
	1/4 in.	5/16 in.	3/8 in.	1/2 in.
Torque (ft-lb)	6	11	19	50

The torque values shown above were used in the test procedures for qualifying the unistrut stud/nut hardware assemblies and includes a minimum safety factor of 3.

3.10.3.2.3 Expansion Anchor Test Results

Expansion anchors were selected for testing by inspecting and testing 25% of the cable tray vertical support members using shell type anchors and 10% of the rigid conduit supports using shell anchors. The lower sampling rate for conduit was used since all Class 1E conduit is rigid and has a very low design load. However, the 2000-lb test load was used on conduit anchors. All expansion anchors were tested on each of the sample supports.

The selected anchors were inspected and load tested to 2000 lb in accordance with RG&E Ginna Station Procedures. The acceptance criteria is that the shell anchors hold the required load without excessive movement.

The results of the shell anchor testing program are summarized in Table 3.10-3.

3.10.3.2.4 Frictional Anchor Test Results

The unistrut stud/nut testing criteria (frictional anchors) used were as follows:

A. All accessible unistrut stud nuts used for cable tray supports were tested. The total number of Class 1E supports is shown in Table 3.10-4.

B. The unistrut nuts/bolts that were tested were those used to attach the strut members to the ceiling Q deck bolts or angle clips. These attachments rely on friction and must be torqued to at least a minimum value which was established to ensure a safety factor of at least 3. Figure 3.10-4 shows the various configurations of strut supports used throughout Ginna Station. The unistrut joints affected by the procedures are marked by an arrow.

C. The "as-found" torque of all the unistrut stud nuts on a particular support was recorded. All inaccessible bolts were identified and recorded. Torque wrench adapters (i.e., crow's foot) were used to reduce the number of inaccessible nuts or bolts. Those bolts still inaccessible were wrench-tightened where possible.

D. The design torque values for the various bolt sizes were derived from the following manufacturer's data:

	Bolt Size			
	1/4 in.	5/16 in.	3/8 in.	1/2 in.
Torque (ft-lb)	6	11	19	50

If the "as found" torque values were less than the minimum values specified by the manufacturer then the proper torque values were applied to each bolt. Both the as-found and final torque values were recorded.

All accessible supports were tested. The results of the friction bolt testing program are summarized in Table 3.10-4.

3.10.3.2.5 Embedded Anchor Test Results

The keystone steel decking test criteria (embedded hardware anchors including embedded unistrut and poured-in-place anchors) were developed and the following generic test was performed to ensure that the load capacity of the Q deck was sufficient to sustain the required loads. Fourteen in-situ tests were performed at different plant locations. These locations were in convenient open areas and not in an actual support location. Ten in-situ unistrut and 12 poured-in-place anchor tests were also completed.

The results of the embedded anchor programs are summarized in Table 3.10-5.

3.10.3.3 Class 1E Equipment Anchorage Qualification Program

As-built drawings were prepared for 115 electrical assemblies. These drawings represent all Class 1E and non-1E equipment which are floor-mounted, mounted on structural steel, poured wall mounted or block wall mounted. Each drawing lists the size, shape, number, and type of existing anchor bolts for a particular assembly. This information was obtained from field measurements.

The weights were assessed based on the area, gauge size of the enclosure steel, and the weights of all the internally mounted components, including wire and terminal blocks. The total equipment weights were then determined including 25% of the enclosure weight for conservatism.

The minimum loading that the existing anchorage must be capable of carrying during a seismic event (safe shutdown earthquake) at Ginna Station was determined during this program. The calculated loads (tensile and shear) were compared to the published load capabilities for the specific anchors used on each assembly. If the calculated load values were within the published capability of the bolts used on a particular assembly, then the calculated loads were used as the test loads for that assembly, provided the bolts were accessible. For wall-mounted equipment that had safety factors in excess of 10, no modification or testing was performed. If it was determined that the existing anchorages were inadequate, then those assemblies were modified taking no credit for the existing anchors.

The horizontal and vertical forces were determined by using one-and-a-half times the peak acceleration shown on the floor response spectrum for each assembly location. All proposed expansion anchor bolts used a minimum safety factor of 5.7 in tension and 4 in shear.

The final phase of the program involved the installation of generic modifications using specific construction drawings for each assembly to be modified. A typical generic modification included the welding of structural plates or angles to the outside of the enclosure frame, the installation of hilti bolts or through bolts depending on location, and the stitch welding of the enclosure cabinets to the frames.

Non-class 1E evaluations were conducted for those assemblies permanently mounted in Seismic Category I buildings that are not safety-related. The anchorage acceptance criteria for those assemblies were the same as for the Class 1E assemblies.

Internally mounted components were categorized and a generic design analysis was developed to evaluate the methods of attaching these components to the cabinets. If any one component is classified Class 1E in an enclosure, then all components were assumed to be Class 1E.

Non-class 1E enclosures were not surveyed. It was assumed that the enclosure will retain any loose component during a safe shutdown earthquake.

3.10.3.4 Conclusions

The NRC has reviewed the RG&E report of the upgrading of anchorage and support of safety-related electrical equipment (*Reference 6*) and concluded that the electrical equipment anchorage design and internal mounted devices and component evaluations and modifications were adequate (*Reference 2*). The required modifications have been completed as designed.

3.10.4 FUNCTIONAL CAPABILITY OF COMPONENTS

The NRC initiated a generic program to develop criteria for the seismic qualification of equipment in operating plants as an Unresolved Safety Issue (USI A-46). Under this program, an explicit set of guidelines (or criteria) to be used to judge the adequacy of the seismic qualifications (both functional capability and structural integrity) of safety-related mechanical and electrical equipment at all operating plants was developed.

The NRC Staff as a result of the seismic review of the R. E. Ginna Nuclear Power Plant has concluded that, since the ground response spectrum (0.2g Regulatory Guide 1.60 spectrum) used for Ginna seismic reevaluation envelops the Ginna site-specific ground response spectrum, additional safety margins in the structures, systems, and components do exist for resisting seismic loadings. The staff also concluded that Ginna Station has an adequate seismic capacity to resist a postulated safe shutdown earthquake, and there is reasonable assurance that the operation of the facility will not endanger the health and safety of the public. (*Reference 2*).

RG&E submitted the Ginna Station response to USI A-46 in January of 1997 (*Reference 13*). In June of 1999 the NRC issued a Safety Evaluation Report (SER) accepting RG&E's analysis and modifications (*Reference 16*).

3.10.5 SEISMIC CATEGORY I TUBING

3.10.5.1 Codes and Standards

The original design of Seismic Category I tubing and tubing supports at Ginna Station was performed to then current (1967) standard industry practice, which was based on the experience of the journeyman instrument installer and did not require conformance to specific industry codes or standards.

Current (1988) design requirements for Seismic Category I tubing and supports include the following:

3.10.5.1.1 Tubing Design Requirements

Instrument Standard of America Standard ISA-S67.02 and Regulatory Guide 1.151 (*References 10* and *11*) are used as guidance for the design, fabrication, installation, and testing of tubing.

Tubing is designed using the stress evaluation equations contained in ANSI B31.1 (1973) with allowable stress limits as included in Table 3.10-6 except that the stress intensification factor, I, applicable to bending moments is taken equal to 1.3 for all joint and fitting configurations because of the relatively low allowable stress permitted by Table 3.10-6 compared to ASME Section III allowables.

Welder qualifications, welding, and examination procedures are in accordance with:
ASME Sections III, V, VIII and XI code; 2004 Edition with no Addenda.
ASME Section IX code; current Edition and Addenda.
ASME Section XI code; 2004 Edition with no Addenda for IWE Containment (metallic liner).
ANSI/ASME B31.1 Power piping; 2004 Edition with no Addenda.

The loads and load causing phenomena to be considered in the qualification and design of tubing shall include the following.

- Dead weight.
- Pressure.
- Temperature.
- Seismic inertia.
- Support motions due to
    1. Thermal.
    2. Seismic.

3.10.5.1.2 Tubing Supports Design Requirements

Tubing supports are standard manufactured tubing supports (clips or clamps) plus any auxiliary steel used to protect tubing (channels) and provide a support path to the building structure. Tubing supports that attach the tubing to auxiliary or building steel shall be standard manufactured tubing supports qualified for their intended use by load rating using the procedure contained in ASME Code Section III-NF-3380, Design by Load Rating, 1986 edition.

Channels or other structural steel used to protect and support tubing and other auxiliary steel used in the tubing support path to the building structure shall be designed to the AISC specification given in *Reference 12* for the limiting loads developed from the spacing tables and charts or as otherwise calculated for individual tubing runs evaluated by analysis. The particular loads and load-causing phenomena used to design supports are the same as given above for tubing, except for pressure. Allowable stresses for the load combinations identified are given in *Reference 12*.

Tubing spans in space, in those areas adjacent to normal personnel access (i.e., within 7 ft 0 in. height of platforms, floor walkway areas, etc.), over 3 ft 0 in. in length, shall be contained in channels or similarly supported or protected against potential damage.

3.10.5.2 Load Conditions

3.10.5.2.1 Tubing

The tubing shall be analyzed for the following loading conditions:

A. Design condition - deadweight plus design pressure.

B. Severe environmental condition$_{(1)}$ - deadweight plus operating pressure plus OBE (inertia).

C. Severe environmental condition$_{(2)}$ - deadweight plus operating pressure plus OBE (inertia) plus OBE (SAM) displacements plus maximum operating thermal effects including thermal support motions.

