North Anna Power Station
Updated Final Safety Analysis Report

Chapter 1

Intentionally Blank

Chapter 1: Introduction and General Description of Plant
Table of Contents

Chapter 1: Introduction and General Description of Plant
Table of Contents (continued)

Chapter 1: Introduction and General Description of Plant

List of Tables

Chapter 1: Introduction and General Description of Plant

List of Figures

CHAPTER 1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

The FSAR was submitted in support of the application of the Virginia Electric and Power Company (VEPCO), Richmond, Virginia, for the Class 103 facility operating licenses, special nuclear materials licenses, by-product materials licenses, and source materials license required for the operation of VEPCO's North Anna Power Station Units 1 and 2. The application was submitted as a combined application for all licenses required for the operation of North Anna Units 1 and 2 as permitted by 10 CFR 50.31.

The North Anna Units 1 and 2 PSAR was filed with the U.S. Atomic Energy Commission (AEC) on March 21, 1969, and Docket Nos. 50-338 (Unit 1) and 50-339 (Unit 2) were assigned. On February 19, 1971, Construction Permits CPPR-77 and CPPR-78 were issued for North Anna Units 1 and 2, respectively.

North Anna Units 1 and 2 are located on a site on the southern shore of Lake Anna in Louisa County, approximately 40 miles north-northwest of Richmond. Lake Anna was created by impounding excess waters of the North Anna River and was developed by VEPCO. Water from Lake Anna is used as a cooling medium for surface condensers and other heat exchanger equipment at the North Anna Power Station.

North Anna Units 1 and 2 each includes a three-coolant-loop pressurized light water reactor nuclear steam supply system and turbine generator furnished by Westinghouse Electric Corporation. The balance of the plant was designed and constructed by VEPCO with the assistance of its agent, Stone & Webster Engineering Corporation.

The nuclear steam supply system is similar in design concept to several such systems licensed by the AEC, including VEPCO's Surry Power Station Units 1 and 2. The containments, which house the major nuclear steam supply system components of each unit, are steel-lined, reinforced-concrete structures that use dry, subatmospheric operation concepts. The containments are similar in design concept to those employed in several projects reviewed by the AEC, including VEPCO's Surry Power Station Units 1 and 2.

Each reactor unit was originally designed for a licensed core power output of 2775 MWt (this corresponds to a nuclear steam supply system rating of 2785 MWt). This core power would result in a gross electrical output of approximately 947 MWe and a net electrical output of approximately 907 MWe with a circulating water temperature of 75°F, and a net electrical output of approximately 898 MWe with a circulating water temperature of 88°F. Each reactor was originally expected to be capable of achieving an ultimate core power level of 2900 MWt (this corresponds to a nuclear steam supply system rating of 2910 MWt). This core power would result in a gross electrical output of approximately 984 MWe and a net electrical output of approximately 944 MWe with a circulating water temperature of 75°F, and a net electrical output of approximately 934 MWe with a circulating water temperature of 88°F. Although the license application was for 2775 MWt (core power), all safety systems, including the containment and engineered safety features, were designed for operation at the expected ultimate power level.

Fuel was loaded in Unit 1 in December 1977, with commercial operation commencing in June 1978. Fuel was loaded in Unit 2 in April 1980, with commercial operation commencing in December 1980. In 1986, both units were uprated to a core power output of 2893 MWt (NSSS Rating of 2905 MWt) with an expected gross electrical output of 982 MWe. In 2010, both units were uprated to a core power output of 2940 MWt (2952 MWt NSSS power rating).

The following sections of this FSAR provide additional information on the design, construction, and operation of North Anna Units 1 and 2. These sections were prepared in accordance with guidelines supplied by the AEC's *Standard Format and Content of Safety Analysis Report for Nuclear Power Plants* issued in February 1972.

Note: As required by the Renewed Operating Licenses for North Anna Units 1 and 2, issued March 20, 2003, various systems, structures, and components discussed within this Updated FSAR are subject to aging management. The programs and activities necessary to manage the aging of these systems, structures, and components are discussed in Chapter 18.

1.2 GENERAL PLANT DESCRIPTION

1.2.1 General

Each unit at North Anna incorporates a closed-cycle pressurized-water nuclear steam supply system, a turbine generator, and the necessary auxiliaries. A radioactive waste disposal system, a fuel handling system, and all auxiliaries, structures, and other onsite facilities required for a complete and operable nuclear power station are also provided.

