SECTION 1.0

INTRODUCTION AND SUMMARY

1.1 PROJECT IDENTIFICATION

This Updated Final Safety Analysis Report (FSAR) is submitted to meet the requirements of 10 CFR 50.71(e). This Updated FSAR is a unique document that contains the changes necessary to reflect information and analyses submitted to the Nuclear Regulatory Commission (NRC) since the submission of the original FSAR and previous amendments. The Updated FSAR was prepared to provide a reference document to be used as the basis for annual revisions. The level of detail of this Updated FSAR is at least the same as that provided in the original FSAR.

The Updated FSAR is comprised of the original FSAR and all subsequent supplements and amendments to the original FSAR which are still applicable to form a single complete and integrated document. The original outline has been maintained with new sections added for material which was not required in the original FSAR. This Updated FSAR is a complete document and does not reference or rely on the original FSAR, as amended.

The original FSAR was submitted in support of the application by the Philadelphia Electric Company (now Exelon Generation Company, LLC (EGC)) for a facility operating license for a two-unit nuclear power plant located at the Peach Bottom site in York County, Pennsylvania, for initial power levels up to 3,293 MWt for each unit, which shall be known as the Peach Bottom Atomic Power Station (PBAPS) Units 2 and 3.

Units 2 and 3 were issued operating licenses on October 25, 1973, and July 2, 1974, respectively; Unit 2 began commercial operation during July 1974, and Unit 3 began commercial operation during December 1974.

Subsequent to issuing these operating licenses, PBAPS Units 2 and 3 were reevaluated with regard to rerating power to 3458 MWt. The acceptability of the rerate evaluations stems from the fact that PBAPS Units 2 and 3 were originally designed for steam flow capabilities at least 5% above its original rating. In addition, improvements in the analytical techniques based on more realistic assumptions, plant performance feedback, and the latest fuel designs resulted in a significant increase in the calculated operational margins related to safety analyses. The licensee received a license amendment for a 5% increase in rated power to 3458 MWt for PBAPS Unit 2 on October 18, 1994, and for Unit 3 on July 18, 1995. The licensee received a license amendment for a 1.62% increase in rated power to 3514 WMt for PBAPS Unit 2 and 3 based on the use of more accurate feedwater flow measurement equipment.

A third reevaluation was done that is termed "Extended Power Uprate (EPU)" in this document, which justified operation of PBAPS Units 2 and 3 with a core thermal power level of 3951 MWt. This EPU power level is 120% of the original licensed core thermal power level of 3293 MWt. The acceptability of the EPU evaluations stems from continuing improvements in the analytical techniques (computer codes) based on several decades of BWR safety technology, plant performance feedback, operating experience, and improved fuel and core designs, which have resulted in significant increases in the design and operating margins between the calculated safety analyses results and the plant licensing limits. The available margins in calculated results, combined with hardware modifications and as-designed excess equipment, system, and component capabilities, provide for power increases up to 20% above original licensed thermal power.

An additional evaluation and license amendment request were submitted to justify expansion of the core flow operating domain in regions with less than rated core flow. This expansion does not increase rated thermal power or rated core flow. The expanded operating domain is identified as Maximum Extended Load Line Limit Analysis Plus (MELLLA+). This evaluation supports operation of Peach Bottom Atomic Power Station at a rated thermal power level of 3,951MWt with a core flow as low as 83% of rated core flow.

An additional Measurement Uncertainty Recapture (MUR) power re-rate evaluation was performed to increase licensed Rated Thermal Power (RTP) level by approximately 1.645% from 3951 MWt to 4016 MWt. These changes are the result of crediting the improved feedwater flow measurement achieved by the existing high accuracy, ultrasonic flow measurement instrumentation. Credit for this instrumentation was not utilized as part of the EPU program since it was not covered by the EPU Licensing Topical Documents. Plant safety, component, and system analyses for which rated power is an input only needs to reflect a 0.34% power measurement uncertainty. Accordingly, the previously existing 2% uncertainty can be allocated as follows: 1.66% is applied to provide sufficient margin to address an uprate to

4016 MWt, and 0.34% is applied in the analysis to continue to account for power measurement uncertainty. The nominal reactor heat balance at 100% rated thermal power and 100% core flow is shown in Figure 1.6.2. The additional MUR power uprate includes operation in the MELLLA+ expanded operating domain. The minimum core flow for l 00% rated power is 85.2% rated core flow (see Figure 3.7.1).

1.1.1 Identification and Qualifications of Contractors

1.1.1.1 Applicant

The Applicant (Philadelphia Electric Company – now EGC) engaged the following contractors to perform engineering, procurement, and construction services for the plant. However, irrespective of the contractual responsibilities discussed below, Philadelphia Electric Company was the sole applicant for the facility license, and was responsible for the design, construction, and operation of the plant. The Applicant was technically qualified to engage in the proposed activities as shown by a summary of previous experience in the field of power generation.

The Applicant was responsible for the design and construction and operation of 13 fossil fuel power plants. Additionally, two large multi-unit hydro-electric generating plants, one of which is a pumped-storage plant, and several diesel engine and gas turbine-driven generator installations also were designed, constructed, and are operated by the Applicant.

The Applicant also had an ownership interest in two additional operating fossil fuel plants. These plants provided capacity and energy to the Applicant's system, but were not operated by the Applicant.

These facilities, which had a net capacity of 6,190,200 kW (summer rating) at the end of 1981, constituted the bulk of the Applicant's fossil electric generating system.

The Applicant was active in the development of atomic energy for electric generation for many years. In 1952 it became a charter member of the Dow Chemical - Detroit Edison Nuclear Power Development Project, which subsequently became Atomic Power Development Associates, Inc. (APDA). This organization designed and developed a fast breeder power reactor for the Atomic Energy Commission's Power Demonstration Program. The Applicant also took part in the formation of Power Reactor Development Company (PRDC), which was organized to finance, construct, own, and operate the fast breeder reactor designed by APDA for the Enrico Fermi Atomic Power Station.

The Applicant's engineers had experience, at various times, in many phases of nuclear projects, including (1) assignments to APDA for design and development of core and fuel elements, shielding design, coordination of research at various levels on the metallurgical and chemical aspects of fuel elements; (2) shift supervisor duties and pre-operational duties, including preparation of plant operation manuals, at the Enrico Fermi Atomic Power Station; (3) assignment to PRDC for coordination of control instrumentation and electrical features; (4) assignment to the Nautilus nuclear submarine project for field engineering and mechanical operations during startup and initial operation; (5) assignment to the nuclear reactor at Shippingport, Pennsylvania for training and operational duties; and (6) assignment to the Knolls Atomic Power Laboratory for participation in prototype design of a sodium boiler for a submarine reactor.

The Applicant also had gained nuclear experience through the construction and operation of its PBAPS Unit 1. This unit was a 200 MWt high temperature, gas-cooled, demonstration power reactor (HTGR). The Atomic Energy Commission authorized operation of this reactor at full power on January 12, 1967. Unit 1 is now in a SAFSTOR status that allows it to be safely stored and subsequently decontaminated to levels that permit release of the facility for unrestricted use. EGC currently holds possession-only license for Unit 1.

The Applicant maintained a staff of qualified engineers. This staff reviewed and approved design features of the plant, conducted with the aid of outside consultants, a quality assurance program, and followed all field construction activity through to the completion of the plant.

1.1.1.2 Engineer-Constructor

The licensee retained Bechtel Power Corporation to provide engineering and construction services for the design and construction of the plant, integrating the items furnished by General Electric Company, the nuclear steam system supplier, with complete balance of plant items. Bechtel Power Corporation was also responsible for shop inspection of all equipment other than the nuclear steam supply system.

Bechtel Power Corporation has been continuously engaged in construction or engineering activities since 1898. Since 1940, Bechtel has been active in the fields of piping, petroleum, power generation and distribution, harbor development, mining and metallurgy, and chemical and industrial processing. The Bechtel organization has grown progressively to become one of the world's largest engineer-constructors for industrial facilities. Thus, Bechtel Power Corporation was qualified to provide the required services for station design, equipment procurement, construction, and startup.

1.1.1.3 Nuclear Steam Supply System Supplier

General Electric Company was awarded a contract to design, fabricate, and deliver the nuclear steam supply system and nuclear fuel for the plant, as well as to provide technical direction for startup of this equipment. The General Electric Company has been engaged in the development, design, construction, and operation of boiling water reactors (BWR's) since 1955. BWR's designed and built by General Electric include the Vallecitos Boiling Water Reactor, Dresden Units 1 and 2, Humboldt Bay, Big Rock Point, KRB (Germany), JPDR (Japan), SENN (Italy), Oyster Creek Unit 1, and Nine Mile Point Unit 1. Among other reactors of General Electric design are Millstone Point Unit 1; Dresden Unit 3; Quad-Cities Units 1 and 2; Monticello Unit 1; Vermont Yankee Unit 1; Browns Ferry Units 1, 2, and 3; and many others since 1970. Thus, General Electric has substantial experience, knowledge, and capability; designed and manufactured the reactors; and furnished technical direction for their startup.

1.1.1.4 Turbine-Generator Supplier

The Applicant awarded a contract to General Electric Company to design, fabricate, and deliver the turbine-generators for the plant, as well as to provide technical assistance for installation and startup of this equipment. General Electric Company has a long history in the application of turbine-generators in nuclear power stations going back to the inception of nuclear facilities for the production of electrical power. General Electric Company is furnishing the turbine-generator units for most stations equipped with its BWR nuclear steam supply systems. General Electric is also supplying turbine-generator units for various other nuclear power plants. The inlet pressures of these units vary between 750 psig to 1,500 psig, and temperatures vary from saturation to approximately 40°F superheat. The ratings of these units range from 500,000 to 1,090,000 kW. General Electric designed, fabricated, and delivered the turbine-generator units, and provided technical direction for the startup of this equipment. The turbines were replaced with turbines manufactured by Alstom Power, which were originally rated for 1,360 MWe. For conditions associated with a reactor Core Thermal Power (CTP) of 4016 MWt, the shaft output of these GE/Alstom turbines is 1391 MW. The turbines were evaluated for the MUR project and were found to be capable of operation at conditions up to, and including, Valves-Wide-Open (VWO). The VWO conditions bound those at a reactor CTP of 4030 MWt. The generators were upgraded to a rating of 1530 MVA as part of the EPU.

1.1.1.5 Consultants During Design and Construction

The licensee has engaged the services of various consultants to perform work and/or verify design concepts for the following: meteorology, hydrology of Conowingo Pond, hydraulic models of Conowingo Pond, ecological studies of the pond, flood calculations for the Susquehanna River basin, seismology, geology, radiation monitoring programs, and quality assurance. The licensee has cooperated in the Water Quality Network program. The licensee has also cooperated with various state and local agencies in matters relating to construction and operation of the Peach Bottom nuclear units.

1.1.1.6 Contractors and Consultants for Plant Modifications

The licensee engaged the services of various contractors and consultants to perform modifications to the plant. Engineering services have been performed by Stone & Webster Engineering Corporation, Chicago Bridge and Iron Corporation, Bechtel Power Corporation, and others. The implementation of the modifications have been performed by Catalytic, Inc. and Chicago Bridge and Iron Corporation.

1.1.1.7 Updated FSAR Contractor

The licensee retained Stone & Webster Engineering Corporation to provide technical direction, preparation, and publication of the initial issue of the Updated FSAR. The licensee has been responsible for all subsequent updated UFSAR revisions.

1.2 DEFINITIONS

The following definitions apply to the terms used in this FSAR:

1. Radioactive Material Barrier - A radioactive material barrier includes the systems, structures, or equipment that together physically prevent the uncontrolled release of radioactive materials. The four barriers are identified as follows:

    a. Reactor Fuel Barrier - Uranium dioxide sealed in metal cladding.

    b. Nuclear System Process Barrier - The systems of vessels, pipes, pumps, tubes, and other process equipment that contain the steam, water, gases, and radioactive materials coming from, going to, or in communication with the reactor core. The actual boundaries of the nuclear system process barrier depend upon the status of plant operation.

       For example, process system isolation valves, when closed, form part of the barrier. The steam-jet air ejector off-gas path forms a planned process opening in the barrier during power operation.

       Because the nuclear system process barrier is designed to be divided by isolation valve action into two major sections under certain conditions, this barrier is considered in two parts as follows:

       (1) Nuclear system primary barrier - The reactor vessel and attached piping out to and including the second isolation valve in each attached pipe. In various codes and standards used in the industry, this barrier is sometimes referred to as the "primary system pressure boundary."

       (2) Nuclear system secondary barrier - That portion of the nuclear system process barrier not included in the nuclear system primary barrier.

    c. Primary Containment - The drywell in which the reactor vessel is located, the pressure suppression chamber, and process line reinforcements out to the first isolation valve outside the containment wall. Portions of the nuclear system process barrier may become part of the primary containment, depending upon the location of a postulated failure. For example, a closed main steam line isolation valve is part of the primary containment barrier when the postulated failure of the main steam line is inside the primary containment.

d. Secondary Containment - The reactor building which completely encloses the primary containment. The reactor building ventilation system and the standby gas treatment system constitute controlled process openings in this barrier.

2. Radioactive Material Barrier Damage - Radioactive material barrier damage is defined as an unplanned, undesirable breach in a barrier, except that the operation of a relief or safety valve does not constitute barrier damage.

3. Nuclear System - The nuclear system generally includes those systems most closely associated with the reactor vessel which are designed to contain or be in communication with the water and steam coming from or going to the reactor core. The nuclear system includes the following:

   a. Reactor vessel

   b. Reactor vessel internals

   c. Reactor core

   d. Main steam lines from reactor vessel to the isolation valves outside the primary containment

   e. Neutron monitoring system

   f. Reactor recirculation system

   g. Control rod drive system

   h. Residual heat removal system

   i. Reactor core isolation cooling system

j. Core standby cooling systems

k. Reactor water cleanup system

l. Feedwater system piping between the reactor vessel and the first valve outside the primary containment.

4. Safety - The word, safety, when used to modify such words as objective, design basis, action, and system, indicates that the objective, design basis, action, or system is related to concerns considered to be of primary safety significance, as opposed to the plant mission - the generation of electrical power. Thus, the word safety is used to identify aspects of the plant which are considered to be of primary importance with respect to safety.

5. Power Generation - The phrase, power generation, when used to modify such words as objective, design basis, action, and system, indicates that the objective, design basis, action, or system is related to the mission of the plant--the generation of electrical power--as opposed to concerns considered to be of primary safety importance. Thus, the phrase, power generation is used to identify aspects of the plant which are not considered to be of primary importance with respect to safety.

6. Operational - The word, operational, is used in reference to the working or functioning of the plant, in contrast to the design of the plant.

7. Scram - Scram is the shutdown of the reactor by rapid insertion of control rods.

8. Safety Limit - A safety limit is an established limit above normal operational limits on the value of a nuclear system process or analytical variable, or an established limit specifying an allowable degree of barrier damage.

9. Planned Operation - Planned operation is normal plant operation under planned conditions in the absence of significant abnormalities. Operations subsequent to an incident (transient, accident, or special event) are not considered planned operations until the actions taken in

the plant are identical to those which would be used had the incident not occurred. The established planned operations can be considered as a chronological sequence: refueling ----> achieving criticality ----> heatup ----> power operation ----> achieving shutdown ----> cooldown ----> refueling.

The following planned operations are identified:

a. Refueling - Refueling includes all of the planned operations associated with a normal refueling except those tests in which the reactor is taken out of and returned to the shutdown (more than one rod subcritical) condition. The following operations are included in refueling:

    (1) Planned physical movement of core components (fuel, curtains, control rods, etc).

    (2) Refueling test operations (except criticality and shutdown margin tests).

    (3) Planned maintenance.

b. Achieving Criticality - Achieving criticality includes the plant actions which are normally accomplished in bringing the reactor from a shutdown condition to a condition in which nuclear criticality is achieved and maintained.

c. Heatup - Heatup begins where achieving criticality ends and includes plant actions normally accomplished in approaching nuclear system rated temperature and pressure by using nuclear power (reactor critical). Heatup extends through warmup and synchronization of the turbine-generator.

d. Power Operation - Power operation begins where heatup ends and includes continued plant operation at power levels in excess of heatup power.

e. Achieving Shutdown - Achieving shutdown begins where power operation ends and includes plant actions normally accomplished in achieving nuclear shutdown (more than one rod subcritical) following power operation.

f. Cooldown - Cooldown begins where achieving shutdown ends and includes plant operations for achieving and maintaining the desired conditions of nuclear system temperature and pressure.

10. Incident - An incident is any event, i.e., abnormal operational transient, accident, special event, or other event, not considered as part of planned operation.

11. Abnormal Operational Transient - The original FSAR definition of an abnormal operational transient includes the events following a single equipment malfunction or single operator error that is reasonably expected during the course of plant operations. Power failures, pump trips, and rod withdrawal errors are typical of the single malfunctions or errors initiating the events in this category. This definition conservatively encompasses all possible operational transients, including those that are not necessarily considered 'abnormal'. Chapter 14 (Plant Safety Analysis) evaluates all plant operational transients and provides a distinction between 'anticipated' (expected) operational transients and 'abnormal' (unexpected) operational transients based on frequency of occurrence and associated event acceptance criteria consistent with the current plant licensing basis documented in NEDE-24011-P-A (GESTAR II).

12. Abnormal Occurrence - Abnormal occurrence refers to the occurrence of any plant condition that:

    a. Exceeds a limiting safety system setting as established in the technical specifications.

    b. Violates a limiting condition for operation as established in the technical specifications.

    c. Causes any abnormal operational transient.

    d. Causes any uncontrolled or unplanned release of radioactive material from the site.

13. Accident - An accident is a single event, not reasonably expected during the course of plant operations, that has been hypothesized for analysis purposes or postulated from unlikely but possible situations, and that causes or threatens a rupture of a radioactive material

barrier. A pipe rupture qualifies as an accident; a fuel cladding defect does not.

14. Design Basis Accident - A design basis accident is a hypothesized accident the characteristics and consequences of which are utilized in the design of those systems and components pertinent to the preservation of radioactive material barriers and the restriction of radioactive material release from the barriers. The potential radiation exposures resulting from a design basis accident are greater than any similar accident postulated from the same general accident assumptions. For example, the consequences of a complete severance of a recirculation loop line are more severe than those resulting from any other single pipeline failure inside the primary containment.

15. Special Event - A special event is an event which qualifies neither as an abnormal operational transient nor an accident, but which is postulated to demonstrate some special capability of the plant or its systems.

16. Safety Action - A safety action is an ultimate action in the plant which is essential to the avoidance of specified conditions considered to be of primary safety significance. The specified conditions are those that are most directly related to the ultimate limits on the integrity of the radioactive material barriers or the release of radioactive material. There are safety actions associated with planned operation, abnormal operational transients, accidents, and special events. Safety actions include such actions as the indication to the operator of the values of certain process variables, reactor scram, core standby cooling, and reactor shutdown from outside the control room (Figures 1.2.1 and 1.2.3 and Table 1.4.2).

17. Power Generation Action - A power generation action is an action in the plant which is essential to the avoidance of specified conditions considered to be of primary significance to the plant mission--the generation of electrical power. The specified conditions are those that are most directly related to the following:

a. The ability to carry out the plant mission, the generation of electrical power, through planned operation.

b. The avoidance of conditions which would limit the ability of the plant to generate electrical power.

c. The avoidance of conditions which would prevent or hinder the return to conditions permitting the use of the plant to generate electrical power following an abnormal operational transient, accident, or special event. There are power generation actions associated with planned operation, abnormal operational transients, accidents, and special events.

18. Protective Action - A protective action is an ultimate action at the system level which contributes to and is essential to the accomplishment of a safety action. System level actions which are essential to accomplishing reactor scram, reactor vessel isolation, containment isolation, pressure relief, automatic depressurization, and core standby cooling are some of the protective actions (Figures 1.2.1, 1.2.2, and 1.2.3).

19. Protective Function - A protective function encompasses the monitoring of one or more plant variables or conditions and the associated initiation of intrasystem actions which eventually result in protective action (Figure 1.2.2).

20. Safety System - A safety system is any system, group of systems, component, or group of components the actions of which are essential to accomplishing a safety action (Figure 1.2.3 and Table 1.4.2).

21. Process Safety System - A process safety system is a safety system the actions of which are essential to a safety action required during planned operation (Figure 1.2.3 and Table 1.4.2). A process safety system in and of itself is not required to be safety related. However, individual components must be reviewed against other quality assurance or regulatory requirements for proper safety classification. Safety-related equipment may be used to satisfy a process safety system design objective (e.g., Primary Containment Pressure Monitoring System).

22. Nuclear Safety System - A nuclear safety system is a safety system the actions of which are essential to a safety action required in response to an abnormal operational transient (Figure 1.2.3 and Table 1.4.2).

23. Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required in response to accidents (Figure 1.2.3 and Table 1.4.2).

24. Protection System - Protection system is a generic term which may be applied to nuclear safety systems and engineered safeguards (Figure 1.2.3 and Table 1.4.2).

25. Special Safety System - A special safety system is a safety system the actions of which are essential to a safety action required in response to a special event (Figure 1.2.3 and Table 1.4.2).

26. Power Generation System - A power generation system is any system the actions of which are not essential to a safety action, but which are essential to a power generation action. Power generation systems are provided for any of the following purposes:

    a. To carry out the mission of the plant, the generation of electrical power, through planned operation.

    b. To avoid conditions which would limit the ability of the plant to generate electrical power.

    c. To facilitate and expedite the return to conditions permitting the use of the plant to generate electrical power following an abnormal operational transient, accident, or special event (Figure 1.2.3 and Table 1.4.2).

27. Safety Objective - A safety objective describes in functional terms the purpose of a system or component as it relates to conditions considered to be of primary significance to the protection of the public. This relationship is stated in terms of radioactive material barriers or radioactive material release. The only systems which have safety objectives are safety systems (Figure 1.2.3).

28. Power Generation Objective - A power generation objective describes in functional terms the purpose of a system or component as it relates to the mission of the plant. This includes objectives which are specifically established so the plant can fulfill the following purposes:

    a. The generation of electrical power through planned operation.

    b. The avoidance of conditions which would limit the ability of the plant to generate electrical power.

    c. The avoidance of conditions which would prevent or hinder the return to conditions permitting the use of the plant to generate electrical power following an abnormal operational transient, accident, or special event (Figure 1.2.3).

       A system or piece of equipment has a power generation objective if it is a power generation system. A safety system can have a power generation objective, in addition to a safety objective, if parts of the system are intended to function for power generation purposes.

29. Analytical Objective - An analytical objective describes the purpose or intent of a portion of this report presenting an analysis.

30. Safety Design Basis - The safety design basis for a safety system states in functional terms the unique design requirements which establish the limits within which the safety objective shall be met.

    A power generation system may have a safety design basis which states in functional terms the unique design requirements that ensure that neither planned operation nor operational failure of the system results in conditions for which plant safety actions would be inadequate.

31. Power Generation Design Basis - The power generation design basis for a power generation system states in functional terms the unique design requirements which

establish the limits within which the power generation objective shall be met.

A safety system may have a power generation design basis which states in functional terms the unique design requirements which establish the limits within which the power generation objective for the system shall be met.

32. Safety Evaluation - A safety evaluation is an evaluation which shows how the system satisfies the safety design basis. A safety evaluation is performed only for those systems having a safety design basis.

33. Power Generation Evaluation - A power generation evaluation is an evaluation which shows how the system satisfies some or all of the power generation design bases. Because power generation evaluations are not directly pertinent to public safety, they are generally not included. However, where a system or component has both safety and power generation objectives, a power generation evaluation can be used to clarify the safety versus power generation capabilities.

34. Operational Nuclear Safety Requirements - An operational nuclear safety requirement is a limitation or restriction on either the value of a process variable or the operability of a plant system. Such operational nuclear safety requirements must be observed in the operation (not necessarily at power) of the plant to satisfy specified operational nuclear safety criteria. The aggregate of all operational nuclear safety requirements defines an operational framework within which actual plant operations must remain.

35. Rated Power - Rated power refers to operation at a reactor power of 3951 MWt; this is also termed 100 percent power. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power.

36. Design Power

    The safety analyses are based on a power level of 4030 MWt (approximately 1.0034 times the rated power level). This margin accounts for the Reg. Guide 1.49 requirements in accordance with the MUR design criteria. Reg. Guide 1.49 does not apply to special events such as anticipated transients without Scram (ATWS), Station Blackout (SBO), and Appendix R. These events were analyzed for the rated power level of 4016 MWt.

