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A.1 STATION BLACKOUT (SBO)

A.1.1 STATION BLACKOUT OVERVIEW

Station Blackout is defined as the complete loss of alternating current electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with a turbine trip and the unavailability of the onsite emergency AC power system). A Station Blackout does not involve the loss of available AC power to buses fed by station batteries through inverters. The event is considered to be terminated upon the restoration of power to the essential switchgear buses from any source, including the alternate AC source which has been qualified as an acceptable coping mechanism. A concurrent single failure or design basis accident need not be assumed during a station blackout event (Reference 2 and Reference 18).

The requirements for Station Blackout are established in 10 CFR 50.63 (Reference 1), which was formally issued in 1988. Guidance for compliance with the regulatory requirements is presented in NUMARC 87-00, Revision 0 (Reference 2) and Regulatory Guide 1.155 (Reference 3). The NRC has not endorsed Revision 1 to NUMARC 87-00, but has accepted specific supplements to NUMARC 87-00 Rev. 0, as described in Appendix K of NUMARC 87-00 Rev. 1 (Reference 2).

The station blackout regulation requires determination of the coping duration category based on criteria provided in Reference 2 and Reference 3. The “required coping duration” is defined as the time between the onset of station blackout and the restoration of off-site AC power to safe shutdown buses. “Coping duration category” is a quantification of the relative risk of a particular facility to the occurrence of a station blackout (loss of all onsite and offsite AC power).

The determination of the required coping duration category is based on several factors, such as the plant design and the probability of severe weather conditions in the area. Once the required coping duration category has been established, the design approach to coping with the station blackout event is demonstrated. This design approach may choose to take credit for either an available alternate AC power source or opt for an AC power-independent design. The plant systems must have the necessary capacity and capability to ensure the core is cooled and containment integrity is maintained for the required station blackout coping duration.

The coping duration categories are 2, 4, 8, or 16 hours, as determined from Table 3-8 of Reference 2. The intent of the regulation is for all domestic nuclear plant sites to fall in either the 2-hour or the 4-hour coping duration category, and then select either the “Alternate AC” or “AC-Independent” coping methodology for their specific plant. The NRC bases for coping duration category objectives are described in Section 2.3.2 of Reference 2. The major contributor to overall station blackout risk is the likelihood of losing off-site power and the duration of power unavailability. The stated objective of the NRC is to reduce the core damage frequency due to station blackout to approximately 10^{-5} per year for the average site. This objective is accomplished by requiring either a four hour coping capability or use of an Alternate AC (AAC) source.

PBNP's original response to the SBO rule concluded that the required coping duration category was 8 hours and used the Gas Turbine Generator (GTG) G-05 as the sole Alternate AC (ACC) source to power the safe shutdown loads of both blacked out units. Because the GTG cannot be

shown to be available within 10 minutes of the onset of station blackout, a one hour coping assessment was performed as required by Section 7.1.2 of Reference 2. (Reference 4 and Reference 8)

The coping duration category was subsequently revised to 4 hours based on a change in the extremely severe weather (ESW) group classification as discussed in Section A.1.2. With the addition of the G-03 and G-04 EDGs, the SBO minimum redundancy requirements of emergency AC (EAC) power supplies for normal safe shutdown of both units is exceeded and utilization of an EDG as an AAC source is allowed. By definition, a unit with an available EAC power supply is not blacked out. However, any EDG credited as an AAC source must be capable of handling the safe shutdown loads in both the blacked out and non-blacked out units (Reference 2). The PBNP EDGs meet this requirement. Therefore, the present coping methodology utilizes the Gas Turbine Generator (GTG) G-05 or an Emergency Diesel Generator (EDG) from the non-blacked out unit as Alternate AC (ACC) sources. An EDG will start, accelerate to rated frequency and voltage, and can be connected to an EAC bus in either unit within ten minutes of SBO initiation. The GTG will be manually started, accelerate to rated frequency and voltage, and be available to power the safe shutdown loads within one hour of SBO initiation (Reference 6, Reference 15). Since PBNP continues to use the GTG as one of the ACC sources, and it cannot be shown to be available within 10 minutes, the one hour coping assessment has been retained and is described in Section A.1.3.

A.1.2 STATION BLACKOUT COPING DURATION CATEGORY DETERMINATION

The potential for long duration loss of off-site power (LOOP) events can have a significant impact on station blackout risk and required coping duration. Long duration LOOP events are typically associated with grid failures due to severe weather conditions or unique transmission system features. Shorter duration LOOP events tend to be associated with plant specific switchyard features. Per Reference 1, the required coping duration shall be based on the following factors:

1. The redundancy of the emergency standby power system
2. The reliability of each of the emergency power sources
3. The expected frequency of a loss of offsite power
4. The probable time required to restore offsite power

Offsite Power Design Characteristic Group

The regulatory guidance (Reference 2, Tables 3-5a and 3-6a; Reference 3, Table 4) has established three basic groups (P1, P2, and P3) for categorizing the design of the preferred offsite power system. A category of P3 is assigned to those plants with a frequency of grid-related loss of offsite power events greater than once in 20 site-years, which is limited to St. Lucie, Turkey Point and Indian Point (Reference 2).

Since PBNP is not included among the three noted plant sites, further evaluation of several factors is necessary to establish the Offsite Power Design Characteristic Group. The applicable group is defined based on combinations of the following three factors:

- extremely severe weather
- severe weather
- offsite power system independence

Extremely Severe Weather (ESW Group)

The estimated frequency of loss of offsite power due to extremely severe weather is determined by the annual expectation of storms at the site with wind velocities equal to or greater than 125 mph. These events are normally associated with the occurrence of hurricanes where high windspeeds may cause widespread transmission system unavailability for extended periods. Since electrical distribution systems are not designed for such conditions, it is assumed the occurrence of such windspeeds will directly result in the loss of offsite power.

The estimated frequency may be determined based on either site-specific data or on data from local weather stations. Table 3-2 of Reference 2 summarizes site-specific National Oceanic Atmospheric Administration (NOAA) data for the estimated frequency of occurrence of extremely severe weather. As published in this table, PBNP has an event frequency of 0.0036, and therefore was categorized in ESW Group 4 (Reference 4). Subsequent review determined the NOAA data for extremely severe weather was overly conservative for PBNP, and that an ESW event frequency supporting an ESW Group 2 category was justified (Reference 5). This departure from the NUMARC 87-00 criteria was reviewed and approved by the NRC (Reference 6).

Severe Weather (SW Group)

Table 6 of Reference 3 and Part 3.2.1.C of Reference 2 define the severe weather factor based on the frequency of a loss of offsite power due to severe weather. The severe weather considered includes snow, tornadoes, high winds, and storms with salt spray. These are related by the equation:

$$\text{frequency} = 1.3 \times 10^{-4} \times h_1 + b \times h_2 + 0.012 \times h_3 + c \times h_4$$

The variables in this equation are defined for PBNP in Reference 2, Section 3.2.1.C:

h_1 = annual expectation of snowfall for site, in inches; this is 42.0 inches for PBNP

h_2 =annual expectation of tornadoes with windspeeds greater than or equal to 113 miles per hour, in events per square mile; this is 0.000035 for PBNP

h_3 =annual expectation of storms with wind velocities between 75 and 124 mph; this is 0.1 for PBNP

h_4 =annual expectation of storms with significant salt spray for the site; this is 0.0 for PBNP.

b =72.3; the PBNP offsite power system design connects four 345 kV transmission circuits to the plant switchyard via a single right-of-way.

c =0; the PBNP site is not considered vulnerable to the effects of salt spray.

These factors, when combined in the severe weather frequency equation, yield an estimated frequency of loss of offsite power due to severe weather of 0.0092. This places PBNP in SW Group 2.

Independence of the Offsite Power System (I Group)

Reference 3, Table 5, defines the offsite power system independence factor, and Reference 2 Section 3.2.1.D simplifies the determination:

If: (a) all offsite power sources are connected to the safe shutdown buses through one switchyard or through multiple electrically connected switchyards, and (b1) the normal power source is from the main generator and there are no automatic and one or more manual transfers of all safe shutdown buses to the preferred or alternate offsite power sources, or (b2) there is one automatic and no manual transfers of the safe shutdown buses to one preferred or one alternate offsite power source, the site falls in the I-3 group. Otherwise, the site is assigned to the I-1/2 group.

The I-1/2 group is characterized by features associated with greater independence and redundancy of sources, and a more desirable transfer scheme. I-3 sites have simpler, less desirable offsite power systems and switchyard capabilities.

Condition a: The PBNP offsite power system consists of four (4) 345 kV transmission circuits, connected via a single right-of-way, to a single switchyard which serves both PBNP units. On this basis, the answer to Condition A is considered to be “YES” for the PBNP site.

Condition b1 and b2: The PBNP auxiliary power distribution system provides offsite power connections to the safety-buses of each unit via the high voltage station auxiliary transformers and the low voltage station auxiliary transformers. This normal supply of power to the safety-related buses is derived from offsite power sources. Upon loss of the preferred offsite power source to the safety-related buses of one unit, the buses will be powered from the preferred power source of the other unit. On this basis, the answers to both Condition b(1) and b(2) are considered to be “NO”, and the PBNP site is classified in the I-1/2 Group.

Offsite AC Power Design Characteristic Group Determination

The combination of the ESW, SW and I factors results in an Offsite Power Design Characteristic Group of P1 for PBNP, based on Reference 3, Table 4.

Emergency AC Power Configuration Group

Regulatory guidance defines four Emergency AC (EAC) Power Configuration groups (A, B, C, and D) based on the availability and redundancy of the emergency power supplies. Reference 2 Section 3.2.2 clarifies the EAC groups, basing it on the number of EAC power supplies required to handle the safe shutdown loads and on the number of additional EAC power supplies available. The PBNP EAC power configuration group is C, based on the following:

PBNP is a two-unit site with four shared Emergency Diesel Generators (EDGs) and one gas turbine generator (GTG). The two Train A EDGs are identical components with a 2000 hour rated output of 2850 kW at 4.16 kV. The two Train B EDGs are identical components with a 2000 hour rated output of 2848 kW at 4.16 kV. All four EDGs are available to support the safe shutdown equipment of either PBNP unit, and a single EDG can supply adequate power to the safe shutdown loads in both units. The GTG has a rating of 23.10 MVA at an output voltage of 13.8 kV, and can supply adequate power to the safe shutdown loads in both units.

Therefore, because only one EDG is necessary to operate safe shutdown equipment for both units following a loss of offsite power, the EAC power configuration group at PBNP is "C", as a 1 out of 2 EDG, dedicated, or 1 out of 3 EDGs, shared configuration per Table 3-7 of Reference 2. Additionally, the PBNP SBO licensing basis permits the use of either the GTG or an EDG as the AAC source.

Target Standby Diesel Generator Reliability

The reliability of the EAC power sources has a key role in the quantification of risk due to SBO. A target value for reliability was therefore made a factor in establishing the required SBO coping duration. The EDG target reliability was selected to be 0.975 based on the original EAC configuration group determination of "D" (i.e., prior to the installation of G-03 and G-04) and the reliability data that existed at the time of the initial SBO evaluation. (Reference 4) These reliability computations utilized the NRC-recommended methodology of EPRI Report NSAC-108 (Reference 7)

Because PBNP offsite power design group is P1, and EAC configuration is C, the target EDG reliability value may be 0.950 or 0.975 per Table 2 of Reference 3. PBNP has retained the reliability target value of 0.975 (Reference 15). PBNP has implemented an EDG reliability program which is based on the methodology of EPRI Report NSAC-108 and conforms to the guidance of RG 1.155, Position 1.2 (Reference 8 and Reference 15).

Coping Duration Category Determination Summary

The previous determinations are summarized below:

Offsite AC Power Design Characteristic Group	=P1
Emergency AC Power Configuration Group	=C
Target Standby Diesel Generator Reliability	=0.975

In accordance with Table 3-8 of Reference 2 and Table 2 of Reference 3, the group determinations listed above result in a coping duration category for PBNP of four hours.

A.1.3 STATION BLACKOUT COPING ANALYSES

Condensate Inventory for Decay Heat Removal

This analysis ensures that PBNP has sufficient condensate inventory to support the decay heat removal function for the SBO event duration. Section 7.2.1 of Reference 2 provides a simplified calculation approach to determine the required condensate volume. This analysis is satisfied by demonstrating that Technical Specification volume requirements envelop the volume estimated by the Reference 2 methodology.

At a core power of 1800 MWt, 14,000 gallons of condensate water are required for the one hour SBO event duration based on the methodology of Reference 2. However in order to maintain the same margin set by the NRC in Reference 8 for subsequent switchover to the long-term AFW water supply, the minimum CST usable volume is set at 15,410 gallons. This volume is bounded by the Technical Specification CST volume requirements which includes additional margin to

account for suction piping losses, vortex prevention, pump NPSH requirements, unusable tank volume, and instrument uncertainty. Therefore, PBNP has sufficient condensate inventory for the SBO event duration, including time to transfer to the long-term source after the one-hour period, Reference 16, Reference 17, Reference 22.

Safety-Related Battery Capacity

This coping analysis assures the plant has adequate battery capacity to support required safe shutdown loads for the SBO event duration. Reference 2, Section 7.2.2 suggests the minimum battery capacity be four hours for plants using the AC-independent coping position and one hour for plants utilizing the Alternate AC power source position. The analysis should consider the factors in IEEE Standard 485, including lowest expected electrolyte temperatures and appropriate load duty cycles. Load shedding, if required, shall commence no sooner than 30 minutes after the SBO event initiation.

The evaluation of PBNP battery capacity is based on design analyses which verify sufficient capacity to support all safety related DC loads on all four DC channels (D05, D06, D105 and D106) for a minimum of one hour. No load shedding is required to meet this requirement (Reference 9). This determination was reviewed and accepted by the NRC (Reference 11).

Compressed Air

This analysis demonstrates that air operated valves required for decay heat removal can be operated as required for the defined coping duration. Section 7.2.3 of Reference 2, requests identification of all required valves and the availability of backup air supplies or manual operation capability. At PBNP, no safety-related air operated valves are required to cope with a SBO event for one hour (Reference 4, Reference 8).

Effects of Loss of Ventilation

This analysis ensures the room temperatures in areas containing equipment required to mitigate a SBO event do not increase to values impacting operability following loss of forced ventilation. Reasonable assurance of equipment operability is based on calculated maximum room temperatures less than or equal to 120 °F (Reference 12). The PBNP analyses evaluated the loss of ventilation effects in key plant areas, as summarized below.

Containment Building

The containment building analysis was based on generic analyses performed for Westinghouse Owners Group Emergency Response Guidelines. Based on engineering review and judgment, the containment building temperature would not reach levels which would impair equipment operability (Reference 12). The NRC found this analysis to be acceptable (Reference 8).

Instrument Inverter Rooms

The instrument inverter DY-03 and DY-04 rooms were evaluated as they have the smallest volume of the PBNP inverter rooms and thus would experience the fastest heatup. The results of room temperature analyses showed a maximum room temperature ≤ 120°F, which was considered acceptable for equipment operability (Reference 12). The NRC found this analysis to be acceptable (Reference 8).

Cable Spreading Room

The cable spreading room was evaluated utilizing the room temperature analysis method described above. This analysis showed a maximum room temperature ≤ 120°F, which was considered acceptable for equipment operability (Reference 12). The NRC found this analysis to be acceptable (Reference 8).

Auxiliary Feedwater Pump Room

The Auxiliary Feedwater Pump room was evaluated utilizing the room temperature analysis method described above. This analysis showed a maximum room temperature ≤ 120°F, which was considered acceptable for equipment operability (Reference 12). The NRC found this analysis to be acceptable (Reference 8).

Control Room

The Control Room was evaluated utilizing the room temperature analysis method described above. This analysis showed a maximum room temperature of 128 °F with all tiles for the suspended ceiling in place and ≤ 120°F with some of the ceiling tiles removed. The latter analysis was considered acceptable for equipment operability, and the control room ceiling was permanently modified to remove some of the suspended ceiling tiles (Reference 12). However, the NRC did not find this analysis to be acceptable, and requested additional information and justification of the analysis results (Reference 8). Additional analyses were performed which confirmed the original results (Reference 13). The NRC found the commitment to perform the confirmatory analyses and permanently modify the control room ceiling to be acceptable (Reference 11).

Computer Room

The Computer Room was evaluated utilizing the room temperature analysis method described above. This analysis showed a maximum room temperature of 174 °F with all tiles for the suspended ceiling in place and ≤ 120°F with some of the ceiling tiles removed. The latter analysis was considered acceptable for equipment operability, and the computer room ceiling was permanently modified to remove some of the suspended ceiling tiles (Reference 12). However, the NRC did not find this analysis to be acceptable, and requested additional information and justification of the analysis results (Reference 8). Additional analyses were performed at a higher initial room temperature which confirmed the viability of the ceiling tile removal modification (Reference 13). The NRC found the commitment to perform the confirmatory analyses and permanently modify the computer room ceiling to be acceptable (Reference 11).

Containment Isolation

One of the key safety functions identified in Reference 1 is containment integrity. In addressing this item, the containment isolation valves were reviewed to verify that valves which must be capable of being closed or that must be operated under station blackout conditions can be positioned, with indication, independent of the blacked-out unit's safety-related power supplies. Based on assessment guidelines provided in Section 7.2.5 of Reference 2, the following isolation valve types may be excluded from consideration as valves of concern:

1. Valves normally locked closed during operation
2. Valves that fail closed on a loss of power
3. Check valves
4. Valves in non-radioactive, closed loop systems not expected to be breached in an SBO (except lines which communicate directly with the containment atmosphere)
5. Valves of less than 3-inch nominal diameter.

Based on these exclusion criteria, there are five penetrations for each PBNP unit for which indication and control would be lost during a SBO event. Four of the five penetrations are associated with motor-operated valves in the component cooling water system. Manual isolation capability for these four valves provides adequate containment isolation. The remaining penetration is associated with the chemical and volume control system, and includes an automatic air-operated valve inside of containment. This valve would close on the loss of power, and the penetration can also be manually isolated (Reference 4). The NRC found the containment isolation analysis to be acceptable (Reference 8).

Reactor Coolant Inventory

The maximum reactor coolant pump (RCP) seal leakage rate of 25 gpm per RCP, assuming a complete loss of cooling, has been evaluated, as recommended by the guidance of Reference 2. The total RCP leakage plus the miscellaneous Technical Specification leakage of 10 gpm results in a total primary system leakage rate of 60 gpm. The conclusion is that the RCS inventory would be reduced by RCP seal leakage, but adequate initial RCS inventory is available for the one hour necessary to bring the AAC power source on line (Reference 4 and Reference 8).

The Extended Power Uprate (EPU) revisited the subject of reactor coolant inventory during an SBO event, and found that because EPU would not affect the leakage rates or initial pressurizer level, the reactor coolant inventory during an SBO would continue to be acceptable (Reference 23 and Reference 24).

A.1.4 ALTERNATE AC SOURCE

The original PBNP emergency power system configuration consisted of two EDGs shared between the two PBNP units. This design resulted in an EAC Power Configuration Group of "D", based on one EDG required of two EDGs shared between the two PBNP units (Reference 4). This configuration prompted the use of gas turbine generator (GTG) G-05 as the AAC source, with commitments to maintain the target GTG reliability at or above 0.95, and to be able to start and load the GTG within 1 hour of a postulated SBO event (Reference 8).

The new EDG installation in 1991-1996 resulted in an EAC Power Configuration Group of "C", based on either one of two EDGs in a dedicated unit configuration or one of three EDGs in a shared unit configuration (after discounting the EDG considered to be the AAC source). This design change also permitted adoption of a SBO coping strategy still utilizing GTG G-05 as the AAC source, and also utilizing one EDG in the non-blacked out unit as an AAC source in addition to or in lieu of G-05 (Reference 5, and Reference 6). The GTG or an EDG are normally available to support the safe shutdown equipment of either PBNP unit, and either power source can supply power sufficient to achieve safe shutdown in both units. The GTG and each EDG are thus

considered fully capable AAC sources per Appendix B of Reference 2; i.e., with sufficient capacity and capability to operate necessary systems for the required 4-hour coping duration.

The EDGs can be connected to the EAC power buses of either unit in a SBO within 10 minutes (Reference 6 , Reference 14). Therefore per Section 3.2.5 of Reference 3, no coping analysis was required for using the EDGs as AAC sources (Reference 6).

