BEFORE THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

NRC Docket Nos. 50-424, 50-425

In the Matter of

GEORGIA POWER COMPANY

SUPPLEMENT 5 TO

APPLICATION FOR LICENSE

UNDER THE ATOMIC ENERGY ACT OF 1954

AS AMENDED

FOR

ALVIN W. VOGTLE NUCLEAR PLANT

UNITS 1, 2

The Applicant, Georgia Power Company, hereby supplements its Application for a Construction Permit and Operating License, originally submitted on August 1, 1972, by the addition of supplementary material attached hereto.

This supplement reflects the conclusions reached in our March 3, 1978 meeting with the NRC Staff on the main steam and feedwater system design, and includes the 1.5 scaling factor used in the design of deeply-embedded Category I structures.

W E Ehrensperger

BY: W. E. Ehrensperger,
Senior Vice President

Sworn to and subscribed before me, this 3rd day of November, 1978

Mary J. Winter

Notary Public

My Commission Expires November 24, 1979

7811200259

Georgia Power Company
Post Office Box 4545
230 Peachtree Street, N.W.
Atlanta, Georgia 30302
Telephone 404 522-6060

W. E. Ehrensperger
Senior Vice President
Power Supply

November 17, 1978

Director of Nuclear Reactor Regulation
ATTN: Roger S. Boyd, Director
Division of Project Management
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

NRC DOCKET NUMBERS 50-424 AND 50-425
CONSTRUCTION PERMIT NUMBERS CPPR-108 AND CPPR-109
ALVIN W. VOGTLE NUCLEAR PLANT-UNITS 1 AND 2
SUPPLEMENT 5 TO APPLICATION

Dear Mr. Boyd:

Georgia Power Company hereby files three (3) signed copies and sixty (60) conformed copies of Supplement 5 to its application for a Construction Permit and Operating License for the Alvin W. Vogtle Nuclear Plant, Units 1 and 2.

This supplement reflects the conclusions reached in our March 3, 1978 meeting with the NRC Staff on the main steam and main feedwater system design for the Vogtle Nuclear Plant. It also includes the 1.5 scaling factor that will be used to multiply the "envelope in-structure response spectra" generated for each deeply-embedded seismic Category I structure.

Yours truly,

W E Ehrensperger

W. E. Ehrensperger

cc: R. A. Thomas
G. F. Trowbridge, Esq.
D. E. Dutton
J. A. Bailey
L. T. Gucwa
B. L. Lex
I. S. Mitchell, III
H. A. Sindt

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INSTRUCTION SHEET
SUPPLEMENT NO. 5

ALVIN W. VOGTLE NUCLEAR PLANT
PRELIMINARY
SAFETY ANALYSIS REPORT

DO NOT REMOVE EXISTING WHITE PAGES

Replace Table of Contents pages S2 v and S2 vi with S5 v and S5 vi; and S2 xvii and S2 xviii with S5 xvii through S5 xviiia.

Insert Table of Contents pages S5 3-i through S5 3-via ahead of page 3-i.

Replace Table of Contents pages S3 3-xv and S3 3-xvi with S5 3-xv through S5 3-xvia.

Insert Table of Contents pages S5 10-i through S5 10-iv ahead of page 10-i.

Insert Chapter 3 pages S5 3.1-5 and S5 3.1-6 ahead of page 3.1-5.

Insert Chapter 3 pages S5 3.2-1 and S5 3.2-2 ahead of page 3.2-1.

Insert Chapter 3 pages S5 3.2-3 through S5 3.2-4 ahead of page 3.2-3.

Insert Chapter 3 pages S5 3.5-31 and S5 3.5-32 ahead of page 3.5-31.

Insert Chapter pages S5 3.6-3 through S5 3.6-4a ahead of page 3.6-3.

Insert Chapter 3 pages S5 3.6-5a through S5 3.6-7 ahead of page 3.6-5a.

Replace Chapter 3 page S4 3.7-47 with S5 3.7-47.

Insert Chapter 10 pages S5 10.3-1 and S5 10.3-2 ahead of page 10.3-1.

Insert Chapter 10 pages S5 10.3-5 through S5 10.3-8 ahead of page 10.3-5.