D. Extreme environmental condition - deadweight plus operating pressure plus SSE (inertia).

E. Abnormal condition - deadweight plus operating pressure plus loss-of-coolant-accident induced thermal effects (application limited to inside containment).

3.10.5.2.2 Tubing Supports

The tubing system supports will be evaluated to the following combinations of tubing system imposed loads:

A. Severe environmental condition$_{(1)}$ (Equation 4 of Table Q1.5.7.1 of *Reference 12*):

   Deadweight plus OBE (inertia).

B. Severe environmental condition$_{(2)}$ (Equation 6 of Table Q1.5.7.1 of *Reference 12*):

   Deadweight plus maximum operating thermal including restraint of free end displacement and thermal support motions plus OBE (inertia) and (seismic anchor motion) effects.

C. Extreme environmental condition (stress limit coefficient from Table Q1.5.7.1 is 1.6, Equation 8 of *Reference 12*):

   Deadweight plus SSE (inertia).

D. Abnormal (stress limit coefficient from Table Q1.5.7.1 is 1.7, Equation 11 of *Reference 12*) (application limited to inside containment):

   Deadweight plus maximum accident thermal including restraint of free end displacement and thermal support motions.

Included in the design of horizontally run channels provided to protect or support tubing runs defined as deadweight shall be a requirement to support an external vertical load of 50 lb, to protect the tubing during construction and normal plant maintenance, placed to cause the highest bending and shear stresses in the channel.

3.10.5.3 Routing Requirements

Instrument sensing lines shall be routed to prevent violating required separation between redundant instrument channels. Separation between redundant instrument sensing lines shall be provided by free air space or barriers, or both, such that no single failure can cause the failure of more than one redundant sensing line.

The minimum separation between redundant instrument sensing lines shall be at least 18 in. in air, in nonmissile, non-high-energy jet stream, non-pipe-whip or nonhostile areas. As an alternative, a suitable barrier shall be used, which extends at least 1 in. beyond the line of sight between redundant sensing lines and shall be designed and mounted to Seismic Category I requirements. In hostile areas potentially subject to high-energy jet stream, missiles, and pipe whip, the separation shall be provided by space in air, steel or concrete barriers, or both, and documented with analyses or calculations as necessary to prove that the separation protects the redundant sensing lines from failure due to a common cause. All barriers shall be designed and mounted to Seismic Category I requirements.

Instrument sensing lines shall be run along walls, columns, or ceilings whenever practical, avoiding persons supporting themselves on the lines or damage of the sensing lines by pipe whip, missiles, jet forces, or falling objects.

Supports, brackets, clips, or hangers shall not be fastened to the instrument sensing lines for the purposes of supporting cable trays or any other equipment.

Routing of the nuclear-safety-related instrument sensing lines shall ensure that the function of the lines is not affected by vibration, abnormal heat, or stress.

REFERENCES FOR SECTION 3.10

1. R. C. Murray, et al., Seismic Review of the Robert E. Ginna Nuclear Power Plant as Part of the Systematic Evaluation Program, NUREG/CR-1821, November 15, 1980.

2. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: SEP Safety Topics III-6, Seismic Design Consideration and III-11, Component Integrity, dated January 29, 1982.

3. Letter from J. E. Maier, RG&E, to D. M. Crutchfield, NRC, Subject: SEP Topic II-6, Seismic Considerations (Seismic Structural Evaluation of the Main Control Board), dated January 9, 1984.

4. Letter from R. M. Kacich, SEP Owners Group, to C. I. Grimes, NRC, Subject: SEP Topic III-6, Seismic Design Considerations, SEP Owners Group Cable Tray/Conduit Test Program, dated October 15, 1984.

5. Letter from L. D. White, Jr., to D. L. Ziemann, NRC, Subject: The Seismic Action Plan, Anchorage and Support of Safety-Related Electrical Equipment, dated February 11, 1980.

6. Letter from J. E. Maier, RG&E, to D. M. Crutchfield, NRC, Subject: Anchorage and Seismic Support of Safety-Related Electrical Equipment, Final Report, dated December 22, 1980.

7. Letter from J. E. Maier, RG&E, to D. M. Crutchfield, NRC, Subject: Anchorage and Support of Safety-Related Electrical Equipment, Final Report, dated February 27, 1981.

8. Gilbert Associates, Inc., Cable Trays and Electrical Circuits Power, Control and Instrumentation, Ginna Station Unit No. 1, Technical Specification SP-5375, dated March 17, 1967.

9. Unistrut General Engineering Catalog No. 9, Unistrut Corporation, Wayne, Michigan.

10. Instrument Society of America, Nuclear Safety-Related Instrument Sensing Line Piping and Tubing Standard - 1980 for Use in Nuclear Power Plants, ISA-S67.02, 1983.

11. U.S. Nuclear Regulatory Commission, Instrumentation Sensing Lines, Regulatory Guide 1.151, July 1983.

12. American Institute of Steel Construction, Nuclear Facilities - Steel Safety-Related Structures for Design, Fabrication, and Erection, Specification ANSI/AISC N690, 1984.

13. Letter from R. C. Mecredy, RG&E, to G. S. Vissing, NRC, Subject: Resolution of Generic Letter 87-02 Supplement 1 and 88-20 Supplements 4 and 5, dated January 31, 1997.

14. Letter from R. C. Mecredy, RG&E, to G. S. Vissing, NRC, Subject: Response to NRC "RAI" on USI A-46, May 27, 1998.

15. Letter from R. C. Mecredy, RG&E, to G. S. Vissing, NRC, Subject: Response to NRC second "RAI" on USI A-46, dated February 2, 1999.

16. Letter to R. C. Mecredy, RG&E, from G. S. Vissing, NRC, Subject: Plant Specific Safety Evaluation Report for USI A-46, dated June 17, 1999.

Table 3.10-1
MAJOR CLASS 1E COMPONENTS AND THE BASIS FOR SEISMIC QUALIFICATION

		System/Component	Basis for Seismic Qualification
I.		**EMERGENCY POWER SYSTEM**	
	A.	Low voltage (600-V) switchgear (excluding unit transformer) (Westinghouse DB 15, 25, 50, and 75 breakers)	Post-construction testing.
	B.	Motor control centers (Westinghouse type W)	Post construction testing and analysis in accordance with IEEE 344-1971. Upgraded by analysis to IEEE 344-1975.
	C.	Motor-operated valve operators (ac/dc)	Post-construction testing.
	D.	Vital 120-V ac Distribution panels 1A and 1C Inverters (Solidstate Controls, Inc.) Constant voltage transformers (CVT)	Postconstruction testing. Installed in 1978 qualified by test in accordance with IEEE 344-1975. CVTs qualified to IEEE 344-1975.
	E.	125-V dc power system 125-V, 60-cell batteries (Gould) and racks Battery chargers	Design specification; 0.52g simultaneous horizontal and vertical. Racks qualified to IEEE 344-1975. Battery cells qualified to IEEE 344-1987.
	F.	Diesel generators (Alco/Westinghouse)	Design specification; 0.47g simultaneous horizontal and vertical acceleration.
	G.	Reactor building cable penetrations (Crouse-Hinds)	Postconstruction testing.
	H.	Conduit supports and tray supports	SEP Owners Group.
	I.	Electrical equipment anchors	Modification program.
II.		**SAFEGUARDS INSTRUMENTATION AND CONTROL**	
	A.	Transmitters (Barton, Foxboro)	Post-construction testing.
	B.	Reactor trip switchgear (DB 50)	Post-construction testing.
	C.	Main control board (Wolf and Mann)	Design specification; 0.52g simultaneous horizontal and vertical acceleration.
	D.	Reactor trip system racks (A/D conversion)	Design specification; 0.52g simultaneous horizontal and vertical acceleration. Modification to racks.

	System/Component	Basis for Seismic Qualification
E.	Protective relay racks (safety injection and reactor trip logic)	Design specification; 0.52g simultaneous horizontal and vertical acceleration.
F.	Safeguards racks (engineered safety features actuation (ESFAS output)	Design specification; 0.52g simultaneous horizontal and vertical acceleration.
G.	Control switches (Westinghouse type W2 and OT2)	Post-construction testing.

Table 3.10-2
ELECTRICAL COMPONENTS SELECTED FOR SEISMIC REVIEW

Item Description	Reason for Selection
Battery racks	Evaluate capacity of the bracing to develop lateral load capacity.
Motor control centers	Typical seismically qualified electrical equipment. Functional design adequacy may not have been demonstrated. Check anchorage to floor structure.
Switchgear	Same as motor control centers.
Control room electrical panels	The control panels appear to be adequately anchored at the base. However, there is a need to check components which are cantilevered off of the front panel and to check front panel stiffness.
Electrical cable raceways	The cable tray support systems did not have any specific seismic qualification testing.