A.1.5 PROCEDURES AND TRAINING

The PBNP plant currently has Emergency Operating Procedures (EOPs) addressing the loss of all AC power, including:

ECA 0.0, Loss of All AC Power
ECA 0.1, Loss of All AC Power Recovery Without SI Required
ECA 0.2, Loss of All AC Power Recovery With SI Required

These procedures were developed from the Westinghouse Owners Group Emergency Response Guidelines, which have been reviewed and approved by the NRC. The ECA 0.0 procedure directs operators to restore power to the safety-related buses by EDG restart, offsite power reconnection, gas turbine generator G-05 start or opposite unit safety-related bus crosstie. The ECA 0.1 and 0.2 procedures provide guidance for recovery from the station blackout condition once AC power has been restored. The SBO recovery guidelines were implemented prior to promulgation of the Station Blackout Rule, and thus additional operator training was not necessary (Reference 4). The NRC considered the procedures to be acceptable, and appropriate training is implemented for any EAC power source configuration changes (Reference 8 and Reference 9).

As part of the G-05 reliability program, testing in a manner similar with that as would be required during a SBO event is performed on at least a quarterly basis. The testing includes starting and running G-05 with its support systems powered by the auxiliary power diesel generator G-501. The testing synchronizes G-05 to the grid and includes operation for at least 1 hour at a load which envelopes the SBO requirements (Reference 11, Reference 13, Reference 19, Reference 20, and Reference 21).

A.1.6 QUALITY ASSURANCE PROGRAM

The PBNP SBO position utilizes equipment to cope with the postulated SBO event which was not previously covered by plant quality assurance (QA) programs. The plant equipment originally classified as safety-related and required for SBO coping is covered by a 10 CFR 50 Appendix B quality assurance program which meets or exceeds the guidelines of Appendix A of Reference 3. Non-safety-related components credited for coping in the PBNP SBO position have been assigned an Augmented Quality (AQ) classification which incorporates the QA program attributes (Reference 9). The NRC found the commitment to include non-safety-related components credited for SBO coping in the AQ program to be acceptable (Reference 11).

A.1.7 REFERENCES

1. 10 CFR 50.63, “Loss of All Alternating Current Power.”

2. NUMARC Document 87-00, “Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors,” Revision 0; November 1987, Revision 1; August, 1991.

3. NRC Regulatory Guide 1.155, “Station Blackout,” Revision 1; August 1988.

4. VPNPD-89-216, NRC-89-043, WEPCo Letter to NRC, “Response to 10 CFR 50.63, TAC. Nos. 68586 and 68587, Loss of All Alternating Current Power, Point Beach Nuclear Plant, Units 1 and 2,” April 17, 1989.

5. NPL 94-0353, WEPCo Letter to NRC, “Supplement to 10 CFR 50.63, TAC. Nos. 68586 and 68587, Loss of All Alternating Current Power, Point Beach Nuclear Plant, Units 1 and 2,”September 22, 1994.

6. NRC Letter to WEPCo, TAC Nos. M90613 and M90614 “Station Blackout Modification, Point Beach Nuclear Plant, Units 1 and 2 ” October 16, 1995.

7. EPRI Report NSAC-108, “The Reliability of Emergency Diesel Generators at U.S. Nuclear Power Plants,” September 1986.

8. NRC Letter to WEPCo, TAC Nos. 68586 and 68587, “Safety Evaluation of the Point Beach Response to the Station Blackout Rule,” October 3, 1990.

9. VPNPD-90-459, NRC-90-110, WEPCo Letter to NRC, “10 CFR 50.63, TAC. Nos. 68586 and 68587, Loss of All Alternating Current Power, Point Beach Nuclear Plant, Units 1 and 2,” November 8, 1990.

10. Not used

11. NRC Letter to WEPCo, TAC 68586, “Supplemental Safety Evaluation of Response to Station Blackout Rule,” March 22, 1991.

12. VPNPD-90-148, NRC-90-030, WEPCo Letter to NRC, TAC. Nos. 68586 and 68587, “Supplement to 10 CFR 50.63, Loss of All Alternating Current Power, Point Beach Nuclear Plant, Units 1 and 2,” March 30, 1990.

13. VPNPD-93-182, NRC-93-112, WEPCo Letter to NRC, TAC. Nos. 68586 and 68587, “Supplement to 10 CFR 50.63, Loss of All Alternating Current Power, Point Beach Nuclear Plant, Units 1 and 2,” October 12, 1993.

14. NPM 2003-0795, Revision 1 to Memo NPM 2003-0646, Time-Critical Operator Actions, November 7, 2003.

15. NPL 95-0284, WEPCo Letter to NRC, TAC. Nos. M90613 and M90614, “Supplement to 10 CFR 50.63, Loss of All Alternating Current Power, Point Beach Nuclear Plant, Units 1 and 2,” June 14, 1995.

16. Calculation CN-SEE-111-08-3, Revision 0, “Point Beach 1 & 2 Minimum Condensate Storage Tank Volume for EPU Program,” November 12, 2008.

17. NRC Safety Evaluation "Point Beach Nuclear Plant (PBNP), Units 1 and 2 - Issuance Of License Amendments Regarding Extended Power Uprate (TAC Nos. ME1044 and ME1045)," dated May 3, 2011.

18. Code of Federal Regulations, Title 10, Section 50.2, "Definitions."

19. VPNPD-89-511, NRC-89-114, WEPCo Letter to NRC, "Supplement to 10 CFR 50.63, TAC. Nos. 68586 and 68587, Loss of All Alternating Current Power, Point Beach Nuclear Plant, Units 1 and 2," September 26, 1989.

20. VPNPD-93-058, "Supplement to 10 CFR 50.63, TAC Nos. 68586 and 68587 Loss Of All Alternating Current Power Point Beach Nuclear Plants, Units 1 and 2," March 5, 1993.

21. NRC Letter to WEPCo, Use of Point Beach Nuclear Plant Gas Turbine Generator (G-05) as an Alternate Ac Power Source; 10 CFR 50.63, TAC Nos. M84420 and M84421, "Loss Of All Alternating Current Power," August 12, 1993.

22. NRC Safety Evaluation, "Point Beach Nuclear Plant (PBNP), Units 1 and 2 -Issuance Of License Amendments Re: Auxiliary Feedwater System Modification (TAC Nos. Me1081 and Me1082)," dated March 25, 2011.

23. FPL Energy Letter to NRC, NRC 2009-0030, "License Amendment Request 261 Extended Power Uprate," dated April 7, 2009.

24. NRC Safety Evaluation, "Issuance of License Amendments Regarding Extended Power Uprate," dated May 3, 2011.

A.2 HIGH ENERGY PIPE FAILURE OUTSIDE CONTAINMENT

A.2.1 INTRODUCTION AND EVALUATION CRITERIA

The High Energy Line Break (HELB) program was initiated with the Atomic energy Commission (AEC) letter to Wisconsin Electric Power (WEP) dated December 19,1972 (Mr. Giambusso to Mr. Quale) (Reference 1 and Reference 2). In that letter, the commission stated that the consequences of postulated pipe failures outside of the containment structure, including the rupture of a main steam or feedwater line, need to be adequately documented and analyzed. The NRC's original acceptance of the PBNP response is contained in Reference 3. As part of the Extended Power Uprate (EPU) project, PNBP submitted several updated aspects of the HELB analysis. The changes were evaluated and accepted by the NRC in Reference 5.

The following specific criteria are applicable to the point Beach HELB analyses:

a. The definition of high energy piping systems are systems which have a combined pressure and temperature rating which exceeds a service temperature of 200°F or greater <u>and</u> a design pressure above 275 psig.

b. Normally depressurized lines which are pressurized only for infrequent periodic testing under controlled diagnostic conditions are not considered in the HELB analyses.

c. Coincident or compounded accidents, including natural events, are not considered in the HELB analysis unless the compound accident can be directly caused by the HELB.

d. Pipe motion and jet forces resulting from breaks shall not impair the ability to safely shut down the reactor or impair the ability to cool the reactor core. Safe shutdown is defined as Hot shutdown (MODE 4) for the HELB analyses.

e. Critical leakage cracks are located and oriented to cause worst effects.

A.2.2 DESCRIPTION OF HIGH ENERGY SYSTEMS

A.2.2.1 High energy piping systems are defined as those which have a service temperature of 200°F or greater <u>and</u> a design pressure above 275 psig (Reference 2). The Plant operational conditions under which the definition applies include normal reactor operation and anticipated operational occurrences. Piping systems 1" nominal pipe size and smaller are excluded from HELB review. The piping systems which fall under the definition listed above as determined in Reference 6 are:

a.Main steam

b.Main feedwater

c.Chemical and volume control system (CVCS) letdown

d.Steam generator blowdown

e.Condensate

f.Heater drain tank pump discharge

g.Turbine extraction steam

h. Feedwater heater and MSR vents and drains

A.2.3 DESCRIPTION OF BREAK AND CRACK LOCATIONS

Introduction

The HELB program requires the identification and application of specific criteria which define where a large break or a leakage crack must be postulated in each high energy system outside containment. This section describes those criteria and how they are applied at PBNP.

Locations of Required Postulated Large Breaks in High Energy Lines

a. For high energy systems that do not have a documented dynamic pipe stress analysis, NRC Generic Letter 87-11 (Reference 4) requires that a large break must be postulated at the piping welds to each fitting, valve, or welded attachment. Thus, for these high energy lines without documented dynamic pipe stress analyses, a large break is postulated to occur in every room or compartment through which the line is routed. The specific large break is postulated to occur at the most restrictive or most bounding location within each room or compartment. Calculation PBNP-994-21-06 (Reference 13) specifies the locations where each of these large breaks is postulated to occur. The high energy systems that fall into this category for large break locations are:

   Chemical and volume control system (CVCS) letdown
   Steam generator blowdown
   Condensate
   Heater drain tank pump discharge
   Turbine extraction steam
   Feedwater heater and MSR vents and drains

b. For high energy systems that have a documented dynamic pipe stress analysis, specific methodologies are used for Point Beach, as described in Section A.2.5. The methods described are summarized in Reference 13 and the tables and data contained therein document therein document the specific locations of postulated large breaks for these lines. The high energy lines which fall into this category are:

   Main Steam
   Main Feedwater

Size and Orientation of Large Breaks

The methods to determine the size and orientation of each large break are described in Section A.2.5. The results of those analyses are presented in Reference 13.

Required Postulated Leakage Cracks in High Energy Lines

Where high energy pipes are routed in the vicinity of structures and systems necessary for safe shutdown of the plant, a leakage crack in the piping system is postulated at the most adverse location possible with respect to the affected equipment. The location of the postulated cracks is provided in Reference 13. The methods of determining the size and orientation of the leakage cracks are described in Section A.2.5.

A.2.4 DESCRIPTION OF NEEDED EQUIPMENT

The December 19, 1972 AEC letter and its enclosure (Reference 1), as clarified in January 1973 (Reference 2), required the identification of those systems and components required to detect and mitigate HELB events and to maintain the plant in a safe shutdown condition and to maintain the ability to cool the reactor core. The equipment needed in the postulated HELB events is listed in Reference 19.

A.2.5 METHODOLOGIES FOR LOCATIONS, SIZE AND ORIENTATION OF BREAKS

Methodology for Postulating Large Breaks Locations in Lines with Dynamic Stress Analyses

As discussed in Section A.2.3, PBNP applies specific requirements to determine the location of required postulated large breaks in high energy lines for which a dynamic stress analysis has been performed. The methodologies described in Generic Letter 87-11 (Reference 4) were applied in Reference 7, Reference 8, Reference 9, Reference 10, Reference 11, Reference 12 to identify applicable stresses.

Point Beach reconciled the piping to the ASME 1977 B&PB Code, Section III, Subsection NC, including the Winter 1978 Addenda (Reference 21). Code Equations 9 and 10 from Reference 21 including stress intensification factors are identical to those utilized in the pipe stress analysis.

The piping systems identified in Section A.2.3 above were designed to the USAS AB31.1-1967 Power Piping Code (Reference 20). For purposes of HELB evaluation, the piping was evaluated against the criteria identified for ASME Section III Code Class 2 and 3. There are no ASME Section III Code Class 1 piping systems outside containment at Point Beach. Thus, a large break is postulated to occur at any location that meets any one of the following criteria:

a. Any terminal end

b. Any intermediate location where the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with an operating basis earthquake (OBE) seismic event and operational plant conditions exceed 0.8 $(1.2S_h + S_A)$. S_h is the allowable stress limit at the operating temperature, and S_A is the allowable stress range for thermal expansion as found in Reference 20.

c. Any intermediate location where the thermal expansion stress term exceeds 0.8 S_A.

Methodology for Determining Size and Orientation of Postulated Large Breaks

Longitudinal breaks in main piping runs or branch runs were examined for pipes of 4" nominal pipe diameter and larger. A longitudinal break is parallel to the pipe axis and oriented at any point around the pipe circumference. The break area is equal to the effective cross-sectional flow area upstream of the break location with the length of the break equivalent to twice the inside pipe diameter. Dynamic forces resulting from such breaks are assumed to cause lateral pipe movements in the direction normal to the pipe axis.

Circumferential breaks were considered in piping runs and branch runs for pipes of less than a 4" nominal diameter. A circumferential break is perpendicular to the pipe axis, and the break area is equivalent to the cross-sectional flow area of the ruptured pipe. Dynamic forces resulting from such breaks are assumed to separate the piping axially and cause whipping in any direction normal to the pipe axis.

Methodology for Determining Size and Orientation of Postulated Leakage Cracks

A leakage crack is defined as a single open crack with a length equal to 1/2 the pipe diameter and a width equal to 1/2 the pipe wall thickness. The orientation of the leakage crack can be in any direction along the pipe at the most adverse location in terms of needed equipment or structures.

A.2.6 METHODOLOGY FOR CALCULATING MASS AND ENERGY RELEASE

This section describes the methodology for calculating mass and energy releases for compartment temperature and pressures. The mass and energy releases used for jet impingement and pipe whip are discussed in Section A.2.8 and Section A.2.9 respectively.

Calculation of mass and energy (M&E) releases from a break or crack are accomplished by running multiple cases to assure that all scenarios are bounded. This assures that the most conservative results are attained, with a trade-off between maximizing break enthalpy and maximizing mass release. Thus, the following two methods are used for temperature limiting cases and pressure limiting cases.

a. For temperature limiting cases throughout most of the Primary Auxiliary Building, Facades and Turbine Hall, the analyses are contained in Reference 28 using the computer code LOFTRAN. The most conservative plant condition is full power at Extended Power Uprate (EPU) conditions. The calculation includes the effects of core thermal power, energy from main feedwater and auxiliary feedwater additions, engineered safeguards systems, sensible heat stored in the RCS and steam generator metal mass and tubing, and reverse steam generator heat transfer. The evaluation methodology maximizes the superheat of the releases.

b. For pressure limiting cases throughout most of the Primary Auxiliary Building, the Facades, and the Turbine Hall, the analyses are found in Reference 14. The most conservative plant condition is hot zero power. These analyses use the Extended Henry-Fauske critical flow model for sub-cooled liquid conditions and the Moody critical flow model for saturated steam conditions.

c. For the CCW Heat Exchanger Room, all pressure and temperature limiting cases are documented in Reference 29 using the RELAP 5 computer code model. The methodology calculates mass and energy using the Moody critical flow model per ANSI/ANS 56.10-1982. The postulated HELB consists of a circumferential guillotine pipe rupture with the two ends completely offset. The flows from each end are comprised of a transient flow followed by a steady state flow which is determined independently. For the transient flow, fictional losses are conservatively neglected. For steady state flow, line losses downsteam of the source to the break are considered.

d. For temperature limiting cases for the Steam Generator Blowdown line in the lower portions of the Facade and for the CVCS Letdown line, the analyses are found in Calculation 2012-0012 (Reference 15).

A.2.7 METHODOLOGY FOR COMPARTMENT PRESSURE AND TEMPERATURE

Calculations of pressure and temperature responses of compartments within the PAB, Facades, and Turbine Hall were performed using the GOTHIC computer code. The buildings, the internal compartments and the net free volumes within them are modeled in a GOTHIC base case, which is based on detailed plant walkdowns and measurements. The heat absorbing slabs considered in the model are also based on plant walkdowns and measurements. All barriers between compartments (wall, floors, ceilings, doors, etc.) are defined in the GOTHIC model. The model for these calculations is found in Reference 15.

The pressure and temperature response calculations are performed by considering the locations of postulated breaks and leakage crack within the plant compartments. The bounding cases are run using the mass and energy releases associated with the limiting break or leakage cracks for the various compartments. Initial conditions for the evaluations include the following:

a. The assumed initial temperature of the volume in each compartment is contained in Reference 17.

b. The assumed initial pressure of each volume is 14.375 psia.

c. The assumed initial relative humidity of each volume is 37 percent. This is based on the minimum value of humidity ratio and maximum outside temperature of 95°F.

The calculations of compartment pressures and temperatures are documented in Reference 17.

A.2.8 METHODOLOGY FOR JET IMPINGEMENT

Jet Thrust Forces

Jet thrust forces that accompany a pipe break are both steady state and time dependent in nature. The steady state forces are due to the blowdown of fluid from some system stagnation pressure, P_o. The time dependent forces are due to the propagation of pressure disturbances in the fluid immediately following pipe break. Both types of forces must be considered in calculating pipe break.

Steady State Thrust Calculation

The generalized steady state thrust equation as developed by Shapiro (Reference 22) is:

$$F = \frac{\dot{m}}{g_c} V_e + (P_e \angle P_a) A_e \qquad (1)$$

Where: $\dot{m}$ = fluid mass flow rate (lb_m/sec)
V_e = fluid exit velocity (ft/sec)
g_c = gravitational constant ($lb_m ft/lb_f\ sec^2$)
P_e = fluid exit pressure (PSF)
P_a = ambient pressure (PSF)
A_e = exit area (ft^2)

A convenient non-dimensional thrust can be defined by dividing through by P_o and A_e obtaining:

$$F/P_oA_e = \frac{\dot{m}V_e}{P_oA_eg_c} \div \frac{(P_c \angle P_a)}{P_o} \qquad (2)$$

One dimensional continuity, m = ρV and the definition, G = m/A, can be used with (2) to obtain the following alternate expressions:

$$F/P_oA_e = \frac{GV_e}{P_og_c} + \frac{(P_e \angle P_a)}{P_o} \qquad (3)$$

and: $$F/P_oA_e = \frac{G^2}{\rho_e g_c P_o} + \frac{(P_c \angle P_a)}{P_o}$$

Where: ρ_e is exit mass density (lb_m/ft^3).

Four blowdown situations are considered for rupture of steam and water lines. They are:

a. Blowdown of steam from superheated conditions

b. Blowdown of a steam-water mixture

c. Blowdown of cold water

d. Blowdown of water with flashing from subcooled water conditions.

Superheated Steam

Superheated steam is usually treated as an ideal gas with the gas constant, R, equal to 85.75 ft. lb_f/ lb_m °R and the ratio of specific heats γ, equal to 1.3 (Reference 23). If the flow is further considered isentropic, the thrust parameter becomes (Reference 23):

$$F/P_oA_e = 1.26 \angle \frac{P_a}{P_o} \tag{4}$$

Friction effects can be considered by assuming the flow process follows the Fanno line as described by Shapiro. (Reference 22) The Fanno analysis predicts thrust parameter will be a function of the pipe friction parameter fL/D, as shown in Figure A.2-8. For the case of fL/D = 0 the Fanno analysis reduces to the inviscid flow case (equation (4)). Flow restrictions will tend to decrease flow rates and can be included using Figure A.2-9 - Figure A.2-11.

Steam Water Mixtures

An equilibrium, two-phase flow model has been developed by Moody (Reference 24) which can be used to predict blowdown of mixtures of steam and water. Moody provides plots of G_{max} as a function of stagnation conditions for friction parameters between 0 and 100. He also provides a plot which can be used to determine exit conditions. For frictionless flows (fL/D = 0), Moody's model gives approximately the same results as (4):

$$F/P_oA_e = \frac{P_a}{P_o} \tag{5}$$

Fauske (Reference 25) has proposed a second model which includes non-equilibrium effects. He compares his model with equilibrium models and concludes that for low steam qualities (x < 2%) and short pipes equilibrium models may not be conservative estimate of thrust for short pipes and low steam qualities. Again, Figure A.2-9 - Figure A.2-11 can be used to include flow restriction effects.

