Chapter 16

DESIGN CRITERIA FOR STRUCTURES AND EQUIPEMENT

16.1 SEISMIC DESIGN CRITERIA FOR STRUCTURES AND EQUIPEMENT

16.1.1 Definition of Seismic Design Classifications

All equipment and structures were classified as seismic Class I, Class II, or Class III as recommended in:

a) TID-7024, “Nuclear Reactors and Earthquakes,” August 1963,
b) G.W. Housner, “Design of Nuclear Power Reactors Against Earthquakes,” Proceedings of the Second World Conference on Earthquake Engineering, Vol. I, Japan 1960, Pg. 133, 134 and 137,
c) 10 CFR 100 Appendix A, “Seismic and Geological Siting Criteria for Nuclear Power plants,”
d) USNRC Reg. Guide 1.29, Rev 3, Sept. 1978, “Seismic Design Classification” and
e) ANSI/ANS-58.14-1993, “Safety and Pressure Integrity Classification Criteria for Light Water Reactors.”

Class I

Those structures, systems and components, which must remain functional during or following a Safe Shutdown Earthquake (SSE)* to ensure: (i) The integrity of the reactor coolant pressure boundary, (ii) the capability to shutdown the reactor and maintain it in a safe shutdown condition, and (iii) the capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the guidelines of 10 CFR 100. Also included in this definition are those structures, systems, and components that do NOT perform a safety-related function, but which must maintain structural integrity during or following a SSE to mitigate deleterious effects of system seismic interaction.

*NOTE: The Safe Shutdown Earthquake defines that earthquake which has commonly been referred to as the Design Basis Earthquake (DBE).

Class II

Those structures, systems and components which are necessary for continued operation without undue risk to the health and safety of the public when subjected to the effects of an Operating Basis Earthquake (OBE) in combination with normal operating loads.

Class III

Those structures, systems and components which are not directly related to reactor operation and containment, and which do not have to maintain structural integrity during or following a SSE.

16.1.2 Classification of Particular Structures and Equipment

The following are examples of particular structural and equipment classifications. These classifications are not intended to be all inclusive.

Item	Class
Buildings and Structures	
Containment (including all penetrations and air locks, the concrete shield, the liner and the interior structures)	I
Fan House	I
Spent Fuel Pit	I
Electrical Tunnels	I
Fuel Storage Building	III
Control Room	I
Waste Holdup Tank	I
Liquid Waste Storage Building	III
Diesel Generator Building	I
Primary Auxiliary Building	I
Containment Access Facility (Structural steel and PAB interfaces only; balance of structure is Class III)	I
Control Building	I
Auxiliary Feedwater System Enclosure (lower portion of Shield Wall area)	I
Intake Structure (The 124 by 58 feet reinforced concrete structure that house the circulating water and service water pumps and their associate equipment)	I
Service Water Pipe Chase and Adjacent Connecting Discharge Canal Wall	I
Turbine Structure	III
Replaced Steam Generator Storage Facility	III
Buildings Containing Conventional Facilities	III

Item	Class
Equipment Piping and Supports*	
Reactor Control and Protection System	I
Radiation Monitoring System	I
Nuclear Process Instrumentation and Controls	I
Reactor • Vessel and its supports • Vessel internals • Fuel Assemblies • RCC Assemblies and Drive Mechanisms • Supporting and positioning members • Incore Instrumentation Structure	I
Reactor Coolant System • Piping and Valves (including safety & relief valves) Steam Generators • Pressurizer • Reactor Coolant pumps • Supporting and positioning members	I
Engineered Safety Features • Safety Injection System (including safety injection and residual heat removal pumps, refueling water storage tank, accumulator tanks, boron injection tank, residual heat exchangers and connecting piping and valving) • Containment Spray System (including spray pumps, spray headers, spray additive tank [retired] and connecting piping and valving) • Containment Air Recirculation Cooling and Filtration System (including fans, coolers, ducts, valves, absolute filters and demisters)	I
Primary Auxiliary Building Ventilation System	I
Condensate Storage Tanks	I

*NOTE: Class I components (equipment, piping, instrumentation, etc.) located in or supported on a Class II structure are protected from earthquake damage or are backed by other Class I components located in or supported by a Class I structure.

Item	Class
Pressurizer Relief Tank (including discharge piping downstream of pressurizer safety relief valves)	II
Residual Heat Removal Loop	I
Containment Penetration and Weld Channel Pressurization System	I
Component Cooling Loop	I
Isolation Valve Seal Water System	I
Sampling System	II
Spent Fuel Pit Cooling Loop	II & III
Fuel Transfer Tube	I
Emergency Power Supply System	I
• Diesel generators and fuel oil storage tank	
• DC power supply steam system	
• Vital AC power supply system (instrument bus inverters)	
• Power distribution lines to equipment required for transformers and switchgear supplying the engineered safety features	
• Control panel boards	
• Motor control centers	
Battery Chargers 31,32, 33 and 35	I
Battery Charger 34	III
Station Service Transformer	III
Control Equipment, facilities and lines necessary for the above seismic Class I items	I
Control Room Air Conditioning System	I
Hot Penetration Cooling System	I, III
Steam Generator Blowdown System	I, II & III
Containment Crane	I
Manipulator, Fuel Storage Building Crane and other cranes	III
Conventional equipment, tanks and piping, other than seismic Class I and Class II	III

Item	Class
Waste Disposal System	I
• Chemical drain tank	
• Waste Holdup Tanks	
• Sump Tank	
• Gas Decay Tanks	
• Spent Resin Storage Tank	
• Reactor Coolant Drain Tank	
• Compressors	
• Waste Holdup Tank Pumps	
• Sump Tank Pumps	
• Interconnecting Waste Gas Piping	
• All elements not listed as seismic Class I	III
Emergency Feed, and Service Water Pumps and Piping	I
Backup Service Water Pumps and piping	III
Fire Protection System (Piping Supports)*	I
• Diesel Generator Building	
• Control Building	
• Primary Auxiliary Building	
• Fan House	
• Auxiliary Feed Pump Room	
• Electrical Tunnel	
• Containment Building	
• Fuel Handling Building	
*Seismically analyzed to Class I criteria in accordance with Section 9.6.2	
Primary Makeup Water Storage Tank	I
De-icing Pit and Pumps	III
Plant Vent	I
The Chemical and Volume Control System is considered seismic Class I except for the items listed below:	
Batch Tank	II
Monitor Tanks	II
Monitor Tank Pumps	II
Chemical Mixing Tank	II
Resin Fill Tank	III

The only portions of the plant which might carry substantial radioactivity, and which are seismic Class I, but which are not required because of safeguards operation or the safe shutdown and isolation of the reactor, are portions of the Chemical and Volume Control System and Waste Disposal System.

The specific components in the Chemical and Volume Control System are the volume control tank and holdup tank with associated piping, valves and supports. These components are all seismic Class I. In addition, the design of the system tanks and their location was based upon the commitment that a vessel rupture would not cause doses in excess of 10 CFR 20 limits at the exclusion radius

The specific components in the Waste Disposal System are the gas decay tanks with the associated piping, valves and supports. These components are all seismic Class I. In addition, the gas decay tanks of the Waste Disposal System have been designed such that the failure of any tank will not exceed 10 CFR 20 doses at the exclusion radius.

The analysis showing that the rupture of the volume control tank or a gas decay tank does not exceed the special dose limits selected for Indian Point Unit 3 is found in Chapter 14.

Those components of the Chemical and Volume Control System that are not seismic Class I are listed above. Those components of the Waste Disposal System which are not seismic Class I are as follows: liquid waste holdup tanks in Liquid Waste Storage Building, regenerant tank, baler, and reactor coolant drain tank. Failure of these components will not result in offsite doses in excess of 10 CFR 20 limits at the site exclusion radius.

16.1.3 General Seismic Design Criteria and Damping Values

The general seismic criteria and methods of analysis described in this section are those which were utilized during the design phase of Indian Point 3. Details of the seismic piping reanalysis for safety related systems conducted by the Authority during 1979 and 1980 are presented in Section 16.3.5.

Class I

All components, systems and structures classified as seismic Class I were designed in accordance with the following criteria:

1. Primary steady state stresses, when combined with the seismic stress resulting from the application of seismic motion with a maximum ground acceleration of 0.05g acting in the vertical and 0.1g acting in the horizontal planes simultaneously, are maintained within the allowable stress limits accepted as good practice and, where applicable, set forth in the appropriate design standards, e.g., ASME Boiler and Pressure Vessel Code, USAS B31.1 Code for Pressure Piping, ACT 318 Building Code Requirements for Reinforced Concrete, and AISC Specifications for the Design and Erection of Structural Steel for Buildings.

2. Primary steady state stresses when combined with the seismic stress resulting from the application of seismic motion with a maximum ground acceleration of 0.10g acting in the vertical and 0.15g acting in the horizontal planes, simultaneously, are

limited so that the function of the component, system or structure shall not be impaired as to prevent a safe and orderly shutdown of the plant.

No loss of function implies that rotating equipment will not freeze, pressure vessels will not rupture, supports will not collapse under the load, systems required to be leak tight remain leak tight and components required to respond actively (such as valves and relays) will respond actively. The criteria for functional adequacy of the structures state that stresses do not exceed yield when subjected to seismic motion with a 0.15g maximum ground acceleration. Seismic Class I equipment associated with the primary reactor coolant loop was designed in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code for Nuclear Vessels for response to a 0.15g maximum ground acceleration earthquake. All seismic Class I piping was designed in accordance with the USAS Code for Pressure Piping B31.1.0 for response to 0.15g maximum ground acceleration earthquake.

For the Containment design, refer to the Containment Design Report (Appendix 5A).

3. The seismic design criteria and qualification testing employed to assure the adequacy of seismic Class I electrical equipment are discussed in Section 7.2. The control board is not considered protection equipment. Typical switches and indicators for safeguards components were tested to determine their ability to withstand seismic forces without malfunction which would defeat automatic operation of the required component. The control boards are stiff, and past experience indicates that amplification of the board structure and accelerations seen by the devices mounted therein is considerably less than the subsequent acceleration which was shown the device could withstand in testing. Some components, for instance most pumps, required no additional restraints in order to meet the seismic criteria. Tanks generally required thicker walls and/or wall stiffeners and heavier support members and anchor bolts. Battery racks and instrument racks generally required heavier supports, cross bracing and heavier anchor bolts. The protection system equipment racks are bolted to the floor; no other seismic restraints were employed or deemed necessary to meet the seismic criteria. The type testing described in Section 7.2 used the same bolting arrangement as employed in the plant installation.

Seismic analysis of selected seismic Class I components including heat exchangers, pumps, tanks and valves, as well as seismic Class I structures, was performed using one of three methods depending on the relative rigidity of the equipment being analyzed:

(1) Equipment which is rigid and rigidly attached to the supporting structure was analyzed for a g-loading equal to the acceleration of the supporting structure at the appropriate elevation,

(2) Equipment which is not rigid, and therefore a potential for response to the support motion exists, was analyzed for the peak of the floor response curve with appropriate damping values;

(3) In some instances, non-rigid equipment was analyzed using a multi-degree-of-freedom modal analysis including the effect of modal participation factors and mode

shapes together with the spectral motions of the floor response spectrum defined at the support of the equipment. The inertial forces, moments, and stresses were determined in each mode. They were then summed using the square-root-sum-of-the-squares method. Where structures were too complex to analyze, testing was performed.

The reactor coolant loop piping and main steam and main feedwater piping inside containment were seismically analyzed by Westinghouse using the computer code WECAN for the Reactor Coolant Loop and the computer code WESTDYN for the main steam and main feedwater lines. Verification of the computer codes for both static, linear and non linear elastic dynamic analysis capability has been performed. Reduced modal analysis method and modal superposition method and modal superposition method are used in the time history seismic analyses.

The reduced modal analysis is used to determine the natural frequencies and mode shapes for a linear, undamped structure. This analysis requires the specification of dynamic or active degrees of freedom (DOF) for the model, which are a subset of the total number of DOF. The selection of dynamic DOF must be such that the low frequency spectrum can accurately be presented while a reduced eigenvalue problem is solved. In other words, the selected or dynamic (or active) DOF should be able to describe the frequency modes at interest.

The modal superposition method gives a time history solution for the response of an arbitrary structure subjected to known modal forces or ground acceleration time histories. The structure may include linear or non-linear elements. The uncoupled modal equations are integrated analytically.

The input to the time history DBE seismic analysis is in the form of time history seismic motions applied individually at the containment base mat in the north-south, east-west and vertical direction. These time histories seismic motions are based on those used in developing response spectra. The total response is obtained by determining the maximum response from combining absolutely one of the horizontal responses with the vertical seismic response.

Seismic Class I piping having a diameter 6” or larger plus the high head safety injection piping were initially designed statically using spacing tables which reflected the simultaneous application of horizontal and vertical spectral accelerations corresponding to 0.67 and 0.5 times the peak of the amplified floor response spectrum, respectively, developed at the support elevation of the piping system. A multi-degree-of-freedom dynamic analysis using the computer code ADLPIPE employing a dynamic model of the system and the applicable floor response spectrum as input motion was then performed to confirm the static design and analysis. The dynamic analysis successfully confirmed the conservatism of the static design.

Seismic Class I piping less than six inches in diameter was statically analyzed using spacing tables for simultaneously applied horizontal and vertical spectral accelerations corresponding to 2.0 and 1.33 times the peak of the amplified floor response spectrum, respectively, developed at the support elevation of the piping system. The coefficient of two times the peak of the amplified floor response spectrum was selected to account conservatively for modal participation factor effects in each mode and the contribution of higher modes. The design conservatism inherent in such a procedure has been verified by earlier comparative studies (Ginna, H.B. Robinson, and IP-2 Plants) relating seismic design stresses determined by coefficients from the peak of applicable floor or ground response spectrum to those determined by multi-degree-of-freedom detailed modal dynamic analysis.

The six inch diameter was selected as the dividing point because the reduction in pipe support hardware made possible by the more rigorous multi-degree-of-freedom detailed modal dynamic analysis below the six inch size (as opposed to the simplified double-the-peak response) did not warrant its use.

Non-rigid components and equipment components and equipment were only analyzed for an equivalent static load for vertical and horizontal seismic inputs if a dynamic analysis of a multi-degree-of-freedom model of similar component or piece of equipment has shown that the equivalent static load used gives conservative results. It is noted that, as described above, for piping having a diameter less than six inches, twice the peak of the floor response spectrum was used to determine the equivalent static loading. Analytical methods employed in the design of other seismic Class I structures, systems, and components are:

	ITEM	METHOD
1.	Reactor coolant loop piping and main stream and main feedwater piping inside containment	Multi-degree-of-freedom modal analysis response spectra
2.	All other Class 1 Piping $\geq$ 6” Dia. (including two inch high head safety injection lines)	Equivalent static analysis and confirmatory multi-degree-of-freedom modal analysis response spectra
	$<$ 6” Dia	Equivalent static analysis
3.	Refueling Water Storage Tank	Multi-degree-of-freedom modal analysis response spectra
4.	Primary Auxiliary Building ventilation system	Equivalent static load
5.	Condensate Storage Tank	Multi-degree-of-freedom modal analysis response spectra
6.	Containment Penetration and Weld Channel Pressurization System	Equivalent static load
7.	Diesel Generators	*See NOTE
8.	Fuel Oil Storage Tanks	No specific seismic design (UL approved, buried, atmospheric design pressure)
9.	DC Power Supply System	Equivalent static load

10.	Power Distribution lines to equipment required for transformers and switch-gear supplying the engineered safety features	Equivalent static load analysis on cable tray supports
11.	Control equipment, facilities and lines necessary for Items 6 through 9	Equivalent static load
12.	Auxiliary Feedwater System and Building	As outlined in the Authority's response to NRC Generic letter No. 81-14 (IPN-81-66, 8/28/81)
13.	Containment crane	Equivalent static load
14.	Emergency Boiler Feed Pumps and Service Water Pumps	Equivalent static load

*NOTE: No seismic design analysis was provided by the manufacturer of the Emergency Diesel Generator. However, the manufacturer of the diesel engine stated the following: "The diesel engine provided (for IP-2 and IP-3) was originally designed as a prime motive power unit for locomotive service. To meet these requirements, all component parts of the engine were designed to withstand minimum shock loads of 2.5g in any direction. This engine when modified for other uses retain this design criteria, as well as all allied equipment required. The engine foundation and sub-base are included."