Table 3.10-3
SHELL ANCHOR TEST SUMMARY

Location	Total Number of Anchors	Number of Anchors That Held Load	Number of Anchors That Did Not Hold Load	Inaccessible
Auxiliary building basement floor	11	11	0	0
Auxiliary building intermediate floor	16	16	0	0
Screen house basement floor	9	9	0	0
Cable tunnel ceiling	5	5	0	0
Containment building basement	2	2	0	0
Relay room	6	5	0	1
Battery rooms	4	4	0	0
Diesel-generator pits	22	21	1	0
Total	75	73	1	1

Table 3.10-4
FRICTION BOLT TEST RESULT SUMMARY

Location	Total Number of Bolts	Acceptable Torque	Bolts Wrench Tightened	Bolts Not Accessible
Auxiliary building basement floor	227	217	1	9
Auxiliary building intermediate floor	202	133	17	52
Intermediate building, elevation 271 ft 0 in	28	14	2	12
Intermediate building, elevation 278 ft 4 in	320	305	11	4
Screen house basement floor	144	142	2	0
Cable tunnel	649	532	15	102
Relay room	361	315	1	45
Battery rooms	215	213	0	2
Diesel-generator pits	84	84	0	0
Containment basement floor	112	112	0	0
Containment intermediate floor	338	337	0	1
Total	2680	2404	49	227

Table 3.10-5
CATEGORY 3 ANCHORS TEST SUMMARY

Location	Number of Poured-In-Place Tested	Unistrut Tests	Q-Deck Tests	Total Tests	Held Load	Did Not Hold Load
Auxiliary building basement floor	0	2	0	2	2	0
Auxiliary building intermediate floor	0	2	0	2	2	0
Intermediate building, elevation 271 ft 0 in	0	0	2	2	2	0
Screen house basement floor	0	2	0	2	2	0
Containment basement floor	0	2	2	4	4	0
Containment intermediate floor	12	2	2	4	4	0
Relay room	0	0	2	2	2	0
Battery rooms	0	0	6	6	6	0
Total	12	10	14	24	24	0

Table 3.10-6
STRESS LIMITS FOR TUBING

Condition	**Stress Limits**
Design	$P_m + P_b \le S_h$
Severe environmental $_1$	$P_m + P_b \le 1.2\ S_h$
Severe environmental $_2$	$P_m + P_b + P_e + P_{SAM} \le (S_h + S_A)$
Extreme environmental	$P_m + P_b \le 1.8\ S_h$
Abnormal [a]	$P_m + P_b + P_e + P_{AAM} \le$ the stress limit for system operability

Where

P_m =	Primary general membrane stress; $P\ D_o / 4\ t_n$
P_b =	Primary bending stress; M_i / Z and M_T / Z
$S_{A'}$, $S_{h'}$, S_e =	Allowable stress from ANSI B31.1 Code for material at design temperature
P_e =	Restraint of free end displacement (thermal and differential support motion stress)
P_{SAM} =	Stresses due to differential OBE seismic support motions
P_{AAM} =	Stress due to accident-induced support motions
M_T =	Torsional moment on pipe
M_i =	Bending moment on pipe

a. Application limited to inside containment.

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

3.11.1 BACKGROUND

3.11.1.1 Initial Design Considerations

Section 6.1.2 discusses environmental considerations in the selection of engineered safety features materials. Sections 6.2.2.1, 6.3.2.1, and 6.5.1.2 discuss environmental protection design features for components of the containment ventilation (containment recirculation fan cooler), emergency core cooling, and containment air filtration systems located inside containment.

3.11.1.2 Review of Environmental Qualification of Safety-Related Electrical Equipment

The review of the environmental qualification of safety-related electrical equipment for Ginna Station was initiated in 1977 under Topic III-12 of the Systematic Evaluation Program (SEP). In February 1980, the NRC redirected the review program for SEP plants and provided Division of Operating Reactors (DOR) guidelines for evaluating environmental qualification and for identifying safety-related equipment for which environmental qualification was to be addressed (*Reference 1*). On June 25, 1982, the NRC issued an interim regulation (*Reference 2*), which suspended the June 30, 1982, deadline for qualification of electrical equipment pursuant to the DOR Guidelines and NUREG 0588. Subsequently, 10 CFR 50.49 was issued (February 22, 1983).

Ginna Station submitted the initial report concerning the environmental qualification of electrical equipment by letter, dated February 24, 1978 (*Reference 3*). This submittal was reformatted and resubmitted on December 1, 1978 (*Reference 4*). It was revised and resubmitted again on April 25, 1980 (*Reference 5*), and on October 31, 1980 (*Reference 6*). On June 1, 1981, the NRC issued its Safety Evaluation Report (SER) for the Environmental Qualifications of Safety-Related Electrical Equipment at the R. E. Ginna Nuclear Power Plant (*Reference 7*). The letter included the SER by the Office of Nuclear Reactor Regulation (NRR), the Draft Interim Technical Evaluation Report (TER C5257-178) by the NRC Consultant, Franklin Research Center, and a request that Ginna Station provide additional information. Ginna Station responded to the June 6, 1981 SER by letters dated September 4, 1981 (*Reference 8*), November 6, 1981 (*Reference 9*), and February 18, 1982 (*Reference 10*). The NRC transmitted an SER by the NRR, and a Technical Evaluation Report by Franklin Research Center, TER C5257-454, on December 13, 1982 (*Reference 11*), based on RG&E responses in *References 8*, *9*, and *10*. Rochester Gas and Electric Corporation provided additional information in *References 12*, *13*, *14*, and *15*. In the responses (*Reference 16*) to NRC Generic Letter 84-24, RG&E certified program compliance with 10 CFR 50.49. It was also noted that the Environmental Qualification Program is not adversely impacted by the IE bulletins and notices listed in Generic Letter 84-24. In *Reference 17*, the NRC concluded that the Environmental Qualification Program complies with the 10 CFR 50.49 and that the issues raised in *Reference 11* are satisfactorily resolved.

Based on the DOR guidelines, the Ginna Station Environmental Qualification Program addresses the safety-related electrical equipment which must function to mitigate the consequences of loss-of-coolant accidents (LOCA) or high-energy line breaks inside or outside containment and whose environment would be adversely affected by the accident.

3.11.2 EQUIPMENT IDENTIFICATION

In accordance with the DOR guidelines, Ginna Station was directed to establish a list of systems and display instrumentation needed to mitigate the consequences of a LOCA or high-energy line break inside or outside containment and to reach a safe shutdown. The display instrumentation selected includes parameters to monitor overall plant performance as well as to monitor the systems on the list. The list of systems was established on the basis of the functions that must be performed for mitigation of the consequences of a LOCA or high-energy line break and to effect safe shutdown without regard to the location of the equipment relative to a potentially hostile environment. The systems considered were those required to achieve or support (1) emergency reactor shutdown, (2) containment isolation, (3) reactor core cooling, (4) containment heat removal, (5) core residual heat removal, and (6) prevention of significant releases of radioactive material to the environment. The list of equipment requiring environmental qualification is included in the Ginna Station October 31, 1980, report (*Reference 6*), as supplemented in *References 8 through 10* and *12 through 14*. The current "Master List" relative to 10 CFR 50.49 is contained in a plant procedure.

3.11.3 IDENTIFICATION OF LIMITING ENVIRONMENTAL CONDITIONS

This section defines the bases for and references to the environmental conditions encountered throughout the plant. A tabular summary is provided in Table 3.11-1.

3.11.3.1 Inside Containment

<u>3.11.3.1.1 Post Loss-of-Coolant Accident Environment</u>

Postaccident environmental conditions inside containment are discussed in Section 6.1.2.1. The limiting conditions resulted from LOCA analyses. The temperature and pressure profiles are given in Figures 6.1-1 and 6.1-2 with peak values being 286°F and 60 psig, respectively. The radiation environment for Ginna Station is presented in Figures 6.1-4 and 6.1-5 from data in Tables 3.11-2 and 3.11-3. Material compatibility with postaccident chemical environment is also discussed in detail in Section 6.1.2.1. For a LOCA, containment conditions were analyzed as part of SEP Topic VI-2.D by the Lawrence Livermore National Laboratory for the NRC (*Reference 18*). It was determined that the peak pressure is 59.3 psig, which is less than the design pressure of 60 psig. In the long term (10,000 to 20,000 sec), the containment temperature stays above the original environmental qualification envelope (250°F versus 225°F). The Ginna Station limiting temperature has thus been increased accordingly. The NRC determined that this small variation had no effect on the qualification status of Ginna Station equipment. The peak temperature of 285.26°F is also less than the design temperature of 286°F. *Reference 36* covers the impact of Extended Power Uprate (EPU).

An evaluation was performed to determine the effect of the BWI replacement steam generators (RSGs) at Ginna Station on the results of the containment response following a LBLOCA. The RSGs have approximately 0.9 percent more mass in the primary system than the original steam generators (OSGs). This would cause the peak reactor building pressure and temperature to increase by approximately 0.5 psi and approximately 1 °F, respectively.

The peak pressure and temperature remain below the acceptance criteria of 60 psig and 286 °F, respectively.

Figure 3.11-1 is of historical interest and shows the nomogram reproduced from Appendix B of the DOR Guidelines. Ginna Station (Pre-uprate power level 1520 MWt, containment volume 997,000 ft^3) is represented by the line shown in Figure 3.11-1.

In June 1984, the NRC issued Revision 1 to Regulatory Guide 1.89. Appendix D of Regulatory Guide 1.89, Revision 1, provides a methodology for determining the qualification radiation dose.

A comparison of the detailed assumptions in developing the dose information contained in Tables D-1 and D-2 of Regulatory Guide 1.89, Revision 1, (reproduced as Tables 3.11-4 and 3.11-5) and Ginna Station is shown in Table 3.11-6.

Although the Ginna Station fan coolers have iodine removal capability, no credit is taken for iodine removal by the filters for conservatism.

The dose rate at the centerline of containment in Tables 3.11-4 and 3.11-5 was determined by the specific activity of the containment atmosphere (i.e., curies/cubic feet). The specific activity is directly proportional to the reactor power level and inversely proportional to the containment volume. The specific activity and therefore the containment centerline dose rate for Ginna Station assuming reactor power of 1811 MWt (or 102% of 1775 MWt which takes into account power measurement uncertainties and is consistent with assumptions used in Section 15.6) is shown below. The equation includes a 4% on reactor power to accomodate variations in the fuel management schemes, a conservative estimate for containment free volume of 997,000 ft^3, and a time dependent scaling factor to address the difference in the fuel cycle length (SF_{BURNUP}).