1.2.2 Structures

The major structures are the reactor containment; auxiliary building; fuel building; turbine building, which includes the main control room. The site arrangement, the plot plan, and the general arrangement of equipment within these structures are shown on the drawings listed in the following tabulation:

Item	Drawing
Site plan	Figure 1.2-1 and Reference Drawing 1
Plot plan	Figure 1.2-2 and Reference Drawing 2
Containment structure and containment auxiliary structures	Reference Drawings 3 through 9
Auxiliary building	Reference Drawings 10 through 16
Fuel building	Reference Drawings 17 and 18
Control area	Figure 1.2-3 and Reference Drawing 19
Service building	Reference Drawings 20 through 22
Turbine building	Reference Drawings 23 through 30
Service water pump house	Reference Drawings 31 and 32
Main circulating pump structure	Reference Drawings 33 and 34
Service water valve house	Reference Drawings 35 and 36
Service water tie-in vault	Reference Drawing 37
Station black-out building	Reference Drawings 38 and 39

Each reactor containment is a steel-lined, reinforced-concrete cylinder with a hemispherical dome and a flat, reinforced-concrete foundation mat. Each containment is designed to withstand the internal pressure accompanying the design basis accident, is leaktight, and provides adequate radiation shielding for both normal operation and design basis accident conditions.

During normal operation, internal pressure is subatmospheric and there is no outleakage of activity from the containment structure. Following the postulated loss-of-coolant accident

(LOCA) described in Section 15.4, the containment peak pressure will be reduced to subatmospheric by the use of redundant spray cooling systems, thereby positively terminating outleakage to the environment.

The general seismic criteria used in the design of the structures and equipment in the station are described in Section 3.7. The operating-basis earthquake results in horizontal ground acceleration of 0.06g for structures on rock and 0.09g for structures on soil. The design-basis earthquake results in horizontal ground accelerations of 0.12g for structures on rock and 0.18g for structures on soil. Damping at these accelerations is generally assumed at 2% and 5% respectively, for major concrete structures. Vertical acceleration is assumed at two-thirds of the horizontal acceleration and is considered to act simultaneously with the horizontal acceleration.

1.2.3 Nuclear Steam Supply System

The nuclear steam supply system consists of a Westinghouse pressurized water reactor and supporting auxiliary systems.

The steam flow of the nuclear steam supply system based on 0% makeup is as follows:

	Unit 1	Unit 2
Thermal output of nuclear steam supply system, MWt	2955	2955
Thermal output of reactor core, MWt	2943	2943
Steam flow from nuclear steam supply system, 10^6 lbm/hr	12.90	12.94
Steam pressure at a steam generator outlet, psia	838	824
Maximum moisture content, %	0.10	0.10
Assumed feedwater temperature at steam generator inlet, °F	437.5	435

The nuclear steam supply system consists of a reactor and closed reactor coolant loops connected in parallel to the reactor vessel, each loop containing a reactor coolant pump and a steam generator. The nuclear steam supply system also contains an electrically heated pressurizer and certain auxiliary systems.

High-pressure water circulates through the reactor core to remove the heat generated by the fission process. The heated water exits from the reactor vessel and passes via the coolant loop piping to the steam generators. Here it gives up its heat to the feedwater to generate steam for the turbine generator. The cycle is completed when the water is pumped back to the reactor vessel. The entire reactor coolant system is composed of leaktight and controlled-leakage components to ensure that the reactor coolant is confined to the system or its auxiliaries.

The core is of the multiregion type. Fuel assemblies within a typical batch are mechanically identical, although the fuel enrichment is typically not the same in all the assemblies. Small differences may also exist between different batches of fuel, as new design features are incorporated into reload fuel assemblies.

In the initial core loading, three fuel enrichments were used. Fuel assemblies with the highest enrichments were placed in the core periphery, or outer region, and the two groups of lower-enrichment fuel assemblies were arranged in a selected pattern in the central region. In subsequent refuelings, approximately one-third of the fuel is discharged and fresh fuel is loaded into the core. The remaining fuel is arranged in the core in such a manner as to achieve optimum power distribution. Details on fuel loading for the first and subsequent cycles appear in Section 4.3.2.1.

