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1.2-1 General Site Plan

SECTION 1

INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 PROJECT IDENTIFICATION

This Updated Final Safety Analysis Report is submitted pursuant to the requirements of 10CFR50.71 by Public Service Electric and Gas Company (PSE&G) for the two nuclear power units at its Salem Generating Station.

PSE&G and Westinghouse Electric Corporation have jointly participated in the design and construction of each unit. On August 21, 2000, the operating licenses for Salem Units 1 & 2 were transferred from PSE&G to PSEG Nuclear LLC. Each unit employs a pressurized water reactor nuclear steam supply system furnished by Westinghouse which is similar in design concept to several other projects licensed by the Nuclear Regulatory Commission. The only systems shared by the two units are Compressed Air, the Control Room Area intake air radiation monitors, parts of the Control Room Area Ventilation System, bulk Nitrogen Supply, Demineralized Water, and the Solid Radwaste Handling System. There are a minimum of shared components; chemical drain and laundry hot shower tanks and pumps and the 20,000 barrel Bulk Fuel Oil Storage Tank are the only components in common.

The licensed core power for both units is 3459 MWt. The approximate values for gross and net electrical outputs are 1178 MWe and 1135 MWe respectively for Unit 1 and 1182 MWe and 1139 MWe respectively for Unit 2. The reactors are expected to be capable of outputs of approximately 3570 MWt, which corresponds to the valves-wide-open rating of the turbine generators of 1209 MWe gross and 1163 MWe net for Unit 1 and 1220 MWe gross and 1174 MWe net for Unit 2. The containment and engineered safety features for both units have been designed and evaluated at the maximum power rating of 3570 MWt. Loss-of-coolant accidents and those postulated accidents having offsite dose consequences have been analyzed at the power rating of 3570 MWt.

The remainder of Section 1 of this report summarizes the principal design features and safety criteria of the nuclear units, pointing out the similarities and differences with respect to other pressurized water nuclear power plants employing the same technology and basic engineering features as the Salem Generating Station.

Section 2 contains a description and evaluation of the site and environs, supporting the suitability of the site for a nuclear plant of the size and type described. Section 3 discusses the identification, description, and discussion of the principal architectural and engineering design of structures, components, equipment, and systems important to safety. The reactor is described in Section 4. Section 5 discusses the Reactor Coolant System and related systems, and Sections 6 through 11 the emergency and auxiliary systems.

Section 13 describes the Company's program for organization and training of plant personnel. Section 14 contains an outline and description of the initial tests and operations associated with plant startup.

Section 15 is a safety evaluation summarizing the analyses which demonstrate the adequacy of the Reactor Protection System, and the engineered safety features. The consequences of various postulated accidents are within the guidelines set forth in the Nuclear Regulatory Commission's Rules and Regulations and 10CFR50.67, Accident Source Term.

1.2 PLANT SITE SUMMARY

1.2.1 Site Description

The approximately 700 acre Salem site is located along the eastern shore of the Delaware River in Lower Alloways Creek Township, Salem County, New Jersey about 8 miles southwest of Salem, New Jersey. The population density of the area surrounding the site is low. Distance to the site boundary is about 4200 feet. The nearest residence is approximately 3.4 miles west of the site in Bay View Beach, Delaware. Other nearby residences are located 3.5 miles east-northeast and 3.5 miles northwest of the site. The population center distance is 15.5 miles. The area is primarily utilized for agricultural pursuits, with heavy industry located generally 15 miles and beyond to the north of the site.

1.2.2 Meteorology

The meteorological data pertinent to the Salem site has been reviewed, and there is no reason to anticipate unusual meteorological problems. The terrain is open and extremely flat, and the land-sea interaction favors a vigorous wind flow.

A meteorological tower facility was established northwest of the reactor area on the site to provide actual site meteorological data. This data collection program has been terminated as sufficient data has been collected and analyzed to describe the dispersion parameters. The tower has been relocated east of the site.

1.2.3 Geology and Hydrology

An investigation of Salem site geology and hydrology was completed in 1967. The nearest known faulting is approximately 25 miles from the site. Test borings at the site indicate that subsurface conditions are adequate to support the structures. The regional direction of ground water movement is toward the Delaware River,

and all surface drainage at the Salem site flows directly into the river.

1.2.4 Seismology

The site is located in a region which has experienced only infrequent minor earthquake activity. No known faults exist in the basement rock or sedimentary deposits in the immediate vicinity of the site. Significant earthquake motion is not expected at the site during the life of the facility.