37. Single Failure - A single failure is a failure that can be ascribed to a single causal event. Single failures are considered in the design of certain systems and are presumed in the evaluations of incidents to investigate the ability of the plant to respond in the required manner under degraded conditions. The nature of single causal event to be presumed depends on the risk of the event being evaluated. Reasonably expected single failures are presumed as the causes of abnormal operational transients. Single failures of passive equipment are assumed sometimes to be the causes of accidents. Safety actions essential in response to abnormal operational transients and accidents must be carried out in spite of single failures in active equipment. In any case, a single failure includes the multiple effects resulting from the single causal event.

38. Operable - Operability - A system, subsystem, division, component, or device shall be operable or have operability when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

39. Operating - A system or component is operating when it is performing its required action in its required manner.

40. Shutdown - The reactor is shut down when the effective multiplication factor (k_{eff}) is sufficiently less than 1.0 that the full withdrawal of any one control rod could not produce criticality under the most restrictive potential conditions of temperature, pressure, core age, and fission product concentration.

41. Operational Transient - see Abnormal Operational Transient (11).

42. Cold Shutdown - The reactor is in cold shutdown when the reactor mode switch is in the shutdown position, and the reactor coolant is maintained at less than or equal to 212°F.

43. Not Used.

44. Place in Cold Shutdown - Place in cold shutdown means conduct an uninterrupted normal plant shutdown operation until cold shutdown is attained.

45. Not Used.

46. Not Used.

47. Not Used.

48. Place in Isolated Condition - Place in isolated condition means conduct an uninterrupted normal isolation of the reactor from the main (turbine) condenser, including the closure of the main steam line isolation valves.

49. Not Used.

50. Not Used.

51. Core Fuel-To-Water Total Power - The core fuel-to-water total power is the sum of (a) the instantaneous integral, over the entire fuel clad outer surface, of the product of heat transfer area increment and position dependent heat flux, and (b) the instantaneous rate of energy deposition by neutron and gamma reactions in all the water and core components, except fuel rods, in the cylindrical volume defined by the active core height and the inner surface of the core shroud.

52. Not Used.

53. Core Alteration - Core Alteration shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be Core Alterations:

a. Movement of wide range neutron monitors, local power range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and

b. Control rod movement, provided there are no fuel assemblies in the associated core cell.

c. Suspension of Core Alterations shall not preclude completion of movement of a component to a safe position.

54. Risk - Risk is the product of the probability of an event and the adverse consequences of the event.

55. Reliability - Reliability is the probability that an item will perform its specified function without failure for a specified time period in a specified environment.

56. Unreliability - Unreliability is the probability that component or system will fail to perform its specified action for a specified time period in a specified environment. (The sum of reliability and unreliability equals unity.)

57. Availability - Availability is the probability that an item will be operable when called upon to perform its specified function.

58. Unavailability - Unavailability is the probability that a component or system will be inoperable when called upon to perform its specified action. (The sum of availability and unavailability equals unity.)

59. Repair Rate - The repair rate is the number of repairs completed per unit time.

60. Failure Rate - The failure rate is the number of failures per unit time.

61. Test Duration - The test duration is the elapsed time between test initiation and test termination.

62. Test Interval - The test interval is the elapsed time between the initiation of identical tests.

63. Phenomenological Event - A phenomenological event is a real, observable event. The complete set of phenomenological events includes all real events, no matter how rare or unpredictable. Thus, from an a

priori point of view, phenomenological events are unknowable; from the a posteriori point of view, phenomenological events are known.

64. Phenomenological Probability - A phenomenological probability is the probability that a system which performs its action perfectly will actually mitigate successfully to alleviate the consequences of a phenomenological event. Thus, it is a measure of the degree to which the action the system is expected to perform matches the true nature of the phenomenological events imposed upon the system.

65. Active Component - A device characterized by an expected significant change of state or discernable mechanical motion in response to an imposed design basis load demand upon the system. Examples are: switch, relay valve, pressure switch, turbine, transistor, motor, damper, pump, analog meter, etc.

66. Passive Component - A device characterized by an expected negligible change of state or negligible mechanical motion in response to an imposed design basis load demand upon the system. Examples are: cable, piping, valve in stationary position, resistor, capacitor, fluid filter, indicator lamp, cabinet, case, etc.

1.3 METHODS OF TECHNICAL PRESENTATION

1.3.1 Purpose

The purpose of the annual updating of the Updated FSAR is to provide a reference document of all changes made to the plant and reported to the NRC.

The purpose of the original FSAR was to provide the technical information required by Section 50.34 of 10CFR50 to establish a basis for evaluation of the plant with respect to the issuance of a facility operating license. Unless otherwise specified, the information in this report is presented in terms of one unit, but is applicable to both units.

1.3.2 Radioactive Material Barrier Concept

Because the safety aspects of this report pertain to the relationship between plant behavior under a variety of circumstances and the radiological effects on persons off-site, the report is oriented to the radioactive material barriers. This orientation facilitates evaluation of the radiological effects of the plant on the environment. Thus the presentation of technical information is considerably different from that which would be expected in an operational manual, maintenance manual, or nuclear engineer's handbook.

The technical information presented about a system or component is that relationship of the system or component to the radioactive material barriers. Systems that must operate to preserve or limit the damage to the radioactive material barriers are described in the greatest detail. Systems that have little relationship to the radioactive material barriers are described only with as much detail as is necessary to establish their functional role in the plant.

1.3.3 Organization of Contents

This Updated FSAR is organized into 14 major sections, each of which consists of a number of subsections. The format is the same as the original FSAR except where new material has been supplied. The new material is placed under new section numbers, usually at the end of a chapter.

A system for classifying the various aspects of the BWR with respect to safety is given in subsection 1.4. This classification system is fundamental to assessing the adequacy of the plant with respect to the relative importances of different safety concerns. The principal architectural and engineering criteria, which define the broad frame of reference within which the plant was designed, are set forth in subsection 1.5. Subsection 1.6 presents a brief description of the plant in which the nuclear safety systems and engineered safeguards are separated from the other plant systems so that those systems essential to safety are clearly identified.

Sections 2.0 through 13.0 present detailed information about the design and operation of the plant. The nuclear safety systems and engineered safeguards are integrated into these sections according to system function (core standby cooling control), system type (electrical, mechanical), or according to their relationship to a particular radioactive material barrier. Section 3.0, "Reactor," describes plant components and presents design details that are most pertinent to the fuel barrier. Section 4.0, "Reactor Coolant System," describes plant components and systems that are most pertinent to the nuclear system process barrier. Section 5.0 describes the primary and secondary containments. Thus, Sections 3.0, 4.0, and 5.0 are arranged according to the four radioactive material barriers. The remainder of the sections group system information according to plant function (radioactive waste control, core standby cooling, power conversion, control) or system type (electrical, structures).

Section 14.0, "Plant Safety Analysis," provides an overall safety evaluation of the plant which demonstrates both the adequacy of equipment designed to protect the radioactive material barriers and the ability of the safeguard features to mitigate the consequences of situations in which one or more radioactive material barriers are assumed damaged.

1.3.4 Format Organization of Subsections

Subsections are numerically identified by representing their order of appearance in a section by two numbers separated by a decimal point, e.g., 3.4, the fourth subsection in Section 3.0. Subsections are further subdivided into major paragraphs by numbers separated by decimal points (3.4.1, 3.4.1.1, etc.). Pages within each subsection are consecutively numbered (3.4-1, 3.4-2, etc).

Tabulations of data are designated "Tables" and are identified by the subsection number followed by a decimal point and the number of the table, e.g., Table 3.4.5 is the fifth table of subsection 3.4. Drawings, pictures, sketches, curves, graphs, and engineering diagrams are identified as "Figures" and are numbered in the same manner as tables. (Drawing M-300 defines the meanings of piping instrumentation symbols used in the figures of this report.)

The general organization of a subsection describing a system or component is as follows:

1. Objective

2. Design Basis

3. Description

4. Evaluation

5. Inspection and Testing

6. Operational Nuclear Safety Requirements (if applicable).

To clearly distinguish the safety versus operational aspects of a system, the objective, design basis, and evaluation titles are modified by the words "safety" or "power generation," according to the definitions given in subsection 1.2. Systems that have safety objectives are safety systems. A safety evaluation is included only when the system has a safety design basis; the evaluation shows how the system satisfies the safety design basis. A power generation evaluation is included when needed to clarify the safety versus power generation aspects of a system that has both safety and power generation functions.

A nuclear safety operational analysis of the plant has been performed to systematically identify the operational limitations or restrictions which must be observed with regard to certain process variables and certain plant systems to satisfy specified nuclear safety operational criteria. The method used for this analysis is described in Appendix G. The resulting operational limits or restrictions are presented in the "Operational Nuclear Safety Requirements" portions of the subsections describing the applicable systems. The operational nuclear safety requirements and limits form the bases for the technical specifications (Appendix B).

Within each subsection of the text, applicable supporting technical material is referenced. References are cited in a list of references at the end of a section or subsection. The references may provide a particular technical basis for BWR plant design and analysis, or document special technical work performed by General Electric, which is specifically applicable to the PBAPS.

1.3.5 Power Level Basis for Analysis of Abnormal Operational Transients and Accidents

For those abnormal operational transients and accidents for which high power operation increases the severity of the results, the analyses assume plant operation at design power as an initial condition. For those events for which an initial condition of low or intermediate power level operation renders the most severe results, the analyses presented in this report represent the most severe case within the operating spectrum.

1.4 CLASSIFICATION OF BWR SYSTEMS, CRITERIA, AND REQUIREMENTS FOR SAFETY EVALUATION

1.4.1 Introduction

To fully evaluate the many aspects of the design and operation of the BWR plant, it is necessary to classify the various systems, criteria, design bases, and operating requirements in light of specified personnel (including the public) hazard considerations. A system has been developed which allows classification of any BWR aspect (criterion, system, design basis, or operating requirement) relative to either personnel hazard or the plant mission, the generation of electrical power.

Table 1.4.1 illustrates the concept used in the classification process. The concept applies to the total plant: design and operation. A major distinction is made between those BWR aspects which are most pertinent to personnel hazard and those which are most pertinent to the plant mission, the generation of electrical power. Those aspects most pertinent to personnel hazard appear under the "safety consideration" side of the table, and the aspects most pertinent to the plant mission appear under the "power generation" side. All plant components contribute in some measure to safety, but those classified under "power generation" considerations are considerably less important to safety than those BWR items classified under "safety" considerations.

Therefore, the right and left sides of the table represent a major difference in importance to safety.

Down the left side of Table 1.4.1 are listed the various types of plant operation, including events resulting in transients and accidents. An allowance is made for a special event in the left column to enable the classification of criteria, systems, and operational requirements not otherwise classifiable. The left-hand column is actually a gross probability scale. Planned operation is certain; abnormal operational transients are reasonably expected; and accidents are very improbable. Any special events would have to be fitted into the probability scale as appropriate.

The rectangular spaces formed under the safety considerations heading and the power generation heading represent potential classification categories for BWR criteria, systems, and operational requirements. This classification concept, when applied, allows an accurate distinction between the importances of the various aspects of BWR design and operation.

1.4.2 Classification Basis

Table 1.4.2 presents the basis for classifying various BWR items. The format of the table is identical to that used in Table 1.4.1, which presented the classification concept. Within each classification category a list of unacceptable results is given. The unacceptable results represent a set of master criteria from which the design and operation of the BWR can be consistently evaluated.

The only unacceptable results listed on the power generation (right) side of the chart are those that are more restrictive than those on the safety (left) side.

In the various columns inside each classification category, generic labels are assigned to the specific elements which appear or would appear, if listed, in the column. A generic label is given only to facilitate discussion and identification of a group of elements united by their common classification. Beneath the generic names are listed some of the more illustrative BWR items which can be classified in the different columns. Some of the listed items are the limits and restrictions found in the Technical Specifications.

Classification analyses have been performed to establish the essentiality of the various BWR systems to the avoidance or prevention of the listed unacceptable results. Such analyses consider any applicable criteria requiring redundancy or specified levels of functional reliability in the avoidance of unacceptable results. Once a system is classified, it is evaluated with reference to the criteria applicable to the group in which it performs an essential action. A classification analysis is not the same as a plant safety analysis. A classification analysis takes no credit whatever for the system under study; whereas, a plant safety analysis represents the true response of the whole plant to an event under specified analytical assumptions.

1.4.3 Use of the Classification Plan

Because Table 1.4.2 permits the classification of any BWR criterion, system, or operational requirement into one or more of the classification categories, the plan facilitates a plant-wide safety overview. The plan explains the reasons for the differences in the designs of apparently similar systems by relating the actions of the systems to specified unacceptable results. With the design complete, the classification plan is used to establish operational requirements and procedures whose differences are consistent with the different importances of unacceptable results.

It should be noted that a system may be classified in several categories. This occurs because classification is the result of a functional analysis of the plant. When classified in more than one category, a system must satisfy all of the requirements for each category with regard to its contributions to the various safety actions within each of the categories.

TABLE 1.4.1

BWR SAFETY ENGINEERING

CONCEPT FOR CLASSIFICATION OF BWR SYSTEMS, CRITERIA, AND REQUIREMENTS

FOR SAFETY EVALUATION

Type of Operation or Event	Actual Plant Design And Operation	
	Safety Considerations	Power Generation Considerations
1. Planned Operation	In this category are classified the unacceptable safety results, criteria, plant actions, systems, and operational requirements pertinent to safety during planned operation. This space represents the aspects of the BWR which must be considered to assure that the BWR operator can operate the plant within specified safety limitations. Certain process indicators, process variable limits, and limits on the release of radioactive material would be classified here.	In this category are classified the unacceptable results for power generation, criteria, plant actions, systems, and operational requirements pertinent to the production of electrical power during planned operation. Process systems and normal operational procedures would be classified here.
2. Abnormal Operational Transients	In this category are classified the unacceptable safety results, criteria, plant actions, systems, and operational requirements pertinent to safety in regard to abnormal operational transients. Certain protection systems, safety limits, and limiting safety system settings would be classified here.	In this category are classified the unacceptable results for power generation, criteria, plant actions, systems, and operational requirements pertinent to the ability to produce electrical power as that ability is affected by abnormal operational transients. Certain systems not used for planned operation would be classified here.
3. Accidents	In this category are classified the unacceptable safety results, criteria, plant actions, systems, and operational requirements pertinent to safety in regard to accidents. Engineered safeguards would be classified here.	In this category are classified the unacceptable results for power generation, criteria, plant actions, systems, and operational requirements pertinent to the ability to produce electrical power as that ability is affected by accidents. Design considerations and post-accident procedures provided to enable the plant to be used for power generation after an accident would be classified here.
4. Special Event	In this category are classified the unacceptable safety results, criteria, plant actions, systems, and operational requirements pertinent to safety in regard to the stated special event. Safety systems provided especially for the special event would be classified here.	In this category are classified the unacceptable results for power generation, criteria, plant actions, systems, and operational requirements pertinent to the ability to produce electrical power as that ability is affected by the stated special event. Systems and procedures provided to enable the plant to be returned to power operation following the special event would be classified here.

TABLE 1.4.2

CLASSIFICATION OF BWR SYSTEMS, CRITERIA, AND REQUIREMENTS FOR SAFETY EVALUATION

ACTUAL PLANT DESIGN AND OPERATION										
SAFETY CONSIDERATIONS						POWER GENERATION CONSIDERATIONS				
TYPE OF OPERATION OR EVENT	UNACCEPTABLE SAFETY RESULTS	TYPES OF APPLICABLE CRITERIA	TYPE OF ACTIONS REQUIRED TO AVOID UNACCEPTABLE RESULTS	TYPES OF SYSTEMS REQUIRED TO CARRY OUT ACTION	TYPES OF REQUIREMENTS TO BE OBSERVED IN OPERATION OF PLANT TO AVOID UNACCEPTABLE RESULTS	UNACCEPTABLE RESULTS FOR POWER GENERATION (WHERE MORE RESTRICTIVE THAN UNACCEPTABLE SAFETY RESULTS)	TYPES OF APPLICABLE CRITERIA	TYPES OF ACTIONS REQUIRED TO AVOID UNACCEPTABLE RESULTS (WHERE NOT REQUIRED AS A SAFETY ACTION)	TYPES OF SYSTEMS REQUIRED TO CARRY OUT ACTION (WHRE NOT REQUIRED AS A SAFETY SYSTEM	TYPES OF REQUIREMENTS TO BE OBSERVED IN OPERATION OF PLANT TO AVOID UNACCEPTABLE RESULTS
1. PLANNED OPERATION	1-1 RELEASE OF RADIOACTIVE MATERIAL TO THE ENVIRONS THAT EXCEEDS THE LIMITS OF 10CFR20. 1-2 FUEL FAILURE TO SUCH AN EXTENT THAT WERE THE FREED FISSION PRODUCTS RELEASED TO THE ENVIRONS VIA THE NORMAL DISCHARGE PATHS FOR RADIOACTIVE MATERIAL, THE LIMITS WOULD BE EXCEEDED. 1-3 NUCLEAR SYSTEM STRESS IN EXCESS OF THAT ALLOWED FOR PLANNED OPERATION BY APPLICABLE ASME AND ANSI CODES. 1-4 EXISTENCE OF A PLANT CONDITION NOT CONSIDERED BY PLANT SAFETY ANALYSIS.	NUCLEAR SAFETY DESIGN CRITERIA TYPE S-1 NUCLEAR SAFETY OPERATIONAL CRITERIA TYPE S-1 PROCESS SAFETY DESIGN CRITERIA PROCESS SAFETY OPERATIONAL CRITERIA VARIOUS INDUSTRY CODES RADWATE CRITERIA LOADING CRITERIA (NORMAL CONDITIONS)	SAFETY ACTION TYPE S-1 PROCESS SAFETY ACTION (A CATEGORY OF SAFETY ACTION) INDICATION OF PROCESS VARIABLES ROD WORTH CONTROL ROD PATTERN CONTROL CONTROL OF PROCESS VARIABLES CONTROL ROD CONTROL REFUELING BLOCK CONTROL ROD CONTROL REFUELING BLOCK CORE REACTIVITY CONTROL	SAFETY SYSTEMS – TYPE S-1 PROCESS SAFETY SYSTEMS (A CATEGORY OF SAFETY SYSTEMS) INDICATORS PROCESS RADIATION MONITORS REFUELING INTERLOCKS NUETRON MONITORING SYSTEM (FLUX INDICATIONS)	OPERATIONAL NUCLEAR SAFETY REQUIREMENTS – TYPE S-1 OPERATIONAL NUCLEAR SAFETY LIMITS – TYPE S-1 TECHNICAL SPECIFICATIONS – TYPE S-1 ENVELOPE LIMITS LIMITING CONDITIONS FOR OPERATION FOR INDICATORS RADIOACTIVE MATERIAL RELEASE LIMITS REACTIVITY LIMITS LIMITING CONDITIONS FOR OPERATION FOR RADWASTE SYSTEMS NUCLEAR SYSTEM LEAKAGE LIMITS	1-1 INABILITY TO GENERATE ELECTRICAL POWER 1-2 FUEL FAILURE 1-3 INABILITY TO PERFORM ROUTINE MAINTENANCE WITH PLANT AT POWER 1-4 INABILITY TO OPTIMIZE FUEL PERFORMANCE 1-5 INABILITY TO REPSOND TO CHANGES IN POWER DEMAND 1-6 INABILITY TO SHUT DOWN REACTOR WITH CONTROL RODS IN THE NORMAL MANNER	POWER GENERATION DESIGN CRITERIA – TYPE PG-1 POWERE GENERATION OPERATIONAL CRITERIA – TYPE PG-1 PROCESS DESIGN CRITERIA- PROCESS OPERATIONAL CRITERIA	POWER GENERATION ACTION – TYPE PG-1 PROCESS ACTION (A CATEGORY POWER GENERATION ACTION) INDICATORS OF PROCESS VARIABLES PROCESS OPERATIONS FUEL PERFOMANCE CALCULATIONS POWER LEVEL CONTROL CONDENSATION OF EXHAUST STEAM	POWER GENERATION SYSTEMS – TYPE PG-1 PROCESS SYSTEMS (A CATEGORY OF POWER GENERATION SYSTEMS) INDICATORS PROCESS COMPUTER SYSTEM RECIRCULATION FLOW CONTROL SYSTEM REACTOR MANUAL CONTROL SYSTEM CONTROL ROD DRIVE SYSTEM FEEDWATER SYSTEM TURBINE-GENERATOR MAIN CONDENSER	OPERATIONAL POWER GENERATION REQUIREMENTS-TYPE PG-1 OPERATIONAL POWER GENERATION LIMITS- TYPE PG-1 NORMAL OPERATING PROCEDURES MAINTENANCE PROCEDURES CALIBRATION PROCEDURES REFUELING PROCEDURES
2. ABNORMAL OPERATIONAL TRANSIENTS	2-1 THE RELEASE OF RADIOACTIVE MATERIAL TO THE ENVIRONS THAT EXCEEDS THE LIMITS OF 10CFR20. 2-2 ANY FUEL FAILURE CALCULATED AS A DIRECT RESULT OF THE TRANSIENT ANALYSES. 2-3 NUCLEAR SYSTEM STRESS IN EXCESS OF THAT ALLOWED FOR TRANSIENTS BY APPLICABLE ASME AND ANSI CODES.	NUCLEAR SAFETY DESIGN CRITERIA - TYPE S-2 NUCLEAR SAFETY OPERATIONAL CRITERIA – TYPE S-2 VARIOUS INDUSTRY CODES IEEE-279 LOADING CRITERIA – UPSET CONDITIONS SINGLE FAILURE CRITERION TESTABILITY CRITERIA	SAFETY ACTION TYPE S-2 SCRAM PRESSURE RELIEF CORE COOLING RESTORE A C POWER	SAFETY SYSTEMS TYPE – S-2 PROTECTION SYSTEM GENERIC TERM NUCLEAR SAFETY SYSTEMS (A CATEGORY OF PROTECTION SYSTEMS) REACTOR PROTECTION SYSTEM (SCRAM) CONTROL ROD DRIVE SYSTEM (SCRAM) NEUTRON MONITORING SYSTEM (WPRM, APRM) PRESSURE RELIEF SYSTEM REACTOR VESSEL ISOLATION CONTROL SYSTEM HIGH PRESSURE COOLANT INJECTION SYSTEM REACTOR CORE ISOLATION COOLING SYSTEM dc POWER SYSTEM STANDBY ac POWER INCIDENT DETECTOR CIRCUTRY	OPERATIONAL NUCLEAR SAFETY REQUIREMENTS – TYPE S-2 OPERATIONAL NUCLEAR SAFETY LIMITS-TYPE S-2 TECHNICAL SPECIFICATIONS – TYPE S-2 SAFETY LIMITS LIMITING SAFETY SYSTEM SETTINGS LIMITING CONDITIONS FOR OPERATION FOR PROTECTION SYSTEMS SURVEILLANCE REQUIREMENTS FOR PROTECTION SYSTEMS	2-1 FUEL FAILURE 2-2 THE LIFTING OF SAFETY VALVES 2-3 CONDITIONS REQUIRING THE OPENING OF THE REACTOR VESSEL FOR INSPECTION OR REPAIR 2-4 INABILITY TO RETURN TO POWER OPERATION 2-5 INADVERTENT CRITICALITY DURING REFUELING	POWER GENERATION DESIGN CRITERIA – TYPE PG-2 POWER GENERATION OPERTIONAL CRITERIA – TYPE PG-2	POWER GENERATION ACTION – TYPE PG-2 ROD BLOCK PRESSURE RELIEF REFUELING BLOCK	POWER GENERATION SYSTEMS – TYPE PG-2 REACTOR MANUAL CONTROL SYSTEM (ROD BLOCK) PRESSURE RELIEF SYSTEM REFUELING INTERLOCKS	OPERATIONAL POWER GENERATION REQUIREMENTS – TYPE PG-2 OPERATIONAL POWER GENERATION LIMITS – TYPE PG-2 NORMAL OPERATING PROCEDURES POST TRANSIENT RECOVERY PROCEDURES REFUELING RESTRICTIONS

TABLE 1.4.2 (Continued)