DO NOT REMOVE EXISTING WHITE PAGES

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Firefighting systems are designed to assure that their rupture or inadvertent operation will not significantly impair systems important to safety (see subsection 9.5).

The fire protection system consists of a reliable, partially automatic system designed and installed in accord with the requirements of the National Fire Protection Association, the American Insurance Association, Nuclear Mutual Limited, and the applicable local codes and regulations.

The fire protection system is provided with test hose valves for periodic testing. All equipment is accessible for periodic inspection. The fire protection system is described in section 9.5.

3.1.4 CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

RESPONSE

Structures, systems, and components important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant, main steam line break, and main
S5 feedwater break accidents. Criteria are presented in section 3.5, 3.6, and 3.9.2.7 while environmental conditions are presented in section 6.2.

These structures, systems, and components are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

Chapter 7 lists the motors, instrumentation, and associated cables of protection and safety features systems located inside the containment. It gives the design requirements in terms of

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

3.2.1 SEISMIC CLASSIFICATION

A two-level system is used for the seismic classification of structures, components, and systems other than Westinghouse's scope of the Alvin W. Vogtle Nuclear Plant (VNP):

Category I Structures, Components, and Systems

Category II Structures, Components, and Systems

For Westinghouse design responsibility components refer to RESAR-3, sections 3.2, 3.9 and 5.2. In addition to the standard systems described in RESAR-3 above, refer to tables 6.2-12, 9.3-2, 9.1-2, and 10.4-1 of the PSAR for the safety classification of the Westinghouse components in the containment spray system, boron recycle system, spent fuel pit cooling system and steam generator blowdown system.

The classification of pipes, valves, and fittings is shown on the piping and instrumentation diagrams in the appropriate PSAR sections.

3.2.1.1 Definitions

Seismic Category I structures, components, and systems are defined in accordance with USAEC Safety Guide No. 29 as those necessary to assure:

The integrity of the reactor coolant pressure boundary.

The capability to shut down the reactor and maintain it in a safe shutdown condition.

The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.

Category I structures, components, and systems are designed to withstand the appropriate seismic loads and other applicable loads without loss of function. Category I structures are sufficiently isolated or protected from Category II structures to ensure that their integrity is maintained at all times.

Category II structures, components, and systems are those whose failure would not result in the release of significant radioactivity and would not prevent reactor shutdown. All equipment not specifically listed as Category I is included as Category II. Specifically, all non-Category I systems, equipment and components that are installed in Category I structures are carefully examined to determine the degree, if any, of detrimental effect on safety related systems, should failure of the non-Category I equipment, systems or components occur. Where the detrimental effect can be shown to effect safe shutdown equipment then the non-Category I item is either upgraded to Category I or separated by distance or barricade. The failure of Category II structures, components and systems may interrupt power generation.

3.2.1.2 <u>Category I Structures</u>

Containment Structure

Enclosure Building*

Auxiliary Building

Control Building

Fuel Handling Building

Nuclear Service Cooling Towers and Basins

Diesel Generator Building

S5 | Auxiliary Feedwater Building

Condensate Storage Tanks

Refueling Water Storage Tank

Reactor Makeup Water Storage Tank

Pipe Tunnels and Electrical Cable Tunnels as follows:

A. Pipe Tunnels

1. Auxiliary Feedwater Piping
2. Nuclear Service Cooling Water Piping
3. Refueling Water Piping

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*For structural integrity of the EB, refer to paragraph 3.8.4.4.1.

3.2.1.3.4 Component Cooling Water System

Component Cooling Water Heat Exchanger (Tube Side)
(Shell Side)

Component Cooling Water Pump

Component Cooling Water Surge Tank

3.2.1.3.5 Engineering Safety Features

Radiation Monitoring System

Containment Isolation System

3.2.1.3.6 Conventional Mechanical Section

Main Steam System

Main Steam Piping (from Steam Generator up to and including five-way restraint) | S5

Safety Valves

Isolation Valves

Atmospheric Dump Valve

Steam Blowdown and Sampling Piping (Steam Generator to and Including Isolation Valve)