Cold Water Flow

Blowdown of cold water can be treated as flow of an incompressible fluid ($\rho = \rho_o$). For inviscid flow the exit pressure becomes the ambient pressure, P_a and the exit velocity becomes:

$$V_e = \sqrt{\frac{2(P_o \angle P_a)g_c}{\rho_o}} \tag{6}$$

The thrust parameter is then (Reference 23):

$$F/P_oA_e = 2 \angle \frac{2P_a}{P_o} \tag{7}$$

Friction effects can be included using the approximate expression:

$$F/P_oA_e = \frac{2}{\left(f\frac{L}{D} + 1\right)} \tag{8}$$

Subcooled Water Flow

Subcooled water blowdown is characterized by flashing of the fluid near the pipe exit. This flashing tends to cause lower thrust levels than those predicted using non-flashing incompressible flow theory. The non-equilibrium model developed by Fauske (Reference 25) is applicable in the subcooled region and can be used to predict subcooled water blowdown. The cold water thrust equation (7) can be used to obtain a quick conservative estimate of subcooled thrust.

Unsteady Flow Thrust

Definition of the unsteady thrust requires an examination of the interaction between the propagation of pressure disturbances and the initiation of blowdown. An approximation can be made by assuming the thrust to be defined by:

$$F/P_oA_e = 1.0 \tag{9}$$

over the period, $0 \leq t \leq t_1$

Where: $t_1 = 2L / C$ (10)

and: L = length of pipe from break to pressure vessel
= length of pipe from break to flow restriction
C = sonic velocity
= 1600 ft/sec from steam
= 4000 ft/sec for water

For $t > t_1$ thrust is equal to the steady state value.

Fluid Jet Impingement Forces

In the event of a pipe break, the fluid flowing through the pipe emerges out as a jet impinging at nearby structures or equipment. Various blowdown situations considered here are described in Section 1. On emerging from the break point, the jet undergoes free rapid expansion to the ambient pressure at relatively short distance - a few diameters of break area. For this asymptotic distance, momentum, and shear interactions with jet environment can reasonably be neglected. As such, applying forward momentum conservation, the total jet force, F_j, is constant throughout its travel; and, therefore as assumed by Moody (Reference 23).

$$F_j = F \tag{11}$$

where, F is the total thrust force defined in Section 1. Methods of calculating F are also given there. For the purpose of this report, it is further assumed that F_j remains constant for all distances beyond the asymptotic area. This assumption is conservative. Therefore, the jet pressure at any location along the axis of the jet is given by:

$$P_j(x) = F/A_j(x) \tag{12}$$

where A_j (x) is the expanded jet area at location x along the jet axis. See Figure A.2-8 for system geometry.

Moody (Reference 23) as developed a simple analytical model for estimating the asymptotic jet area for steam, saturated water, and steam/water blowdown situation. Evaluations of LOFT (Reference 26) experimental results tend to indicate that for subcooled water and steam blowdown situations, the jet area expands uniformly at half angle of about 15°, where as steam/ water blowdown expands much more rapidly because of large scale water flashing. Results of Moody's analytical analysis agree, at least qualitatively, with LOFT results. In addition, Moody's analytical analysis predicts results of other experiments, as discussed in Reference 23.

In this report, an empirical approach has been adopted combining Moody's analytical model with the uniform half angle approach, as shown in Figure A.2-8. The half angle is conservatively assumed to be $\phi = 10°$.

According to this empirical mode, the distance of jet travel is divided into 3 regions. Region 1 extends to the asymptotic area, at which point jet expansion area is calculated according to Moody's method; in Region 2, jet area remains constant; then in Region 3, the jet expands at half angle $\phi = 10°$. For subcooled water blowdown, this model assumes half angle approach, ϕ 10°, uniformly in all the three regions, since Moody's model is not well applicable for this case.

To follow Moody, extent of region 1 is taken as;

$$x_i = 5D_e \tag{13}$$

and the jet area at location x_1 is given by the equation:

$$\begin{aligned} A_j(x_i) &= \pi \cdot R_{j1}{}^2 \\ &= (A_e G)^2 v_i / g_c F_j \end{aligned} \qquad (14)$$

Where:

D_e = Equivalent diameter of pipe break area
A_e = Pipe break area
R_{j1} = Radius of the expanded jet at location x1. R_j1 is constant in Region 2.
F_j = F, thrust force (Eq. 11)
v_1 = Specific volume. v_1 is calculated as described in Reference 23

Other terms in Eq. 14 are described in 1. Region 2 extends to the location x_2 given by:

$$A_j(x_i) = A_j(x), x = x_2 \qquad (15)$$

where A_j (x) is the jet area in Region 3 and is calculated by any one of the following equations;

1. Guillotine break:

$$A_j(x) = A_e\left(1 + \frac{2x}{D_e}\tan\phi\right)^2 \qquad (16)$$

where $\phi = 10°$ is the half angle of jet expansion

2. Longitudinal (slot) break;

$$A_j(x) = A_e\left(1 + \frac{2x}{l}\tan\phi\right)\left(1 + \frac{2x}{w}\tan\phi\right) \qquad (17)$$

where 1 and w are slot dimensions.

3. Circumferential crack:

$$A_j(x) = A_e\left(1 + \frac{2x}{w}\tan\phi\right)\left(1 + \frac{2x}{l}[1 + 2\tan\phi]\right) \qquad (18)$$

In Region 1, additional conservative assumption is made that the jet area increases uniformly from A_e at x = 0, to A_j (x_1) at $x = x_1$, or:

$$A_j(x) = A_e\left[1 + \frac{X}{X_i}\left[\frac{R_{j1}}{R_e} \angle 1\right]\right]2, for 0 \leq x \leq x_1 \quad (19)$$

where $R_e = D_e/2 = A_e/\pi$, and R_{j1} is given by Eq. 14.

Impingement Loads on Targets

Once the jet area A_j is calculated by the method described above, the jet pressure is readily calculated according to Eq. 12, i.e.:

$$P_j = F_j/A_j \quad (20)$$

and the jet impingement load on the target is given by;

$$F_T = P_j \cdot A_w \quad (21)$$

where A_{te} is the effective target area. Calculation of A_{te} for various geometries is outlined below:

1. Flat Surface

If the target with physical area A_t cancels all the fluid momentum in the jet, then;

$$A_{te} = A_t$$

For the case where target is oriented at angle θ with respect to the jet axis and there is no flow reversal:

$$A_{te} = A_t \sin\theta$$

2. Pipe Surface

Let: D_P = Diameter of pipe, and D_j = Diameter of jet impinging on pipe = $\sqrt{\frac{4A_j}{\pi}}$

Then, for $D_P < D_j$:

$$A_{te} = CA_j$$

where C is pipe curvature factor and $C = 2/\pi$ for $D_p < D_j$;

$$A_{te} = C \cdot A_t$$

where $A_t = D_P \cdot D_j$ (conservative approximation)

3. Deflecting Surface

Effective target surface area, A_{te}, for the targets which deflect the jet rather than totally cancel the fluid momentum in the jet, is calculated as described by Moody (Reference 23)

A.2.9 METHODOLOGY FOR PIPE WHIP

The maximum operating pressure (P) used in the calculations is 900 psig and the maximum temperature, 534°F. The maximum jet thrust (F) for a steam line is equal to 1.26 PA, where A equals the flow area of the pipe. Here the maximum jet thrust is equal to 8.4 kips. Assuming no strain hardening, the hinge moment for the 3 inch Schedule 40 piping is:

$$M_h = 1.28 T_y S_m$$

Where:

T_y = Minimum yield strength at 534°F for A106-GR. B

= 27.5 KSI

S_m = Section modulus

= 1.724 in^3

M_h = 60,700 in-obs

A ruptured pipe will then whip if the cantilevered length of pipe (L_h) normal to the direction of thrust is greater than:

$$L_h = \frac{M_h}{f} = 7.25 \text{ inches}$$

Jet Impingement Forces

Equipment	Jet Force	Local Pressure
Boric Acid Tank	8.4 Kips	14.5 psig
Component Cooling Heat Exchanger	8.4 Kips	14.5 psig
Component Cooling Surge Tank	-----------	2.5 psig

A.2.10 REFERENCES

1. ”General Information Required for Consideration of the Effects of a Piping System Break Outside of containment”, AEC, December 19, 1972.

2. Errata Sheet for “General Information Required for Consideration of the Effects of a Piping System Break Outside of Containment”, AEC, January 24, 1973.

3. “Safety Evaluation Report-High Energy Line Failure Outside of Containment, Point Beach Nuclear Plant Units 1 and 2”, NRC, Dated May 7, 1976.

4. NRC Generic Letter 87-11, “Relaxation in Arbitrary Pipe Rupture Requirements”, June 19,1987.

5. “Point Beach Nuclear Plant Units 1 and 2 - Issuance of License Amendments Regarding Extended Power Uprate (TAC NOS. ME 1044 AND ME 1045), NRC, May 3,2011.

6. Calculation PNBP-994-21-02, Rev.0, High Energy System Selection.

7. Calculation PBNP-994-21-05-P01, Rev.1, Unit 1 Main Steam.

8. Calculation PBNP-994-21-05-P02, Rev.1, Unit 2 Main Steam.

9. Calculation PBNP-994-21-05-P03, Rev.2, Unit 1 Main Feedwater.

10. Calculation PBNP-994-21-05-P04, Rev.2, Unit 2 Main Feedwater.

11. Calculation PBNP-994-21-05-P05, Rev. 1, Unit 1 Steam Supply to AFWPT.

12. Calculation PBNP-994-21-05-P06, Rev.1, Unit 2 Steam Supply to AFWPT.

13. Calculation PBNP-994-21-06, Rev.2, Break and Crack Size/Location Selection.

14. Calculation PBNP-994-21-07, Rev.0, Mass and Energy Releases.

15. Calculation 2012-0012, “HELB GOTHIC Base Model.

16. Not Used.

17. Calculation 2012-0013, “HELB GOTHIC Analysis.”

18. Not Used.

19. Calculation PBNP-994-21-15, Rev.2, Required Equipment List.

20. USAS B31.1-1967, “Power Piping Code”.

21. ASME B&PV Code, Section III, 1977 with Winter 1978 Addenda.

22. Shapiro, A.H., “The Dynamics and Thermodynamics of Compressible Fluid Flow, “Vol.1, Ronald Press, New Your, New York, 1953.

23. Moody, F.J., “Prediction of Blowdown forces and Jet Thrusts,” ASME Publication, June, 1969.

24. Moody, F.J., “Maximum Two Phase Vessel Blowdown form Pipes,” G.E. Report APED-4827, 65 APE 11, April 20, 1965.

25. Fauske, Hans K. and Henry, Robert E., “The Two Phase Critical Flow of One-component Mixtures in Nozzles, Orifices, and Short Tubes,” ASME Transactions, Journal of Heat Transfer, May 1971.

26. Loft Test, IDO-17242, Idaho Nuclear Corporation, 1968.

27. Not Used.

28. Westinghouse Letter WEP-08-174, “Transmittal of Steamline Break Mass/Energy Releases Outside Containment for the Point Beach Extended Power Uprate Program,” December 29, 2008.

29. Point Beach Calculation 2008-12355, Rev.0 “Mass and Energy Calculation for High Energy Line Break in 3” Steam Line within CCW HX Room”.

Table A.2-1 JET IMPINGEMENT FORCES ON CABLE SPREADING ROOM WALLS

1. Slot break jet impingement forces on cable-spreading room door and wall from:

   a. 24 in. main steam line nearest wall

(1) Aimed directly at wall -	pressure	83.5 psi
	area	41 $ft.^2$
(2) Aimed towards the door -	pressure	11.51 psi
	area	304.0 $ft.^2$

   b. 24 in. main steam line farthest from wall

(1) Aim to impinge on wall or door -	pressure	20.74 psi
	area	164.5 $ft.^2$

2. Slot break jet impingement forces on cable-spreading room door and wall:

   a. 24 in. turbine bypass line nearest wall

(1) Aimed directly at wall -	pressure	83.5 psi
	area	41 $ft.^2$
(2) Aimed towards the door -	pressure	49.3 psi
	area	68.8 $ft.^2$

   b. 24 in. turbine bypass line farthest from wall:

	pressure	42.41 psi
	area	79.6 $ft.^2$

Table A.2-2 JET IMPINGEMENT FORCES (VARIOUS LOCATIONS)

1. Critical crack jet impingement forces on barrier and non-vital switchgear from:

   24 in. main steam line nearest switchgear

(1)	On barrier -	pressure	7.25 psi
		area	6.5 $ft.^2$
(2)	On switchgear 1A01 -	pressure	1.23 psi
		area	36.6 $ft.^2$
(3)	On switchgear 1A02 -	pressure	1.13 psi
		area	53.0 $ft.^2$

2. Critical crack jet impingement forces on the corner control room window from:

24 in. main steam line -	pressure	.258 psi

3. Critical crack jet impingement on concrete block wall between electrical equipment room and main steam pipe chase from:

30 in. main steam line -	pressure	71.4 psi

4. Critical crack jet impingement forces on boric acid tanks from:

3 in. main steam line to auxiliary feed pump -	pressure	0.44 psi
	area	3.36 $ft.^2$

5. Critical crack jet impingement forces on component cooling heat exchangers from:

3 in. main steam line to auxiliary feed pump -	pressure	0.04 psi
	area	34.6 $ft.^2$

Table A.2-3 MASS AND ENERGY USED IN JET IMPINGEMENT AND PIPE WHIP

Assuming Stop Valve Closure on Signal:

T (sec)	Mass Flow (lbm/sec)	Enthalpy (BTU/lbm)	STM Quality
$T_0 = 0$	5,630	1,190	100
$T_1 = 1.85$ or 3.3*	12,670	708	
$T_2 = 1.85$ or 3.5*	19,700	570	4
$T_3 = 7.55$**	19,700	570	4

* 1.85 sec is shortest time until entrainment - assumes only mass from break to SG plus ½ SG steam mass. 3.3 sec. includes decompression of SG, all piping upstream and down, and total steam mass of SG.

** Assumes 6 sec. valve closure time plus 1.55 sec. for piping blowdown.

(1)Data transmitted from Westinghouse via telecon R. Henderson/J. Kendall on April 25, 1973.

Table A.2-4 LIST OF CREDITED PROTECTION FEATURES FOR JET IMPINGEMENT AND PIPE WHIP

FSAR FIGURE No.	DESCRIPTION
Figure A.2-1	Cable Spreading Room Wall Barrier
Figure A.2-2	Non-Vital Switchgear Room Wall Barrier
Figure A.2-3	Control room Window Impingement
Figure A.2-4	Restraint R1 Aux. Steam Supply to Waste Disposal (Valve 1MOV-2020)
Figure A.2-5	Restraint R2 Aux. Steam Supply to Waste Disposal (Valve 1MOV-2020)
Figure A.2-6	Restraint R3 Aux. Steam Supply to Waste Disposal (Valve 2MOV-2020)
Figure A.2-7	Restraint R4 Aux. Steam Supply to Waste Disposal (Valve 2MOV-2020)

Figure A.2-1 CABLE SPREADING ROOM WALL BARRIER

Sheet 1 of 3

[redacted]

Figure A.2-1 CABLE SPREADING ROOM WALL BARRIER

Sheet 2 of 3

[redacted]

Figure A.2-1 CABLE SPREADING ROOM WALL BARRIER

Sheet 3 of 3

[redacted]

Figure A.2-2 NON-VITAL SWITCHGEAR ROOM WALL BARRIER

[redacted]

Figure A.2-3 CONTROL ROOM WINDOW IMPINGEMENT

[redacted]

Figure A.2-4 RESTRAINT R1

[redacted]

Figure A.2-5 RESTRAINT R2

[redacted]

Figure A.2-6 RESTRAINT R3

[redacted]

Figure A.2-7 RESTRAINT R4

[redacted]

Figure A.2-8 PIPE BREAK SCHEMATIC

[redacted]

Figure A.2-9 GUILLOTINE

[redacted]

Figure A.2-10 SLOT - LONGITUDINAL

[redacted]

Figure A.2-11 CIRCUMFERENTIAL CRACK

[redacted]

A.3 CONTROL OF HEAVY LOADS

A.3.1 OVERVIEW

NUREG-0612 (Reference 1) established guidelines to ensure that the probability and consequences of dropping a heavy load on irradiated fuel or equipment required to achieve safe shutdown and continue decay heat removal are acceptably small. Reference 2 and Reference 4 required licensees to submit information detailing how they met or intended to meet the requirements of NUREG-0612.

Reference 2 established a six-month response for “Phase I” submittals, and a nine-month response for “Phase II” submittals. Reference 4 corrected minor errors in Reference 2.

“Phase I” required compliance with Section 5.1.1 of NUREG-0612. The intent was to ensure that all load handling systems at nuclear power plants are designed and operated so their probability of failure is appropriately small for the critical tasks in which they are employed. Phase I consisted of seven general items:

- Defined Safe Load Paths for heavy load handling over or in the vicinity of irradiated fuel or safe shutdown equipment,
- Procedures for heavy load handling over or in the vicinity of irradiated fuel or safe shutdown equipment,
- Training and qualifications for crane operators,
- Design, testing, and inspection of special lifting devices,
- Installation and use of other lifting devices,
- Inspection, testing, and maintenance of overhead cranes, and
- Design standards for overhead cranes

Reference 5 is the Technical Evaluation Report (TER) that accepted the Point Beach responses for Phase I.

“Phase II” required compliance with Section 5.1.2 through 5.1.6 of NUREG-0612 to ensure that, for load handling systems used in areas where their failure might result in significant consequences, either:

1. The cranes and associated lifting devices satisfy the single failure proof criteria, or
2. Conservative evaluations of load handling accidents indicate that the potential consequences of any load drop are acceptably small.

Reference 6 concluded that the risks associated with damage to safe shutdown systems are relatively small because:

1. nearly all load paths avoid this equipment
2. most equipment is protected by an intervening floor
3. of the general independence between crane failure probability and safety-related systems which has been observed
4. redundancy of components.

Reference 6 also stated that the single most important example of the heavy load concern is the loads handled over the open reactor vessel during refueling (such as the reactor vessel head). It also stated that precautions have been and are being taken such that no accidents have occurred. Reference 6 concluded that the objective identified in Section 5.1 of NUREG-0612 for providing "maximum practical defense, in depth" is satisfied by Phase I compliance and that the Phase II analysis did not indicate the need to require further generic action at that time."

Bulletin 96-02 (Reference 15) alerted licensees to the importance of complying with existing regulatory guidelines associated with the control and handling of heavy loads in all MODES other than cold shutdown, refueling, and de-fueled. It further reminded licensees of their responsibilities for ensuring that heavy load activities carried out under their license are performed safely and within the requirements of Title 10 of the Code of Federal Regulations.

The bulletin further required a review of plans and capabilities for handling heavy loads in accordance with NUREG-0612 (Phase I), and Generic Letter 85-11. Reference 16 reported the results of this review for Point Beach to the NRC. Specific representations of that response are included where applicable in the following sections. In Reference 17, the NRC acknowledged receipt of the Bulletin 96-02 response, accepted it, and closed the NRC review of the matter at Point Beach.

In response to specific questions regarding measures that were or would be in place for the final lift of the original Unit 2 reactor vessel head from the vessel to remove fuel, specific one-time commitments and representations were made by Reference 18. Additional supplemental information pertaining to the risk of a reactor vessel head drop was not specific to the one-time lift.

The following sections summarize the PBNP licensing basis for control of heavy loads.

A.3.2 NUREG-0612 PHASE I REQUIREMENTS AND COMMITMENTS

(Reference 5 unless otherwise noted)

Safe Load Paths

The following cranes are cited as needing to comply with NUREG-0612:

Containment Polar Crane (both units)
Auxiliary Building Main Crane
Turbine Building Main Crane
Circulating Water Pumphouse Monorails (N-S and E-W)*
Reactor Pressure Vessel Head Monorails*
Containment Buttress Jib Cranes*
Main Shop Crane*
Jib Crane Over Core Instrumentation Seal Tables*

*Reference 7 and Reference 9 justified removal of these cranes from the scope of NUREG-0612 based on the separation and redundancy of safety-related equipment that could be impacted by a dropped load. However, in Reference 12 it was affirmed that these cranes either met the

provisions of NUREG-0612 Section 5.1.1, or would by the completion of the next refueling outage. In the final TER (Reference 5), these cranes are cited as being in scope, but that Safe Load Paths were established only for the other four cranes (Containment Polar, Turbine Hall, and Primary Auxiliary Building cranes).