(1811 MWt/ 4100 MWt) x 1.04 x (2,520,000 ft^3 / 997,000 ft^3) × SF_{BURNUP} x tabulated values shown in Tables 3.11-4 and 3.11-5

or

1.161 x SF_{BURNUP} x the tabulated values of Tables 3.11-4 and 3.11-5.

The time-dependent dose at the containment centerline of Ginna Station is contained in Tables 3.11-2 and 3.11-3.

Reference 38 through 42 cover containment radiation dose due to EPU.

Submergence of valves inside containment is discussed in *Reference 19* where it has been shown that operation following submergence is not required. Submergence of instrumentation is discussed in *Reference 20*. All instrumentation required to function for postaccident monitoring has been elevated to prevent submergence with the exception of two resistance temperature detectors (RTDs) for the reactor vessel level indication system, which included submergence in their environmental qualification.

3.11.3.1.2 Post Main Steam Line Break Environment

The peak pressure following a main steam line break is contained in Section 6.2.1.2.3.2. The temperature associated with the main steam line break is higher than that of the LBLOCA, but was determined by the NRC not to be limiting, however, for qualification of equipment required following a main steam line break because

A. The high temperature transient is very brief and there is super-heated steam (with a lower heat transfer capability), as opposed to saturated steam.

B. The equipment is protected from the direct effects of the steam line break by concrete floors and shields.

C. The sensitive portions of the electrical equipment are not directly exposed to the environment but are protected by housing, cable jackets, and the like.

For these reasons, the humidity and steam environment following a LOCA remains limiting. This is consistent with the NRC Position 4.2 of the Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors. Radiation levels in containment following a main steam line break are not limiting since fuel failures are not projected to result from a main steam line break. Chemical environment and submergence are bounded by the LOCA conditions.

The NRC further examined a generic issue concerning main steam line break with continued feedwater addition. In a February 9, 1983, SER (*Reference 21*) the NRC concluded that the results of SEP Topic VI-2.D calculations were acceptable because (1) the main feedwater system is automatically isolated and the preferred auxiliary feedwater system limits flow to the steam generators, (2) the preferred auxiliary feedwater pumps are protected from the effects of runout flow, and (3) all potential water sources were identified and although a reactor return to power would occur, there is no violation of specified acceptable fuel design limits.

3.11.3.2 Auxiliary Building

3.11.3.2.1 Heating, Ventilation, and Air Conditioning

The auxiliary building has a heating, ventilation, and air conditioning system which provides clean, filtered, and tempered air to the operating floor of the auxiliary building. Air from within the Auxiliary Building sweeps the surface of the decontamination pit and spent fuel storage pool. The system exhausts air from the equipment rooms and open areas of the auxiliary building, and from the decontamination pit and spent fuel pool (SFP) through a closed exhaust system. The exhaust system includes a 100%-capacity bank of high efficiency particulate air filters and redundant 100%-capacity fans discharging to the atmosphere via the plant vent. The auxiliary building ventilation system (ABVS) is included in Drawings 33013-1869 through 33013-1872 and is discussed in Section 9.4.2. This arrangement ensures the proper direction of air flow for removal of airborne radioactivity from the auxiliary building.

Included in the auxiliary building exhaust system is a separate charcoal filter circuit, which exhausts from rooms where fission-product activity may accumulate during MODES 1 and 2 in concentrations exceeding the average levels expected in the rest of the building. Although no credit for this system is assumed in the plant safety analysis, this circuit is capable of providing exhaust ventilation from the areas containing pumps and related piping and valving which are used to recirculate containment sump liquid following a LOCA. A full-flow charcoal filter bank is provided in the circuit, along with two 50%-capacity exhaust fans. The air-operated suction and discharge dampers associated with each fan are interlocked with the fan such that they are fully open when the fan is operating and fully closed when the fan is stopped. These dampers fail to the open position on loss of control signal or control air. The fans discharge to the main auxiliary building exhaust system containing the high efficiency particulate air (HEPA) filter bank. To ensure a path for the charcoal (and HEPA) filtered exhaust to the plant vent if the main exhaust fans are not operating, a fail-open damper is installed in a bypass circuit around the two main exhaust fans. In addition to the main auxiliary building ventilation system (ABVS), the residual heat removal, safety injection, containment spray, and charging pump motors are provided with additional cooling provisions when the pumps are operating. The safety injection and containment spray pump motors are located in an open area in the basement of the auxiliary building and share three service-water-cooled heat exchangers. In 1992, service water to these heat exchangers was blanked off (see Section 9.4.9.1). The charging pumps and residual heat removal pumps are located in individual rooms, each room being provided with two cooling units consisting of redundant fans, water-cooled heat exchangers, and ductwork for circulating the cooled air. The capacity of each charging pump cooling unit is sufficient to maintain acceptable room-ambient temperatures with the minimum number of pumps required for system operation in service. The cooling units in the residual heat removal pump pit are not required for the operation of the residual heat removal pumps, even if both pumps are operating.

In the event of a loss of offsite power, the auxiliary building ventilation system (ABVS) main supply and exhaust fans would be inoperable. However, all other fans in the auxiliary building ventilation system (ABVS) are supplied by emergency diesel power, including the pump cooling circuits for safety-related pump motors, as described above. Analysis has shown that the three levels of the auxiliary building and the residual heat removal pump pit would remain within acceptable limits when the outside air was at its maximum expected temperature and there were no cooling units operating. Since the auxiliary building is a very large volume building, it is not expected that there would be a significant postaccident temperature increase except in some local areas near hot piping and large motors. This situation exists in the basement of the auxiliary building where the safety-related pumps and recirculated sump fluid piping are located.

For the case where a loss-of-coolant accident (LOCA) occurs concurrently with the loss of offsite power, a temperature increase in the auxiliary building operating level could also occur due to spent fuel pool (SFP) heatup, in the event that service water to the spent fuel pool heat exchangers were required to be isolated. The safety-related pumps and associated equipment are qualified for the resulting environments.

3.11.3.2.2 Loss of Ventilation

Normal convective cooling, supplemented by the ventilation system as described above, is adequate to maintain the postaccident temperature within normal ambient levels. In the event that all ventilation were lost, it has been determined that the pumps and associated valves would be capable of operating in the resultant environment for the time required to mitigate the accident without significant reduction in the available operating life of the equipment (see Section 9.4.2.4).

As part of SEP Topic III-5.B, an extensive review was performed of high- and moderate-energy pipe breaks. In the auxiliary building it was determined that steam heating line breaks would adversely affect the environmental qualification of safety-related electrical equipment. In response to this postulated pipe break scenario, RG&E provided pipe whip and jet impingement protection for a 6-in. steam line to protect certain cable trays. Also, RG&E made available spare electrical breakers and cable required for operation of a charging pump, as well as procedures and administrative controls. The calculated peak pressure and temperature conditions in the auxiliary building for the event are 150°F and 0.1 psig.

3.11.3.2.3 Radiation Levels

The radiation levels in the auxiliary building would increase in the event of a LOCA. Using conservative postaccident fission-product activity levels, the postaccident environment in the auxiliary building was calculated. This is discussed in detail in Section 12.4.3.3. The only major radiation field in terms of equipment qualification is in the vicinity of the recirculating fluid and in front of containment penetrations. *Reference 43* addresses radiation in front of containment penetrations.. *Reference 6*, as amended by the evaluation performed for the extended power uprate and discussed in *References 35, 38, 39 and 43*, addresses the required qualification doses for all the affected equipment.

3.11.3.2.4 Flooding

Flooding is not a concern in the auxiliary building. A review of potential equipment failures was conducted as part of the *Appendix R* fire protection review as well as SEP Topic III-6, Seismic Design Considerations. It was determined that actuation of the fire protection sprinklers or failure of all nonseismic tanks would not flood required safety-related equipment.

3.11.3.3 Intermediate Building

Implementation of an augmented inservice inspection program for high-energy piping outside containment has reduced the probability of pipe breaks in these systems to acceptably low levels (Section 3.6.2.1). A 6-in. main steam line branch connection break is the intermediate building design-basis event. An analysis of this event resulted in calculated steam conditions of 0.25 psig and 212°F (*References 32*, *33*, and *34*). A pipe crack or branch line that could not be isolated is the limiting design-basis event for the intermediate building environment. Mass and energy release in this case would be limited by the dryout of the steam generators with the duration of the environment dependent on the size of the leak or break. Based on flow through a main steam safety valve (a 6-in. line) of 247 lb/sec at a steam line pressure of 1100 psia and the inventory available for release from a main steam break (see Table 15.6-7), the mass and energy flow will continue for at least 11 minutes. Smaller leaks may continue substantially longer. It is expected that within 30 minutes to 1 hour, action could be taken to provide added ventilation to the building by opening doors. Within several hours, return to near ambient conditions could be accomplished. The exact duration is not critical in terms of affected equipment qualification; therefore, no explicit calculations have been performed. Chemical spray is not a design consideration in this building. The effects of submergence need not be considered, as discussed in *References 22*, *23*, and *24*. *Reference 8* presents the result of an analysis performed to ensure that safety-related equipment would not be flooded in the event of a feed line break in the intermediate building.

The turbine-driven auxiliary feedwater pump (TDAFW) area was analyzed to determine the resultant environmental conditions if all ventilation were lost. The purpose was to obtain data to assess the feasibility of performing manual operation of certain valves in the area. The analysis showed that the peak temperature would reach 145°F within the first hour and then stabilize (*Reference 31*).