The plant was conservatively designed to respond elastically, with no loss of function, to horizontal ground accelerations as high as 10 percent of gravity, and the design was checked for a hypothetical acceleration of 20 percent of gravity.

1.2.5 Marine Ecology

A thorough study of the biological makeup of the Delaware Estuary is being conducted. The study has continued since the plant went into operation to determine the effects (if any) of plant operation on the ecology of the Estuary.

1.2.6 Environmental Radiation Monitoring

An environmental radiation monitoring program for the site and surrounding area is being conducted. This program has continued since the plant went into operation to determine the effects (if any) of plant operation on radiation levels in the environment.

1.2.7 Facility Safety Conclusions

The safety of the public and plant operating personnel and reliability of plant equipment and systems have been the primary considerations in the plant design. The approach taken in fulfilling the safety consideration is three-fold. First, careful attention has been given to design so as to prevent the release of

radioactivity to the environment under conditions which could be hazardous to the health and safety of the public. Second, the plant has been designed so as to provide adequate protection for plant personnel wherever a potential radiation hazard exists. Third, Engineered Safety Features have been designed with redundancy and diversity, and to stringent quality standards.

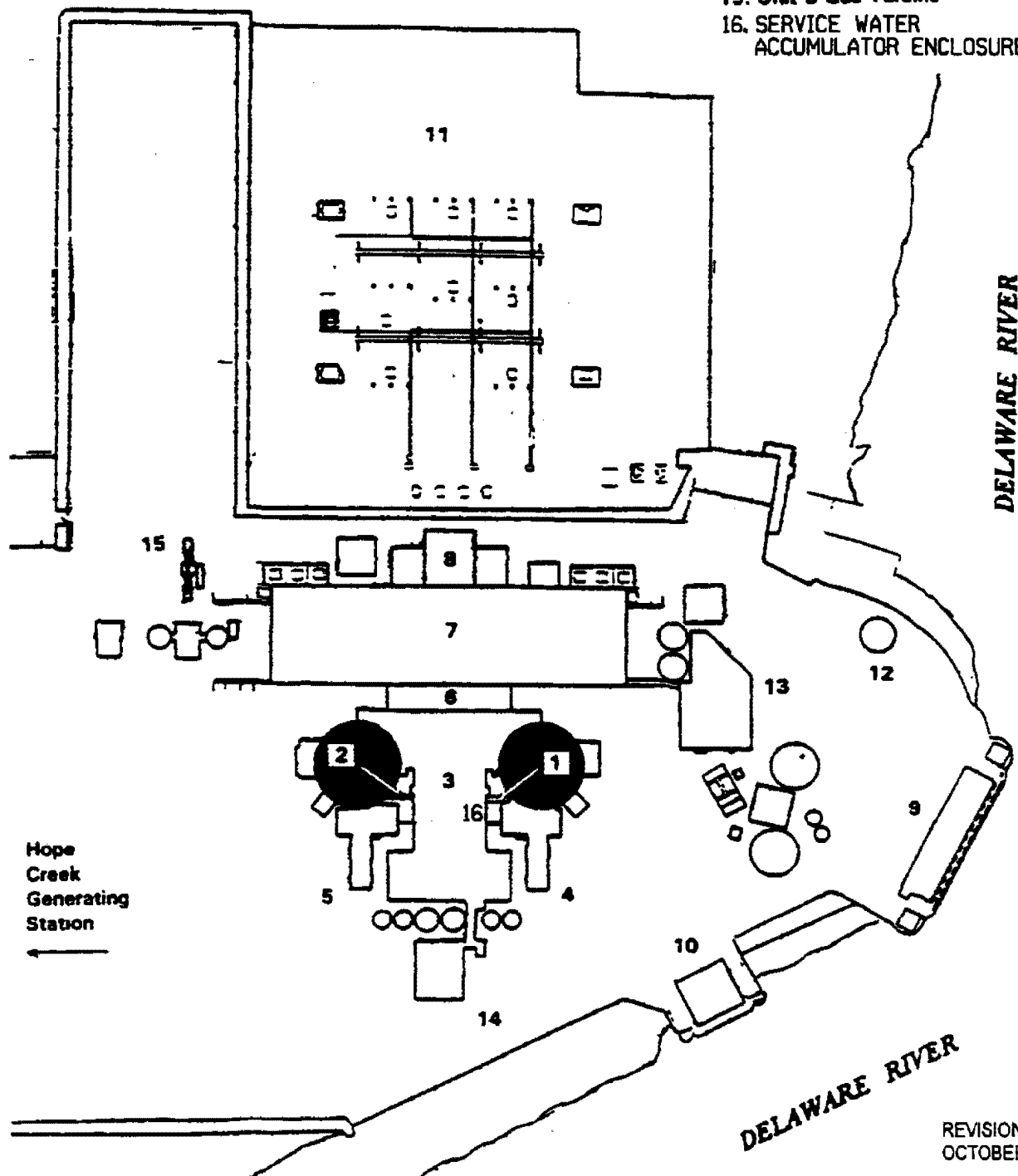
Based on the over-all design of the plant including its safety features and the analyses of possible incidents including the design basis accident, it is concluded that the Salem Generating Station can be operated without undue risk to the health and safety of the public.