In addition, the manufacturer of the generator portion of the units stated: "Machines of this type have been transported via rail shipment all over the United States without experiencing difficulty. Rail shipment experience indicate that shock loads of a magnitude of 2G's are common."

The methods utilized to determine the seismic input to these components are stated in the seismic design criteria.

The following criteria and procedures were used in formulating the mathematical model for the reactor coolant loop. Each portion of the piping system (straight runs and elbows) was subdivided into discrete elements. The mass of each of these elements was concentrated at the center of gravity of the elements. The major components were subdivided into discrete elements and the masses were located so as to (1) maintain the proper total mass of the component (2) maintain its moment of inertia about the center of gravity of the component (3) to maintain the position of the center of gravity of the component.

Structures having significant eccentricities between the centers of mass and centers of shear were modeled mathematically so that torsional effects could be considered. These models consisted of lumped masses having effective translational and rotational inertia connected by springs simulating the elastic restraints which included effective torsional stiffness.

A rational basis for the effect of seismic torsion has been developed by N.M. Newmark.[1] [2] A key parameter in Newmark's work is the transit time of the soil wave motion to pass over the long dimension of the building.

The shorter the time the less the torsional effect. Because Indian Point is on hard rock, the transit time is quite small. Therefore, the seismic torsional effect is not significant and can be neglected.

For the Containment, the concrete was assumed not to participate in resisting seismic shear even though experimental evidence suggested such contribution is significant, even for biaxially loaded concrete in tension. Therefore, the ductility of the shear resisting mechanism was taken to be provided entirely by the reinforcing steel acting in tension to carry diagonal tension loads.

For all other seismic Class I structures, the standard horizontal and vertical reinforcing in each face of walls and slabs provided the mechanism to resist shear loads which included torsional effects. The design was in accordance with the procedures in ACI-318-63 "Building Code Requirements for Reinforced Concrete," June 1963.

The locations of seismic supports and restraints for all seismic Class I piping, down to ¾ inch in diameter, were determined by the Architect Engineer and shown on installation drawings. In the event that a support or restraint could not be located as specified on the installation drawings, a reanalysis was performed prior to relocation. An as-built verification program has produced revised drawings, which reflect actual field conditions.

Site Quality Control inspectors verified the final location, correct type of each device, and its proper installation. This verification was made by actual physical check using approved Architect-Engineer drawings, manufacturer drawings, contractor drawings. and benchmarks established for this purpose.

<u>Splice Stagger in the Containment and Other Seismic Class I Structures.</u>

In the Containment, seismic design criteria required that vertical rebar splices be staggered a minimum of 1'-2" and that seismic diagonal bar splices be staggered 1'-2" vertically in each direction. In the dome a 2'-0" stagger pattern was specified throughout for the Cadweld splices as well as the reinforcing splice plates, except for final closure pieces at the apex of the dome. Horizontal rebar splices were specified in elevation and in cross-section (bars or bar pairs) with 2'-4" nominal and 2'-0" minimum stagger.

The above requirements were generally satisfied during construction except in special cases where physical or layout problems occurred in isolated areas in the Containment.

For all seismic Class I structures, other than the containment, rebars were specified to be lap spliced in accordance with the requirements of

ACI-318-63 "Building Code Requirements for Reinforced Concrete." No other specific stagger requirements were formulated. In the Containment, mechanical splices were included in the design because of biaxial tensile stress conditions in the concrete which eliminate bond and require continuous rebar, and because of the ACI-318 requirement that lapped splices in tension cannot be used for bars greater than No. 11.

Splicing of Reinforcing Steel by Welding

Welding of rebar for splicing is not permitted. Strength welding of rebar to structural steel elements or other heavy rebar was not permitted. Tack welding of rebar was not permitted.

Although rebar was not welded it should be pointed out that transition and closure splices in the Containment Dome employed Cadweld splice sleeves welded to structural steel plate (ASTM A 516 GR60). In addition to the destructive testing of random samples employed for all cadwelding, the root and final pass of each weld was magnetic particle inspected.

Class II and III

All seismic Class II* structures and components were designed on the basis of a static analysis for a ground acceleration of 0.05g acting in the vertical and 0.1g acting in the horizontal directions simultaneously. The structural design of all seismic Class III structures met the requirements of the applicable building code which was the “State Building Construction Code,” State of New York, 1961. This code does not reference the Uniform Building Code.

The design of seismic Class I piping was subject to loading combination and corresponding stress limits which included loads due to the Design Basis Earthquake and Operating Basis Earthquake while the seismic Class II piping was subject to loads associated only with the Operating Basis Earthquake.

It has been found that in some cases, in the Containment Building the loading combinations and stress limits involving the Operating Basis Earthquake will govern the design of piping systems with respect to seismic criteria. In these cases, seismic Class I or Class II piping were designed for Operating Basis Earthquakes. It was therefore designed for the governing condition.

In those cases where it was shown the loading combinations involving the Design Basis Earthquake governed, the adjacent seismic Class II piping and supports were designed to the seismic Class I criteria.

*NOTE: There are no seismic Class II structures.

Effects of Failure of Class III Equipment on Safety-Related Equipment

A review of potential failures of seismic Class III equipment and the potential adverse effects of such failures on safety related equipment was conducted.

The review consisted of determining the seismic Class III lines in the Diesel Generator Building, Vapor Containment, Fuel Handling Building, Service Water Pump Area, Control Building, Turbine Hall, Primary Auxiliary Building and the Auxiliary Boiler feed Pump Room and assessing the flooding potential from each line. This was accomplished by identifying the seismic Class III systems and portions of systems and tracing them through drawings for location and arrangement in the plant. It was determined from the review, that failure of seismic Class III equipment would not potentially adversely affect the performance of safety related equipment in the following buildings: Diesel Generator Building, Vapor Containment, Fuel Handling Building, Service Water Pump Area and Turbine Hall.

The portion of the Indian Point 3 water-medium Fire Protection System in the Diesel Generator Building, although classified as a seismic Class I system, was investigated for the effect of inadvertent actuation. Two 500 gpm sump pumps provided in this building were sized to accept water from the rupture of the diesel generator cooling water system. These pumps are controlled by independent float switch assemblies, each set at a different elevation to start the pumps in sequence. In case of a rupture of the service water pipe supplying water to the diesel generator cooling system, 24” drain lines collect water from the Diesel Generator Building and discharge into the river. The Diesel Generator Rooms are protected with a CO_2 fires suppression system, and the air supply to the engines is via a snorkel directly from the outside. Performance of the Diesel Generators would not be adversely affected by either actuation of the Fire Protection System or rupture of the service water piping in the building.

The original design of the CO_2 fire suppression systems protecting the diesel generator rooms was susceptible to inadvertent operation during a seismic event, seismic interaction, interaction resulting from a tornado generated missile or an adverse environment resulting from a high energy line break in the Turbine Building. This is because the control panels are non-safety-related, seismic Class III and are located in a non-safety-related, seismic Class III structure. Subsequent to the original design, a design change was implemented to install an interfacing safety-related, seismic Class I auxiliary control panel which prevents an inadvertent operation of the systems from resulting in a CO_2 discharge or result in an unacceptable loss of the ventilation systems which serve these rooms.

Essentially, all the equipment in containment is seismic Class I. Flooding in containment would be indicated within a few minutes by various methods, including humidity detectors and sump level sensors. A description of the leak detection systems is provided in Section 6.7.1.

A portion of the spent fuel cooling loop in the Fuel Storage Building and Primary Auxiliary Building is classified as seismic Class III. The largest source of water in this building is the storage pool. The spent fuel storage pool cooling connections enter near the water level at the top of the pool with this system.

The overhead piping for the new screen wash system has been designed such that in case of failure:

1) All operating pumps will be stopped by a low pressure switch;
2) Failed piping shall drain through the screens and pump back into the bays; and
3) Any water spilled on 15 foot deck shall be insignificant to cause flooding.

No safety related equipment is located in the Turbine Hall. However, flooding from the Turbine Hall could potentially affect the performance of the 480 volt switchgear located in the Control Building at Elevation 15’. Since the Circulating Water System is an open system having absolutely no valves, and therefore no means of producing a high dynamic head, the probability of a failure is practically zero. However, to assure that the 480 volt switchgear would not be adversely affected by flooding , redundant level alarm switches were installed in the pipe tunnel at Elevation 3’3” of the Turbine Hall. These switches sense high water in the pipe tunnel and give an indication to the Control Room. In addition, a barrier was installed at the doorway to the switchgear room to provide protection from flooding up to 19’. The operators have ample time to investigate any flooding problem and take appropriate action by shutting down the circulating pumps to prevent flooding to Elevation 19’.

A DBE seismically induced break of the Turbine Hall Elevation 15'-0" fire protection header could result in flooding of the 480 volt switchgear with a potential loss of all four 480 volt emergency buses. Modification 93-3-433 FRW added six seismic Safety Related supports so that the portion of the Augmented Quality Related piping and deluge valves for the water spray systems for Main, Station, Auxiliary, and Unit Auxiliary Transformers, located in the Control Building, will be capable of withstanding a Design Basis seismic event.

Inadvertent actuation of the Fire Protection System in the electrical tunnels will not potentially affect the performance of safety-related equipment in the Control Building. The electrical tunnels are provided with floor drains to handle water from the cable tray fire protection spray system. These drains discharge to grade outside the tunnel.

The original design of the CO_2 fire suppression systems protecting the 480 V switchgear room and cable spreading room was susceptible to inadvertent operation during a seismic event, seismic interaction, interaction resulting from a tornado generated missile or an adverse environment resulting from a high energy line break in the Turbine Building. This is because the control panels are non-safety-related, seismic Class III and are located in a non-safety-related, seismic Class III structure. Subsequent to the original design, a design change was implemented to install an interfacing safety-related seismic Class I auxiliary control panel which prevents an inadvertent operation of the systems from resulting in a CO_2 discharge in either room or results in an unacceptable loss of the ventilation systems which serve these rooms."

The Primary Auxiliary Building was so designed that flooding from any elevation will result in the water settling at the lowest level (Elevation 15') as each room has various floor penetrations which permit drainage to this elevation. In addition, the stairways provide substantial flow area. Originally it was believed that the performance of the two Residual Heat Removal Pumps located in the 15' elevation of the Primary Auxiliary Building would be affected by flooding only if the water reached an Elevation of 19'. As stated in Con Edison's January 23, 1973 letter to the AEC, approximately 120,000 gallons of water would be required to cause flooding to elevation 19'. The combined volume, approximately 2,800 gallons, of all non-Class I tanks in the Primary Auxiliary Building would cause negligible flooding if they failed. There are several seismic Class III lines and fire protection systems in the Primary Auxiliary Building that have sufficient capacity to cause flooding of the Residual Heat Removal Pumps. The seismic Class III line in the Primary Auxiliary Building with the largest nominal flow rate would take approximately 6 hours to flood to Elevation 19'. Although it is evident from the above that operators would have sufficient time to discover that a failure in seismic Class III line has occurred and take appropriate actions to prevent flooding to the 19' elevation in the Primary Auxiliary Building, modifications were made to assure that there is adequate drainage area to preclude flooding of the Residual Heat Removal Pumps in the unlikely event that the flooding is not discovered. It was later determined that the pump motors would be impacted by flood water at an elevation lower than 19' and a modification was made in 2008 to install a flood control drain line and valve from the PAB to a manhole in the transformer yard. This will ensure flood level does not exceed the 18' 2 3/4" elevation. The drainage area is also adequate to preclude flooding from the Fire Protection System. (See Section 9.6.2)

The Systems Interaction Study determined the effects of the internal flooding in the Primary Auxiliary Building due to failure of Class II and III piping. The study concluded that Class II and Class III pipe breaks will result in a water level of 18' 5" after 9.9 hours.

Evaluation of the Auxiliary Boiler Feed Pump Area, located between the Containment and the Shield Wall, revealed that safety related equipment would not be affected by failure of the seismic Class II portion of the main steam system. Failure of the main feedwater lines, located above and the outside of the Auxiliary Boiler Feed Pump Room, would result in water accumulating at the 18'6" elevation. Performance of the Auxiliary Boiler Feed Pumps could be adversely affected only if the water reached Elevation 19'8" in the Auxiliary Boiler Feed Pump Room. Provisions were made to assure adequate drainage under the worst postulated conditions of the main feedwater line failure.

Installation of level alarm switches in the Turbine Hall and provisions in the Primary Auxiliary Building and the Auxiliary Boiler Feed Pump Area were made during the normal course of construction of the plant.

Also included in the review was the potential effect of chemical releases on safety related equipment. It was determined that chemical releases caused by failure of seismic Class III equipment would have no potential adverse effect on safety related equipment.

Ground Response Spectra

The seismic ground response spectra used in the design of Indian Point 3 are shown in Figure 16.1-1 for the Operating Basis (smaller) Earthquake maximum ground acceleration of 0.10g, and in Figure 16.1-4 for the Design Basis (larger) Earthquake maximum ground acceleration velocity 0.15g. The response spectra were developed from the average acceleration velocity displacement curves presented in TID-7024, Nuclear Reactors and Earthquake, for large-magnitude earthquakes at moderate distances from the epicenter. As such, the curves are made of the combined normalized response spectrum determined from components of four strong-motion ground accelerations: El Centro, California, December 30, 1934; El Centro, California, May 18,1940, Olympia, Washington, April 13, 1949 and Taft, California, July 21, 1952.

Figures 16.1-2 and 16.1-3 are plots of the smoothed site ground response spectra and the ground response spectra derived from the earthquake records from 2 and 5 percent damping. The smoothed response spectra plot was taken from Figure 16.1-1. The system period interval where response spectra acceleration values were calculated was 0.02 seconds.

As seen in Figure 16.1-2 and 16.1-3, the computed ground response spectra using a system period interval of 0.02 seconds is equal to or greater than the smooth response spectra for the site.

To assure that the flow response spectra conservatively reflect the effects of variations in assumptions made for structural properties, dampings, and soil structure interactions, the response spectra peaks were widened. This widening effect was applied to peak values and is proportional to the response frequency. The increase in width is greater at high frequencies. The sharp valleys due to discontinuities in the plot were raised by an averaging technique.

In order to reflect in a conservative manner the expected variations of the periods of vibration of the structures in the seismic response curves for Seismic Class I buildings, the response spectra peaks were extended in the period scale by an amount equal to or greater than ± 8.5%.