The radiation environment was reviewed in response to the TMI Lessons Learned commitments. With the exception of areas in front of containment penetrations (*Reference 43*), the radiation environment is not significant in terms of equipment qualification.

As part of SEP Topic III-5.B, a review was made of high-energy line failures which could affect the steam and feedwater lines in the intermediate building. Potential cracks in the steam and feedwater piping were determined to be insignificant in terms of damaging required safe shutdown equipment. An evaluation was made of the postulated consequences of intermediate building block wall failure due to a high-energy line break in the turbine building. It was determined that failure of the safety and relief valves would not be limiting and that auxiliary feedwater flow would be maintained. However, RG&E did commit to evaluate, and modify as necessary, the structural integrity of steam and feedwater lines, main steam isolation valves, and auxiliary feedwater connections in conjunction with the Ginna Station Structural Upgrade Program (*Reference 25*) in order to provide protection from the failure of the adjacent wall. This information is provided in more detail in Section 3.6.2.

3.11.3.4 Cable Tunnel

Since the cable tunnel is effectively open to the intermediate building, the limiting environmental conditions for the cable tunnel are identical to the intermediate building conditions. However, physical separation is such that no concern exists with respect to direct effects such as jet impingement due to postulated high-energy line breaks.

3.11.3.5 Control Building

The limiting environmental conditions of the control building, which includes the control room, relay room, and battery rooms, is normal ambient conditions. Protection against high-energy line breaks and circulating water line breaks which could occur outside the control building and affect the control building environment are identified and discussed in *References 20 through 24* and *26 through 30*.

The air conditioning system for the control room is described in *Sections 6.4*, and consists of a single train of non-safety related NORMAL Control Room HVAC, plus two trains of Safety Related Control Room Emergency Air Treatment System (CREATS). Any of these 3 trains is capable of maintaining Control Room temperatures in a comfortable range for continuous long-term human occupancy, however, the value for post accident service conditions in the Control Room remains at 104°F so that future equipment specified for installation in the Control Room will be specified to withstand the higher localized temperatures that occur inside of cabinets and control cabinets.

The relay room is normally cooled by two non-safety-related air conditioning systems, which can be manually aligned to the emergency buses by closing the proper bus-tie breakers. Use of portable air conditioning units and fans are options available to maintain environmental conditions within the required specifications.

The battery rooms have a set of inlet and exhaust fans, as well as an air conditioning system. Additional fans powered directly from the batteries have also been installed.

As part of the SEP Topic III-5.B review, RG&E determined that steam heating coils in the control building would result in a harsh environment due to a postulated failure. These sources of steam have been removed from the control building.

3.11.3.6 Diesel Generator Rooms

The emergency diesel generator rooms each have their own heating, ventilation, and air conditioning systems, powered from the diesels. As soon as the diesels are brought up to speed, stabilized, and their respective circuit breakers closed to their emergency buses, the heating, ventilation, and air conditioning systems (ventilating fans) are energized.

Failure of a steam heating line would affect only one diesel. The other diesel, as well as offsite power, would still be available. This configuration has been reviewed by the NRC in *Reference 28* and found acceptable. Protection against events outside the rooms is described in *References 20*, *23*, *26*, *27*, and *30*. The limiting environment in the diesel generator rooms, therefore, is normal ambient conditions.

To provide protection from flooding in the diesel-generator rooms due to a circulating water line break, 18-in.-high steel curbs were installed in the diesel generator rooms. Subsequent installation of the "superwall" at the turbine building interface precludes the necessity for the curbs at that location.

3.11.3.7 Turbine Building

The turbine building does not require a heating, ventilation, and air conditioning system per se, but rather utilizes roof vent fans, wall vent fans, windows, and unit heaters for control of the turbine building environment.

In the event of loss of power to fans in this building, there would be no significant temperature rise since it is a large volume building with sufficient openings (windows and access doors) to adequately circulate the outside air.

Analyses have shown that the limiting pressure is caused by an instantaneous break in the 20-in. feed line in the turbine building (see Section 3.6.2.5.1). Peak pressures are 1.14 psig on the lower two levels of the building and 0.70 psig on the operating floor. Failure of portions of the exterior wall limits the duration of the pressure pulse to a few seconds. Pressure and temperature is limited by the failure capacity of the exterior walls. Assuming saturation conditions, the limiting temperature is approximately 220°F. A 100% humidity steam-air mixture is assumed. Isolation of the main steam and feed system will isolate the source of energy to the turbine building. For conservatism, it has been assumed that the peak pressure and temperature condition persists for 30 minutes with return to ambient being accomplished in a total of 3 hours. The exact duration of high environmental conditions is not critical in terms of affected equipment qualification; therefore, no explicit calculations have been performed.

The limiting flood condition resulting from a circulating water system pipe break is 18 in. of water level in the basement of the building (*Reference 20*).

3.11.3.8 Auxiliary Building Annex

This structure houses the standby auxiliary feedwater system. The limiting environment in this structure is normal ambient conditions. The cooling system for this building is redundant and seismically qualified. Flooding is not a concern since all safety-related equipment associated with the standby auxiliary feedwater system (SAFW) is elevated and there is no large volume of water stored in the building.

3.11.3.9 Screen House

The screen house, like the turbine building, does not require a heating, ventilation, and air conditioning system, but utilizes roof vent fans, wall vent fans, windows and unit heaters for control of the environment. In the event of a loss of power to the fans, there would be no significant temperature rise, since it is a large volume building with sufficient openings to adequately circulate outside air.

The limiting environment in the screen house is normal ambient conditions. A review was conducted as part of SEP Topic III-5.B to evaluate the effects of high- and moderate-energy line breaks in the screen house. It was determined that no protection was needed because alternative shutdown means are available, which do not rely upon service water from the screen house. Curbs were installed in the screen house in 1975 to protect critical equipment from the flooding source of a potential circulating water line break.

3.11.4 EQUIPMENT QUALIFICATION INFORMATION

Complete and auditable records which include supporting documentation for environmental qualification of safety-related electrical equipment are maintained by Ginna Station. The documentation which includes test results, specifications, reports, and other data has been identified by documentation reference citings in the Ginna Station reports to the NRC on the environmental qualification program.

3.11.5 ENVIRONMENTAL QUALIFICATION PROGRAM

The Nuclear Policy Manual defines the additional quality assurance program requirements for replacement and maintenance of environmentally qualified equipment to ensure compliance with the requirements of 10 CFR 50.49. The environmental qualification program is embedded in procedures for design, installation, and maintenance of systems and components. The Equipment Qualification Master List is arranged by system. The Nuclear Policy Manual is the controlling document for the environmental qualification program and assigns the Engineering Department the responsibility for establishing an evaluation process that documents the basis for any changes in the Equipment Qualification Master List.

REFERENCES FOR SECTION 3.11

1. Letter from D. L. Ziemann, NRC, to L. D. White, Jr., RG&E, Subject: Electrical Equipment Environmental Qualification, dated February 15, 1980.

2. NRC Interim Rule, 10 CFR Part 50.49, Environmental Qualification of Electric Equipment, June 25,1982.

3. Letter from L. D. White, Jr., RG&E, to A. Schwencer, NRC, Subject: Environmental Qualification of Electrical Equipment, dated February 24, 1978.

4. Letter from L. D. White, Jr., RG&E, to D. L. Ziemann, NRC, Subject: Environmental Qualification of Electrical Equipment, dated December 1, 1978.

5. Letter from L. D. White, Jr., RG&E, to D. L. Ziemann, NRC, Subject: Environmental Qualification of Electrical Equipment, Revision 2, dated April 25, 1980.

6. Letter from J. E. Maier, RG&E, to D. G. Eisenhut, NRC, Subject: Environmental Qualification of Electrical Equipment, dated October 31, 1980.

7. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: Equipment Qualification of Safety-Related Electrical Equipment, dated June 1, 1981.

8. Letter from J. E. Maier, RG&E, to D. M. Crutchfield, NRC, Subject: Environmental Qualification of Safety-Related Electrical Equipment, dated September 4, 1981.

9. Letter from J. E. Maier, RG&E, to D. M. Crutchfield, NRC, Subject: Environmental Qualification of Electrical Equipment, dated November 6, 1981.

10. Letter from J. E. Maier, RG&E, to D. M. Crutchfield, NRC, Subject: Schedule for Environmental Qualification of Electrical Equipment, dated February 18, 1982.

11. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: Safety-Related Evaluation Report for Environmental Qualification of Safety-Related Electrical Equipment, dated December 13, 1982.

12. Letter from J. E. Maier, RG&E, to D. M. Crutchfield, NRC, Subject: 10 CFR 50.49, Environmental Qualification of Electrical Equipment, dated May 19, 1983.

13. Letter from J. E. Maier, RG&E, to D. M. Crutchfield, NRC, Subject: Environmental Qualification of Electrical Equipment, dated February 1, 1983.

14. Letter from R. W. Kober, RG&E, to D. M. Crutchfield, NRC, Subject: Environmental Qualification of Electrical Equipment, dated March 30, 1984.

15. Letter from R. W. Kober, RG&E, to W. Paulson, NRC, Subject: Environmental Qualification of Electrical Equipment, dated August 30, 1984.

16. Letter from R. W. Kober, RG&E, to J. A. Zwolinski, NRC, Subject: Generic Letter 84-24, Environmental Qualification of Electrical Equipment, dated January 24, 1985.

17. Letter from J. A. Zwolinski, NRC, to R. W. Kober, RG&E, Subject: Environmental Qualification of Electrical Equipment Important to Safety, dated February 28, 1985.

18. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: SEP Topics VI-2.D and VI-3, dated November 3, 1981.

19. Letter from L. D. White, Jr., RG&E, to R. A. Purple, NRC, Subject: Valves Subject to Flooding, dated June 16, 1975.

20. Letter from R. A. Purple, NRC, to L. D. White, Jr., RG&E, Subject: Emergency Core Cooling System Valve Modification, dated July 3, 1975.

21. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: Main Steam Line Break with Continued Feedwater Addition, dated February 9, 1983.

22. Letter from L. D. White, Jr., RG&E, to D. L. Ziemann, NRC, Subject: High-Energy Line Breaks Outside Containment, dated June 27, 1979.

23. Letter from K. W. Amish, RG&E, to A. Giambusso, NRC, Subject: Transmittal of GAI Report No. 1815 on Effects of Postulated Pipe Breaks Outside the Containment Building, dated November 1, 1973.

24. Letter from K. W. Amish, RG&E, to E. G. Case, NRC, Subject: Pipe Breaks Outside Containment, dated November 1, 1974.

25. Letter from J. E. Maier, RG&E, to D. M. Crutchfield, NRC, Subject: SEP Topic III-5.B, Pipe Break Outside Containment, dated July 20, 1983.

26. Letter from K. W. Amish, RG&E, to A. Schwencer, NRC, Subject: Pressure Shielding Steel Diaphragm in Turbine Building, dated February 6, 1978.

27. Letter from R. A. Purple, NRC, to L. D. White, Jr., RG&E, Subject: Amendment No. 7 to Provisional Operating License DPR-18, and transmittal, dated May 14, 1975.

28. Letter from L. D. White, Jr., RG&E, to D. M. Crutchfield, NRC, Subject: SEP Topic III-5.B, Pipe Break Outside Containment, dated August 7, 1980.

29. Letter from D. M. Crutchfield, NRC, to L. D. White, Jr., RG&E, Subject: SEP Topic III-5.B, Pipe Break Outside Containment, dated June 24, 1980.

30. Letter from L. D. White, RG&E, to B. C. Rusche, NRC, Subject: Long-Term Cooling, dated May 13, 1975.

31. Devonrue, Engineering Evaluation of R. E. Ginna Nuclear Power Plant Ventilation System, dated July 1998.

32. Letter from JoEllen West, SAIC, to George Wrobel, subject: Analysis of steam line break in the Intermediate Building, dated October 17, 1986 (ref. UCNs 2/1014 and 2/620).

33. Letter from JoEllen West, SAIC, to George Wrobel, subject: Analysis of steam line break in the Intermediate Building, dated October 29, 1986 (ref. UCNs 2/1014 and 2/620).

34. Letter from JoEllen West, SAIC, to George Wrobel, subject: Analysis of steam line break in the Intermediate Building, dated December 12, 1986 (ref. UCNs 2/1014 and 2/620).

35. Letter from M.G. Korsnick, Ginna Station, to NRC, Subject: "R.E. Ginna Nulear Power Plant, Licensing Amendment Request Regarding Extended Power Uprate, dated July 7, 2005.

36. Letter from Westinghouse Electric Company, Nuclear Services, to D. Graves, Stone & Webster Engineering, RGE-05-52, Ginna Extended Poser Uprate Program, Transmittal of Final Containment Accident Heat Loads, Pressures, Temperatures, and Sump Water Temperatures to Stone & Webster and Ken Rubin Enterprises, dated June 24, 2005.

37. Stone & Webster, Calculation No. 109682-M-014, HVAC System EPU Evaluation, Revision 0, dated March 18, 2005.

38. Stone & Webster, Calculation No. 109682-UR-002, Impact of EPU on Normal Operation Radiation Levels, Shielding Adequacy and Normal Operation Radiation Environments in EQ Zones, Revision 0, dated January 19, 2005.

39. Stone & Webster, Calculation No. 109682-UR-006, Impact of EPU on Post-Accident Radiation Environments in EQ Zones, Revision 0, dated December 29, 2004.

40. Stone & Webster, Calculation No. 109682-UR-007, Post-LOCA Direct Shine Dose from the Containment Recirculation Fan Cooler (CRCF) Charcoal Filters, Revision 0, dated April 25, 2005.

41. Stone & Webster, Calculation No. 109682-UR-008, Post-LOCA Direct Shine Dose through Containment Wall in the Intermediate Building due to Airborne activity within Containment, Revision 0, dated April, 27, 2005.

42. Stone & Webster, Calculation No. 109682-UR-009, Post-LOCA Direct Shine Dose from the Containment Recirculation Fan Cooler (CRCF) HEPA Filters, Revision 1, dated July 7, 2014.

43. Constellation Energy, Constellation Generation Group, Fuel Operations Support Unit, Calculation CA06589, Post Accident Penetration Streaming Doses in the Ginna Intermediate and Auxiliary Buildings, Revision 0, dated June 3, 2005.

44. Stone & Webster, Calculation No. 109682-UR-003, Post-Accident Dose and Dose Rate Scaling Factors to Address the Impact of EPU on Environmental Service Zones (EQ) and Vital Area Access Mission Dose, Revision 1, dated July 7, 2014.

Table 3.11-1
ENVIRONMENTAL SERVICE CONDITIONS FOR EQUIPMENT DESIGNED TO MITIGATE DESIGN-BASIS EVENTS

INSIDE CONTAINMENT

Normal Operation (MODES 1 and 2)

Temperature	60°F to 125°F
Pressure	0 psig
Humidity	50% (nominal)
Radiation[a]	Less than 1 rad/hr. general. Can be higher or lower near specific components.

Accident Conditions (LOCA)

Temperature	Figure 6.1-1 (286°F maximum)
Pressure	Figure 6.1-2 (60 psig design)
Humidity	100%
Radiation[b]	Tables 3.11-2 and 3.11-3; 1.82×10^7 rads gamma; 2.99×10^8 rads beta
Chemical spray	Solution of boric acid (2750 to 3050 ppm boron) plus NaOH in water. Sump solution pH between 7.8 and 9.5, spray pH < 10.25.
Flooding	7-feet (approximately). Maximum submergence elevation is 242 ft. 8 in.

AUXILIARY BUILDING

Normal Operation (MODES 1 and 2)

Temperature	50°F to 104°F
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Less than 24 mrad/hr. general, with areas near residual heat removal piping less than 120 mrad/hr. during shutdown operation.

Accident Conditions (LOCA or steam line break in containment)	
Pressure	0 psig
Humidity	60% (nominal)
Operating floor - 271-ft. elevation	
Temperature	Peak of 131o F within 26 hours, due to terminating SFP Cooling immediately following a LOCA. The temperature cycles between 120o F and 130o F over a 24 hour period; due to solar gain effects, until SFP temperature is reduced by reestablishing SFP cooling.
Radiation near bus 14 and motor control center 1C and 1L.[c]	132 rad
Radiation[c] at other areas.	Less than 50 rad total
Intermediate floor - 253-ft. elevation	
Temperature	Peak of 102°F within 20 hours; stabilizes at less than 100°F after 24 hours.
Radiation[c] near bus 16 and motor control center 1D and 1M.	1190 rad
Radiation[c] at other areas.	Less than 500 rad total
Basement floor - 236-ft. elevation	
Temperature; basement level, West.	Peak of 104°F within 20 hours; stabilizes at less than 100°F after 24 hours.
Temperature; basement level, East near safety injection and containment spray pumps.	Peak of 111°F within 4 hours; stabilizes at less than 100°F after 24 hours.
Radiation[c]; basement level, near containment spray near residual heat removal, and safety injection pumps and piping.	3.7×10^6 rad total (at contact); 6.6×10^4 rad total 10 feet distance.

Radiation[c] at other areas.	Less than 10^4 rad total

Residual heat removal pump pit

Temperature	Temperature range of 162°F to 142°F from 10 hours to 24 hours after loss-of-coolant accident (LOCA). Peak of 166°F following an assumed 50 gpm residual heat removal (RHR) pump seal leak after 24 hours. Peak temperature lasts less than one hour. Room temperature decreases to 150°F, 40 hours after loss-of-coolant accident (LOCA).
Flooding	8.2 inches

Accident Conditions Based Upon High-Energy Line Breaks or Moderate-Energy Line Breaks:

Temperature (peak)	150°F
Pressure (peak)	0.1 psig
Humidity	≈100%
Radiation	Not applicable
Flooding	0 feet

INTERMEDIATE BUILDING

Normal Operation (MODES 1 and 2)

Temperature	50°F to 104°F
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Less than 6 mrad/hr. (higher near reactor coolant sampling lines).

Accident Conditions Based Upon High-Energy Line Breaks or Moderate-Energy Lines Breaks

Temperature	212°F for 30 minutes; then reducing to 104°F within 3 hours.
Pressure	0.25 psig for 30 minutes; then reducing to 0 psig within 3 hours
Humidity	≈100% indefinitely
Radiation	Not applicable
Flooding	0 feet

Accident Conditions Based Upon LOCA Conditions:

Temperature	115°F indefinitely[d] near large motors and feedwater and steam line piping. 104°F in open areas.
Pressure	0 psig
Humidity	≈100%
Radiation[d]	Negligible
Flooding	None of consequence. (*See Reference 8*)

CABLE TUNNEL

Same as INTERMEDIATE BUILDING

CONTROL BUILDING

Control Room

Normal operation (MODES 1 and 2)

Temperature	50°F to 104°F (usually 70°F to 78°F)
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Negligible

Accident Conditions

Temperature	Less than 104°F
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Negligible
Flooding	Not applicable

Relay Room & Relay Room Annex

Normal operation (MODES 1 and 2)

Temperature	50°F to 104°F
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Negligible

Accident Conditions

Temperature	Less than 104°F
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Negligible
Flooding	Not applicable

Battery Rooms

Normal operation (MODES 1 and 2)

Temperature	50°F to 104°F
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Negligible

Accident Conditions

Temperature	Less than 104°F
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Negligible
Flooding	Not applicable

Mechanical Equipment Room

Normal operation (MODES 1 and 2)

Temperature	50°F to 104°F
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Negligible

Accident Conditions:

Temperature	Less than 104°F
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Negligible
Flooding	3 feet (estimated for a service water line leak).