Use of Safe Load Paths are therefore required for the Containment Polar, Turbine Building and Primary Auxiliary Building Cranes when handling heavy loads (defined as 1750 lbs or greater). Use of a second individual with duties defined by procedure to ensure that the crane operator follows approved Safe Load Paths was deemed an acceptable alternative to permanently marking the Safe Load Paths. In addition, approval by the onsite Safety Review Committee (variously termed Manager's Supervisory Staff or MSS, and Plant Operations Review Committee or PORC, over the life of the plant) of deviations from established Safe Load Paths was found to meet the intent of the guideline.

Various plant modifications, which were installed after the final TER (Reference 5) was issued, led to the establishment of additional Safe Load Paths.

The plant was modified to install two additional emergency diesel generators. As part of the installation, safety related cabling was relocated to beneath the floor of the Unit 2 Turbine Hall truck bay. This was under an existing Safe Load Path. The relocation was reviewed for conformance to NUREG-0612 and found to be acceptable (Reference 24).

To protect buried SSCs that are needed for safe shutdown, a Safe Load Path was established in the east yard area for mobile cranes.

The Unit 1 and Unit 2 Steam Generator Blowdown Heat Exchanger Cranes were installed by Reference 22 and Reference 23 and Safe Load Paths established.

Load Handling Procedures

Load specific procedures meeting the requirements of NUREG-0612 5.1.1(2) are required for the following lifts:

1. Spent Fuel Shipping Cask
2. Resin Cask
3. Reactor Vessel Head
4. Reactor Vessel Internals

Generic procedures incorporating the requirements of NUREG-0612 5.1.1(2) are acceptable for all other lifts.

Crane Operator Training

All crane operators shall be trained, qualified, and conduct themselves in accordance with ANSI B30.2-1976, Chapter 2-3 (“Qualifications for and Conduct of Operators”) with the following approved exceptions:

1. The warning bell will be actuated only as required to advise personnel of crane movement, rather than continuously during crane motion.

2. The main line disconnect switch will not be left open. Use of main or local disconnect switches allow the crane to be deenergized for servicing.
3. The cranes will not be deenergized for normal maintenance since some maintenance requires that the power be on. Alternative safety practices will be used when servicing cranes.
4. Crane controls will not be tested at the beginning of each shift. They will be tested at the beginning of each lifting operation.
5. Medical examinations will include eye examinations to meet the requirements of ANSI B30.2-1976 Sections 2-3.1.2b, 1 and 2.

Special Lifting Devices

The special lifting devices identified in scope are:

1. Reactor Head Lifting Device
2. Upper Internals Lifting Device
3. Reactor Coolant Pump Motor Lifting Device
4. MSB Transfer Cask (MTC) and Associated Lifting Yoke
5. NUHOMS® OS197-PB Lift Beam
6. NUHOMS® OS197-1 Transfer Cask

The following discussions for design, fabrication and testing and inspection are only applicable to the special lifting devices for the Reactor Head, Upper Internals and RCP Motor. Information on the MSB Transfer Cask (MTC) and Associated Lifting Yoke can be found in the Final Safety Analysis Report for the VCS-24 Ventilated Storage Cask System. Information on the NUHOMS® OS197-PB Lift Beam and Transfer Cask can be found in the Point Beach 10 CFR 72.212 and Certificate of Compliance Evaluation Report for NUHOMS®-32PT System, Appendix B.

Design

These special lifting devices generally meet the criteria of ANSI N14.6-1978. Specific approved exceptions are taken for:

- Dynamic loads are considered minimal (due to low crane speeds) and have been disregarded,
- Load stress design safety factors less than the Code-required 3 or 5 for the following components of the internals lift rig:
  - Adaptor pin
  - Lift lug pin
  - Side lug pin
  - Sling leg pin

To justify these exceptions, Reference 14 contained extensive design margin analyses, diagrams, and detailed NDE requirements for the special lifting devices.

Fabrication

While a formal quality assurance program was not used in the fabrication of the special lifting devices, a completed review of the manufacturing process (including material selection, welders, welding procedures, and conformance to drawing requirements) was found to be an acceptable alternative.

Testing, Inspection, and Continued Compliance

Annual 150% load testing was waived due to impracticality (internals and head lift rigs) or substantial design margins (reactor pump motor lift rig).

Refueling interval visual inspections and 10-year NDE inspections of critical welds was approved in lieu of annual visual inspections. In the event of major maintenance or application of substantial stresses, it was agreed that lifting the designated loads a short distance for 10 minutes, and visually inspecting critical welds of concern was acceptable.

Lifting Devices (Not Specifically Designed)

All other lifting devices shall be designed, fabricated, and proof-tested per the requirements of ANSI B30.9-1971 and NUREG-0612 Section 5.1.1(5) with the following exceptions:

- Slings used in the Turbine Building south of column line 10 and north of column line 13,
- Slings used in the transport of the turbine rotors
- Rather than inspections on a regular basis per Section 9-2.8.1 of ANSI B30.9-1971, inspections may be performed prior to each use.
- Any existing slings that do not meet the requirements of ANSI B30.9-1971 are to be retired. Until retired, such slings are to be derated by a factor of two.

Inspection, Testing And Maintenance Of Cranes

In-scope cranes are inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976 with the exception of the containment polar cranes. It is not practical to meet the frequencies of ANSI B30.2 for periodic inspection and tests of these limited use cranes. Therefore, these cranes are given an initial inspection in accordance with OSHA requirements prior to use. The major annual inspection, fulfilling the requirements of Chapter 2-2 is performed prior to use in refueling outages.

At the time of the original submittal, refueling outages were every 12 months. Extension of the inspection interval to 18 months coinciding with a longer fuel cycle is consistent with Section 5.1.1(6) of NUREG-0612.

Crane Design

NUREG-0612 Section 5.1.1(7) requires that in scope cranes be designed in accordance with ANSI B30.2-1976, “Overhead and Gantry Cranes”, and CMAA-70, “Specifications for Electric Overhead Traveling Cranes.”

The in scope cranes at Point Beach were designed to EOCI-61, which was later superseded by CMAA-70. Reference 5 contains a reconciliation of 18 specific design criteria that differ between the Code of Construction and those endorsed by NUREG-0612.

It was also committed that the primary auxiliary building crane would be upgraded to meet the single failure proof requirements of Section 5.1.6 and Appendix C of NUREG-0612 (Reference 3).

Interim Protection Measures

In addition to the general requirements delineated above, six interim measures were required to be implemented. Four of these interim measures were encompassed by the Safe Load Paths, Load Handling Procedures, Crane Operator Training, and Crane Inspection, Testing, and Maintenance items discussed above. The two additional items were:

- A revision of the "Refueling and Spent Fuel Assembly Storage" Technical Specification to prohibit movement of heavy loads over and in the spent fuel pool until such time as a single-failure-proof crane was installed, and
- A special review for Heavy Loads Over the Core. Implementation of interim safe load paths, training, maintenance practices, etc. Acceptable completion of this activity was documented in Reference 5.

Reference 3 approved the removal of the Technical Specification restrictions on the movement of heavy loads over the spent fuel pool following completion of modifications to make the primary auxiliary building crane single failure proof.

Phase II Submittals

Phase II implementation of NUREG-0612 was to ensure that all heavy loads that could cause damage to irradiated fuel or safe shutdown equipment due to a load drop would either:

1. Be handled by cranes and lifting devices that satisfy single-failure-proof criteria, or
2. Be analyzed to demonstrate that the consequences of such a load drop would be acceptably small

To be successful, an analysis performed under the second option was required to meet four criteria. Based on calculations assuming an accidental dropping of a postulated heavy load,

I. Releases of radioactive material must produce doses that are equal to or less than 25% of the 10 CFR 100 limits, and

II Damage to fuel and fuel storage racks must not result in a configuration of the fuel such that k_{eff} is larger than 0.95, and

III Damage to the reactor vessel or spent fuel pool is limited so as not to result in water leakage that could uncover the fuel. Makeup water to overcome the leakage must be from a borated source of sufficient concentration if the water being lost is borated, and

IV Damage to redundant or dual safe shutdown equipment will be limited so as not to result in loss of required safe shutdown functions.

Reference 7 transmitted the initial responses to Phase II, and was later supplemented by Reference 8. These responses delineated the cranes capable of carrying loads which could, if dropped, land or fall into the spent fuel pool, on or in the reactor vessel, and/or on equipment necessary to maintain safe shutdown conditions, continued decay heat removal, or spent fuel pool cooling.

Only the primary auxiliary building crane and the containment polar cranes met the criteria for inclusion in the scope of this section of NUREG-0612. Since it was shown that the remaining plant cranes are incapable of impacting redundant or dual safe shutdown equipment from a single heavy load drop event, they were screened out of the requirement to be analyzed to Criterion IV (Reference 7).

A.3.3 AUXILIARY BUILDING CRANE

It was committed that the primary auxiliary building crane would be modified to make it single failure proof (Reference 7). Further commitments to communicate Design Rated Loads (DRL) and Maximum Critical Loads (MCL), etc. were deferred pending selection of a supplier of the modifications. A DRL and MCL of 100 tons were communicated in Reference 11, and was later increased to 125 tons by Reference 13. The commitment to upgrade the crane was part of the basis for Phase I, and was re-iterated in the commitments for Phase II.

The PAB superstructure has been analyzed for the capability of the structure to support and hold the crane with its full rated lift load of 125 tons plus a roof snow load and a concurrent seismic (OBE or SSE) event or a lift of 125 tons plus a roof snow load and design wind loads (Reference 21).

A.3.4 CONTAINMENT POLAR CRANE

A reactor vessel head drop analysis was initiated to determine the consequences of such an event. Reference 8 indicated that the initial analysis (limited to assessing the potential damage to the RCS, and not addressing the potential dose consequences) concluded that severe damage to the safety injection lines and primary loop piping could occur.

Consideration of boron concentration, the maximum permissible K_{eff}, and the potential for positive reactivity addition from a heavy load drop was limited to times when the Programmed and Remote (PaR) inspection device was being handled above a fueled core. This was because procedural controls limit the handling of heavy loads above an exposed core to the PaR device, the upper internals, and the reactor head. Due a combination of geometry and handling precautions for the latter two objects, it is not possible for them to contact fuel assemblies in the core in a way that could credibly cause core geometry changes. Therefore, movement of the PaR device over a core containing fuel assemblies is not allowed (Reference 19).

Subsequent developments prompted the creation of a new type of accident analysis for Point Beach: The Reactor Vessel Head Drop. The details of this analysis are located in FSAR Section 14.3.6. PBNP committed to incorporate the PBNP method of NUREG-0612

Phase 1 compliance into the PBNP FSAR by letter NRC 2005-0094, dated July 24, 2005. (Reference 19)

Reactor Vessel Head Lift

The integrated replacement reactor vessel heads at PBNP involve low headroom lifts. Therefore, measures to specifically minimize the potential for "two-blocking," as defined in NUREG-0612, have been developed. These measures include testing of controls and limit switches and operational restrictions when the load is near its maximum lift height.

NUREG-0612 defines a "two-blocking" event as the act of continued hoisting to the extent that the upper head block and the load block are brought into contact, and, unless additional measures are taken to prevent further movement of the load block, excessive loads will be created in the rope reeving system, with the potential for rope failure and dropping of the load. Of particular applicability to industry-standard handling systems is the potential for the wire rope supporting the load block to be cut or overloaded. This is of special concern with low headroom lifts where the load block is deliberately raised near the upper block in order for the load to clear obstructions. From this position, a stuck relay or operator error, combined with failure of the upper limit switch, could cause a load drop before corrective measures, such as removing power to the crane, could be implemented.

The main hoist of each polar crane is equipped with two independent upper travel limit switches to prevent the possibility of a "two-blocking" incident. The two independent upper travel limit devices are of different design and are activated by independent mechanical means. These devices independently de-energize either the hoist drive motor or the main power supply.

The first limit switch that would be activated is a gear-actuated travel limit. This limit switch is activated by the rotation of the main hoist drum. It is set to actuate prior to a potential "two-blocking" incident. This limit switch de-energizes the hoist drive motor and prevents further movement in the upward direction.

A second counterweight-activated limit switch would be relied upon if the geared limit switch fails. This limit switch would be activated by the physical contact of the lower block with a counterweight connected to the limit switch. The initial contact of the lower block with the counterweight will occur approximately six inches prior to the close of the limit switch. The actual close of the limit switch will occur sufficiently prior to "two-blocking." The limit switch circuit will de-energize the main power supply and consequently apply the hoist braking system.

The independent limit switches are set to ensure that sufficient margin exists between the actuation of these switches and physical contact of the upper and lower blocks. The counterweight-activated limit switch is functionally tested in accordance with maintenance procedures, prior to use if the crane has been idle greater than six months. The gear-actuated upper limit switch is tested daily when the containment crane is in use in accordance with maintenance procedures. Polar crane controls are also checked in accordance with procedures. These checks include initial checks following installation of the radio controls, as well as daily checks when the crane is in use. In addition, prior to lifting the RVH, procedures require a pre-lift inspection to be performed that includes a functional check of the main hoist gear-actuated upper limit switch.

The evolution to move the reactor head from the vessel to the storage stand involves lifting the head vertically, directly above the vessel, to an elevation that permits clearance of the 66-foot elevation interferences. The head is then moved horizontally from above the vessel to a location directly above the storage stand and finally lowered onto the stand. The movement from the stand to the vessel is a reverse of the above movements. The movement of the RVH is controlled by Safe Load Path (SLP) procedures. The SLP provides a path that minimizes crane manipulations and movement while the head is over the vessel.

The head is lifted straight up to the height needed to clear containment 66-foot elevation before any bridge or trolley moves are made. The PBNP refueling cavity design does not have adequate room to allow the head to be moved to a position that completely clears the reactor vessel once the head is above the guide studs. When the head is replaced it is again lifted to a height necessary to clear the containment 66-foot elevation and then moved directly above the vessel, using crane reference marks and then lowered onto the vessel. This sequence minimizes crane manipulations and thus the potential for a crane failure or human performance induced failure.

The integrated RVH assembly has an overall height that is taller than the original RVH assembly. To move the head between the storage stand and the vessel, the bottom flange of the head needs to clear the 66-foot elevation and other physical obstructions attached to the 66-foot elevation, which results in a lift height of 26.4 feet. The installed polar crane main hook has a maximum lift height, as determined by the physical design and limit switch settings. Based on the maximum physical hook elevation prior to "two-blocking" and the replacement head assembly height, the maximum height of the bottom flange of the replacement head is approximately the 69-foot, 5-inch plant elevation (without limit switch settings). The inclusion of the main hook limit switch settings results in a maximum height of the bottom flange of the replacement head at approximately the 67.5-foot plant elevation.

Operational restrictions will be included in procedures height of the bottom of the RVH flange to the 67-foot elevation to prevent actuation of the upper travel limit switches. The restriction will maintain adequate margin below the actuation setpoint of the first upper travel limit switch. This limit is maintained by using physical references and visual level checks during the lift.

The two independent limit switch designs, combined with the testing and operational restriction of the lift height, provide assurance that a “two-blocking” incident potential is minimal.

A.3.5 REFERENCES

1. NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” July 1980.
2. NRC Letter, “Control of Heavy Loads,” December 22, 1980.
3. NRC Safety Evaluation, “NUREG-0612 Control of Heavy Loads,” September 3, 1985.
4. NRC Generic Letter 81-07, “Control of Heavy Loads,” February 3, 1981.
5. TER-C5506-382/383, “Control of Heavy Loads at Point Beach Nuclear Plant, Units 1 and 2,” March 2, 1984 (incorporated into NRC Safety Evaluation March 27, 1984).

6. NRC Generic Letter 85-11, Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants" NUREG-0612", June 28, 1985.

7. WEPCo Letter to NRC, "NUREG-0612 - Control of Heavy Loads at Nuclear Power Plants Transmittal of Nine-Month Response and Updated Six-Month Response Information," January 11, 1982.

8. WEPCo Letter to NRC, "Submittal of Outstanding Response Items NUREG-0612 - Control of Heavy Loads," November 22, 1982.

9. WEPCo Letter to NRC, "Submittal of Additional Information in Response to Draft Technical Evaluation Report NUREG-0612, Control of Heavy Loads," June 30, 1982.

10. Not Used.

11. WEPCo Letter to NRC, "Submittal of Outstanding Information NUREG-0612, Control of Heavy Loads," September 16, 1982.

12. WEPCo Letter to NRC, "Submittal of Additional Information in Response to Draft Technical Evaluation Report NUREG-0612, Control of Heavy Loads," September 28, 1983.

13. WEPCo Letter to NRC, "Transmittal of Additional Information NUREG-0612 - Control of Heavy Loads," February 15, 1983.

14. WEPCo Letter to NRC, "Transmittal of Additional Information NUREG-0612-Control of Heavy Loads," July 23, 1982.

15. NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel In the Reactor Core, or Over Safety-Related Equipment," April 11, 1996.

16. WEPCo Letter to NRC, "Response to NRC Bulletin 96-02, Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," May 9, 1996.

17. NRC Letter to WEPCo, "Completion of Licensing Action for NRC Bulletin 96-02, Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," April 16, 1998.

18. NRC 2005-0050A, NMC Letter to NRC, "Response to Request for Additional Information, Revision 1 NUREG-0612, Control of Heavy Loads Reactor Vessel Head Drop Analysis," April 20, 2005.

19. NRC 2005-0094, NMC Letter to NRC, "Request for Review of Heavy Load Analysis," July 24, 2005.

20. "Point Beach Nuclear Plants, Units 1 and 2 - Issuance of Amendment Re: Incorporation of Reactor Vessel Head Drop Accident Analysis into the Final Safety Analysis Report," September 23, 2005.

21. Automated Engineering Services Corp. Calculation PBNP-305336-S01, Rev. 1, "Structural Analysis of Central PAB with Crane Load of 125 Tons," dated April 3, 2006.

22. Engineering Change EC 12268, Modification to Leave the Crane Runway for the SGBD Heat Exchanger Replacement in Place, Closed April 7, 2009.

23. Engineering Change EC 13249, U2 SGBD HX Handling Equipment, Closed August 3, 2010.

24. SCR 2010-0204, "Heavy Load Handling considerations for Installation of Vital Cabling Beneath U2 Turbine Hall Truck Bay," September 29, 2010.

25. SCR 2012-0005, "Establish a Safe Load Path for Mobile Cranes in the East Yard Area," January 25, 2012.

A.4 (DELETED)

A.5 SEISMIC DESIGN ANALYSIS

A.5.1 SEISMIC DESIGN CLASSIFICATIONS

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

1. TID-7024, "Nuclear Reactors and Earthquakes" August, 1963 and,
2. 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.

Class I

Those structures and components including instruments and controls whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of excessive amounts of radioactivity. Also, those structures and components vital to safe shutdown and isolation of the reactor.

Class II

Those structures and components which are important to reactor operation but not essential to safe shutdown and isolation of the reactor and whose failure could not result in the release of substantial amounts of radioactivity.

Class III

Those structures and components which are not directly related to reactor operation or containment.

When required to maintain a system's safety related functions, the interface between a Class I system and lower Class system is at a normally closed valve, a valve which is capable of remote operation from the control room, or a valve which is capable of self actuation.

Non-Seismic SSC over Seismic SSC (Also known as Seismic II/I or Seismic 2/1)

Class III systems and equipment including pipe are generally not designed to withstand any seismic loads. However, for the "Generic Letter 87-02 Unresolved Safety Issue (USI) A-46" effort, the seismic adequacy of certain PBNP equipment, including the potential interaction between class III and Class I structures, systems and components (SSC), was evaluated (Reference 19). Modifications were made to resolve seismic concerns identified by the review. Subsequent to the Generic Letter 87-02/USI A-46 seismic review effort, Class III structures, systems and components in the power block are now reviewed for earthquake loads if the potential for interaction with safety related SSCs exists. Class III SSC that could interact with Class I SSC are typically identified as Seismic II/I or Seismic 2/1.

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

1. Primary steady state stresses, when combined with the seismic stresses resulting from a response spectrum normalized to a maximum ground acceleration of 0.04g in the vertical direction and 0.06g in the horizontal direction 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, ACI 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 a response spectrum normalized to a maximum ground acceleration of 0.08g acting in the vertical direction and 0.12g acting in the horizontal direction 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.

All Class II components are designed on the basis of a static analysis for a maximum acceleration of 0.04g acting in the vertical direction and 0.06g acting in the horizontal direction simultaneously.