For seismic Class I structures, having peaks occurring above 5 cps, the peaks were widened by more than ±10%. The only structures which are widened by less than ± 10% are the Containment structure and the Shield Wall. The difference between the widening percentage used for these structures, whose significant peaks occur below 5 cps, and the widening percentage of ±10% was less than 0.1 Hz which is negligible compared to the significant peak frequency. Where this difference can be significant, peaks occurring at frequencies above 5 cps, the peaks were widened by more than ±10%.

Since no strong motion records were available for the Eastern United States, the method used appeared to be the most rational considering the amount of earthquake data currently available. In addition, this method was consistent with the procedure being carried out on the majority of the nuclear plants under construction at that time in the United States.

There was not sufficient data available at that time, particularly in the Eastern United States, to attempt to correlate specific site conditions to a particular response spectrum.

Damping Factors

Table 16.1-1 gives the damping factors used in the design of seismic Class I components and structures.

Combined Horizontal and Vertical Amplified Response Loading

Evaluations were made for the simultaneous occurrence of horizontal and vertical seismic input motions. The results of analyses for each of two orthogonal, horizontal directions of excitation were combined directly with the results for vertical excitation on the basis of absolute sums for piping systems analyzed by Westinghouse and on the basis of algebraic summation for piping systems analyzed by UE&C to verify the static design. Vertical response was assumed to be amplified to the same degree as horizontal motions were amplified in determining floor response motions. Since vertical ground motions were assumed equal to two-thirds of the horizontal ground motions, the resultant vertical floor response spectrum values are two-thirds of those values determined for horizontal floor response. If the combined modal responses for the two horizontal and the vertical directions were combined by the square-root-sum-of-the-squares, bases on statistical independence, the resulting stresses would bot be significantly different because of the conservative vertical floor response spectra values assumed on the absolute sum analysis method employed by Westinghouse.

Floor response spectra were generated by plotting maximum dynamic response (acceleration and / or velocity and / or displacement) versus the natural frequency of a series of single-degree-of-freedom oscillators for the floor time history input motion. The time history motion of a floor was determined by dynamic analysis of a multi-degree-of-freedom lumped mass and elastic spring model of the building using a time history ground input motion. The time history ground input was defined such that its ground response spectrum simulates the defined ground response spectrum for the site.

For analysis of mechanical components and piping systems, the modal deflections, forces, and stresses for each mode were computed utilizing the spectral response method for seismic analysis.

The combined total response was obtained by adding the individual modal responses utilizing the square-root-sum-of-the-squares method. Combined total response for closely spaced modal frequencies whose eigen vectors were orthogonal were handled in the above mentioned manner. In the rare event when two significant closely spaced modal frequencies occurred and their eigen vectors were parallel, the combined total response was obtained by adding the RMS values of all other modes to the absolute value of one of the closely spaced modes for the main reactor coolant piping. Since the probability of such a rare event was small, this was disregarded for all systems other than the reactor coolant piping. Forces, moments, deflections, etc., were determined in each mode separately and then combined to determine resultant values. Resultant shears and stresses were computed from these resultant forces and moments.

Natural modes of vibration are normally considered statistically independent and, therefore, a realistic total response was obtained by taking the square root of the sum of the squares of the individual modal responses. However, if significant natural frequencies were closely spaced and their eigen vectors were parallel, the natural modes were assumed to be statistically dependent. Therefore, the absolute value of the response in one of the significant closely spaced modes was added to the square-root-of-the-sum –of-the-squares of all the other modal responses. Two natural frequencies were considered to be closely spaced if their difference was less than ten percent of either value.

16.1.4 Seismic Class I Design Criteria for Vessels and Piping

The loading combinations and stress limits which were employed in the design of seismic Class I piping, vessels, supports, and other applicable components are shown in Table 16.1.-2. The stress limits presented in Table 16.1-2 were used only in conjunction with elastic system dynamic analyses and elastic components analyses.

The emergency condition stress limits were applied to all seismic Class I piping systems outside of the Reactor Coolant Pressure Boundary under load combination of Normal + DBE. This included the steam and feedwater lines inside the Containment, up to and including the isolation valves outside the Containment.

Where restraints on any pipe line were necessary in order to prevent impact on and subsequent damage to neighboring equipment or piping comprising the Reactor Coolant Pressure Boundary, etc., the piping restraint was designed such that a plastic hinge mechanism was not formed. For these systems, the stresses due to postulated pipe rupture loads were maintained within the faulted condition limits.

The design criteria in Table 16.1-2 list loading combinations and stress limits for piping and supports for normal, upset, and faulted categories. Criteria for restraints required that stress limits of supported equipment not exceed code limits for the applicable category. For the seismic Class I portion of the main steam and feedwater systems, loading due to pressure, deadweight, thermal, transient pressure, transient temperature, operating basis earthquake, design basis earthquake and pipe break were considered. For the seismic Class III portion of these systems, loadings due to pressure, deadweight, thermal, transient temperature were considered. In addition, operating basis earthquake loads were considered to the extent that they affect the seismic Class I portion and pipe breaks were considered insofar as a break in the seismic Class III portion may not cause a failure of the seismic Class I portion nor cause a violation of the Containment.

The water hammer effect during a postulated loss-of-offsite-power and / or a loss-of-coolant accident were considered in the design of seismic Class I service water piping and pipe supports in containment. This transient (water hammer) could be the result of an earthquake but the effect would be separated in time wherein seismic and transient (water hammer) loading are not combined.

Allowable stress or rated load criteria are contained in the Power Piping Code ANSI B31.1 (1967), the Manufacture's Standardization Society standard MSS-SP-58 for standard supports, or AISC-69 for non-standard supports.

For the seismic Class I portion of the main steam line out to the isolation valves, the restraints at the steam stop valves were designed for a steam pipe break load of 340 kips. Under this load, the maximum applied primary load or stress was limited to the yield strength of the material. The analytical methods used in designing and evaluating the design of the main steam line restraints are in fact steel structures, most of the calculations were based on beam diagrams and formulas for various static loading conditions.

The design of all mechanical supports and restraints of the main steam and feedwater lines was evaluated by an individual other than the designer. Both the designer and the evaluator were graduated structural engineers qualified in structural stress analysis. In the evaluation, consideration was given to the design criteria, allowable stresses and loading combinations, and the analytical methods used in the design.

To perform their function, i.e., allow core shutdown and cooling, the reactor vessel internals must satisfy deformation limits. For this reason the reactor vessel internals were treated separately in Section 14.3.4.

Piping, Vessels and Supports

The reasoning for selection of the above mentioned loading combinations and stress limits was as follows:

1) For the operating basis earthquake, the nuclear steam supply system was designed to be capable of continued safe operation. Equipment and supports needed for this purpose were required to operate within normal design limits as shown in Line 2 of Table 16.1-2.

2) In the case of the design basis earthquake, it was necessary to ensure that components required to shut the plant down and maintain it in safe shutdown condition do not lose their capability to perform their safety function. This capability was ensured by maintaining the stress limits as shown in Line 3 of Table 16.1-2. No rupture of a seismic Class I pipe can be caused by the occurrence of the design basis earthquake.

3) For the assumed case of a reactor coolant pipe rupture, limit stresses in the unbroken reactor coolant system legs and other seismic Class I vessels and pipes were again as noted in Line 4 of Table 16.1-2.

4) For the extremely unlikely event of the simultaneous occurrence of the design basis earthquake and a reactor coolant system pipe rupture the design of seismic Class I piping and components, excluding the broken pipe, was checked for no loss of function, i.e., the capability to contain fluid and allow fluid flow. Again this was assured by limiting the various stress combinations within the limits shown in Line 5 of Table 16.1-2.

Reactor Vessel Internals

Design Criteria for Normal Operation

The internals and core were designed for normal operating conditions and subjected to loads of mechanical, hydraulic, and thermal origin. The response of the structure under the operating basis earthquake was included in this category.

The stress criteria established in Section III of the ASME Boiler AND Pressure Vessel Code, Article 4, were adopted as a guide for the design of the internals and core with exception of those fabrication techniques and materials which were not covered by the Code, such as the fuel rod cladding. Seismic stresses were conservatively combined and considered primary stresses.

The members were designed under the basic principles of:

1) Maintaining distortions within acceptable limits,
2) Keeping the stress levels within acceptable limits, and
3) Preventing fatigue failures.

Design Criteria for Abnormal Operation

The abnormal design condition assumed blowdown effects due to a pipe break combined in the most unfavorable manner with the effects associated with the design basis earthquake.

For this condition the criteria for acceptability were that the reactor be capable of safe shutdown and that the engineered safety features be able to operate as designed. Consequently, the limitations established on the internals for these types of loads were concerned principally with the maximum allowable deflections. The deflection and stress criteria for critical components under normal operation, plus the design basis earthquake and blowdown excitation are presented in Section 14.3.4.

Movement of Reactor Coolant System Components

The criterion for movement of the reactor vessel, under the worst combination of loads, i.e., normal plus the design basis earthquake plus reactor coolant pipe rupture loads, was that movement of the reactor vessel not exceed the clearance between a reactor coolant pipe and the surrounding concrete to prevent excessive shear load on the RCS pipe should this limit be more restrictive than those listed in Table 16.1-2.

The relative motions between reactor coolant system components are controlled by the structures which are used to support the reactor vessel, the steam generators, the pressurizer and the reactor coolant pumps in such a way that the stresses in the carious components and pipes do not exceed the limits established in Table 16.1-2.

Effect of Fabrication and Environment on Materials Properties

The employment of qualified welding procedure and qualified welders and thorough inspections assured that welds on seismic Class I components and piping have little, if any, effect on the tensile properties of base materials. Tests performed by Westinghouse revealed no difference in tensile properties between welded and non-welded pipes.

Accidental imperfections of the order of magnitude of those that pass inspection were also expected to be of no significance. Because of chemistry control of the employed coolants and periodic inspections, corrosion was not anticipated to be a problem.

The only component affected by irradiation is the reactor vessel. Irradiation of the reactor vessel is significant only in the area adjacent to the core. High stress areas, i.e., nozzle to shell junctures, are only slightly affected by irradiation. The neutron exposure to these areas was calculated and its effect on the stress-strain curve evaluated. The corrected stress-strain curve was then used in the development of the limit curves.

Development of the Faulted Condition Stress Limits

The design limit curves that give the allowable piping and vessel stresses for faulted conditions were developed by using the approach presented in WCAP 5890, Rev. 1. [(3)] This report developed limit curves by using 50 percent of the ultimate strain as the maximum allowable membrane strain. Subsequent to the submission of WCAP 5890, Rev. 1, the allowable membrane strain was limited to 20 percent of the uniform strain. Design limit curves were developed by using the following procedure:

1) Use material data to develop stress-strain curves.

   Stress-strain curves of type 304 stainless steel Inconel 600 and SA 302B low alloy steel at 600 F were generated from tests using graphs of applied load versus cross-head displacement as automatically plotted by the recorder of the tensile test apparatus. The scale and sensitivity of the test apparatus recorder assure accurate measurement of the uniform strain.

   For materials other than these three, stress-strain curves were developed by conservative use of pertinent available material data (i.e., lowest values of uniform strain and initial strain hardening). When the available data was not sufficient to develop a reliable stress-strain curve, three standard ASTM tensile tests of the material in question were performed at design temperature. These data were conservatively applied in developing a stress-strain curve.

2) Normalize the ordinate *stress) of the stress-strain curves to the measured yield strength. (Figures 16.1-5, 16.1-6, and 16.1-7)

3) Use 20% of uniform strain as defined on the curve developed under Item 2 as the allowed membrane strain.

4) Establish the normalized stress ratio at 20% of uniform strain on the normalized stress ratio-strain curves developed under Item 2.

5) Establish the value of the absolute membrane stress limit.

Multiply the normalized stress ratio in Item 4 by the applicable code yield strength at the design temperature to get the membrane stress limit which represents a minimum value. As an alternate, the actual physical properties as determined from standard ASTM tensile tests on specimens from the same heats were used to determine the membrane stress limits. If such an approach was adopted, sufficient documentation was provided to support the actual material properties used.

6) Develop limit curves for the combination of local membrane and bending stresses.

The limit curves were developed by using the analytical approach presented in WCAP 5890, Rev.1, and the stress-strain curve up to the membrane stress limit as developed under Item 5. Stress and stability analysis results were compared with these limits.

Examples of design limit curves as developed by using the above procedure are given in Figures 16.1-8 and 16.1-9.

16.1.5 Seismic Design Bases

Design Organization Involved

The design organization which were involved in the seismic design of Indian Point 3 and their responsibilities were as follows:

- Westinghouse
  Responsible for performing a dynamic analysis of the plant seismic Class I structures using a modal analysis approach. Also responsible for preparing response acceleration spectra at selected points in the plant structures for use in the seismic analysis of piping and equipment. Also responsible for the seismic analysis and design of main coolant piping and nuclear steam supply system equipment.

- United Engineers and Constructors
  Responsible for overall coordination of seismic design. Also responsible for seismic design of structures based upon accelerations, shears and moments determined by Westinghouse in their dynamic analysis. Also responsible for analysis and design of balance-of-plant piping and equipment.

The design of all structures and equipment for the plant were either within the scope of supply of Westinghouse or United Engineers and Constructors (UE&C). UE&C had the overall responsibility for the proper execution of the seismic design.

The safety related items of equipment furnished with the nuclear steam supply system underwent seismic analysis by Westinghouse and (where applicable) Westinghouse subcontractors. Westinghouse had the responsibility for approving analysis performed by its subcontractors.

The overall program and the criteria employed were evaluated by Westinghouse. Records of the documented procedures which were followed in this work are applicable to all phases of design, interchange of design information among the involved organizations, revisions thereto, and coordination of all aspects of design (including seismic design), were maintained by the Authority, now Entergy.

All items within the plant were clearly identified as to their importance to overall safety and were classified as seismic Class I, II or III. Major structures, system and components and their respective classifications are listed in Section 16.1.2.

The design engineer utilized the appropriate generated response acceleration spectra to determine the appropriate earthquake loadings.

In order to assure that UE&C responsibilities were met with regard to structural seismic design, each member of the design group was issued a document containing design procedures to convert the analyses results to working drawings for construction. The result of the Westinghouse dynamic analysis, transmitted to UE&C, included shear and moments at critical portions of each Class I Structure. These results were used to design the structural elements to resist these loads in accordance with the criteria and the applicable codes referenced. In those cases where the structure had already been designed by UE&C using the peak of the applicable response curve, the Westinghouse results were checked to insure that they were less than all shears and moments used in design.

Documentation Procedures

The major interface regarding seismic design information was the flow of information between the designer of the structure and designer of the equipment and components which are attached to the structures. The cognizant Structural, Mechanical, Electrical and Instrumentation engineers each had structures and/or equipment for which they had lead responsibility. All of these groups were serviced by a mechanical "analysis" group which performed appropriate analyses which could be translated into loads and stresses and other design information for use by the various designers. The required seismic information was transmitted in writing to the cognizant engineer and this information became part of the design basis information for the structures or components. The design information was reviewed by the designer, the cognizant engineer and the independent second level reviewer.

Upon completion of the Westinghouse modal analyses of seismic Class I structures, the shear and moments at various elevations were transmitted to UE&C for their use in design. A copy of this information was given to the responsible designer for use in implementing the design. All correspondence was kept in a separate job file by chronological order to insure that the latest information was available. Although there were several revisions to some of the Westinghouse information (received after the initial UE&C designs were completed), this revised Westinghouse information was reviewed to assure adequacy of the original design. Where necessary, revisions were made.

