APPENDIX C

STRUCTURAL QUALIFICATION OF SUBSYSTEMS AND COMPONENTS

Introduction

Appendix C received a general update in FSAR Amendment 10 to describe currently applicable criteria and methods for structural qualification (and verification) of subsystems and components for the operating BFN Units. The updated Appendix C (for Amendment 10 and later Amendments) is maintained in accordance with 10CFR50.71.

Prior to Amendment 10, Appendix C described basic structural loading criteria and qualification methods used in the original design of BFN components and piping subsystems. Those earlier versions of Appendix C are significant for historical purposes only.

C.1 Scope

Appendix C presents the criteria and qualification methods for the following Seismic Class I (hereafter referred to as Class I) mechanical and electrical subsystems/components:

- Piping and Pipe Supports
- Major Components
- Primary Containment System and Penetrations
- Equipment
- HVAC Ductwork and Supports
- Conduit and Supports
- Cable Tray Systems

C.2 Loading Conditions, Definitions, and Overview

C.2.1 Seismic Classification

The design basis for Class I subsystems and components considers all applied loads such as pressure, temperature, deadweight, seismic, and hydrodynamic loads. Definitions of Class I, Class II, plant conditions, and seismic loading are as follows:

Class I

This class includes those structures, systems, and components whose failure or malfunction might cause, or increase the severity of, an accident which would endanger the public health and safety. This category includes those

structures, systems, and components required for safe shutdown and isolation of the reactor.

Class II

This class includes those structures, systems, and components which are important to reactor operation, but are not essential for preventing an accident that would endanger the public health and safety, and are not essential for the mitigation of the consequences of these accidents. A Class II designated item shall not degrade the integrity of any item designated Class I.

C.2.2 Loading Conditions

Normal Conditions

A normal condition is any condition in the course of operation of the plant under planned and anticipated conditions, in the absence of upset, emergency, or faulted conditions.

Upset Conditions

Upset conditions are any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand these conditions. The upset conditions include abnormal operational transients caused by a fault in a system component requiring its isolation from the system, transients due to loss of load or power, and any system upset not resulting in a forced outage. The upset conditions include the effect of the Operating Basis Earthquake.

Emergency Conditions

Emergency conditions are any deviations from normal conditions that require shutdown for correction of the conditions or repair of damage in the system. A fire event is an emergency condition, and its effects of weight and pressure on the components or systems are evaluated. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of specific damage developed in the system.

Faulted Conditions

Faulted conditions are those combinations of conditions associated with extremely low-probability postulated events whose consequences are such that the integrity and operability of the nuclear system may be impaired to the extent where considerations of public health and safety are involved. Such

considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

Test Conditions

A test condition is any condition, such as, when hydrostatic testing of component or system is conducted in the course of operation of the plant under planned and anticipated conditions and in the absence of upset, emergency, or faulted conditions.

C.2.3 Definitions

Operating Basis Earthquake

The Operating Basis Earthquake (OBE) is defined as that earthquake for which structures, systems, and components of the nuclear power plant that must continue operation without undue risk to the health and safety of the public are designed to remain functional.

Design Basis Earthquake [Also referred to as Safe Shutdown Earthquake (SSE)]

The Design Basis Earthquake (DBE) is defined as that earthquake for which the structures, systems, and components of the nuclear power plant that must be capable of safe shutdown and maintain the plant in a safe condition without undue risk to the health and safety of the public are designed to remain functional.

Amplified Response Spectra

Amplified Response Spectra (ARS) are defined as plots of maximum seismic response versus frequency for single degree-of-freedom subsystem at various locations in Class I structures subjected to seismic loading. The generation of amplified response spectra, building accelerations, and displacements used for subsystem and component analyses are described in Section 12.2 of the UFSAR.

Zero-Period Acceleration

Zero-period Acceleration (ZPA) is defined as the peak of the building floor seismic acceleration time history.

Primary Loads

Those loads which produce stresses which are not self-limiting, such as dead weight, pressure, seismic inertia loads, and hydrodynamic loads.

Secondary Loads

Those loads which produce stresses which are self-limiting, such as thermal effects and seismic anchor movements.

Torus Attached Piping Systems

Piping and tubing systems attached directly or indirectly to the torus shell to the point where effects of torus motion are demonstrated to be insignificant and including piping up to at least the first containment isolation valve. These systems include the main steam relief valve discharge piping systems.

C.2.4 Seismic Design Input

The input ground motions and seismic structural analysis methods used to develop design inputs for subsystem and component design and qualification are addressed in UFSAR Section 12.2 for each seismic Class I structure housing safety related subsystems and components. The seismic inputs used for subsystems and components addressed in Appendix C are summarized as follows:

(a) The original design basis El Centro earthquake input motion, described in Section 2.5.4, is used for qualification of

1) Major components (Section C.4),
2) The primary containment system and penetrations (Section C.5),
3) Torus attached piping systems (Section C.3.5).

Specifically, original design basis El Centro input is utilized in the following ways. Time history responses from the dynamic earthquake analyses described in Section 12.2.2.8 are used to seismically qualify the reactor pressure vessel (RPV), RPV internals, and drywell. Amplified response spectra are used to seismically qualify torus attached piping systems. Equivalent static coefficients are used to seismically qualify the torus and vent system as well as the primary system components addressed in Section C.4.2.

(b) The alternate design basis artificial earthquake input motion, described in Section 2.5.4, is used for qualification of piping and pipe supports (except torus attached piping systems) and HVAC ductwork and

supports. It will be used for verification of seismic adequacy of mechanical and electrical equipment and for long-term qualification of conduit and cable tray subsystems.

Specifically, the artificial earthquake amplified response spectra are used to qualify piping and pipe supports (except torus attached piping systems) and HVAC ductwork and supports. These response spectra will also be used in verification of seismic adequacy of equipment and long-term qualification of conduit and cable tray subsystems.

C.2.5 Overview of Structural Qualification

An overview of the structural qualification of BFN Class I subsystems and components follows.

1. Piping and Pipe Supports (Section C.3)

   Qualification of Class I piping and tubing systems, including the large bore piping, small bore piping/CRDH piping, buried piping, and tubing is in accordance with USAS-B31.1.0-1967 (Reference 1). Plant conditions and associated stress limits not addressed in USAS-B31.1.0-1967 are delineated in Section C.3.

   Qualification of torus-attached piping systems has been implemented within the scope of the Long Term Torus Integrity Program (LTTIP) as described in the Browns Ferry Plant Unique Analysis Report (PUAR) (Reference 12) and NRC Safety Evaluation (Reference 22). Torus attached piping system stresses, nozzle loads, concrete anchor loads, and other interface loads are maintained below the allowable described in the PUAR and Section C.3.5.

   Pipe support design criteria for all Class I piping and instrument tubing supports are based on the AISC Manual of Steel Construction (Reference 4), in conjunction with the Manufacturers Standardization Society (MSS) SP-58 (Reference 5). Plant conditions and associated stress limits not addressed in References 4 and 5 are delineated in Section C.3.

2. Major Components (Section C.4)

   Seismic design adequacy of reactor pressure vessel (RPV) and internals (RPV/Internals) have been reassessed (Reference 21). This assessment was based on the coupled seismic model described in Section 12.2.2.8.2. It qualified the

RPV/Internals for the updated loads. Critical stresses and interface loads for the RPV/Internals and primary system components are maintained below the allowables described in Tables C.4-1 and C.4-2, respectively.

3. Primary Containment System and Penetrations (Section C.5)

   The torus (wetwell), connecting vent system between drywell and wetwell, penetrations, and equipment have been qualified to Long Term Torus Integrity Program (LTTIP) criteria as documented in the BFN-Plant Unique Analysis Report (PUAR) (Reference 12). Stresses are maintained below allowables described in the PUAR. In addition, the drywell and its penetrations have been qualified for pressure, deadweight, and applicable loading components as described in Section C.5.1.

4. Equipment (Section C.6)

   Browns Ferry Nuclear Plant was identified as one of the operating plants to be reviewed for the NRC Unresolved Safety Issue (USI) A-46 requirements. The existing BFN safety-related equipment was originally designed to have sufficient margin of safety to withstand BFN design basis seismic loading. Plant-specific verification of seismic adequacy of equipment has been conducted in accordance with References 24, 25, 44, and 45. Qualification of new equipment and replacements for existing equipment from March 1988 until July 2007 is described in Section C.6.3. Equipment Seismic/Structural Qualification (ESQ) after July 2007 is described in Section C.6.4. Qualification of equipment subjected to hydrodynamic loads of the Long Term Torus Integrity Program (LTTIP) is described in Section C.6.5.

5. HVAC Ductwork and Supports (Section C.7)

   Seismic qualification of Class I HVAC ductworks and supports is based on AISC allowable stress design methods (Reference 4) and SMACNA standards (References 26, 27, 28) as described in Section C.7.

C.2.6 Safety Margins

This section describes and justifies the basic structural safety margins used in the original BFN design basis for Class I subsystems and components. In some cases BFN has committed to more specific

safety margins or allowables as described in other Appendix C sections. This section remains applicable for those cases where more specific commitments have not been made.

In addition to the generic definitions in Section C.2.2, the meaning of these terms is expanded in quantitative probabilistic language. The purpose of this expansion is to clarify the classification of any hypothesized accident or sequence of loading events so that the appropriate structural safety margins for reactor vessel and reactor vessel internals are applied. Knowledge of the event probability is necessary to establish meaningful and adequate safety factors for structural design. The table in the next paragraph illustrates the quantitative event classifications.

	P40 = 40-year event encounter probability
Upset (likely)	$1.0 > P_{40} > 10^{-1}$
Emergency (low probability)	$10^{-1} > P_{40} > 10^{-3}$
Faulted (extremely low probability)	$10^{-3} > P_{40} > 10^{-6}$

These probabilities have been assigned to establish the appropriate structural design safety margins for these loading criteria. A summary of these criteria is shown in the table listed below.

Deformation Limit	Table C.2-1
Primary Stress Limit	Table C.2-2
Buckling Stability Limit	Table C.2-3
Fatigue Limit	Table C.2-4

There are many places where, through the exercise of designer judgment, it is unnecessary to actually carry out a formal analysis for each of these limits. A simple example consists of the case where two pieces of pipe of differing wall thicknesses are joined at a butt weld. If they are both subjected to the same loading, only the thinner piece would require a formal analysis to demonstrate that the primary stress limit has been satisfied.