3.2.1.3.7 Auxiliary and Main Feedwater System | S5

Auxiliary Feedwater Pump

Isolation Valves

Piping (from Steam Generator up to and including five-way restraint) | S5

3.2.1.3.8 Nuclear Service Cooling Water System

Cooling Tower and Holdup Basin

Makeup Pump

Circulating Pump

Piping, Valves, and Fittings

3.2.1.3.9 Diesel Generation System

Diesel Generator

Diesel Fuel Storage Tank

Diesel Fuel Day Tank

Diesel Generator Air Tank

Diesel Fuel Transfer Pump

Diesel Fuel Filter

3.2.1.3.10 Heating, Ventilating, and Air Conditioning

A. Control Room

Emergency Supply Air Charcoal Filter

Emergency Supply Air Absolute Filter

Air Conditioning Unit

B. Control Building Class I Switchgear Ventilation

Supports

Ventilation Fans

The content of this page has been deleted.

3.5.3.3 Tornado Missiles

The characteristics of potential tornado missiles are given in paragraph 3.5.2.4.1 and tables 3.5-6 and 3.5-7.

Using the allowables and analytical techniques given in Appendix 3N, it is determined that a 2'-0" thick reinforced concrete wall could resist the impact effects of all the listed missiles without perforation, spalling or failure due to structural response.

3.5.3.4 Protection of Structures, Systems and Components Against External Missiles

The Category I structures listed below are analyzed for missile damage protection capabilities and for compliance with the protection criteria of this section. The bases for selection of these structures for analysis are: (1) they house or service systems and components required for the safe shutdown of the reactor and to maintain it in a safe shutdown condition, (2) if damaged, they could cause uncontrolled release of radioactivity that could result in potential offsite exposure comparable to the guideline exposures of 10 CFR 100.

Containment excluding enclosure building

Auxiliary building including penetration rooms and MSIV/MFIV area

S5

Control building including penetration rooms and MSIV/MFIV area

Nuclear service cooling towers

Condensate storage tank

S5 Auxiliary feedwater pump building and piping tunnel

Fuel Handling Building

Diesel Generator Building and Fuel Facility

To effect a safe shutdown during a tornado, all essential systems and components are protected with adequate barriers designed against external missiles which may be generated in a tornado as discussed in paragraph 3.5.2.4.1. The control, sensing, power supply and piping associated with safety train oriented systems in areas outside of the plant structures are located below plant grade.

may be considered (on a reasonably conservative basis) in analyzing the accident sequence if it can be shown that unacceptable plant conditions will not ensue or be approached.

C. When analyzing the damage inflicted following the postulated pipe failure or jet impingement, conservative assumptions are to be employed.

D. Pipe whip is the result of the formation of a plastic hinge at the point of highest moment in a piping run that has experienced a circumferential pipe break. It is assumed that a whipping pipe has the potential to move perpendicular to the break orientation unless that pipe has been restrained against motion in that particular direction. A whipping pipe is considered to contain sufficient energy to rupture an impacted pipe of smaller nominal pipe size and lighter wall thickness.

E. In assessing accident effects, the loading conditions of a pipe or branch run in terms of internal pressure and temperature, prior to postulated rupture, should be the normal and upset conditions associated with reactor operation.

F. Class 2 and 3 systems on the Vogtle Plant in which pipe breaks are postulated to occur are supported such that the following mechanistic break is not possible. The mechanistic break assumed is a circumferential break wherein the pipe ends translate laterally with respect to each other up to a distance of 1 diameter. To an adjacent piece of equipment, the mechanistic break could potentially cause a full area blowdown load and concurrently a full area thrust load. For the Vogtle Plant the only relative motion of pipe ends after a guillotine rupture is due to thermal growth.

G. Limited breaks (i.e., single-area breaks) are postulated for those portions of the high-energy main steam and main feedwater fluid piping systems between anchors adjacent to containment isolation valves (including any rigid connection to the containment penetration). The limited break zone shall not exceed 60 feet, piping runs shall be reasonably straight, and the ASME Boiler and Pressure Vessel Code, Section III, safety Class 2, forged anchor shall be provided at the terminal point S5 downstream of the isolation valves. Limited breaks are postulated for only those portions of the fluid piping system extending from the forged anchor, adjacent to the outside isolation valve, to the rigid pipe connection at the containment penetration.

