ATTACHMENT (1)

UNIT 1

TECHNICAL SPECIFICATION

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3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 Safety Valves

psia

The **OPERABILITY** of the main steam line code safety valves ensures that the Secondary System pressure will be limited to within 110% of its design pressure of 1000 ~~psig~~ during the most severe anticipated system operational transient. The total relieving capacity for all valves on all of the steam lines is 12.18 x 10^6 lbs/hr at 100% **RATED THERMAL POWER.** The maximum relieving capacity is associated with a turbine trip from 100% **RATED THERMAL POWER** coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser). The main steam line code safety valves are tested and maintained in accordance with the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. ~~The as-left lift settings will be no less than 985 psig to ensure that the lift setpoints will remain within specifications during the cycle.~~

In **MODE** 3, two main steam safety valves are required **OPERABLE** per steam generator. These valves will provide adequate relieving capacity for removal of both decay heat and reactor coolant pump heat from the Reactor Coolant System via either of the two steam generators. This requirement is provided to facilitate the post-overhaul setting and **OPERABILITY** testing of the safety valves which can only be conducted when the RCS is at or above 500°F. It allows entry into **MODE** 3 with a minimum number of main steam safety valves **OPERABLE** so that the set pressure for the remaining valves can be adjusted in the plant. This is the most accurate means for adjusting safety valve set pressures since the valves will be in thermal equilibrium with the operating environment.

STARTUP and/or **POWER OPERATION** is allowable with safety valves inoperable within the limitations of the **ACTION** requirements on the basis of the reduction in Secondary System steam flow and **THERMAL POWER** required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following bases:

For two loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times 106.5$$

For single loop operation (two reactor coolant pumps operating in the same loop)

$$SP = \frac{(X) - (Y)(U)}{X} \times 46.8$$

Replace with Insert A

~~In the spectrum of events analyzed in which automatic initiation of auxiliary feedwater occurs, the following flow conditions are allowed with an operator action time of 10 minutes.~~

~~(1)~~	~~Loss of Feedwater~~	~~0 gpm Auxiliary Feedwater Flow~~
~~(2)~~	~~Feedline Break~~	~~0 gpm Auxiliary Feedwater Flow~~
~~(3)~~	~~Main Steam Line Break~~	~~1550 gpm Auxiliary Feedwater Flow (This being the maximum flow through the AFW suction line, with one unit requiring flow, prior to pump cavitation due to low NPSH).~~

At 10 minutes after an Auxiliary Feedwater Actuation Signal the operator is assumed to be available to increase or decrease auxiliary feedwater flow to that required by the existing plant conditions.

3/4.7.1.3 Condensate Storage Tank

The **OPERABILITY** of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at **HOT STANDBY** conditions for 6 hours with steam discharge to atmosphere with concurrent and total loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 Activity

The limitations on Secondary System specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line and concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 Main Steam Line Isolation Valves

The **OPERABILITY** of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The **OPERABILITY** of

INSERT A

Auxiliary Feedwater flow and response times are conservatively accounted for in the analyses of Design Basis Events. In Main Steam Line Break and Excess Load analyses where Auxiliary Feedwater flow would increase the consequences of the accidents, the delay time for Auxiliary Feedwater actuation is minimized and Auxiliary Feedwater flow is maximized. In Feedline Break, Loss of Feedwater, and Loss of Non-Emergency AC Power analyses, in which Auxiliary Feedwater flow would decrease the consequences of the accidents, the delay time for Auxiliary Feedwater actuation is maximized and Auxiliary Feedwater flow is minimized.

accident conditions. The **OPERABILITY** of this system in conjunction with Control Room design provisions is based on limiting the radiation exposure to personnel occupying the Control Room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of 10 CFR Part 50, Appendix A, General Design Criteria ~~10~~. 19

3/4.7.7 ECCS PUMP ROOM EXHAUST AIR FILTRATION SYSTEM

The **OPERABILITY** of the ECCS Pump Room Exhaust Air Filtration System ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effects on offsite dosage calculations was assumed in the accident analyses.

3/4.7.8 SNUBBERS

All safety related snubbers are required **OPERABLE** to ensure that the structural integrity of the Reactor Coolant System and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on non-safety related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during the previous inspection, the total population or category size, and the previous inspection interval.

Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, that decision must be made and documented before any inspection and that decision shall be used as the basis upon which to determine the next inspection interval for that category. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that

3/4.11.1.3 Liquid Radwaste Treatment System

The requirement that the appropriate portions of this system be used, when specified, provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR 50.36a, of 10 CFR Part 50, Appendix A, General Design Criterion 60 and the design objective given in 10 CFR Part 50, Appendix I, Section II.D. The specified limits governing the use of appropriate portions of the Liquid Radwaste Treatment System were specified as a suitable fraction of the dose design objectives set forth in 10 CFR Part 50, Appendix I, Section II.A for liquid effluents. The dose limits apply to the combined effluent of the plant (e.g. two units).