LEGEND

1. Unit 1 Containment
2. Unit 2 Containment
3. Auxiliary Building
4. Unit 1 Fuel Handling
5. Unit 2 Fuel Handling
6. Service Building
7. Turbine Generator Area

8. Administration Building
9. Circulating Water Intake
10. Service Water Intake
11. 500 KV Switchyard
12. Fuel Oil Storage Tank
13. Clean Facilities Building
14. Controlled Facilities Building
15. Unit 3 Gas Turbine
16. SERVICE WATER ACCUMULATOR ENCLOSURE



REVISION 17
OCTOBER 16 1998

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

General Site Plan

Figure F1.2-2 intentionally deleted.
Refer to plant drawing 204803 in DCRMS

1.3 SUMMARY PLANT DESCRIPTION

The inherent design of the pressurized water, closed-cycle reactor minimizes the quantities of fission products released to the atmosphere. Four barriers exist between the fission product accumulation and the environment. These are the uranium dioxide fuel matrix, the fuel cladding, the reactor vessel and coolant loops, and the reactor containment. The consequences of a breach of the fuel cladding are greatly reduced by the ability of the uranium dioxide lattice to retain fission products. Escape of fission products through a fuel cladding defect would be contained within the pressure vessel, loops, and auxiliary systems. Breach of these systems or equipment would release the fission products to the reactor containment where they would be retained. The reactor containment is designed to adequately retain these fission products under the most severe accident conditions, as analyzed in Section 15.

Several engineered safety features have been incorporated into the plant design to reduce the consequences of a loss-of-coolant accident (LOCA). These safety features include an Emergency Core Cooling System (ECCS). This system automatically delivers borated water to the reactor vessel for cooling the core under high and low reactor coolant pressure conditions. The ECCS also serves to insert negative reactivity into the core in the form of borated water during plant cooldown following a steam line break or an accidental steam release. Other safety features which have been included in the reactor containment design are a Containment Fan Cooler System which acts to effect depressurization of the containment following a LOCA and to remove particulate matter from the containment atmosphere, and a Containment Spray System which acts to depressurize the containment and remove elemental iodine from the atmosphere by washing action. The Containment Spray System provides redundant backup by an alternate principle for the Containment Fan Cooler System for heat removal.

1.3.1 Structures

The major structures include a separate and independent Containment and Fuel Handling Building for each reactor, a common Auxiliary Building with holdup tank vault, a common Turbine Building and a common Administration and Service Building. General layouts of the Reactor Containment, Auxiliary Building, and interior component arrangements are shown on Figures 1.2-1, 5.1-12 and Plant Drawings 204803, 204804, 204805, 204806, 204807 and 204808.

Seismic Criteria

For Category I (seismic) equipment, dynamic methods or conservative static equivalents were used to determine that components and structures will operate or maintain their integrity, as required. For Category II (seismic) equipment, static methods were used and non-seismic equipment meets applicable codes.

Definition of Seismic Categories

Particular structures and equipment are classified according to seismic design.

The seismic definitions are:

1. Category I (seismic)

Those structures, mechanical components, the Reactor Protection System, and Engineered Safety Features Actuation System whose failure might cause or increase the severity of a LOCA. Also, those structures and components vital to safe shutdown and isolation.

2. Category II (seismic)

Those structures and mechanical components that are not Category I (seismic), but which function in direct support of reactor operation.

1.3.2 Nuclear Steam Supply System

The Nuclear Steam Supply System for each unit consists of a pressurized water reactor, Reactor Coolant System (RCS), and associated auxiliary fluid systems. The RCS is arranged as four closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump and a steam generator. An electrically heated pressurizer is connected to one of the loops.