The term SF_{min} that appears in the tables is similar to the classical definition of a minimum safety factor on load or deflection. SF_{min} is related to the event probability by the following equation:

$$SF_{min} = \frac{9}{3 - \log_{10} P_{40}} \quad \text{(Equation A)}$$

where

10^{-1}	$> P_{40}$	$> 10^{-5}$	(Equation A applies)
10^{-5}	$> P_{40}$	$> 10^{-6}$	($SF_{min} = 1.125$)
1.0	$> P_{40}$	$> 10^{-1}$	($SF_{min} = 2.25$)

These expressions show the probabilistic significance of the classical safety factor concept as applied to reactor safety. The SF_{min} values corresponding to the current governing accident event probabilities are summarized as follows:

Item	Governing Loading Conditions	P_{40}	SF_{min}
Upset	N and AD	10^{-1}	2.25
	or N and U	10^{-1}	2.25
Emergency	N and R	10^{-3}	1.5
	or N and Am	10^{-3}	1.5
	or other	10^{-1} to 10^{-3}	2.25 to 1.5
Faulted	N and Am and R	1.5×10^{-6}	1.125
	or other	10^{-3} to 10^{-6}	1.5 to 1.125

where:

N = normal loads

U = upset loads excluding earthquake

AD = Operating Basis Earthquake including any associated transients

Am = Design Basis Earthquake including any associated transients

R = Any pipe rupture loading including any associated transients

The minimum safety factor decreases as the event probability diminishes; and if the event is too improbable (incredible = $P_{40} < 10^{-6}$), no safety factor is appropriate or required.

C.3 Piping and Pipe Supports

Piping design criteria for Class I piping and tubing, with the exception of torus attached piping, are in accordance with USAS-B31.1.0-1967 (Reference 1). Since this early code is incomplete relative to plant

operating conditions and code equations, the later ASME Section III code has been used in the development of load combinations and allowable stress criteria. Section III of the 1971 ASME Boiler and Pressure Vessel Code, including the Summer 1973 addenda (Reference 2), are used as guidance. However, analysis parameters, such as material allowable stresses, Stress Intensification Factor (SIF), coefficient of thermal expansion, elastic modulus, etc., are in accordance with the USAS-B31.1.0-1967 Code (Reference 1).

The design criteria for torus attached piping systems are described in Section C.3.5.

Pipe support design criteria for Class I piping and tubing supports are based on the AISC Manual of Steel Construction (Reference 4), in conjunction with the Manufacturers Standardization Society (MSS) SP-58 (Reference 5) as described in Section C.3.6.

C.3.1 Large Bore Piping

Large bore piping is defined as all Class I, 2½-inch Nominal Pipe Size (NPS) and larger, process piping which is not subject to Main Steam Relief Valve (MSRV) and postulated loss of coolant accident (LOCA) hydrodynamic loads due to torus excitation.

C.3.1.1 Analytical Models of Piping Systems

Analytical models of rigorously analyzed large bore Seismic Class I Piping Systems are consistent with as-built configurations in accordance with the requirements of References 13 and 14.

Large bore Class I piping systems are analyzed by rigorous analysis, which is a detailed analysis of the piping system, generally computer-aided, to assure that the system or design and support locations meet all code requirements. Computer programs used are described in Section C.3.7.

The continuous piping system is mathematically idealized as an assembly of elastic structural members connecting discrete nodal points. Nodal points are placed in such a manner that force deformation characteristics for piping elements such as straight runs of pipe, elbows, etc., can be formulated.

System loads such as internal pressure, weight, thermal expansion, fluid transients, and inertia forces are applied at the nodal points. The stiffness matrix of the interconnecting members is computed and

modified to account for flexibility characteristics specified in USAS-B31.1.0 (Reference 1).

Branch lines off the header piping are either modeled and analyzed as part of the header or decoupled from the header if its moment of inertia is less than $^1/_{25}$ of that of the run pipe. For decoupled branch line analysis, header responses are considered. The header pipe analysis includes the applicable Stress Intensification Factor (SIF) due to branch pipe.

C.3.1.2 <u>Seismic Analysis of Piping Systems</u>

The seismic analysis of large bore piping systems is performed by the response spectrum method. The damping is 0.5 percent of critical for both OBE and DBE. The seismic input for both horizontal and vertical directions is developed from structural analysis using artificial (Housner) time history described in Section 2.5.4.

The Amplified Response Spectra (ARS) developed from the artificial time history are broadened 10 percent for rock supported structures and 15 percent for soil supported structures. The seismic loading considers the dynamic inertia response of the system based on the ARS and the effects of differential Seismic Anchor Movement (SAM) of the structure to which the system is attached.

The rigorous analyses using ARS consider all modes of vibration below 20 Hz as flexural modes. Modes of frequencies 20 Hz and above are considered as rigid modes. Rigid mode responses are computed by using the maximum Zero Period Acceleration (ZPA) as applicable to the building floor (structure) to which the system is attached.

Response spectra analysis is based on either uniform (enveloped) or Independent Support Motion (ISM) techniques. In case of uniform motion, the spectra input is developed by enveloping all applicable spectra of the system attachment points for each direction. When ISM technique is used, zonal responses are combined by absolute summation.

The flexural mode responses are combined by the Square Root of Sum of Squares (SRSS) method except for closely spaced modes (frequencies within 10 percent of each other) which are combined by absolute summation. Rigid mode is combined by SRSS method with flexural modes. The spatial responses are combined by absolute summation of either East-West or North-South responses with Vertical responses. The net seismic responses are obtained by enveloping the East-West/Vertical and North-South/Vertical combinations.

SAM (including decoupled branch lines) effects are combined with either seismic inertia or thermal expansion loads. When combined with seismic inertia loads, absolute method with uniform technique and SRSS method with ISM technique is used. When combined with thermal expansion loads, the effects are combined by absolute summation.

C.3.1.3 Load Combinations and Acceptance Criteria

The load combinations and allowable stress limits for the Class I piping are presented in Tables C.3-1A, C.3-1B, and C.3-1C. These load combinations are categorized in terms of Test, Normal, Upset, Emergency and Faulted conditions as defined in Section C.2.2. Additional definitions are listed below.

Sustained Loads: Effects of live weight (weight of fluid being handled or of fluid used for testing or cleaning) and dead weight (weight of piping, insulation, and fluid or other loads permanently imposed on piping).

Fire Event Loads: Effects of fire events. Loads include sustained loads and internal pressure.

Fluid Transients: Effects of fluid transients including steam hammer, water hammer (excluding check valve slam), and main steam relief valve actuation.

Differential Settlement: Effects of loads on piping due to movement caused by adjacent soil and building structures or by relative settlement of buildings.

C.3.2 Small Bore Piping

Small bore piping is defined as all Class I, 2-inch Nominal Pipe Size (NPS) and smaller, process piping which is not subject to main steam relief valve actuation and LOCA hydrodynamic loads due to torus excitation.

The Control Rod Drive Hydraulic (CRDH) piping system, which consists of 185 1-inch NPS insert and 185 ¾-inch NPS withdrawal lines in generally bundled arrangements, is a unique group of small bore piping. Qualification of small bore piping other than the CRDH piping is described in Section C.3.2.1. Qualification of CRDH piping system is performed separately as described in Section C.3.2.2.

Small bore piping lines which have been modeled and rigorously analyzed as part of the large bore piping system are excluded from the scope of this

section. The details of the rigorous analysis for small bore and CRDH piping are consistent with the large bore piping analysis described in Section C.3.1.

C.3.2.1 General Small Bore Piping

Class I general small bore piping is qualified by field verification, evaluation, and analysis using a generic attribute methodology to meet the same criteria of load combination and allowable stress as described in Section C.3.1. Qualification of pipe supports for Class I small bore piping is described in Section C.3.6.2.

The Class I small bore piping qualification program performed rigorous analyses on a representative sample of problems in accordance with the methodology described in Section C.3.1. The sample size is approximately ten percent (10%) of the total program scope. The sample problems were selected from the as-built Class I small bore piping and pipe supports that are representative of the critical loading conditions and plant locations.

The 10 percent sample problem attributes were applied in evaluating the remaining 90 percent of small bore piping within the program scope. Small bore piping can also be qualified by rigorous analysis.

Design changes within the bounds of the LTTIP are qualified in accordance with Section C.3.5. Class I small bore piping design changes outside these bounds are qualified in accordance with large bore piping criteria (Section C.3.1) using rigorous analysis or equivalent static analysis techniques.

C.3.2.2 Control Rod Drive Hydraulic Piping

The Control Rod Drive Hydraulic (CRDH) system consists of 185 1-inch diameter insert and 185 ¾-inch diameter withdrawal lines routed in generally bundled arrangements with up to 100 pipes per bundle. It was not necessary to explicitly analyze each line; however, rigorous analysis has been performed for each of the typical configuration groups of lines which represent the range of line configurations within each bundle.

Lines are arranged in groups based on similar geometry, size, and span length to enable justification of typical configurations for analysis. These typical configuration lines have been analyzed to assure that both maximum primary and primary plus secondary loads have been evaluated for pipe stress levels and for each CRDH support frame.

CRDH insert and withdrawal lines are attached to the CRD housings. Analytical models and seismic input of CRDH lines are consistent with the TVA

commitment to complete installation of seismic lateral restraints on the CRD housings prior to restart (References 3, 35), as described in Section C.4.1.6.

Support reaction forces for all CRDH pipes have been compiled based on the typical line analyses. These reaction forces are then combined for qualification of the CRDH support frames as described in Section C.3.6.2.2.

CRDH insert and withdrawal pipe guides have been removed from a few locations on the support frames in order to accommodate thermal expansion of the pipes and the drywell vessel. Seismic pipe stresses and support loads are calculated by omitting these unidirectional supports in the piping dynamic models. However, seismic support loads and local pipe stresses due to impact of the pipes at the unidirectional support points are included when the piping models indicate that impact will occur.

These unidirectional loads are included in the reaction forces applied to the CRDH support frames as described in C.3.6.2.2.

Load combinations and allowable stress criteria for CRDH piping are described in Table C.3-1C. CRDH piping stresses are maintained below these allowables. Interface loads with CRD housing, drywell penetrations, and CRD hydraulic control unit components are also maintained within established limits.

C.3.3 Instrument Tubing

Class I tubing required for the safe shutdown is qualified by field verification, evaluation, and analysis using generic attributes to meet the same methodology and criteria as specified in Section C.3.1. Qualification of tubing supports is described in Section C.3.6.3.

The tubing qualification program performed rigorous analyses on a sample of representative tubing problems consisting of approximately 25 percent of the total program scope. The sample problems were selected from the as-built tubing and supports on a plant wide basis. The details of the rigorous analysis for tubing are consistent with the large bore piping analysis described in Section C.3.1. The 25 percent sample problem attributes were applied in evaluating the remaining 75 percent of the tubing within the program scope.

Design changes for tubing within the bounds of the LTTIP are qualified in accordance with Section C.3.5. Class I tubing design changes outside these bounds are qualified in accordance with large bore piping criteria (Section C.3.1) using rigorous analysis or equivalent static analysis techniques.

C.3.4 Buried Piping

Buried Class I piping has been qualified by rigorous analysis of representative models from all the systems containing buried piping. Similar configurations and embedment depths exist in each system. Rigorous analyses have been performed for worst case models from each system containing critical components such as elbows and tees along various depths and pipe diameters.