ACTUAL PLANT DESIGN AND OPERATION										
SAFETY CONSIDERATIONS						POWER GENERATION CONSIDERATIONS				
TYPE OF OPERATION OR EVENT	UNACCEPTABLE SAFETY RESULTS	TYPES OF APPLICABLE CRITERIA	TYPE OF ACTIONS REQUIRED TO AVOID UNACCEPTABLE RESULTS	TYPES OF SYSTEMS REQUIRED TO CARRY OUT ACTION	TYPES OF REQUIREMENTS TO BE OBSERVED IN OPERATION OF PLANT TO AVOID UNACCEPTABLE RESULTS	UNACCEPTABLE RESULTS FOR POWER GENERATION (WHERE MORE RESTRICTIVE THAN UNACCEPTABLE SAFETY RESULTS)	TYPES OF APPLICABLE CRITERIA	TYPES OF ACTIONS REQUIRED TO AVOID UNACCEPTABLE RESULTS (WHERE NOT REQUIRED AS A SAFETY ACTION)	TYPES OF SYSTEMS REQUIRED TO CARRY OUT ACTION (WHRE NOT REQUIRED AS A SAFETY SYSTEM	TYPES OF REQUIREMENTS TO BE OBSERVED IN OPERATION OF PLANT TO AVOID UNACCEPTABLE RESULTS
3. ACCIDENTS	3-1 RADIOACTIVE MATERIAL RELEASE EXCEEDING THE GUIDELINE VALUES OF 10CFR100. 3-2 CATASTROPHIC FAILURE OF THE FUEL BARRIER AS A RESULT OF EXCEEDING MECHANICAL OR THERMAL LIMITS. 3-3 NUCLEAR SYSTEM STRESSES EXCEEDING THAT ALLOWED FOR ACCIDENTS BY APPLICABLE ASME AND ANSI CODES 3-4 CONTAINMENT STRESSES EXCEEDING THAT ALLOWED FOR ACCIDENTS BY APPLICABLE ASME AND ANSI CODES WHEN CONTAINMENT IS REQUIRED. 3-5 OVEREXPOSURE TO RADIATION OF OPERATING PERSONNEL IN THE CONTROL ROOM,	NUCLEAR SAFETY DESIGN CRITERIA TYPE S-3 NUCLEAR SAFETY OPERATIONAL CRITERIA TYPE S-3 - - - - - - - - - - - - - VARIOUS INDUSTRY CODES IEEE-279 LOADING CRITERIA EMERGENCY AND FAULTED CONDITIONS SINGLE FAILURE CRITERION TESTABILITY CRITERIA	SAFETY ACTION TYPE S-3 - - - - - - - - - - - - - SCRAM CORE COOLING CONTAINMENT COOLIING STOP CONTROL ROD EJECTION LIMIT REACTIVIITY INSERTION RATE PRESSURE RELIEF REACTOR VESSEL ISOLATION ESTABLISH PRIMARY CONTAINMENT ESTABLISH SECONDARY CONTAINMENT ISOLATION RESTRICTION OF COOLANT LOSS RATE CONTROL ROOM ENVIRONMENTAL CONTROL.	SAFETY SYSTEMS – TYPE S-3 - - - - - - - - - - - - - PROTECTION SYSTEM GENERIC - - - - - - - - - - - - - ENGINEERED SAFEGUARDS - - - - - - - - - - - - - REACTOR PROTECTION SYSTEM CONTROL ROD DRIVE SYSTEM NEUTRON MONITORING SYSTEM PRESSURE RELIEF SYSTEM SAFETY VALVES REACTOR VESSEL ISOLATION CONTROL SYSTEM PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM PRIMARY CONTAINMENT SECONDARY CONTAINMENT MAIN STEAM LINE ISOLATION VALVES MAIN STEAM LINE FLOW RESTRICTOR HIGH PRESSURE COOLANT INJECTION SYSTEM AUTOMATIC DEPRESSURIZATION SYSTEM LOW PRESSURE COOLANT COOLANT INJECTION CORE SPRAY SYSTEM RHRS CONTAINMENT COOLING CONTROL ROD VELOCITY LIMITER CONTROL ROD DRIVE HOUSING SUPPORTS STANDBY GAS TREATMENT SYSTEM STANDBY ac POWER SYSTEM dc POWER SYSTEM MAIN STEAM LINE RADIATION MONITORING SYSTEM REACTOR BUILDING VENTILATION RADIATION MONITORING SYSTEM RHRS SERVICE WATER SYSTEM STANDBY LIQUID CONTROL SYSTEM	OPERATIONAL NUCLEAR SAFETY REQUIREMENTS – TYPE S-3 OPERATIONAL NUCLEAR SAFETY LIMITS – TYPE S-3 - - - - - - - - - - - - - TECHNICAL SPECIFICATIONS – TYPE S-3 - - - - - - - - - - - - - LIMITING SAFETY SYSTEM SETTINGS LIMITING CONDITIONS FOR OPERATION FOR PROTECTION SYSTEMS SURVEILLANCE REQUIREMENTS FOR PROTECTION SYSTEMS SURVEILLANCE REQUIREMENTS FOR NUCLEAR SYSTEM	3-1 INABILITY TO RETURN TO POWER OPERATION	POWER GENERATION DESIGN CRITERIA – TYPE PG-3 POWER GENERATION OPERATIONAL CRITERIA – TYPE PG-3 - - - - - - - - - - - - -	POWER GENERATION ACTION – TYPE PG-3 - - - - - - - - - - - - -	POWER GENERATION SYSTEMS – TYPE PG-3 - - - - - - - - - - - - -	OPERATIONAL POWER GENERATION REQUIREMENTS-TYPE PG-3 OPERATIONAL POWER GENERATION LIMITS- TYPE PG-3 - - - - - - - - - - - - - POST ACCIDENT RECOVERY PROCEDURES
4A. SPECIAL EVENT – SHUT DOWN FROM OUTSIDE THE CONTROL ROOM. 4B. SPECIAL EVENT – SHUT DOWN WITHOUT CONTROL RODS. 4C. SPECIAL EVENT – LOSS OF NORMAL HEAT SINK.	4-1 INABILITY TO SHUT DOWN THE REACTOR FROM OUTSIDE THE CONTROL ROOM. 4-2 INABILITY TO PERFORM EMERGENCY PROCEDURES BRING THE REACTOR TO THE COLD SHUTDOWN CONDITION FROM OUTSIDE THE CONTROL ROOM. 4-3 INABILITY TO SHUT DOWN THE REACTOR INDEPENDENT OF CONTROL RODS. 4-4 INABILITY TO SAFELY MAINTAIN THE PLANT IN THE SHUTDOWN CONDITION UPON LOSS OF THE NORMAL HEAT SINK.	NUCLEAR SAFETY DESIGN CRITERIA - TYPE S-4 NUCLEAR SAFETY OPERATIONAL CRITERIA – TYPE S-4 - - - - - - - - - - - - - SPECIAL SAFETY DESIGN CRITERIA SPECIAL SAFETY OPERATIONAL CRITERIA - - - - - - - - - - - - -	SAFETY ACTION TYPE S-4 - - - - - - - - - - - - - SPECIAL SAFETY ACTION - - - - - - - - - - - - - SHUT DOWN FROM OUTSIDE THE CONTROL ROOM COOL DOWN FROM OUTSIDE CONTROL ROOM SHUTDOWN WITHOUT CONTROL RODS MAINTAIN SHUTDOWN DURING REACTOR COOLDOWN COOLDOWN WITHOUT NORMAL HEAT SINK	SAFETY SYSTEMS TYPE – S-4 - - - - - - - - - - - - - SPECIAL SAFETY SYSTEMS - - - - - - - - - - - - - CONTROLS OUTSIDE CONTROL ROOM INDICATORS OUTSIDE CONTROL ROOM CONDENSATE STORAGE SYSTEM REACTOR CORE ISOLATION COOLING SYSTEM PRESSURE RELIEF SYSTEM REACTOR PROTECTION SYSTEM CONTROL ROD DRIVE SYSTEM STANDBY LIQUID CONTROL SYSTEM EMERGENCY HEAT SINK	OPERATIONAL NUCLEAR SAFETY REQUIREMENTS – TYPE S-4 OPERATIONAL NUCLEAR SAFETY LIMITS-TYPE S-4 - - - - - - - - - - - - - TECHNICAL SPECIFICATIONS TYPE S-4 - - - - - - - - - - - - - LIMITING CONDITIONS FOR OPERATION FOR SPECIAL SAFETY SYSTEMS SURVEILLANCE REQUIREMENTS FOR SPECIAL SAFETY SYSTEMS	4-1 INABILITY TO RETURN TO POWER OPERATION	POWER GENERATION DESIGN CRITERIA – TYPE PG-4 POWER GENERATION OPERTIONAL CRITERIA – TYPE PG-4 - - - - - - - - - - - - -	POWER GENERATION ACTION – TYPE PG-4 - - - - - - - - - - - - -	POWER GENERATION SYSTEMS – TYPE PG-4 - - - - - - - - - - - - -	OPERATIONAL POWER GENERATION REQUIREMENTS – TYPE PG-4 OPERATIONAL POWER GENERATION LIMITS – TYPE PG-4 - - - - - - - - - - - - - POST EVENT RECOVERY PROCEDURES

1.5 PRINCIPAL DESIGN CRITERIA

There are two ways of considering principal design criteria. One way is to consider the criteria on a system-by-system (or system group) basis. The second way is to consider criteria classification-by-classification as given in Table 1.4.2.

In the classification-by-classification approach, the criteria must be stated in sufficient detail to allow placement of each criterion into one classification category. Thus, there may be closely related criteria pertaining to any given system in each classification category. This is a natural outgrowth of the function (unacceptable result) approach to classification. The actual design of a system must reflect all of the criteria that pertain to it; thus, the less restrictive (but more important) criteria pertaining to the system in the classification approach will be masked by the more restrictive (and less important) criteria.

Safety analysis requires the information gained in the classification-by-classification approach to criteria, but system description is more easily understood through the system-by-system method. In this section both approaches to criteria are given; both are useful.

1.5.1 Principal Design Criteria, Classification-By-Classification

The principal architectural and engineering criteria for the design and construction of the plant are summarized below. The criteria are grouped according to the classification plan given in Table 1.4.2. Some of the more general criteria are so broad that they are applicable, at least in part, to more than one classification. In these very general cases, all of the affected classifications are indicated. Specific design bases and design features are detailed in other sections of this report. Criteria pertaining to operation of the plant are given in Appendix G.

1.5.1.1 General Criteria

Applicable Classifications	Criteria
PG-1, S-1, S-2, S-3	1. The plant shall be designed so that it can be fabricated, erected, and operated to produce electric power in

a safe and reliable manner. The plant design shall be in accordance with applicable codes and regulations.

S-1, S-2, S-3

2. The plant shall be designed in such a way that the release of radioactive materials to the environment is limited, so that the limits and guideline values of applicable regulations pertaining to the release of radioactive materials are not exceeded.

S-1, S-2, S-3, S-4

3. The reactor core and reactivity control system shall be designed so that control rod action shall be capable of bringing the core subcritical and maintaining it so, even with the rod of highest reactivity worth fully withdrawn and unavailable for insertion.

S-1, S-2, S-3

4. Adequate strength and stiffness with appropriate safety factors shall be provided so that a hazardous release of radioactive material shall not occur.

1.5.1.2 Power Generation Design Criteria, Type PG-1 (Planned Operation)

1. The nuclear system shall employ a General Electric BWR to produce steam for direct use in a turbine-generator.

2. The fuel cladding shall be designed to retain integrity as a radioactive material barrier for the design power range.

3. The fuel cladding shall be designed to accommodate without loss of integrity the pressures generated by the fission gases released from the fuel material throughout the design life of the fuel.

4. Heat removal systems shall be provided in sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions from plant shutdown to design power. The capacity of such system shall be adequate to prevent excessive fuel clad temperatures.

5. Control equipment shall be provided to allow the reactor to respond automatically to minor load changes.

6. It shall be possible to manually control the reactor power level.

7. Control of the nuclear system shall be possible from a single location.

8. Nuclear system process controls shall be arranged to allow the operator to rapidly assess the condition of the nuclear system and to locate process system malfunctions.

9. Fuel handling and storage facilities shall be designed to maintain adequate shielding and cooling for spent fuel.

10. Interlocks or other automatic equipment shall be provided as a backup to procedural controls to avoid conditions requiring the functioning of nuclear safety systems or engineered safeguards.

1.5.1.3 <u>Power Generation Design Criteria, Type PG-2 (Abnormal Operational Transients)</u>

1. The fuel cladding, in conjunction with other plant systems, shall be designed to retain integrity throughout any abnormal operational transient.

2. Heat removal systems shall be provided in sufficient capacity and operational adequacy to remove heat generated in the reactor core for any abnormal operational transient. The capacity of such systems shall be adequate to prevent excessive fuel clad temperatures.

3. Heat removal systems shall be provided to remove decay heat generated in the core under circumstances wherein the normal operational heat removal systems become inoperative. The capacity of such systems shall be adequate to prevent excessive fuel clad temperatures.

4. Standby electrical power sources shall be provided to allow removal of decay heat under circumstances where normal auxiliary power is not available.

5. Fuel handling and storage facilities shall be designed to prevent inadvertent criticality.

1.5.1.4 Nuclear Safety Design Criteria, Type S-1 (Planned Operation)

1. The plant shall be designed so that fuel failure during planned operation is limited to such an extent that, were the freed fission products released to the environs via the normal discharge paths for radioactive materials, the limits of 10CFR20 would not be exceeded.

2. The reactor core shall be designed so that its nuclear characteristics do not contribute to a divergent power transient.

3. The nuclear system shall be designed so there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the nuclear system with other appropriate plant systems.

4. Gaseous, liquid, and solid waste disposal facilities shall be designed so that the discharge and off-site shipment of radioactive effluents can be made in accordance with applicable regulations.

5. The design shall provide means by which plant operations personnel can be informed whenever limits on the release of radioactive material are exceeded.

6. Sufficient indications shall be provided to allow determination that the reactor is operating within the envelope of conditions considered by plant safety analysis.

7. Radiation shielding shall be provided and access control patterns shall be established to allow a properly trained operating staff to control radiation doses within the limits of applicable regulations in any mode of normal plant operation.

1.5.1.5 <u>Nuclear Safety Design Criteria, Type S-2 (Abnormal Operational Transients)</u>

1. The plant shall be designed so that fuel failure as a result of any abnormal operational transient is limited to such an extent that were the freed fission products released to the environs via the normal discharge paths for radioactive materials, the limits of 10CFR20 would not be exceeded.

2. Those portions of the nuclear system which form part of the nuclear system process barrier shall be designed to retain integrity as a radioactive material barrier following abnormal operational transients.

3. Nuclear safety systems shall act to assure that no damage to the nuclear system process barrier results from internal pressures caused by abnormal operational transients.

4. Where positive, precise action is immediately required in response to abnormal operational transients, such action shall be automatic and shall require no decision or manipulation of controls by plant operations personnel.

5. Essential safety actions shall be carried out by equipment of sufficient redundance and independence that no single failure of active components can prevent the required actions. For systems or components to which IEEE-279 (1968) is applicable, single failures of passive electrical components are considered, as well as

single failures of active components, in recognition of the higher anticipated failure rates of passive electrical components relative to passive mechanical components.

6. The design of nuclear safety systems shall include allowances for environmental phenomena at the site.

7. Provision shall be made for control of active components of nuclear safety systems from the control room.

8. Nuclear safety systems shall be designed to permit demonstration of their functional performance requirements.

9. Standby electrical power sources shall be provided to allow prompt reactor shutdown and removal of decay heat under circumstances where normal auxiliary power is not available.

10. Standby electrical power sources shall have sufficient capacity to power all nuclear safety systems requiring electrical power.

1.5.1.6 Nuclear Safety Design Criteria, Type S-3 (Accidents)

1. Those portions of the nuclear system which form part of the nuclear system process barrier shall be designed to retain integrity as a radioactive material barrier following accidents. For accidents in which one breach in the nuclear system process barrier is postulated, such breach shall not cause additional breaches in the nuclear system process barrier.

2. Engineered safeguards shall act to assure that no damage to the nuclear system process barrier results from internal pressures caused by an accident.

3. Where positive, precise action is immediately required in response to accidents, such action shall be automatic and shall require no decision or manipulation of controls by plant operations personnel.

4. Essential safety actions shall be carried out by equipment or sufficient redundance and independence that no single failure of active components can prevent the

required actions. For systems or components to which IEEE-279 (1968) is applicable, single failures of passive electrical components are considered, as well as single failures of active components, in recognition of the higher anticipated failure rates of passive electrical components relative to passive mechanical components.

5. Features of the plant which are essential to the mitigation of accident consequences shall be designed so that they can be fabricated and erected to quality standards which reflect the importance of the safety action to be performed.

6. The design of engineered safeguards shall include allowances for natural phenomena at the site.

7. Provision shall be made for control of active components of engineered safeguards from the control room.

8. Engineered safeguards shall be designed to permit demonstration of their functional performance requirements.

9. A primary containment shall be provided that completely encloses the reactor vessel.

10. The primary containment shall be designed to retain integrity as a radioactive material barrier during and following accidents that release radioactive material into the primary containment volume.

11. It shall be possible to test primary containment operability and leak tightness at periodic intervals.

12. A secondary containment shall be provided that completely encloses both the primary containment and fuel storage areas.

13. The secondary containment shall be designed to act as a radioactive material barrier under the same conditions that require the primary containment to act as a radioactive material barrier.

14. The secondary containment shall be designed to act as a radioactive material barrier, if required, whenever the

primary containment is open for expected operational purposes.

15. The primary and secondary containments, in conjunction with other engineered safeguards, shall act to prevent the radiological effects of accidents resulting in the release of radioactive material to the containment volumes from exceeding the guideline values of applicable regulations.

16. Provisions shall be made for the removal of energy from within the primary containment as necessary to maintain the operability of the containment system following accidents that release energy to the primary containment.

17. Piping that both penetrates the primary containment structure and could serve as a path for the uncontrolled release of radioactive material, which could result in a dose to the environs in excess of guideline values of applicable regulations, shall be automatically isolated whenever such uncontrolled radioactive material release is threatened. Such isolation shall be effected in time to prevent radiological effects from exceeding the guideline values of applicable regulations.

18. Core standby cooling systems shall be provided to prevent excessive fuel clad temperatures as a result of a loss of coolant accident (LOCA).

19. The core standby cooling systems shall provide for continuity of core cooling over the complete range of postulated break sizes in the nuclear system process barrier.

20. The core standby cooling systems shall be diverse, reliable, and redundant.

21. Operation of the core standby cooling systems shall be initiated automatically when required regardless of the availability of off-site power supplies and the normal generating system of the plant.

22. Standby electrical power sources shall have sufficient capacity to power all engineered safeguards requiring electrical power.

23. The control room shall be shielded against radiation so that occupancy under accident conditions is possible.

1.5.1.7 Nuclear Safety Design Criteria, Type S-4 (Special Event)

1. In the event that the control room becomes inaccessible, it shall be possible to bring the reactor from power range operation to a shutdown condition by manipulation of the local controls and equipment which are available outside of the control room. Furthermore, plant design shall not preclude the ability, in this event, to bring the reactor to a cold shutdown condition from the hot shutdown condition.

1.5.1.8 Nuclear Safety Design Criteria, Type S-5 (Special Event)

1. Backup reactor shutdown capability shall be provided independent of normal reactivity control provisions. This backup system shall have the capability to shut down the reactor from any normal operating condition, and subsequently to maintain the shutdown condition.

1.5.2 Principal Design Criteria, System-By-System

The principal architectural and engineering criteria for design are summarized below on a system-by-system or system group basis. The system-by-system presentation facilitates the understanding of the actual design of any one system, but significant distinctions in the importance to safety of different criteria pertaining to a system cannot be clearly made as they are in the classification-by-classification presentation. To make consistent judgments regarding plant safety, the classification-by-classification approach to criteria must be used.

In the system-by-system presentation of criteria, only the most restrictive of any related criteria are stated for a system. Where the most restrictive criterion is one which is classified as a power generation consideration in Table 1.4.2, less restrictive but more important safety criteria may be hidden (not stated) in the system-by-system presentation.

1.5.2.1 General Criteria

1. The plant shall be designed so that it can be fabricated, erected, and operated to produce electric

power in a safe and reliable manner. The plant design shall be in accordance with applicable codes and regulations.

2. The plant shall be designed in such a way that the release of radioactive materials to the environment is limited, so that the limits and guideline values of applicable regulations pertaining to the release of radioactive materials are not exceeded.

1.5.2.2 Nuclear System Criteria

1. The nuclear system shall employ a General Electric BWR to produce steam for direct use in a turbine-generator.

2. The fuel cladding shall be designed to retain integrity as a radioactive material barrier for the design power range and for any abnormal operational transient.

3. Those portions of the nuclear system which form part of the nuclear system process barrier shall be designed to retain integrity as a radioactive material barrier following abnormal operational transients and accidents. For accidents in which one breach in the nuclear system process barrier is postulated, such breach shall not cause additional breaches in the nuclear system process barrier.

4. The fuel cladding shall be designed to accommodate without loss of integrity the pressures generated by the fission gases released from the fuel material throughout the design life of the fuel.

5. Heat removal systems shall be provided with sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions from plant shutdown to design power, and for any abnormal operational transient. The capacity of such systems shall be adequate to prevent excessive fuel clad temperatures.

6. Heat removal systems shall be provided to remove decay heat generated in the core under circumstances wherein the normal operational heat removal systems become inoperative. The capacity of such systems shall be adequate to prevent excessive fuel clad temperatures.

7. The reactor core and reactivity control system shall be designed so that control rod action shall be capable of bringing the core subcritical and maintaining it so, even with the rod of highest reactivity worth fully withdrawn and unavailable for insertion.

8. The nuclear system shall be designed so there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the nuclear system with other appropriate plant systems.

9. The reactor core shall be designed so that its nuclear characteristics do not contribute to a divergent power transient.

1.5.2.3 Power Conversion System Criteria

1. The power conversion system components shall be designed to produce electrical power from the steam coming from the reactor, condense the steam into water, and return the water to the reactor as heated feedwater, with a major portion of its gases and particulate impurities removed.

2. The power conversion system components shall be designed to assure that any fission products or radioactivity associated with the steam and condensate are safely contained inside the system, or are released under control conditions within the guideline limits of applicable regulations.

1.5.2.4 Electrical Power System Criteria

1. The station electrical power systems shall be designed to efficiently deliver the electrical power generated to the 500-kV transmission system.

2. Sufficient normal and standby auxiliary sources of electrical power shall be provided to attain an orderly shutdown and continued maintenance of the station in a safe condition. The capacity of the power sources shall be adequate to accomplish all required engineered safeguard functions under postulated design basis accident conditions.

1.5.2.5 Radioactive Waste Disposal Criteria

1. Gaseous, liquid, and solid waste disposal facilities shall be designed so that the discharge and off-site shipment of radioactive effluents can be made in accordance with applicable regulations.

2. The design shall provide means whereby plant operations personnel are informed whenever operational limits on the release of radioactive material are exceeded.

1.5.2.6 Nuclear Safety Systems and Engineered Safeguards Criteria

1.5.2.6.1 General

1. Nuclear safety systems shall act in response to abnormal operational transients to limit fuel damage such that, were the freed fission products released to the environs via the normal discharge paths for radioactive material, the limits of 10CFR20 would not be exceeded.

2. Nuclear safety systems and engineered safeguards shall act to assure that no damage to the nuclear system process barrier results from internal pressures caused by abnormal operational transients or accidents.

3. Where positive, precise action is immediately required in response to accidents, such action shall be automatic and shall require no decision or manipulation of controls by plant operations personnel.

4. Essential safety actions shall be carried out by equipment of sufficient redundance and independence that no single failure of active components can prevent the required actions. For systems or components to which IEEE-279 (1968) is applicable, single failures of passive electrical components will be considered, as well as single failures of active components, in recognition of the higher anticipated failure rates of passive electrical components relative to passive mechanical components.

5. Features of the plant which are essential to the mitigation of accident consequences shall be designed so that they can be fabricated and erected to quality standards which reflect the importance of the safety function to be performed.

6. The design of nuclear safety systems and engineered safeguards shall include allowances for natural phenomena at the site.

7. Provision shall be made for control of active components of nuclear safety systems and engineered safeguards from the control room.

8. Nuclear safety systems and engineered safeguards shall be designed to permit demonstration of their functional performance requirements.

1.5.2.6.2 Containment and Isolation Criteria

1. A primary containment shall be provided that completely encloses the reactor vessel.

2. The primary containment shall be designed to retain integrity as a radioactive material barrier during and following accidents that release radioactive material into the primary containment volume.

3. It shall be possible to test primary containment operability and leak tightness.

4. A secondary containment shall be provided that completely encloses both the primary containment and fuel storage areas.

5. The secondary containment shall be designed to act as a radioactive material barrier under the same conditions that require the primary containment to act as a radioactive material barrier.