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 Dose Rate

This specification is provided to ensure that the dose at any time at and beyond the **SITE BOUNDARY** from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to **UNRESTRICTED AREAS**. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a **MEMBER OF THE PUBLIC** in an **UNRESTRICTED AREA**, exceeding the limits specified in 10 CFR Part 20, Appendix B, Table II (10 CFR 20.106(b)). For **MEMBERS OF THE PUBLIC** who may at times be within the **SITE BOUNDARY**, the occupancy of that **MEMBER OF THE PUBLIC** will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the **SITE BOUNDARY**.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300, Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J.K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977, Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977, and NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants". These equations also provide for determining the actual doses based upon the historical annual average atmospheric conditions. The release rate specifications for iodine-131 and radionuclides in particulate form with half lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the **SITE BOUNDARY**. The pathways that were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable".

This specification implements the requirements of 10 CFR 50.36a, 10 CFR Part 50, Appendix A, General Design Criterion 60 and the design objectives given in 10 CFR Part 50, Appendix I, Section II.D. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in 10 CFR Part 50, Appendix I, Sections II.B and II.C for gaseous effluents. (The dose limits apply to the combined effluent of the plant (e.g. two units).)

3/4.11.2.5 Explosive Gas Mixture

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the Waste Gas Holdup System is maintained below the flammability limit of oxygen. Maintaining the concentration of oxygen below its flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of 10 CFR Part 50, Appendix A, General Design Criterion 60.

ATTACHMENT (2)

UNIT 2

TECHNICAL SPECIFICATION

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3/4.7.1 TURBINE CYCLE

3/4.7.1.1 Safety Valves

psia

The **OPERABILITY** of the main steam line code safety valves ensures that the Secondary System pressure will be limited to within 110% of its design pressure of 1000 psig during the most severe anticipated system operational transient. The total relieving capacity for all valves on all of the steam lines is 12.18×10^6 lbs/hr at 100% **RATED THERMAL POWER**. The maximum relieving capacity is associated with a turbine trip from 100% **RATED THERMAL POWER** coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser). The main steam line code safety valves are tested and maintained in accordance with the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. ~~The as-left lift settings will be no less than 985 psig to ensure that the lift setpoints will remain within specification during the cycle.~~

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INSERT A

Auxiliary Feedwater flow and response times are conservatively accounted for in the analyses of Design Basis Events. In Main Steam Line Break and Excess Load analyses where Auxiliary Feedwater flow would increase the consequences of the accidents, the delay time for Auxiliary Feedwater actuation is minimized and Auxiliary Feedwater flow is maximized. In Feedline Break, Loss of Feedwater, and Loss of Non-Emergency AC Power analyses, in which Auxiliary Feedwater flow would decrease the consequences of the accidents, the delay time for Auxiliary Feedwater actuation is maximized and Auxiliary Feedwater flow is minimized.

accident conditions. The OPERABILITY of this system in conjunction with Control Room design provisions is based on limiting the radiation exposure to personnel occupying the Control Room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of 10 CFR Part 50, Appendix A, General Design Criteria ~~20~~ 19.

3/4.7.7 ECCS PUMP ROOM EXHAUST AIR FILTRATION SYSTEM

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3/4.7.8 SNUBBERS

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3/4.11.1.3 Liquid Radwaste Treatment System

The requirement that the appropriate portions of this system be used, when specified, provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR 50.36a, 10 CFR Part 50, Appendix A, General Design Criterion 60 and the design objective given in 10 CFR Part 50, Appendix I, Section II.D. The specified limits governing the use of appropriate portions of the Liquid Radwaste Treatment System were specified as a suitable fraction of the dose design objectives set forth in 10 CFR Part 50, Appendix I, Section II.A for liquid effluents. (The dose limits apply to the combined effluent of the plant (e.g. two units).)

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 Dose Rate

This specification is provided to ensure that the dose at any time at and beyond the **SITE BOUNDARY** from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to **UNRESTRICTED AREAS**. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a **MEMBER OF THE PUBLIC** in an **UNRESTRICTED AREA**, exceeding the limits specified in Appendix B, Table II of 10 CFR 20.106(b). For **MEMBERS OF THE PUBLIC** who may at times be within the **SITE BOUNDARY**, the occupancy of that **MEMBER OF THE PUBLIC** will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the **SITE BOUNDARY**.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300, Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J.K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.2.2 Dose - Noble Gases

This specification is provided to implement the requirements of 10 CFR Part 50, Appendix I, Sections II.B, III.A and IV.A. The Limiting Condition for Operation implements the guides set forth in Appendix I, of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977, and NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants". These equations also provide for determining the actual doses based upon the historical annual average atmospheric conditions. The release rate specifications for iodine-131 and radionuclides in particulate form with half lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the **SITE BOUNDARY**. The pathways that were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable".

This specification implements the requirements of 10 CFR 50.36a, 10 CFR Part 50, Appendix A, General Design Criterion 60 and the design objectives given in 10 CFR Part 50, Appendix I, Section II.D. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in 10 CFR Part 50, Appendix I, Sections II.B and II.C for gaseous effluents. The dose limits apply to the combined effluent of the plant (eg. two units).

3/4.11.2.5 Explosive Gas Mixture

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the Waste Gas Holdup System is maintained below the flammability limit of oxygen. Maintaining the concentration of oxygen below its flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of 10 CFR Part 50, Appendix A, General Design Criterion 60.

3/4.11.2.6 Gas Storage Tanks

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification to a quantity that is less than the quantity that provides assurance that in the event of an