For the Containment Building and Control Building, UE&C performed an independent modal analysis to verify the Westinghouse analysis. The results were sent to Westinghouse for their records and information.

When UE&C drawings were completed or revised, they were issued for construction. Copies were transmitted to WEDCO, Consolidated Edison, and Westinghouse. In the UE&C offices, revised drawings were removed from all files and marked void. The latest revisions were then substituted.

Design Control Measures

Each Engineering Division within United Engineers and Constructors and Westinghouse was responsible for the adequacy of the design produced by those divisions. As such it was the responsibility of the managers of the respective divisions to provide adequate controls to assure satisfactory designs.

After receipt of Westinghouse information, UE&C proceeded with drawing preparation. All completed drawings were independently checked by another designer to insure adequacy of design with regard to design criteria. In addition, the drawings were given an overall check by the design leader and a cursory check by the structural discipline engineer. The drawing was finally signed by the project manager. In addition, all containment structural drawings were transmitted to Westinghouse for approval prior to issue for construction. When the drawing was issued for construction, the letter giving Westinghouse approval was documented in the drawing title block for quick reference.

Purchase Requirements

Specifications issued by both Westinghouse and United Engineers and Constructors included, as a minimum, loading criteria equal to or greater than those developed for a given location in the Indian Point 3 structures. The equipment and component suppliers were required to perform analyses or tests verifying the design and integrity of safety related components, using the appropriate criteria as inputs, or the analysis was done by UE&C or Westinghouse.

All seismic analyses submitted by vendors supplying seismic Class I equipment, which were not under the scope of the Nuclear Steam Supply System Contract, were reviewed by the architect-engineer (UE&C). This review consisted of the following:

1) All seismic calculations or test reports were reviewed by the responsible principal engineer.

2) A separate independent review was performed by the responsible Analytical Division.

3) The results of the above reviews were coordinated by the principal engineer and comments were returned to the vendor.

4) Final acceptance of the adequacy of the seismic design was confirmed in writing by UE&C after all comments had been resolved.

All seismic analyses submitted by subcontractors to Westinghouse were reviewed by Westinghouse equipment and analytical engineering departments in a manner similar to that described above.

For safety related seismic Class I electrical and control equipment, type tests or analyses were conducted under seismic accelerations based upon the results of a multi-degree-of-freedom, time-history analysis of the structure and applicable frequencies to demonstrate the ability of the equipment to perform its functions.

The analyses, test procedures, and test reports as submitted, whichever were applicable, were reviewed by either Westinghouse or UE&C, depending upon the origin of the specification. The requirements for submittal and approval was included in the specification.

16.1.6 Procedure for Utilization of Station Seismic Monitoring Equipment Following an Earthquake

The purpose of this procedure is to provide a plan for the utilization of data from the seismic monitoring equipment installed at Indian Point 3 following an earthquake.

Use of this data, as specified in the procedure, enables the station operating personnel to determine what course of action to take following an earthquake.

Equipment

The seismic monitoring system consists of equipment located as follows:

Containment

1) Three Engdahl Enterprises Peak Shock Recorders, Model PSR 1200-H-V-12A, installed in a tri-axial mount at Elevation 46'- 0" on the base mat. These provide a plot of eleven points on the 2% damping curve for the vertical axis and two horizontal area. The eleven points are within the frequency range of 2.26Hz.

2) Two Kinemetrics, Inc. FBA-3 tri-axial force balance accelerometer one installed at Elevation 46'-0" on the base mat and one installed on the Containment Structure Wall at Elevation 99'-0" directly above the lower unit.

3) One Teledyne PRA-103 Peak Recording Accelerograph installed on each of the three following pieces of equipment:
    a) One steam generator
    b) One reactor coolant pump
    c) The pressurizer

Control Room, Elevation 53'

1) Two Kinemetric, Inc. Etna digital recorders to receive and record the data from the two FBA-3 accelerometer in containment.

2) An Engdahl Enterprise Model PSA-1575 Peak Shock Annunciator for visual warning that predetermined acceleration limits making up the 2% damping response spectrum have been exceeded at any or all of the eleven frequencies monitored.

Alarm

In the event of a strong motion earthquake, magnitude 0.01g or greater, an alarm will be annunciated in the Control Room that a seismic event is being recorded by the strong motion accelerographs.

Action Required

The actions required following an alarm are detailed in operating procedures available at the plant site.

16.1.7 Categorization of Structures, Systems and Components

The structures, systems and components of Indian Point 3 can be classified to lie within the following categories:

Safety Related

A system, part of a system, structure, and/or component shall be deemed Safety Related if it is necessary to ensure: 1) the integrity of the reactor coolant pressure boundary, or 2) the capability to shut down the reactor and maintain it in a safe, shutdown condition, or 3) the capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the guidelines of 10 CFR 100. Some Safety Related structures, systems and components are listed in Table 16.1-4. A detailed listing may be found in the EN-DC-167 Reference Document and the equipment database.

Non-Safety Related

A system, part of a system, structure, and/or component shall be deemed Non-Safety Related, if it is not essential for a safe shutdown, i.e., hot shutdown. Failures of this equipment could result in loss of power generation but would not endanger public safety.

Augmented Quality Related

A Non-Safety Related system, part of a system, structure and/or component shall be deemed Augmented Quality Related if it is a system, structure or component that performs a function which may have some significance to safety with respect to design criteria to which the Quality Assurance Program must be applied as applicable. Some of these systems and components are listed in Table 16.1-5. A detailed listing may be found in the EN-DC-167 Reference Document and the equipment database.

16.1.8 Use of Generic Implementation Procedure (GIP) for Seismic Adequacy of Equipment and Parts

The GIP (Reference 4), as modified and supplemented by the U.S. Nuclear Regulatory Commission Supplemental Safety Evaluation Report No. 2 (Reference 5), may be used as an alternative method to existing methods for the seismic design and verification of existing, modified, new and replacement equipment and parts classified as Seismic Class I except for the following:

1. Regulatory Guide 1.97 Category I Items
2. Residual Heat Removal System motor operators MOV-730 & MOV-731

Only those portions of the GIP listed in "Use of Generic Implementation Procedure (GIP) for New and Replacement Equipment and Parts (NARE)" (Reference 6) shall be used. The other portions of the GIP are not applicable since they contain administrative, licensing, and documentation information which is applicable only to the Unresolved Safety Issue (USI) A-46 program. GIP shall be used with limitations stated in IP3 Nuclear Safety Evaluation NSE 94-3-029 (Reference 7).

References

1) Newmark, N. M., "Torsion in Symmetrical Buildings," Proceedings of the Fourth World Conference on Earthquake Engineering, Santiago, Chile, 2, A-3, 1969, p. 19-32.

2) Newmark, N. M. and Rosenblueth, E. Fundamentals of Earthquake Engineering, Prentice-Hall, Inc., New Jersey, 1971.

3) Wiesemann, R. E., R. E. Tome and R. Salvatori, "Ultimate Strength Criteria to Ensure No Loss of Function of Piping and Vessels under Earthquake Loading", WCAP-5890, Revision 1.

4) Seismic Qualification Utility Group (SQUG), "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment", Revision 2, 02/14/1992.

5) Nuclear Regulatory Commission, "Supplement No. 1 to Generic Letter (GL) 87-02 That Transmits Supplemental Safety Evaluation Report No. 2 (SSER No. 2) on SQUG Generic Implementation Procedure Revision 2 As Corrected on February 14, 1992 (GIP-2)", May 22, 1992.

6) Seismic Qualification Utility Group (SQUG), "Use of Generic Implementation Procedure (GIP) for New and Replacement Equipment and Parts (NARE)", Revision 2, October 25, 1999.

7) Indian Point 3 Nuclear Safety Evaluation NSE 94-3-029, "Seismic Verification of Equipment by SQUG Generic Implementation Procedure (GIP)", Rev. 1, 11/23/1999.

TABLE 16.1-1

DAMPING FACTORS FOR CLASS I COMPONENTS AND STRUCTURES

Component	Per Cent of Critical Damping
Containment Structure:	
(a) Design Basis Earthquake (larger)	5.0
(b) Operating Basis Earthquake (smaller)	2.0
Concrete Support Structure of Reactor Vessel:	2.0
Steel Assemblies:	
(a) Bolted or Riveted	2.5
(b) Welded	1.0
Concrete Structures above Ground:	
(a) Shear Wall	5.0
(b) Rigid Frame	5.0
Piping	0.5

TABLE 16.1-2

LOADING COMBINATIONS AND STRESS LIMITS

OPERATING CONDITION AND LOADING COMBINATIONS	VESSELS	PIPING	SUPPORTS	PUMPS	VALVES
1. Normal (Deadweight, Thermal and Pressure)	$P_m \leq S_m$ $P_L \leq 1.5\ S_m$ P_m (or P_L) + $P_B \leq 1.5\ S_m$	P $\leq$ *	Within stress limits as provided by applicable code either AISC-69 or MSS-SP-58	ASME Section III, 1968 Edition	USAS 16.5 or MSS-SP-66
2. Upset (Normal + Operating Basis Earthquake)	P_m (or P_L) + P_B + Q $\leq 3.0\ S_m$ (See Notes 1 & 2)	P $\leq$ 1.2 *			
3. Faulted (Normal + Design Basis Earthquake Loads)	(a) $P_m < 1.25_m$ or S_y' Whichever is larger $P_L < (1.25_m)$ or $1.5\ S_y$' Whichever is larger P_m (or P_L) + $P_B < 1.5\ (1.25_m)$ Or $1.5\ S_y$ whichever is larger (See Note 3) or (b) Faulted Condition Stress Limits in Table 16.1-3	Design Limit Curves as discussed in the text (also see Note 4)	Permanent Deflections of Supports Limited to Maintain Supported Equipment Within Faulted Condition Stress Limits.	Maximum Average Membrane Stress $\leq$ 2.4 S	Maximum Average Membrane Stress $\leq$ 2.4 S
4. Faulted (Normal + Pipe Rupture Loads)					
5. Faulted (Normal + Design Basis Earthquake + Pipe Rupture Loads).					

TABLE 16.1-2
(Cont.)

LOADING COMBINATIONS AND STRESS LIMITS

Where	P_m	=	primary general membrane stress intensity
	P_L	=	primary local membrane stress intensity
	P_B	=	primary bending stress intensity
	S	=	allowable value as specified in design codes
	S_m	=	stress intensity value from ASME B&PV Code, Section III
	P	=	piping stress calculated per USAS B31.1 Code for Power Piping.
	s	=	allowable stresses from USAS B31.1 Code for Power Piping.

These limits may also apply to ASME Class C vessels

	Q	=	secondary stress intensity
	S_y	=	minimum specified material yield (ASME B&PV Code, Section III, Table N-421 or equivalent)

Note 1: The limits on local membrane stress intensity ($P_L < 1.5S_m$) and primary membrane plus primary bending stress intensity (P_m (or P_L) + $P_B < 1.5S_m$) need not be satisfied at a specific location if it can be shown by means of limit analysis or by tests that the specified loadings do not exceed 2/3 of the lower bound collapse load as per paragraph N-417.6 (b) of the ASME B&PV Code, Section III, Nuclear Vessels.

Note 2: In lieu of satisfying the specific requirements for the local membrane ($P_L < 1.5S_m$) or the primary plus secondary stress intensity (P_m (or P_L) + P_B + $Q < 3S_m$) at a specific location, the structural action may be calculated on a plastic basis and the design will be considered to be acceptable if shakedown occurs, as opposed to continuing deformation, and if the deformations prior to shakedown do not exceed specified limits, as per paragraph N-417.6(a) (2) of the ASME B&PV Code, Section III, Nuclear Vessels.

Note 3: The limits on local membrane stress intensity ($P_L < 1.8S_m$ or $1.5S_y$) and primary membrane plus primary bending stress intensity (P_M (or P_L) + $P_B < 1.8S_m$ or $1.5S_y$) need not be satisfied at a specific location if it can be shown by means of limit analysis or by tests that the specified loadings meet the requirements of paragraph N-417.10 (c) of the ASME B&PV Code, Section III, Nuclear Vessels; or , for Steam Generators, that the specified loadings do not exceed eighty percent of the lower bound collapse load.

Note 4: As an alternate to the design limit curves which represent a pseudo plastic instability analysis, a plastic instability analysis may be performed in some specific cases considering the actual strain-hardening characteristics of the material, but with yield strength adjusted to correspond to the tabulated value at the appropriate temperature in Table N-424 or N-425, as per paragraph N-417.11 (c) of the ASME B&PV Code, Section III, Nuclear Vessels. These specific cases will be justified on an individual basis.

TABLE 16.1-3

FAULTED CONDITION STRESS LIMITS FOR CLASS I VESSELS

System (or Subsystem) Analysis	Component Analysis	Stress Limits for Vessels		Test
		P_m	$P_m + P_B$	
ELASTIC	Elastic	Smaller of $2.4\ S_m$ and $0.70\ S_u$	Smaller of (2) $3.6\ S_m$ and $1.05\ S_u$	0.8 L_T (3) (4)
	Plastic	Larger of (3) $0.70\ S_u$ or $S_y + 1/3\ (S_u - S_y)$	Larger of (3) $0.70\ S_{ut}$ or $S_y + 1/3\ (S_{ut} - S_y)$	
	Limit Analysis	0.9 L_1	(3) (1)	
PLASTIC	Plastic Elastic	Larger of $0.70\ S_u$ or $S_y + 1/3\ (S_u - S_y)$	Larger of $0.70\ S_{ut}$ or $S_y + 1/3\ (S_{ut} - S_y)$	

NOTE:

(1) L_1 = Lower bound limit with assumed yield point equal to 2.3 S_m

(2) These limits are based on a bending shape factor of 1.5. For simple bending cases with different shape factors, the limits will be changed proportionally.

TABLE 16.1-3
(Cont.)

FAULTED CONDITION STRESS LIMITS FOR CLASS I VESSELS

(3) When elastic system analysis is performed, the effect of component plastic deformation on the dynamic system response will be checked. When this method is used, justification will be provided to show that the results of the elastic system analysis are valid.

(4) The limits established for the analysis need not be satisfied if it can be shown from the test of a prototype or model that the specified loads (dynamic or static equivalent) do not exceed 80 percent of L_T, where L_T is the ultimate load or load combination used in the test. In using this method, account shall be taken of the size effect and dimensional tolerances (similitude relationships) which may exist between the actual component and the tested models to assure that the loads obtained from the test are a conservative representation of the load carrying capability of the actual component under postulated loading for faulted conditions.