6. The secondary containment shall be designed to act as a radioactive material barrier, if required, whenever the primary containment is open for expected operational purposes.

7. The primary and secondary containments, in conjunction with other engineered safeguards, shall act to prevent the radiological effect of accidents resulting in the release of radioactive material to the containment volumes from exceeding the guideline values of applicable regulations.

8. Provisions shall be made for the removal of energy from within the primary containment as necessary to maintain the operability of the containment system following accidents that release energy to the primary containment.

9. Piping that penetrates the primary containment structure and could serve as a path for the uncontrolled release of radioactive material, which could result in a dose to the environs in excess of guideline values of applicable regulations, shall be automatically isolated whenever such uncontrolled radioactive material release is threatened. Such isolation shall be effected in time to prevent radiological effects from exceeding the guideline values of applicable regulations.

1.5.2.6.3 Core Standby Cooling Criteria

1. Core standby cooling systems shall be provided to prevent excessive fuel clad temperature as a result of a postulated design basis LOCA.

2. The core standby cooling systems shall provide for continuity of core cooling over the complete range of postulated break sizes in the nuclear system process barrier.

3. The core standby cooling systems shall be diverse, reliable, and redundant.

4. Operation of the core standby cooling systems shall be initiated automatically when required, regardless of the availability of off-site power supplies and the normal generating system of the plant.

1.5.2.6.4 Standby Power Criteria

1. Standby electrical power sources shall be provided to allow reactor shutdown and removal of decay heat under circumstances where normal auxiliary power is not available.

2. The standby electrical power sources shall also provide sufficient power to all engineered safeguards requiring electrical power.

1.5.2.7 Reactivity Control Criteria

1. Backup reactor shutdown capability shall be provided independent of normal reactivity control provisions. This backup system shall have the capability to shut down the reactor from any operating condition, and subsequently to maintain the shutdown condition.

2. In the event that the control room is inaccessible, it shall be possible to bring the reactor from power range operation to a hot shutdown condition by manipulation of controls and equipment which are available outside of the control room. Furthermore, plant design shall not preclude the ability, in this event, to bring the reactor to a cold shutdown condition from the hot shutdown condition.

1.5.2.8 Process Control Systems Criteria

1.5.2.8.1 Nuclear System Process Control Criteria

1. Control equipment shall be provided to allow the reactor to respond automatically to minor load changes.

2. It shall be possible to manually control the reactor power level.

3. Control of the nuclear system shall be possible from a single location.

4. Nuclear system process controls shall be arranged to allow the operator to rapidly assess the condition of the nuclear system and to locate process system malfunctions.

5. Interlocks or other automatic equipment shall be provided as a backup to procedural controls to avoid conditions requiring the actuation of nuclear safety systems or engineered safeguards.

1.5.2.8.2 Power Conversion Systems Process Control Criteria

1. Controls shall be provided to maintain temperature and pressure within design limitations.

2. Controls shall be designed to provide indication of system trouble.

3. Control of the power conversion system shall be possible from a single location.

4. Controls shall be provided to ensure adequate cooling of power conversion system equipment.

5. Controls shall be provided to ensure adequate condensate purity.

1.5.2.8.3 Electrical Power System Process Control Criteria

1. Controls shall be provided to ensure that sufficient electrical power is provided for startup, normal operation, and to attain shutdown and maintain the plant in a safe condition.

2. Control of the electrical power system shall be possible from the main control room.

1.5.2.9 Auxiliary Systems Criteria

1. Multiple independent auxiliary systems shall be provided for the purpose of cooling and servicing the plant, the reactor, and the plant containment systems under various normal and abnormal conditions.

2. Fuel handling and storage facilities shall be designed to prevent criticality of and to maintain adequate shielding and cooling for spent fuel.

1.5.2.10 Shielding and Access Control Criteria

1. Radiation shielding shall be provided and access control patterns shall be established to allow a properly trained operating staff to control radiation doses within the limits of applicable regulations in any mode of normal plant operation.

2. The control room shall be shielded against radiation so that occupancy under accident conditions is permitted.

1.5.2.11 Structural Loading Criteria

Plant structures shall be designed with adequate strength and stiffness, and with appropriate safety factors, so that a hazardous release of radioactive material shall not occur.

1.6 PLANT DESCRIPTION

1.6.1 General

1.6.1.1 Site and Environs

1.6.1.1.1 Location and Size of Site

The site is located partly in Peach Bottom Township, York County, partly in Drumore Township, Lancaster County, and partly in Fulton Township, Lancaster County, in southeastern Pennsylvania on the westerly shore of Conowingo Pond at the mouth of Rock Run Creek. It is about 38 mi north-northeast of Baltimore, Maryland, and 63 mi west-southwest of Philadelphia, Pennsylvania. Figures 2.2.1 through 2.2.4 show the site location with respect to surrounding communities.

1.6.1.1.2 Site Ownership

The licensee owns the 620-acre property except that the immediate area on which Units 2 and 3 stand are owned by the licensee and PSEG Nuclear, LLC, as tenants in common.

1.6.1.1.3 Activities at the Site

Approximately 700 ft and 1,000 ft downstream from Units 2 and 3, respectively, and included in their exclusion area is Unit 1. Unit 1 is now in SAFSTOR status that allows it to be safely stored and subsequently decontaminated to levels that permit release of the facility for unrestricted use.

An Independent Spent Fuel Storage Installation is located approximately 1500 feet south of Rock Run Creek and the Training Center.

A 500-kV unattended substation is located within the site property about 2,000 ft northwest of Unit 3, and another 500-kV unattended substation is located about 1,600 ft south-southwest of Unit 2.

1.6.1.1.4 Access to Site

U.S. Highway Route 1 intersects Maryland Route 623 about 8 miles south-southeast of the site. From this point, 7 miles of bituminous all-weather road leads to the Atom Road, a hard-surface road which leads to Unit 1 plant area. Approximately 1.5 miles beyond Atom Road is Lay Road, a hard-surface, bituminous road which leads to Units 2 and 3. This road passes close by the north

500-kV substation. Two dirt roads also enter the exclusion area and have limited access. A spur track of the former Maryland and Pennsylvania Railroad was extended to the site and has since been abandoned.

1.6.1.1.5 Description of the Environs

1. General

   The plant is located in a cut taken from a formerly wooded, gentle slope near the point at which Rock Run Creek discharges into Conowingo Pond. Within a 1-mi radius of the plant, and on both sides of the pond, steep sloping hills rise directly up to about 300 ft above plant grade with rock outcroppings apparent at many locations. Plant grade is approximately 116 ft Conowingo Datum (C.D.).

2. Population

   The area within a 10-mi radius of the site, when proposed, had a total population of less than 25,000. There is no large center of population within this radius. The city of Lancaster, Pennsylvania lies 19.4 mi north of the site, has a population according to the 1970 Census of Population of 57,690 residents, and is the city located at the corresponding "population center distance":, i.e., "the distance from the reactor to the nearest boundary of a densely populated center containing more than 25,000 residents." For the PBAPS site this distance is 17.9 mi.

3. Land Use

   The major portion of the land in the five-county area surrounding the site is used for farming.

1.6.1.1.6 Geology

Geological studies were conducted which included geologic mapping, geologic reconnaissance of the site and environs, a study of the surface water and ground water conditions of the site and surrounding area, a series of test borings and test pits, literature research, and interviews with local geologists. Core borings were drilled in the plant area and at other locations in the site area. Soil samples of the overburden and NX-size cores of the underlying rock were extracted and tested to evaluate their physical properties. The conclusions were that no geologic features exist which adversely influence or affect the use of this site for a nuclear facility, and that bedrock at the site is sound and provides adequate foundation support for all major structures.

1.6.1.1.7 <u>Seismology</u>

Seismologic studies, including field geophysical surveys to evaluate the dynamic properties of the foundation materials, literature research, analyses of the tectonics of the region, and development of a seismic design parameters, were conducted by the consultant. Conclusions reached as a result of the studies includes (1) no major earthquakes have had epicenters closer than 350 mi, and (2) minor earthquake activity closer to the site has been observed. It is concluded that the site may be subjected to slight ground motion during the life of the plant, but that the foundation rock will respond well and will not experience adverse consolidation effects or any reduction in strength when subjected to earthquake motion.

Critical plant structures were designed in accordance with the earthquake criteria recommended by the consultant.

1.6.1.1.8 <u>Hydrology</u>

The Susquehanna River drains an area of 27,500 sq mi in New York, Pennsylvania, and Maryland. The Peach Bottom site is about 14 mi north of the river's mouth at the head of the Chesapeake Bay. At this point the drainage area is approximately 27,000 sq mi. Along the lower 35 mi, where the river flows between steep hills, are located four major hydroelectric plants: Safe Harbor, Holtwood, Muddy Run, and Conowingo. Peach Bottom is on Conowingo Pond 8.5 mi above Conowingo Dam and 6 mi below Holtwood Dam. The pond varies in width between 0.6 and 1.5 mi and contains when full to Elevation 109.25 ft (C.D.) 246,000 acre feet or 82 billion gal of water. The top 10 ft, or about 80,000 acre feet of water, are used as pondage to regulate power generation.

The observed natural river flows on the Susquehanna River have ranged from a minimum daily average (1964) of 1,400 cubic feet per second (cfs) to a peak (1977) of 972,000 cfs. The average discharge is 36,200 cfs. Conowingo Dam passed the peak flood without difficulty. Peak flows are now reduced somewhat by flood control dams on upland tributaries.

During low flow, when water is standing in Conowingo Pond, wind and temperature variations cause general diffusion and mixing.

Turning on of the hydroplants sets up eddies which are gradually damped as operation continues. Dye tracer studies have been made on a model of Conowingo Pond to determine the circulation and displacement of water under different conditions of river flow and plant operation.

The river below Peach Bottom is at present used as a source of domestic water supply for the city of Havre deGrace, the Perry Point Veteran's Hospital, the Conowingo Power Plant, the city of Baltimore, and the city of Chester.

The plant site is in the Piedmont region where ground water occurs in the relatively shallow overburden and may be collected in quantities suitable for domestic use. These small ground water supplies are derived from rainwater that soaks into and through the soil in limited areas surrounding each well. This water percolates into drilled wells through fissures and cracks that thin out and disappear rapidly with depth in the rock. Ground water moves in the overburden and rock fissures in the direction of the nearest stream or spring. Its discharge under natural conditions supports the continuous dry weather flow of the abundant small streams in the area.

At Peach Bottom the water table rises from Conowingo Pond, up through the building site into the hill to the rear. Under these conditions ground water discharges directly into Conowingo Pond.

Based upon the results of studies, it is concluded that there are no hydrologic conditions which are unfavorable to the location and operation of a nuclear facility at this site.

1.6.1.1.9 Meteorology

The meteorology of the site and surrounding area has been studied extensively beginning in 1959 for Unit 1 and expanded in 1967 for Units 2 and 3. Nothing in the existing site data or the general records of the area suggests any particularly unusual or disturbing meteorological features.

Instrumentation installed at the site has defined wind patterns along the shore, on the slope behind the units, over the Conowingo Pond, and at higher elevations. Analysis of data confirms the existence of two rather distinct dispersion zones, one pertaining to the valley and the other representative of the more synoptic patterns above it.

There is clear evidence of channeling in the valley and of slope-flow on the western shore of Conowingo Pond where the plant is located, but these patterns are not difficult to define. Close to the steep, western side of the valley, virtually all stable atmospheric conditions are accompanied by down slope motion, with the air flowing out over the pond and then either up or downstream. In unstable conditions, accompanied by local solar heating, the pattern is more varied, but channeling is also quite apparent.

At higher elevations, the wind flow becomes progressively divorced from valley effects until, at Elevation 688 ft mean sea level (MSL) near the top of the microwave tower, it is difficult to see the channeling, and the distribution of wind directions and speeds is typical of the unrestricted flow in the eastern United States. The frequency of stable dispersion conditions is normal for a location of this type, averaging approximately 30 percent of all hours annually and reaching a peak of about 45 percent during the most stable months.

The analysis of data has not uncovered any unanticipated specialized conditions, and the fate of any release over or above the site can be estimated from available data.

1.6.1.1.10 Design Bases Dependent Upon Site and Environs

Information relating to the site and environs has been summarized and this information, as applicable, has been used in the design of the plant. The several design features which are dependent upon or affected by the site characteristics are summarized below:

a. Off-Gas System

   Based upon available meteorological data, plant operational characteristics, and the stack design, the off-site doses arising from plant operation will be in compliance with 10CFR20.

b. Liquid Waste Effluent

   Liquid wastes are discharged to the Conowingo Pond through the circulating water discharge canal. The concentration of such wastes at the point of discharge into the pond will be in compliance with 10CFR20.

c. Wind Loading Design

Plant structures are designed to withstand the wind loads described in Appendix C. Features of the plant important to continuity of core cooling are either designed to withstand tornados having 300 mph winds, or are contained in a structure which is designed to the same criteria.

d. <u>Central Power Supply</u>

Two independent off-site sources of power are available for startup or emergency plant loads. However, to account for possible power line outages, the plant is designed so that it may be shut down and maintained in a safe condition without an off-site power source.

e. <u>Hydrology</u>

The plant is designed such that the theoretical maximum water level due to probable maximum flood shall not prevent safe shutdown or removal of residual heat. To provide for effective use of Conowingo Pond as a heat sink, earth dikes and mechanical-draft cooling towers are used.

f. <u>Geology</u>

The geology of the area indicates that the underlying bedrock is capable of supporting loads imposed by the plant structures.

g. <u>Seismic Design</u>

The seismic design for structures and equipment important to the plant safety features is based on dynamic analyses using response spectrum curves for the site based on ground motion accelerations of 0.05g (design earthquake) and 0.12g (maximum credible earthquake).

The natural periods of vibration are calculated for buildings and equipment which are vital to the safety of the plant. Damping factors are based upon the materials and methods of construction used.

Seismic design is based on normal allowable stress as set forth in applicable codes, but is more conservative for critical structures because the usual one-third increase in normal allowable working stresses is not used. As an additional requirement, the design is such that a safe shutdown can be made with the containment and heat removal facilities intact during the maximum credible earthquake. However, the stress under this condition may approach yield stress.

1.6.1.2 Facility Arrangement

The plant is arranged such that the main turbine-generator axis is parallel to the pond, and the reactor buildings are located to the west, or land side, of the turbine building. The overall arrangement is shown in Drawing C-1, "Site Plan," and Drawing C-2, "Plot Plan." The reactor buildings are separated by the reactor building auxiliary bays and the radwaste building. The main control room is located in the turbine building between the two turbine-generator units.

The administration office and shop is located between the turbine building and the intake pump structure. The water treatment building is located adjacent to, and to the south of, the pump structure. The diesel generator building is located to the south of Unit 2. The warehouse building is located east of the emergency cooling towers.

The stack is located on the hill about 700 ft to the west of the plant.

The circulating water cooling tower facility is located to the south-east of Unit 1, along the river bank, separated by the discharge canal on the land side.

Units 2 and 3 are located a minimum distance of 2,600 ft from the nearest site boundary. The minimum distance to the site boundary in a downstream direction is about 3,300 ft, and in an inland direction about 3,100 ft from Unit 2. The minimum distance across the pond from either the Unit 2 or the Unit 3 reactor to the far shore of the pond (to the northeast) is 7,600 ft. The minimum distance from the stack to the site boundary is 2,350 ft.

Approximately 500 ft downstream from Units 2 and 3, and included in the exclusion area, is PBAPS Unit 1, a decommissioned high temperature, gas-cooled, 40 MWe nuclear power plant (Docket No. 50-171). The Independent Spent Fuel Storage Installation is located approximately 1500 ft. south of the Rock Run Creek.

1.6.1.3 Nuclear System

The nuclear system includes a single cycle, forced circulation, General Electric BWR producing steam for direct use in the steam turbine (Figure 1.6.1). A heat balance showing the major parameters of the nuclear system for the rated power condition is shown in Figure 1.6.2.

1.6.1.3.1 <u>Reactor Core and Control Rods</u>

The fuel for the reactor core consists of slightly enriched uranium-dioxide pellets contained in sealed Zircaloy-2 tubes. These fuel rods are assembled into individual fuel assemblies.

Gross control of the core is achieved by movable, bottom-entry control rods. The control rods are of cruciform shape and are dispersed throughout the lattice of fuel assemblies. The rods are controlled by individual hydraulic systems.

1.6.1.3.2 <u>Reactor Vessel and Internals</u>

The reactor vessel contains the core and supporting structure, the steam separators and dryers, the jet pumps, the control rod guide tubes, distribution lines for the feedwater, core spray, and standby liquid control, the in-core instrumentation, and other components. The main connections to the vessel include the steam lines, the coolant recirculation lines, feedwater lines, control rod drive housings, and core standby cooling lines.

The reactor vessel is designed and fabricated in accordance with applicable codes for a pressure of 1,250 psig. The nominal operating pressure is 1,050 psia in the steam space above the separators.

The reactor core is cooled by demineralized water which enters the lower portion of the core and boils as it flows upward around the fuel rods. The steam leaving the core is dried by steam separators and dryers located in the upper portion of the reactor vessel. The steam is then directed to the turbine through the main steam lines. Each steam line is provided with two isolation valves in series, one on each side of the primary containment barrier.

1.6.1.3.3 <u>Reactor Recirculation System</u>

The reactor recirculation system pumps reactor coolant through the core to remove the energy generated in the fuel.

This is accomplished by two recirculation loops external to the reactor vessel but inside the primary containment. Each loop has one motor-driven recirculation pump. Recirculation pump speed can be varied to allow some control of reactor power level through the effects of coolant flow rate on moderator void content.

1.6.1.3.4 Residual Heat Removal System

The residual heat removal system (RHRS) is a system of pumps, heat exchangers, and piping that fulfills the following functions:

1. Removal of residual heat during and after plant shutdown.

2. Injection of water into the reactor vessel, following a LOCA, rapidly enough to reflood the core and prevent excessive fuel clad temperatures, independent of other core cooling systems. This is discussed in paragraph 1.6.2, "Nuclear Safety Systems and Engineered Safeguards."

3. Removal of heat from the primary containment following a LOCA to limit the increase in primary containment pressure. This is accomplished by cooling and recirculating the water inside the primary containment. The redundancy of the equipment provided for containment cooling is further extended by a separate part of the RHRS which sprays cooling water into the drywell. This latter capability is discussed in paragraph 1.6.2.12.

1.6.1.3.5 Reactor Water Cleanup System

A reactor water cleanup system, which includes a filter-demineralizer, is provided to clean up the reactor water, to reduce the amounts of activated corrosion products in the water, and to remove reactor coolant from the nuclear system under controlled conditions.

1.6.1.4 Power Conversion System

Each unit utilizes a power conversion system, including a turbine-generator, a main condenser, air ejector and turbine steam packing exhauster condensers, condensate filter-demineralizers, and a feedwater system, to produce electrical power from the steam coming from the reactor, condense the steam into water, and return the feedwater to the reactor. The heat rejected to the main condenser is removed by the circulating water system

(Figure 1.6.1). The turbine-generator heat balance for guaranteed power is shown in Figure 1.6.3.

1.6.1.4.1 Turbine-Generator

The high-pressure turbines (1 each per unit) are General Electric casing and Alstom steam path tandem-compound, six-flow, non-reheat, 1,800-rpm double flow. The low pressure turbines (3 each per unit) are 1,800-rpm, tandem-compound, six flow, non-reheat steam turbines manufactured by Alstom. Exhaust steam from the high-pressure turbine passes through moisture separators before entering the three low-pressure turbines. Steam is extracted for five stages of feedwater heating and also supplies three reactor feed pump drive turbines. Turbine controls include a speed governor, steam admission valves, stop valves, and supervisory and operating instruments. The generator is a direct-driven, conductor-cooled unit with a direct-driven exciter.

1.6.1.4.2 Turbine Bypass System

A bypass system passes steam directly to the main condenser under the control of the pressure regulator. Steam is bypassed to the main condenser whenever the reactor steaming rate exceeds the loading of the turbine-generator.

1.6.1.4.3 Main Condenser

Three horizontal, single-pass, single-pressure deaerating type, divided water box condenser sections per unit are provided to condense the steam from each low-pressure turbine. A condenser section is located below the low-pressure elements of the turbine, with the tubes oriented transversely to the turbine-generator axis. The hotwells of each condenser are designed to provide a minimum condensate retention time of 2 min.

Deaeration removes dissolved gases entrained in the condensate draining to the hotwell. The oxygen content leaving the condenser hotwells should not exceed 0.0035 cc/liter at standard conditions. During low-load operation, steam may be used to assist in deaeration.

1.6.1.4.4 Main Condenser Air Ejector and Turbine Steam Sealing Systems

Two 2-stage steam jet air ejectors, complete with inter- and after-condensers, are provided for each unit to remove air and non-condensibles from the main condensers. One motor-driven vacuum pump is provided for each unit to produce a vacuum in the condensers during startup. Non-condensible gases removed from the main condenser system are discharged to the off-gas stack after a holdup time for decay of radioactive gases.

Each turbine gland seal system includes a steam seal regulator and two steam packing exhausters, consisting of an exhaust blower and one condenser, for control of shaft leakage. This system discharges non-condensible gases from the gland seal system to the stack through a piping system which provides holdup time for decay of radioactive gases. The mechanical vacuum pump discharges to the off-gas stack.

1.6.1.4.5 Circulating Water and Cooling Tower System

Three circulating water pumps per unit deliver water to the condenser water boxes. The pump pits are sectionalized to permit dewatering of one pit for maintenance while the remaining pumps are in operation.

The circulating water is screened to intercept and remove debris at the screen structure along the pond, about 700 ft east of the pump structure; it is again screened at the inlet to the circulating water pumps. Additional screens are provided at the pump structure to protect the high-pressure, emergency, and normal service water pumps.

The flow out of the condensers is pumped to three mechanical-draft cooling towers.

1.6.1.4.6 Condensate Filter-Demineralizer System

Each unit is furnished with a full-flow condensate filter-demineralizer system to ensure the reactor receives water of the required purity. Corrosion products that originate in the turbine, condenser, piping system, and shell side of the feedwater heaters are removed from the condensate by this system. The system also protects the reactor against condenser tube leaks and removes impurities which might enter the condensate system with makeup water.

The demineralizer vessels are located in a shielded area. Spent resins are sluiced from a demineralizer vessel to a receiving tank in the radwaste system for disposal.

The condensate filter demineralizer system may operate with 2 out of 12 filter demineralizers not precoated with resin.

1.6.1.4.7 Condensate and Feedwater System

The condensate and feedwater system takes condensate from the main condenser and delivers it to the reactor to maintain reactor water level. Three condensate pumps per unit take suction from the condenser hotwell and discharge, in series, through the steam jet air ejector inter- and after-condensers, the turbine gland seal condenser, the condensate demineralizer system, and five stages of feedwater heaters to the suction of the reactor feed pumps.

The feedwater heaters are of the closed shell and tube type. All heaters, with the exception of the heaters utilizing steam from the turbine's lowest pressure extraction stage, are provided with integral drain coolers. All drains cascade by pressure differential from the heater through the drain cooler to the next lower pressure heater and drain cooler and finally to the main condenser.
Three horizontal reactor feed pumps, with direct-connected variable speed turbine drives, are provided per unit. The dual-admission turbines normally take steam from the main turbine crossaround, with the auxiliary source from the main steam line. A three-element control system regulates feedwater to the reactor by controlling the admission of steam to the turbines.

1.6.1.5 Electrical Power System

Each generator is connected to a transformer bank which steps up from generator voltage (22 kV) to 500 kV. Each transformer bank is connected into a separate 500-kV substation. The substations are spaced approximately 3,300 ft apart with two tie lines between them. Multiple outgoing lines tie these substations into the 500-kV grid in the area.

The normal auxiliary power supply for the plant is from the unit auxiliary power transformer connected to the generator leads. Start-up and the emergency auxiliary power are provided from three offsite sources. One source is the Nottingham-Cooper 220 kV line which is stepped down to 13kV through a 220/13 kV transformer. The second source is the Newlinville-Peach Bottom 220 kV line stepped down to 13 kV through a 220/13kV transformer. The third source is the Muddy Run-Peach Bottom 500/220 kV, 1000 MVA autotransformer tertiary at 13 kV. Only two of the offsite sources are connected to the emergency auxiliary transformers, the third serves non-essential loads and is a spare connectable source.

The main generator leads are isolated-phase, metal-clad bus from generator terminals to transformer terminals.