Rod cluster control assemblies consisting of cylindrical absorber rods are used for reactor control. The absorber rods move within guide tubes in certain fuel assemblies. The absorber rods are attached to a spider connector to form a rod cluster control assembly (RCCA). The spider of the RCCA is attached to a drive shaft, which is raised and lowered by a drive mechanism mounted on the reactor vessel head. The downward trip of the rod cluster control assemblies is by gravity.

The reactor coolant pumps are Westinghouse vertical, single-stage, centrifugal pumps of the shaft-seal type. The power supply systems for the pumps are designed so that coolant flow adequate to cool the reactor core under all required conditions is maintained.

The steam generators are Westinghouse vertical U-tube units that contain Inconel tubes. Integral moisture separators reduce the moisture content of the steam to 0.25% or less.

The reactor coolant piping and all of the pressure-containing and heat transfer surfaces in contact with reactor coolant are stainless steel or stainless steel clad, or are made of an equivalent corrosion-resistant material. The steam generator tubes and fuel tubes are Inconel and Zircaloy, respectively. The reactor core internals, including the control-rod drive shafts, are primarily stainless steel.

An electrically heated pressurizer connected to one reactor coolant loop maintains reactor coolant system pressure during normal operation, limits pressure variations during load transients, and keeps the system pressure within design limits during abnormal conditions.

Auxiliary system components are provided to charge the reactor coolant system and add makeup water, purify reactor coolant water, provide chemicals for corrosion inhibition and reactivity control, cool system components, remove decay heat when the reactor is shut down, and provide for emergency safety injection.

1.2.4 Control and Instrumentation

The reactor is controlled by rod cluster control motion, which is required for load-follow transients and for start-up and shutdown; and by a soluble neutron absorber, boron in the form of boric acid, which is inserted during cold shutdown, partially removed at start-up, and adjusted in concentration during core lifetime to compensate for such effects as fuel consumption and the accumulation of fission products that tend to slow the nuclear chain reaction.

The control system allows the plant to accept step-load increases of 10% and ramp-load increases of 5% per minute over the load range of 15% to 100% of full power. Equal step- and ramp-load reductions are possible over the range of 100% to 15% of full power.

The supervision of both the nuclear and turbine-generator plants is accomplished from the main control room.

1.2.5 Waste Disposal Systems

The waste disposal systems provide all equipment necessary to collect, process, and prepare for disposal all radioactive liquid, gaseous, and solid wastes produced as a result of station operation. The waste disposal systems are capable of handling the wastes produced by both units.

Liquid wastes are collected and processed through the Ion Exchange Filtration System (IEFS) and/or the demineralizers in the waste disposal building. Continuous radiation monitoring is provided for treated liquid waste before its release to the circulating water discharge tunnel. Liquid waste is analyzed and monitored to ensure that discharge concentrations are maintained as low as practicable and well within the limits of applicable regulations.

Spent resins are placed into approved containers, dewatered, and shipped from the site for ultimate disposal at an authorized location.

Gaseous wastes are diluted, filtered, and discharged to the environment with a yearly average activity level as low as practicable.

1.2.6 Fuel Handling System

The reactor is refueled with equipment that handles spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site. The underwater transfer of spent fuel provides an economic and transparent radiation shield, as well as a reliable coolant for the removal of decay heat.

The fuel handling system is divided into two pool regions: the refueling cavity, which is flooded for refueling; and the spent-fuel pit, which is external to the reactor containment and is always accessible to plant personnel. The two pools are connected by the fuel transfer system, which transports the fuel from the refueling cavity to the transfer canal.

Spent fuel is removed from the reactor vessel by a manipulator crane and placed in the fuel transfer system. In the spent-fuel pit, the fuel is removed from the transfer system and placed into storage racks. After a suitable decay period, spent fuel can be moved from storage in the spent fuel pool and loaded into casks for storage at the North Anna Independent Spent Fuel Storage Installation (ISFSI), or loaded into casks for shipment off the site.

1.2.7 Turbines and Auxiliaries

Each turbine is a tandem-compound, 3-element, 1800-rpm unit having 57-inch last-stage exhaust blading in the low-pressure elements. Four combination single-stage moisture separators-reheaters are employed to dry and superheat the steam between the high- and low-pressure turbine cylinders for each unit. A single-pass, deaerating, surface condenser installed in two sections, two 100%-capacity steam jet air ejectors, three 50%-capacity condensate pumps, three 50% steam generator feedwater pumps, two motor-driven and one turbine-driven auxiliary steam generator feedwater pumps, and six stages of feedwater heating are provided.