The reactor core is composed of uranium dioxide pellets enclosed in Zircaloy-based tubing with welded end plugs. The tubes are supported in assemblies by spring clip grid structures. The control rods consist of clusters of stainless steel clad silver-indium-cadmium absorber rods located within the fuel assemblies. The nuclear fuel is typically loaded in three regions, with the new fuel being introduced into the core interior and by its third cycle of operation being discharged from the core's outermost region to spent fuel storage.

The reactor vessel and reactor internals contain and support the fuel and control rods. The reactor vessel is cylindrical with hemispherical heads and is clad with stainless steel.

The pressurizer is a cylindrical pressure vessel with hemispherical heads and is equipped with electrical heaters and spray nozzles for system pressure control.

The steam generators are vertical U-tube units utilizing Inconel tubes. Integral moisture separating equipment reduces the moisture content of the steam at the turbine throttle to ≤ 0.25 percent for Unit 1 and ≤ 0.1 percent for Unit 2.

The reactor coolant pumps are vertical single stage centrifugal pumps equipped with controlled leakage shaft seals.

Auxiliary systems are provided to charge the RCS, add makeup water, purify reactor coolant water, provide chemicals for corrosion inhibition and reactor control, cool system components, remove residual heat when the reactor is shutdown, cool the spent fuel storage pool, sample reactor coolant water, provide for emergency safety injection, and vent and drain the RCS.

1.3.3 Reactor and Plant Control

The reactor is controlled by a coordinated combination of soluble neutron absorbers and mechanical control rods. The control system allows the plant to accept step load changes of 10 percent and ramp load changes of 5 percent per minute over the load range of 15 to 95 percent power under normal operating conditions.

Complete supervision of each reactor and turbine generator is accomplished from each unit's control room.

1.3.4 Waste Disposal System

The Waste Disposal Systems provide all the equipment necessary to collect, process, and prepare for disposal, all radioactive liquid, gaseous, and solid wastes produced as a result of reactor operation.

After collection, liquid wastes are evaporated and/or demineralized if necessary to reduce activity levels. The treated water from the demineralizers or the evaporator distillate may be recycled for use in the plant or may be discharged via the

condenser discharge at concentrations well within the limits set forth in 10CFR20. The evaporator concentrates and spent demineralizer resins are solidified, drummed, and shipped from the site for ultimate disposal in an authorized location.

Gaseous wastes are collected and held up for radioactive decay, after which they may be reused for blanketing tanks. Decayed gases are discharged to the environment in a controlled manner which maintains the offsite dose well below the limits set forth in 10CFR20.

1.3.5 Fuel Handling System

Each reactor is refueled with equipment designed to handle spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for dry storage at the Independent Spent Fuel Storage Installation (ISFSI) or for shipment from the site. Underwater transfer of spent fuel provides an optically transparent radiation shield as well as a reliable source of coolant for removal of decay heat. This system also provides capability for receiving, handling, and storing new fuel. The Spent Fuel Pool Cooling System has been redesigned to include a second permanent spent fuel pool cooling pump.

1.3.6 Turbine and Auxiliaries

The turbine is a four casing, tandem-compound, six flow exhaust, 1800 rpm unit with 44-inch last stage blades. There are six combination moisture separator-steam reheater assemblies. The turbine generators are rated as described in Section 1, with saturated inlet steam conditions of 765 psia, exhausting at 1.5 inches of mercury absolute, at zero percent makeup. There are six stages of feedwater heating.

The turbine is equipped with an Electro-Hydraulic Control System, which uses an electronic controller and a high-pressure fire resistant fluid system to control valve movement.

The condenser is of the single pass deaerating type. There are three strings of feedwater heaters, three one-third size condensate and heater drain pumps and two one-half size feedwater pumps. Drains from the two highest feedwater heaters are pumped into the Condensate System and drains from the four lowest feedwater heaters are cascaded to the condenser.

1.3.7 Electrical System

Each main generator is a 1300 MVA, 25 kV, 3 phase, 60 cycle, 0.9 pf, 1800 rpm, 75 psig hydrogen inner-cooled unit with water cooled stator windings. Field excitation is provided by a direct shaft driven brushless excitation system. Each generator is connected to the primary side of three single phase main stepup transformers through isolated phase buses. The secondary side of each main transformer delivers power to the 500 kV switchyard.

The station service systems consist of a 13.8 kV north ring bus, and 13.8 kV south bus sections, auxiliary and station power transformers, 4160 V, 460 V, 230 V, and 115 V ac and 250 V, 125 V, and 28 V dc buses and equipment. A third 500 kV system tie, the 13.8 kV north ring bus and 13.8 kV south bus sections, arrangement replaces the 69 kV single source described in the Preliminary Safety Analysis Report. This provides a superior power supply system to the station.