S_y = Yield stress at temperature

S_u = Ultimate stress from engineering stress-strain curve at temperature

S_{ut} = Ultimate stress from true stess-strain curve at temperature

S_m = Stress intensity from ASME Section III at temperature

TABLE 16.1-4

SAFETY-RELATED SYSTEMS

- Reactor Coolant System
  Includes: Pressurizer System and Associated Safety and Relief Valves
- Secondary Coolant System up to Second Isolation Valve
  Includes: Secondary Relief, Auxiliary Feedwater and Boiler Blowdown, with the exception of the motor-operated block valves and low-flow bypass valves which are exempted per NUREG-0800 criteria
- Chemical and Volume Control System
- Sampling System
- Containment Ventilation System
  Includes: Containment Air Recirculation Cooling and Filtration System
- Containment Spray
- Waste Disposal System
- Service Water-Essential Header
- Instrument Air System
- Fuel Handling System
- Reactor Protection System
- Engineering Safety Systems Protective System
- Process and Area Radiation Monitoring System
- Emergency Power System
- Containment Penetration and Weld Channel Pressurization System
- Isolation Valve Seal Water System
- Hydrogen Recombiner System [Historical Information]
- Safety Injection System
- Component Cooling System
- Residual Heat Removal System
- Spent Fuel Cooling System
- Control Room Ventilation System
- Fuel Building Emergency Exhaust System

SAFETY-RELATED STRUCTURES

- Containment

SAFETY-RELATED COMPONENTS

- Core and Reactor Internals
- Control Rods and Drives
- Incore Thermocouples
- Temperature Sensors in Auxiliary Feedwater Pump Room

TABLE 16.1-4
(Cont.)

SAFETY-RELATED CONSUMABLES

- Diesel Generator Fuel Oil
- Boric Acid
- Lubricating Oils for Safety-Related Components
- Sodium Hydroxide for Containment Spray System
- Weld Rod for Safety-Related Items
- Hydraulic Snubber Fluid

SAFETY-RELATED PROGRAM COMMITMENTS

- All times designated in Design Specification as ASME Section III, Classes 1, 2, and 3.
- Generic Letter 89-10 Motor Operators

NOTE: A detailed listing of structures and systems is provided in the EN-DC-167 Reference Document. “Safety Related” denoted the highest classification applicable to the system, structure or component. Lower classifications may exist within these systems, structures or components. Component QA classifications may be found in the equipment database.

TABLE 16.1-5

AUGMENTED QUALITY RELATED STRUCTURES, SYSTEMS AND COMPONENTS

- Packaging of Radioactive Materials for Transport and transportation of Radioactive Materials Under Certain Conditions
- Low Level Radiation Waste Storage Tanks (additions)
- Fire Protection System
- Meteorological Tower
- Temperature Sensors in Penetration Area of Primary Auxiliary Building
- Level Sensors – Lower Level Turbine Building
- Seismic Monitoring System
- Manipulator Crane
- Containment Polar Crane
- Instrumentation (e.g., indicators, recorders, alarms, etc.) not already specifically classified as Safety Related by other sections of the FSAR that is required for:
    - Executing emergency procedures,
    - Verifying that plant conditions are within limits of Technical Specifications, or
    - Determining the status of Safety Related equipment including bypasses and permissives
- Level Transmitters: LT-181A, 181B
- Hot Penetration Blower No.'s 31, 32, 33, & 34
- Steam Generator Feed Flow, Steam Flow and Level Recorders
- Spent Fuel Pit Bridge
- Six Pipe Plugs Located on the RCP Motor Flywheel
- Emergency Diesel Generator Starting Air Compressors and Controls
- Manual Handwheel Actuators
- Retainer Clips and Bolts for Closure Head O-Rings on Reactor Vessel
- Turbine Control Oil Auto Stop Trip
- Cotter Pins for VC Airlock and Equipment Hatch
- Changing Pump O-Rings and Gaskets
- AMSAC System
- Fuel Storage Building Crane
- Fuel Storage Building Emergency Ventilation System (FSBEVS) Pressure Boundary Doors

NOTE: This lists only a portion of those "Augmented Quality" non-safety related structures, systems, and components and instrumentation to which the QA Program must be applied, as applicable. A detailed listing may be found in the EN-DC-167 Reference Document and the equipment database.

16.2 TORNADO DESIGN CRITERIA

16.2.1 Definition of Design Basis Tornado

The plant is safeguarded from the tornados by the combined use of buildings and structures designed to withstand tornados, and by redundancy of components. All Class I buildings and structures were designed to withstand tornado winds corresponding to 300 mph tangential velocities, traverse velocities of 60 mph and a differential pressure drop of 3 psi in 3 seconds with no loss of function. The exception to this includes areas without safety related equipment or redundant equipment as discussed in FSAR Section 16.2.2.

All Class I buildings and structures were also designed to withstand various postulated tornado-generated missiles, including the following:

Horizontal Missiles

1) 4” x 12” x 12’ plank at 300 mph
2) 4000 lb. passenger car at 50 mph less than 25 ft. above the ground.

Vertical Missiles

1) 4” x 12” x 12’ plank at 90 mph
2) 4000 lb passenger car at 17 mph less than 25 ft. above the ground.

16.2.2 Tornado-Proof Systems and Equipment

Systems and Equipment Protected by Enclosure

All of the equipment which must be protected from tornados and tornado-generated missiles is contained within structures designed to withstand such loadings. The equipment or systems located within these structures include the following:

Primary Auxiliary Building

1) Safety Injection Pumps

2) Residual Heat Removal Pumps

3) Component Cooling Systems except portions of the piping loop in the Fuel Storage Building (Component Cooling Water operation is assured by the ability to provide make-up from the Primary Water Storage Tank).

4) Waste Disposal System (except for Waste Holdup Tank in Waste Holdup Tank Pit and Reactor Coolant Drain Tank and Pumps in the Containment)

5) Chemical and Volume Control System (except for Excess Letdown and Regenerative Heat Exchangers inside the Containment and Holdup Tanks in the Waste Holdup Tank Pit)

6) Refueling Water Purification Pump

7) Sampling Systems

8) Auxiliary Building Ventilation System (ducts and supply fans only)

9) Containment Spray Pumps

10) Spray Additive Tanks [retired]

11) Pressurization Air Receivers

12) Electrical Tunnels

13) Waste Hold-up Tank Pit

Control Building

1) Instrumentation Readouts and Controls
2) Control Room Ventilation System
3) Batteries and Battery Chargers
4) Instrumentation Air System
5) Additional CCR HVAC Cooling Condenser Units (restrained to the Control Building roof to prevent them from becoming missiles but are not tornado missile protected)

Containment

1) Reactor Vessel, Core, Instrumentation, and Controls
2) Primary Coolant System (including Pressurizer and Pressurizer Relief Tank)
3) Steam Generators
4) Residual Heat Removal Heat Exchangers
5) Reactor Coolant Drain Tank and Pumps
6) Excess Letdown and Regenerative Heat Exchangers
7) Accumulators
8) Recirculation Pumps
9) Containment Air Recirculation Cooling and Filtration System

Diesel Generator Building

Auxiliary Feedwater System Building

The service water pump motors are protected by the service water enclosure, which is surrounded by the Intake Structure Enclosure (ISE) Building. The service water enclosure is designed as seismic Class I structure. The sidings and roofings of the ISE are postulated to be airborne during a tornado but will be prevented from coming in contact with the service water pump motors by the service water enclosure.

The potential for damage to spent fuel assemblies stored in the fuel pool from either turbine-generated or tornado-generated missiles is very low. See Appendix 14A for the worst case assumptions of offsite exposures due to turbine missile damaged fuel assemblies. See WCAP-7572, “Effect of Tornado Missiles on Stored Spent Fuel” for analysis of offsite exposures due to tornado missile damaged fuel assemblies. In both cases, the resultant site boundary doses are well below the 10 CFR 100 guidelines.

Service Water Pipe Chase

The two redundant service water supply lines crossing the Discharge Canal are protected by the concrete pipe chase from tornado effects. A postulated tornado generated missile can collapse the 8” concrete slab (at the top of the pipe chase) locally and hit the upper supply line. The pipe is capable to withstand the impact of the missile and the fallen concrete. Pipe stress is still below the allowable stress limit permitted by code.

Systems and Equipment Protected by Redundancy

All components and equipment for safe shutdown and isolation of the reactor are housed within the tornado-proof structures described above, with the following exceptions. For these components and systems, adequate tornado protection is provided by redundancy:

1) Redundancy is provided for the vital 480 volt system by three independent systems. Onsite there are three emergency diesel generators which are redundant and tornado protected; offsite there is a 138 kV above-ground system and a 13.8 kV under-ground system.

2) The emergency feed requirements of the steam generators are assured by tornado protected pumps and redundant water supplies.

3) The water requirements of the primary system are assured by the availability of primary water storage tank, the refueling water tank and the boric acid tanks.

4) Service water supply is assured by redundancy of two supply lines, four screens and six pumps of which only two pumps, one screen and one supply line are required for prolonged shut-down. The intake structure itself is tornado proof. The Backup Service Water System is an additional source of service water independent of the intake structure. The redundant service water supply lines are either buried underground with a minimum of 2’-10” cover or are protected by a minimum of two feed of concrete or a 8 inch thick slab for their entire run. The minimum distance between the headers is one foot. This protection is sufficient for the missiles considered.

Design Procedures

Specific design procedures employed to evaluate the capability for the reinforced concrete structures to withstand tornado loadings were as follows:

1) The tornado loads were investigated considering overall structural effects. Overturning moments, base shears and toe pressure were checked considering the wind load, missile load, dead load and live load. The tornado loads were investigated considering local structural effects. Concrete and rebar stresses were checked considering wind loads, missile loads, dead loads and live loads.

2) For Items (1) and (2) above, the external wind loads, 3 psi negative pressure, and missile loads were considered in combinations yielding the most conservative load combination and thus the highest stress condition. Only one missile was considered acting at any time simultaneously with the wind loadings.

3) Missile penetrations into the reinforced concrete structure and corresponding loads on the structure were calculated by the following general procedure:

a) Calculate depth of penetration of the missile using the modified Petry Formula (1).

b) Calculate the impulsive force considering conservation of energy. The depth of penetration of the missile and the deflection of the structure are considered in calculated the impulsive force.

c) Calculate the equivalent static force by multiplying the impulsive force by the dynamic load factor considering a rectangular load pulse acting for the duration of missile impact.

d) Design the structure to resist the equivalent static force using recommended stress indices (2). The tornado protection structures, which are constructed of reinforced concrete, were designed to prevent missile penetration and spalling (by selection of moderate degree of damage allowable stress indices for structural design in accordance with Reference 1) of concrete from the walls, roof slab or dome impacted by the missile. Therefore, secondary missiles are not created which could damage or make inoperable Class I systems which must be protected from tornados.

For a more detailed description of the containment structure tornado analysis refer to Sections 2.2 and 2.4 and to the Containment Design Report in Appendix 5A.

Equipment and systems contained within tornado proof structures are protected from tornados and tornado missiles. Components and systems not housed within tornado-proof structures (but essential for safe shutdown and isolation of the reactor) are provided with protection to that function by component or system redundancy. The prior subheading, “Systems and Equipment Protected by Redundancy” discusses this.

Typical objects that could be postulated as potential tornado missiles were selected. These typical objects were approximated by the shape of simple objects like straight cylinders and slabs.

Assuming 300 mph tornado, an analysis was performed using the modified shapes. The results indicated which objects could be sustained or moved by the winds. Based on the above, the missiles for which plant protection was required were selected. These missiles are listed in Section 16.2.1.

16.2.3 Tornado Design Criteria

Tornado wind loads are converted to equivalent static structural loadings in accordance with the applicable portions of the wind design methods described in ASCE Paper No. 3269 “Wind Forces on Structures.” The provisions for gust factors and variation of wind velocity with height do not apply. The following factored load equation is used for those structures designed to resist tornado wind effects:

$C = (1 \pm 0.05)D + 1.0W'$ For containment structure

$C = 1.0D + 1.0W'$ For all other seismic Class I Buildings and Structures

where:

C = Required load capacity of section.

D = Dead load of the structure plus any normal operating live loads.

W' = Tornado wind load to include pressure drop effect where applicable.

The stress criteria used for this load criterion were for no gross yield of the primary structure with the yield stress levels reduced by the capacity reduction factors as defined in Chapter 5.

Three general criteria were adopted for the design of Indian Point 3 in tornado conditions:

I. A tornado will not cause a Loss-of-Coolant Accident.

II. A tornado will not impair the ability to safety shut the plant down.

III. A tornado, following Loss-of-Coolant Accident, will not impair the long term safety of the plant.

Criterion I

The Reactor Coolant System is contained entirely within the confines of the containment vessel. For the tornado to cause a Loss-of-Coolant Accident the tornado or tornado-produced missiles must penetrate the containment vessel. The design is such that penetration of the containment vessel is not credible.

Criterion II

There are two phases of reactor shutdown that must be considered; a shutdown to hot shutdown condition and a shutdown to cold condition.

Shutdown to Hot Shutdown Condition

The Reactor requires a number of basic services when held for an extended period in the hot standby condition:

a) Residual Heat Removal
b) Reactivity Control, i.e., as fission poisons decay
c) Pressurizer Pressure and Level Control
d) Auxiliary Building and Control Room Ventilation
e) Electrical Systems

These services require that a number of systems and equipment will continue to operate following a tornado:

a) Residual Heat Removal

Following a normal plant shutdown an automatic steam dump control system bypasses steam to the condenser and maintains the reactor coolant temperature at its no load value. This implies the continued operation of the steam dump system, condensate circuit, condenser cooling water, feedwater pumps and steam generator instrumentation. Failure to maintain water supply to the steam generators would result in steam generator dry out after some 34 minutes and loss of the secondary system for decay heat removal.

Redundancy and full protection where necessary is built into the system to ensure the continued operation of the steam generator units. If the automatic steam dump control system is not available independently controlled relief valves for each steam generator maintain the steam pressure. These relief valves are further backed up by code safety valves for each steam generator. Numerous calculations, verified by startup tests have shown that with the steam generator safety valves operating alone the Reactor Coolant System maintains itself close to the nominal no load condition. The steam relief facility is adequately protected by redundancy and local protection. For decay heat removal, it is only necessary to maintain the control on one steam generator.

For the continued use of the steam generators for decay heat removal, it is necessary to provide a source of water, a means of delivering that water and, finally, instrumentation for pressure and level indication.

The normal source of water supply is the secondary feed circuit; this implies satisfactory operation of the condenser, air ejector, condenser cooling circuit, etc. In addition to the normal feed circuit the plant may fall back on:

1) The condensate storage tanks
2) The city water storage tank
3) The city water supply

Feedwater may be supplied to the steam generators by either the electrical feedwater pumps or by the steam driven feedwater pump; these pumps and associated valves may be controlled both locally and remotely from the Control Room. In the event of loss of compressed air, local operation would be adopted.

For continued operation of the electrical feedwater pumps, the 480 volt system must be assured. This is discussed under item (e).

In addition, the diesel generators require the continued supply of fuel oil and service water; adequate redundancy and protection exist for this purpose.

Vital instruments and controls are provided both locally and in the Control Room.

b) Reactivity Control

Following a normal plant shutdown to hot shutdown condition, soluble poison is added to the primary system to maintain subcriticality.

For boron addition the Chemical and Volume Control System is used; control may be local or from the Control Room. Routine boration requires the use of:

- Charging pumps and volume control tank with associate piping.
- Boric Acid transfer pumps and tanks and associated piping.
- Letdown station.
- Non-regenerative heat exchanger and associated equipment.
- Component Cooling and Service Water Systems.
- Periodic operation of one reactor coolant pump for pressurizer homogenization; the auxiliary spray/heaters could be used if necessary.
- Compressed air for valve operation – manual could be adopted if necessary.

The vital items of this equipment are housed within the containment and the reinforced concrete auxiliary building. The Service Water System is protected by means of redundancy. In order to guarantee the operation of the system the 480 volt system must again be assured.