These limited breaks are postulated provided the following design stress limits are met:

S5

1. The maximum stress ranges, as calculated by the sum of equations (9) and (10) in Paragraph NC-3652, ASME Code, Section III, considering normal and upset plant conditions, (i.e., sustained loads, occasional loads, and thermal expansion) and an OBE event, do not exceed 0.8 $(1.2S_h + S_A)$.

   The maximum stress, as calculated by equation (9) in Paragraph NC-3652, under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of piping, does not exceed $1.8S_h$.

2. To prevent a breach of containment, the piping run downstream of the valve outside the containment and upstream of the penetration inside the containment is restrained such that excessive pipe loads are not transmitted to the penetration, penetration piping and containment isolation valve following a postulated pipe failure.

3. For all high energy line penetrations, provisions are made to permit full volumetric in-service inspection of all longitudinal and circumferential welds between the penetration and the isolation valve.

4. When break locations are not postulate per 1. above, longitudinal and circumferential welds on the piping or branch runs are limited to a practical minimum. Transitions between different wall thicknesses are made with a gentle slope to diminish stress discontinuities. Piping restraints are attached to the piping in such a manner as not to cause excessive stress discontinuities.

H. The measures taken for the protection of structures, systems, and components important to safety should not preclude the conduct of inservice examinations of ASME Class 2 and 3 pressure-retaining components as required by the rules of ASME Boiler and Pressure Vessel Code - Section XI, "Inservice Inspection of Nuclear Power Plant Components."

I. In piping systems (moderate energy), the failure for the purpose of assessing accident effects is assumed to be a "critical crack." The critical crack postulated for these evaluations is defined to have an effect opening of an area equivalent to the product of one-half the pipe internal diameter and one-half the pipe wall thickness.

3.6.2.1 Design Basis Piping Break Criteria for Postulated Breaks Inside the Containment

The criteria for postulated breaks inside the containment is defined for ASME Code Class 2 and 3 components as follows (this criteria also applies to Class 1 branch lines):

3.6.2.1.1 Postulated Break Locations

The locations for which breaks should be postulated to occur for ASME Code Class 2 and 3 piping are based on the regions of piping runs or branch runs with the greatest potential for failure under intensities associated with specified seismic events and normal operational plant conditions. On the basis of high stress intensities and related considerations which indicate a greater probability of failure relative to straight pipe, piping break locations should be postulated to occur at the terminal ends of the piping run or branch run. Similarly, breaks should be assumed to occur at any intermediate locations between terminal ends of piping runs or branch runs that exhibit stress intensities above conservatively derived limits based on the stress levels actually existing in the fluid system piping. The limits selected on this basis are elastically calculated primary plus secondary stress intensities per Regulatory Guide 1.46.

C. Longitudinal or circumferential breaks are postulated perpendicular to the direction of the maximum calculated stress rather than having both longitudinal and circumferential breaks at the same location.

3.6.2.1.3 Pipe Whip Restraint Measures

Measures for restraint against pipe whipping as a result of the design basis breaks postulated to occur at the locations specified in paragraph 3.6.2.1.1 need not be provided for piping where any one of the following applies:

A. The piping is physically separated (or isolated) from other piping or components by protective barriers or is restrained from whipping by plant design features such as concrete encasement;

B. Following a single break, the unrestrained pipe movement of either end of the ruptured pipe in any direction about a plastic hinge formed at the nearest pipe whip restraint cannot damage any structure, system, or component important to safety;

C. The energy associated with the whipping pipe can be demonstrated to be insufficient to impair the safety function of any structure, system, or component important to safety to an unacceptable level;

D. Both of the following piping system conditions are met:

1. The design temperature is 200 F or less, and

2. The design pressure is 275 psig or less.

3.6.2.2 <u>Design Basis Piping Break Criteria for Postulated Breaks Outside the Containment</u>

The design basis piping break criteria for breaks outside the containment is as follows:

3.6.2.2.1 High-Energy-Fluid Systems

A. For piping systems that by plant arrangement and layout are isolated by remote location from structures, systems, and components important to safety, pipe breaks need not be postulated provided the requirements of paragraph 3.6.2.G are satisfied.