The spectrum response curves for the equipment inside the building are generated by the time history technique of seismic analysis. The sample earthquake utilized is that recorded at Olympia, Washington 45N-120W on April 13, 1949. The originally recorded earthquake is scaled to that of .06g. Essentially, the curves are generated by applying the recorded earthquake to a single degree of freedom system, for which the values for damping and natural frequency are varied. Some averaging of the curves is provided to smooth out the erratic response of the earthquake's random behavior. At the high frequency end of the curve, the acceleration levels converge to the peak input value at the location inside the building. Table A.5-2 gives the damping factors used in the design of components and structures. The 2% and 5% damping values given in the table for the containment structure include the soil-structure interaction damping.

The design of Class I Nuclear Steam Supply System Equipment and Supports employs one of three approaches to the problem. The first utilizes the “response spectrum” approach and modal analysis of the dynamic loads imparted by the earthquake. The supports are provided in accordance with actual system response to the earthquake. The second approach provides adequate supports to remove the piping system from the “resonance range.”

The third method of analysis applied to Class I Nuclear Steam Supply Equipment and Supports proceeds as follows. A conservative analysis is accomplished by assuming that the natural period of vibration of the structure or components lies at the peak of the floor response spectrum, and that the corresponding response acceleration is used for the analysis at the appropriate damping value. Independent of the method adopted, the following applies:

1. Stresses and deflections resulting from the combined influence of normal loads and the seismic loads due to the design earthquake (ground accelerations of 0.04g acting in the vertical and 0.06g acting in the horizontal direction simultaneously) are calculated and checked against the limits imposed by the design standard.

2. Stresses and deflections resulting from the combined influence of normal loads and the seismic loads due to the assumed hypothetical earthquake (ground accelerations of 0.08g acting in the vertical and 0.12g acting in the horizontal direction simultaneously) are calculated and checked to verify that deflections do not cause loss of function and that stresses do not produce rupture.

For mechanical components of Engineered Safeguards Systems, analysis will be performed on a worst plant basis to determine the response in the frequency range of interest. Modifications will be made as necessary considering potential for resonance responses. The component will then be analyzed using seismic loads as obtained from building response calculations to show that stresses and deflections are within allowable limits and will not result in loss of function.

The seismic design of Class I piping systems in the nuclear plant employed one of three methods: (1) the main steam line in the containment and several other piping systems were treated using the response spectrum techniques coupled with a multidegree of freedom modal analysis including the effect of modal participation factors. The piping seismic restraints were designed in accordance with the dynamic response of the system. (2) Some of the piping systems were provided with adequate restraints to remove the fundamental frequency of the system from the "resonance range." Where the piping system can be assumed rigid with respect to the building, the maximum accelerations and displacements at the attachment points to the building, as determined from the building analysis, are used for design. (3) The major portion of the Class I piping seismic restraints are designed so as to withstand the peak accelerations determined from floor response spectra.

NOTE: The following describes the original method used to seismically qualify Class I equipment. Additional verification of the seismic adequacy of plant mechanical and electrical equipment was performed as discussed in Section A.5.6, Verification of Seismic Adequacy of Equipment per Generic Letter 87-02.

Class I equipment, including heat exchangers, pumps, tanks, valves, motors, and electrical equipment components, are analyzed in one of four methods depending on the relative rigidity of the equipment being analyzed. (1) Equipment which is rigid and rigidly attached to its support structure is 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, is analyzed for the spectral peak of the floor response curve for appropriate damping values. (3) In some instances non-rigid equipment is analyzed using a multidegree 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 are determined in each mode. The final seismic stress is determined by summation of individual modal stresses on a square-root-sum-of-the-squares basis. (4) Type testing of selected electrical equipment has been conducted to demonstrate seismic design adequacy as described in WCAP 7397-L. For the analysis of equipment to resist the vertical seismic component, 2/3 of the ground response spectrum curves for the design or hypothetical earthquake are used to determine the acceleration appropriate to the vertical frequency.

Class I equipment, which is located at or below ground level, e.g., pumps and emergency diesel generators (DG), is analyzed using the Housner ground response spectrum defined for the Point Beach site in the FSAR and repeated in Figure A.5-1 and Figure A.5-2. The "g-values" were obtained from the response curves for the appropriate damping values as defined in Table A.5-2.

Engineered Safeguards tanks, e.g., Spray Additive (CS) and Refueling Water Storage (SI), are analyzed for at least ground acceleration of 0.06g in any direction horizontally and 0.04g vertically occurring simultaneously, and in conjunction with other loads, without exceeding allowable stresses. Hydrodynamic analyses of these tanks have been performed using the methods described in Chapter 6 of "U.S. Atomic Energy Commission - TID 7024".

Heat exchangers associated with the Engineered Safeguards Systems have been analyzed to determine the response in the frequency range of interest to show that stresses and deflections are within allowable limits. The method of dynamic analysis uses a proprietary computer code called WESDYN. This code uses as input; inertia values, member sectional properties, elastic characteristics, support and restraint data characteristics, and the 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 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 root mean square method.

Some Class I ventilation fans are mounted on elastomer shock isolation pads with flexible characteristics in the longitudinal, transverse, and vertical directions. The manufacturer supplied the pads' frequency relationships. The spectral acceleration was obtained using the frequency, appropriate damping value, and proper floor response spectrum.

The procedure for determining base shear for the design earthquake acceleration for a Class II item is the same as the procedure used for a Class I item.

A.5.2 SEISMIC CLASSIFICATION OF STRUCTURES AND EQUIPMENT

As-built construction drawings for all structures, equipment, and systems were compiled as construction progressed by the contractor and architect engineers. Copies of these "as-built" drawings are retained by the Licensee.

The contract between the Licensee and prime contractor specified that the Licensee shall be furnished copies of all appropriate calculations. In some instances, Licensee has agreed with the designers that calculations will be retained in the designers' permanent files and made available to the utility as required on request. This is particularly appropriate where the Licensee believes designers' participation is essential for proper interpretation and analysis of applicable future events. Particular structure and equipment classifications are given below:

Buildings and Structures	Seismic Class
Containment, including all penetrations and airlocks, the concrete shield, the liner, the interior structures, and the dome truss (for support of the containment spray piping and the containment air recirculation cooling system (VNCC) ductwork)	I
Spent fuel pool	I
Control room	I
Diesel generator room	I
Pumphouse (to the extent that water is always available to the service water pumps)	I
Auxiliary building (except for steel superstructure)	I
Emergency Diesel Generator Building (except stairway enclosure) (Reference 16 and Reference 17)	I
Turbine structure (Reference 4, Reference 5 and Section 10.2.5)	III
Buildings containing conventional facilities, auxiliary building superstructure, and diesel generator building stairway enclosure (See Section 10.2.5)	III
Steam Generator Storage Bldg. (Reference 1)	III
Equipment, Piping, and Supports	
Reactor Control and Protection System including miscellaneous relay racks (Reference 2) (except main feedwater flow transmitters, which are Class III)	I
Radiation Monitoring System	III
ATWS circuit isolation devices (Reference 3)	III
Process Instrumentation and Controls	I
Reactor	I
Vessel and its supports Vessel internals	

Buildings and Structures	Seismic Class
Fuel assemblies	
RCC assemblies and drive mechanisms	
Supporting and positioning members	
In-core instrumentation structure	
Reactor Coolant System	I
Piping and valves containing full system pressure (including safety & relief valves)	
Steam generators	
Pressurizer	
Reactor coolant pumps	
RCP Oil Collection	
Supporting and positioning members	
Reactor Coolant Gas Vent System (Reference 7 & Reference 8)	
LTOP System (Reference 6)	
Engineered Safety Features	I
Safety Injection System (including safety injection and residual heat removal pumps (ACS), refueling water storage tank, accumulator tanks, residual heat exchangers (ACS), and primary connecting piping and valving)	
Containment Spray System (including spray pumps, spray headers, spray additive tank and primary connecting piping and valving)	
Containment Ventilation System (including fans, coolers, ducts and valves) (Containment Air Sampling System is an ESF but is not Seismic Class I.)	
Main Steam supply to TDAFWPs (Reference 9)	I
Auxiliary Building Ventilation System	III (See Section 9.5)
VNCR - CREFS Subsystem	I (See Section A.5.6.3)
CREFS Backup Filtration System	I (See Section A.5.6.3)
Condensate Storage tanks	III

Buildings and Structures	Seismic Class
Pressurizer relief tank	II
Residual heat removal loop	I
PAB Battery and Electrical Equipment Room Ventilating System	I
Component cooling loop	I
Sampling System	II
Spent Fuel Storage Racks (Reference 10)	I
Spent fuel pool cooling loop	I
Spent fuel pool purification loop	I
Fuel transfer tube	I
Emergency Power Supply System	I
Diesel generators, associated fuel oil storage tanks and fuel oil transfer system DC power supply system Power distribution lines to equipment, transformers, switchgear supplying the engineered safety features Control panel boards Motor control centers	
Control Equipment, facilities and lines necessary for the above Class I items	I
The control air supply from the accumulators for pressurizer PORV's (Reference 11)	I
Waste Disposal System	I
Waste holdup tank Sump tank Gas decay tanks Reactor coolant drain tank Waste gas compressor package Waste evaporator Waste evaporator feed pump Sump tank pumps Interconnecting waste gas piping	
Waste Disposal System	III

Buildings and Structures	Seismic Class
All elements not listed as Class I (including Blowdown Evaporator)	
Containment crane	I
Manipulator and other cranes	III
Conventional equipment, tanks and piping, other than Class I and II	III
Service Water pumps and piping, including service water for fire protection of Class I components where required	I
Main Feedwater equipment that isolates the supply of Main Feedwater to the Steam Generators (Reference 12)	III
Fire protection pumps and piping except as noted above	III
Auxiliary Feedwater System (except for Condensate Storage Tanks and some interconnected branch piping) (standby Steam Generator feedpumps and associated components are seismic class I for system pressure boundary integrity.) (Reference 5, Reference 13, Reference 14, and Reference 15)	I
The Chemical and Volume Control System is Class I except:	
Boric Acid Storage Tank	II
Batching tank	II
Evaporator condensate demineralizers	II
Condensate filter	II
Monitor tanks	II
Monitor tank pumps	II
Deborating demineralizers	II
Concentrates holding tank	II
Concentrates holding tank transfer pumps	II
Chemical mixing tank	III
Resin fill tank	III

A.5.3 CLASS I DESIGN CRITERIA FOR VESSELS AND PIPING

All components of the reactor coolant system and associated systems are designed to the standards of the applicable ASME Code or USAS Code. The loading combinations which are employed in the design of Class I components of these systems, i.e., vessels, piping, supports, vessel internals and other applicable components, are given in Table A.5-3.

This table also indicates the stress limits which are used in the design of the listed equipment for the various loading combinations.

To be able to perform their function, i.e., allow core shutdown and cooling, the reactor vessel internals must satisfy deformation limits presented in Table A.5-1 as well as the stress limits shown in Section 14.3.3. For this reason the reactor vessel internals are treated separately. The load combinations used in reactor coolant system component design also include a case assuming simultaneous occurrence of a hypothetical earthquake and design basis accident. This is the case of a hypothetical earthquake occurring during the steady-state portion of the design base accident. The analysis shows that RCS integrity would not be further compromised by this occurrence.

Piping, Vessels, and Supports

The reasoning for selection of the load combinations and stress limits given in Table A.5-3 is as follows. For the design earthquake, the Class I components are designed to be capable of continued safe operation, i.e., for the combination of normal loads and design earthquake loading.

In the case of the assumed hypothetical earthquake, it is only necessary to ensure that critical components do not lose their capability to perform their safety function, i.e., shut the plant down and maintain it in a safe condition. This capability is ensured by maintaining the stress limits as shown in Table A.5-3. No rupture of a Class I pipe can be caused by the occurrence of the assumed hypothetical earthquake.

Careful design and thorough quality control during manufacture and construction and periodic inspection during plant life, ensures that the independent occurrence of a reactor coolant pipe rupture is extremely remote. If it is assumed that a reactor coolant pipe ruptures, the stresses in the unbroken legs will be equal to or less than those allowed per loading condition 4 of Table A.5-3.

For the extremely remote events represented by the hypothetical earthquake, or the design basis accident, or the hypothetical earthquake in combination with the steady state portion of the design basis accident, the design of Class I piping and components is checked for no loss of function, i.e., contain fluid and allow fluid flow. This is assured by limiting the various stress combinations within the limit curves as presented in WCAP 5890, Revision 1, as modified by Note 1 of this Section. This minimum margin of safety between the design stress limit and the expected collapse condition is that for the case of pure tension and is defined as:

$$\frac{S_{ultimate} \angle S_{design}}{S_{design}}$$

In the more practical cases of design, piping and vessels will always experience some combination of tension and bending. For these combinations of loads the margin of safety is larger than that for pure tension, as shown by the limit curves contained in WCAP-5890, Revision 1. Therefore, it is conservative to base the margin of safety on pure tension.

Reactor Vessel Internals - Design Criteria for Normal Operation

The internals and core are designed for normal operation conditions and subjected to loads of mechanical, hydraulic, and thermal origin. The loading of the structure due to the design earthquake as well as the operational transients is considered as an upset condition.

The stress criteria of ASME Section III, which are used as a guide in the design of the internals and core with exception of those fabrication techniques and materials which are not covered by the Code such as the fuel rod cladding including the operating earthquake, are based on a limit design theory with the assumption that the material behavior is perfectly plastic with no strain-hardening. The criteria are chosen so that the structure has a sufficient margin against the limit load for primary stresses and that shakedown to elastic behavior is assured for secondary stresses.

Section 14.3.3 lists the stress criteria for the core and internals integrity analysis in the case of primary system pipe rupture. The limitations established on the internals are concerned principally with the maximum allowable deflections and/or stability of the parts. For the blowdown accident the assumption of a perfectly plastic material with yield stress equal to the initial yield stress of the actual material at temperature is too conservative. Therefore, for this case we are in agreement with the NRC developed stress criteria, which are based on the same limit analysis concept as the criteria of Section III, but take credit for the strain hardening capabilities of the materials. The allowable stress values given in Section 14.3.3 are based on the actual stress-strain curve of 304 SS at 600°F.

The members are designed under the basic principles of: (1) maintaining distortions within acceptable limits, (2) keeping the stress levels within acceptable limits, and (3) prevention of fatigue failures.

To study the seismic response of the reactor internals, a dynamic, elastic study is performed as follows. The maximum stresses are obtained by combining the contributions from the horizontal and vertical earthquakes by adding components. These stresses are then superimposed on the normal operating stresses. The following paragraphs describe the horizontal and vertical contributions for the standard 2-loop, 12 ft. core, reactor internals.

<u>Horizontal Earthquake Model and Procedure</u>

The reactor building with the reactor vessel support, the reactor vessel, and the reactor internals are included in this analysis. The mathematical model of the building, attached to ground, is similar to that used to evaluate the building structure. The reactor internals are mathematically modeled by beams, concentrated masses, and linear springs.

All masses, water, and metal are included on the mathematical model. All beam elements have the component weight or mass distributed uniformly, e.g., the fuel assembly mass and barrel mass. Additionally, wherever components are attached somewhat uniformly their mass is included as an additional uniform mass, e.g., baffles and formers acting on the core barrel. The water near and about the beam elements is also included as a distributed mass. Horizontal components are considered as concentrated mass acting on the barrel. This concentrated mass also includes components attached to the horizontal members since this is the media through which the reaction is transmitted. The water near and about these separated components is considered as being additive at these concentrated mass points.

The concentrated masses attached to the barrel represent the following; (1) the upper core support structure, including the upper vessel head and one-half the upper internals; (2) the upper core plate, including the thermal shield and the other half of the upper internals; (3) the lower core plate, including one-half of the lower core support columns; (4) the lower one-half of the thermal shield; and (5) the lower core support, including the lower instrumentation and the remaining half of the lower core support columns.

The modulus of elasticity is chosen at its hot value for the three major materials found in the vessel, internals, and fuel assemblies. In considering shear deformation, the appropriate cross-sectional area is selected along with a value of Poisson's ratio. The fuel assembly moment of inertia is derived from experimental results by static and dynamic tests performed on fuel assembly models. These tests provide stiffness values for use in this analysis.

Modal analysis, plus the response spectrum method (Note 2) is used in this analysis. Natural frequencies and normal modes are obtained by the use of a transfer matrix method.

The maximum deflection, acceleration, is determined at each particular point by summing the absolute values obtained for all modes. Shear forces and bending moments are determined, and the earthquake stresses are calculated.

Analytical Model for Vertical Earthquake Model and Procedure

The reactor internals are modeled as a single degree of freedom system for vertical earthquake analysis using all the spring constants from the ground to the core.

Spent Fuel Storage

The spent fuel storage pool is constructed of reinforced concrete and is Class I seismic design. The entire interior basin face and transfer canal is lined with stainless steel plate. The spent fuel storage racks are designed in accordance with Regulatory Guide 1.29, Revision 2 as seismic Class I components. The structural analysis of the racks has considered all the loads and the load combinations specified in the NRC Standard Review Plan. The honeycomb steel structure of the rack not only provides a smooth all welded stainless steel box structure to preclude damage during normal and abnormal load conditions, but also provides an additional margin of safety in the form of internal structural damping created by the large areas of load bearing surface between boxes with array.

Design Criteria for Abnormal Operation

The abnormal design condition assumes blowdown effects due to a reactor coolant pipe double-ended break. For this condition, the criteria for acceptability are that the reactor be capable of safe shutdown and that the engineered safety features are able to operate as designed. Consequently, the limitations established on the internals for these types of loads are concerned principally with the maximum allowable deflections. The deflection criteria for critical structures under abnormal operation are presented in Table A.5-1.

Reactor Vessel

The criteria for movement of the reactor vessel, under the worst combination of loads, i.e., normal plus the assumed hypothetical earthquake or normal plus reactor coolant pipe rupture loads, assures that the movement of the reactor vessel will not exceed the clearance between the reactor coolant piping and the surrounding concrete nor cause stresses in excess of the levels set forth in Table A.5-3.

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, and will result in stress levels as set forth in Table A.5-3.

The relative motions between components will be controlled to within Table A.5-1 limits by the stiffness of the supporting structure. Where provisions for thermal growth are necessary, snubbers will be provided to serve as limit stops under seismic and pipe rupture loading.

ANALYSIS NOTES

Note 1

1. Use material data to develop stress-strain curves. Typical stress-strain curves of Type 304 Stainless Steel, Inconel 600, and SA 302B low alloy steel at 600° F have been 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 have been 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 as described above.

2. Normalize the ordinate (stress) of the stress-strain curves to the measured yield strength.

3. Use 20% of uniform strain as defined on the curve developed under Item 1 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 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. 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 limit. If such an approach was adopted, sufficient documentation was provided to support the actual material properties used.

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

Note 2

Shock and Vibration Handbook, edited by Harris and Crede, Volume 3, Chapter 50, “Vibration of Structures Induced by Seismic Waves”, by George W. Housner.

A.5.4 SEISMIC DESIGN OF CLASS I STRUCTURES

Introduction

The following supplementary information is provided in support of the seismic design of structures and equipment for the Point Beach Nuclear Plant.

a. With reference to FSAR Figure 5.1-14, the sum of the lumped masses representing the containment structure (designated WT) is 39,000,000 lbs. The sum of the weights of the interior concrete and equipment shown as dashed circles is 22,000,000 lbs, of which the steam generators contribute about 1,500,000 lbs.

   Higher modes (more than two) have been checked and found to be insignificant. Absolute values of the forces are added instead of the RMS (as in Section 5.1.2.4) when the RMS at the ground is smaller than the actual ground acceleration.

b. The following is a description of a sample calculation demonstrating the method used in determining the seismic response of a Class I building for the purpose of restraining piping.

The results of the seismic analysis conducted on the Point Beach Nuclear Power Plant for the control room building are presented herewith. This same procedure has been utilized for the purpose of providing a seismic design of other structural systems and Class I equipment.

The control room building is enclosed in the turbine building but is considered as an independent structure, since no fixed connections exist between the two buildings. Essentially the structure consists of exterior and interior concrete shear walls in both N-S and E-W direction connected by lighter concrete slabs. For the purpose of the seismic analysis a mathematical model is constructed consisting of lumped masses and stiffness coefficients. A brief sketch of the building and a superimposed outline of the model is shown on Figure A.5-3 and Figure A.5-8.

The control room building is subjected analytically to a horizontal ground acceleration of 0.06g (g = unit acceleration of gravity) for the design earthquake and a horizontal ground acceleration of 0.12g for the hypothetical earthquake. The results of the analysis are discussed in the form of internal forces and geometric behavior. The methods utilized are presented with a discussion of how the seismic analysis is conducted.

Results

A summary of the mass model values is shown in Figure A.5-8 and Figure A.5-9.