DIESEL GENERATOR ROOMS

Normal operation (MODES 1 and 2)

Temperature	60°F to 104°F
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Negligible

Accident Conditions (Maximum Design Temperature Day)

Temperature	Less than 125°F
Pressure	0 psig
Humidity	90% (estimated)
Radiation	Negligible
Spray	Not applicable
Flooding [e]	0 ft

One Ventilation Fan Operating (Maximum Design Temperature Day)

Temperature	Less than 140°F

TURBINE BUILDING

Normal operation (MODES 1 and 2)

Temperature	50°F to 104°F
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Negligible

Accident Conditions (High-Energy Line Break)

Temperature	220°F for 30 minutes, reduce to 100°F within 3 hours
Pressure	1.14 psig on mezzanine and basement levels, 0.7 psig on operating floor for 30 minutes, reduce to ambient 3 hours.
Humidity	100 %
Radiation	Negligible
Flooding	18 inches in basement (circulating water break)

AUXILIARY BUILDING ANNEX

Normal operation (MODES 1 and 2)

Temperature	60°F to 120°F
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Negligible

Accident Conditions

Temperature	60°F to 120°F
Pressure	0 psig
Humidity	60% (normal)
Radiation	Negligible
Flooding	Approximately 2 feet

SCREEN HOUSE

Normal operation (MODES 1 and 2)

Temperature	50°F to 104°F
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Negligible

Accident Conditions

Temperature	Less than 104°F
Pressure	0 psig
Humidity	60% (nominal)
Radiation	Negligible
Flooding	18 inches (circulation water break)

NOTE:—Temperature considerations for station blackout are contained in Section 8.1.4.5.2

a. Areas where the dose rates are expected to be higher are: (1) Reactor Cavity area. (2) Areas near components that contain reactor coolant, such as RCS loop cubicles and the regenerative heat exchanger area. The appropriate dose rates for these areas are 40 rad/hr. See *Reference 39*.

b. Dose estimates in areas adjacent to the containment recirculation fan cooler charcoal and HEPA filters will be higher than the containment general area doses. For such cases component location specific assessments are utilized as needed. See *References 40, 42 and 43.*

c. Dose estimates are determined for a LOCA with one (1) train of Engineered Safety Feature (ESF) cooling operating. Dose estimates in areas in front of containment penetrations will be higher than that estimated for the zone. For such cases, component location specific assessments are utilized as needed. See *Reference 43*.

d. Estimated (no explicit calculations performed).

e. Service water line crack would affect only one room.

Table 3.11-2
ESTIMATES FOR TOTAL AIRBORNE GAMMA DOSE CONTRIBUTORS IN CONTAINMENT TO A POINT IN THE CONTAINMENT CENTER - GINNA STATION

Time (hr.)	Airborne Iodine Dose (Rem)	Airborne Noble Gas Dose (Rem)	Plateout Iodine Dose (Rem)	Total Dose (Rem)
0.00	---	---	---	---
0.03	5.63E+04	8.86E+04	1.97E+03	1.47E+05
0.06	1.00E+05	1.66E+05	4.65E+03	2.71E+05
0.09	1.27E+05	2.37E+05	8.43E+03	3.72E+05
0.12	1.46E+05	3.00E+05	1.28E+04	4.59E+05
0.15	1.61E+05	3.60E+05	1.77E+04	5.39E+05
0.18	1.71E+05	4.16E+05	2.29E+04	6.10E+05
0.21	1.81E+05	4.69E+05	2.81E+04	6.78E+05
0.25	1.92E+05	5.37E+05	3.54E+04	7.63E+05
0.38	2.19E+05	7.41E+05	5.89E+04	1.02E+06
0.5	2.37E+05	9.11E+05	8.05E+04	1.23E+06
0.75	2.75E+05	1.23E+06	1.24E+05	1.63E+06
1	3.10E+05	1.51E+06	1.63E+05	1.98E+06
2	4.22E+05	2.45E+06	3.05E+05	3.18E+06
5	6.41E+05	4.30E+06	6.30E+05	5.57E+06
8	7.74E+05	5.30E+06	8.71E+05	6.95E+06
24	1.18E+06	7.58E+06	1.69E+06	1.05E+07
60	1.53E+06	8.60E+06	2.45E+06	1.26E+07
96	1.69E+06	9.03E+06	2.78E+06	1.35E+07
192	1.96E+06	9.85E+06	3.33E+06	1.52E+07
298	2.15E+06	1.04E+07	3.71E+06	1.62E+07
394	2.27E+06	1.05E+07	3.97E+06	1.67E+07
560	2.41E+06	1.08E+07	4.24E+06	1.73E+07
720	2.47E+06	1.09E+07	4.37E+06	1.76E+07
888	2.51E+06	1.09E+07	4.45E+06	1.79E+07
1060	2.53E+06	1.09E+07	4.49E+06	1.80E+07
1220	2.54E+06	1.09E+07	4.52E+06	1.80E+07
1390	2.55E+06	1.09E+07	4.53E+06	1.80E+07

Time (hr.)	Airborne Iodine Dose (R)	Airborne Noble Gas Dose (R)	Plateout Iodine Dose (R)	Total Dose (R)
1560	2.55E+06	1.09E+07	4.55E+06	1.81E+07
1730	2.55E+06	1.09E+07	4.55E+06	1.81E+07
1900	2.55E+06	1.10E+07	4.56E+06	1.81E+07
2060	2.55E+06	1.10E+07	4.56E+06	1.81E+07
2230	2.55E+06	1.10E+07	4.56E+06	1.81E+07
2950	2.55E+06	1.10E+07	4.56E+06	1.81E+07
3670	2.55E+06	1.10E+07	4.56E+06	1.81E+07
4390	2.55E+06	1.10E+07	4.56E+06	1.81E+07
5110	2.55E+06	1.10E+07	4.56E+06	1.81E+07
5830	2.55E+06	1.10E+07	4.56E+06	1.81E+07
6550	2.55E+06	1.10E+07	4.56E+06	1.81E+07
7270	2.55E+06	1.10E+07	4.56E+06	1.81E+07
8000	2.55E+06	1.10E+07	4.56E+06	1.81E+07
8710	2.55E+06	1.10E+07	4.56E+06	1.82E+07
			TOTAL	1.82E+07

Table 3.11-3
ESTIMATES FOR TOTAL AIRBORNE BETA DOSE CONTRIBUTORS IN CONTAINMENT TO A POINT IN THE CONTAINMENT CENTER - GINNA STATION

Time (hr)	Airborne Iodine Dose (rads)[a]	Airborne Noble Gas Dose (rads)[a]	Total Dose (rads)[a]
0.00	---	---	---
0.03	1.68E+05	6.34E+05	8.02E+05
0.06	2.99E+05	1.14E+06	1.44E+06
0.09	3.80E+05	1.56E+06	1.94E+06
0.12	4.37E+05	1.91E+06	2.35E+06
0.15	4.79E+05	2.22E+06	2.70E+06
0.18	5.11E+05	2.49E+06	3.00E+06
0.21	5.39E+05	2.73E+06	3.27E+06
0.25	5.69E+05	3.02E+06	3.59E+06
0.38	6.45E+05	3.85E+06	4.49E+06
0.5	7.00E+05	4.51E+06	5.21E+06
0.75	8.11E+05	5.72E+06	6.53E+06
1	9.10E+05	6.81E+06	7.72E+06
2	1.22E+06	1.06E+07	1.18E+07
5	1.80E+06	1.95E+07	2.13E+07
8	2.14E+06	2.61E+07	2.82E+07
24	>3.26E+06	4.85E+07	5.18E+07
60	4.42E+06	7.27E+07	7.71E+07
96	4.96E+06	8.81E+07<	9.30E+07
192	5.83E+06	1.17E+08	1.23E+08
298	6.40E+06	1.37E+08	1.43E+08
394	6.79E+06	1.46E+08	1.53E+08
560	7.19E+06	1.57E+06	1.65E+08
720	7.40E+06	1.64E+08	1.71E+08
888	7.51E+06	1.68E+08	1.75E+08
1060	7.58E+06	1.71E+08	1.78E+08
1220	7.63E+06	1.72E+08	1.80E+08

Time (hr)	Airborne Iodine Dose (rads)[a]	Airborne Noble Gas Dose (rads)[a]	Total Dose (rads)[a]
1390	7.65E+06	1.75E+08	1.83E+08
1560	7.66E+06	1.78E+08	1.86E+08
1730	7.66E+06	1.81E+08	1.89E+08
1900	7.66E+06	1.83E+08	1.90E+08
2060	7.66E+06	1.86E+08	1.93E+08
2230	7.67E+06	1.87E+08	1.95E+08
2950	7.67E+06	1.98E+08	2.06E+08
3670	7.67E+06	2.09E+08	2.17E+08<
4390	7.67E+06	2.20E+08	2.27E+08
5110	>7.67+06	2.31E+08	2.38E+08
5830	7.67E+06	2.40E+08	2.48E+08
6550	7.67E+06	2.51E+08	2.59E+08
7270	7.67+06	2.62E+08	2.70E+08
8000	7.67E+06	2.72E+08	2.80E+08
8710	7.67E+06	2.83E+08	2.91E+08

a. Dose conversion factor is based on absorption by tissue.