Three diesel-generators per unit are provided as onsite sources of power in the event of complete loss of normal ac power. These generators power the post-accident containment cooling equipment as well as the safety injection, centrifugal charging, and residual heat removal pumps to assure an acceptable post loss-of-coolant containment pressure transient and adequate core cooling. Two-out-of-the three diesel-generators can handle the electrical load required for a unit in the event of a LOCA.

1.3.8 Engineered Safety Features

The engineered safety features provided for each unit have sufficient redundancy of components and power sources such that under the conditions of a LOCA they can maintain the integrity of the containment and maintain the exposure of the public below the limits set forth in 10CFR50.67, even when operating with partial effectiveness. The engineered safety features incorporated in the design of each unit and the functions they serve are summarized below.

1. The ECCS injects borated water into the RCS. This system limits damage to the core and limits the energy and fission products released into the containment following a LOCA.

The system has been extensively redesigned by Westinghouse. The basic changes in the redesigned system are the use of two charging pumps from the Chemical and Volume Control System for high head injection in addition to their normal charging function and the relocation of the boron injection tank to the discharge side of these pumps. The design of these pumps was changed from reciprocating to centrifugal. Piping, valving, and instrumentation were also revised as a result of the system redesign.

2. A steel-lined concrete containment vessel consisting of reinforced concrete cylindrical wall, a hemispherical dome, and a reinforced concrete base with testable high integrity penetrations.

3. Reactor containment fan coolers and filters to reduce containment pressure and filter particulate matter following a LOCA.

4. A Containment Spray System to reduce containment pressure and remove iodine from the containment atmosphere.

5. The Containment Isolation System incorporates valves and controls on piping systems penetrating the containment structure. These valves are arranged to provide two barriers between the RCS or containment atmosphere and the environment. System design is such that failure of one valve to close will not prevent isolation, and no manual operation is required for immediate isolation.

Automatic isolation is initiated by a containment isolation signal, derived for Phase A isolation by the safety injection signal and high-high containment pressure signal for Phase B isolation.

6. Power sources for the engineered safety features for each unit are provided by two 4 kV power circuits fed from the 500 kV system through the south 13 kV substation in the 500 kV switchyard.

The 500 kV switchyard arrangement consists of three 500 kV transmission lines connected to a breaker-and-a-half design with four 500-13 kV transformers. Two of them are connected to the 500 kV main bus section 1, the other two are connected to Section 2.

Two 500-13 kV transformers provide power to the south 13kV bus sections (one transformer per section) while the other two transformers feed the north 13kV ring bus. Each south 13kV bus section feeds two 13-4 kV transformers, one for each unit, to provide off-site power for the engineered safety features and new Circulating Water Switchgear. The north 13 kV ring bus is normally operated split to allow one 500-13 kV transformer to feed two (one for each unit) 13-4 kV transformers for Group buses.

Should one out of two 500-13kV transformers feeding the north 13kV ring bus be out of service, the ring bus will be realigned to provide power to all four 13-4kV transformers for both unit group buses from the remaining transformer.

If one out of two 500-13kV transformers feeding the south 13kV bus is out of service, transformers connected to the ring bus will be realigned in such a way that one transformer replaces the lost one while the other provides power to all four 13-4kV transformers for the group buses. During this 500-13kV transformer swap over period, the double ended 4kV vital buses receive power from the second off-site power source.

Reliable diesel-generator power is provided for the engineered safeguards loads in the event of failure of station auxiliary power. In addition, if external auxiliary power to the station is lost concurrent with an accident, power is available for the engineered safeguards from the diesel-generators, which are capable of supplying the engineered safeguards load to assure protection of the health and safety of the public in the event of a LOCA.

7. All components necessary for the proper operation of the engineered safety features are operable from the control room.

1.4 IDENTIFICATION OF CONTRACTORS

The Salem Generating Station was designed and constructed by Public Service Electric & Gas (PSE&G). Westinghouse Electric Corporation designed and furnished the nuclear steam supply equipment and systems including the fuel assemblies.

PSE&G contracted United Engineers and Constructors Inc. of Philadelphia, Pennsylvania, to supervise field erection. PSE&G also engaged several consultants to provide technical assistance in various areas. These consultants are listed below.