Analysis

Qualification of Class I buried piping is based on BFNP site specific geotechnical and seismic input data. Effects of surrounding soil on piping is simulated by horizontal and vertical soil springs. The spring rate and spacing requirements are according to References 9, 10 and 36.

Bounding analyses of all buried piping configurations are performed by using TPIPE computer program. Thermal, seismic, internal pressure, overburden pressure, and differential movement loading conditions are applied and qualified to satisfy the criteria depicted in Table C.3-1A.

Seismic analysis is done by statically applying the axial strain, which is calculated from the peak ground velocity and Raleigh wave velocity values as established in References 10 and 11. The bending strain is ignored as it is negligible.

Class I buried piping at penetrations into the secondary containment, entry points into the intake structure, and penetrations into other structures are analyzed for the differential movements of the soil and the structure. Typically, the analysis of piping within a structure includes a portion of the buried piping to a length sufficient enough to simulate the effects of an anchor. In some cases, the soil structure interface is protected from the effects of differential movements by using flexible couplings and/or guard boxes.

C.3.5 Torus Attached Piping Systems

Torus attached piping systems, as defined in Section C.2.3, are within the scope of the BFN Long Term Torus Integrity Program (LTTIP) design criteria. These systems are qualified to withstand the hydrodynamic loads associated with postulated loss of coolant accident (LOCA) loads and main steam relief valve discharges, seismic, static, and thermal loads defined in References 6 and 19. Structural qualifications, modifications, and design criteria for the torus attached piping systems, including the main steam relief valve discharge piping systems, are presented in sections 4, 7, 8, Appendix A, and

Appendix B of the LTTIP Plant Unique Analysis Report (PUAR),(Reference 12). The NRC safety evaluation report for the LTTIP is Reference 22.

Some refinements and clarifications were made to the torus attached piping systems design criteria and associated methods after the LTTIP SER (Reference 22) was issued in May 1985. These changes neither increase allowable stresses nor reduce structural margins relative to the acceptance criteria considered in LTTIP SER. Significant changes include:

1. Deflection limits were added for rigid pipe supports which are not attached to the torus and not included on the piping model. The deflection limit for supports existing on 7-31-87 is $^5/_{32}$-inch, when normalized loads are applied. The corresponding limit for supports added after 7-31-87 is $^1/_8$-inch. Normalized loads are obtained by dividing the piping analysis loads by the stress factors tabulated in LTTIP PUAR section 4.3.4.1.

2. Friction loads due to piping deadweight and thermal expansion effects were added to the load combinations for qualification of rigid supports which act as pipe guides. A friction factor of 0.3 is used and the friction load is considered in support qualification if the pipe thermal movement in the unrestrained direction is $^1/_{16}$-inch or more. Friction loads are considered in deadweight plus thermal expansion (primary plus secondary) loading combinations which exclude dynamic loads.

3. An additional loading condition for hydrostatic testing was added. Normal (service level A) stress limits are applied for this testing condition.

4. A fire event was added to the emergency condition (service level C) load combinations. This new load combination excludes any dynamic loading. It involves pressure plus deadweight effects only.

5. Clarification was added that pipe support gaps up to $^1/_{16}$-inch are considered in some thermal expansion/contraction analyses. However, pipe support flexibility is not considered in these analyses unless the supports are directly attached to the torus and included in the piping models for all loading conditions.

6. The active valve list was updated to add some new valves to the list. The new active valves are evaluated by the same criteria as the existing active valves listed in PUAR Table 2-3 (i.e., the criteria in PUAR section 4.3.3).

7. Interface requirements between torus attached piping and other piping (e.g., large bore piping) were clarified. When a torus attached piping model terminates in a lapping zone with other Class I piping, loads and stresses calculated from the separate analytical models are enveloped within the lapping zone and LTTIP criteria is satisfied for the enveloped loads and stresses.

BFN design criteria documents and engineering procedures control the qualification of torus attached piping systems for design changes. Compliance with these documents ensures that the allowable stresses, deflection limits, nozzle load limits, and other interface limits described in LTTIP PUAR section 4, appendix A, appendix B, and the clarification/refinements described above are satisfied.

Discrepancies between the initial as-designed and as-built conditions of LTTIP torus attached piping system modifications, described in the PUAR, were identified by reinspections and corrected. Those modifications now comply with engineering requirements assuring compatibility with the LTTIP design criteria.

The Emergency Core Cooling System (ECCS) suction strainers were initially modified to satisfy LTTIP structural design criteria requirements as described in PUAR Section 8.5.2 and Figure 8-4. Later, in response to NRC Bulletin 96-03 concerning potential plugging by debris (Reference 42), those strainers were replaced by larger, more functionally efficient strainers designed by GE.

The replacement strainers are securely fastened to the previously existing 30-inch diameter flanges by twenty-four 3/4-inch bolts. Each bolting flange face is located approximately one foot inside its associated ECCS suction penetration.

The replacement ECCS suction strainers and associated header/piping and penetrations were structurally qualified to LTTIP design criteria and analytical methodology described in the PUAR with the following refinements and clarifications:

1. The ECCS suction header/piping models were modified to simulate the added mass of the new strainers and associated water mass. Strainer stiffness was simulated based on structural properties determined from a detailed model of the strainers. The effective water mass was determined based on GE Research & Development (R&D) test data for the strainer.

2. Applied hydrodynamic drag loads for the LOCA and Main Steam Relief Valve (MSRV) LTTIP load cases were defined by extrapolation of applied drag loads for the previously existing strainers. Hydrodynamic drag load factors were based on comparison of the size, location, hydrodynamic mass, and drag coefficients for the replacement and previously existing strainers. The effective hydrodynamic masses and drag coefficients for the replacement strainers were based on GE R&D test data.

3. LOCA and MSRV drag load responses for the new strainers were determined by multiplying the applied hydrodynamic drag loads by Dynamic Load Factors (DLFs) in three orthogonal directions. LOCA Pool Swell DLFs were maintained at 2.0. LOCA Condensation Oscillation (CO)/Chugging and MSRV DLFs were calculated based on characteristic frequencies of the submerged strainers mounted to the ECCS suction header/piping models. Load combination techniques for the harmonic source and fluid structure interaction CO and Chugging drag loads on the strainer were per LTTIP criteria requirements (PUAR Sections 4.2.3 and 4.2.4).

4. Load reduction ("knockdown") factors for the single and multiple main steam relief valve torus dynamic response effects were justified based on correlation of the test results from the LTTIP inplant MSRV tests described in Appendix C of the PUAR with the ECCS suction header/piping analyses for those conditions. These load reduction factors conservatively account for increased MSRV flow rates due to 3% setpoint tolerance and anticipated 5% power uprate conditions, plus a four-inch increase in maximum pressure suppression pool level. The MSRV load reduction factors were applied to the torus dynamic response MSRV inputs to the ECCS suction header/piping and associated torus penetrations analyses. They were not applied to MSRV hydrodynamic drag loads.

5. Compliance of the ECCS header/piping systems and penetrations with LTTIP structural criteria (PUAR Section 4.3) was demonstrated for the updated models. No additional modifications were required to the header/piping systems or penetrations.

6. Structural integrity of the replacement strainers was demonstrated for enveloping loads. Strainer stresses comply with stresses from ASME Section III, 1989 Edition allowable stresses. Service levels for the various LTTIP load combinations were conservatively established based on the Mark I Containment Program Structural Acceptance Criteria Plant Unique Analysis Application Guide (PUAR Reference 13) Table 5.1 "Class MC Components and Supports".

BFN LTTIP design criteria documents have been changed to: 1) Require the definition of MSRV load reduction factors for the ECCS suction header/piping systems, penetrations and strainers; 2) establish the allowable stress criteria for the new strainers; and permit use of the new strainer drag coefficients and hydrodynamic masses based on test data for the strainers.

These changes are justified by supporting test/analysis correlation and by compliance with the intent of NUREG 0661 (Reference 6) and the applicable Mark I Containment Program documents. The other methodology changes described above are permitted by the LTTIP design criteria and PUAR Section 4.

These design criteria changes are limited to the replacement strainers, installed in response to concerns identified by NRC Bulletin 96-03 (Reference 42), and the associated ECCS suction header/piping and penetrations. PUAR Section 4.2.2.1 included a provision to allow use of the MSRV inplant test results to "address future NRC concerns regarding the BFN containment system." Definition and application of the MSRV load reduction factors, as described above, is consistent with that PUAR provision which was considered in the LTTIP SER (Reference 22). It is also consistent with NUREG-0661, Appendix A, Section 2.13.9, "MSRV Load Assessment By In-Plant Tests." Therefore, the design criteria changes neither increase allowable stresses nor reduce structural margins relative to the acceptance criteria considered in the LTTIP SER.

C.3.6 Pipe and Tubing Supports

Pipe/tubing supports, except LTTIP pipe supports, for Class I piping and tubing are designed based on the AISC manual of Steel Construction (Reference 4), in conjunction with the Manufacturer's Standardization Society SP-58 (Reference 5). Pipe/tubing supports are classified into the three general categories with respect to its load combination and stress allowables as following:

Linear Support - Any support which resists load essentially through a single component of direct stress. These supports provide resistance to movement of the pipe in a particular direction or directions from all load sources. A linear support is any variety of restraint configurations designed and fabricated from structural shapes and plates.

Dynamic Snubber - Provides resistance to dynamic movement without restricting gradually applied motion (e.g., piping thermal expansion) in the direction specified.

Component Standard Support - A support assembly consisting of one or more units which are catalog items and generally mass produced.

C.3.6.1 Large Bore Pipe Supports

Class I pipe supports installed on Class I large bore piping limit deflections from any one or all of the applicable load and movement sources as follows:

DW	-	Deadweight of sustained loads (includes applicable fluid weight in test condition or fire event)
E1	-	Operational basis earthquake (OBE)
E2	-	Safe shutdown earthquake (SSE)
Ti	-	Thermal mode i=1, 2, --- (includes directional anchor movements)
VT	-	Valve thrust (relief forces)
WH	-	Waterhammer
S1	-	OBE anchor movements
S2	-	DBE (SSE) anchor movements
PR	-	Pipe rupture

C.3.6.1.1 Large Bore Pipe Supports Design Criteria Analysis

The design load combinations and allowable stresses for the Class I large bore pipe supports are presented in Table C.3-2. These design load combinations are categorized with respect to hydrotest, normal, upset, emergency, and faulted conditions as defined in Section C.2.2. The basic computer programs used for large bore pipe support analysis are described in Section 3.7.

Concrete Anchors

Concrete Anchors for pipe supports are ductile type or expansion anchors. Ductile anchors at BFN are predominantly welded studs or regular length undercut anchors. To ensure that the ductile anchor capacity is controlled by the anchor steel capacity, the allowable load is limited to one-fourth the anchor concrete pullout capacity.