It is worthy of note that with the reactor held at hot shutdown conditions, boration of the plant is not required immediately after shutdown. The xenon transient does not decay to the equilibrium level until at least 9 hours after shutdown and a further period would elapse before the reactivity shutdown margin provided by the full length control rods have been cancelled. This delay would provide useful time for emergency measures although the essential systems are considered to be adequately protected within the auxiliary building and Containment Building. For loss of CCW due to a missile strike in the Fuel Storage Building, city water is available for hook-up (IPN-02-040).

c) Pressurizer Pressure Level Control

Following a reactor trip, the primary coolant temperature will automatically reduce to the no load temperature condition as dictated by the steam generator conditions. This reduction in the primary water temperature reduces the primary water volume and if continued pressure control is to be maintained primary water makeup is required. The pressurizer pressure level is controlled in normal circumstances by the Chemical and Volume Control System. This requirement implies the charging pump duty referred to for boration plus a guaranteed borated water supply. The facility for boration is safety protected within the Primary Auxiliary Building; it is only necessary to supply water for makeup. Water may readily be obtained from separate sources: that in the volume control tank, boric acid tanks, monitor tanks, primary storage tank, and refueling water storage tank.

Similarly to the two previous service requirements, the 480 volt system must be assured with the additional electrical load of the pressurizer heaters. Vital instruments and controls are provided both locally and in the Control Room.

d) Ventilation

The most essential ventilation requirements apply to the containment since in order to guarantee the satisfactory operation of the instrumentation and control systems the containment air temperature must be controlled to a tolerable level. This system again requires the satisfactory operation of the Service Water and Electrical Systems.

e) Electrical Systems

Protection from tornado is provided for the 480 volt switchgear and supply redundancy is provided by the diesel generators, the two above-ground incoming lines and the one below ground incoming line. The 6.9kV is fed by an underground 13.8 kV feeder from the Buchanan substation. The Buchanan substation consists of four buses.

Shutdown to Cold Condition

Plant cooldown is not an immediate requirement following major damage due to a tornado. For a cooldown, the basic services required are:

a) Residual Heat Removal
b) Reactivity Control
c) Pressurizer Pressure Level Control
d) Ventilation
e) Electrical Systems

A cooldown would not be attempted until full equipment facilities had been guaranteed.

Tornado missile damage to a small bore pipe in the Containment Cooling Loop in the Fuel Storage Building (FSB) would require isolation and repair or isolation of piping. Prior to establishing Residual Heat Removal during plant cooldown the CCW System would have to be refilled using operator action. The Primary Water Storage Tank is available to replace lost water inventory.

Criterion III

Following a Loss-of-Coolant Accident the residual heat is removed through internal recirculation conditions with the facility for external recirculation if required. The duty implies the continued operation of the Auxiliary Feedwater System together with the associated electrical and service water supplies. The recirculation systems are protected by the tornado proof containment and auxiliary buildings. The Electrical and Service Water Systems are assured by redundancy as previously discussed.

References:

(1) "Design of Protective Structures" by Arsham Amirikian, Navy Docks P-51, Bureau of Yards and Docks Department of the Navy, Washington, D.C., August 1950.

(2) TM5-855-1, Department of the Army Technical Manual, "Fundamentals of Protective Design (Non-Nuclear)," 1965.

16.3 DEMONSTRATION OF ADEQUACY OF SELECTED SEISMIC CLASS I ITEMS

16.3.1 Design of Seismic Class I Structures

A multi degree-of-freedom modal analysis was performed on all Class I building structures for Indian Point 3. The results indicate that all except the containment structure are rigid.

16.3.2 Analysis of Seismic Class I Equipment Other Than Reactor Coolant Pressure Boundary*

The ability of Class I equipment including heat exchangers, pumps, tanks, valves, motors, and electrical equipment components to withstand seismic loads was verified using one of the following methods:

(1) Equipment which is rigid and rigidly attached to its support structure was analyzed for a 'g' loading equal to the peak acceleration of the supporting structure at the appropriate elevation.

(2) Equipment which is not rigid and therefore potential for response to the support motion exists, was analyzed for the peak of the floor response curve for appropriate damping values.

(3) In some instances non-rigid equipment was analyzed using a multi-degree of freedom modal analysis. All contributing modes are considered. In addition, it should be pointed out that a sufficient number of masses is included in the mathematical models to insure that coupling effects of members within the component are properly considered. The results of these analyses indicate that the models contain more masses than necessary, and that future analyses of comparable equipment could be considerably simplified by considering fewer masses. The method of dynamic analysis uses a proprietary computer code called WESTDYN. This code uses as input, inertia values, member sectional properties, elastic characteristics, support and restraint data characteristics, and appropriate seismic response spectrum. Both horizontal and vertical components of the seismic response spectrum are applied simultaneously. The modal participation factors are combined with the mode shapes and the appropriate seismic response spectra acceleration to give the structural response for each mode. The internal forces and moments are computed for each mode from which the modal stresses are determined. The stresses are then summed using the square root of the sum of squares method.

(4) Type testing of selected electrical equipment has been conducted to demonstrate seismic design adequacy as described in WCAP-7817 and Section 16.3.3.

*NOTE: The analysis of the Reactor Coolant System is discussed in Appendix 4B.

For the analysis of equipment to resist the vertical seismic component, 2/3 of the horizontal response spectrum curves were used to determine the acceleration appropriate to the vertical frequency.

Engineered Safeguards tanks, e.g., Boric Acid, Accumulator Spray Additive [retired] and Surge, were analyzed using method (3) above, for combined horizontal and vertical seismic excitation occurring simultaneously, and in conjunction with normal loads. Hydrodynamic analyses of these tanks were performed using the methods described in Chapter 6 of the U.S. Atomic Energy Commission –TID 7024.

Heat exchangers associated with the Engineered Safeguards Systems, e.g., Component Cooling and Residual Heat Removal, were analyzed using method (3) above, and the results show that stresses and deflections are within allowable limits.

Selected critical Engineered Safeguards valves are analyzed using method (3) above and the results indicate that their fundamental natural frequency is sufficiently separated from the building frequency that they will see little or no amplification of building motion. The results further indicated that the total stresses, considering all modes, is far below the allowable stress limits.

Damping values used for each item of equipment are in conformity with Table 16.1-1.

Non-linearities such as gaps, frictional forces, joint slippage, etc., were not considered explicitly in the model. It was felt that these non-linearities would tend to detune the system, hence act as if to increase the percentage of critical damping thus decreasing the response.

Appendages, such as motors attached to motor operated valves, were included in the mathematical models.

16.3.3 Seismic Testing of Instrumentation and Control Equipment

Mathematical models were not used for seismic design of instrumentation. Ability to withstand the seismic condition was determined by actual vibration type testing of typical instrumentation equipment under simulated seismic accelerations to demonstrate its ability to perform its functions. The seismic testing was reported in Westinghouse reports WCAP-7817, titled "Seismic Testing of Electrical and Control Equipment," by E.L. Vogeding, dated December 1971. The following is a summary:

> In a nuclear power plant, electrical and control equipment that initiates reactor trips, actuates safeguards systems and/or monitors radioactive releases from the plant must be capable of performing their functions during and after an earthquake that has occurred at the plant site. To demonstrate the ability of this equipment to perform under earthquake conditions, selected types of this essential equipment representative

of all protection and safeguard circuits and equipment were subjected to vibration tests which simulated the seismic conditions for the "low seismic" class of plants.* During the tests, equipment operation was monitored to prove proper performance of functions. The results show that there were no electrical malfunctions. Based on these results, it is concluded that the equipment will perform their design functions during as well as following a "low seismic" earthquake.

*NOTE: Those having Design Basis Earthquake horizontal acceleration less than or equal to 0.2g.

The low seismic test envelope is given in WCAP-7817 is appropriate for the locations of this protection and safeguards control and electrical equipment in Indian Point 3. The test curve developed for Indian Point 2 is conservative when applied to Indian Point 3 since the most adverse location, seismically, in Indian Point 2 is steel framed and relatively flexible, while that for Indian Point 3 is of reinforced concrete and therefore relatively rigid.

A typical path taken by a safeguards actuation signal is traced below to show that it is generated, transmitted and conditioned by the through equipment whose seismic adequacy has been demonstrated by test or analysis. A similar exercise may be carried out for reactor protection system signals.

A safeguards signal may be initiated by an instrument or transmitter which has the ability to withstand seismic forces as demonstrated in WCAP-7817, Sec. 4.8. This signal is carried in conduit and cable trays whose supports have been studied for resistance to seismic forces. The signal passes to the process control racks proven as described in WCAP-7817, Sec. 4.2. The signal is sent next to the safeguards actuation racks proven as described in WCAP-7817, Sec. 4.3. The actuation signal proceeds to the appropriate switchgear or active type controller.

The control board is not a Class I component. Typical switches and indicators for safeguards components were tested to determine their ability to withstand seismic forces without malfunction which would defeat automatic operation of the required component. Experience on previous control boards indicated that during shipment, "g" forces considerably greater than those required by the design basis earthquake are applied to the board and no failures of board mounted devices for engineered safeguard circuits had occurred. Past experience also indicated that the amplification due to the board structure can be measured during shipment. <u>W</u>NES instrumented the control boards during shipment to determine this amplification factor. Verification of no loss of function due to switches and indicators in the engineered safeguards circuits was completed by showing that the amplified "g" forces imposed on the devices were considerably less than the devices have shown to be able to withstand testing.

The safeguards circuits employ Westinghouse Series <u>W</u> motor control centers, and type DS circuit breakers and associated metal-enclosed or metal-clad switchgear. Review of these switchgear for proof of adequacy of the seismic resistant design determined that these motor control centers mounted in the metal enclosures, have been shock tested and proven to remain fully operable for shocks of at least 3g in any direction. Proof of resistance of the DS metal-clad switchgear to a seismic response spectrum established to "low seismic" plants have been demonstrated by vibration testing.

The switchgear supplies the power to operate the safeguards equipment completing the actuation train. The seismic design of this equipment is described in Section 16.3.2. The DC power supply may be considered as a branch to this main train of actuation. The source of DC power is the station batteries. The batteries and battery racks present a simple structural problem which was analyzed and found adequate for the forces imparted by the floor upon which they are located. Specially designed styrofoam spacers are installed in the intercell groups to provide additional seismic damping for the cell group. The conduit and cable trays carrying the DC power to the main station train received the same study for seismic support as described above.

16.3.4 Ability of Service Water Lines to Accept Seismic Ground Displacement

The service water lines consist of two 24" diameter carbon steel pipes. They run in a common trench which is backfilled. Assuming that the ends of a pipe are free to displace vertically but not rotate and that the maximum permissible stress is restricted to 30,000 psi, a parametric study concluded that the following maximum allowable relative displacements may occur during a seismic disturbance without overstressing the pipe:

Length (ft.)	1	10	25	50	75	100
Displacement (inches)	0.002	0.20	1.25	5.01	11.25	20.04

This parametric study consisted of investigating the maximum allowable relative displacements of the ends of the buried 24" service water piping for all lengths of straight pipe segments. The length of pipe was varied as a parameter to ascertain the magnitude of displacement required to stress the pipe to 30,000 psi. The corresponding displacements were then reviewed to determine whether it was feasible that the underlying bedrock could sustain such motion without catastrophic consequence. It was concluded that the displacements required to stress the service water pipe to 30,000 psi were in excess of that which could be reasonably imposed by the bedrock. Pipes entering the containment and other structures are effectively anchored at the points of penetration. When piping was routed from one building to another with restraint at or near entry points, differences in seismic responses between the two buildings were accommodated in the following manner: The floor response curves for the two entry points were overlaid and the envelope of both curves was used as input to the dynamic analysis of the entire piping run between the two buildings.

When a piping system was routed from one building to another, piping and supports arrangements were made in such a way that the relative movement between supports was accommodated by the flexibility of the pipe.

To evaluate the stresses imposed by relative motion, the initial stress analysis utilized the absolute sum of supports displacements found from seismic analysis of structures; using a static approach, the stress was calculated. The resulting stress was combined with other secondary stresses. The total stress was evaluated against B31.1 code allowable stress.

It was concluded that the service water lines can withstand, without being overstressed, relative bedrock displacements associated with the earthquakes defined for the Indian Point site. The Service Water System piping was reanalyzed in the seismic piping reanalysis effort described in Section 16.3.5.

16.3.5 Analysis of Seismic Class I Piping

During the design phase of Indian Point 3, all seismic Class I piping 6 inches in diameter or larger (other than the reactor coolant loop piping and main steam and main feedwater piping inside containment) together with the two inch diameter high head safety injection lines were initially statically designed by UE&C using spacing tables. Subsequently, these lines were dynamically analyzed for seismic response to confirm the static design; all other Class I piping (less than six inches in diameter) was statically designed and analyzed also using spacing tables. During 1979 and 1980 a seismic reanalysis of safety related piping systems was performed. The two design approaches and the reanalysis program outlined below. As indicated earlier, Westinghouse was responsible for seismically analyzing the reactor coolant loop, main steam, and main feedwater piping inside containment. Westinghouse was also responsible for other aspects associated with the design of the reactor coolant loop.

The design placement of seismic restraints was predicted on the principle of containing the seismic stresses without restricting the free thermal expansion of the piping system. The systems were designed to have sufficient flexibility to prevent the movements from causing failure of piping or anchors from overstress.

Each of the seismic supports was verified to agree with the as-built location.

Relative displacement between anchor points was considered in the seismic analysis of the main steam lines for Indian Point 3. Analysis indicated that the stresses at the highest stressed point were affected by less than 10% when relative anchor displacements were considered.

Dynamic Analysis of Seismic Class I Piping During Design Phase

Class I piping systems, 6 inches in diameter and larger plus the 2 inch diameter high head safety injection lines were modeled and dynamic flexibility analysis performed. A detailed description of the method of analysis is given below.

The analysis was performed using the proprietary computer code ADLPIPE. The code used as input, system geometry, inertia values, member sectional properties, elastic characteristics, support and restraint data characteristics, and the appropriate Indian Point 3 seismic floor response spectrum for 0.5% critical damping. Both horizontal and vertical components of the seismic response spectrum were applied simultaneously.

With this input data, the overall stiffness matrix of the three dimensional piping system was generated (including translational and rotational stiffness). The modal participation factors were computed and combined with the mode shapes and the appropriate seismic response spectra to give the structural response for each mode.

Each piping run was modeled as a three dimensional system which consisted of straight segments, curved segments, and restraints. Straight segments were distinguished from the curved segments during data output.

The computer code required that the piping be represented by a discrete mass model. Each mass included the contribution of both the steel encasement and conveyed fluid. Where valves or other concreted masses existed in the piping system, these were included in the model.

Restraints were included in the model at their proper location. The directionality of the restraints was also considered of the restraints was also considered.

Some averaging of the response spectra was performed to smooth out the erratic response of the earthquake's random behavior. At the high frequency end of the spectra, the acceleration levels of the smoothed spectra converged to the values of the unsmoothed spectra.

The computer code ADLPIPE utilized an algebraic summation option for intramodal response combination and the square root of the sum of the squares option for intermodal response combinations. The algebraic summation method of combination was later considered unacceptable as it may have predicted nonconservative results in the piping reanalysis. This determination of the inadequacy of the computer code ADLPIPE gave rise to the piping reanalysis described later in this section which was performed in accordance with the guidelines provided in IE Bulletin No. 79-07 ("Seismic Stress Analysis of Safety-Related Piping").