<u>Consultant</u>	<u>Program</u>
Southwest Research Institute, San Antonio, Texas	Quality Control
S. M. Stoller Corporation, New York, New York	Reactor Core and Nuclear Fuel Cycle
Smith - Singer Meteorologists, Inc., (Now Meteorological Evaluation Systems, Inc.) Amityville, Long Island, New York	Meteorology
Dames and Moore, Cranford, New Jersey	Geology, Hydrology, Seismology
Pritchard - Carpenter, Consultants, Coral Gables, Florida	Hydrology
Radiation Management Corporation, Philadelphia, Pennsylvania	Radiation Monitoring, Emergency Planning
Ichthyological Associates, Middletown, Delaware	Marine Ecology

Porter-Gertz, Consultants, Inc.,
(Now Porter Consultants)
Ardmore, Pennsylvania

Radiation Monitoring,
Emergency Planning

Framatome Technologies, Inc., (FTI)
of Lynchburg, Va. and Raytheon Corp.

Unit 1 Steam Generator
changeout

During the operational phase, various consultants and contractors have been employed to support station operation. These organizations are selected and perform the applicable service in accordance with the Salem QA Manual.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

One of the design bases for the Salem Generating Station has been to utilize well-developed and proven design concepts, systems, and equipment, in order to minimize the potential for cost and schedule overruns and to enhance the reliability of operation. As a consequence, there have been few requirements for research and development programs to confirm the adequacy of the design. Those programs identified for Salem have been satisfactorily completed, as described in Section 1.5.1. Other programs were identified as valuable to define margins of conservatism or possible design improvements. Relevant programs in this latter category are described in Section 1.5.2

1.5.1 17 x 17 Fuel Assembly

A comprehensive test program for the 17 x 17 assembly has been successfully completed by Westinghouse. Reference 1 contains a summary discussion of the program. The following sections present specific references documenting individual portions of the program.

1.5.1.1 Rod Cluster Control Spider Tests

Rod cluster control spider tests have been completed. For a further discussion of these tests, refer to Section 4.2.3.4.

1.5.1.2 Grid Tests

Verification tests of the structural adequacy of the grid design have been completed. Refer to Section 4.2.3.4 and Reference 2 for a discussion of these tests.

1.5.1.3 Fuel Assembly Structural Tests

Fuel assembly structural tests have been completed. Refer to References 2 and 3 for a discussion of these tests.

1.5.1.4 Guide Tube Tests

Verification tests of the structural adequacy of the guide tubes have been completed. Refer to References 3 and 4 for a discussion of these tests.

1.5.1.5 Prototype Assembly Tests

Verification tests of the integrated fuel assembly and rod cluster control performance have been completed. Refer to References 3 and 4 for a discussion of these tests.

1.5.1.6 Departure from Nucleate Boiling Tests

The test program for experimentally determining the effect of the fuel assembly geometry on the departure from nucleate boiling (DNB) heat flux has been completed. Refer to Reference 5 for a discussion of these tests.

1.5.1.7 Incore Flow Mixing

The experimental test program to determine the effects of the fuel assembly geometry on mixing has been completed. Refer to Reference 6 for a discussion of these tests.

1.5.2 Other Programs

1.5.2.1 Generic Programs of Westinghouse

Reference 7 summarizes ongoing safety-related research and development programs that are being carried out for, or by, or in conjunction with the Westinghouse Nuclear Energy System Division and that are applicable to Westinghouse pressurized water reactors.

1.5.2.2 LOCA Heat Transfer Tests

Experimental test programs to determine the thermal-hydraulic characteristics of 17 x 17 fuel assemblies and to obtain experimental reflooding transfer data under simulated loss-of-coolant accident (LOCA) conditions have been completed. Refer to Reference 8 for a discussion of these tests. A single rod burst test program to quantify the maximum assembly flow blockage which is assumed in the LOCA analyses has been completed. Refer to Reference 9 for a discussion of these tests. The results of these two test programs have been used in the Emergency Core Cooling System analyses in Chapter 15.

1.5.3 References for Section 1.5

1. Eggleston, F. T., "Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries - Spring 1976," June 1976.
2. Gesinski, L. and Chiang, D., "Safety Analysis of the 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident," WCAP-8236 (Proprietary) and WCAP-8288 (Non-Proprietary), December 1973.
3. DeMario, E. E., "Hydraulic Flow Test of the 17 x 17 Fuel Assembly," WCAP-8278 (Proprietary) and WCAP-8279 (Non-Proprietary), February 1974.
4. Cooper, F. W., Jr., "17 x 17 Driveline Component Tests - Phase IB, II, III, D-Loop Drop and Deflection," WCAP-8446 (Proprietary) and WCAP-8449 (Non-Proprietary), December 1974.
5. Hill, K. W., et al., "Effects of 17 x 17 Fuel Assembly Geometry on DNB," WCAP-8296-P-A (Proprietary) and WCAP-8297-A (Non-Proprietary), February 1975.