Expansion anchors transfer loads to the concrete by expanding laterally against the side of a hole drilled in hardened concrete. Expansion anchors at BFN are predominantly wedge bolts and expansion shell anchors. The tensile allowable loads for all support loading conditions for the wedge bolts and expansion shell anchors are limited to one-fourth and one-fifth, respectively, of the anchor concrete pullout capacity.

C.3.6.2 Small Bore Pipe Supports

C.3.6.2.1 General Small Bore Pipe Supports

The design criteria for all Class I small bore pipe supports, other than LTTIP and the CRDH small bore pipe supports, is the same as that for Class I large bore pipe supports as described in Section C.3.6.1.1. (See Section C.3.2.1).

C.3.6.2.2 CRDH Pipe Supports

For Control Rod Drive Hydraulic (CRDH) insert and withdrawal piping supports, in addition to the load combinations listed in Table C.3-2 the following load combinations are also evaluated.

Load Combination	Direction	Design Load Combination
Upset	+	DW + $T2^{(+)}$
	-	DW + $T2^{(-)}$
Emergency	+	DW + $T3^{(+)}$
	-	DW + $T3^{(-)}$

where: T2 is abnormal scram thermal mode, and
T3 is normal scram with post-LOCA thermal mode.

Normal full scram thermal loads are combined with seismic loads in the upset, emergency, and faulted load combinations of Table C.3-2.

Since the CRDH piping are routed in generally bundled arrangements as described in Section C.3.2.2, special types of CRDH pipe supports are defined as follows.

Rack - A two-dimensional frame which provides support/restraint for a CRDH pipe bundle.

Bar - An individual member in a support rack which provides direct support/restraint for a row of pipes within a CRDH pipe bundle.

The following methodology is used to determine and evaluate CRDH piping support stresses and stress-related load effects, thereby accounting for the fact that the peak seismic forces from multiple individual pipes will not occur simultaneously.

a. To calculate support stresses due to pipe seismic restraints on a pipe bundle support rack and on each individual support bar in that rack, the stresses caused by application of individual pipe peak seismic inertia forces in each separate orthogonal restraint direction are combined by a factored Absolute Sum (ABSUM) method. By this method, the support stresses due to individual pipe seismic inertia forces in an orthogonal restraint direction are multiplied by a factor and then combined by ABSUM. The multiplication factor is 0.5 for pipe seismic inertia (not impact) forces on racks which support 23 pipes or more; it is 0.75 for pipe seismic inertia forces on racks and individual bars which support from 9 to 22 pipes; and it is 1.0 for pipe seismic inertia forces on racks and individual bars which support from 1 to 8 pipes.

   Pipe seismic impact force effects and the net (combined) pipe seismic inertia force effects are combined by the SRSS method.

   The net (combined) pipe seismic inertial and impact force effects are combined with the support frame seismic self-weight excitation effects by the ABSUM method.

   This process is repeated for each orthogonal restraint direction.

b. Support frame stresses from seismic forces applied in each separate orthogonal restraint direction are combined by ABSUM. Similarly, seismic stresses from multiple racks in an overall support frame are combined by ABSUM. This is done either directly or by simultaneous application of the forces, for each rack and each orthogonal direction, to the overall frame in the directions which maximize overall frame seismic stresses and displacements. Each net (combined) seismic force effect is the larger of the North-South plus Vertical, or East-West plus Vertical seismic input effects. The seismic support stresses and displacements all have a + and - value for combination with static load effects.

c. Individual pipe forces are applied to the support frames for each static load case (deadweight, normal scram thermal, abnormal scram thermal, and post-LOCA thermal), and stresses are calculated for each case by algebraic summation.

d. Static and seismic frame stresses are combined and compared to the associated allowable stresses for each load combination defined by Table C.3-2 and this Section, considering the directional sense of each static stress and the + and - values of the seismic stress. Both general frame and local bar stresses are evaluated in this manner.

In addition, the combined deflection from seismic pipe forces, seismic self-weight excitation of the frames, and dead-weight of the pipes are limited to ⅛-inch at the point of pipe inertial load application for each direction of restraint.

e. Each pipe clamp and guide is evaluated for the combined effects of the peak seismic and static forces for each load combination. The forces in each orthogonal direction of restraint are applied simultaneously and compared to the rated capacity of the clamp/guide.

The design criteria for CRDH pipe supports, with the exceptions that above additional load combinations are added to those specified in Table C.3-2, as well as the special evaluation methodology and provisions described above, are the same as that for Class I large bore pipe supports as described in Section C.3.6.1.1.

C.3.6.3 Tubing Supports

The design criteria for Class I tubing supports is the same as that for Class I large bore pipe supports as described in Section C.3.6.1.1.

C.3.7 Computer Programs for Class I Piping System Analysis

The following is a list of the principal computer programs used for dynamic and/or static analysis of Class I piping and pipe supports. Each program's scope, background, applicability, and method of validation is discussed in the program descriptions below. As required, additional computer programs are used to support these analyses.

Program Name	Application
TPIPE	Static and Dynamic
ANSYS	Static, Dynamic and Non-linear
FAPPS	Static
GTSTRUDL	Static and Dynamic

TPIPE--for the linear elastic structural analysis of arbitrary, 3-dimensional piping systems subject to static and dynamic loadings. Analyses are performed to requirements for ASME Classes 1, 2, 3 systems, and ANSI B31.1.0 Power Piping Code.

Piping system is idealized as a mathematical model consisting of lumped weights connected by weightless elastic members. The locations of the lumped weights are chosen to adequately represent the dynamic characteristics of the system for dynamic considerations. The direct stiffness method of structural analysis is used to form the stiffness matrix, including stiffness modifications for curved components.

Diagonal mass and damping matrices are assumed. The equations of equilibrium are solved to determine the system displacements, and hence member forces and moments for the applied loading and/or displacements, using a Gaussian elimination procedure.

TPIPE analyzes piping systems subject to applied static loading conditions using the method discussed in the preceding paragraph; however, the piping dead load analysis considers both distributed weight properties of the piping and any concentrated weights.

TPIPE analyzes piping systems for dynamic excitation using the analysis technique known as the response spectrum modal superposition method. A direct integration or modal superposition time history capability is also available. Seismic options include a multiple support zone capability or independent support motion (ISM) technique for response spectrum analysis. The dynamic properties of the system (periods of vibration and normal mode shapes) are determined using a modified subspace iteration technique, and the system response is then computed by the modal superposition procedure. The seismic analysis capability includes the contribution of rigid modes (ZPA effect).

TPIPE has been benchmarked against the NRC program EPIPE in accordance with the Standard Review Plans, NUREG-0800, Section 3.9.1.II and NUREG/CR-1677. TPIPE is verified and maintained by using formal software QA procedures.

ANSYS - The ANSYS computer program is a large-scale general purpose computer program for the solution of several classes of engineering analysis problems. ANSYS is capable of analyzing structures with static and dynamic loadings, elastic and plastic member properties, creep and swelling, buckling, and small and large deflections.

The matrix displacement method of analysis based upon finite element idealization is employed throughout the program.

FAPPS - The Frame Analysis Program for Pipe Supports (FAPPS) is an inter-active computer program specifically developed for the

analysis and design of 18 standard frames as well as any non-standard frame for pipe support. It optimizes member sizes, welds, base plates, and embedments based upon various user specified design limitations.

The process of optimization of member sizes is controlled completely by the user to achieve an economical solution. In this way the user can limit the range of the member size selection process, subject to the available shapes, consideration of members connectivity and installation feasibility, and also the possible anticipation of future additional loads or revised design loads.

The FAPPS program also has the flexibility to perform normal load condition code checks for either the AISC, ASME Section III Subsection NF or AIJ codes.

The FAPPS also performs code check for upset, emergency, and faulted load conditions.

The FAPPS allows use of various types of load sets for simplification of input to allow algebraic, absolute, and/or SRSS combination of results due to each load vector within a load set as well as each load set that is to be combined in one load set.

FAPPS is verified and maintained by using formal software QA procedures.

GTSTRUDL provides the ability to specify characteristics of structural problems, perform analyses, reduce and combine results, perform design, and output any part or all of the information stored in the structural problem data based on a selective basis.

Analytical procedures apply to any combination of framed structures and continuum mechanics problems of arbitrary configuration and composition. Force boundary conditions on member ends, and force and displacement boundary conditions at support joints, may be specified implicitly by means of structural type and orientation commands, or explicitly for a member or joint. Continuum mechanics problems are treated using the finite element method in which the domain of the problem consists of different shapes connected at a finite number of joints.

GTSTRUDL permits elements (members and finite elements) of different types to be mixed in the same problem solution whether they have the same or a different number of degrees-of-freedom per joint.

Properties of member elements may be specified by providing section properties of prismatic or variable section members, naming a section from a pre-established table of properties (such as "W14X237"), or specifying flexibility of stiffness matrices for special member elements.

GTSTRUDL analysis procedures perform linear small displacement static and dynamic analysis of structures composed of any combination of member and finite elements with the same or variable number of degrees-of-freedom per joint.

GTSTRUDL design procedures include steel design and code checking for member elements by the 1969 and 1978 AISC (American Institute of Steel Construction) Specifications for general steel structures.

GTSTRUDL is verified and maintained by using formal software QA procedures.

C.4 Major Components

C.4.1 Reactor Pressure Vessel (RPV), RPV Internals, and Supports

General Electric Company (GE) originally designed and qualified the BFN RPVs, RPV internals, and supports as described in Sections C.4.1.1 through C.4.1.5. In 1989, an upgraded seismic analysis was performed and a reassessment of the combined loading effects was made as described in Section C.4.1.6. In 1997, an additional reassessment was made for a 105% power uprate operation which included the current seismic loads. Stresses and loads remain within design basis allowables, as indicated by Table C.4-1.

C.4.1.1 RPV Stress Analysis

The RPVs were designed, fabricated, inspected, and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III, its interpretations, and applicable requirements for Class A vessels as defined therein, as of the date that the reactor vessel order was placed (Reference Appendices J, K, and L for BFN 1, 2, and 3, respectively).

Stress analysis requirements and load combinations for the RPVs were evaluated for the cyclic conditions expected throughout the 40-year life, with the conclusion that ASME code limits are satisfied.

The 60 year operating life has been evaluated as a Time Limited Aging Analyses (TLAA). The summary of these evaluations are provided in Appendix O, Sections O.3.2.1 and O.2.4.

The RPV design report (original Appendix J that was redocketed by Reference 23 on June 23, 1989) provides the results of the original detailed design stress analyses performed for the RPV to meet the code requirements. Selected RPV components considered to have possibly higher than code design primary stresses, as a result of rare events or a combination of rare events, were also analyzed in accordance with the requirements of the loading criteria in this section and the safety margins in Section C.2.6.

Results (critical load combinations, locations, and allowables) for the most critical of those original analyses are included in Table C.4-1. The allowables were met in all cases.