1.6.1.6 Radioactive Waste Systems

The radioactive waste systems are designed to control the release of plant produced radioactive material to within the limits specified in 10CFR20. This is done by various methods such as collection, filtration, holdup for decay, dilution, and concentration. The methods employed for the controlled release of these contaminants are dependent primarily upon the state of the material: liquid, solid, or gaseous.

1.6.1.6.1 Liquid Radwaste System

The liquid radwaste system collects, treats, stores, and disposes of all radioactive liquid wastes. These wastes are collected in sumps and drain tanks at various locations throughout the plant and then transferred to the appropriate collection tanks in the radwaste building for treatment, storage, and disposal. Wastes to be discharged from the system are processed on a batch basis, each batch being processed by methods appropriate for the quality and quantity of materials determined to be present. Processed liquid wastes may be returned to the condensate system or discharged to the environs through the circulating water discharge canal. The liquid wastes in the discharge canal are diluted with condenser effluent circulating water to achieve a permissible concentration.

Equipment is selected, arranged, and shielded to permit operation, inspection, and maintenance with minimum personnel exposure. For example, tanks and processing equipment which will contain significant radiation sources are located behind shielding, and sumps, pumps, instruments, and valves are located in controlled access rooms or spaces. Processing equipment is selected and designed to require a minimum of maintenance.

Protection against accidental discharge of liquid radioactive waste is provided by valving redundance, instrumentation for detection, alarms of abnormal conditions, and procedural controls. Additionally, the radioactive discharges must be pumped to the environs.

1.6.1.6.2 Solid Radwaste System

Solid wastes originating from nuclear system equipment are stored for radioactive decay in the fuel storage pool and prepared for offsite shipment in approved shipping containers. Examples of these wastes are spent fuel, spent control rods, in-core ion chambers, etc.

Approved spent fuel may also be removed from the spent fuel pool and stored in an approved dry cask storage system located on site at the Independent Spent Fuel Storage Installation.

Process solid wastes are collected, dewatered, and prepared for temporary onsite storage or offsite shipment in approved containers. Examples of these solid wastes are filter residue, spent resins, paper, air filters, rags, and used clothing.

1.6.1.6.3 Gaseous Radwaste System

The gaseous radwaste system collects, processes, and delivers gases from each main condenser air ejector, startup vacuum pump, and gland seal condenser to the stack for elevated release to the environment. Gases from each main condenser air ejector are passed through a recombiner-adsorber train and high efficiency filters and exhausted through the stack. The adsorber train consists of ambient charcoal delay beds which provide time for decay of radioactive gases. In addition, the delay provides the operator time to take appropriate action in the event the noble gas release rate exceeds permissible limits.

Gland seal and startup vacuum pump gases are delayed to allow sufficient decay of N-16 and O-19, and then passed directly to the stack for release.

1.6.2 Nuclear Safety Systems and Engineered Safeguards

1.6.2.1 Reactor Protection System

The reactor protection system (RPS) initiates a rapid, automatic shutdown (scram) of the reactor. This action is taken in time to prevent excessive fuel cladding temperatures and any nuclear system process barrier damage following abnormal operational transients. The RPS overrides all operator actions and process controls.

1.6.2.2 Neutron Monitoring System

Although not all of the neutron monitoring system qualifies as a nuclear safety system, those portions that provide high neutron flux signals to the reactor protection system do. The wide range neutron monitors (WRNM), average power range monitors (APRM), and oscillation power range monitors (OPRM) which monitor neutron flux via in-core detectors, signal the RPS to scram in time to prevent excessive fuel clad temperatures as a result of abnormal operational transients.

1.6.2.3 Control Rod Drive System

When a scram is initiated by the RPS, the control rod drive system (CRDS) inserts the negative reactivity necessary to shut down the reactor. Each control rod is controlled individually by a hydraulic control unit. When a scram signal is received, high-pressure water from an accumulator for each rod forces each control rod rapidly into the core.

1.6.2.4 Nuclear System Pressure Relief System

A pressure relief system, consisting of relief and safety valves mounted on the main steam lines, prevents excessive pressure inside the nuclear system following either abnormal operational transients or accidents.

1.6.2.5 Reactor Core Isolation Cooling System

The reactor core isolation cooling system (RCICS) provides makeup water to the reactor vessel whenever the vessel is isolated or during shutdown whenever normal water supply is not available. The RCICS uses a steam driven turbine-pump unit and operates automatically, in time and with sufficient coolant flow, to maintain adequate reactor vessel water level.

1.6.2.6 Primary Containment

A pressure suppression primary containment houses the reactor vessel, the reactor coolant recirculating loops, and other branch connections of the reactor primary system. The pressure suppression system consists of a drywell, a pressure suppression chamber storing a large volume of water, a connecting vent system between the drywell and the water pool, isolation valves, containment cooling systems, and other service equipment. In the event of a process system piping failure within the drywell, reactor water and steam would be released into the drywell air space. The resulting increased drywell pressure would then force a mixture of air, steam, and water through the vents into the pool of water stored in the suppression chamber. The steam would condense rapidly in the suppression pool, resulting in a rapid pressure reduction in the drywell. Air transferred to the suppression chamber pressurizes the suppression chamber and is subsequently vented to the drywell to equalize the pressure between the two vessels. Cooling systems remove heat from the reactor core, the drywell, and from the water in the suppression chamber, thus providing continuous cooling of the primary containment under accident conditions. Appropriate isolation valves are actuated during this period to ensure containment of radioactive materials within the primary containment.

1.6.2.7 Primary Containment and Reactor Vessel Isolation Control System

The primary containment and reactor vessel isolation control system automatically initiates closure of isolation valves to close off all process lines which are potential leakage paths for radioactive material to the environs. This action is taken upon indication of a potential breach in the nuclear system process barrier.

1.6.2.8 Secondary Containment

The secondary containment includes the reactor building, the reactor building heating and ventilating system, and the standby gas treatment system, and is designed to provide for controlled, filtered, and elevated release of airborne activity.

1.6.2.9 Main Steam Line Isolation Valves

Although process lines which penetrate the primary containment and offer a potential release path for radioactive material are provided with redundant isolation capabilities, the main steam lines, because of their large size and large mass flow rates, are given special isolation consideration. Two automatic isolation valves, each powered by both air pressure and spring force, are provided in each main steam line. These valves fulfill the following objectives:

1. To prevent excessive damage to the fuel barrier by limiting the loss of reactor coolant from the reactor vessel resulting either from a major leak from the steam piping outside the primary containment or from a malfunction of the pressure control system resulting in excessive steam flow from the reactor vessel.

2. To limit the release of radioactive materials, by closing the nuclear system process barrier, in case of a

gross release of radioactive materials from the fuel to the reactor coolant and steam.

3. To limit the release of radioactive materials, by closing the primary containment barrier, in case of a major leak from the nuclear system inside the primary containment.

1.6.2.10 Main Steam Line Flow Restrictors

A venturi-type flow restrictor is installed in each steam line close to the reactor vessel. These devices limit the loss of coolant from the reactor vessel and prevent uncovering of the core before the main steam line isolation valves are closed in case of a main steam line break outside the primary containment.

1.6.2.11 Core Standby Cooling Systems

A number of core standby cooling systems (CSCS's) are provided to prevent excessive fuel clad temperatures in the event a breach in the nuclear system process barrier results in a loss of reactor coolant. The four CSCS's are:

High-pressure coolant injection system (HPCIS)
Automatic depressurization system (ADS)
Core spray system
Low-pressure coolant injection (an operating mode of the RHRS) (LPCI)

1. High-Pressure Coolant Injection System

The HPCIS provides and maintains an adequate coolant inventory inside the reactor vessel to prevent excessive fuel clad temperatures as a result of postulated small breaks in the nuclear system process barrier. A high-pressure system is needed for such breaks because the reactor vessel depressurizes slowly, preventing low-pressure systems from injecting coolant. The HPCIS includes a turbine-pump powered by reactor steam. The system is designed to accomplish its function on a short-term basis without reliance on plant auxiliary power supplies other than the DC power supply. The HPCI steam supply inboard isolation valve is AC powered, but is normally maintained open.

2. Automatic Depressurization System

The ADS acts to rapidly reduce reactor vessel pressure in a LOCA situation in which the HPCIS fails to automatically maintain reactor vessel water level. The depressurization provided enables the low-pressure standby cooling systems to deliver cooling water to the reactor vessel. The ADS uses some of the relief valves which are part of the nuclear system pressure relief system. The automatic relief valves are arranged to open upon conditions indicating both that a break in the nuclear system process barrier has occurred and that the HPCIS is not delivering sufficient cooling water to the reactor vessel to maintain the water level above a pre-selected value. The ADS will not be activated unless either the core spray system or the LPCIS is operating.

3. Core Spray System

The core spray system consists of two independent pump loops that deliver cooling water to spray spargers over the core. The system is actuated by conditions indicating that a breach exists in the nuclear system process barrier, but water is delivered to the core only after reactor vessel pressure is reduced. This system provides the capability to cool the fuel by spraying water onto the core. Either core spray loop is capable of preventing excessive fuel clad temperatures following a LOCA.

4. Low Pressure Coolant Injection

Low-pressure coolant injection (LPCI) is an operating mode of the RHRS and is an engineered safeguard. LPCI uses the pump loops of the RHRS to inject cooling water at low pressure into the reactor recirculation loops. LPCI is actuated by conditions indicating a breach in the nuclear system process barrier, but water is delivered to the core only after reactor vessel pressure is reduced. LPCI operation, together with the core shroud and jet pump arrangement, provides the capability of core reflooding following a LOCA in time to prevent excessive fuel clad temperatures.

1.6.2.12 Residual Heat Removal System (Containment Cooling)

The RHRS for containment cooling is placed in operation to limit the temperature of the water in the suppression pool following a design basis LOCA. In the containment cooling mode of operation, the RHRS pumps take suction from the suppression pool and deliver the water through the RHRS heat exchangers, where cooling takes place by transferring heat to the High Pressure Service Water System. The fluid is then discharged back to the suppression pool.

Another portion of the RHRS is provided to spray water into the primary containment as a means of reducing containment pressure following a LOCA. This capability is in excess of the required energy removal capability and can be placed into service at the discretion of the operator.

1.6.2.13 Control Rod Velocity Limiter

A control rod velocity limiter is a part of each control rod and limits the velocity at which a control rod can fall out of the core should it become detached from its CRD. The rate of reactivity insertion resulting from a rod drop accident is limited by this feature. The limiters contain no moving parts.

1.6.2.14 Control Rod Drive Housing Supports

CRD housing supports are located underneath the reactor vessel near the control rod housings. The supports limit the travel of a control rod in the event that a control rod housing is ruptured. The supports prevent a nuclear excursion as a result of a housing failure, thus protecting the fuel barrier.

1.6.2.15 Standby Gas Treatment System

The standby gas treatment system is part of the secondary containment and has the capability of maintaining a negative pressure within the reactor building, with respect to the outside atmosphere, to limit ground level release of radioactive material. The reactor building atmosphere, normally discharged through the building ventilation exhaust, can be processed through the standby gas treatment system for filtration before being discharged to the stack when a high radiation condition occurs.

1.6.2.16 Standby ac Power Supply

The standby ac power supply consists of four diesel-generator sets. The diesel-generators are sized so that three diesels can supply all necessary power requirements for one unit under postulated design basis accident conditions plus necessary power requirements for the safe shutdown of the second unit. The diesel-generators start and attain rated voltage and frequency within 10 sec. The diesel-generator system is arranged with four independent 4-kV buses for each unit, each bus being connected to one diesel-generator. Each diesel-generator starts automatically upon loss of off-site power or detection of a nuclear accident. The necessary engineered safeguard system loads are applied on a preset time sequence. Each generator operates independently without paralleling.

1.6.2.17 dc Power Supply

Three independent sets of 125/250-V batteries are provided for each reactor unit. The sets are not interconnected. One battery charger panel consisting of two-100% chargers is provided for each battery. In addition, one non-Class 1E alternate battery charger provides an alternate means to maintain 1E battery voltage in the event of inoperability of one of the Class 1E battery chargers.

Two safety-related 125/250-V dc systems are designed to provide an adequate power source for supplying the engineered safeguard loads of one unit, and the required shutdown loads of the second unit, with concurrent loss of off-site power and any single failure in the dc system.

One independent balance-of-plant 125/250-V dc system is used for the turbine generator emergency bearing oil pump and other nonsafety-related loads.

1.6.2.18 High-Pressure Service Water System

A high-pressure service water system removes the heat rejected by the RHRS during shutdown operation and accident conditions.

1.6.2.19 Emergency Service Water System

The emergency service water system supplies water for cooling standby diesel-generators and CSCS equipment rooms.

1.6.2.20 Main Steam Line Radiation Monitoring System

The main steam line radiation monitoring system consists of four gamma radiation monitors located external to the four main steam lines just outside of the primary containment. The monitors are designed to detect a gross release of fission products from the fuel. Upon detection of high radiation, an alarm signal is initiated and trending of radiation levels will be utilized to monitor radiation levels and determine if additional action is required to maintain radiation levels within limits. Actuation of high-radiation alarm alerts Operators to close any open reactor coolant sample lines.

1.6.2.21 Ventilation Radiation Monitoring System

The ventilation radiation monitoring system consists of a number of radiation monitors arranged to indicate and record the activity level of the ventilation exhaust from the buildings during planned operations. Upon detection of high radiation, the reactor building is automatically isolated and the standby gas treatment system is started.

1.6.2.22 Standby Liquid Control System

The Standby Liquid Control System provides an alternate method of bringing the reactor subcritical and maintaining it subcritical as described in Section 1.6.3.1. The system is also credited with injection during a LOCA to maintain the suppression pool pH greater than 7 throughout the accident duration. This ensures that sufficient iodine will be retained in the Suppression Pool water, and offsite doses remain within 10 CFR 50.67 limits.

1.6.3 Special Safety Systems

1.6.3.1 Standby Liquid Control System

Although not intended to provide rapid reactor shutdown, the standby liquid control system provides a redundant, independent, and different way from the control rods to bring the reactor subcritical and to maintain it subcritical as the reactor cools. The system makes possible an orderly and safe shutdown in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner. The system is sized to counteract the positive reactivity effect from rated power to the cold shutdown condition, and meets the performance requirements of 10CFR50.62 as described in Section 3.8.

1.6.3.2 Shutdown Capability Outside the Control Room

Means are provided outside the main control room to maintain the reactors in the shutdown conditions if the main control room becomes uninhabitable.

1.6.3.3 Emergency Heat Sink

An emergency heat sink is provided to assure adequate shutdown cooling in the event of the unavailability of the normal heat sink. This system consists of a closed-cycle cooling tower system used in conjunction with the high pressure service water and emergency service water systems.

1.6.3.4 Alternate Rod Insertion

An alternate rod insertion system (ARI) is provided for mitigating an anticipated transient without scram (ATWS). The ARI system provides an alternate means of reactor shutdown which is independent of the reactor protection system. An ARI signal opens solenoid valves on the scram air header to bleed air from the header which in turn allows the scram inlet and discharge valves to open. The control rod drives then insert the control rods which shut down the reactor. Venting of the scram air header also closes the scram discharge volume vent and drain valves.

The ARI system meets the requirements of 10CFR50.62 including the guidance listed in the June 26, 1984 Federal Register. The ARI system also meets the intent of NRC Generic Letter 85-06 dated April 16, 1985.

Additional information on ARI is provided in subsection 7.9.4.4.2 and in Drawings M-356, M-357, M-1-CC-4, Sheets 6, 7, and 12, M-1-CC-42, Sheets 1, 2, 10, and 11, and M-1-CC-46, Sheet 1.

1.6.4 Process Control and Instrumentation

1.6.4.1 Nuclear System Process Control and Instrumentation

1.6.4.1.1 Reactor Manual Control System

The reactor manual control system provides the means by which control rods are manipulated from the control room for gross power control. The system controls valves in the CRD hydraulic system. Only one control rod can be manipulated at a time. The reactor manual control system includes the controls that restrict control rod movement (rod block) under certain conditions as a backup to procedural controls.

1.6.4.1.2 Recirculation Flow Control System

The recirculation flow control system controls the speed of the reactor recirculation pumps. Adjusting the pump speed changes the coolant flow rate through the core. This effects changes in core power level. The system is arranged to manually change reactor power output by varying the frequency of the electrical power supply for the reactor recirculation pumps.

1.6.4.1.3 Neutron Monitoring System

The neutron monitoring system is a system of in-core neutron detectors and electronic monitoring equipment. The system provides indication of neutron flux, which can be correlated to thermal power level, for the entire range of flux conditions that may exist in the core. The wide range neutron monitors (WRNM) provide flux level indications during reactor startup and low power operation. The local power range monitors (LPRM) and APRM's allow assessment of local and overall flux conditions during power range operation. A rod block monitor (RBM) is provided to prevent rod withdrawal when reactor power should not be increased. The Traversing In-core Probe System (TIPS) provides a means to calibrate the individual LPRM's.

1.6.4.1.4 Refueling Interlocks

A system of interlocks, restricting the movements of refueling equipment and control rods when the reactor is in the refuel mode, is provided to prevent an inadvertent criticality during refueling operations. The interlocks back up procedural controls that have the same objective. The interlocks affect the refueling bridge, the refueling bridge hoists, the fuel grapple, control rods, and the service platform hoist.

1.6.4.1.5 Reactor Vessel Instrumentation

In addition to instrumentation provided for the nuclear safety systems and engineered safeguards, instrumentation is provided to monitor and transmit information that can be used to assess conditions existing inside the reactor vessel and the physical condition of the vessel itself. The instrumentation provided monitors reactor vessel pressure, water level, surface temperature, internal differential pressures and coolant flow rates, and top head flange leakage.

1.6.4.1.6 Process Computer System (PMS)

An on-line process computer is provided to monitor and log process variables and to make certain analytical computations. In conjunction with approved operating procedures, the rod worth minimizer function prevents improper rod withdrawal under low power conditions. The effect of the rod block is to limit the reactivity worth of the control rods by enforcing adherence to the pre-planned rod pattern.

1.6.4.2 Power Conversion Systems Process Control and Instrumentation

1.6.4.2.1 Pressure Regulator and Turbine Control

The pressure regulator controls both the turbine admission valves and the turbine bypass valves and maintains constant reactor pressure. Pressure regulation is coordinated with the turbine speed and load control systems. The turbine control utilizes an electrohydraulic control system arranged for remote operation.

1.6.4.2.2 Feedwater System Control

A three-element control system is used to regulate the feedwater system so that proper water level is maintained in the reactor vessel. The feedwater control signal is used to control the speed of the steam-turbine-driven feedwater pumps.

1.6.4.3 Electrical Power System Control

Controls for the electrical power system are located in the control room to permit safe startup, operation, and shutdown of the plant.

1.6.4.4 Radiation Monitoring

1.6.4.4.1 Process Radiation Monitoring

Radiation monitors are provided on various lines to monitor for radioactive materials released to the environs via process liquids and gases or for detection of process system malfunctions. These monitors annunciate alarms and/or provide signals to initiate isolation and corrective actions.

1.6.4.4.2 Area Radiation Monitors

Radiation monitors are provided to monitor for abnormal radiation at various locations in the reactor building, turbine building, and radwaste building. These monitors annunciate alarms when abnormal radiation levels are detected.

1.6.4.4.3 Site Environs Radiation Monitors

Radiation monitors are provided outside the plant buildings to monitor radiation levels. These data are used for determining the contribution of plant operations to on-site and off-site radiation levels.

1.6.4.4.4 Liquid Radwaste System Control

Liquid wastes to be discharged are handled on a batch basis with protection against accidental discharge provided by procedural controls. Instrumentation, with alarms, to detect abnormal concentration of the radwastes, is provided.

1.6.4.4.5 Solid Radwaste Control

The solid radwaste system collects, treats, and prepares solid radioactive wastes for off-site shipment. Wastes are handled on a batch basis. Radiation levels of the various batches are determined by the operator.

1.6.4.4.6 Gaseous Radwaste System Control

The gaseous radwaste system is continuously monitored by the off-gas vent radiation monitor, the air ejector off-gas radiation monitor and the adsorber post treatment monitor. A high level signal will annunciate alarms in the main control room.

1.6.5 Auxiliary Systems

1.6.5.1 Normal Auxiliary ac Power

Normal auxiliary power is supplied from the main generator to the auxiliary buses through the unit auxiliary transformers.

Three independent off-site sources of auxiliary power are available to serve the plant. Any of the three sources can be connected to any auxiliary bus in the plant. Each off-site source has the capacity for operation of all systems required to shutdown the plant and maintain it in a safe condition.

1.6.5.2 Reactor Building Cooling Water System

The reactor building cooling water system provides adequate cooling water to designated auxiliary plant equipment.

1.6.5.3 Turbine Building Cooling Water System

The turbine building cooling water system provides adequate cooling water to designated auxiliary equipment.

1.6.5.4 Service Water System

The service water system supplies river water for plant makeup and supplies the turbine building and the reactor building equipment coolers.

1.6.5.5 Fire Protection System

A fire protection system is provided to supply fire fighting water to points throughout the plant. Chemical and CO_2 protection systems, as well as portable fire extinguishers, are also provided.

1.6.5.6 Heating, Ventilation, and Air Conditioning Systems

The ventilation systems of the reactor buildings, turbine building, and radwaste building supply and circulate filtered outside air for personnel comfort and equipment cooling, and discharge to the ventilation exhausts. A separate air conditioning system is provided for the main control room. Two auxiliary boilers provide plant heating.

1.6.5.7 New and Spent Fuel Storage

New and spent fuel is stored in high density storage racks located in the spent fuel pool. Fuel transfer during refueling is conducted underwater.

Approved spent fuel may also be removed from the spent fuel pool and stored in an approved dry cask storage system located on site at the Independent Spent Fuel Storage Installation. Loading of casks for dry storage is conducted underwater in the spent fuel pool.

1.6.5.8 Fuel Pool Cooling and Cleanup System

A fuel pool cooling and cleanup system is provided to remove decay heat from spent fuel stored in the fuel pool and to maintain a specified water temperature, purity, clarity, and level.

1.6.5.9 Service and Instrument Air System

A service and instrument air system supplies compressed air.

1.6.5.10 Makeup Water System

The makeup water system furnishes clarified water, filtered, and demineralized water for various plant requirements.

1.6.5.11 Potable and Sanitary Water Systems

A potable water system for drinking and sanitary uses is provided for the plant.

1.6.5.12 Equipment and Floor Drainage System

The equipment and floor drainage system handles both radioactive and nonradioactive drains. Drains which may contain radioactive materials are pumped to the radwaste system for cleanup, reuse, or discharge. Nonradioactive drains are discharged to the storm drain system.

1.6.5.13 Process Sampling System

The plant process sampling system monitors the operation of plant equipment and provides information for making operational decisions.

1.6.5.14 Station Communications System

A station communications system provides communications between various plant buildings and locations.

1.6.6 Shielding

Shielding is provided to meet the occupancy requirements of the various areas of the plant.

1.6.7 Loading Criteria

Structures and equipment are designed to resist structural and mechanical damage due to both dead and live loads, including environmental forces. Structural loading criteria are discussed in detail in Appendix C.

Definitions of seismic Class I and seismic Class II structures follow.

a. Seismic Class I

   Seismic Class I structures and equipment are those whose failure could increase the severity of the design basis accident, cause release of radioactivity in excess of 10CFR100 limits, or those essential for safe shutdown and removal of decay heat following a LOCA.

b. Seismic Class II

   Seismic Class II structures and equipment are those whose failure would not result in the release of significant radioactivity and would not prevent reactor shutdown. The failure of seismic Class II structures may interrupt power generation.

A structure designated seismic Class II shall not degrade the integrity of any structure designated seismic Class I. Although a structure, as a whole, may be seismic Class I, less essential portions may be considered seismic Class II if they are not associated with loss of function, and their failure does not render the seismic Class I portion inoperable.

1.7 COMPARISON OF PRINCIPAL DESIGN CHARACTERISTICS

This section provides a comparison of the major features with other BWR facilities at the time the application for operating licenses was made.

The design of this facility is based upon proven technology attained during the development, design, construction, and operation of BWR's of similar or identical types.

1.7.1 Nuclear System Design Characteristics

Table 1.7.1 summarizes the nuclear system design characteristics for PBAPS Units 2 and 3. Design characteristics are also presented for the nuclear systems of Vermont Yankee, Cooper, and Browns Ferry nuclear power stations.

1.7.2 Power Conversion Systems Design Characteristics

Table 1.7.2 summarizes the power conversion systems design characteristics for PBAPS Units 2 and 3. Design characteristics are also presented for the power conversion systems of Vermont Yankee, Cooper, and Browns Ferry nuclear power stations.