1.2.8 Electrical Systems

The main generator for each unit is an 1800-rpm, 22-kV, 3-phase, 60-Hz, hydrogen inner-cooled unit. Each main step-up transformer delivers power to the high-voltage switchyard.

The station service system for each unit consists of auxiliary transformers, 4160V and 480V switchgear and buses, 480V motor control centers, 120V ac vital buses, and 125V dc batteries and equipment. Non-safety-related buses are powered from the main generator via three station service transformers, while the emergency buses are powered from the switchyard via three reserve station service transformers.

Emergency power is supplied from two separate and similar emergency diesel-driven generators per unit. Each diesel-driven generator is capable of supplying necessary power for the postaccident containment depressurization subsystems, as well as charging pumps and low-head safety injection pumps, to ensure the operation of minimum safeguards for the design basis accident.

1.2.9 Engineered Safety Features

The engineered safety features provided for each unit have sufficient redundancy and independence of components and power sources that, under the conditions of the assumed design basis accident, the systems can, even when operating with partial effectiveness, maintain the integrity of the containment and reduce the exposure of the public well below the criteria in 10 CFR 50.67.

Each unit is independent in terms of its engineered safety features. The systems to be provided for each unit are summarized below:

1. The steel-lined concrete containment structure provides a reliable barrier against the uncontrolled escape of fission products due to accidents, and permits subatmospheric operation by limiting air inleakage. The structure and all penetrations, including access openings and ventilation ducts, are of proven design.

2. The safety injection system injects borated water into the reactor coolant loops to cool the core by the operation of on-line accumulators, and by high- or low-head pumps subsequent to LOCAs.
3. The containment quench and recirculation spray systems can provide a spray of cool, basic, borated water to the containment atmosphere.

   Following the assumed design basis accident, the containment pressure would be reduced rapidly to subatmospheric pressure by these systems, thereby positively terminating leakage to the atmosphere. The subsequent long-term maintenance of subatmospheric conditions would be accomplished by the recirculation spray and containment atmosphere cleanup system.

1.2.10 Auxiliary Systems

Auxiliary systems are provided as described below:

1. The component cooling system, an intermediate cooling system common to both units, transfers heat from heat exchangers containing reactor coolant or other radioactive or potentially radioactive liquids and gases to the service water system. The maximum heat load occurs during the initial stages of residual heat removal during reactor cooldown. The component cooling system and the residual heat removal system are designed to reduce the temperature of the reactor coolant to approximately 140°F within 20 hours after a reactor shutdown.
2. The service water system removes heat from the component cooling system during the normal operation or cooldown of two reactor units and from the recirculation spray subsystem during a LOCA. This heat is transferred to the environment via the service water reservoir or the Waste Heat Treatment Facility. The service water system also provides cooling to the miscellaneous components requiring an assured supply of cooling water during a loss-of-coolant or loss-of-station-power accident.
3. The boron recovery system is a common system serving both units. This system degasifies and stores borated radioactive water from the reactor coolant system letdown by the Chemical and Volume Control System. The system processes this letdown by evaporation, filtration, and demineralization to produce primary-grade water and concentrated boric acid solution for plant reuse or disposal. Stripped gases are sent to the gaseous waste disposal system.
4. The sampling system transmits representative liquid and gaseous samples to the sampling sinks for laboratory analysis.
5. The vent and drain system collects potentially radioactive fluids and gases from various systems and sends these fluids and gases to the boron recovery system or the appropriate waste disposal system.

6. The fuel pit cooling and refueling purification system removes the residual heat from spent fuel stored in the spent-fuel pit and purifies the water in the reactor cavity and spent-fuel pit.
7. Ventilation systems are provided for the containment and other structures. The containment ventilation system recirculates and cools the containment atmosphere. A purge system, which includes charcoal filters, is provided for use during periods of shutdown. The ventilation and air conditioning system serving the main control room is designed to provide uninterrupted service during all conditions, including accident conditions. The ventilation air areas of possible radioactive contamination are discharged through a monitored ventilation vent and can be routed through filters.
8. The fire protection system furnishes water and other extinguishing agents with the capability of extinguishing any single or probable combination of simultaneous fires that might occur at the station. The system consists of a water system, low- and high-pressure carbon dioxide systems, a Halon 1301 system, and a foam system.
9. The circulating water system provides water for cooling the main condensers and can provide water to the bearing cooling water system. The water is pumped from the North Anna Reservoir created by damming the North Anna River. The water discharges to the Waste Heat Treatment Facility.
10. The compressed air system supplies station service and instrument air. Dryers are provided for the instrument air compressors. Separate, redundant supplies of instrument air are provided in each of the containment structures.