C.4.1.2 RPV Fatigue Analysis

The analysis of the RPV shows that all components are adequate for cyclic operation by the rules of Section III of the ASME Boiler and Pressure Vessel Code. The critical components of the vessel are evaluated on a fatigue basis, calculating cumulative usage factors (ratios of required cycles to allowed cycles-to-failure) for all operating cycle conditions. The cumulative usage factors for the critical components of the RPV are below the code allowable of 1.0.

C.4.1.3 RPV Internals and Supports Stress Analysis

The RPV internals are designed using Section III of the ASME Boiler and Pressure Vessel Code as a guide. The material used for fabrication of most of the internals is solution heat-treated, unstabilized, Type 304 austenitic stainless steel conforming to ASTM specifications. Allowable stresses for the internals materials under normal operating conditions are taken directly from Section III.

For rare events or a combination of rare events, the RPV internals and supports were analyzed in accordance with the requirements of the loading criteria in this section and the safety margins in Section C.2.6, and results (critical load combinations, locations, and allowables) for the most critical of those analyses are included in Table C.4-1. The allowables were met in all cases.

C.4.1.4 RPV Internals Deformation Analysis

C.4.1.4.1 Control Rod System

If there were to be excessive deformation of the Control Rod System, made up of the control rod drive, control rod drive housing, control rod, control rod guide tube and fuel channels, and the core structural elements which support them (top guide, core support, and shroud and shroud support), it could possibly impede control rod insertion. The maximum loading condition that would tend to deform these long, slender components is the Design Basis Earthquake. The highest calculated stresses occur where the Design Basis Earthquake and loads resulting from the DBA pipe break are considered to occur simultaneously. Even in these cases the general stress levels are relatively low and for uprated conditions, control rod guide tube buckling potential is within allowable criteria. No significant deformation is associated with these calculated stresses; therefore, rod insertion would not be impeded after an assumed simultaneous Design Basis Earthquake and pipe break accident. The addition of control rod drive housing lateral restraints (reference Section C.4.1.6) provides added assurance in this regard.

C.4.1.4.2 Core Support

The core support sustains the pressure drop across the fuel. This pressure drop is the only load which causes significant deflection of the core support. Excessive core support deflection could lift the control rod guide tubes off their seats on the control rod drive housings and thereby increase core bypass leakage. This upward deflection would have to be ½-inch to begin to lift guide tubes. The maximum deflections under normal operating conditions and pipe rupture differential pressures for the core support are calculated to be very small as compared to ½-inch. The guide tubes will, therefore, not be lifted off. For uprated conditions, the core support beams have been evaluated for buckling as a result of the pressure drop and have been determined to have margin within allowable criteria.

C.4.1.5 RPV Internals Fatigue Analysis

The fatigue analysis was performed using as a guide the ASME Boiler and Pressure Vessel Code, Section III. The method of analysis used to determine the cumulative fatigue usage is described in APED-5460, 'Design and Performance of GE-BWR Jet Pumps,' September 1968. The most significant fatigue loading occurs in the jet-pump shroud support area of the internals.

The analysis was performed for a plant where the configuration (leg-type shroud support) was almost identical to Browns Ferry.

Therefore, the calculated fatigue usage is expected to be a reasonable approximation for BFN.

The following loading combinations and transients were considered.

1. Normal startup and shutdown;

2. Operating Basis and Design Basis Earthquakes;

3. Ten-minute blowdown from a stuck main steam relief valve;

4. HPCI operation;

5. LPCI operation (DBA); and

6. Improper start of a recirculation loop.

Calculated Cumulative Fatigue Usage was less than the allowable of 1.0.

Remarks

The location of maximum calculated fatigue usage is at the bottom side of the baffle plate at the point where the baffle plate attaches to the shroud in the vicinity of the minimum ligament.

C.4.1.6 RPV, RPV Internals, and Supports Seismic Analysis

GE originally performed a detailed seismic analysis of the RPV, RPV internals, and supports as described in redocketed Appendix J. Loads from other load sources were combined with the seismic loads from that analysis, with the results described in Sections C.4.1.1 through C.4.1.5.

The seismic loads on the RPV, RPV internals, and supports are now based on a dynamic analysis of an upgraded model of the RPV and RPV internals coupled with the Reactor Building as described in Section 12.2.2.8.2. The results of the upgraded seismic analyses were combined with the results of other loads for the various loading conditions in the reassessment documented by Reference 21. All stresses and loads remain within the allowables given in Table C.4-1.

The upgraded RPV, RPV internals, and supports seismic model includes consideration of the Control Rod Drive (CRD) housing lateral restraints which were added (References 3 and 35) to limit CRD housing and attached CRD hydraulic piping seismic stresses and displacements.

C.4.2 Primary System Components Stress Analysis

GE supplied, specified the allowables, and performed the original qualifications for the primary system components listed in Table C.4-2. Seismic loads used in these qualifications were based on El Centro input motion as described in Section C.2.4.

The extent of GE's stress analyses performed on equipment/components were dependent upon the type of equipment/components and the type of fabrication. Fabricated shapes are generally made from plate or rolled shapes with uniform thickness and shapes with regular geometric configurations. Cast shapes are generally made with nonuniform material thickness in complicated shapes that are not regular geometric configurations. Manufacturers have traditionally designed cast shapes conservatively since they do not lend themselves to rational analysis. Usually a design is developed based on extensive tests and experience. Selected components considered to have possibly higher than code design primary stress as a result of rare events or a combination of rare events were analyzed in accordance with the requirements of the loading criteria in this section and the safety margins in Section C.2.6. Results (critical load combinations, locations, and allowables) for the most critical of those analyses are included in Table C.4-2.

Class I large bore and torus attached piping systems have been qualified to satisfy requirements described in Section C.3. As part of this qualification the piping nozzle loads on primary system components are maintained within the allowable nozzle loads described in Table C.4-2.

In addition, primary system component stresses are within the allowable stresses and component thicknesses are greater than the required thicknesses described in Table C.4-2.

C.5 Primary Containment System and Penetrations

Each Browns Ferry unit employs a pressure suppression primary containment system which houses the reactor vessel, the reactor coolant recirculation loops, and other branch connections of the Reactor Primary System (RPS). The main functions of the primary containment system are:

1) to withstand the pressures resulting from a loss-of-coolant accident (LOCA) and/or main steam relief valve discharge,

2) to provide enclosure for the decay of any radioactive material which may ultimately be released,

3) to store sufficient water to condense steam released as a result of a LOCA and/or MSRV discharge, and

4) to supply water to the Emergency Core Cooling Systems (ECCS).

The primary containment system consists of a drywell, a pressure suppression chamber (wetwell or torus) which stores a large volume of water, a connecting vent system between the drywell and the wetwell, isolation valves, a vacuum relief system, containment cooling systems, equipment for establishing and maintaining a pressure differential between the drywell and the wetwell, and other service equipment. Section C.5.2 provides a more complete description of the primary containment system.

The following C.5 sections describe the structural qualification of the primary containment system except for the torus attached piping systems. Qualification of torus attached piping systems, including MSRV discharge piping systems, is described in Section C.3.5.

C.5.1 <u>Primary Containment Vessels Stress Analysis</u>

The primary containment vessels (consisting of the drywell, torus, vent system, and associated integral penetrations) were originally designed, fabricated, inspected, and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III, its interpretations, and applicable requirements for Class B vessels as defined therein as of the date that the vessel order was placed.

The Browns Ferry Containment vessels for Units 1 and 2 were built to Section III of the ASME Boiler and Pressure Vessel Code, 1965 edition, and Addenda through Winter 1966, inclusive (Reference 17). The Unit 3 containment vessel was built to Section III of the ASME code, 1965 edition, and Addenda through Summer 1967, inclusive. The ASME code design condition categories presently defined as Normal, Upset, and Emergency were not defined in these editions.

The loading conditions and allowable stresses for the drywell are presented in Table C.5-1. The accident condition allowables are based on 1968 ASME Section III Code with Addenda through Summer of 1969. Stresses in the drywell vessel including its integral penetrations are maintained within these allowables. These primary containment loadings were validated to be acceptable for 105% power uprate operation (Reference 43).

See Sections C.5.3 and C.5.4 for qualification of the torus vent system, non-safety related internal structures, and wetwell/drywell vacuum breakers.

C.5.2 Primary Containment Bellows Stress Analysis

The vent pipes bellows were designed, fabricated, and tested in accordance with ASME Boiler and Pressure Vessel Code, 1965 edition to Winter 1966 Addenda, and including Code Case Interpretations 1177-5 and 1330-1.

In accordance with Code Case 1177-5, the membrane stress in the bellows is limited to the tabulated maximum allowable stress value of 15,500 psi for the material (SA-240-T304) at 300°F in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, Table UHA-23.

Pressure tests were conducted on each bellows by sealing the annulus between the bellows and protective guard plate and pressurizing and monitoring the pressure decay for several hours. All bellows were found to be leak-tight.

The bellows-type expansion joints for containment penetrations and vent pipes are designed for an internal pressure of 56 psig at 281°F and an external pressure of 2 psig (7 psig in the steam vault room) at 281°F. The joints are also designed to permit an axial extension of 0.6 in., an axial compression of 2.0 in., and a lateral offset of from 0.3 to 1.55 in. depending on the elevation of the penetration. The design fatigue life of the joints is 7000 cycles.

The containment penetration expansion joints are designed, fabricated, and tested to meet Interpretations of the ASME Boiler and Pressure Vessel Code, Cases 1177 and 1330, and to meet the standards of the Expansion Joint Manufacturer's Association, Inc. The primary stresses are limited in accordance with Code Cases 1177 and 1330 to ASME Boiler and Pressure Vessel Code, Section VIII (Reference 18) allowable stress intensities. The secondary stresses are limited to the design fatigue life stress values for the expansion joint at 7000 cycles.

The longitudinal butt welds in the expansion joints are radiographed in accordance with the ASME Boiler and Pressure Vessel Code prior to forming. The welds attaching the bellows elements to the transition elements are made and inspected in accordance with Code Case 1330. The vent pipe bellows were evaluated further for the effects of postulated hydrodynamic loads in the LTTIP (Section C.5.3).

C.5.3 Long Term Torus Integrity Program (LTTIP)

In July 1980, the Nuclear Regulatory Commission (NRC) issued NUREG-0661, "Safety Evaluation Report, Mark I Containment Long-Term Program" (Reference 6) to address the NRC acceptance criteria for the Mark I nuclear plant containment system evaluation on the identified loss-of-coolant accident (LOCA) and main steam relief valve hydrodynamic loads.

The BFN Long Term Torus Integrity Program (LTTIP) reevaluated the plant-specific responses, in compliance with the intent of NUREG-0661. Results of the reevaluation, qualification, and implemented modifications are documented in the BFN-LTTIP Plant Unique Analysis Report (PUAR) (Reference 12) and the NRC safety evaluation report (Reference 22). This program addressed the torus (wetwell), vent system, torus attached piping systems, and non-safety related internal structures portions of the BFN primary containment system.