S5 B. For the main steam and feedwater piping systems that are enclosed in suitably designed concrete structures or compartments to protect structures, systems, and components important to safety, pipe breaks should be postulated at the following locations in each piping or branch run within each protective structure:

S5 1. A nonmechanistic single-area break of the largest line for area pressurization and environmental qualification of equipment.

2. A minimum of one break of the largest branch run within each protective structure or compartment at a location that results in the maximum loading from the impact of the postulated ruptured pipe and jet discharge force on wall, floor, and roof of the structure or compartment, including internal pressurization, and taking into account any piping restraints provided to limit pipe motions. S5

C. Break locations for the main steam piping located in the main steam tunnel shall be postulated only at both terminal points, two intermediate points, and in proximity to the auxiliary feedwater system if all the following conditions are met:

1. Stress criteria in accordance with ANSI B31.1.0 assuming seismic Category 1 loads.

2. Tunnel, supports, and restraints dynamically analyzed for SSE loads. S5

3. Full volumetric inspection of all circumferential and longitudinal welds, either in the shop or in the field, of all main steam piping in proximity to the diesel generator building.

D. A single pipe failure in (seismic Category I or non-seismic Category I) piping system is postulated to occur as the initiating event (accident). Such an event is not concurrent with any other incident, accident, or natural phenomena. In a high energy system (operating pressure and temperature greater

The variation of shear modulus with shear strain for the lower sand stratum is shown in figure 3.7-22C. This is based on the standard curve proposed by Seed and Idriss. The shear modulus corresponding to low shear strain levels (10^{-4} % or less) for this stratum was computed based on the measured shear wave velocity of 1800 ft/sec. To account for the variation in the measured shear wave velocity with the depth, a range of shear moduli with upper-bound values equal to 1.5 times the mean values and lower-bound values equal to the mean values divided by 1.5, will be used in the analysis. S3

As discussed in paragraph 3.7.1.3.1, the damping values for the compacted sand backfill, the clay marl bearing stratum and the lower sand stratum shown in figures 3.7-21, 3.7-22, and 3.7-22A will be used in the analysis.

In general, the soil properties are nonlinear in character. An iterative process is used to obtain equivalent linear properties which are strain dependent. The methods generally used for such an analysis are included in the computer program FLUSH.

The generation of design time history motions is described in section 3.7.1.2. This ground motion is defined for the free field and applied at the finished grade level (El 220'-0") of the site.

The time history at the base of the idealized soil profile is obtained through deconvolution analysis of the design time history specified at finished grade level, using appropriate soil properties. The time history, thus obtained, is applied at the base of the soil-structure interaction system, with appropriate soil properties for soil-structure interaction analysis. The resulting time history response will be used to generate the in-structure response spectra at selected floor elevations. The analysis is performed with consideration S4 given to the variation of soil parameters as indicated above using appropriate cut-off frequencies such that the acceleration profile in the free field is realistic. The "envelope in-structure response spectra" shall be developed by enveloping the response spectra obtained by considering the variation of soil properties.

Response spectra corresponding to the free field time history motions calculated at the elevations of Category I structural foundations are generated. Considering the variation of soil properties, "envelope response spectra" for each Category I foundation level is developed. A scaling factor of 1.5 has been established such that when the "envelope response spectra" curves are multiplied by the scaling factor, the 60% design spectra curves will be essentially enveloped. The "envelope S5 in-structure response spectra" curves generated for each deeply embedded seismic Category I structure will be multiplied by the scaling factor of 1.5 to obtain the "design in-structure response spectra" curves.

10.3 MAIN STEAM SUPPLY SYSTEM

10.3.1 DESCRIPTION

Steam at the outlet from the four steam generators, a total of 15,155,582 lb/hr at 985 psia, 542.8 F, and 0.25 percent moisture is routed in the main steam piping to the turbine at a total pressure drop of 20 psi at the turbine guaranteed steam flow condition. As percent power demand increases, the steam generator outlet steam pressure decreases from 1107 psia, 556 F near zero percent power to 985 psia, 542.8 F at 100 percent power. Consequently, the turbine inlet pressure decreases as the load increases due to this characteristic of the steam generators of the NSSS.