1.7.3 Electrical Power Systems Design Characteristics

Table 1.7.3 summarizes the electrical power systems design characteristics for PBAPS Units 2 and 3. Design characteristics are also presented for the electrical power systems of Vermont Yankee, Cooper, and Browns Ferry nuclear power stations.

1.7.4 Containment Design Characteristics

Table 1.7.4 summarizes the design characteristics for the primary and secondary containments of PBAPS Units 2 and 3. Design characteristics are also presented for the primary and secondary containment systems employed for Vermont Yankee, Cooper, and Browns Ferry nuclear power stations.

1.7.5 Structural Design Characteristics

Table 1.7.5 summarizes the structural design characteristics of PBAPS Units 2 and 3. Design characteristics are also presented for the structures of Vermont Yankee, Cooper, and Browns Ferry nuclear power stations.

TABLE 1.7.1

COMPARISON OF NUCLEAR SYSTEM DESIGN CHARACTERISTICS

(Parameters are related to Rated Power Output
for a single plant unless otherwise noted.)

THERMAL AND HYDRAULIC DESIGN	Peach Bottom* Unit 2	Peach Bottom* Unit 3	Original Design Browns Ferry	Original Design Vermont Yankee	Original Design Cooper
Rated Power, MWt	3,293	3,293	3,292	1,593	2,381
Design Power, MWt	3,440	3,440	3,440	1,665	2,500
Core Coolant Flow Rate, lb/hr	102.5×10^6	102.5×10^6	102.5×10^6	48.0×10^6	73.5×10^6
Feedwater Flow Rate, lb/hr	13.30×10^6	13.331×10^6	13.33×10^6	6.43×10^6	9.81×10^6
Feedwater Temperature, °F	376.1	376.1	376.1	372	367.1
System Pressure, Nominal in Steam Dome, psia	1,020	1,020	1,020	1,020	1,020
Average Power Density, kW/liter	50.7	50.0	50.7	51.0	50.6
Design Limit Maximum Output, kW/ft	18.5	18.5	18.5	18.5	18.5
Average Thermal Output, kW/ft	7.04	6.95	7.04	7.07	7.02
Maximum Heat Flux, Btu/hr-sq ft	428,400	428,400	428,400	427,300	427,820
Average Heat Flux, Btu/hr-sq ft	163,220	160,996	163,220	163,610	162,480
Maximum CO Centerline Temperature, °F	4,493	4,493	4,493	4,493	4,493
Maximum Fuel Volumetric Average Temperature, °F	2,781	2,781	2,781	2,781	2,781
Maximum Fuel Rod Outside Surface Temperature, °F	565	565	565	565	565
Minimum Critical Heat Flux Ratio (MCHFR)	≥1.9	≥1.9	≥1.9	≥1.9	>1.9
Coolant Enthalpy at Core Inlet, Btu/lb	521.3	521.3	521.3	519.8	520.1
Core Maximum Exit Voids within Assemblies, %	76	76	76	79	75
Core Average Exit Quality, % Steam	12.9	12.9	12.9	13.6	12.9

* At the time the original operating license application was made.

TABLE 1.7.1(Continued)

THERMAL AND HYDRAULIC DESIGN (Continued)	Peach Bottom* Unit 2	Peach Bottom* Unit 3	Original Design Browns Ferry	Vermont Yankee	Cooper
MCHFR Design Power Peaking Factors					
Maximum Relative Assembly Power	1.4	1.4	1.4	1.4	1.4
Local Peaking Factor	1.24	1.24	1.24	1.24	1.24
Axial Peaking Factor	1.4	1.4	1.4	1.5	1.4
Total Peaking Factor	2.43	2.43	2.43	2.6	2.43
NUCLEAR DESIGN (First Core)					
Water/UO Volume Ratio (Cold)	2.43 Type 1 2.53 Types 2&3	2.53 Types 1,2,&3	2.43 Type 1 2.53 Types 2&3	2.41	2.41 Types 1,2,&3
Reactivity with Strongest Control Rod Out, k	<0.99	<0.99	<0.99	<0.99	<0.99
Moderator Temperature Coefficient					
At 68°F, k/k - °F Water	-3.5×10^{-5}	-3.5×10^{-5}	-3.5×10^{-5}	-5.0×10^{-5}	-3.5×10^{-5}
Hot, no voids, k/k - °F Water	-11.6×10^{-5}	-11.6×10^{-5}	-11.6×10^{-5}	-17.0×10^{-5}	-11.6×10^{-5}
Moderator Void Coefficient					
Hot, no voids, k/k - % Void	-8.7×10^{-4}	-8.7×10^{-4}	-8.7×10^{-4}	-1.0×10^{-3}	-8.7×10^{-4}
At Rated Output, k/k - % Void	-1.05×10^{-3}	-1.05×10^{-3}	-1.05×10^{-3}	-1.5×10^{-3}	-1.05×10^{-3}
Fuel Temperature Doppler Coefficient					
At 68 °F, k/k - °F Fuel	-0.9×10^{-5}	0.9×10^{-5}	-0.9×10^{-5}	-1.3×10^{-5}	-1.3×10^{-5}
Hot, No Void, k/k - °F Fuel	-1.0×10^{-5}	-1.0×10^{-5}	-1.0×10^{-5}	-1.2×10^{-5}	-1.2×10^{-5}
At Rated Output, k/k - °F Fuel	-0.9×10^{-5}	-0.9×10^{-5}	-0.9×10^{-5}	-1.3×10^{-5}	-1.3×10^{-5}
Initial Average U-235 Enrichment, W/O	2.19	2.19	2.19	2.50	2.15
Fuel Average Discharge Exposure, MWD/Ton	19,000	19,000	19,000	19,000	19,000

* At the time the original operating license application was made.

TABLE 1.7.1(Continued)

CORE MECHANICAL DESIGN	Peach Bottom* Unit 2	Peach Bottom* Unit 3	Original Design Browns Ferry	Vermont Yankee	Cooper
Fuel Assembly					
Number of Fuel Assemblies	764	764	764	368	548
Fuel Rod Array	7 x 7	7 x 7	7 x 7	7 x 7	7 x 7
Overall Dimensions, in	175.98	175.98	175.98	175.98	175.98
Weight of UO per Assembly, lb	Low Enrichment 490.489 Type 1 High Enrichment 468.90 Type 2 468.81 Type 3	Low Enrichment 475.72 Type 1 High Enrichment 474.32 Type 2 473.99 Type 3	Low Enrichment 490.489 Type 1 High Enrichment 468.90 Type 2 468.81 Type 3	Undished - 487.4	Low Enrichment 490.4 High Enrichment 474.4 Type 2 474.1 Type 3
Weight of Fuel Assembly, lb	Low Enrichment 682.489 Type 1 High Enrichment 674.79 Type 2 674.70 Type 3	Low Enrichment 681.61 Type 1 High Enrichment 681.21 Type 2 681.11 Type 3	Low Enrichment 682.489 Type 1 High Enrichment 674.79 Type 2 674.70 Type 3	Undished - 682	Low Enrichment 681.4 Type 1 High Enrichment 681.3 Type 2 681.2 Type 3
Fuel Rods					
Number per Fuel Assembly	49	49	49	49	49
Outside Diameter, in	0.563	0.563	0.563	0.563	0.563
Clad Thickness, in	0.032 Type 1 0.037 Types 2&3	0.37 Types 1, 2 & 3	0.032 Type 1 0.037 Types 2&3	0.032	0.032 Type 1 0.037 Types 2,3
Gap - Pellet to Clad, in	0.006	0.006	0.006	0.006	0.006
Length of Gas Plenum, in	16	14	16	16	16 Type 1 14 Types 2,3
Clad Material	Zircaloy-2	Zircaloy-2	Zircaloy-2	Zircaloy-2	Zircaloy-2
Cladding Process	Freestanding loaded tubes	Freestanding loaded tubes	Freestanding loaded tubes	Freestanding loaded tubes	Freestanding loaded tubes
Fuel Pellets					
Material	Uranium Dioxide	Uranium Dioxide	Uranium Dioxide	Uranium Dioxide	Uranium Dioxide

* At the time the original operating license application was made.

TABLE 1.7.1(Continued)

CORE MECHANICAL DESIGN (Continued)	Peach Bottom* Unit 2	Peach Bottom* Unit 3	Original Design Browns Ferry	Original Design Vermont Yankee	Original Design Cooper
Fuel Pellets (Continued)					
Effective Stacked Density, % of theoretical	94	94	94	93	93
Diameter, in	0.487 Type 1 0.477 Types 2&3	0.477 Type 1, 2 & 3	0.487 Type 1 0.477 Types 2&3	0.487	0.487 Type 1 0.477 Types 2,3
Length, in	-0.5 Types 2&3 -0.75 Type 1	-0.5 Types 1, 2&3	0.5 Types 2&3 0.75 Type 1	0.75 2 & 3	0.75 Type 1 0.50 Types 2,3
Fuel Channel					
Overall Dimension, in (length)	166.906	166.906	166.906	166.906	166.906
Thickness, in	0.080	0.080	0.080	0.080	0.080
Cross Section Dimensions, in	5.438 x 5.438	5.438 x 5.438	5.438 x 5.438	5.438. x 5.438	5.438 x 5.438
Material	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4
Core Assembly					
Fuel Weight as UO, lb	361,837	361,837	361,837	179,370	267,095
Zirconium Weight, lb (Z-2 + Z-4 Spacers)	140,307	140,307	140,397	63,300	94,305
Core Diameter (equivalent), in	187.1	187.1	187.1	129.9	158.5
Core Height (Active Fuel), in	144	144	144	144	144
Reactor Control System					
Method of Variation of Reactor Power	Movable Control Rods & Variable Coolant Pumping	Movable Control Rods & Variable Coolant Pumping	Movable Control Rods & Variable Cooling Pumping	Movable Control Rods & Variable Coolant Pumping	Movable Control Rods & Variable Coolant Pumping
Number of Movable Control Rods	185	185	185	89	137
Shape of Movable Control Rods	Cruciform	Cruciform	Cruciform	Cruciform	Cruciform
Pitch of Movable Control Rods	12.0	12.0	12.0	12.0	12.0

* At the time the original operating license application was made.

TABLE 1.7.1(Continued)

CORE MECHANICAL DESIGN (Continued)	Peach Bottom* Unit 2	Peach Bottom* Unit 3	Original Design Browns Ferry	Vermont Yankee	Cooper
Control Material in Movable Rods	B C granules Compacted in SS Tubes	B C granules Compacted in SS Tubes	B C granules Compacted in SS Tubes	B C granules Compacted in SS Tubes	B C granules Compacted in SS Tubes
Type of Control Rod Drives	Bottom entry, Locking Piston	Bottom entry, Locking Piston	Bottom entry, Locking Piston	Bottom entry, Locking Piston	Bottom entry, Locking Piston
Supplementary Reactivity Control	Gadolinia Burnable Poison	Gadolinia Burnable Poison	Gadolinia Burnable Poison	Flat boron--stainless steel Control Curtain	Gadolinia Burnable Poison
In-Core Neutron Instrumentation					
Number of In-Core Neutron Detectors (Fixed)	172	172	172	80	124
Number of In-Core Detector Assemblies	43	43	43	20	31
Number of Detectors per Assembly	4	4	4	4	4
Number of Flux Mapping Neutron Detectors	5	5	5	3	4
Range (and Number) of Detectors					
Source Range Monitor	WRNM 0-100% power (8)	WRNM 0-100% power (8)	Source to 10-3% power (4)	Source to 10-3% power (4)	Source to 10-3% power (4)
Intermediate Range Monitor	WRNM 0-100% power (8)	WRNM 0-100% power (8)	10-4% to 10% power (8)	10-4% to 10% power (6)	10-4% to 10% power (8)
Local Power Range Monitor	5% to 125% power (172)	5% to 125% power (172)	5% to 125% power (172)	5% to 125% power (80)	5% to 125% power (124)
Average Power Range Monitor	2.5% to 125% power (6)	2.5% to 125% power (6)	2.5% to 125% power (6)	2.5% to 125% power (6)	2.5% to 125% power (6)
Number and Type of In-Core Neutron Sources	7 Sb-Be	7 Sb-Be	7 Sb-Be	8 Sb-Be	5 Sb-Be
REACTOR VESSEL DESIGN					
Material	Carbon Steel/Clad Stainless Steel (ASME SA-336 & SA-302B)				
Design Pressure, psia	1265	1265	1265	1265	1265

* At the time the original operating license application was made.

TABLE 1.7.1(Continued)

REACTOR VESSEL DESIGN (Continued)	Peach Bottom* Unit 2	Peach Bottom* Unit 3	Original Design Browns Ferry	Vermont Yankee	Cooper
Design Temperature, °F	575	575	575	575	575Inside Diameter
Inside Diameter ft-in	20 - 11	20 - 11	20 - 11	17 - 2	18 - 2
Inside Height, ft-in	72 - 11	72 - 11	72 - 11/18	63 - 1.5	69 - 4
Side Thickness (including clad)	6 - 5/16	6 - 5/16	6- 5/16	5.187	5.531
Minimum Clad Thickness, in	1/8	1/8	1/8	1/8	1/8
REACTOR COOLANT RECIRCULATION DESIGN					
Number of Recirculation Loops	2	2	2	2	2
Design Pressure					
Inlet Leg, psig	1,250	1,250	1,148	1,175	1,148
Outlet Leg, psig	1,500	1,500	1,326	1,274	1,274
Design Temperature, °F	575	575	562	562	562
Pipe Diameter, in	28	28	28	28	28
Pipe Material	316 NG (Controlled Chemistry)	316 NG (Controlled Chemistry)	304/316	304/316	304/316
Recirculation Pump Flow Rate, gpm	45,200	45,200	45,200	32,500	45,200
Number of Jet Pumps in Reactor	20	20	20	20	20
MAIN STEAM LINES					
Number of Steam Lines	4	4	4	4	4
Design Pressure, psig	1,115	1,115	1,146	1,146	1,146
Design Temperature, °F	583	583	563	563	563
Pipe Diameter, in	26	26	26	20	24
Pipe Material	C.S. A155, KC-70 Carbon Steel (ASTM A155 KC70 or ASTM A106 Grade B)				

* At the time the original operating license application was made.

TABLE 1.7.1(Continued)

CORE STANDBY COOLING SYSTEMS (These systems are sized on design power.)	Peach Bottom* Unit 2	Peach Bottom* Unit 3	Original Design Browns Ferry	Original Design Vermont Yankee	Original Design Cooper
Core Spray System					
Number of Loops	2	2	2	2	2
Flow Rate, gpm	6,250 at	6,250 at	6,250	6,250	6,250
High-Pressure Coolant Injection System (No.)	1	1	1	1	1
Number of Loops	1	1	1	1	1
Flow Rate, gpm	5,000	5,000	5,000	4,250	4,220
Automatic Depressurization System (No.)	1	1	1	1	1
Low-Pressure Coolant Injection (No.)	1	1	1	1	1
Number of pumps	4	4	4	4	4
Flow Rate, gpm/pump	10,000 at 20 psid	10,000 at 20 psid	10,000 at 20 psid	4,800 20 psid	7,000 20 psid
AUXILIARY SYSTEMS					
Residual Heat Removal Systems					
Reactor Shutdown Cooling (number of pumps)	4	4	4	4	4
Flow Rate, gpm/pump(1)	10,000	10,000	10,000	7,000	7,700
Capacity, Btu/hr/heat exchanger(2)	70×10^6	70×10^6	70×10^6	57.5×10^6	70×10^6
Number of heat exchangers	4	4	4	2	2
Primary Containment Cooling					
Flow rate, gpm	40,000	40,000	40,000	28,000	30,800
High-Pressure Service Water System					
Flow Rate, gpm/pump	4,500	4,500	4,500	2,700	8,000
Number of pumps	4	4	8(3)	4	4

* At the time the original operating license application was made.

TABLE 1.7.1(Continued)

AUXILIARY SYSTEMS (Continued)	Peach Bottom* Unit 2	Peach Bottom* Unit 3	Original Design Browns Ferry	Vermont Yankee	Cooper
Reactor Core Isolation Cooling System					
Flow Rate, gpm	616 at 1,120 psid	616 at 1,120 psid	616 at 1,120 psid	400	416 at 1,120 psid
Fuel Pool Cooling and Cleanup System					
Capacity, Btu/hr	11.25 x 10^6	11.25 x 10^6	8.8 x 10^6	2.37 x 10^6	3.4 x 10^6

* At the time the original operating license application was made.

(1) Capacity during reactor flooding mode with three of four pumps running.

(2) Capacity during post-accident cooling mode with 165°F shell side inlet temperature, maximum service water temperature, and one RHR pump and one RHR service water pump in operation.

(3) For all three units.

TABLE 1.7.2

COMPARISON OF POWER CONVERSION SYSTEM DESIGN CHARACTERISTICS

Turbine-Generator	Peach Bottom* Units 2 and 3	Vermont Yankees	Cooper	Browns Ferry
Design Power, MWt	3,440	1,665	2,487	3,440
Design Power, MWe	1,230	564	836	1,152
Generator Speed, rpm	1,800	1,800	1,800	1,800
Design Steam Flow, lb/hr	14.27×10^6	6.423×10^6	10.049×10^6	14.049×10^6
Turbine Inlet Pressure, psig	983.9	950	970	965
Turbine Bypass System				
Capacity, Percent of Turbine Design Steam Flow	25	100	25	25
Main Condenser				
Heat Removal Capacity, Btu/hr	$7,600 \times 10^6$	$3,500 \times 10^6$	$5,367 \times 10^6$	$7,770 \times 10^6$
Circulating Water System				
Number of Pumps	3	3	4	3
Flow Rate, gpm/pump	250,000	117,000	162,500	200,000
Condensate and Feedwater Systems				
Design Flow Rate, lb/hr	13.999×10^6	6.4×10^6	9.773×10^6	13.999×10^6
Number Condensate Pumps	3	2	3	3
Number Condensate Booster Pumps	-	-	-	3
Number Feedwater Pumps	3	2	2	3
Condensate Pump Drive	ac power	ac power	ac power	ac power
Condensate Booster Pump Drive	-	-	-	ac power
Feedwater Pump Drive	Turbine	ac power	Turbine	Turbine

* At the time the original license application was made.

TABLE 1.7.3

COMPARISON OF ELECTRICAL POWER SYSTEMS DESIGN CHARACTERISTICS

Transmission System	Peach Bottom* Units 2 and 3	Vermont Yankee	Cooper	Browns Ferry
Outgoing Lines (number-rating)	4-500 kV	2-345 kV	4-345 kV	6-500 kV
Normal Auxiliary AC Power				
Incoming Lines (number-rating)	1-230 kV 1-13.8 kV	2-345 kV 1-230 kV 1-115 kV 1-4,160 kV	1-115 kV 1-69 kV	2-161 kV
Auxiliary Transformers	2	1	1	3
Startup Transformers	2	1	2	2
Emergency Transformers	2	-	-	-
Standby ac Power Supply				
Number Diesel-Generators	4	2	4	4
Number of 4,160-V Standby Buses	8	2	2	4
Number of 480-V Standby Buses	8	3	3	8
dc Power Supply				
Number of 125-V Batteries	12-125 V	2	2	4
Number of 125-V or 250-V Buses	8-250 V 8-125 V	4	4	4

* At the time the original operating license application was made.

TABLE 1.7.4

COMPARISON OF CONTAINMENT CHARACTERISTICS

Primary Containment *	Peach Bottom Units 2 and 3**	Vermont Yankees	Cooper	Browns Ferry
Type	Pressure Suppression	Pressure Suppression	Pressure Suppression	Pressure Suppression
Construction				
Drywell	Light bulb shape; steel vessel	Light bulb shape; steel vessel	Light bulb shape; steel vessel	Light bulb shape; steel vessel
Pressure Suppression Chamber	Torus; steel vessel	Torus; steel vessel	Torus; steel vessel	Torus; steel vessel
Pressure Suppression Chamber - Internal Design Pressure, psig	56	56	56	56
Pressure Suppression Chamber - External Design Pressure, psi	2	2	2	2
Drywell-Internal Design Pressure, psig	56	56	56	56
Drywell-External Design Pressure, psi	2	2	2	2
Drywell Free Volume including Vent Lines, Vent Header, and Downcomers, cu ft	175,800	134,000	145,430	159,000
Pressure Suppression Chamber Free Volume, cu ft	127,700 to 132,000	99,000	109,810	119,000
Pressure Suppression Pool Water Volume at Minimum Water Level, cu ft	122,900	78,000	87,660	135,000
Minimum Submergence of Vent Pipe Below Pressure Pool Surface, ft	4	4	4	4
Design Temperature of Drywell, °F	281	281	281	281
Design Temperature of Pressure Suppression Chamber, °F	281	281	281	281

* Where applicable, containment parameters are based on design power.

** At the time the original operating license application was made.

TABLE 1.7.4 (Continued)

Primary Containment (Continued)	Peach Bottom Units 2 and 3**	Vermont Yankees	Cooper	Browns Ferry
Downcomer Vent Pressure Loss Factor	6.21	6.21	6.21	6.21
Break Area/Total Vent Area	0.0146	0.019	0.019	0.019
Calculated Maximum Pressure After Blowdown				
Drywell, psig	41.5	35	46	46.6
Pressure Suppression Chamber, psig	27	22	28	27
Initial Pressure Suppression Pool Temperature Rise, °F	42	35	50	50
Leakage Rate, % by weight/day at 56 psig and 281°F	0.5	0.5	0.5	0.5
Secondary Containment				
Type	Controlled leakage, elevated release	Controlled leakage, elevated release	Controlled leakage, elevated release	Controlled leakage, elevated release
Construction				
Lower Levels	Reinforced concrete w/ com-posite concrete steel forms	Reinforced concrete	Reinforced concrete	Reinforced concrete
Upper Levels	Steel super-structure and siding	Steel super-structure and siding	Steel super-structure and siding	Steel super-structure and siding
Roof	Steel decking	Steel sheeting	Steel sheeting	Steel sheeting
Internal Design Pressure, psig	0.25	0.25	0.25	0.25
Design Inleakage Rate, % free volume/day at 0.25 in H_2O)	100	100	100	100
Elevated Release Point				
Type	Stack	Stack	Stack	Stack
Construction	Reinforced concrete	Steel	Steel	Reinforced concrete
Height (above ground)	500 ft	318 ft	100 m	600 ft

** At the time the original operating license application was made.

TABLE 1.7.5

COMPARISON OF ELECTRICAL POWER SYSTEMS DESIGN CHARACTERISTICS

Seismic Design	Peach Bottom Units 2 and 3*	Vermont Yankee	Cooper	Browns Ferry
Design Earthquake (horizontal g)	0.05	0.07	0.10	0.10
Maximum Earthquake (horizontal g)	0.12	0.14	0.20	0.20
Wind Design				
Maximum Sustained (mph)	87	80	100	100
Tornados (mph)	300	300	300	300

* At the time the original operating license application was made.

1.8 SUMMARY OF RADIATION EFFECTS

1.8.1 Normal Operation

The gaseous and liquid radioactive waste systems are designed so that the dose to any person at the site boundary does not exceed that permitted by 10CFR20. The expectancy, based on operating experience, is that the dose to any person at the site boundary from gaseous waste discharge will not average more than about 1 percent of the permissible dose, and that concentrations of liquid waste at the point of discharge will average less than 1 percent of the concentrations permitted by 10CFR20. Both effects are only a small fraction of the effect of natural background radiation.

1.8.2 Abnormal Operational Transients

Analysis of abnormal operational transients, described in Section 14.0, "Plant Safety Analysis," shows that they do not result in any significant increase of radioactive material release to the environs over that experienced during normal operation.

1.8.3 Accidents

The ability of the plant to withstand the consequences of accidents without posing an undue hazard to the health and safety of the public is evaluated by analyzing a variety of postulated accidents. The calculated consequences of the design basis accidents are described in Section 14.0, "Plant Safety Analysis."

These doses are below the doses given in 10CFR50.67.

1.8.4 Interaction with Unit 1

Unit 1 was defueled and partially decontaminated in the late 1970s to allow it to be placed in NRC SAFSTOR status.
There is no interaction between Unit 1 and Units 2 and 3, except Unit 1’s liquid waste may be moved to the radwaste facility between Units 2 and 3 for processing and discharge.

1.8.5 Independent Spent Fuel Storage Installation

During normal operation of the Independent Spent Fuel Storage Installation only direct radiation is emitted from loaded dry storage casks. The radiation dose is limited by the requirements of 10CFR72.104, which considers the direct dose from the storage casks in combination with the normal plant effluents.