6. Cadek, F. F.; Motley, F. E.; and Dominicis, D. P., "Effect of Axial Spacing on Interchannel Thermal Mixing with the R Mixing Vane Grid," WCAP-7941-P-A (Proprietary) and WCAP-7959-A (Non-Proprietary), January 1975.
7. Eggleston, F. T., "Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries - Winter 1977 - Summer 1978," WCAP-8768, Revision 2, October 1978.
8. "Westinghouse ECCS Evaluation Model - October 1975 Version," WCAP-8622 (Proprietary) and WCAP-8623 (Non-Proprietary), November 1975.
9. Kuchirka, P. J., "17 x 17 Design Fuel Rod Behavior During Simulated Loss-of-Coolant Accident Conditions," WCAP-8289 (Proprietary) and WCAP-8290 (Non-Proprietary), November 1974.

1.6 LIST OF ACRONYMS

The following is an alphabetical listing of the most frequently used acronyms in this report.

AEC - Atomic Energy Commission

AFST - Auxiliary Feedwater Storage Tank

AFW - Auxiliary Feedwater

AIF - Atomic Industrial Forum

ALARA - As Low as is Reasonably Achievable

ALP - Actuation Logic Processor (AMSAC)

ALS - Actuation Logic System (AMSAC)

AMSAC - Actuation Mitigation System Actuation Circuitry

ANS - American Nuclear Society

ANSI - American National Standards Institute

AO - Axial Offset

ASTM - American Society for Testing and Materials

ATWS - Anticipated Transient Without SCRAM

BIT - Boron Injection Tank

BNWL - Battelle Northwest Laboratory

BOL - Beginning-of-Life

BOP - Balance-of-Plant

BTP - Branch Technical Position

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BWR - Boiling Water Reactor

CAACS - Control Area Air Conditioning System

CAP - Chemical Analysis Panel

CASP - Containment Air Sampling Panel

CCP - Centrifugal Charging Pump

CERC - Coastal Engineering Research Center

CFR - Code of Federal Regulations

CIS - Containment Isolation System

CPS - Condensate Polishing System

CRDM - Control Rod Drive Mechanism

CRS - Control Room Supervisor

CSAS - Containment Spray Actuation System

CSS - Containment Spray System

CVCS - Chemical and Volume Control System

CVTR - Carolina-Virginia Tube Reactor

CWS - Circulating Water System

DAS - Data Acquisition System

DBA - Design Basis Accident

DBE - Design Basis Earthquake

DCRDR - Detailed Control Room Design Review

DEPS - Double-Ended Pump Suction

DF - Decontamination Factor

DNB - Departure from Nucleate Boiling

DNBR - Departure from Nucleate Boiling Ratio

DOT - Department of Transportation

d/p - differential pressure

DRF - Dose Reduction Factor

DTT - Ductility Transition Temperature

DVRPC - Delaware Valley Regional Planning Commission

E&CD - Engineering and Construction Department

EACS - Emergency Air Conditioning System

ECCS - Emergency Core Cooling System

EOL - End-of-Life

EPD - Electric Production Department

EPRI - Electrical Power Research Institute

EPZ - Emergency Planning Area

ESF - Engineered Safety Features

FPS - Fire Protection System

FSAR - Final Safety Analysis Report
 GDC - General Design Criteria
 GM - Geiger-Mueller
 GPM - Gallons Per Minute
 GWD - Gigawatt Day
 GWS - Gaseous Waste System
 HED - Human Engineering Deficiency
 HEPA - High-Efficiency Particulate Air
 HFP - Hot Full Power
 HHW - High-High Water
 HP - High Pressure
 I&C - Instrumentation and Control
 ICE - Instrumentation Controls and Electrical
 IEEE - Institute of Electrical and Electronics Engineers
 I/O - Input/Output
 LCR - License Change Request
 LDP - Lighting Distribution Panel
 LP - Low Pressure
 LPG - Liquefied Petroleum Gas