Other auxiliary systems include the domestic water system, communications system, primary plant gas supply system, and auxiliary steam system.

1.2.11 Common and Separate Facilities

Separate and similar systems and equipment are provided for each unit except as noted below. Only those components that are shared by the two units are included in this list.

Electrical systems (Chapter 8**)**

Standby station service transformer facility

Chemical and volume control system (Section 9.3.4)

Chemical mixing tank

Boric acid storage tanks (3)

Boric acid pumps (4)

Boric acid batching tank

Resin fill tank

Boron recovery system (Section 9.3.5)

Component cooling water system (Section 9.2.2)

Component cooling surge tank

Component cooling water pumps (4)

Component cooling water heat exchangers (4)

Chilled water mechanical chiller

Mechanical chilled water circulating pumps (3)

Containment Atmosphere Cleanup System (Section 6.2.5)

Hydrogen recombiners (2)

Hydrogen analyzers (2)

Purge blowers (2)

Fuel pit cooling and refueling purification systems (Section 9.1.3)

Spent-fuel pit cooling pumps (2)

Fuel pit coolers (2)

Fuel pit skimmer assemblies (2)

Refueling purification pumps (3)

Refueling purification ion exchanger (1)

Refueling purification filters (2)

Sampling system (Section 9.3.2)

Sample coolers for boron recovery system samplers (2)

Sample coolers for auxiliary boiler systems (3)

Vent and drain system (Section 9.3.3)

Auxiliary building sump pumps (2)

Fuel building sump pumps (2)

Service water system (Section 9.2.1)

Fire protection system (Section 9.5.1)

Water supply and treatment system (Section 9.2.3)

Ventilation system (Section 9.4), other than containment ventilation

Primary plant gas supply system (Section 9.5.10)

Auxiliary steam system (Section 9.4.1)

Lubricating-oil system (Section 10.4.5)

Clean and dirty lube-oil storage tanks (2)

Transfer pump

Portable centrifuge

Radioactive waste systems (Chapter 11)

Structures, buildings, and miscellaneous equipment

Auxiliary building

Fuel building

Turbine building and turbine room crane

Service building

Main control area

Decontamination facility

Office building

General station services (nonelectrical)

Fuel-oil system

Service water pump house

Waste disposal building

Boron recovery tank building

Waste gas decay tank vault

Station black-out building

1.2.12 Special Designs

There are no significant extrapolations in the technology or solutions to particularly difficult engineering problems involved in the design and construction of the station.

The reservoir formed by damming the North Anna River may be considered an unusual site characteristic. This reservoir serves as the cooling water supply for the station. A portion of the lake, called the Waste Heat Treatment Facility, dissipates waste solution heat from the circulating water discharge before the return of this water to the main body of the lake, the North Anna Reservoir. The North Anna Reservoir and the Service Water Reservoir form the ultimate heat sink for the station.

1.2 REFERENCE DRAWINGS

The list of Station Drawings below is provided for information only. The referenced drawings are not part of the UFSAR. This is not intended to be a complete listing of all Station Drawings referenced from this section of the UFSAR. The contents of Station Drawings are controlled by station procedure.