LTTIP structural qualification criteria, methods, and modifications for the torus and torus penetrations, vent system and vent pipe bellows, and non-safety related internal structures are described in Sections 4, 5, 6, 9, and Appendix B of the LTTIP PUAR. There have been no significant changes in the criteria or methods described therein since the LTTIP SER was issued in May 1985, except as follows:

a. Analyses have been performed to account for a maximum analytical pressure suppression pool water level elevation of 536'-10", which is 4 inches higher than the previously analyzed value. The new analytical water level includes consideration of potential instrument error and a margin for future use.

b. Applied pool swell and vent thrust loads were generated based on a more refined Design Basis Accident blowdown analysis performed using the LAMB vessel blowdown model described in NEDO-20566 (Reference 41). Structural analysis methods were not changed.

c. Stresses and loads remain below the acceptance limits considered in the LTTIP SER.

d. The LTTIP requirements are also demonstrated to be satisfied for 105% power uprate operation (Reference 43).

BFN design criteria documents and engineering procedures control the qualification of these primary containment system components for design changes. Compliance with these documents ensures that the allowable stresses and interface limits described in PUAR Section 4 and Appendix B are satisfied.

Discrepancies between the initial as-designed and as-built conditions of the LTTIP modifications for these components were identified by re-inspection and corrected. Those modifications now comply with engineering requirements assuring compatibility with the design criteria.

See Section C.3.5 for structural qualification of the torus attached piping system.

Changes in LTTIP design criteria and methodology for structural qualification of replacement ECCS suction strainers and associated header/piping systems and penetrations are described and justified in Section C.3.5.

C.5.4 Wetwell/Drywell Vacuum Breakers

Wetwell/drywell vacuum breaker dynamic loads associated with the LOCA chugging phenomena were identified during full scale tests for the Mark I Containment Program. Those loads were not included in NUREG-0661 and, consequently, were not addressed by the LTTIP PUAR. They were the basis of NRC Generic Letter 83-08 (Reference 37). The BFN wetwell/drywell vacuum breakers were evaluated and modified for these chugging dynamic loads in response to GL 83-08, as described by Reference 38. The LOCA hydrodynamic loads are also demonstrated to be satisfied for 105% power uprate operation (Reference 43).

C.6 Equipment

The general question regarding the adequacy of seismic qualification of safety-related equipment in operating plants has been recognized as an industry wide concern by the nuclear industry. The NRC established this concern in December 1980 as Unresolved Safety Issue (USI) A-46, "Seismic Qualification of Equipment in Operating Plants", (NUREG-0606 and -0705). It is important to note, however, that the A-46 statement of the issue recognized the industry consensus position that the application of the rules and procedures in existence at the time of operating plant design served to ensure that conservative margins were incorporated into safety equipment design.

In 1987, Browns Ferry Nuclear Plant was identified as one of the operating plants applicable to A-46 requirements. The existing Browns Ferry safety equipment was qualified originally to have sufficient margin of safety to withstand seismic loading. Plant-specific verification of seismic adequacy of equipment was implemented in accordance with References 24, 25, 43, and 44.

For historical purposes, the original equipment seismic loading and analysis methods are described in Sections C.6.1 and C.6.2. Qualification of replacements for existing equipment and new equipment from March 1988 to July 2007, while the A-46 Generic Implementation Procedure (GIP) was being developed by the Seismic Qualification Utilities Group (SQUG), approved by the NRC, and implemented at BFN, is described in Section C.6.3. Seismic/Structural Qualification (S/SQ) of new and replacement equipment after July 2007 is described in Section C.6.4.

C.6.1 Equipment Seismic Loading (Historical)

For GE-supplied equipment, seismic design conditions were included in the purchase specification of seismic Class I equipment. These were in the form of equivalent static seismic coefficients both in the horizontal and vertical directions. Vendor design submittals in this area (as well as all other functional areas) were reviewed for adequacy and applicability by the design engineer. Qualifications of GE-supplied mechanical equipment, which are primary system components, are described in Section C.4.2.

For TVA purchased Class I mechanical equipment, static seismic coefficients were specified for pumps, motors, etc., that were known to have a natural frequency greater than 20 Hz. The DBE building response in the vertical and horizontal directions at the equipment location was specified. Valves, relays, etc., were specified to withstand seismic loads equal to or greater than resonance response at the equipment location. For equipment whose natural frequency was felt to be less than 20 Hz, TVA required the vendor to determine the natural frequency. Appropriate spectral curves were included in the specifications for equipment suspected of having a low natural frequency (≤ 20 Hz), and the vendor was required to make a thorough dynamic analysis or test of this equipment.

The support structure of all Class I equipment was designed to adequately protect the equipment it supports.

For TVA-purchased Class I electrical equipment, the following requirements were included in the purchase specifications to assure adequate design and functional integrity under the seismic design conditions.

1. The equipment, the devices mounted on it, and its supports shall be designed to withstand seismic forces determined from floor accelerations of (A) g horizontal, and (B) g vertical, provided that all of the natural frequencies of vibration of the equipment are greater than 20 Hz.

2. Devices located in equipment or in areas of the equipment which have natural frequencies of vibration less than 20 Hz shall be designed to withstand seismic forces determined from resonant acceleration of (C) g horizontal and (B) g vertical.

3. The stresses in the supports and the anchor bolts due to seismic loads shall be combined with the stresses due to other live and dead loads and operating loads. The allowable stress for this combination of loads shall be based on the ordinary allowable stresses set forth in the applicable codes.

4. All equipment shall be anchored or fastened in such a way that it will remain in place when friction is considered nonexistent with the following accelerations:

   (A) specified from 0.20g to 1.70g, dependent upon location,
   (B) specified as 0.134g, and
   (C) specified from 1.24g to 29.80g, dependent upon location.

Provisions were included in the purchase specification of electrical equipment to ensure that the seismic requirements were satisfied. The specifications required the vendor to certify that his equipment meets the seismic requirements and to submit a verification report giving the following information.

1. Rigid equipment (has no natural frequencies of vibration less than 20 Hz): Show that the lowest natural frequency of more than 20 Hz and that all components on, or in, the equipment will continue to function properly when subjected to the seismic forces as determined from the floor accelerations specified.

2. Nonrigid equipment (has natural frequencies of vibration less than 20 Hz): Show that all components on, or in, the equipment will continue to function properly when subjected to the seismic forces as determined from the resonant accelerations specified.

C.6.2 Equipment Seismic Analysis (Historical)

Equipment was typically analyzed statically to determine its response to earthquake loads. The equivalent static coefficients for the equipment were obtained from the floor response spectra corresponding to the support elevations of the equipment. In lieu of determining the natural frequency of the equipment, the peak value of

the applicable floor response spectrum was used in calculating the earthquake-induced loads. In cases where the allowable stress limits were exceeded, the corresponding input acceleration was obtained from the appropriate floor response spectra.

Equipment which was outside the reactor coolant pressure boundary (RCPB) used the design conditions of pressure, thermal, and deadweight plus Operating Basis Earthquake (OBE), which was equivalent to Normal plus Upset. Substituting the Design Basis Earthquake for the OBE was equivalent to the Emergency condition. The Faulted condition was not applicable.

C.6.3 Seismic Qualification of New Equipment and Replacements for Existing Equipment From March 1988 Until July 2007

NRC Generic Letter 87-02 (Reference 8) specified that replacements should be "verified for seismic adequacy either by using A-46 criteria and methods or, as an option, qualifying by current licensing criteria." In the interim time period from March 1988 until July 2007, BFN optional equipment seismic qualification criteria for new items and replacements, which were plant modifications, were generally in accordance with NRC Regulatory Guide 1.100 (Reference 20) and IEEE 344-1975 (Reference 32). This optional, “current criteria” was used for a majority of BFN design changes for Class I equipment. This practice was continued until the A-46 Generic Implementation Procedure (GIP) methodology was included in a licensing basis revision per Section C.6.4.

Alternately, during this interim time period, qualification of Class I replacements and new items were addressed by the following approaches:

1. Qualification of a new or replacement item of equipment was accomplished by similarity to an existing installed item. Any dissimilarities were evaluated to show that the new or replacement item was no less capable of withstanding seismic loading than the installed item upon which the qualification by similarity was based.

2. Qualification was accomplished by comparison to an identical existing component. The two components were required to meet all of the following conditions.

    a. The form, fit, function, weight and weight distribution, and materials of construction were identical.

    b. The parts were from the same manufacturer and had the same model number.

c. It was positively shown that the mounting configuration and location of the existing item created a seismic loading condition which was equal to or greater than the seismic loading condition predicted for the new or replacement item.

3. Qualification was accomplished by use of the draft guidelines and criteria of USI A-46 and the associated Seismic Experience Data Base (See References 8, 33, 34) and then verified by A-46 GIP implementation.

In each of these alternative approaches the objective was to assure that the required seismic adequacy was maintained or achieved so that it could be verified later by NRC-approved A-46 criteria. Results of the BFN A-46 GIP implementation activities (References 24, 25, 44, and 45) confirmed that these approaches were successful.

C.6.4 Equipment Seismic/Structural Qualification (ESQ) After July 2007

After July 2007, S/SQ of BFN Class I and Class II electrical and mechanical equipment of all types, including electrical assemblies and devices, electrical conduit and cable tray raceway systems, and mechanical equipment and fluid system components (pump, tank and vessel assemblies, valves, and other in-line fluid system components) is performed in accordance with BFN ESQ design criteria, which replaced the interim design criteria described in Section C.6.3.

The ESQ design criteria requires new Class I equipment to be seismically qualified either by compliance with “current criteria” (based on IEEE 344-1975, NRC Regulatory Guide 1.100 R1, and applicable ASME codes) or SQUG GIP 3A methods (Reference 46). It also requires modifications to existing Class I equipment to comply with “current criteria” when there are specific BFN licensing commitments for the existing equipment to comply with IEEE 344-1975 (or 1971). In addition, it requires modifications to existing Class I equipment to comply with “current criteria” if the existing equipment S/SQ documentation is in accordance with “current criteria.” Otherwise, GIP 3A methods (Reference 46) are identified as an alternative approach for Seismic Qualification of Class I equipment when applied in accordance with SQUG Implementation Guidelines for Seismic Qualification of New and Replacement Equipment/Parts (NARE) (Reference 47).

Seismic Qualification of Class I equipment includes compliance with applicable criteria for normal operating loads plus seismic loads. However, S/SQ also entails qualification of Class I equipment in some locations for other BFN design basis structural loading conditions. For example, the ESQ design

criteria also requires Class I equipment within the LTTIP boundaries to be qualified for load combinations and acceptance criteria described in the LTTIP PUAR (Reference 12). In addition, Class I equipment, which is exposed to the outside environment, is required to be qualified for BFN design basis wind, tornado missiles, snow, and ice as applicable at the equipment location. The requirements for these additional loading conditions are considered "current criteria" because those conditions are not addressed by GIP 3A.