The re-analysis was limited to address the concerns cited in the subject IE Bulletin and as such the original analysis criteria (e.g., system modeling) were maintained.

The reactor coolant loop, main steam, and main feedwater piping inside containment were originally analyzed by Westinghouse in a manner acceptable within the requirements of I.E. Bulletin 79-07, and as such the concerns of the subject IE Bulletin were not acceptable to these piping lines.

UE&C Static Analysis of Seismic Class I Piping During Design Phase

Class I piping and supports, other than those dynamically analyzed (i.e., piping less than six inches in diameter except the two inch high head safety injection lines), were analyzed for equivalent static load. With a ground acceleration of 0.15g horizontal and 0.10g vertical, the spectral accelerations corresponding to two times and 1.33 times the maximum point on the 0.5% critical damping amplified response curve was used to calculate an equivalent static force imparted to the pipe and its support points for the horizontal and vertical directions, respectively. The sum of the resulting additional stress plus the normal stresses was limited to 1.2 times the B31.1 code allowable stress for piping. The stresses in the pipe supports and hangers were likewise limited to 1.33 times the allowable stress in accordance with the American Institute of Steel Construction (AISC).

Seismic Reanalysis for Safety Related Piping Systems

As discussed above, the original UE&C confirmatory dynamic analyses for the Indian Point 3 safety related piping systems greater than or equal to six inches diameter plus the high head safety injection piping utilized the computer code ADLPIPE. As discussed in IE Bulletin No. 79-07, the algebraic summation method of combination for intramodal responses was judged unacceptable as it may predict nonconservative results. The following piping system or portions thereof were affected by the subject IE Bulletin and reanalyzed by UE&C:

1) Condensate System
    - Auxiliary feedwater pump suction from condensate storage tank.

2) Auxiliary Feedwater System
    - Turbine driven auxiliary feedwater pump discharge.

3) Service Water System

4) Reactor Coolant System
   - Connections to reactor coolant systems from second check valve
   - Pressurizer surge line
   - Pressurizer relief lines

5) Safety Injection System including
   - Containment spray system
   - Accumulator discharge lines
   - Refueling water
   - Residual heat removal loop
   - Boron injection

6) Auxiliary Coolant System
   - Component cooling loop

7) Waste Disposal System
   - Recirculation fan cooling coil drains

Method of Reanalysis

The following method of reanalysis was submitted to and approved by the NRC staff for use in addressing the concerns cited in IE Bulletin No. 79-07. The seismic reanalysis was performed for the Operating Basis Earthquake (OBE) loading condition using the response spectrum analysis approach. The Amplified Response Spectra (ARS) associated with one horizontal (X) component and the vertical (Y) component of the seismic excitation were considered simultaneously. The analysis was repeated for the horizontal (Z) component and the vertical (Y) component. The reanalyses were performed with the UES&C –ADLPIPE- 2 computer code and a computer user option which use the square root of the sum of the squares for both intramodal and intermodal responses.

From these two cases, worst case values for the pipe seismic stresses, support loads and component nozzle loads were multiplied by a factor of 1.38 and then combined with other applicable loadings. The factor of 1.38 was found acceptable by NRC to reflect adequate conservatism in the calculations. Results from loading conditions other than seismic were not recalculated since they were not affected by IE Bulletin No. 79-07.

The factor 1.38, when used in combination with the computer user option, addressed the most conservative interpretation of the FSAR commitments regarding the intramodal response combination (i.e., this factor was utilized to account for the difference between absolute vs. SRSS summations). When a result calculated exceeded the applicable allowable limit, a reanalysis was performed using an equivalent analytical approach which included all three earthquake components and used the square root of the sum of the squares method for both the intermodal and intramodal responses without utilizing the factor 1.38. However, this latter approach was not employed as its use was not deemed necessary.

The results obtained from the OBE seismic reanalyses were multiplied by 1.5 to yield the Design Basis Earthquake (DBE) seismic condition values.

The safety-related lines 15 and 51 from the discharge of the containment spray pumps to the point where they penetrate the containment from Primary Auxiliary Building were further re-analyzed. The results of this re-analysis are presented in Reference 1.

As-Built Configuration of Safety Related Piping System

As part of the analytical effort required to conduct the seismic piping reanalysis program, an "As-Built" verification of those safety related piping systems subjected to the piping reanalysis was performed.

The field verification program for normally accessible areas consisted of line walks of 194 static span table analyzed lines and 117 dynamically analyzed lines. The program was conducted in accordance with approved plant procedures.

Valve Weight Corrections

A program was conducted to collect information relevant to Velan valve weight data. The program identified the specific swing check valves incorporated into the Indian Point 2 facility, the original weight data used in the piping analysis, the system and line in which the valve is installed, the weight variations, and the present deviation in weight compared with the specific line weight between supports.

The valves range in size from 3" to 12" and are installed in either the Auxiliary Coolant System, the Chemical and Volume Control System or the Safety Injection System. The actual weights of the 18 valves in question were included in the seismic piping reanalysis calculations.

Concrete Expansion Anchor Bolts in Class I Systems

A review of pipe support base plates using concrete expansion anchor bolts demonstrated the existence of QC documentation verifying the compliance with anchor bolt design requirements.

To further verify and complement this documentation, several elements of the program, in addition to the field verification effort, addressed the various criteria and concerns for concrete anchor bolts, as follows:

a) A verification survey incorporating two hundred and fifty (250) base plates with seven hundred (700) anchors was carried out for over twenty (20) normally accessible lines. The survey and sampling effort verified that the engineering, design and installation requirements were carefully carried out.

b) A UT sampling effort for the determination of anchor belt imbedment incorporated more than one hundred seventy five (175) normally accessible (outside Containment) supports. This sampling verified that the design requirements and installation procedures for concrete anchor imbedment were followed.

c) A preload upgrade/test retorque effort subjected to evaluation approximately two thousand (2000) normally accessible supports and more than seven hundred and fifty (750) normally inaccessible supports. This effort insured that supports not addressed in the repair or modification efforts are properly preloaded.

d) On site torque/preload testing was conducted for Hilti Kwik-Bolt Concrete Expansion Anchors. This testing and surveillance verified the appropriate torque values for a corresponding preloading of these anchors.

e) A field inspection effort was run to determine the extent to which anchor bolts on safety related lines were installed in concrete block walls. Seismic Class I system or safety related system supports which utilize concrete anchors in block walls were modified to eliminate the anchor bolt installations in these concrete block walls.

f) A hanger support repair effort resolved minor variances and problems identified during the line walk and inspection efforts. Approximately eight hundred (800) supports were repaired. Of these, some one hundred and fifty (150) are normally accessible.

g) To further insure proper preloading of concrete anchors, normally inaccessible supports involved in the modification effort had spring disc washers installed.

Results of Piping Reanalysis

All of the piping systems identified previously were reanalyzed for seismic loading using the method of analysis discussed above. The “As-Built” verification was performed in accordance with approved plant procedures. The results of the “As-Built” verification were incorporated in the reanalysis of the piping lines as well as the re-evaluation of valve weights, pipe supports, equipment nozzles and containment piping penetrations.

The results of the line reanalyses show that the total stresses, for both upset and emergency plant operating conditions, are within their respective applicable allowable limits.

Pipe supports, hangers, snubbers and pipe whip restraint components, including the base plate and anchor bolts, were re-evaluated for the new applied piping loads for both upset and emergency conditions. Those not found capable of performing their safety functions within their respective applicable allowable limits were modified as necessary.

Equipment nozzles and containment piping penetrations were reevaluated. The results confirmed that the new applied piping loads, both for the upset and emergency plant operating conditions, are within their respective applicable allowable limits.

16.3.6 Seismic Design of Spent Fuel Pool

Procedures outlined in Section 6.5 of TID-7024, “Nuclear Reactors and Earthquakes,” were used for the seismic design of the spent fuel pool. The effects of water in the pool is accounted for in this design approach.

The Fuel Storage Building outside the pool was evaluated for seismic capability to establish that unacceptable damage to the CCW piping would not occur. The methods and criteria were submitted for review in letters IPN-01-034 and IPN-02-040 and established that no unacceptable damage occurs.

16.3.7 Seismic Design of Intake Structure

Procedures outlined in Section 6.5 of TID-7024, "Nuclear Reactors and Earthquakes," were used for the seismic design of the Intake Structure walls. The effect of water sloshing on the walls is accounted for in this design approach. The controlling factor in the design of the Intake Structure was the hydrostatic load, with the worst combination being one chamber empty and the adjacent chamber being filled with water.

References

1) Letter from C.A. McNeill, Jr. (NYPA) to S.A. Varga (NRC) dated February 8, 1985 entitled "Revision of Results Previously Reported for IE Bulletin No. 79-07 (Seismic Stress Analysis of Safety-Related Piping) Line 51 of Problem 413."

16.4 DETAILS OF STRUCTURAL DESIGN

16.4.1 Design of Containment Interior Structures

The interior structure was designed as five separate main structural components. They are:

- 3' thick fill slab
- 3' thick crane wall
- 4' to 6' thick refueling canal
- 2' thick operating floor slab
- Primary Shield Wall

The method of design, stress analysis, critical stresses and locations were as follows:

3' Thick Fill Slab – The controlling loads on the 3' fill slab occur at the reactions from the primary equipment supports due to various postulated pipe breaks. The slab was designed as a series of radial beams running under the equipment supports and spanning between the reactor support wall and the crane wall. Stresses in reinforcing were limited to fy. Maximum stresses occur immediately below the primary equipment supports.

3' Thick Crane Wall – The crane wall is designed for a 7 psi differential pressure occurring immediately after a primary pipe break and prior to pressure equalization.

Although the stress level associated with this pressure differential were sufficiently low to establish that the concrete could resist the pressure loading, sufficient reinforcing was provided to resist all membrane forces without any contribution from the concrete. Stresses are limited to 0.9 fy. The membrane hoop stress was 13 ksi and the axial vertical rebar stress was 3.13 ksi.

A two dimensional Finite Element Analysis was performed to determine the area which would be affected by the Jet Force. The analysis indicated that in local areas (near the application of the force) some minor yielding of the crane wall rebar occurs. The yielding, which occurs only in the horizontal steel, is very local in nature. There is sufficient steel available in the vertical direction to accommodate any redistribution of load from the horizontal direction. In addition, redistribution will take place with the adjacent understressed facets. The load was assumed to act at the mid-height of the wall, thus causing maximum bending moment.

Further stability of the crane wall was demonstrated by determining the ultimate failure load by means of a yield line analysis. This analysis indicated that the structure has the capacity, through strain energy of structural response, to resist uniform Jet Force load of 2100 kips acting simultaneously with the 7 psi pressure differential without failure.

The containment internal concrete is essentially rigid (fundamental frequency ~17 cps), therefore, seismic loads were calculated using the Design Basis Earthquake maximum ground acceleration (0.15g).

The crane wall was considered as a cantilever beam and the base shear determined by the response spectrum approach. The base shear was distributed to the individual nodes by the formula:

$$F_x = W_x h_x V / \Sigma W_x h_x$$

Where: V = Base Shear

W_x = Weight of node under consideration

h_x = Distance from base to section under consideration.

The moment at the base was determined and the uplift calculated by considering a circular ring of thickness equal to the area of steel per inch. This maximum uplift which occurs at the point at the base of the structure stresses the rebar to 1.1 ksi. This load is insignificant when compared with the Jet Force load, therefore, consideration of simultaneous blowdown and earthquake loads do not affect the conclusions above.

The crane wall was also designed to resist steam and feedwater pipe break reactions of 340 kips and 200 kips where supports are connected to the wall. This extra steel provided for pipe break loads is available, in the form of steel buttresses, to resist pressure, jet force and seismic loads; however, it was not considered in the analysis.

<u>4’ to 6’ Thick Refueling Canal</u>

The refueling canal was designed for the 7 psi pressure differential. The wall resists the pressure by spanning vertically between the refueling floor and the operating floor. Stresses were limited to 0.9 fy.

A Finite Element Analysis was also performed to check the effects of the Jet Force load. Some local yielding was indicated; however, the cross section is sufficient to provide stability since the moment capacity is slightly greater than that of the crane wall. A yield line analysis was performed and provided the basis for the above.

The seismic load was determined by the same procedure used for the crane wall. The average load in kips/ft was distributed over the wall and the vertical span was conservatively assumed to carry the entire load. The resulting bending movement produced a stress of approximately 3 ksi in the rebar. This had an insignificant effect on the conclusions concerning the Jet Force loads when blowdown and earthquake were considered simultaneously.

2’ Thick Operating Floor Slab

Because of the many openings in the floor for equipment, the floor is designed as a series of beams. Principal loadings are (D.L. + 500 psf live load) and (7 psi pressure differential + D.L.). The first loading, (D.L. + 500 psf live load), was designed in accordance with Part IV-B of ACI 318. Stresses for the pressure differential case were limited to 0.9 fy.

The operating floor was investigated for Jet Force loads. There appears to be very little area of the operating floor which can be reached by the expanding jet of water from a break in the Reactor Coolant System. The jet is greatly dispersed in the distance between the primary coolant piping and the underside of the operating floor. The only area of the floor which can be struck by a jet spans between the areas of the floor heavily reinforced as beams. The span cross section consists of a T-beam with a 2’-0 thick floor acting as the flange and 7’-0 high biological shielding wall as the web. This section resists the jet force load within the 0.9 fy stress limit on the rebar.

Primary Shield Wall

The reactor pressure vessel is enclosed by a 6’-0” thick circular reinforced concrete shield wall which is designed to sustain the internal pressure and provide missile protection for the Containment and liner in the highly unlikely failure of the reactor vessel due to the longitudinal split. All stresses were maintained within 95 percent of specified minimum ultimate rebar tensile stress.

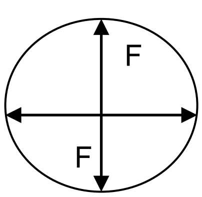
In the event of a circumferential reactor break, the ¼” base mat liner plate at the bottom of the Containment Reactor Cavity Pit, directly under the reactor vessel, is protected by 2’-0” of concrete with a 1” steel liner plate embedded in the top of the concrete. Below the containment base mat liner plate is 4 ½ feet of concrete poured on rock.

The cavity wall was designed to withstand the forces and internal pressurization associated with a longitudinal split without gross damage. The assumed accident condition was a longitudinal split of the cylindrical part of the reactor vessel (i.e., 24.4 feet long) having an average width of 1.9 foot. As a result of the assumed accident, the following two loading cases were considered in the analysis:

Load Condition I

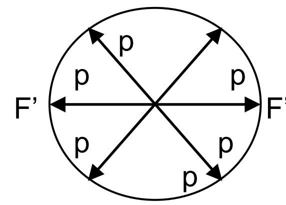
Load on cavity walls at the instant of vessel rupture -

F= 650 kips/ft equivalent static line load at the instant of vessel rupture applied as shown in sketch based on a dynamic load factor of 2 applied to the subcooled pressure of 2250 psi times the average width of the break



Load Condition 2

Load on cavity walls as shown in sketch -



F' = 188 kips/ft equivalent

P = 600 psi equivalent static pressure

The line load was based on saturated pressure of 1300 psi times the average width of the break and the pressure load was based on energy released and vent area available. The maximum stress level in the rebar under these loading conditions was limited to the 0.95 ultimate strength of the rebar. For Load Condition 1 and Load Condition 2, maximum rebar stresses assuming the concrete to be cracked were 63 ksi and 82.6 ksi, respectively. The rebar used is ASTM A 432 (Revised ASTM 615-63, grade 60) with specified yield of 60 ksi and ultimate tensile strength of 90 ksi.