Table 3.11-4
ESTIMATES FOR TOTAL AIRBORNE GAMMA DOSE CONTRIBUTORS IN CONTAINMENT TO A POINT IN THE CONTAINMENT CENTER, REGULATORY GUIDE 1.89, REVISION 1

Time (hr)	Airborne Iodine Dose (R)	Airborne Noble Gas Dose (R)	Plateout Iodine Dose (R)	Total Dose (R)
0.00	---	---	---	---
0.03	4.82E+4	7.42E+4	1.69E+3	1.24E+5
0.06	8.57E+4	1.39E+5	3.98E+3	2.29E+5
0.09	1.09E+5	1.98E+5	7.22E+3	3.14E+5
0.12	1.25E+5	2.51E+5	1.10E+4	3.87E+5
0.15	1.38E+5	3.01E+5	1.52E+4	4.54E+5
0.18	1.47E+5	3.48E+5	1.96E+4	5.15E+5
0.21	1.55E+5	3.92E+5	2.41E+4	5.71E+5
0.25	1.64E+5	4.49E+5	3.03E+4	6.43E+5
0.38	1.87E+5	6.19E+5	5.05E+4	8.57E+5
0.50	2.03E+5	7.61E+5	6.90E+4	1.03E+6
0.75	2.36E+5	1.03E+6	1.06E+5	1.37E+6
1.00	2.66E+5	1.26E+6	1.40E+5	1.67E+6
2.00	3.62E+5	2.04E+6	2.61E+5	2.66E+6
5.00	5.50E+5	3.56E+6	5.40E+5	4.65E+6
8.00	6.63E+5	4.38E+6	7.47E+5	5.79E+6
24.0	1.01E+6	6.26E+6	1.45E+6	8.72E+6
60.0	1.31E+6	7.16E+6	2.10E+6	1.06E+7
96.0	1.45E+6	7.56E+6	2.39E+6	1.14E+7
192	1.68E+6	8.29E+6	2.86E+6	1.28E+7
298	1.85E+6	8.76E+6	3.19E+6	1.38E+7
394	1.95E+6	8.85E+6	3.41E+6	1.42E+7
560	2.07E+6	9.06E+6	3.64E+6	1.48E+7
720	2.13E+6	9.15E+6	3.76E+6	1.50E+7
888	2.16E+6	9.19E+6	3.83E+6	1.52E+7
1060	2.18E+6	9.21E+6	3.87E+6	1.53E+7
1220	2.19E+6	9.21E+6	3.89E+6	1.53E+7

Time (hr)	Airborne Iodine Dose (R)	Airborne Noble Gas Dose (R)	Plateout Iodine Dose (R)	Total Dose (R)
1390	2.20E+6	9.21E+6	3.90E+6	1.53E+7
1560	2.20E+6	9.22E+6	3.91E+6	1.53E+7
1730	2.20E+6	9.22E+6	3.91E+6	1.53E+7
1900	2.20E+6	9.22E+6	3.92E+6	1.53E+7
2060	2.20E+6	9.22E+6	3.92E+6	1.53E+7
2230	2.20E+6	9.22E+6	3.92E+6	1.53E+7
2950	2.20E+6	9.23E+6	3.92E+6	1.54E+7
3670	2.20E+6	9.24E+6	3.92E+6	1.54E+7
4390	2.20E+6	9.24E+6	3.92E+6	1.54E+7
5110	2.20E+6	9.25E+6	3.92E+6	1.54E+7
5830	2.20E+6	9.25E+6	3.92E+6	1.54E+7
6550	2.20E+6	9.26E+6	3.92E+6	1.54E+7
7270	2.20E+6	9.27E+6	3.92E+6	1.54E+7
8000	2.20E+6	9.27E+6	3.92E+6	1.54E+7
8710	2.20E+6	9.28E+6	3.92E+6	1.54E+7
			TOTAL	1.54E+7

Table 3.11-5
ESTIMATES FOR TOTAL AIRBORNE BETA DOSE CONTRIBUTORS IN CONTAINMENT TO A POINT IN THE CONTAINMENT CENTER, REGULATORY GUIDE 1.89, REVISION 1

Time (hr)	Airborne Iodine Dose (rads)[a]	Airborne Noble Gas Dose (rads)[a]	Total Dose (rads)[a]
0.00	---	---	---
0.03	1.47E+5	5.48E+5	6.95E+5
0.06	2.62E+5	9.86E+5	1.25E+6
0.09	3.33E+5	1.35E+5	1.68E+6
0.12	3.83E+5	1.65E+6	2.03E+6
0.15	4.20E+5	1.91E+6	2.33E+6
0.18	4.49E+5	2.14E+6	2.59E+6
0.21	4.73E+5	2.35E+6	2.82E+6
0.25	5.00E+5	2.60E+6	3.10E+6
0.38	5.67E+5	3.30E+6	3.87E+6
0.50	6.15E+5	3.86E+6	4.48E+6
0.75	7.13E+5	4.89E+6	5.60E+6
1.00	8.00E+5	5.81E+6	6.61E+6
2.00	1.07E+6	9.02E+6	1.01E+7
5.00	1.58E+6	1.65E+7	1.81E+7
8.00	1.88E+6	2.20E+7	2.39E+7
24.0	2.87E+6	4.08E+7	4.37E+7
60.0	3.89E+6	6.15E+7	6.54E+7
96.0	4.37E+6	7.48E+7	7.92E+7
192	5.14E+6	1.00E+8	1.05E+8
298	5.64E+6	1.17E+8	1.23E+8
394	5.99E+6	1.25E+8	1.31E+8
560	6.34E+6	1.34E+8	1.40E+8
720	6.53E+6	1.39E+8	1.46E+8
888	6.63E+6	1.42E+8	1.49E+8
1060	6.69E+6	1.44E+8	1.51E+8
1220	6.73E+6	1.45E+8	1.52E+8

Time (hr)	Airborne Iodine Dose (rads)[a]	Airborne Noble Gas Dose (rads)[a]	Total Dose (rads)[a]
1390	6.75E+6	1.47E+8	1.54E+8
1560	6.76E+6	1.49E+8	1.56E+8
1730	6.76E+6	1.51E+8	1.58E+8
1900	6.76E+6	1.52E+8	1.59E+8
2060	6.76E+6	1.54E+8	1.61E+8
2230	6.77E+6	1.55E+8	1.62E+8
2950	6.77E+6	1.62E+8	1.69E+8
3670	6.77E+6	1.69E+8	1.76E+8
4390	6.77E+6	1.76E+8	1.83E+8
5110	6.77E+6	1.83E+8	1.90E+8
5830	6.77E+6	1.89E+8	1.96E+8
6550	6.77E+6	1.96E+8	2.03E+8
7270	6.77E+6	2.03E+8	2.10E+8
8000	6.77E+6	2.09E+8	2.16E+8
8710	6.77E+6	2.16E+8	2.23E+8
		TOTAL	2.23E+8

a. Dose conversion factor is based on absorption by tissue.

Table 3.11-6
GINNA STATION/REGULATORY GUIDE 1.89, APPENDIX D, COMPARISON OF POSTACCIDENT RADIATION ENVIRONMENT ASSUMPTIONS

The in-containment post-LOCA radiation environments provided in Appendix D of Regulatory Guide 1.89, Rev. 1 is based on a core power level of 4100 MWt and a 12 month fuel cycle length. The core power level utilized to develop the radiation environment at Ginna is 1811 MWt (includes 2% margin for power measurement uncertainties). The fuel cycle length utilized for Ginna Station is 18 months.

Appendix D Paragraph	Regulatory Guide 1.89	Ginna Station
2.1.1	Release of 50% of the iodine and 100% of the noble gas inventory to the containment atmosphere.	Release of 50% of the iodine and 100% of the noble gas inventory to the containment atmo- sphere.
2.1.2	Containment free volume of 2.52 x 10^6 ft^3 74% or 1.86 x 10^6 ft^3 directly covered by containment spray.	Containment free volume of 1.00 x 10^6 ft^3. 78% (minimum) or 7.8 x $10^5 ft^3$ covered by containment spray.
2.1.3	Large release uniformly dis- tributed in a relatively open containment.	Large release uniformly distributed in a relatively open containment.
2.1.4	ESF fans with a flow rate of 220,000 cfm. Mixing between all major unsprayed regions and the main spray region.	Four fan coolers produce approximately 132,000 cfm. Thorough mixing is obtained.[a]
2.1.6	Containment spray from two equal capacity trains each rated for 3000 gpm boric acid solution.	Containment spray from two equal capacity trains each bounded by 1200 to 1800 gpm boric acid solution.[b]

a. The Regulatory Guide 1.89 fan cooler flow rate of 220,000 cfm results in a complete recirculation of 2.52 x 10^6 ft^3 of the containment atmosphere every 11.45 min. The Ginna Station fan coolers recircu- late the atmosphere once every 7.58 min.

b. The Regulatory Guide 1.89 spray system provides for a spray flow of 1 gpm for every 310 ft^3 of sprayed volume. The Ginna Station spray system provides a spray flow of 1 gpm for every 325 ft^3 of sprayed volume.