During accident conditions the dry cask sealing system is postulated to fail producing a gaseous effluent. This effluent is limited by the requirements of 10CFR72.106, which does not require consideration of simultaneous contributions to dose from the plant.

Normal and accident doses from the dry cask storage system are subject to control by the plant's radiation protection program including the Offsite Dose Calculation Manual.

1.9 PLANT MANAGEMENT

1.9.1 Organizational Structure

The organizational structure and reporting relationships are described in the Quality Assurance Topical Report, NO-AA-10.

1.9.2 Operator Training

The operating, maintenance, technical, and administrative staffs receive extensive training and instruction in academic subjects and practical operations. These instructions are given both within and outside the plant to qualify the staff for their responsibilities and to enable them to obtain NRC operator and senior operator licenses where required (Section 13.0).

1.9.3 Safety Responsibilities

The licensee is responsible for personnel selection and training, all plant operations, and the execution of written normal and emergency procedures. The General Electric Company is responsible for the design of the nuclear steam supply system and for providing technical direction as requested.

1.9.4 Emergency Plans

Detailed, written procedures cover all anticipated emergencies. The appropriate personnel are trained in these procedures, and periodic drills and reviews are conducted. See Appendix O.

1.9.5 Cooperation With Other Agencies

A coordinated program is established to cover emergencies at the site. It includes liaison with such agencies as local fire and police departments, state police, public health authorities, and local hospitals. See Appendix O.

1.10 QUALITY ASSURANCE PROGRAM

The basic objectives of the quality assurance (QA) program are to ensure the required degree of functional integrity, safety, and reliability of the safety-related structures, systems, and components of Units 2 and 3, and to ensure the required availability for the electric power generating capability of the plant for generation of electricity to meet the system requirements.

The responsibility for the design, construction, testing, and operation of Peach Bottom Units 2 and 3 rests with the licensee.

QA programs were developed by the licensee, the General Electric Company, and Bechtel Corporation and implemented during the design, construction, startup, and operations phase of the units.

The licensee appointed experienced graduate engineers, functioning as the Quality Assurance Engineers for the Peach Bottom Project, to coordinate the development and implementation of the QA program during design, construction, and startup. The licensee also retained MPR Associates, Incorporated, an independent QA consultant, to consult during this phase.

The General Electric Company, Atomic Power Equipment Department, had the responsibility of implementing the QA program for the nuclear steam supply system.

Bechtel Corporation had the responsibility of implementing the QA program for the balance of plant equipment and the field construction work.

The QA program is detailed in Appendix D.

1.11 STATION RESEARCH DEVELOPMENT AND FURTHER INFORMATION; REQUIREMENTS AND RESOLUTIONS SUMMARY

1.11.1 General

The design of the General Electric BWR for this station is based upon proven technological concepts developed during the development, design, and operation of numerous similar reactors. The AEC, in reviewing the Browns Ferry and Peach Bottom dockets at the construction permit stage identified several areas where further R&D efforts were required to more definitely assure safe operation of this station. Also, both the AEC Staff and the Advisory Committee for Reactor Safeguards (ACRS) in their review of other reactor projects, identified several additional technical areas for which further detailed support information was requested. All of these development efforts thus far were of three general types: (1) those which pertain to the broad category of water-cooled reactors, (2) those which pertain specifically to BWR's, and (3) those which have been noted particularly for a facility during the construction permit licensing activities by the AEC Staff and ACRS reviews.

Appendix J of this FSAR provides a complete, comprehensive examination and discussion of each of these concern areas and the Peach Bottom construction permit concerns, indicating the planned or accomplished resolution. A summary conclusion of the analysis is provided in this subsection by Tables 1.11.2 through 1.11.5.

1. Areas Specified in the Peach Bottom AEC-ACRS Construction Permit Letters (refer to Table 1.11.2).

2. Areas Specified in the Peach Bottom AEC-ACRS Construction Permit Safety Evaluation (refer to Table 1.11.3).

3. Areas Specified in Other Related AEC-ACRS Construction and Operating Permit Letters (refer to Table 1.11.4).

4. Areas Specified in Other Somewhat Related AEC-Staff Construction and Operating Permit Evaluation Reports (refer to Table 1.11.5).

The scope of many of the areas of technology for items in 1, 2, and 3 is discussed in detail as part of an official response[1] by the General Electric Company to the various ACRS concern subjects. The General Electric Company has submitted many topical reports to the AEC/NRC in support of the original FSAR application and those of other GE-BWR facilities (refer to Table 1.11.1). Some of the reports that have been submitted since the original FSAR filing are referenced in the particular section to which they apply.

1.11 STATION RESEARCH DEVELOPMENT AND FURTHER INFORMATION; REQUIREMENTS AND RESOLUTIONS SUMMARY

REFERENCE

1. Bray, A.P., et al, APED-5608, "The General Electric Company, Analytical and Experimental programs for Resolution of ACRS Safety Concerns," APED-5608, April, 1968.

TABLE 1.11.1

PEACH BOTTOM ATOMIC POWER STATION UNITS 2 AND 3

TOPICAL REPORTS SUBMITTED TO THE AEC IN SUPPORT OF DOCKET

GE Report No.		Title
1.	APED-5286	Design Basis for Critical Heat Flux in Boiling Water Reactors (September, 1966)
2.	APED-5446	Control Rod Velocity Limiter (March, 1967)
3.	APED-5449	Control Rod Worth Minimizer (March, 1967)
4.	APED-5450	Design Provisions for Inservice Inspection (April, 1967)
5.	APED-5453	Vibration Analysis and Testing of Reactor Internals (April, 1967)
6.	APED-5555	Impact Testing on Collet Assembly for Control Rod Drive Mechanism 7RDB144A (November, 1967)
7.	TR67SL211	An Analysis of Turbine Missiles Resulting from Last Stage Wheel Failure (October, 1967) (No longer applicable)
8.	APED-5608	General Electric Company Analytical and Experimental Program for Resolution of ACRS Safety Concerns (April, 1968) (Not Class I)
9.	APED-5455	The Mechanical Effects of Reactivity Transients (January, 1968)
10.	APED-5528	Nuclear Excursion Technology (August, 1967)
11.	APED-5448	Analysis Methods of Hypothetical Super-Prompt Critical Reactivity Transients in Large Power Reactors (April, 1968)
12.	APED-5458	Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors (March, 1968)

TABLE 1.11.1 (Continued)

GE Report No.		Title
13.	APED-5640	Xenon Considerations in Design of Large Boiling Water Reactors (June 1968)
14.	APED-5454	Metal Water Reactions - Effects on Core Cooling and Containment (March 1968)
15.	APED-5456	In-Core Nuclear Instrumentation Systems for Oyster Creek Unit 1 and Nine Mile Point Unit 1 Reactors (August 1968)
16.	APED-5460	Design and Performance of General Electric Boiling Water Reactor Jet Pumps (September 1968)
17.	APED-5654	Considerations Pertaining to Containment Inerting (August 1968)
18.	APED-5696	Tornado Protection for the Spent Fuel Fuel Storage Pool (November 1968)
19.	APED-5706	In-Core Neutron Monitoring System for General Electric Boiling Water Reactors Rev. 1 (April 1969)
20.	APED-5703	Design and Analysis of Control Rod Drive Reactor Vessel Penetrations (November 1968)
21.	APED-5698	Summary of Results Obtained from a Typical Startup and Power Test Program for a General Electric Boiling Water Reactor (February 1969)
22.	APED-5750	Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves (March 1969)
23.	APED-5756	Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor (March 1969)
24.	APED-5652	Stability and Dynamic Performance of the General Electric Boiling Water Reactor (April 1969)

TABLE 1.11.1 (Continued)

GE Report No.		Title
25.	APED-5736	Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards (April 1969)
26.	APED-5447	Depressurization Performance of the General Electric Boiling Water Reactor High Pressure Coolant Injection System (June 1969)
27.	NEDO-10017	Field Testing Requirements for Fuel, Curtains, and Control Rods (June 1969)
28.	NEDO-10029	An Analytical Study on Brittle Fracture of GE-BWR Vessel Subject to the Design Basis Accident (July 1969)
29.	NEDO-10045	Consequences of a Steam Line Break for a General Electric Boiling Water Reactor (October 1969)
30.	NEDO-10173	Current State of High Performance BWR Zircaloy-Clad UO_2 Fuel (May 1970)
31.	NEDO-10139	Compliance of Protection Systems to Industry Criteria: GE BWR Nuclear Steam Supply System (June 1970)
32.	NEDO-10179	Effects of Cladding Temperature and Material on ECCS Performance (June 1970)
33.	NEDO-10208	Effects of Fuel Rod Failure on ECCS Performance (August 1970)
34.	NEDO-10174	Consequences of a Postulated Flow Blockage Incident in a Boiling Water Reactor (May 1970)
35.	NEDO-10189	An Analysis of Functional Common-Mode Failures in GE BWR Protection and Control Instrumentation (July 1970)

Other Reports Submitted to the AEC Which Are Not Topicals

GE Report No.	Title
NEDO-10183	Core Standby Cooling Systems for the General Electric 1969 BWR Standard Plants (May 1970)
APED-5479	Fuel Rod Failures During Simulated Loss-of-Coolant Conditions (March 1968)
APED-5529	Core Spray and Core Flooding Heat Transfer Effectiveness in a Full-Scale Boiling Water Reactor Bundle (June 1968)

TABLE 1.11.2

AEC-ACRS CONCERNS - RESOLUTIONS

Identification Section No.	AEC-ACRS Concern	Peach Bottom Resolutions
J.2.2	Browns Ferry ACRS Comments (3/14/67) Applicable to Peach Bottom	
J.2.2.1	Effects of Fuel Failure on CSCS Performance	Topical Report (GE-APED-5608) Topical Report (GE-NEDO-10179)
J.2.2.2	Effects of Fuel Bundle Flow Blockage	Topical Report (GE-APED-5608) Topical Report (To be submitted August, 1970)
J.2.2.3	Verification of Fuel Damage Limit Criterion	Topical Report (GE-APED-5608) Dresden 2/3 - Amendment 14/15 Topical Report (GE-NEDO-10173)
J.2.2.4	Effects of Cladding Temperature and Materials on CSCS Performance	Topical Report (GE-APED-5608) Topical Report (GE-APED-5458) Topical Report (GE-NEDO-10179)
J.2.2.5	Quality Assurance and Inspection of the Reactor Primary System	FSAR (Incorporated in design Section 4.0 and Appendices D and I)

TABLE 1.11.2 (Continued)

Identification Section No.	AEC-ACRS Concern	Peach Bottom Resolutions
J.2.2.6	Control Rod Block Monitor Design	FSAR (Incorporated in design Section 7.0 and Appendix G) Dresden 2/3 - Amendments 17/18 and 19/20 Brunswick 1/2 - Supplement 5
J.2.2.7	Plant Startup Program	Topical Report (GE-APED-5698) FSAR (Incorporated in design - Section 13.0)
J.2.2.8	Main Steam Line Isolation Valve Testing Under Simulated Accident Conditions	FSAR (Incorporated in design subsection 4.6) Topical Report (GE-APED-5608) Topical Report (GE-APED-5750) Topical Report (GE-NEDO-10045)
J.2.2.9	Performance Testing of the Station Standby Diesel-Generator System	Not applicable to Peach Bottom 2 and 3 (FSAR subsection 8.5 and Appendix J)
J.2.2.10	Formulation of an In-Service Inspection Program	FSAR (Incorporated in design Appendix I)
J.2.2.11	Diversification of CSCS Initiation Signals	FSAR (Incorporated in design - Sections 6.0 and 7.0)
J.2.2.12	Control Systems for	FSAR (Incorpor-

	Emergency Power	ated in design - subsection 8.5 and Appendix J)

TABLE 1.11.2 (Continued)

Identification Section No.	AEC-ACRS Concern	Peach Bottom Resolutions
J.2.2.13	Misorientation of Fuel Assemblies	FSAR (Incorporated in design Section 3.0)
J.2.3	Failure of Conowingo Dam-Alternate Heat Removal Capability	FSAR (Incorporated in design - subsection 10.24)
J.2.4	Ring Header Leakage Protection Capability	FSAR (Incorporated in design - Appendix J)
J.2.5	Station Thermal Effect - Commonwealth of Pennsylvania Limits	FSAR (Incorporated in design - Sections 11.0 and 12.0)
J.2.6	HPCIS - Depressurization Capability	Topical Report (GE-APED-5608) Topical Report (GE-APED-5947) FSAR (Incorporated in design - Section 6.0)
J.2.7	Station Startup Program	Topical Report (GE-APED-5698) FSAR (Incorporated in design -

TABLE 1.11.3

AEC-STAFF CONCERNS - RESOLUTIONS

Identification Section No.	AEC-Staff Concern	Peach Bottom Resolutions
J.3.2	AEC-Staff SER Section 2.0 Concerns	
J.3.2.2	RPS--IEEE-279 Design	FSAR (Incorporated in design - Sections 6.0 and 7.0. See also Table 1.11.2, item J.2.2.6 and Table 1.11.5, item J.5.6) Brunswick 1/2 - Supplements 5 and 6. Dresden 2/3 - Amendments 17/18 and 19/20. Hatch 1, Amendment 6.
J.3.3	AEC-Staff SER Section 3.0 Concerns	
J.3.3.2	Station Meteorological Program	FSAR (Incorporated in design - Section 2.0)
J.3.3.3	Station-Site Slope Cut Program Studies	FSAR (Incorporated in design - subsection 2.9)
J.3.3.4	Station-Site Flood Protection Studies	FSAR (Incorporated in design - Section 12.0)
J.3.3.5	Station-Site Diffusion and Dispersion Studies-Radiological Effects Determination	FSAR (Incorporated in design - subsection 2.4)

TABLE 1.11.3 (Continued)

Identification Section No.	AEC-Staff Concern	Peach Bottom Resolutions
J.3.3.6	Station Alternate Heat Sink in Event of Dam Failure - Design Capability	See Table 1.11.2, item J.2.3
J.3.4	AEC-Staff SER Section 4.0 Concerns	
J.3.4.2	Suction Piping System Supply Water to ECCS (CSCS) Design Aspects	FSAR (Incorporated in design - Appendix J). See Table 1.11.2, item J.2.6
J.3.4.3	Adequacy of HPCIS as a Depressurizer	
J.3.4.4	Engineered Safety Features-Electrical Equipment Inside Primary Containment-Design Capabilities	FSAR (Incorporated in design - Section 7.0) Millstone 1-Amendment 18
J.3.5	AEC-Staff SER Section 5.0 Concerns	
J.3.5.2	Steam Line Break Fuel Rod Integrity-Thermal Hydraulic Analytical Justification	See Table 1.11.2, item J.2.2.8
J.3.6	AEC-Staff SER Section 6.0 Concerns	
J.3.6.2	Development Program of Significance for All Large Water-Cooled Power Reactors	
	a. Linear Heat Generation Rate-Fuel Damage Limit	See Table 1.11.2, item J.2.2.3

TABLE 1.11.3 (Continued)

Identification Section No.	AEC-Staff Concern	Peach Bottom Resolutions
	b. Local Fuel Melting Resulting from Inlet Coolant Orifice Blockage	See Table 1.11.2, item J.2.2.2
	c. Effect of Fuel Clad Failure on Emergency Core Cooling	See Table 1.11.2, item J.2.2.1
J.3.6.3	Development Program of Significance for BWR's in General	
	a. Core Spray Effective-ness	See Table 1.11.2, item J.2.2.4
	b. Steam Line Isolation Valve Testing Under Simulated Accident Conditions	See Table 1.11.2, item J.2.2.8
	c. Control Rod Worth Minimizer	Topical Report (GE-APED-5449) FSAR (Incorpo-rated in design - subsection 7.16)
	d. Control Rod Velocity Limiter	Topical Report (GE-APED-5446) FSAR (Incorpo-rated in design - subsection 3.4)
	e. In-Core Neutron Monitor System	Topical Report (GE-APED-5456) Topical Report (GE-APED-5706) FSAR (Incorpo-rated in design - subsection 7.5)

f. Jet Pump Development	Topical Report (GE-APED-5460)

TABLE 1.11.3 (Continued)

Identification Section No.	AEC-Staff Concern	Peach Bottom Resolutions
J.3.6.4	Areas Requiring Further Technical Information	
	a. CSCS Thermal Effects on the Reactor Vessel and Internals	Topical Report (GE-NEDO-10029) FSAR (Incorporated in design - Sections 3.0 and 4.0 and Appendix C)
J.3.6.4	b. Interchannel Flow Stability	FSAR (Incorporated in design - subsection 7.17) Topical Report (GE-APED-5652) Topical Report (GE-APED-5640) GE Memorandum SCER-60, July, 1967
	c. In-Service Inspection	FSAR (Incorporated in design - Appendix I)
	d. Primary System Leakage Detection	FSAR (Incorporated in design - subsection 4.10)

TABLE 1.11.4

AEC-ACRS CONCERNS ON OTHER DOCKETS

CAPABILITY FOR RESOLUTION

Identification Section No.	AEC-ACRS Concern	Peach Bottom Capability for Resolution
J.4.2	Instrumentation for Prompt Detection of Gross Fuel Failures	Brunswick 1/2-Supplements 3 and 4 Duane Arnold 1
J.4.3	AEC General Design Criteria #35 Design Intent and Conformance	FSAR (Incorporated in design - Appendix H)
J.4.4	Scram Reliability Study	Brunswick 1/2, Supplement 6 Study Results (To be available late 1970)
J.4.5	Design Basis of Engineered Safety Features	FSAR (Examined capability of design - sub-section 14.9)
J.4.6	Hydrogen Generation Study	Topical Report (GE-APED-5454) Topical Report (GE-APED-5654) Brunswick 1/2, Supplement 4
J.4.7	Seismic Design and Analysis Models	FSAR (Confirma-tion of design - Appendices A and C)
J.4.8	Automatic Pressure Relief System-Single Component Failure Capability-Manual Operation	FSAR (Incorpo-rated in design - Sections 6.0 and 8.0)

J.4.9	Flow Reference Scram	FSAR (Incorpo- rated in design -

TABLE 1.11.4 (Continued)

Identification Section No.	AEC-ACRS Concern	Peach Bottom Capability for Resolution
J.4.10	Main Steam Lines - Standards for Fabrication, Q/C, and Inspection	FSAR (Incorporated in design - Appendices A and I)
J.4.11	Main Steam Line Isolation Valve Leakage	FSAR (Incorporated in design - Appendix J)
J.4.12	Reactor Startup Vibration Testing Capability	FSAR (Incorporated in design - Appendix J)

TABLE 1.11.5

AEC-STAFF CONCERNS ON OTHER DOCKETS

CAPABILITY FOR RESOLUTION

Identification Section No.	AEC - Staff Concern	Peach Bottom Capability for Resolution
J.5.2	Tornado and Missile Protection - GE-BWR - Spent Fuel Storage Pool	FSAR (Incorporated in design - Sections 10.0 and 12.0) Topical Report (GE-APED-5696) Topical Report (Bechtel Corporation, July, 1969)
J.5.3	BWR System Stability Analysis	FSAR (Incorporated in design - subsection 7.17) Topical Report (GE-APED-5652) Topical Report (GE-APED-5640) GE Memorandum SCER-60, July, 1967
J.5.4	RPV-Stub Tube Design	FSAR (Incorporated in design - Section 4.0) Topical Report (GE-APED-5703)
J.5.5	RPS and CSCS Instrumentation - Cable Markings and Identification	FSAR (Incorporated in design - Appendix J)
J.5.6	RPS and CSCS Instrumentation - Design Criteria (IEEE-279)	FSAR (Incorporated in design - Sections 5.0, 6.0, 7.0, Appendix G) Topical Report (To be available mid-1970) Dresden 2/3 - Amendments 17/18 and 19/20

1.12 EXTENDED POWER UPRATE COMPUTER CODES/METHODOLOGIES

1.12.1 General

The EPU License Amendment Request (LAR) was submitted in September of 2012, ML122860201. The NRC reviewed and approved the LAR in August 2014. License Amendments 293 and 296 were issued by the NRC which approves EPU at Peach Bottom Units 2 & 3. The LAR was prepared following the guidelines contained in NRC-approved General Electric (GE) Licensing Topical Report (LTR) NEDC-33004P-A, "Constant Pressure Power Uprate." The Constant Pressure Power Uprate (CPPU) LTR is commonly referred to as the "CLTR."

The evaluation methods and conclusions of the CLTR were approved for GE fuel up through GE14 fuel assemblies. The PBAPS, Units 2 and 3 cores at the time of EPU implementation consist only of GNF2 fuel. As such, certain evaluations and conclusions of the CLTR are not applicable for fuel design-dependent evaluations supporting the PBAPS EPU. For fuel-dependent topics, the PBAPS application used the guidance in NRC-approved GE LTRs NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," and NEOC-32S23P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate." These two LTRs are commonly referred to as "ELTRl" and "ELTR2," respectively.

Table 1.12.1 lists the computer codes used for EPU analyses. These codes were used in various EPU task reports which helped form the basis of the EPU LAR.

TABLE 1.12.1

COMPUTER CODES USED FOR EXTENDED POWER UPRATE ANALYSIS

Task	Computer Code*	Version or Revision	NRC Approved	Comments
Nominal Reactor Heat Balance	ISCOR	09	Y(2)	NEDE-24011P Rev. 0 SER
Power/Flow Map	ISCOR	09	Y(2)	NEDE-24011P Rev. 0 SER
Reactor Core and Fuel Performance	TGBLA PANACEA ISCOR	06 11 09	Y Y Y(2)	NEDE-30130-P-A (4) NEDE-30130-P-A (4) NEDE-24011P Rev. 0 SER
Thermal Hydraulic Stability	ISCOR PANACEA ODYSY TRACG	09 11 05 04	Y(2) Y Y N(15)	NEDE-24011P Rev. 0 SER NEDE-30130-P-A (4) NEDE-33213P-A NED0-32465-A
Reactor Pressure Vessel (RPV) Fluence	TGBLA DORTG	06 01	Y N	14) (12) and (13)
Reactor Internal Pressure Differences (RIPDs)	ISCOR LAMB TRACG	09 07 02	Y(2) (3) Y	NEDE-24011P Rev. 0 SER NEDE-20566-P-A NEDE-32176P Rev. 2 NEDC-32177P Rev. 2 NRC TAC No. M90270
Reactor Vessel Integrity - Stress and Fatigue Evaluation	ANSYS FatiguePro ANSYS Mechanical-APDL and PrepPost VESLFAT PIPEFAT	11 3.0 12.1 x 64 2.0 1.03	N N(17) N(17) N(17) N(17)	(1) (17) (17) (17) (17)
RPV Fluid Induced Vibration	ANSYS	11	N	(1)
Reactor Recirculation System (RRS)	BILBO	04V	N/A	NEDE-23504, February 1977 (1)
Reactor Coolant Pressure Boundary Piping	ME-101	N9/ May 2004	N	(9)
Piping Components Flow Induced Vibration (FIV)	SAP4G07	07	N	GE NED0-10909 (1)

TABLE 1.12.1 (continued)

COMPUTER CODES USED FOR EXTENDED POWER UPRATE ANALYSIS

Task	Computer Code*	Version or Revision	NRC Approved	Comments
Transient Analysis	PANACEA	11	Y	NEDE-30130-P-A, NEDE-24011-P-A (4)
	ISCOR	09	Y(2)	NEDE-24011P Rev.0 SER
	ODYN	10	Y	NEDO-24154-A
	SAFER	04	(5)	NEDE-23785-1-PA, Rev. 1; NEDC-30996P-A (8)(10)
	TASC	03	Y	NEDC-32084P-A Rev. 2
Anticipated Transient Without Scram	ODYN	10	Y	NEDE-24154P-A Supp. 1, Vol. 4
	STEMP	04	(6)	
	PANACEA	11	Y(4)	NEDE-30130-P-A
	ISCOR	09	Y(2)	NEDE-24011P Rev. 0 SER
	TASC	03	Y	NEDC-32084P-A Rev. 2 (11)
Containment System Response	SHEX	06	Y	(7)
	M3CPT	05	Y	NED0-10320, Apr. 1971 (NUREG-0661)
	LAMB	08	(3)	NEDE-20566-P-A September 1986
Annulus Pressurization (AP)	ISCOR	09	Y(2)	NEDE-24011P Rev. 0 SER
Appendix R Fire Protection	GESTR	08	(5)	NEDE-23785-1-PA Rev. 1
	SAFER	04	(5)	(8)(10)
	SHEX	06	Y	(7)
Decay Heat for Spent Fuel Pool Heat Load	TGBLA	06	Y(4)	NEDE-30130-P-A
	PANACEA	11	Y(4)	NEDE-30130-P-A
	DECAY	1	N(1)	Based on ANSI/ANS-5.1-1979
ECCS-LOCA	LAMB	08	Y	NED0-20566A
	GESTR	08	Y	NEOE-23785-1-PA Rev. 1
	SAFER	04	Y	(8)(10)
	ISCOR	09	Y(2)	NEDE-24011P Rev. 0 SER
	TASC	03	Y	NEDC-32084P-A Rev. 2 (11)
Fission Product Inventory	ORIGEN	2.1	N(16)	Isotope Generation and Depletion Code
Station Blackout	SHEX	06A	Y	(7)
Accident Radiological Analysis	RAD TRAD	1998/1999/2002	Y	NUREG/CR-6604
	PAVAN	02	Y	Oak Ridge National Laboratory

* The application of these codes to the EPU analyses complies with the limitations, restrictions, and conditions specified in the approving NRC SER

where applicable for each code. The application of the codes also complies with the SERs for the EPU programs.