The shield wall analysis showed rebar stresses of 64.6 ksi assuming all concrete was cracked assuming a pressure buildup of 600 psi inside the pit due to release of reactor contents. Since the integrity of the wall is not jeopardized the integrity of the vessel support which is supported on the wall will not be jeopardized. Deflection of the shield wall will not cause large stresses in the vessel support since a sliding surface is provided on the shoes, allowing the vessel support to slide.

Circumferential Cracking

The worst circumferential crack location from the standpoint of downward missiles is just below the RCS piping nozzles. As the following calculations show, the missile will not violate the containment structure and liner integrity.

As a consequence of this circumferential crack, the downward missile represented by the bottom vessel head has the following characteristics at the time of impact on the cavity floor:

1. Weight: 381,000 lbs
2. Cross sectional area of crater: 63 ft
3. Downward velocity: 213 ft/sec
4. Concrete crushing strength: 4,000 psi

The depth of penetration was calculated by using the Petri formula for penetration into an infinitely thick concrete slab, as reported in Nav. Docket P-51.

$$D=K\,(W/A)\log_{10}(1 + V^2/215{,}000)$$

Where: D=depth of penetration, ft.
K=penetration coefficient for 4,000 psi concrete
W=missile weight, lb.
A=missile area, ft^2
V=missile velocity, lb/sec.

The following parameters were used:

$K=2.8 \times 10^{-3}$
W=381,000 lb.
$A=63 \text{ ft}^2$
V=213 ft/sec.

The result is a depth of penetration of 1.4 feet.

Since the ¼" base mat liner is covered by 2' – 0" of concrete and topped with a 1" steel plate, it can be readily seen that the liner will not be reached, even neglecting the 1" steel plate in the penetration calculations.

Loading Due to Temperature Gradient

During normal operations, the only significant transient temperature gradients in the reactor containment interior structures occur during startup. The minimum containment internal temperature is limited to 50°F. The maximum operating containment internal temperature is 130°F. Forced movement of containment air is used to limit the concrete temperature surrounding the reactor vessel. This forced air movement of the containment air, as well as, normal convection and radiation is expected to limit the concrete temperature differentials in the range of 5 to 10°F. To demonstrate the large margin available in the concrete crane wall and the primary shield wall, a conservative assumption of a 300°F temperature gradient was evaluated. The evaluation included the gradient effect through the crane wall, the 6' thick portion of the primary shield wall below the reactor coolant pipe nozzle, the 5' thick portion of the primary shield wall where the nozzles penetrate the wall, and the 4'thick wall above the shield wall.

The maximum rebar stress was found to be 4500 psi and occurred in the vertical rebar in the crane wall. The maximum compressive concrete stress was found to be 226 psi and occurred in the hoop direction on the 5' portion of the primary shield wall. These stresses are approximately 20% of the allowable working stress values and have no significant effect on the design adequacy of the structures analyzed.

16.4.2. Class I Structures and Components Potentially Endangered by Failure of Class II or Class III Structures and Components

Seismic Class I structures and components which are so located that they could be potentially endangered by failure of seismic Class III structures are the Control Building, and the main steam piping, and feedwater piping, which could be endangered by seismic Class III Turbine Building. The Turbine Building was analyzed, using a multidegree of freedom modal dynamic analysis, for the Design Basis Earthquake, (0.15g maximum ground acceleration) and the building as constructed is capable of carrying the load without failure. A similar dynamic analysis was also performed to insure that no potential gross failure of the Indian Point 1 stack or superheater building could occur for the design basis earthquake and the design basis tornado for Indian Point 3.

The Containment Access Facility, which is situated atop the west end of the seismic Class I Primary Auxiliary Building, is partially a seismic Class III structure, however, the structural steel for this facility, as well as the structural interfaces with the PAB, were procured and installed to meet seismic Class I requirements. Although the Containment Access Facility is not a safety related structure, it has been designed to retain its structural integrity during a design basis seismic event. Also, the PAB and adjoining pipe penetration tunnel have been seismically evaluated to demonstrate their ability to resist seismic loads with the addition of the Containment Access Facility. Postulated failure of the Containment Access Facility due to design basis tornado loads would not adversely affect the operation of safe shutdown equipment located in the PAB or elsewhere. The seismic Class I PAB Ventilation System would not be adversely affected by any postulated failures of connected exhaust ductwork in the Containment Access Facility.

A Systems Interaction (SI) Study was conducted to determine the potential endangerment of seismic Class I components by failure of seismic Class II and Class III components (Refer to References 2,3, and 4). The Authority has resolved all potentially unacceptable interactions identified by this study.

The Fuel Storage Building overhead crane is a seismic Class III crane. The crane bridge, trolley, and building crane supports were dynamically analyzed at various positions of the trolley, both loaded and unloaded using response spectra modal analysis for the design basis earthquake. The analysis showed that neither the crane bridge or trolley would derail or overturn during the DBE and thus would not endanger the spent fuel pit or other seismic Class I functions.

The seismic Class III Jib crane, hoist and associated control equipment, located in the NE sector of the containment building operating floor, were analyzed for seismic loading and found to maintain structural integrity during a DBE. Therefore no seismic Class I structures and components would be affected by its failure.

The manipulator crane in the Containment Building, a seismic Class III crane, is restrained from overturning and will not endanger seismic Class I structures.

16.4.3 Tornado Protection

As discussed in Section 16.2.2, all equipment which must be protected from tornados and tornado generated missiles is contained within tornado proof structures or protected by redundancy.

The tornado proof structures, which were constructed of reinforced concrete, were designed to prevent missile penetration and spalling (by selection of moderate degree of damage allowable stress indices for structural design in accordance with Reference (1) of concrete from the walls, roof slab or dome impacted by the missile). Therefore, secondary missiles are not created which could damage or make inoperable seismic Class I systems which must be protected from tornados.

Further discussion of criteria for determining missile protection requirements is presented in Section 16.2.3.

Tornado Load Capacity of Structures

Containment Structure

The containment can withstand all loads put on it by the design tornado specified in Section 16.2. Details are given in the Containment Design Report (Appendix 5A).

Primary Auxiliary Building (PAB) and Control Building

These structures are capable of resisting any wind loads generated by the design tornado specified.

Fuel Storage Building

Based on information furnished by the siding manufacturer, the siding panel on this structure will blow out at 170 psf (i.e., 1.18 psi) negative pressure. Panels fail at 60 psf external pressure which is equivalent to 162 mph external wind load. The girts will fail at 90 psf (i.e., 0.62 psi) negative pressure.

The 60 psf mentioned above controls the external loading condition.

Block walls are located below Elevation 95’0” on the south and east sides of the Fuel Storage Building. The Primary Auxiliary Building protects the south wall from tornado loads. The block wall located on the west side above elevation 95’0” does not present an interaction concern. The east wall would fail under tornado wind but would not affect safety related equipment. A missile through the east wall could damage small bore piping associated with the CCW system. FSAR Section 9.3 discusses operator action to maintain CCW function.

Intake Structure

The concrete sub-structure and the structural steel super structure of the service water enclosure are capable of resisting tornado wind loads.

Tornado Missile Resistance of Structures

Containment Structure, Primary Auxiliary Building, Control Building, Diesel Generator Building and Auxiliary Feedwater System Building

The Containment, Primary Auxiliary Building, Diesel Generator Building and Auxiliary Feedwater System Building will not be penetrated by the design tornado missiles. These missiles are:

Horizontal missiles

a) 4” x 12” x 12’ wood plank at 300 mph
b) 4000 lb auto at 50 mph less than 25’ above the ground

Vertical missiles

a) 4” x 12” x 12’ wood plant at 90 mph
b) 4000 lb auto at 17 mph less than 25’ above the ground

Fuel Storage Building

The 3” thick siding panels on this structure are not capable of resisting any tornado generated missiles.

Intake Structure

The intake structure is capable of resisting any missile loads generated by a tornado. This is true only for the structure and does not necessarily include equipment, although the circulating water pump and service pump motors are dispersed so that a single missile could not cause all of them to fail.

Spent Fuel Pool Dewatering by Tornado

Dewatering of the Spent Fuel Pool is discussed in proprietary report WCAP-7313-L “Tornado Induced Water Removal from Spent Fuel Storage Pool,” submitted in May 1969. Two geometric configurations were considered: One in which the tornado funnel passes at such a distance from the pool center as to produce over it the largest pressure gradients and wind velocities. The other in which the tornado funnel centers over the pool. The results of this study indicate that for the non-aligned tornado, the pool water level will drop 6 feet at the most, leaving over 17 feet of water over the top of fuel assemblies. A centered tornado of such strength that its tangential wind velocity at the pool rim equals 300 mph, will leave at least 10 feet of water over the top of the spent fuel assemblies, if such a tornado remained stationary over the pool center from some 100 seconds. Even if the tornado residence time were that long which, according to field observations, is an unusually long period, the ability of the pool to cool the spent fuel assemblies and to offer radiation protection will not be impaired.

16.4.4 Cathodic Corrosion Protection

During the initial Licensing process, a complete survey and tests to determine the need for cathodic protection on Indian Point 3 was made by the A.V. Smith Engineering Company of Narberth, Pennsylvania. Electrical resistivity measurements and visual inspection of the area away from the river, where the Turbine Generator Building, Reactor Building, Primary Auxiliary Building, and associated facilities are located, indicated that the environment is mostly rock with areas of dry sandy clay. The electrical resistivity of the soil ranged from 3,500 to 30,000 ohm-centimeters with the majority of the readings being above 10,000 ohm-centimeters. On this basis, it was determined that cathodic protection was not required on underground facilities in areas of the containment building liner away from the river, although protective coating on pipes was recommended to eliminate any random localized corrosion attack.

An analysis of Hudson River water data, obtained from the Consolidated Edison plant chemist, showed the electrical resistivity of the water to vary over an extremely wide range due to salt intrusion from the ocean. The range of resistivity has been from 59 to 10,000 ohm-centimeters with a large number of readings in the 300 ohm-centimeter area. This value was considered to be extremely corrosive and the following structures in the area near the river were placed under cathodic protection:

1) De-icing lines
2) Bearing piles
3) Sheet piling (earth and water side) and wing wall anchorage system
4) Metallic structures inside intake structure (traveling screens, bar racks, circulating water pump suction, service water pump suction).

The cathodic protection system was not functional and was removed for the Intake Structure Enclosure modification to facilitate installation of the building's structural steel. While the original intent was to design and install new cathodic protection systems, new systems were not installed and the potential adverse effects caused by corrosion are addressed as follows:

1) The de-icing system was retired and no longer requires cathodic protection.
2) The sheet piling system, consisting of bearing piles, walers, and sheet piles is subject to inspections and degraded conditions are addressed as required.
3) The cathodic protection system for the Traveling Water Screens and Bar Racks were found not be effective and the installed cathodic protective systems were retired. The original Traveling Water Screens which were carbon steel were upgraded to stainless steel frames, baskets, and chains. The splash housings are of fiberglass construction. The Bar Racks were last replaced in 2009, are of galvanized steel construction and are epoxy coated with a coal tar epoxy expected to provide corrosion protection over a longer than 20-year period. The guides for the Screens and Racks are carbon steel channels mounted in a concrete trough. The rate of corrosion is slow and the Screens and Racks are on a regular Preventive Maintenance (PM) cycle that checks for degraded conditions.
4) The circulating water lines and service water lines are protected by concrete encasement in areas of high corrosion and do not require cathodic protection.
5) The Service Water and Circulating Water pumps suction are not cathodically protected. Rather, the Service Water Pumps suction is inspected and refurbished as part of the Service Water Pumps PM activities. The Circulating Water Pumps are inspected and refurbished according to Preventive Maintenance Program requirements.
6) The Service Water Pumps structural steel grating in the intake bay was replaced with materials of a greater resistance to corrosion.

In 2009, a guided wave assessment of buried piping at Indian Point Unit 3 was performed by Structural Integrity Associates, Inc. of Centennial Colorado. The assessment identified potential corrosion on the Unit 3 Condensate Storage Tank Supply and Return lines in the vicinity of the AFW Pump Building. As a result this piping was placed under cathodic protection.

16.4.5 Thermal Stresses in Walls of Spent Fuel Pit

The thermal stresses in the walls of the spent fuel pool resulting from temperature gradients were evaluated by the procedure outlined in ACI 349-80 "Code Requirements for Nuclear Safety-Related Concrete Structures" and presented in a plant specific report applicable to the maximum density spent fuel pool racks, in which every cell is presumed to be fully loaded with a fuel assembly and a control rod assembly (Reference 6). For the portion of the pool below grade a linear gradient with 200°F water temperature and a 50°F outside temperature was assumed for the analysis. A gradient of 200°F water temperature and 0°F outside temperature was used for the structure above grade. For accident conditions (loss of pool cooling), a water temperature of 212°F was used. Under these conditions, maximum linear and pool anchor strain were calculated, as well as strain-induced loads, maximum average shear and maximum bending moment. In all cases, the resultant values for the pool mat, the interior wall, the exterior wall and the canal mat were within allowable limits.

Provisions were made to limit cracking and prevent leakage through the concrete by means of porous intercept channels, even though the pit is lined with a leak proof stainless steel liner. All welds were vacuum-box tested during construction to assure a leak tight membrane, and all shop welds were dye penetrant inspected on the water side in the shop. In addition, there is a leak collection system behind all field welds. The effect of a thermal gradient would be to compress the linear, thereby preventing any leakage. This leakage collection system is brought to common drain line provided with a manually operated isolating valve. The valve serves as a backup means of limiting leakage from the pit should cracks develop in any of the pit liner joints.

16.4.6 Rainfall Accumulation

Buildings or structures housing safety related items were evaluated for effects from rainfall accumulation. Roof drains were sized to handle 5-5-½ inches rainfall per hour. Roof design loadings were 40 lbs/ft^2 max. The following rainfall accumulations were evaluated:

Hours	Rainfall Accumulations*	Rainfall Accumulation**
1	9.1 inches	4.1 inches
2	12.7 inches	2.7 inches
3	16.0 inches	1.0 inches
6	23.9 inches	---
12	29.0 inches	---

NOTE: *Rainfall accumulation only
**Rainfall accumulation with 5"/hr roof drainage

For these accumulations** the largest roof loading realized was 21.4 lbs/ft^2 which occurred during the first hour.

References

1) TM5-855-1, Department of the Army Technical Manual, "Fundamentals of Protective Design (Non-Nuclear)," 1965

2) Letter from J.P. Bayne to S.A. Varga dated November 30, 1983, entitled "Indian Point 3 (IP-3) Systems Interaction (SI) Study" – Attachment A.

3) Letter from J.P. Bayne to S.A. Varga dated March 6, 1984, entitled "Indian Point 3 (IP-3) Systems Interaction (SI) Study."

4) Letter from J.P. Bayne to S.A. Varga dated October 31, 1984, entitled "IP-3 Systems Interaction (SI) Study."

5) Nuclear Safety Evaluation No. 86-03-138 IS, Rev. 5: Intake Structure Enclosure Building – Phases I, II, & III

6) "Structural Evaluation of the Spent Fuel Storage Building for Storage of U.S. Tool and Die Maximum Density Racks Containing 1345 Fuel Assemblies," dated March 25, 1988, Ebasco Services, Inc.