The main steam piping headers conduct the total steam flow from the steam generators to the turbine; one line size is 38 inches, the other is 44 inches as shown in figure 10.1-3 and 10.1-4. Each main steam line is sized to provide a required equal balanced steam pressure at the inlet to the turbine stop valves. The pipe sizes selected are based on different pipe lengths of the unsymmetrical steam piping layout with the present reactor building orientation to the turbine building.

The design pressure-temperature rating of the main steam piping is 1185 psig 600 F, which matches the design pressure-temperature rating of the steam generator secondary feedwater-steam side.

10.3.2 PIPING COMPONENTS AND SYSTEMS

In evaluating the design of the main steam system piping, the following components and/or systems are considered:

A. The attachment of the main steam piping to the steam generators takes into account the allowable nozzle loading movements and stresses as specified by the steam generator manufacturer for all four units operating or with units out of service.

B. The routing, loading, and support of the main steam piping through the penetration section of the reactor containment building for various unit operating conditions are determined.

C. The loading and support of the main steam line isolation valves just outside the containment building are determined.

D. The sequential blowing effect of the steam safety valves and atmospheric power relief valves located in the main steam piping between the containment building and the main steam line isolation valves and the resultant valve reaction thrusts on the piping is determined.

E. The routing and support of interconnected steam piping to the reheaters, the turbine bypass system to the condenser, and the main steam generator turbine driven feedwater pumps for normal, upset, and emergency conditions are determined.

F. The interfacing of the main steam lines with the main turbine stop/control valves is determined, taking into account allowable loadings, movements, and stresses as specified by the turbine-generator manufacturer for normal conditions, on line stop valve testing conditions, and upset conditions.

G. For the foregoing systems and components relating to the main steam piping, the requirements are calculated as outlined by the specific applicable codes and classes as shown on the respective system diagrams.

S5 Steam is conducted from each steam generator in 26-inch lines through the containment building, the lines being anchored at the containment wall. The lines are designed in accord with subsection 3.6.2.G.2, up to and including the anchor forging at the end of the MSIV/MFIV area. The isolation valves and the spring-loaded safety and atmospheric power relief valves are located outside the containment building.

10.3.3 VALVES

10.3.3.1 <u>Safety Valves</u>

Five spring-loaded safety valves in each 36-inch steam outlet line as shown on figure 10.1-4 are designed and selected in accord with the requirements of the ASME Boiler and Pressure Vessel Code, Section III Nuclear Vessels. The lowest safety valve is set at 1175 psig, the design pressure of the steam generator (minus 10 psi piping loss). The highest safety valve setting is 105 percent of the design pressure at 1234 psig.

valve is suitable for remote adjustment of the relieving pressure. Control is automatic based on steam line pressure. Local manual operators are provided in case of complete loss of automatic control.

10.3.3.3 Main Steam Line Isolation Valves

Two automatically operated bi-directional main steam isolation valves will be installed in the steam line from each steam generator outside of the reactor containment. The isolation valve system will prevent the uncontrolled blowdown of more than one steam generator for postulated steam line ruptures in accordance with subsection 3.6.2.G.2. This valve system fulfills the requirements of General Design Criteria 57 of 10 CFR 50. | S5

The following design bases apply:

A. Safety General Design Criteria 57 of Appendix A to 10 CFR 50.

B. Design maximum valve closure time of 10 seconds. | S5

C. Valves will be designed and manufactured to ASME Section III, Class 2.

D. Valves will be designed for seismic Category I.

If the break is within the containment, steam is discharged into the containment. The other steam generators would act to feed steam through the interconnecting piping to reverse the flow into the damaged line if reverse flow protection were not provided to prevent discharge of more than one steam generator.

Closure of the automatic bi-directional valves within 10 seconds from receipt of signal prevents a reverse flow of steam. | S5

If the break is downstream of the isolation valves, the
S5 automatic isolation valves would close within 10 seconds from
receipt of the initiating signal.

For the condition of a steam generator tube rupture, the isolation valves serve to limit the total radioactive release to the environment by isolating the damaged steam generator.

A flow restrictor is installed within each steam generator outlet nozzle and is primarily used to limit the steam flow release in case of a main steam line rupture. Steam flow is measured using the pressure drop between the steam generators and pressure taps in the main steam lines downstream of the flow restrictors.