The ESQ design criteria requires new Class I electrical conduit and cable tray raceway systems and modifications to existing Class I electrical raceway systems to be seismically qualified in accordance with GIP 3A methods and the SQUG Implementation Guidelines for NARE (References 46 and 47).

Per the ESQ design criteria, new Class II equipment and modifications to existing Class II equipment are seismically qualified by demonstration of structural and pressure boundary integrity by "current criteria" or by GIP 3A methods.

Replacement items (e.g. parts) for Class I and Class II equipment for plant maintenance (not plant modifications) are verified to ensure that the S/SQ of the replacement item or its host equipment is not degraded. This is accomplished by application of a TVA design standard, in accordance with the ESQ design criteria. The design standard implements the Seismic Technical Evaluation of Replacement Items (STERI) process per References 48 and 49. It ensures that S/SQ of the equipment is maintained (not degraded) in accordance with "current criteria" and the SQUG Implementation Guidelines for NARE (Reference 47).

C.6.5 Qualification of Equipment in Torus Attached Piping Systems

Equipment in torus attached piping systems is also qualified to the requirements defined in section 4.3 of the LTTIP PUAR (Reference 12). Section C.3.5 of this Appendix describes the qualification of torus attached piping systems.

C.6.6 Interface Loads from Class I Piping Analysis

Interface loads between Class I equipment and piping which is rigorously analyzed, as described in Section C.3, are maintained within acceptable limits justified by TVA, TVA-contractors, or equipment vendors. For example, nozzle loads on primary system components are maintained within allowables described in Section C.4.2 and Table C.4-2.

C.7 Heating, Ventilation, and Air Conditioning (HVAC) Ductwork and Supports

C.7.1 Scope

Seismic qualification of the Class I HVAC ductwork and associated supports is described in this section. The BFN Class I HVAC ductwork consists of rectangular and round sheet metal ducts (References 26, 27, 28), as well as scheduled pipe used as ductwork. For rectangular ducts, the Companion Angle (CA) and Pocket Lock (PL) type transverse joint constructions as specified by the Sheet Metal and Air Conditioning Contractors' National Association (SMACNA) are used. Additionally, welded joint constructions are also used.

The analytical methods are described in Section C.7.2 and C.7.3. When scheduled pipe is used as ductwork, qualification may be done according to methods and stress limits described in Section C.3.1 and C.3.6. The buried HVAC ductwork for Standby Gas Treatment (SGT), constructed of scheduled pipe, is evaluated in accordance with Section C.3.4.

C.7.2 Ductwork System Seismic Analysis

The ductwork system consists of a duct run and a series of supports. The original ductwork design was based on Amplified Response Spectra (ARS) using the El Centro earthquake input ground motion as described in Section 2.5.4. Subsequently, all Class I ductwork systems have been reevaluated using ARS developed from artificial time history (Housner) and impact assessments have been made. The impact assessments include combining the two directional responses absolutely, as opposed to the SRSS method used with the El Centro response spectra. The results of the impact assessments have been used to determine whether the ductwork previously qualified using the El Centro response spectra (SRSS combination) meets the allowables specified in Section C.7.3. For any ductwork system not meeting the specified allowables, modifications have been made for compliance with the requirements of Section C.7.3.

For the seismic evaluation, computer aided modal response method is used. The ducts are modeled as beam elements with effective bending and shear properties. For rectangular ducts with Companion Angle and Pocket Lock type transverse joints, frequency correction factors of 0.87 and 0.59, respectively, have been applied to more accurately predict the frequency. However, for round ducts (welded joints), the frequency correction factor used is 1.0. Modes of frequencies 20 Hz and above are considered as rigid. The effects of Zero Period Acceleration (ZPA) are also considered, which is applied to the rigid mode with the effective values as the maximum building floor accelerations at or above 20 Hz.

The flexural mode responses are combined by SRSS method for each direction, except for closely spaced modes (frequencies within 10 percent of each other), which are combined by absolute summation. The rigid mode is combined by SRSS method with flexural modes. Two sets (xy and zy) of resultant seismic responses are generated, where x and z represent two horizontal directions and y represents the vertical direction. A set is formed by absolute summation of responses of the two directions. The controlling response is the larger of the two sets of responses.

Alternatively, an equivalent static method with the peak acceleration values corresponding to the system fundamental frequency has been used. A multimode correction factor of 1.5 is applied to the peak acceleration in an equivalent static analysis method.

The ARS analysis considers all effective concentrated weights lumped along the ductwork. Differential building seismic movements are also considered. In addition, weather induced loads (as applicable) are considered for ductwork exposed to exterior conditions.

The damping ratio of DBE (SSE) response spectra used on each type of the ductwork are as follow:

Ductwork	Critical Damping %
Rectangular, companion angle or pocket lock	7
Rectangular, all welded	2
Round duct, all types except scheduled pipe	2
Scheduled pipe	1

C.7.3 Ductwork Load Combinations and Allowable Stresses

Ductwork systems are designed for Normal and Emergency loading conditions. Normal condition consists of Deadweight loads. Emergency condition is a combination of Deadweight and DBE (SSE) seismic loads.

C.7.3.1 Duct Allowable Stresses

Bending

Bending stresses are limited to the following allowables:

Rectangular Duct:	8000	psi (Normal)
	12000	psi (Emergency)

*Round Duct: 10000 psi (Normal)
15000 psi (Emergency)

* Not applicable to scheduled pipe analyzed in accordance with Sections C.3.1 and C.3.4.

Shear

1) Rectangular Duct:

$Va = 5.5\ w\ (6.4)\ (Ie/w)^{.25}$ (Emergency)

Where,

Va = Allowable shear capacity of a rectangular duct cross section, 1b.

w = Uniform weight per foot length of ductwork, lb/ft.

Ie = Effective moment of inertia (bending) of cross section, $in.^4$ (based on approach in Reference 26).

Notes

a)For evaluating the shear load at a section with an unreinforced opening, Va is reduced in proportion to the reduction in gross shear area.

b)For ductwork with heavier gauge steel than the SMACNA construction, an alternative method of qualification by AISI Specification (Reference 39) is adopted. If this method is used, a ⅓ increase is allowed for the Emergency load combination.

2) Round Duct

Maximum shear stress: $0.53\ F_y$ (Emergency)

where, F_y = Minimum specified yield stress of duct section, psi.

Buckling

Maximum allowable = 0.9 x critical buckling for axial compression (Emergency)

C.7.3.2 Ductwork Supports Allowable Stresses

The allowables for Class I ductwork supports, connecting bolts, and welds are as follows:

Normal:	AISC Manual allowables (Reference 4)		
Emergency:	Tension and Bending	:	1.5 x Normal *
	Compression	:	1.5 x Normal *
	Shear	:	0.9 F_y / 1.7321
	Bolt stress in tension	:	1.5 x Normal
	Bolt stress in shear	:	1.4 x Normal
	Weld stress	:	1.5 x Normal *
	Buckling	:	0.9 x Critical

*Maximum allowable: 0.9 F_y.

C.8 Control of Heavy Loads

C.8.1 Introduction/Licensing Background

Generic Letter (GL) 81-07, “Control of Heavy Loads,” was initiated for all Licensees of Operating units to review the controls of handling heavy loads in regard to the requirements of NUREG-0612. GL 81-07 was in response to the occurrence of load drops in the industry.

The implementation of Phase I of NUREG-0612 included the development and evaluation of critical lift zones; improved designation and inspection of rigging equipment; improved crane operator and rigging training; the development of station procedures to control heavy lifts; review of reactor building or containment crane systems for single failure proof capabilities; and for stations without single failure proof capabilities, the performance of load drop analyses.

Twelve sites (twenty reactors in all) including both PWRs and BWRs were selected as a pilot program to review the effectiveness of the implementation of Phase I of NUREG-0612. BFN Units 1, 2, and 3 were included in this pilot assessment by the NRC. Due to the improvements made with the incorporation of NUREG-0612 at the pilot plants, the conclusion by the NRC was to not proceed with the implementation of Phase II of NUREG-0612. Generic Letter 85-11 cancelled Phase II of NUREG-0612.

The Control of Heavy Loads program at BFN was established through a number of letters submitted to the NRC. The submittals and letters include:

- H. G. Parris (TVA) letter to A. Schwencer (NRC), Comparison of the Browns Ferry Nuclear plant reactor building crane design, testing and maintenance requirements with the positions given in the document entitled “Branch Technical Position APCSB 9-1, Overhead Handling Systems for Nuclear Power Plants,” dated June 30, 1976

- L.M. Mills (TVA) letter to T. A. Ippolito (NRC), Requested information on the Browns Ferry Reactor Building crane, dated February 10, 1981

- L.M. Mills (TVA) letter to D. B. Vassallo (NRC), Requested information regarding NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants”, dated April 12, 1982

- L.M. Mills (TVA) letter to D. B. Vassallo (NRC), Requested information regarding NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants” - Enclosed is information regarding section 2.1 of NUREG-0612, dated June 3, 1982

- L.M. Mills (TVA) letter to D. B. Vassallo (NRC), Requested information regarding NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants” - Enclosed is information regarding section 2.2 and 2.3 of NUREG-0612, dated September 28, 1982

- L.M. Mills (TVA) letter to D. B. Vassallo (NRC), NRC / TVA telecon review of the technical evaluation report for the Browns Ferry entitled “Control of Heavy Loads - NUREG-0554,” dated December 14, 1982

In response to these submittals, the NRC issued a Safety Evaluation Report (SER) on the Control of Heavy Loads (Phase I) for BFN. This SER was transmitted to TVA by a letter from D. B. Vassallo (NRC) to H. G. Parris “Control of Heavy Loads - (Phase 1)” {review and acceptance of BFN’s response to Control of Heavy Loads (Phase 1), 1980 Generic Letter “Control of Heavy Loads”}, dated June 6, 1984.

C.8.2 Safety Basis

The safety basis for the Control of Heavy Loads is provided by assuring the risks associated with load-handling failures is acceptably low. This assurance is provided by meeting the requirements of NUREG-0612, Section 5.1.1, the use of an equivalent single-failure-proof crane for the reactor head lift, drywell head lift, lifts associated with reactor disassembly and inspections; and for dry cask lifts on the refuel floor. Critical lifts performed in plant areas other than the refuel floor are conducted in accordance with program documents and procedures.