LPM - Loose Parts Monitoring

SGS-UFSAR

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LNG - Liquified Natural Gas
 LOCA - Loss-of-Coolant Accident
 LOFT - Loss of Fluid Test
 LSP - Liquid Sampling Panel
 LWS - Liquid Waste System
 MCD - Minor Civil Division
 MEL - Master Equipment List (Section 17.2)
 MEL - Moderate Energy Lines
 MI - Mechanical and Integrated
 MIG - Manual Inert Gas
 MMA - Manual Metal Arc
 MMI - Modified Mercalli Intensity
 MOL - Middle-of-Life
 MSIV - Main Steam Isolation Valve
 MSL - Mean Sea Level
 MSR - Moisture Separator-Reheater
 MWD/
 MTU - Megawatt Days per Metric Ton of Uranium
 NBS - National Bureau of Standards
 NBU - Nuclear Business Unit
 NCO - Nuclear Control Operators

NDT - Nil Ductility Transition
 NEMA - National Electric Manufacturers' Association
 NFPA - National Fire Protection Association
 NIS - Nuclear Instrumentation System
 NML - Nuclear Mutual Limited
 NPSH - Net Positive Suction Head
 NOS - Nuclear Oversight Department
 NRB - Nuclear Review Board
 NRC - Nuclear Regulatory Commission
 NSR - Nuclear Safety Review Department
 NSSS - Nuclear Steam Supplier System
 NWS - National Weather Service
 OBE - Operating Basis Earthquake
 OD - Outside Diameter
 O&M - Operations and Maintenance
 ORNL - Oak Ridge National Laboratories
 OS - Operations Superintendent
 OSHA - Occupational and Safety Health Act
 OTG - Operational Test Group
 PASS - Post Accident Sampling System

PLUS - Parcel Land Use System

PMH - Probable Maximum Hurricane

POPS - Pressurizer Overpressure Protection System

PORC - Preoperational Testing Review Committee

PORV - Power-Operated Relief Valve

PRT - Pressurizer Relief Tank

PSAR - Preliminary Safety Analysis Report

PSD - Public Service Datum

PSE&G - Public Service Electric & Gas

PSSUG - Public Service Electric & Gas Startup Group

PVRC - Pressure Vessel Research Committee

PWR - Pressurized Water Reactor

QA - Quality Assurance

QC - Quality Control

RA - Reduction Area

RAMPS - Repair and Maintenance Procedure System

RCC - Rod Cluster Control

RCS - Reactor Coolant System

RCCA - Rod Cluster Control Assembly

RCFC - Reactor Containment Fan Cooler
 RCS - Reactor Coolant System
 RCL - Reactor Coolant Loop
 RCP - Reactor Coolant Pump
 RCPB - Reactor Coolant Pressure Boundary
 RCS - Reactor Coolant System
 REMP - Radiological Environmental Monitoring Program
 REP - Radiation Exposure Permit
 RG - Regulatory Guide
 RHR - Residual Heat Removal
 RMS - Radiation Monitoring System
 RMS - Root-Mean-Square
 RPS - Reactor Protection System
 RSE - Reload Safety Evaluation
 RTD - Resistance Temperature Detector
 RVED - Reactor Vessel Examination Device
 RWST - Refueling Water Storage Tank
 SBO - Station Blackout
 SEC - Safeguards Equipment Control
 SER - Safety Evaluation Report

SGB - Steam Generator Blowdown

SGS - Salem Generating Station

SIS - Safety Injection System

SMSA - Standard Metropolitan Statistical Area

SORC - Station Operations Review Committee

SPDS - Safety Parameter Display System

SRP - Standard Review Plan

SRSS - Square-Root-of-the-Sum-of-the-Square

SSE - Safe Shutdown Earthquake

SSG - Salem Startup Group

SSPS - Solid State Protection System

STA - Shift Technical Advisor

STGD - Steam Turbine - Generator Division

SWRI - Southwest Research Institute

SWS - Service Water System

TDC - Thermal Diffusion Coefficient

TDH - Total Dynamic Head

TDS - Total Dissolved Solid

TLD - Thermoluminescent Dosimeter

TMI - Three Mile Island
 T/MS - Test/Maintenance System (AMSAC)
 TSC - Technical Support Center
 UE&C - United Engineers & Constructors
 UFSAR - Updated Final Safety Analysis Report
 UPS - Uninterruptible Power System
 USE - Upper Shelf Energy
 UTG - United Engineers and Constructors Test Group
 UTS - Ultimate Tensile Stress
 VCT - Volume Control Tank
 WDS - Waste Disposal System
 w.g. - water gage
 WILMAPCO - Wilmington Metropolitan Area Planning Council
 WMID - Wisconsin-Michigan Inspection Device
 WOL - Wedge Opening Loading