The results of analyzing the model for natural frequencies and mode shapes are presented in Figure A.5-10. The mode shapes are plotted and labeled to show how the structure vibrates at its various natural frequencies. Damping values of the various materials are also presented in this analysis.

The result of the seismic analysis due to the design earthquake are presented in Figure A.5-4 through Figure A.5-7 showing internal forces and geometric behavior. For the hypothetical earthquake, all values have to be increased by a factor of 2.0.

This analysis is provided for the E-W direction. However, the building acts not as much as a flexible structure but as a rigid body interacting with the soil. The analysis performed on the N-S direction provided results identical to the E-W. On this basis the analysis applies to both directions.

Method Of Analysis

The methods used in conducting the seismic analysis consist essentially of five steps. The first step involves the formulation of a mathematical model. The natural frequencies and mode shapes of the model are determined during the second step. Appropriate damping values are selected in the third step upon evaluation of the materials and mode shapes. The fourth step is the appropriate description of the earthquake. The response of the structure to the earthquake is determined in the fifth step.

The mathematical model of the structure is constructed in terms of lumped masses and stiffness coefficients. At appropriate locations within the building, points are chosen to lump the weights of the structure. Between these locations, properties are calculated for moments of inertia, cross-sectional areas, effective shear areas, and lengths (Figure A.5-8 and Figure A.5-9). Appropriate properties are obtained for the soil upon which the building rests. These properties are utilized to obtain soil stiffness coefficients. The properties of the model are utilized in an IBM computer program, STRESS, along with the unit loads to obtain the flexibility coefficients of the building at the mass locations.

The natural frequencies and mode shapes of the structure are obtained by Bechtel computer program, CE617 (Bechtel proprietary program). This program utilizes the flexibility coefficients and lumped weights of the model. The flexibility coefficients are formulated into a matrix and inverted to form a stiffness matrix. The program then uses the technique of diagonalization by successive rotations to obtain the natural frequencies and mode shapes. The results are shown in Figure A.5-10.

Damping values for the structural system are selected based upon evaluation of the materials and mode shapes. Appropriate damping values of individual materials are presented in Table A.5-2. Evaluation of the mode shapes makes possible the selection of damping values to be associated with each mode. Both first and second mode indicate mainly activity due to the elasticity of the underlying soil. First mode shows the soil to be contributing to a translating effect and only a little rocking of the building. The second mode indicated also translation but the amount of rocking is considerably larger. For both modes flexure of structure is negligible.

Due to this strong effect from soil elasticity and the relatively small flexibility of the structure, no proportional combining of damping values are necessary. In determining the response of the building to the earthquake the spectrum response technique is utilized. For this technique the earthquake is described by a spectrum response curve as shown in Figure A.5-1 and Figure A.5-2. Curves are provided for both the design and hypothetical earthquake. From the curves, acceleration levels are determined as associated with the natural frequency and damping value of each mode. These acceleration levels are shown on Figure A.5-1 and Figure A.5-2. The standard spectrum response technique uses these values to determine inertial forces, shears, moments, and displacements per mode. These results are then combined on the basis of the sum of the absolute values to obtain the structural response. The process is accomplished by a Bechtel computer program CE641 (Bechtel proprietary program).

A.5.5 SEISMIC DESIGN OF SERVICE WATER PIPING

(The scope of this section applies only to the service water piping in the "Pump House" extending from the discharge of the service water pumps to the point where the piping leaves the pump house building.)

A static "g" load analysis was performed in each of the two horizontal directions as well as in the vertical direction. The direction for one horizontal loading is taken normal to the axis of the most slender piping profile and the other direction is taken along that axis. In this load analysis, no restraint credit is taken for sliding supports.

For the Class I system, the boundaries of the piping system model used in the seismic analysis extends well beyond the stress analysis boundaries set by the first normally closed valve. This is done to provide confidence that the loading influence of the non-essential piping outside of (but attached to) the critical Class I portion of the system model is adequately accounted for.

At a given point, the largest stress due to one horizontal seismic loading is combined with the stress due to the vertical seismic loading to obtain the total seismic stress. All stresses include stress intensification factors as recommended by USAS B31.1.0-1967.

The static "g" factor used in the before mentioned loading analysis was obtained from the ground response spectra for the Hypothetical Basis Earthquake (0.12g) between 0.5% and 1.0% critical damping as follows:

1. The use of the ground response spectra is justified because the "Pump House" is a very rigid low profile structure. The ground response is essentially transmitted to the building contents without amplification.
2. From the before mentioned response spectra, the peak acceleration between 0.5% and 1.0% of critical damping is 0.5g. The peak acceleration occurs at a period of about 0.2 seconds.
3. The unrestrained piping system has calculated vibrational periods of approximately T=.23 seconds and T=.13 seconds. Since both of these periods are near the peak of the response curve described in sub paragraph (2), the static analysis was based on an acceleration of 0.5g. This is a conservative practice whereby any restraints that are imposed to satisfy stress requirements tend to stiffen the system so that the acceleration imposed in service will be less than the 0.5g designed for.
4. The longitudinal stress at a cross section are calculated per Paragraph 102.3.2 (d) of USAS B31.1.0-1967. The stresses are combined as follows:

   $$\sigma = S_p + S_{ew} + S_{ee}$$

   Where S_{ee} is the larger of:

   $$S_{ee} = S_{e1} + 2/3S_{e2} \text{ or } S_{e3} + 2/3S_{e2}$$

   Where:

S_P = Longitudinal stress due to internal pressure

S_{ew} = Longitudinal stress due to dead load weight

S_{e1} & S_{e3} = Longitudinal stress due to the horizontal seismic loading

S_{e2} = Longitudinal stress due to the vertical seismic loading

For the design earthquake (.06g), the values for S_{e1} , S_{e2} , and S_{e3} are taken as one half of the values used for the Hypothetical Earthquake.

The stresses calculated per the before mentioned procedure for both the Hypothetical Earthquake and Design Earthquake have been compared with the FSAR criteria and all conditions have been satisfied.

NOTE: Seismic response spectra have been developed for the Pump House. In lieu of the above static "g" load analysis, seismic analysis may be performed using the Response Spectrum Methodology, see Section A.5.7.

A.5.6 VERIFICATION OF SEISMIC ADEQUACY OF EQUIPMENT PER NRC GENERIC LETTER 87-02

A.5.6.1 Evaluation of Existing Plant Equipment

Seismic adequacy evaluation of then-existing plant equipment necessary to bring the plant to, and maintain it in, a safe shutdown condition during the first 72 hours following a safe shutdown earthquake (SSE) was performed in response to Generic Letter (GL) 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46." This was done using the SQUG "Generic Implementation Procedure (GIP) for Verification of Nuclear Plant Equipment," Revision 2. For these evaluations, safe shutdown was defined as the reactor subcritical with a minimum shutdown margin between 1% and 2.77% and the reactor coolant average temperature at or greater than 540°F. Documentation of the methodology used, equipment evaluated and the results of these evaluations are contained in Reference 18 and Reference 19 is the NRC SE of the USI A-46 implementation program.

A.5.6.2 Seismic Design and Verification of Modified, New and Replacement Equipment

Modified, new, or replacement equipment classified as Seismic Class I may be seismically designed and verified (after installation) for seismic adequacy using seismic experience data in accordance with a methodology developed by the Seismic Qualification Utility Group and approved by the NRC as documented in both of the following:

1. Seismic Qualification Utility Group (SQUG), GENERIC IMPLEMENTATION PROCEDURE (GIP) FOR SEISMIC VERIFICATION OF NUCLEAR PLANT EQUIPMENT, Revision 2, Corrected February 14, 1992; as modified by

2. U. S. 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.

The scope of equipment to which the SQUG Methodology above may be applied includes certain classes of active mechanical and electrical equipment as specified in the SQUG GIP, electrical relays, cable trays and conduit, heat exchangers, and tanks (modification of existing tanks only). As stated in SSER-2, "For new installations and newly designed anchorages in modifications or replacements, the GIP-2 criteria and procedures may also be applied, except that the factor of safety currently recommended for new nuclear power plants in determining the allowable anchorage loads shall be met."

A.5.6.3 Control Room Emergency Filtration System (CREFS)

CREFS has two parts: 1 – a portion of existing Control Room Ventilation (VNCR) and 2 - the CREFS Backup Filtration System. These two subsystems do perform a safety function and are required to meet Quality Related (QR) requirements.

VNCR - CREFS Subsystem

The AST SE (NRC Safety Evaluation, Reference 20) allows application of seismic experience data evaluations for AST only. Use of seismic experience data for Ventilation or HVAC other than AST will require NRC approval – see Reference 20, Section 2.4.2.2.

Existing, new and replacement (non-like-for-like) equipment, except as noted below in CREFS Backup Filtration System, can be qualified using one or more of the following:

1. Seismic experience data as provided for in the SQUG GIP,
   a. See FSAR Section A.5.6.1 for existing equipment and
   b. See FSAR Section A.5.6.2 for modified, new or replacement equipment,
2. EPRI Topical Report 1014608, "Seismic Evaluation Guidelines for HVAC Duct and Damper Systems, Revision to 1007896" dated December 2006, or
3. Full seismic I qualification.

CREFS Backup Filtration System

As stated in Reference 20, the CREFS Backup Filtration units along with associated ductwork and bubble tight dampers are to be installed and supported to Class I requirements, as defined in FSAR Appendix A.5. Since the CREFS Backup Filtration subsystem was added c. 2011, the supports and ductwork did not exist prior to implementation of AST. The supports were designed to meet both OBE and SSE requirements. Supports and bubble tight dampers are not qualified by use of seismic experience data. Modifications and repairs to the CREFS Backup Filtration subsystem are to be designed and installed to Class I requirements. Modifications and repairs cannot use seismic experience data as a means of seismic qualification.

A.5.7 SEISMIC ANALYSIS OF PIPING SYSTEMS

Piping may be generally classified according to the dynamic response of the system. Systems are considered rigid if they are supported and restrained in such a manner so as to cause the first mode of vibration to occur in the rigid range of the response spectrum curve. All other piping is considered flexible.

The rigid range of the response spectrum curve is defined as that portion in which there is no significant change in spectral acceleration with increasing frequencies. If piping is supported and restrained so that the first mode of vibration occurs in this range, it is classified as rigid.

Rigid piping systems are analyzed with static equivalent loads corresponding to the acceleration in the rigid range of the response spectrum curves for the applicable floor elevations. Both horizontal and vertical static equivalent loads are applied to the rigid piping systems. The amplitude of the component for the horizontal and vertical direction are combined on an absolute sum basis. The larger of the combined N-S and vertical or E-W and vertical components are used in the stress computations. The stresses are then computed in accordance with “ASME Boiler and Pressure Vessel Code Section III-Nuclear Power Plant Components, 1971,” hereafter referred to as ASME Section III. The rigid range is dependent on site seismicity and building response and as such will be determined on a case basis. The rigid range typically begins between 20 to 33 cps.

Piping that cannot be classified as rigid by the method defined above is assumed to be flexible and the analytical technique must incorporate consideration of pipe natural frequencies in addition to the fundamental frequency.

The dynamic analysis of flexible piping systems is performed using the response spectrum method. A flexible piping system is idealized as a mathematical model consisting of lumped masses connected by massless elastic members. The lumped masses are carefully located so as to adequately represent the dynamic and elastic properties of the piping system. The three-dimensional stiffness matrix of the mathematical model is determined by the direct stiffness method. Axial, shear, flexural, and torsional deformations of each member are included. For curved members, a decreased stiffness is used in accordance with ASME Section III. The mass matrix is also calculated.

After the stiffness and mass matrix of the mathematical model are calculated, the natural frequencies of piping system and corresponding mode shapes are determined using the following equation:

$$(\underline{K} \angle W_n^2 \underline{M})\underline{\phi}_n = \underline{0}$$

Where:

$\underline{K}$ = stiffness matrix
W_n = natural circular frequency for the nth mode
$\underline{M}$ = mass matrix
$\underline{\phi}_n$ = mode shape matrix for the nth mode
$\underline{0}$ = zero matrix

The Given's or the Jacobi method is used in the solution of the above equation. The mode shapes are normalized as follows:

$$\underline{\phi}_n{}^t \underline{M} \underline{\phi}_n = 1$$

A generalized mass matrix is calculated, and should correspond to:

$$\underline{\phi}^t \underline{M} \underline{\phi} = \underline{I}$$

Where:

$\underline{\phi}$ = matrix of mode shapes
$\underline{\phi}^t$ = transpose of $\underline{\phi}$
$\underline{I}$ = identity matrix.
The response spectrum method is then used to find the maximum response of each mode:

$$\underline{Y}_n(t)_{max} = \frac{\underline{\phi}_n^t \underline{M} \underline{D} S_{an}}{W_n^2 M_n}$$

Where:

S_{an} = spectral acceleration value for the nth mode
$\underline{D}$ = earthquake vector matrix, used to introduce earthquake direction to the response analysis
$\underline{\phi}_n^t$ = transpose of the nth mode shape
M_n = generalized mass of the nth mode; equals one by Equation
$\underline{Y}_n$ = generalized coordinate matrix for the nth mode.

Using the maximum generalized coordinate for each mode, the maximum displacements associated with each mode are calculate.

$$\underline{V}_n = \phi \underline{Y}_n(t)_{max}$$

The square root of the sum of the squares method is used to combine the modal responses:

$$V_i = \sqrt{V_{i1}^2 + V_{i2}^2 + \ldots V_{in}^2}$$

Where:

V_i = displacement at ith due to the response of n modes
V_{in} = displacement at ith point due to nth mode.

Once the appropriate displacements have been determined for each mass and each mode, the effective inertia forces for each mode are computed:

$$\underline{Q}_n = \underline{K} \underline{V}_n$$

Where:

$\underline{Q}_n$ = effective inertia force matrix due to nth mode

$\underline{V}_n$ = displacement matrix due to nth mode.

The effective acceleration for each mode is calculated:

$$\underline{a}_n = \underline{M}^{-1}\underline{Q}_n$$

Where:

$\underline{a}_n$ = effective acceleration matrix due to n^{th} mode
$\underline{M}^{-1}$ = the inverse of mass matrix.

After the effective inertia forces have been determined, the internal forces and moments for each mode are also calculated:

$$\underline{S}_n = \underline{b}\underline{Q}_n$$

Where:

$\underline{S}_n$ = internal force and moment matrix due to the nth mode
$\underline{b}$ = force transformation matrix

The effective inertia forces, the effective accelerations, and the internal forces and moments are combined with the square root of the sum of the squares method. For each piping system, the analysis is performed three times; once for horizontal excitation in the N-S direction, once for the E-W direction, and once for vertical excitation. Each horizontal analysis is combined with vertical analysis. The basis of combination is the square root of the sum of the squares. The maximum internal force or moment, restraining forces or moments, effective inertia force, effective acceleration, or displacement is the larger number as obtained from either of the horizontal (combined with vertical) analyses. The stresses are then computed from the internal forces and moments and are combined with other loadings (e.g., weight pressure and thermal).

A.5.8 MASONRY WALL DESIGN

NRC Bulletin No. 80-11, "Masonry Wall Design," required identifying all masonry walls in the plant which are in proximity to or have attachments from safety-related piping or equipment such that wall failure could affect a safety related system. The Bulletin also required a reevaluation of the design adequacy of these walls to determine whether they will perform their intended function under all postulated loads and load combinations.

In response to Bulletin 80-11, masonry walls that could affect safety related equipment were identified and re-evaluated using criteria submitted to the NRC by Wisconsin Electric letter dated August 14, 1981. The criteria is enclosed with the letter as Appendix B, Criteria For the Reevaluation of Concrete Masonry Walls For the Point Beach Nuclear Plant, Revision 1, August 15, 1981. The NRC accepted use of these criteria in Safety Evaluation Report, "Masonry Wall Design," transmitted to Wisconsin Electric by letter dated May 11, 1982. These criteria remain applicable to future modifications affecting masonry walls whose failure could affect a safety-related system.

A.5.9 SEISMIC ANALYSIS OF THE DIESEL GENERATOR BUILDING (DGB)

The mathematical model of the DGB consisted of several stick elements representing the reinforced concrete shear walls with nodes at each floor level. Each of these nodes was connected by rigid links, representing the rigid diaphragm action of the floor slab. The soil-structure interaction was accounted for by using six soil springs (three translations and three rotations in a Cartesian system), attached to the rigid foundation mat. The Housner horizontal design spectra with a peak ground acceleration of 0.06g for an operating basis earthquake and 0.12g for a safe shutdown earthquake were used as ground input motions. The vertical component of ground acceleration was 2/3 of the magnitude of the horizontal component. The responses (deflections, moments, shears, etc.) of the building were obtained through the response spectrum method using one set of soil spring values.

Response spectra curves for equipment located in the DGB were obtained through time history analysis. The analysis started with the design earthquake time histories input at the bottom of the mathematic model of the DGB. The time histories for the three directions of motion (two horizontal and one vertical), at each floor were then obtained as a result of the analysis. By applying these floor time histories to a single-degree-of-freedom oscillator, response spectra curves were obtained for each of the floors of the DGB. (Reference 16 and Reference 17)

A.5.10 REFERENCES

1. NRC Safety Evaluation dated September 30, 1983, Amendment No. 75 to Facility Operating License No. DPR-24.

2. WE Letter to NRC, VPNPD-91-112, “Status Update Electrical Distribution System Functional Inspection Point Beach Nuclear Plant Units 1 and 2,” dated March 28, 1991.

3. NRC Safety Evaluation Dated September 17, 1986, “Safety Evaluation of Topical Report (WCAP-10858),” “AMSAC Generic Design Package.”

4. WE Letter to NRC, “Additional Response To NRC Generic Letter 81-14,” Point Beach Nuclear Plant, Units 1 and 2, dated May 4, 1982.

5. NRC Letter, Status Report and Technical Evaluation Report, “Seismic Qualification Of The Auxiliary Feedwater System,” Point Beach Nuclear Plant Units 1 and 2, dated January 16, 1985.

6. NRC Safety Evaluation, Amendment Nos. 45/50 to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Units 1 and 2, “Low Temperature Overpressure Mitigating Systems,” dated May 20, 1980.

7. NRC Letter, “NUREG-0737 Item II.B.l, Reactor Coolant System Vents - Point Beach Nuclear Plant Units 1 And 2,” dated September 22, 1983.

8. WE Letter to NRC, “Reactor Coolant System Gas Vent System Point Beach Nuclear Plant, Units 1 and 2,” dated June 18, 1982.

9. NRC Safety Evaluation, Addendum No. 5 to the Safety Evaluation in the Matter of Point Beach Nuclear Plant Units 1 and 2, dated November 2, 1971.

10. NRC Safety Evaluation, Amendment Nos. 35/41 to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Units 1 and 2, "Modification of The Spent Fuel Storage Pool," dated April 4, 1979.

11. WE Letter to NRC, "Reactor Vessel Overpressurization," Point Beach Nuclear Plant, Units 1 and 2, dated December 20, 1976.

12. NRC Safety Evaluation, "Main Steam Line Break with Continued Feedwater Addition," Point Beach Nuclear Plant, Units 1 and 2, dated October 8, 1982.

13. WE Letter to NRC, "Final Resolution of Generic Letter 81-14 Seismic Qualification of Auxiliary Feedwater System," Point Beach Nuclear Plant, Units 1 And 2, dated April 26, 1985.

14. NRC Safety Evaluation, "Seismic Qualification of the Auxiliary Feedwater System," Point Beach Nuclear Plant Units 1 And 2, dated September 16, 1986.

15. WE Letter to NRC, "Seismic Qualification of the Auxiliary Feedwater System," Point Beach Nuclear Plant Units 1 and 2, dated December 15, 1982.

16. VPNPD-93-171, "Design Summary for the Installation of Two additional Emergency Diesel Generators - Point Beach Nuclear Plants, Unit 1 and 2," dated September 24, 1993 and attached Report REP-0026, "PBNP Diesel Project Design Submittal," Revision 0, dated September 21, 1993.

17. NRC Safety Evaluation 94-003, "Emergency Diesel Generator Addition Project, Point Beach Nuclear Plant," October 24, 1994.

18. US NRC Generic Letter 87-02, USI A-46 Resolution, Seismic Evaluation Report, Revision 1, dated January 1996.

19. NRC SE, "Response to Supplement No. 1 to Generic Letter 87-02 for the Point Beach Nuclear Plant, Units 1 and 2," dated July 7, 1998.