C.8.3 Scope of Heavy Load Handling Systems

A heavy load for BFN is defined as any load weighing in excess of 1,000 lbs that is lifted in an area designated as a critical lift zone. Critical Lift Zones (CLZ) are defined as zones in designated strategic regions of the plant where a load drop impact could potentially release radioactive material into the environment or prevent equipment from functioning that may be required to achieve safe shutdown and continued decay heat removal. Overhead handling systems, including mobile, lifting lugs and monorail devices, that meet these criteria are:

- Reactor Building overhead crane
- Mobile crane lifts over the Units 1 & 2 and Unit 3 Standby Diesel Generator Buildings
- Mobile crane lifts over Intake Pumping Station
- Lifting lugs or temporary rigging over the Core Spray Pumps
- Monorail and hoist assembly over the Recirculation Pumps

In addition, other overhead handling systems were reviewed and excluded from this list on the basis that a load drop would not result in damage to any system required for plant shutdown or decay heat removal for one of the following reasons:

1. There is sufficient physical separation of the overhead handling system from any system or component required for safe shutdown or decay heat removal.

2. The system or component over which the load is carried is out of service while the load handling system is used.

3. The load weighs less than 1,000 lbs and is not considered to be a heavy load.

C.8.4 Control of Heavy Loads Program

The Control of Heavy Loads Program consists of the following:

1. BFN commitments in response to NUREG-0612, Section 5.1.1 elements

2. For reactor disassembly/reassembly and inspections and refuel associated lifts, a single-failure-proof crane

3. For spent fuel cask lifts over the spent fuel pool, a single-failure-proof crane

4. Performance of critical lifts in accordance with program documents and procedures

5. Performance of preventive maintenance, inspection, and testing of cranes, special lifting devices, and rigging equipment in accordance with program documents and procedures

6. Use of task qualified crane operators and rigging personnel

C.8.4.1 BFN Commitments in Response to NUREG-0612, Section 5.1.1

The control of heavy loads is performed by compliance with the seven guidelines outlined in NUREG-0612, Section 5.1.1.

These guidelines are met through the following:

Guideline	Compliance Method
1	<u>Safe load paths</u> - Safe load paths are contained in maintenance instructions. Directions contained within these instructions provide requirements for control of any lift greater than 1,000 pounds, lifts on the refuel floor of the Reactor Building in the regions adjacent to and over the spent fuel pool and over the reactor cavity, lifts in the Reactor Building inside the drywell over the Recirculation system pumps, lifts in the Reactor Building over the Core Spray system pumps, lifts over the Units 1 & 2 and Unit 3 Standby Diesel Generator Buildings, and lifts at the Intake Pumping Station over the Residual Heat Removal Service Water (RHRSW) pumps in those areas designated as critical lift zones (CLZ). The critical lifting zones are defined as follows:

- Reactor building refuel floor CLZ - The region defined as the Spent Fuel Pool CLZ, within 15 feet of the spent fuel pool. Also, the region defined as the Reactor Well CLZ, within 15 feet of the reactor well when at least one of the lower horizontal reactor well shield blocks has been removed and when spent fuel is in the reactor vessel. Also, when Alternate Decay Heat Removal (ADHR) is being used as the primary heat removal source for a particular unit's fuel pool, the CLZ shall extend over the affected ADHR piping service when it is in service.

- Standby Diesel Generator Building CLZ - The region, over the roof of the Unit 3 Standby Diesel Generator Building, when any one of the four diesel generators is inoperable; and the Unit 1 & 2 Standby Diesel Generator Building when Diesel Generator Auxiliary Board "A" is inoperable.

- Intake Pumping Station CLZ - The region over the remaining operable RHRSW pumps when limiting conditions per Technical Specification exist and/or the region over the remaining operable High Pressure Fire pumps.
- Core Spray CLZ - The region, when irradiated fuel is in the

reactor vessel, with the potential of associated Core Spray pumps and piping being impacted by equipment transported through the hatches on El. 565.0’.

- Recirculation Pump Motor CLZ - The region, with irradiated fuel in the reactor vessel, when removing or replacing a recirculation pump motor. The Recirculation pump piping CLZ is not considered an impact target area for NUREG-0612 Heavy Loads when all of the following conditions are in place: 1) affected unit is in cold shutdown; 2) reactor is depressurized with the reactor head removed (irradiated fuel may be in the reactor), 3) recirculation nozzle and jet pump plugs are installed, and 4) the recirculation system is tagged out of service.

To control load movement, maintenance instructions direct the crane operator to raise and transfer the load to its destination following safe load paths which have been designated in the instructions. To ensure that the established load paths are followed, all lifts performed per these instructions are done under the supervision of a designated individual (person-in-charge) who will verify the load path is clear prior to load movement. Deviations from approved load paths require prior approval of the Plant Operations Review Committee (PORC).

2 <u>Procedures</u> - Load handling procedures for the reactor building crane, placement of mobile cranes adjacent to the Units 1 & 2 and Unit 3 Standby Diesel Generator Building and the Intake Pumping Structure; and rigging over the Core Spray system pumps and the Recirculation system pumps are contained in maintenance instructions. These instructions contain sections covering scope of control, references, prerequisites, precautions and limitations, acceptance criteria, performance, inspections, tables of approved heavy load lifts, and drawings identifying safe load paths. Tables of the various approved heavy load lifts identify the crane to be used, approved rigging or lifting devices, component weights, and reference drawings and procedures.

3 <u>Crane Operators</u> - Programs for crane operator training, qualification, and conduct are contained in TVA Safety Procedures. The training programs include:

- Operating Practices and Functional Characteristics
- Rigging Fundamentals
- Electrical Maintenance
- Certification Skills for Overhead cab-operated Cranes
- Crane Operator Medical

These training programs incorporate all of Chapter 2.2 of ANSI (ASME) B30.2 - 1976.

4 Special lifting devices - BFN Special Lifting Devices are the reactor pressure vessel carousel head strong back, the drywell head strong back, the Preferred Engineering reactor cavity inspection platform lifting device, the dryer separator slings, recirculation pump motor lifting beam, and hardware to support dry cask storage lifts (Hi-Trac Lift Yoke, Hi-Storm Lifting Bracket, MPC Lift Cleats, Hi-Trac Lift Link Assembly and Mating Device-125D). Qualification of these devices was performed in accordance with NUREG-0612 and is documented in design documents. Inspection of these lift devices is performed pre-use and on a periodic bases in accordance with plant procedures. Refer to Chapter 10.4.4 and the Holtec International FSAR Report for the Hi-Storm 100 Cask System and for a description of Special Lifting Devices for the Hi-Storm 100 Cask System used for the dry cask storage of spent fuel.

5 Lifting devices that are not specially designed - All slings and other lifting devices not specially designed and used with cranes subject to NUREG-0612, Section 5.1, are designed, inspected, and tested in accordance with ANSI B30.9 - 1971.

6 Cranes are inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2 - 1976 - Cranes and hoists at BFN are inspected, tested, and maintained in accordance with specific site maintenance (MI) and preventative maintenance (PM) instructions which implement the requirements of the applicable ANSI (ASME) standard. Each handling system as listed below has its own unique instruction or procedure to control inspection and testing. The load handling system and applicable standard are as follows:

Handling System	Procedure	Reference Standard
Reactor Building Crane	MI	ANSI B30.2 - 1976

	Mobile Cranes	PM	ANSI B30.5 - 1976

7 The reactor building crane was designed to meet the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976 and CMAA-70 - 1975. Refer to FSAR Chapter 12.2.2.4, 12.2.2.5, 12.2.2.8, and the Holtec International FSAR Report for the Hi-Storm 100 Cask System for additional details on the design requirements for the reactor building overhead crane and the supporting structural steel features. Analysis results for the reactor building overhead crane, supporting structural steel features, and special lifting devices are documented in design calculations, vendor technical reports, and / or other design documents.

C.8.4.2 Reactor Pressure Vessel Head (RPVH) Lifting Procedures

BFN operations and maintenance instructions are used to control lifts associated with reactor disassembly and reassembly, refuel activities, and dry cask storage activities. These instructions and TVA Safety Procedures contain requirements to ensure the single-failure-proof equivalency of the reactor building crane is maintained. Additionally sections 3.9.4, 3.9.5, 3.9.6, and 3.9.7 of the BFN Units 1, 2, and 3 Technical Requirements Manual contain requirements for the establishment of operability of the reactor building overhead crane and requirements for the performance of lifts and equipment staging in conjunction with reactor disassembly / reassembly and refueling activities. These requirements include:

- Crane ambient temperature operating limits.
- Preventive maintenance, inspection, and functional testing of the reactor building crane.
- Crane safety functions verification requirements.

The BFN reactor building crane was evaluated against NUREG-0554, “Single Failure-Proof Cranes for Nuclear Power Plants,” as part of the station response to NUREG-0612, Section 5.1.3 (1) (and thus Section 5.1.6) compliance. This evaluation indicated that the crane is equipped with numerous single-failure-proof features. These features, also as described in greater detail in FSAR Chapter 12.2.2.5 and the Holtec International FSAR Report for the Hi-Storm 100 Cask System, include:

- The Dry Cask Storage spent fuel lift yoke is designed to higher factors of safety than specified in ANSI N14.6 - 1993

- Cab Mounted Emergency Stop Button
- Floor Mounted Emergency Stop Buttons
- Overload Protection
- Overspeed Detection
- Main hoist is equipped with a single-failure proof hoist system designed in accordance with Ederer's Generic Licensing Topical Report EDR-1 to provide compliance with NUREG-0554.
- All the crane controls are spring-returned to "off".
- Undervoltage protection is provided on all motions to sense low, or loss of, control voltage and cause the driven equipment to stop.
- Two overhoist limit switches and one down-travel limit switch are provided on each hoist.
- A torque-proving circuit checks that current is actually flowing in the main and auxiliary hoist drive motor's armatures before the motor brakes are permitted to be released.
- Designed for Safe Shutdown Earthquake with the Maximum Critical Load

NEI 08-05, "Industry Initiative on Control of Heavy Loads," defines the requirements for an equivalent single-failure-proof crane for the purposes of lifting of the reactor head. In addition to having the required safety features, the following measures are provided for the reactor head/reactor internals lifts associated with refueling activities over the reactor, drywell head lifts, and dry cask storage associated lift over the spent fuel pool:

- All safety functions of the crane are verified to be operational prior to performing the lift
- Direct communications are provided between the Crane Operator, Person-In-Charge and Signal Person via headsets
- Emergency stop buttons are manned during lift
- Backup Emergency Stop Signal is provided
- Pre-job brief performed that includes identification of supervisory oversight, establishment of lift management protocol, acceptable travel limits of crane, verification of load travel path
- Maintenance rule (a)(4) measures addressed in outage safety plan

C.8.5 Safety Evaluation

Heavy load lifts at BFN are performed safely and in accordance with NUREG-0612. Basis is provided by:

- Controls implemented by NUREG-0612, Section 5.1.1, make the risk of a load drop very unlikely.

- The use of a single-failure-proof crane makes the risk of a reactor head load drop, drywell head load drop, or a load drop associated with refueling activities over the reactor extremely unlikely and acceptably low.

- The risk associated with the movement of heavy loads is evaluated and controlled by station maintenance instructions, operations procedures, and site procedures.

C.9 References

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