Three-inch lines connected upstream of the isolation valves (see figure 10.1-4) in the steam outlet line from three steam generators provide steam to the turbine-driven auxiliary feedwater pumps. This assures a source of steam to the turbine-driven auxiliary pump when the steam generators are isolated and are producing steam from reactor decay heat. The 3-inch steam piping to the auxiliary turbine feed pump is designed to the ASME Code Section III Class 2.

Main steam piping is cross-connected downstream of the main
steam line isolation valves. The bypass valve around the main
S5 steam isolation valves will be designed for Nuclear Safety
Class 2 and seismic Category I. Branch piping from the cross-connections provides steam to the reheaters, gland steam sealing system, steam air ejectors, the steam generator feedwater pump turbines, and the turbine bypass steam to the condensers (see figure 10.1-3).

Main steam piping in the main steam tunnel downstream of
the anchor forging at the end of the MSIV/MFIV area is
designed in accord with the Power Piping Code ANSI-B31.1.0.
S5 Anchor forgings shall be provided at both terminal points for
main steam and feedwater piping located in the main steam tunnel.
In addition, pipe whip restraints shall be provided at the two
intermediate high-stress points, as well as in the proximity of
the auxiliary feedwater tunnel.

The specification for the main steam and feedwater line
isolation valves will require each prospective vender to
provide a detailed description of environmental qualification
as well as the provisions for inservice tests. Environmental
S5 qualification will satisfy the environment which results
from the breaks postulated in accordance with subsection
3.6.2.G.2. Inservice inspection tests will be performed
periodically to demonstrate that these valves will function
in accordance with design. The procedure will be available
after a proposal is accepted.

10.3.3.4 Valve Testing

The steam safety valves located in the main steam piping at the outlet from each steam generator are individually tested during initial startup or during the shutdown operation by checking the actual pop and closing pressures against the required design opening and closing pressures listed describing the safety valves.

Isolation valve leakage rates and acceptance criteria will be dependent on the types of valve or valves purchased. These valves will be manufactured in accordance with ASME Sec III Class 2.

The inplant operational testing will be in accordance with the technical specification of section 16.4.7. The inplant leakage rate test and acceptance criteria will be dependent upon the type of valve selected. After selection these tests will be included in the technical section of the PSAR.

These documents will be specified in the valve specifications and more detailed information will be available when a proposal is accepted.

The opening and closing of the atmospheric power relief valves are likewise checked prior to initial startup or during shutdown.

The main steam line isolation valves are located approximately twenty feet from the containment building for the following reasons:

A. The main steam line safety and relief valves are to be placed directly on the main steam line which requires that the isolation valve be located further downstream. The safety and relief valves alone require approximately seventeen feet of line. This arrangement is considered far more practical and safer than placing the valves on a header.

B The safety valves are located outside the enclosure building so that a negative pressure can be maintained in the enclosure building. The enclosure building wall is approximately thirteen feet from containment. Having the safety valves inside the enclosure building would require an additional twenty-four enclosure penetrations per unit to discharge the safety and relief valves to atmosphere. Since umbrella type discharge stacks are to be used as safety valve vents it would be impossible to maintain the negative pressure in the enclosure building.

C. The present arrangement of the safety and isolation valves provides a room to isolate these valves from other facilities. It also optimizes the support arrangements and expansion loops for the lowest practical stress loading. See figure 1.2-5.

10.3.3.5 Pipe Testing

S5 Main steam piping located in the main steam tunnel in the proximity of the diesel generator building will be volumetrically inspected either in the shop or in the field. Tests and/or inspection will include circumferential as well as longitudinal welds.

10.3.4 SYSTEM TESTING

The various alarm and pressure trip points to isolate the steam generator feedwater pumps to prevent overpressurization are checked by comparing design setpoints versus actual measured trip settings. The main steam line is hydrostatically tested to confirm leaktightness. Visual inspection of pipe weld joints confirms the exterior condition of the weld. Pipeline expansion and movement from the cold condition to the hot normal operating condition is checked by measuring movement from field bench marks such as steel columns or pipe supports as specified on design isometric piping drawings indicating calculated movements along the x, y, and z axes.