20. US NRC SE, "Amendment No. 240 to Renewed Facility Operating License No. DPR-24 and Amendment No. 244 to Renewed Facility Operating License No. DPR-27, NextEra Energy Point Beach, LLC, Point Beach Nuclear Plant, Units 1 and 2, Docket Nos. 50-266 and 50-301," dated April 14, 2011.

Table A.5-1 INTERNALS DEFLECTIONS UNDER ABNORMAL OPERATION (INCHES)

	Calculated Deflection (Preliminary)	Allowable Limit	No Loss-of-Function Limit
Upper Barrel expansion/compression (to assume sufficient inlet flow area/and to prevent the barrel from touching any guide tube to avoid disturbing the RCC guide structure).	0.072	3	6
Upper Package axial deflection (to maintain the control rod guide structure geometry).	0.005	1	2
RCC Guide Tube cross section distortion (to avoid interference between the RCC elements and the guides.)	0	0.035	0.072
RCC Guide Tube deflection as a beam (to be consistent with conditions under which ability to trip has been tested).	0.2	1.0	1.5
Fuel Assembly Thimbles cross section distortion (to avoid interference between the control rods and the guides).	0	0.035	0.072

Table A.5-2 DAMPING FACTORS

Type of Condition and Structure	Design Earthquake	Hypothetical Earthquake
Welded Steel Plate Assemblies	1%	2%
Welded Steel Framed Structures	2%	2%
Bolted Steel Framed Structures	2.5%	5%
Interior Concrete Equip. Supports	2%	2%
Reinforced Concrete Structures on Soil	5%	7.5%
Prestressed Concrete Containment Structure on Piles	2%	5%
Vital Piping Systems*	0.5%	0.5%
Soil Damping	5%	5%
Verification of Electrical and Mechanical Equipment and Anchorage**		5%
Verification of Vertical Welded Steel Tanks**		4%

* For the Unit 1 main steam line outside of containment, with a support configuration that includes energy absorbers, the damping factors range from 0.5% - 4.3% for the design earthquake, and from 0.5% - 17% for the hypothetical earthquake. For the Unit 2 main steam line outside of containment, with a support configuration that includes energy absorbers, the damping factors range from 0.5% - 3.6% for the design earthquake, and from 0.5% - 20% for the hypothetical earthquake.

** Refer to Section A.5.6

Table A.5-3 LOADING CONDITIONS AND STRESS LIMITS
Sheet 1 of 4

Definitions[1]

1. Normal Conditions: Any condition in the course of system start-up, operation in the design power range, and system shutdown, in the absence of Upset, Emergency, or Faulted Conditions.

2. Upset Conditions: Any deviations from Normal Conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The Upset Conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, transients due to loss of load or power, and any system upset not resulting in a forced outage. The estimated duration of an Upset Condition shall be included in the Design Specifications -- The Upset Conditions include the effect of the specified earthquake for which the system must remain operational or must regain its operational status.

3. Emergency Conditions: Any deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. The total number of postulated occurrences for such events shall not exceed twenty-five (25).

4. Faulted Conditions: Those combinations of conditions associated with extremely low probability postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent where considerations of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

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1. Summer 1968 Addenda to the ASME B&PV Code, Section III.

Table A.5-3 LOADING CONDITIONS AND STRESS LIMITS
Sheet 2 of 4

Among the Faulted Conditions may be a specified earthquake for which safe shutdown is required.

For loading combination 1, the seismic loading was considered as a loading of short duration and the allowable stress limit increased by 20%.

For loading combination 2 (normal plus hypothetical earthquake loads) and 3, (normal plus pipe rupture loads) the rules used for nuclear vessels in ASME Section III were adopted as a guide. However, with this amendment, loading combinations 2 and 3 have been revised in order to conform to present design definitions and are now under category 4, i.e., faulted conditions.

For loading combinations for vessel supports, the stress limits are shown. The calculated stresses do not exceed these limits. The supports do not impose loadings on the vessel wall which exceed paragraph N-473 of ASME Section III. According to ASME Section III Code, the allowables for pressure retaining components are specified as follows:

	Primary Stress Intensity	
Loading Conditions	General Membrane (P_m)	Local Membrane (P_L) Plus Bending (P_b)
1. Normal Conditions	$P_m < S_m$	$P_L + P_b < 1.5S_m$
2. Upset Conditions (Normal + OBE)	$P_m < S_m$	$P_L + P_b < 1.5S_m$
3. Emergency Conditions	$P_m < 1.2S_m$	$P_L + P_b < 1.8S_m$ (or $2.25S_m$ for piping)
4. Faulted Conditions (Normal + DBE, Normal + DBA, Normal + DBE + DBA') See Note 5	N/A	$P_L + P_b < 1.8S_m$ (or $3.0S_m$ for piping)

where:

Pm	= primary general membrane stress intensity	DBE	= Hypothetical Earthquake
P_L	= primary local membrane stress intensity	OBE	= Design Earthquake
P_b	= primary bending stress intensity	DBA	= Design Basis Accident
Q	= Secondary stress intensity	DBA'	= Steady-state Portion of Design Basis Accident
S_m	= stress intensity value from ASME B&PV Code, Section, III, Nuclear Vessels.		
S_y	= minimum specified material yield strength		

Table A.5-3 LOADING CONDITIONS AND STRESS LIMITS
Sheet 3 of 4

According to ASME Section III, Subsection NF component supports, the allowables for component supports are specified as follows:

Loading Conditions	Primary Stress Intensity General Membrane (P_m)	Primary Stress Intensity General Membrane (P_m) Plus Bending (P_b)
1. Normal Conditions	$P_m < S_m$	$P_m + P_b < 1.5S_m$
2. Upset Conditions (Normal + OBE)	N/A	N/A
3. Emergency Conditions	$P_m < 1.2S_m$	$P_m + P_b < 1.8S_m$
4. Faulted Conditions (Normal + DBE, Normal + DBA, Normal + DBE + DBA')	$P_m < 1.5S_m$	N/A

where:

DBE = Hypothetical Earthquake
OBE = Design Earthquake
DBA = Design Basis Accident
DBA' = Steady-state Portion of Design Basis Accident

Material	F (y) (KSI)	Stress Limits	F (B) (KSI)	F (T) (KSI)	F (V) (KSI)
ASTM - A36	36	Working	24	22	14.5
		Yield	32.4	32.4	19.4
ASTM - A514		Working	66	60	40
ASTM - A517	100	Yield	90	90	54
ASTM - A490	125	Working	82.5	75	32
		Yield	112.5	112.5	67.5

Working stress limits correspond to loading conditions 1 & 2.
Yield stress limits correspond to loading conditions 3 & 4.

F (y) = Yield Stress
F (B) = Bending Stress
F (T) = Tensile Stress
F (V) = Shear Stress

Table A.5-3 LOADING CONDITIONS AND STRESS LIMITS

Sheet 4 of 4

NOTES FOR TABLE Table A.5-3

NOTE: 1 The limits on local membrane stress intensity ($P_L \leq 1.5S_m$) and primary membrane plus primary bending stress intensity (P_m(or P_L) $+P_b \leq 1.5S_m$) 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 ($P_L \leq 1.5S_m$) or the primary plus secondary stress intensity ($P_L + P_b + Q \leq 3S_m$) 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 which occur 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 ($P_L \leq 1.5S_m$) and primary membrane plus primary bending stress intensity (P_m (or P_L) $+ P_b \leq 1.5S_m$) need not be satisfied at a specific location if it can be shown by means of limit analysis or by test that the specified loadings do not exceed 120% of 2/3 of the lower bound collapse load, as per paragraph N-417.10(c) of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

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 the yield strength adjusted to correspond to the tabulated value at the appropriate temperature in Table N-424 or N-425, as per paragraph N-417.11c of the ASME B&PV Code, Section III, Nuclear Vessels. These specific cases will be justified on an individual basis.

NOTE: 5 The Faulted Condition load combination for the replacement reactor vessel closure heads and CRDM pressure housings consists of Normal + SRSS (DBE + DBA), where SRSS refers to the square-root-of-the sum-of-squares load combination methodology.

Figure A.5-1 EARTHQUAKE RESPONSE SPECTRUM - .06g

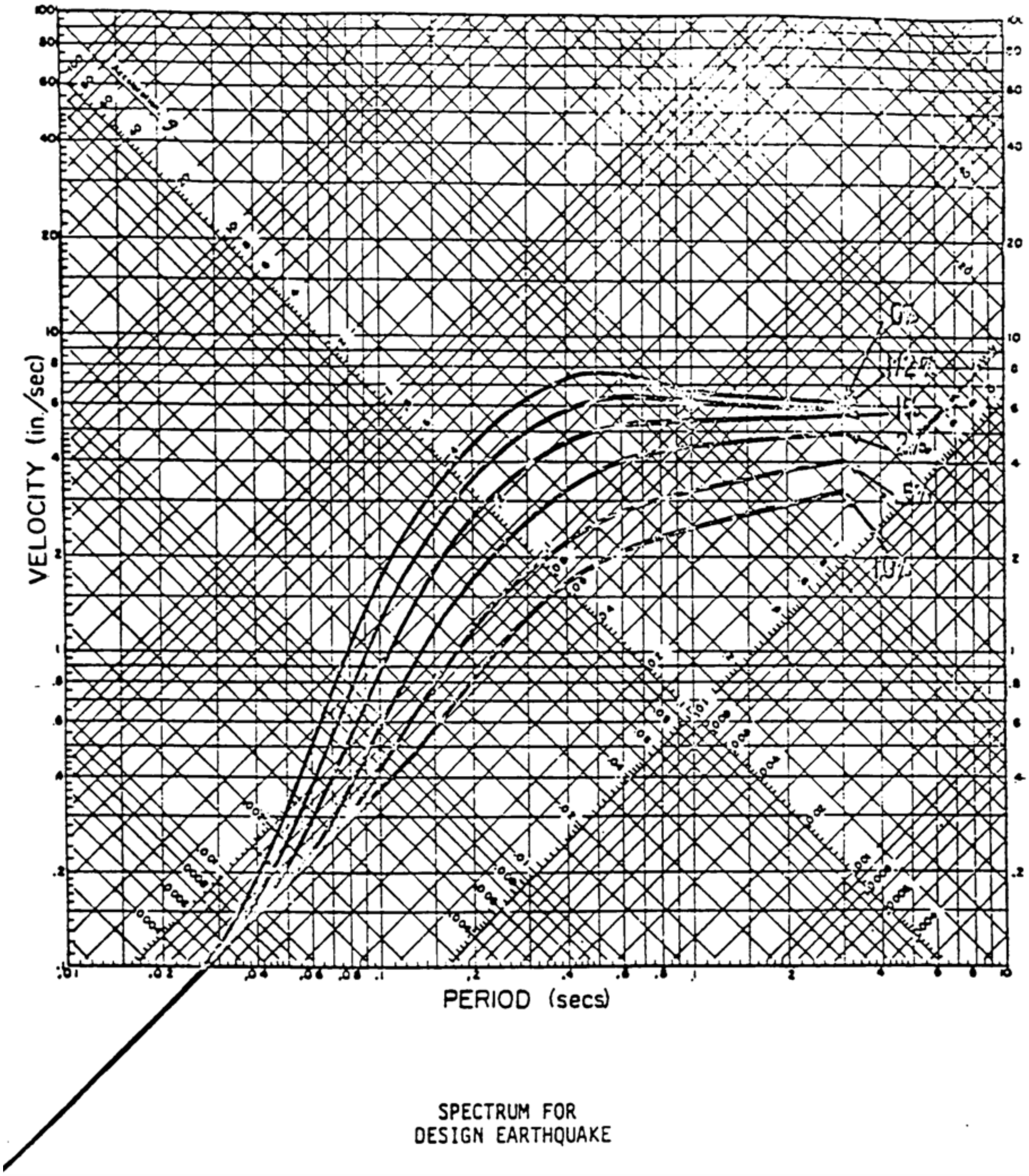

Figure A.5-2 EARTHQUAKE RESPONSE SPECTRUM - 0.12g

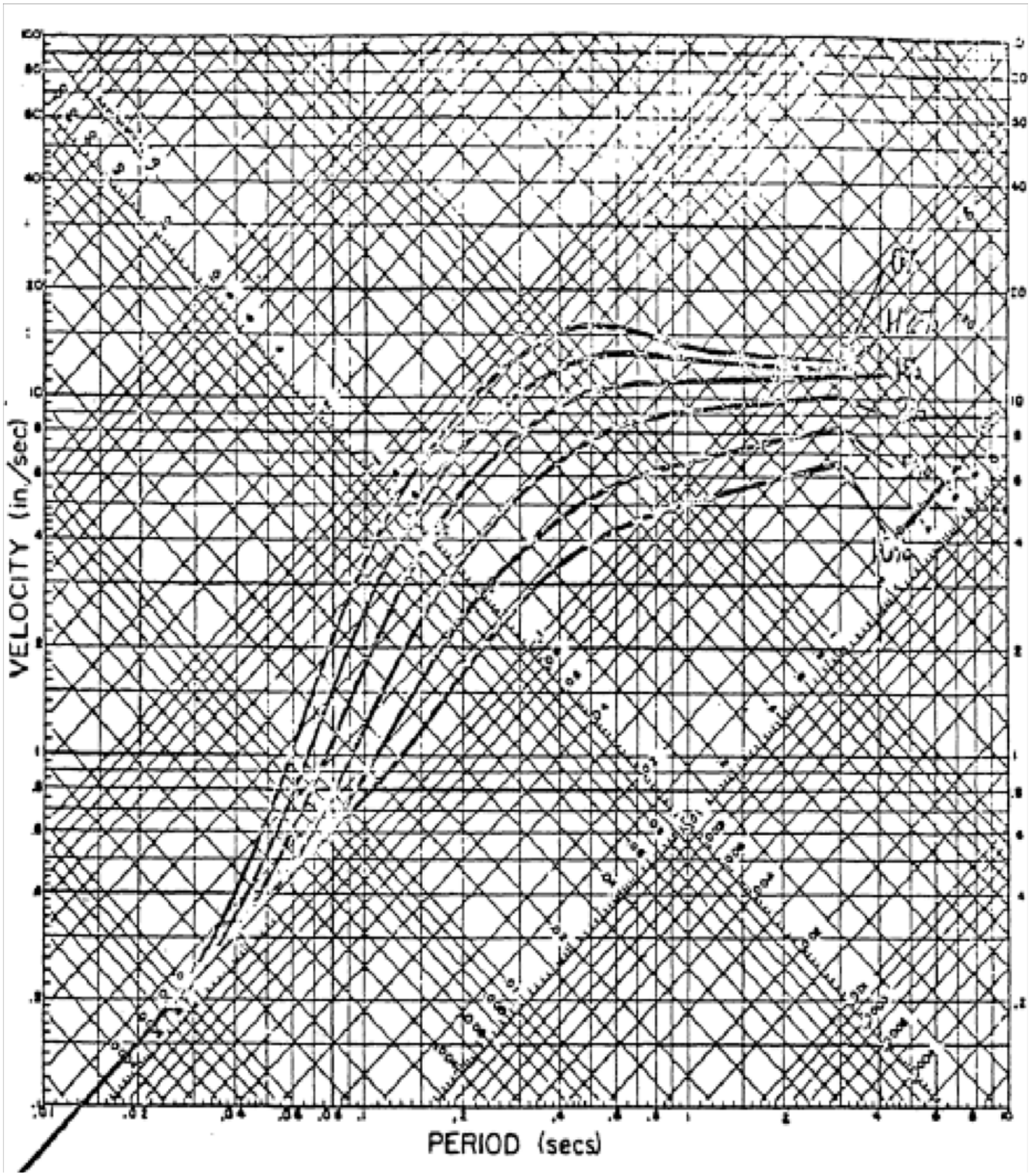

Figure A.5-3 CONTROL ROOM BUILDING SECTION, N-S DIRECTION

[redacted]

Figure A.5-4 CONTROL ROOM BUILDING BENDING MOMENT - HEIGHT

[redacted]

Figure A.5-5 CONTROL ROOM BUILDING SHEAR - HEIGHT

[redacted]

Figure A.5-6 CONTROL ROOM BUILDING - ACCELERATION ENVELOPE

[redacted]

Figure A.5-7 CONTROL ROOM BUILDING - DISPLACEMENT ENVELOPE

[redacted]

Figure A.5-8 CONTROL ROOM - MODEL FOR STRESS

[redacted]

Figure A.5-9 CONTROL ROOM - PROPERTIES OF LUMP MASSES AND CONNECTING MEMBERS

Mass #	Weight Kips	Member #	Shear Area Ft^2	Total Area Ft^2	I_2 $Ft^4 \times 10^3$
1	3726.0				
		1	427.50	921.20	820.2
2	4700.0				
		2	237.0	362.25	326.2
3	2742.0				
		3	223.20	341.70	323.3
4	1670.0				
		4	210.75	468.15	343.5
5	1331.0				

Figure A.5-10 CONTROL ROOM BUILDING - MODE SHAPES AND FREQUENCIES

[redacted]

A.6 SHARED SYSTEMS ANALYSIS

Certain components of plant systems are shared by the two units as stated in Section 1.2.9. The purpose of this Appendix is to present a failure analysis of shared components (Table A.6-1) to demonstrate that the following GDC is met:

> Sharing of Systems
> Reactor facilities may share systems or components if it can be shown that such sharing will not result in undue risk to the health and safety of the public. (GDC 4)

In addition to listing shared components by system, Table A.6-1 also includes the corresponding equipment functions which are shared between the two units, each unit's condition that places the maximum demand on the shared system/component, and the ability of the shared system to tolerate the failure of any single active component without loss of the shared function.