TABLE 1.12.1 (continued)

COMPUTER CODES USED FOR EXTENDED POWER UPRATE ANALYSIS

(1) Not a safety analysis code that requires NRC approval. The code application is reviewed and approved by GEH for "Level-2" application and is part of GEH's standard design process. Also, the application of this code has been used in previous power uprate submittals.

(2) The ISCOR code is not approved by name. However, the SER supporting approval of NEDE-2401lP Rev. 0 by the May 12, 1978, letter from D.G. Eisenhut (NRC) to R. Gridley (GE) finds the models and methods acceptable, and mentions the use of a digital computer code. The referenced digital computer code is ISCOR. The use of ISCOR to provide core thermal-hydraulic information in RIPDs, Transient, ATWS, Stability, Reactor Core and Fuel Performance and LOCA applications is consistent with the approved models and methods.

(3) The LAMB code is approved for use in ECCS-LOCA applications (NEDE-20566-P-A and NED0-20566A), but no approving SER exists for the use of LAMB in the evaluation of RlPDs or containment system response. The use of LAMB for these applications is consistent with the model description of NEDE-20566-P-A.

(4) The physics code PANACEA provides inputs to the transient code ODYN. The improvements to PANACEA that were documented in NEDE-30130-P-A were incorporated into ODYN by way of Amendment 11 of GESTAR II (NEDE-24011-PA). The use of TGBLA Version 06 and PANACEA Version 11 in this application was initiated following approval of Amendment 26 of GEST AR II by letter from S.A. Richards (NRC) to G.A. Watford (GE) Subject: "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GEST AR II Implementing Improved GE Steady- State Methods," (TAC NO. MA6481), November 10, 1999. TGBLA06 with Error Correction 6 was used in the PBAPS Core Design analysis and it meets the requirements established by the Safety Evaluation for Licensing Topical Report NEDC-33l73P.

(5) The ECCS-LOCA codes are not explicitly approved for Transient or Appendix R usage. The staff concluded that SAFER is qualified as a code for best estimate modeling of loss-of-coolant accidents and loss of inventory events via the approval letter and evaluation for NEDE-23785P, Revision 1, Volume II, (Letter, C.0. Thomas (See NRC) to J. F. Quirk (GE), "Review of NEDE-23785-1(P), "GESTRLOCA and SAFER Models for the Evaluation of the Loss-of-Coolant
Accident, Volumes I and II," August 29, 1983.)). In addition, the use of SAFER in the analysis of long term Loss-of-Feedwater (LOFW) events is specified in the approved L TRs for power uprate: "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate,"NEDC-32424P-A, February 1999 and "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32523P-A, February 2000. The Appendix R events are similar to the LOFW and small break LOCA events.

(6) The STEMP code uses fundamental mass and energy conservation laws to calculate the suppression pool heatup. The use of STEMP was noted in NEDE-24222, "Assessment of BWR Mitigation of ATWS, Volume I & II (NUREG-0460 Alternate No. 3) December 1, 1979." The code has been used in ATWS applications since that time. It has also recently been accepted in the NRC

review ofNEDC-33270, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)." There is no formal NRC review and approval of STEMP.

TABLE 1.12.1 (continued)

COMPUTER CODES USED FOR EXTENDED POWER UPRATE ANALYSIS

(7) The application of the methodology in the SHEX code to the containment response is approved by the NRC in the letter to G. L. Sozzi (GE) from A. Thadani (NRC), "Use of the SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," July 13, 1993.

(8) Letter, Richard E. Kingston (GEH) to NRC, "Transmittal of Revision 1 of NEDC- 32950, Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model," MFN 07-406, July 31, 2007.

(9) ME-10l is a Bechtel Corporation linear elastic analysis of piping program used by Exelon for analysis of the Main Steam (MS) piping. ME-101 is not a safety analysis code that requires NRC approval. Exelon validation and verification of the ME-101 program and related approval data is stored in Exelon APPID Number EX0006876.

(10) SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants," NEDE-30996P-A, General Electric Company, October 1987.

(11) The NRC approved the TASC-03A code by letter from S.A. Richards (NRC) to J.F. Klapproth (GE Nuclear Energy), Subject: "Review of NEDC-32084P, TASC-03A, A Computer Code for Transient Analysis of a Single Fuel Channel," TAC NO. MB0564, March 13, 2002.

(12) CCC-543, "TORT-DORT Two-and Three-Dimensional Discrete Ordinates Transport Version 2.8.14," Radiation Shielding Information Center (RSlC), January 1994.

(13) Letter, H.N. Berkow (USNRC) to G.B. Stramback (GE), "Final Safety Evaluation Regarding Removal of Methodology Limitations for NEDC-32983P-A, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations (TAC No. MC3788)," November 17, 2005.

(14) Letter, S.A. Richards (USNRC) to G. A. Watford (GE), "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II-Implementing Improved GE Steady-State Methods (TAC No. MA6481)," November 10, 1999.

(15) TRACG02 has been approved in NED0-32465-A by the U.S. NRC for the stability Delta CPR over Initial CPR Versus Oscillation Magnitude (DIVOM) analysis. The CLTP stability analysis is based on TRACG04, which has been shown to provide essentially the same or more conservative results in DIVOM applications as the previous version, TRACG02.

(16) The use of ORIGEN 2.1 to calculate the core source term is accepted for use per Section 3.1 of Regulatory Guides l.183 and 1.195. NRC approval requires the review (and approval) of a Licensing Topical Report (LTR) regarding the use of ORIGEN 2.1 for certain applications.

(17) Software used for Environmental Assisted Fatigue analysis. Results of these codes have been reviewed by the NRC in previous industry analysis.

1.13 MELLLA+ COMPUTER CODES/METHODOLOGIES

1.13.1 General

The MELLLA+ License Amendment Request (LAR) was submitted in September of 2014 (ML14247A503). The NRC reviewed and approved the LAR in March 2016. License Amendments 305 and 309 were issued by the NRC which approves MELLLA+ at Peach Bottom Units 2 & 3. The LAR was prepared following the guidelines contained in NRC-approved General Electric (GE) Licensing Topical Report (LTR) NEDC-33006P-A, “Maximum Extended Load Line Limit Analysis Plus.” This report provides a systematic disposition of the M+LTR subjects applied to PBAPS, including performance of plant-specific assessments and confirmation of the applicability of generic assessments to support a MELLLA+ core flow operating domain expansion. Table 1.13.1 lists the computer codes used for MELLLA+ analyses. These codes were used in various MELLLA+ task reports which helped form the basis of the MELLLA+ LAR.

TABLE 1.13.1

COMPUTER CODES USED FOR MELLLA+ ANALYSIS

Task	Computer Code*	Version or Revision	NRC Approved	Comments
Reactor Heat Balance	ISCOR	09	Y(1)	NEDE-24011P Rev. 0 SER
Reactor Core and Fuel Performance	TGBLA PANAC ISCOR	06 11 09	Y(2) Y(2) Y(1)	NEDE-30130-P-A NEDE-30130-P-A NEDE-24011P Rev. 0 SER
Thermal Hydraulic Stability	ODYSY ISCOR PANAC TRACG	05 09 11 04	Y Y(1) Y(3) N(14)	NEDE-33213P-A NEDE-24011P Rev. 0 SER NEDE-30130P-A NEDE-33147P-A Rev. 4
Reactor Internal Pressure Differences	LAMB TRACG ISCOR	07 02 09	(4) (5) Y(1)	NEDE-20566P-A, September 1986 NRC TAC No. M90270, September 1994 NEDE-24011P Rev. 0 SER
Reactor Recirculation System	BILBO	04V	(8)	NEDE-23504, February 1977
Reactor Pressure Vessel (RPV) Fluence	TGBLA DORTG	06 01	Y N	(2) (11) (12)
Containment System Response	M3CPT LAMB	05 08	Y (4)	NEDM-10320, March 1971 NEDE-20566-P-A, September 1986
Annulus Pressurization Loads	ISCOR	09	Y(1)	NEDE-24011P Rev. 0 SER
ECCS-Loss-of-Coolant-Accident (LOCA)	LAMB PRIME SAFER ISCOR TASC	08 03 04 09 03	Y Y(15) Y Y(1) Y	NEDE-20566P-A NEDC-332S6P-A Rev.1, NEDC-33257PA Rev. 1, NEDC-332S8P-A Rev. 1 (9) (10) NEDE-24011P Rev. 0 SER NEDC-32084P-A Rev. 2
Transient Analysis	PANAC ISCOR TRACG	11 09 04	Y(6) Y(1) Y	NEDE-30130P-A NEDE-24011P Rev. 0 SER NEDE-32906P-A Rev. 3, NEDE-32906P Supp. 3-A, Rev. 1 (6)

TABLE 1.13.1 (continued)

COMPUTER CODES USED FOR MELLLA+ ANALYSIS

Task	Computer Code*	Version or Revision	NRC Approved	Comments
Anticipated Transient Without Scram	ODYN	10	Y	NEDE-24154P-A Supp. 1, Vol. 4
	STEMP	04	(7)	
	PANAC	11	Y(6)	NEDE-30130P-A
	TASC	03	Y	NEDC-32084P-A Rev. 2
	ISCOR	09	Y(1)	NEDE-24011P Rev. 0 SER
	TRACG	04	Y(13)	NEDE-32906P Supp. 3-A, Rev. 1

* The application of these codes to the MELLLA+ analyses complies with the limitations, restrictions, and conditions specified in the approving NRC SER where applicable for each code. The application of the codes also complies with the SERs for the MELLLA+ programs.

(1) The ISCOR code is not approved by name. However, in the SER supporting approval of NEDE-24011P Revision 0 by the May 12, 1978 letter from D. G. Eisenhut (NRC) to R. Gridley (GE), the NRC finds the models and methods acceptable for steady-state thermal-hydraulic analysis, and mentions the use of a digital computer code. The referenced digital computer code is ISCOR. The use of ISCOR to provide core thermal-hydraulic information in reactor internal pressure differences (RIPDs), transient, ATWS, stability, and LOCA applications is consistent with the approved models and methods.

(2) The use of TGBLA Version 06 and PANAC Version 11 was initiated following approval of Amendment 26 of GESTAR II by letter from S. A. Richards (NRC) to G. A. Watford (GE) Subject: “Amendment 26 to GE Licensing Topical Report NEDE-24011P-A, GESTAR II Implementing Improved GE Steady-State Methods,” (TAC NO. MA6481), November 10, 1999.

(3) The use of PANAC Version 11 was initiated following approval of Amendment 26 of GESTAR II by letter from S.A. Richards (NRC) to G.A. Watford (GE) Subject: "Amendment 26 to GE Licensing Topical Report NEDE-24011P-A, GESTAR II Implementing Improved GE Steady-State Methods," (TAC NO. MA6481), November 10, 1999.

(4) The LAMB code is approved for use in ECCS-LOCA applications (NEDE-20566-P-A), but no approving SER exists for the use of LAMB for the evaluation of RIPDs or containment system response. The use of LAMB for these applications is consistent with the model description of NEDE-20566-P-A.

(5) NRC has reviewed and accepted the TRACG application for the flow-induced loads on the core shroud as stated in NRC SER TAC No. M90270.

(6) The physics code PANAC provides inputs to the transient codes ODYN and TRACG04. The use of PANAC Version 11 in conjunction with TRACG04 was approved by the NRC SE for NEDE-32906P Supplement 3-A, Revision 1, April 2010.

(7) The STEMP code uses fundamental mass and energy conservation laws to calculate the suppression pool heatup. The use of STEMP was noted in NEDE-24222, "Assessment of BWR Mitigation of ATWS, Volume I & II (NUREG-0460 Alternate No. 3)," December 1, 1979. The code has been used in ATWS applications since that time. There is no formal NRC review and approval of STEMP or the ATWS topical report.

(8) Not a safety analysis code that requires NRC approval. The code application is reviewed and approved by GEH for "Level-2" application and is part of GEH's standard design process. Also, the application of this code has been used in other MELLLA+ and power uprate submittals.

(9) General Electric Company, "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants," NEDE-30996P-A, October 1987.

(10) Letter, Richard E. Kingston (GEH) to NRC, "Transmittal of Revision 1 of NEDC-32950, Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model," MFN 07-406, July 31, 2007.

(11) CCC-543, "TORT-DORT Two- and Three-Dimensional Discrete Ordinates Transport Version 2.8.14," Radiation Shielding Information Center (RSIC), January 1994.

(12) Letter, H. N. Berkow (NRC) to G. B. Stramback (GE), "Final Safety Evaluation Regarding Removal of Methodology Limitations for NEDC-32983P-A, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations (TAC No. MC3788)," November 17, 2005.

(13) TRACG04 is approved by the NRC for application to ATWS overpressure transients in NEDE-32906P Supplement 3-A, "Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients," April 2010. ODYN remains the licensing basis code for ATWS long-term analysis consistent with the NRC SE for NEDC-33006P. The use of TRACG04 for the best estimate TRACG ATWS with depressurization and ATWS with instability (ATWSI) analysis is required by the NRC SE for NEDC-33006P.

(14) TRACG04 application for DSS-CD is documented in NEDE-33147P-A Revision 4.

(15) Application of PRIME models and data to downstream methods is approved by NEDO-33173 Supplement 4-A, "Implementation of PRIME Models and Data in Downstream Methods," Revision 1, November 2012.

1.14 MEASUREMENT UNCERTAINTY RECAPTURE COMPUTER CODES/METHODOLOGIES

1.14.1 General

The MUR License Amendment Request (LAR) was submitted in February of 2017, ML17048A444. The NRC reviewed and approved the LAR in November 2017. License Amendments 316 and 319 were issued by the NRC which approves MUR at Peach Bottom, Units 2 and 3. The LAR was prepared following the guidelines contained in NRC-approved General Electric (GE) Licensing Topical Report (LTR) NEDC-32938P-A, "Thermal Power Optimization." The Thermal Power Optimization (TPO) LTR is limited to a maximum rated thermal power level 120 percent of the Original Licensed Thermal Power (OLTP) level. The LTR states that an MUR uprate that would result in the licensed thermal power level in excess of 120 percent of OLTP must provide plant-specific evaluations for those evaluations not performed at 102 percent the Previous Licensed Thermal Power (PLTP) level. Based on previous power uprates, PBAPS, Units 2 and 3, PLTP level of 3,951 MWt is 120 percent of the OLTP level of 3,293 MWt. This MUR power uprate power level of 4,016 MWt is approximately 122 percent of the OLTP level. As such, consistent with the requirements in the GE LTR, additional plant-specific evaluations were provided to support the amendment request. Table 1.14.1 lists the computer codes used for MUR analyses. These codes were used in various MUR task reports which helped form the basis of the MUR LAR.

TABLE 1.14.1

COMPUTER CODES USED FOR MUR ANALYSIS

Task	Computer Code*	Version or Revision	NRC Approved	Comments
Reactor Recirculation System	BILBO	04V	(1)	NEDE-23504, February 1977
Nominal Reactor Heat Balance	ISCOR	09	Y(2)	NEDE-24011P, Rev. 0 SER
Nominal Reactor Pressure Differences	ISCOR	09	Y(2)	NEDE-24011P, Rev. 0 SER
Station Blackout	SHEX	06	Y(3)	
Reactor Core and Fuel Performance	TGBLA PANAC ISCOR	06 11 09	Y(4) Y(4) Y(2)	NEDE-30l30P-A NEDE-30l30P-A NEDE-24011P, Rev. 0 SER
Thermal-Hydraulic Stability	ODYSY ISCOR PANAC TRACG	05 09 11 04	Y Y(4) Y(2) Y	NEDE-33213P-A NEDE-24011P, Rev. 0 SER NEDE-31130P-A NEDE-33147P-A, Rev.4
Piping Components Flow Induced Vibration	SAP4G07P	07	(1)	
Anticipated Transient Without Scram	ODYN STEMP PANACEA ISCOR TRACG TASC	10 04 11 09 04 03	Y (1) Y(4) Y(2) Y Y	NEDE-24154P-A, Suppl. 1, Vol. 4 NEDE-30130P-A NEDE-24011P, Rev. 0 SER NEDE-32906P, Suppl. 3-A, Rev. 1 NEDC-32084P-A, Rev.2
Anticipated Transient Without Scram with Instability	TRACG	04	Y(5)	
Appendix R Fire Protection	SHEX SAFER PRIME	06 04 03	Y(3) Y(6,7) Y(8)	

*The application of these codes to the PBAPS MELLLA+ and MUR analyses complies with the limitations, restrictions, and conditions specified in the approving NRC SER where applicable for each code. The application of the codes also complies with the SERs for the MELLLA+ and MUR programs.

TABLE 1.14.1 (continued)

COMPUTER CODES USED FOR MUR ANALYSIS

Table Notes:

(1) Not a safety analysis code that requires NRC approval. The code application is reviewed and approved by GEH for "Level 2" application and is part of GEH's standard design process. The application of this code has been used in previous power uprate submittals.

(2) The ISCOR code is not approved by name. However, in the SER supporting approval of NEDE-2401lP, Revision 0 by the May 12, 1978, letter from D. G. Eisenhut (NRC) to R. Gridley (GE), the NRC finds the models and methods acceptable, and mentions the use of a digital computer code. The referenced digital computer code is ISCOR. The use of ISCOR to provide core thermal-hydraulic information in reactor internal pressure differences, transient, ATWS, stability, reactor core and fuel performance, and LOCA applications is consistent with the approved models and methods.

(3) The application of the methodology in the SHEX code to the containment response is approved by the NRC in the letter to Gary L. Sozzi (GE) from Ashok Thadani (NRC), "Use of the SHEX Computer Program and ANSI American Nuclear Society (ANS) 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," July 13, 1993 (Reference 20).

(4) The use of TGBLA Version 06 and PANAC Version 11 was initiated following approval of Amendment 26 of GEST AR II by letter from S. A. Richards (NRC) to G. A Watford (GE) Subject: "Amendment 26 to GE LTR NEDE-24011P-A, GESTAR II Implementing Improved GE Steady-State Methods," (TAC NO. MA6481), November 10, 1999.

(5) The TRACG04 code is not approved by the NRC for long-term ATWS calculations including ATWS with depressurization and A TWS with core instability. However, the use of TRACG04 for the best-estimate TRACG ATWS analysis is consistent with the NRC Safety Evaluation (SE) for NEDC-33006P.

(6) General Electric Company, "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants," NEDE-30996P-A, October 1987.

(7) Letter, Richard E. Kingston (GEH) to NRC, "Transmittal of Revision l ofNEDC-32950, Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model," MFN 07-406, July 31, 2007.

(8) Application of PRIME models and data to downstream methods is approved by NED0-33173 Supplement 4-A, "Implementation of PRIME Models and Data in Downstream Methods," Revision l, November 2012.

1.15 RISK INFORMED CATEGORIZATION AND TREATMENT OF SSCS

1.15.1 Introduction

10 CFR 50.69 provides a risk-informed process for classifying systems, structures and components (SSCs). In the traditional approach, SSCs are categorized as either "safety-related" (as defined in 10 CFR 50.2) or "nonsafety-related." By applying risk insights, SSCs can be further classified as being either "safety significant" or "low safety significant." This results in four Risk-Informed Safety Class (RISC) categories:

- RISC-1: Safety-related SSCs that perform safety significant functions.
- RISC-2: Nonsafety-related SSCs that perform safety significant functions.
- RISC-3: Safety-related SSCs that perform low safety-significant functions.
- RISC-4: Nonsafety-related SSCs that perform low safety significant functions.

PBAPS received approval to implement 10 CFR 50.69 by License Amendment 321/324 (Reference 1) on October 25, 2018, in accordance with the methodology described in NEI 00-04 Reference 2), as endorsed by NRC Regulatory Guide (RG) 1.201 (Reference 3).

1.15.2 Scope

This process can be applied to selected systems or structures and implemented over a period of time. Implementation is conducted on entire systems or structures, not selected components within a system. This ensures that all functions for an SSC within a system or structure are appropriately considered when determining safety significance. Systems that have been categorized under this process are identified in UFSAR chapters for each system.

1.15.3 SSC Categorization

Categorization of SSCs is done in accordance with NEI 00-04. The process consists of the following:

A. Risk Characterization.

1. Use of PRA models to evaluate risk associated with internal events, including internal flooding and fire.

2. Apply shutdown safety assessment process (Reference 4) and the station shutdown risk management program to assess shutdown risk.

3. Implement the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method (Reference 5) to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports.

4. Use non-PRA evaluation methods that are based on the IPEEE Screening Assessment for External Hazards (Reference 6).

B. Defense in Depth (DID) assessment to ensure that adequate redundancy and diversity are retained.

C. Risk Sensitivity Study using PRA methods based on a postulated change in reliability.

D. Review by an Integrated Decision-making Panel (IDP) to ensure appropriate considerations have been made for plant design, operating practices and operating experience.

1.15.4 Alternative Treatment

Safety-related SSCs are subject to a specific set of regulations (special treatment) that are not applicable to non-safety related equipment. Compliance with 10 CFR 50.69 provides an alternative to meeting the regulations identified in 10 CFR 50.69(b)(1) for RISC-3 and RISC-4 SSCs.

The performance of RISC-1 and RISC-2 SSCs are monitored to determine if adjustments to the categorization assumptions or treatment processes are necessary. This monitoring can be performed in the same manner as done for 10 CFR 50.65 (Maintenance Rule) except the monitoring addresses all functional failures, not just maintenance preventable functional failures. Since RISC-2 SSCs are non-safety related, enhanced treatment may be warranted to improve the reliability and availability of the SSC in support of its safety significant function.

RISC-3 SSCs must remain capable of performing their safety-related functions under design basis conditions, including seismic and environmental conditions. Periodic inspection and testing activities must be conducted to verify they will remain capable of performing their safety-related functions. Appropriate safety margins must be maintained for RISC-3 SSCs and any increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment must be small, within the sensitivity limits of the risk methods used in categorization.

1.15.5 Periodic Reviews

A periodic review is performed at least once every two refueling outages. The review includes evaluating changes to the plant, operational practices, plant and industry operating experience, SSC performance, impact of updated PRA and other factors that may affect SSC categorization and treatment. This review maintains safety margins for categorized SSCs and any increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment within the sensitivity limits of the risk methods used in categorization.

1.15.6 References

1. U.S. Nuclear Regulatory Commission, "Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendment Nos. 321 And 324 to Adopt 10 CFR 50.69, Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," October 25, 2018 (ML18263A232).

2. Nuclear Energy Institute, "10CFR50.69 SSC Categorization Guideline," NEI 00-04, Rev. 0, July 2005 (ML052900163)

3. U.S. Nuclear Regulatory Commission, "Guidelines for Categorizing Structures, Systems and Components in Nuclear Power Plants According to Their Safety Significance, For Trial Use," Regulatory Guide 1.201, Rev. 1, May 2006 (ML061090627)

4. Nuclear Management and Resources Council, "Guidelines for Industry Actions to Assess Shutdown Management," NUMARC 91-06, December 1991 (ML14365A203)

5. U.S. Nuclear Regulatory Commission, "Arkansas Nuclear One, Unit 2 – Approval of Request for Alternative ANO-2 R&R-004,

Revision 1, Request to Use Risk Informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems," April 22, 2009 (ML090930246)

6. U.S. Nuclear Regulatory Commission, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities – 10 CFR 50.54(f)," Generic Letter 88-20, Supplement 4, June 1991.