Table A.6-1 SHARED SYSTEMS ANALYSIS

Shared System	Shared Components	Shared Fuction	Quantity Provided	Explanation	Conditions of Maximum Demand on the System	Quantity req'd for Maximum Demand	Able to tolerate the single failure of an active component
Chemical and Volume Control System	Boric Acid Storage Tanks	Storage of boric acid for refueling and emergency shutdown	3	Three tanks are provided such that all the boric acid required during the operating cycles of both units may be stored in them. Depending on boron concentration, either one or two tanks are required to shutdown a unit to cold xenon-free concentration, and assuming the most reactive rod fully withdrawn	Simultaneous shutdown of both units	1	No (See Note 1)
	Batching Tank	Makeup of fresh concentrated boric acid solution	1	One tank is provided for the two units. It is used infrequently after initial boration.	N/A	N/A	N/A
	Holdup Tanks	Storage of dilute boric acid prior to recycle processing	3	Three tanks are provided to handle the rejected chemical shim solution from all expected operating and start-up transients for two unit plant operation.	N/A	N/A	N/A
	Recirculation Pump	Handling of tank inventory	1	Serves the common hold up tanks infrequently	N/A	N/A	N/A
	Gas Stripper Feed Pumps	Pumping of chemical shim solution to gas stripper/boric acid evaporator processing train	2	Two pumps are provided each with sufficient capacity to supply both processing trains simultaneously. One pump serves as a spare to the other.	N/A	N/A	N/A
	Evaporator Feed Ion Exchangers	Remove cesium and lithium activity from boric acid to be processed for reuse.	4	Four vessels are provided. Two vessels in series have sufficient capacity to supply both processing trains simultaneously. Two resin beds serve as a spare to the other two.	N/A	N/A	N/A

Table A.6-1 SHARED SYSTEMS ANALYSIS

Shared System	Shared Components	Shared Fuction	Quantity Provided	Explanation	Conditions of Maximum Demand on the System	Quantity req'd for Maximum Demand	Able to tolerate the single failure of an active component
	Gas Strip per Boric Acid Evaporator Train	Processing used chemical shim solution to produce clean, re-usable reactor makeup water and concentrated boric acid solution.	2	Two processing trains serve as common equipment for the two units. One train serves as a spare to the other although both may be operated simultaneously.	N/A	N/A	N/A
	Monitor Tanks	Reservoirs for processed water for analysis prior to storage in reactor makeup water tank.	4	Four tanks are provided to permit continuous operation of each evaporator train and so that one may be filling while the other is examined and emptied in each train.	N/A	N/A	N/A

Table A.6-1 SHARED SYSTEMS ANALYSIS

Shared System	Shared Components	Shared Fuction	Quantity Provided	Explanation	Conditions of Maximum Demand on the System	Quantity req'd for Maximum Demand	Able to tolerate the single failure of an active component
	Monitor Tank Pumps	Pump water from the monitor tanks to the reactor makeup water tank	2	Two pumps are provided, each with adequate capacity to handle both units. One pump serves as a spare to the other.	N/A	N/A	N/A
	Evaporator Condensate Demineralizers	Remove trace amounts of boric acid from processed water	3	Three demineralizers are provided each with sufficient capacity to serve both units. Thus adequate spare capacity is provided.	N/A	N/A	N/A
	Reactor Makeup Water Tank	Storage of clean makeup water	1	One tank is provided which is adequately sized to serve both units.	N/A	N/A	N/A
	Reactor Makeup Water Tank Pumps	Supply miscellaneous reactor makeup	2	Two pumps are provided, each with sufficient capacity to serve needs of the two units. One pump serves as a spare to the other.	N/A	N/A	N/A
	Concentrates Holding Tank	Storage of boric acid evaporator bottoms for sampling	1	One tank holds the production of concentrates from one batch of evaporator operation.	N/A	N/A	N/A
	Concentrates Holding tank Transfer Pumps	Discharge of boric acid solution from concentrates holding tank.	2	Two pumps provided to service the common concentrates holding tank.	N/A	N/A	N/A

Table A.6-1 SHARED SYSTEMS ANALYSIS

Shared System	Shared Components	Shared Fuction	Quantity Provided	Explanation	Conditions of Maximum Demand on the System	Quantity req'd for Maximum Demand	Able to tolerate the single failure of an active component
Auxiliary Coolant System	Component Cooling Heat Exchangers	Intermediate heat exchanger between service water and component cooling water.	4	Four exchangers are provided to serve both units. Normally one exchanger will provide adequate cooling for each unit. Two exchangers serve as spare units. The spare exchangers may be utilized to speed the shutdown of either unit as required. See Note 4	Simultaneous initiation of shutdown cooling at RHR cut-in conditions (350°F) on both units	2	yes
	Component Cooling Water Pumps	Circulate component cooling water for miscellaneous services in both units.	4	Four pumps are provided to serve both units. Normally one pump will provide adequate circulation to cool each unit, with the other pump assigned to that unit serving as a standby spare. The spare pumps may be used to speed the shutdown of either unit as required.	Simultaneous initiation of shutdown cooling at RHR cut-in conditions (350°F) on both units	2	yes
	Spent Fuel Pool Pumps	Recirculation of spent fuel pool water	2	Two pumps are provided to service the common spent fuel pool.	See Note 2	2	N/A
	Spent Fuel Pool Demineralizer	Purification of the spent fuel pool water and refueling water	1	One demineralizer is provided. It is operated intermittently and may be bypassed when the resin is replaced.	See Note 2	N/A	N/A

Table A.6-1 SHARED SYSTEMS ANALYSIS

Shared System	Shared Components	Shared Fuction	Quantity Provided	Explanation	Conditions of Maximum Demand on the System	Quantity req'd for Maximum Demand	Able to tolerate the single failure of an active component
	Spent Fuel Pool Filter	Purification of the spent fuel pool water and refueling water	1	One filter is provided. The purification loop is by-passed when the cartridge is replaced.	See Note 2	N/A	N/A
	Spent Fuel Pool Heat Exchanger	Cooling Spent Fuel Pool Water	2	Sufficient capacity is provided to maintain reasonable pool temperatures	See Note 2	2	N/A
	Refueling Water Circulating Pump	Circulation of refueling water if required for purification	1	One pump provides in frequent purification service for both refueling water tanks.	N/A	N/A	N/A
Fuel Handling System	Spent Fuel Storage Pool	Storage of spent fuel elements from refueling until shipment	1	A common area is provided with adequate rack storage space to meet the requirements of two units.	See Note 2	N/A	N/A
	New Fuel Storage	Storage of new fuel elements from delivery until loading into the reactors.	1	A common area with new fuel storage rack is provided with adequate space to serve both units.	N/A	N/A	N/A
	Decontamination Area	Easily cleaned area for decontamination of equipment.	1	A common area is provided with adequate space to serve both units.	N/A	N/A	N/A
	Spent Fuel Pool Bridge	Transfer of fuel elements between storage and fuel transfer system.	1	A common bridge is provided serving the common spent fuel pool.	N/A	N/A	N/A

Table A.6-1 SHARED SYSTEMS ANALYSIS

Shared System	Shared Components	Shared Fuction	Quantity Provided	Explanation	Conditions of Maximum Demand on the System	Quantity req'd for Maximum Demand	Able to tolerate the single failure of an active component
Service Water System	Pumphouse and Headers	Environment for service water pumping equipment.	1	A common pumphouse is provided for the service water pumping equipment.	The recirculation phase of the post-LOCA condition in one unit with normal power operation in the second unit.		See Service Water Pumps
	Service Water Pumps	Provide cooling water water for various common and Unit specific loads as described in Section 9.6.	6	Six service water pumps are provided to supply water to the dual, common loop piped system for the two units. Normally, pumps will supply both units; the additional pumps provide increased capacity when required and serve as spares.	The recirculation phase of the post-LOCA condition in one unit with normal power operation in the second unit.	3	yes
Electrical System	Diesel Generators	Supply emergency power in the event of a loss of the AC power supply.	4	Four Diesel generators are supplied as common to both units. Each will have adequate capacity to safely control a LOCA in one unit and a concurrent trip of the second unit to the hot shutdown condition (See Note 3).	LOCA in one unit with concurrent trip of the second unit (to the hot shutdown condition) when all AC power supply is simultaneously lost.	1	yes
	Gas Turbine	Supply power during a blackout and certain App. R fire scenarios	1	One gas turbine unit is supplied in the event of a blackout, to supply spinning reserve and for peaking purposes.	N/A	N/A	N/A

Waste Disposal	A common waste disposal system is to be used for the two units. Each containment structure has its own reactor coolant drain tank, and containment sump, and each is serviced by two reactor coolant drain tank pumps. All other waste disposal equipment is sized to adequately serve two units and the common auxiliary and service buildings. This shared equipment includes:			The Waste Disposal System serves no emergency function.
	Laundry and Hot Shower Tank Chemical Drain Tank Sump Tank Waste Hold-up Tank	Waste Gas Compressors Waste Evaporator Train Drumming Station Gas Analyzer	Boric Acid Evaporator Letdown Gas Stripper Cryogenic System Waste Condensate Tanks	Waste Distillate Tanks Blowdown Evaporator Gas Decay Tanks Gas Manifolds Filtration/Demineralization System

1 Boric acid injection affords backup reactivity shutdown capability, independent of control rod cluster which normally serve this function in the short term situation. Normally, boric acid injection is only used either to supplement rod control for xenon decay or for reactor cooldown. At the lower allowed acid concentrations, one full storage tank will not be sufficient to achieve the required shut down margin. Additional boric acid solution from a second tank will be required. However, sufficient storage exists in all three tanks to support shutdown of both units.

2 Operation of the Spent Fuel Handling System is only required when nuclear fuel is to be moved underwater. The spent fuel pool cooling system is designed for a heat load greater than that generated by a complete core offload with about 1381 assemblies already in the pool.

3 See Section 8.0.

4 License condition for amendment 178 requires that each unit will utilize only one CCW heat exchanger until such time that analyses are completed and the SW system is reconfigured as necessary to allow operation of one or both units with two heat exchangers in service.

A.7 PLANT FLOODING

The Point Beach Site selection is inherently resistant to external flooding risks as discussed in Section 2.5 "Hydrology." The plant design and equipment layout provide additional protection from postulated internal and external flooding sources. This section provides information on affected systems, critical equipment heights and protective strategies for addressing internal and external plant flooding.

A.7.1 AFFECTED SYSTEMS AND PROTECTION METHODS

Systems and components that must be protected from external flooding were specified in the original plant Safety Evaluation Report (SER) (Reference 22) as "critical plant components."

The Point Beach internal flooding basis was initiated by a 1972 Atomic Energy Commission (AEC) communication and specifies that no failure of a non-Category I (seismic) component can result in a flooding condition that could adversely affect equipment needed to get the plant to safe shutdown or to limit the consequences of an accident (Reference 1) (Reference 3) (Reference 8).

Protection Methods

Acceptable methods for providing flood protection for plant systems and equipment are diverse. The basic strategies include:

Equipment Height:

- If the elevation of the potentially vulnerable equipment exceeds the design basis flood level for the affected room, then adequate protection exists. (Reference 8) (Reference 22)

Topography

- Lake bottom contour, construction of the bank, and distance from shore can be credited for mitigating the effects of wave run-up events. Property slope can be credited for mitigating the effects of precipitation events and for providing a relief path for internal flooding.

Barriers:

- Interior or exterior barriers that protect vulnerable equipment from the effects of flooding can be used to provide adequate protection from flooding from internal (Reference 3) or external sources. The use of sandbags is an acceptable option to provide for the protection of plant equipment from internal or external flood sources at Point Beach (Reference 24) (Reference 25).

Separation:

- Separation and redundancy of trains or equipment is sufficient if both trains, or redundant equipment, cannot be impacted by the same flooding event. This provides acceptable protection from internal flood sources only (Reference 3) (Reference 26).

Operator Actions:

- Operator actions as specified in plant procedures can be credited in response to both internal and external flooding sources (Reference 2) (Reference 4) (Reference 8) (Reference 24) (Reference 25).

Detection:

- Water level alarms can be credited if they process an alarm to the Control Room and are redundant (Reference 3) (Reference 51).

Relief Paths:

- Passages or piping and other openings may be credited as a relief path if they are designed for the safe shutdown earthquake (SSE), including seismically induced wave action of water inside the affected compartment during the SSE (Reference 3).

A.7.2 EXTERNAL FLOODING

The bounding external flooding event can be either a maximum wave run-up or a maximum precipitation event. The site topography and hydrology, as discussed in Section 2.5, both serve to minimize potential flooding vulnerability.

Flooding Conveyance Paths

The site topography provides sufficient drainage capacity and conveyance to the lake to address potential impacts from a design basis precipitation event (see FSAR Section 2.5.2 "Lake Levels and Flooding").

The site layout, consisting of the intake structure and rip-rap bank topography, are credited in the flooding evaluations, which demonstrate that the calculated flood level is bounded by the license basis flood level of +9.0 feet. Protection to +9.0 feet is provided by procedurally driven installation of temporary barriers at entrances to the CWPH and Turbine Building. When lake level exceeds administratively controlled limits, both units are brought to cold shutdown and barriers are installed. The Circulating Water, Condensate and Feedwater Systems are secured prior to installation of barriers at the Turbine Building doors/flood dampers in order to eliminate the major sources of internal flooding while the Turbine Building relief paths are blocked.

There are no openings in the pumphouse walls, other than tight fitting doors, that have a bottom elevation less than +9.0 feet (Reference 25). In addition, storm drains are provided outside each of the pump house doors.

The turbine building, which is the structure next closest to the lake, is more than 100 feet from the top of the bank. The combination of this distance, the shoreline riprap and the storm drains mitigate lake effect flooding.

Table A.7-2 lists the design features credited for mitigating external plant flooding.

Underground Conduits and Trenches

The external manhole and cable trenches are designed to remove water through a cascade system utilizing a combination of gravity drains and pumps. The manholes are monitored through the Facilities Monitoring Program and the Cable Condition Monitoring Program.

A.7.3 INTERNAL FLOODING

Design Basis Flood Level

Design Basis Flood Levels for internal flooding sources are based upon protecting designated equipment needed to get the plant to safe shutdown (Reference 1), (Reference 3), (Reference 8), (Reference 50), and (Reference 52) and are therefore unique to each room and the associated limiting flood source. Timelines for system/operator response are based on a mass/flow balance that determines equilibrium flood levels (Reference 28).

The consequences of a single failure of any non-Category I (Seismic) SSC that has the potential to cause flooding have been evaluated for all areas where such a failure could have an impact on safe shutdown equipment (Reference 28).

Loss of Offsite Power (LOOP)

A loss of offsite power (LOOP) is assumed to occur during internal flood events unless the LOOP results in a less limiting consequence (Reference 3). Design features that rely on electric power to operate (such as sump pumps) can only be credited for flood protection if they can be powered by site emergency power sources.

Separation and Redundancy

For internal flooding, separation (including barriers) and redundancy of trains or equipment is sufficient to mitigate the effects of a flood if both trains, or redundant equipment, cannot be impacted by the same flooding event. However, the features listed in Table A.7-1 are still included in the license basis.

[redacted]

Internal Flooding Sources

The Point Beach internal flooding license basis as established by the Safety Evaluation Report (SER) dated November 20, 1975 did not require Category I (Seismic) SSCs to be postulated as internal flooding sources (Reference 1) (Reference 8). In addition, components which can withstand a SSE are not postulated as flood sources.

The main condensers in the turbine hall were evaluated and determined not to fail during a SSE (Reference 48) (Reference 49). The evaluations address:

- the condenser and its anchorage,
- the thermal and seismic movements of the upper flange of the circulating water pipe at the expansion joint,
- pipe alignment being retained following a full circumference expansion joint failure,
- seismic integrity of CW piping entering and exiting the condenser.

Evaluation (Reference 28) has demonstrated that, under both normal operating conditions and while the internal flood drain paths are blocked by the External Wave Run up Flood Mitigation Strategy, there is sufficient time available to eliminate the source of internal flooding prior to impacting safety related/safe shutdown equipment.

Internal flooding from a failure of the fire protection piping in the G03/G04 Emergency Diesel Generator Building is a through-wall leakage crack, sized in accordance with Branch Technical Position (BTP) MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment" (Reference 47).

Table A.7-1 lists the applicable rooms, the flood sources, and the design features credited for mitigation of plant internal flooding.

Table A.7-1 DESIGN FEATURES CREDITED FOR MITIGATING INTERNAL PLANT FLOODING Page 1 of 3

Room Description	Design Feature
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]

Table A.7-1 DESIGN FEATURES CREDITED FOR MITIGATING INTERNAL PLANT FLOODING   Page 2 of 3

[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]

Table A.7-1 DESIGN FEATURES CREDITED FOR MITIGATING INTERNAL PLANT FLOODING Page 3 of 3

Residual Heat [redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]

Table A.7-2 DESIGN FEATURES CREDITED FOR MITIGATING EXTERNAL PLANT FLOODING Page 1 of 2

Room Description	Design Feature (Reference 46)
[redacted]	[redacted]
[redacted]	[redacted]

Table A.7-2 DESIGN FEATURES CREDITED FOR MITIGATING EXTERNAL PLANT FLOODING Page 2 of 2

[redacted]	[redacted]

REFERENCES

1. AEC letter to WE, Request for Review of Non-Category I (Seismic) Systems as Possible Flood Sources, dated September 26, 1972.

2. WE letter to NRC, Flooding Resulting from Non-Category I Failure Point Beach Nuclear Plant - Units 1 and 2, dated February 20, 1973.

3. AEC letter to WE, Supplemental Request for Flooding Analysis of Non-Category I System Sources, dated December 10, 1974.

4. WE letter to NRC, Flooding Resulting from Non-Category I Failure Point Beach Nuclear Plant - Units 1 and 2, dated February 17, 1975.

5. WE letter to NRC, Flooding Resulting from Failures of Non-Category I Equipment Point Beach Nuclear Plant - Units 1 and 2, date April 28, 1975.

6. NRC letter to WE, Request for Additional Information Flooding of Equipment Important to Safety of Point Beach, Units 1 and 2, dated September 29, 1975.

7. WE letter to NRC, Potential for Flooding of Safety Related Equipment Point Beach Nuclear Plant Unit Nos. 1 and 2, dated October 24, 1975.

8. NRC letter to WE, Safety Evaluation Regarding the Potential for Flooding from Postulated Ruptures of Non-Category I (Seismic) Systems, dated November 20, 1975 and associated Safety Evaluation.

9. WE letter to NRC, Flooding Resulting from Non-Category I Failures Point Beach Nuclear Plants Units 1 and 2, dated May 26, 1976.

10. SCR 2014-0127, “Seal Manhole and Cleanouts in the Subsoil Drain System,” EC-281710, “Seal Manhole and Cleanouts in the Subsoil Drain System.

11. SCR 2009-0057, USAR 01141895 change to FSAR A.7, “Plant Internal Flooding,” dated May 5, 2009.

12. WE letter to NRC, “Response to Generic Letter 81-14,” dated October 31, 1986.

13. Evaluation 2009-003, USAR 01141895 changes to FSAR A.7, “Plant Internal Flooding,” dated May 21, 2009.

14. SCR 2007-150-01, “MOD EC 11174 CWPH Flood Relief Modification,” dated September 29, 2009.

15. EC 11416 Revision 1, “Façade Flooding and Tendon Gallery Grout Removal.”

16. SER 88-089-10, EOP-1.3 Transfer to Containment Sump Recirculation, Revision 10, March 12, 1992.

17. SCR 2013-0124, “Remove Auxiliary Feedwater Room Continuous Flood Watch from CLB,” dated July 23, 2013.

18. EC 279915, Revision 0, “Issue Calculation PBNP-26256-026-S01, Evaluation of Block Wall West of CST’s, CMU 68-3/19.”

19. EC 271960, Revision 1, “Reinforce Masonary Wall on the East Side of the Non-vital Switchgear Room, Reference Work Order 40081601.”

20. EC 280179, “Define ‘Waterproof’ as used in FSAR A.7.”

21. SCR 2013-0093, “Shim F-16 (Cont. Rm. Charcoal Filter) Drain Line to Provide HVAC Equipment Room Drainage.”

22. NRC Safety Evaluation dated July 15, 1970.

23. NUREG-0800 sections 2.4.1 and 3.4.1.

24. NUREG/CR-4458 Section 3.4.2 “External Flood Vulnerability.”

25. Original Plant FFDSAR.

26. NUREG/CR-4458 section 3.3.2 “Internal Flooding Vulnerability.”

27. Calculation FPL-076-CALC-014, “Precipitation and Snow Intensity Determination and Roof Drainage,” dated December 18, 2013.

28. Calculation 2014-0007, “Allowable Flood Levels,” dated October 31, 2014.

29. EC 282741, Revision 0, “Strengthen NVSGR North Wall.”

30. Calculation 91C2696-C-014, “USI A-46/IPEEE, Equipment Fragilities for T-24A and T-24B” Stevenson & Associates report dated November 5, 2002.

31. Calculation 11Q0975-C-002, “Seismic Assessment of Waste Distillate Tanks and Supporting Frame,” Revision 1 dated May 21, 2012.

32. Not Used

33. Calculation 11Q0975-C-001, “Seismic Assessment of Monitor Tanks T10A, B, C, D” Stevenson & Associates dated May 2, 2011.

34. PBNP Letter to NRC dated June 20, 1977 section 5.1.4, “Fire System Failure Effects on Safety Related Equipment.

35. NRC Safety Evaluation Report dated August 2, 1979.

36. Not Used

37. Calculation 2014-02058, “Design Basis Turbine Hall and PAB Internal Flooding,” dated October 2, 2014.

38. Calculation 2009-0008, “Circulating Water Pump House Internal/External Flooding,” Revision 2 dated October 2, 2014.

39. EC 282054, “Exterior Barrier Strategy,” Revision 1 dated October 25, 2014 (Proprietary).

40. Calculation 2014-0002, “Effects on Safety Equipment of Bypassing the Installed Wave Run-up Barriers,” Revision 1 dated August 11, 2014.

41. Not Used

42. Not Used

43. EC 281314, “Sealing of Unnumbered Penetrations Between the 5’ el. Trench in the Central PAB and Pipeways 2 and 3.”

44. EC 282101, “Exterior Flood Barriers For Doors 3 and 8 and VNDG-4151A Thru 4151F,” Revision 1 dated October 26, 2014 (Proprietary).

45. WE letter to NRC, Fire Protection Provisions, dated June 20, 1977.

46. Eval 2014-005, “EC 281811, External Wave Run-up Flood Mitigation Strategy,” dated November 6, 2014.

47. VPNPD-93-171, “Design Summary for the Installation of Two additional Emergency Diesel Generators - Point Beach Nuclear Plants, Unit 1 and 2," dated September 24, 1993 and attached Report REP-0026, "PBNP Diesel Project Design Submittal,” Revision 0, dated September 21, 1993.

48. EC 282394, “Comprehensive 95002 Internal Flooding for Turbine Hall.”

49. Calculation S-13206-424-01, “Circulating Water Piping Expansion Joint Flow Limiter.”

50. NRC Letter to WE, Seismic Qualification of the Auxiliary Feedwater System at Point Beach Nuclear Plant, Units 1 and 2, dated September 16, 1986.

51. EC 282055, “High Condenser Pit and Lake Level Alarm Inputs.”

52. NP 8.4.17, “PBNP Flooding Program,” Revision 22, October 15, 2015.

53. EC 283362, “Flood Mitigation Strategy - DGB, NVSG and HVAC Rooms.”