	Drawing Number	Description
1.	11715-FY-1B	Site Plan, Units 1 & 2
2.	11715-FY-1A	Plot Plan, Units 1 & 2
3.	11715-FM-1A	Machine Location: Reactor Containment, Plan, Elevation 291'- 10", Unit 1
4.	11715-FM-1B	Machine Location: Reactor Containment, Plan, Elevation 262'- 10", Unit 1
5.	11715-FM-1C	Machine Location: Reactor Containment, Plan, Elevation 241'- 0", Unit 1
6.	11715-FM-1D	Machine Location: Reactor Containment, Plan, Elevation 216'- 11", Unit 1
7.	11715-FM-1E	Machine Location: Reactor Containment, Sections 1-1 & 5-5, Unit 1
8.	11715-FM-1F	Machine Location: Reactor Containment; Sections 2-2, 6-6, 7-7, & 10-10; Unit 1
9.	11715-FM-1G	Machine Location: Reactor Containment, Sections 3-3 & 4-4, Unit 1
10.	11715-FM-2A	Arrangement: Auxiliary Building, Plan, Elevation 244'- 6"
11.	11715-FM-2B	Arrangement: Auxiliary Building, Plan, Elevation 259'- 6"
12.	11715-FM-2C	Arrangement: Auxiliary Building, Plan, Elevation 274'- 0"
13.	11715-FM-2D	Arrangement: Auxiliary Building, Plan, Elevation 291'- 10"
14.	11715-FM-2E	Arrangement: Auxiliary Building, Sections 1-1 & 2-2
15.	11715-FM-2F	Arrangement: Auxiliary Building; Sections 3-3, 4-4, & 5-5
16.	11715-FM-2G	Arrangement: Auxiliary Building, Sections 6-6 & 7-7
17.	11715-FM-3A	Arrangement: Fuel Building, Sheet 1
18.	11715-FM-3B	Arrangement: Fuel Building, Sheet 2
19.	11715-FE-27B	Arrangement: Main Control Room, Elevation 276'- 9", Units 1 & 2
20.	11715-FM-5A	Arrangement: Service Building, Sheet 1
21.	11715-FM-5B	Arrangement: Service Building, Sheet 2

22.	11715-FM-5C	Arrangement: Service Building, Sheet 3
23.	11715-FM-4A	Machine Location: Turbine Area, Plan, Operating Level
24.	11715-FM-4B	Machine Location: Turbine Area, Plan, Mezzanine Level
25.	11715-FM-4C	Machine Location: Turbine Area, Plan, Ground Floor
26.	11715-FM-4D	Machine Location: Turbine Area, Sections, Sheet 1
27.	11715-FM-4E	Machine Location: Turbine Area, Sections, Sheet 2
28.	11715-FM-4F	Machine Location: Turbine Area, Sections, Sheet 3
29.	11715-FM-4G	Machine Location: Turbine Area, Sections, Sheet 4
30.	11715-FM-4H	Machine Location: Turbine Area, Sections, Sheet 5
31.	11715-FM-8A	Arrangement: Service Water Pump House, Sheet 1
32.	11715-FM-8B	Arrangement: Service Water Pump House, Sheet 2
33.	11715-FM-6A	Arrangement: Intake Structure, Sheet 1
34.	11715-FM-6B	Arrangement: Intake Structure, Sheet 2
35.	11715-FP-5AM	Service Water Valve House Piping, Plan and Sections, Units 1 & 2
36.	11715-FP-5AN	Service Water Valve House Piping, Plan and Sections, Units 1 & 2
37.	11715-FP-5AK	Service Water, Buried Piping Tie-In, Units 1 & 2
38.	11715-FM-11D	Arrangement: Station Black Out Building, Plan, Units 1 & 2
39.	11715-FM-11E	Arrangement: Station Black Out Building, Sections, Units 1 & 2

Withheld under 10 CFR 2.390 (d) (1)

[redacted]

Withheld under 10 CFR 2.390 (d) (1)

[redacted]

Withheld under 10 CFR 2.390 (d) (1)

[redacted]

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

1.3 COMPARISON TABLES

1.3.1 Comparisons with Similar Facility Designs

The comparison tables which follow reflect the designs at the time of the original FSAR submittal, unless otherwise noted.

1.3.1.1 Comparison of Nuclear Steam Supply Systems

Table 1.3-1 presents a comparison of the design of the nuclear steam supply system for the North Anna Power Station with those for the Surry Power Station and the Beaver Valley Nuclear Station Unit 1.

1.3.1.2 Comparison of Engineered Safety Features

Table 1.3-2 presents a summary of the design and operational data on the engineered safety features for North Anna Units 1 and 2 together with comparable data derived from the FSARs for Surry Units 1 and 2 and Beaver Valley Unit 1. The Surry Units 1 and 2 (Docket Nos. 50-280 and 50-281) and Beaver Valley Unit 1 (Docket Nos. 50-334) FSARs were selected because these units are closely related technically to the North Anna units and serve as examples of Stone & Webster facilities that received operating licenses before the North Anna units.

The North Anna Power Station units are also generally comparable with the PWRs of the Pacific Gas and Electric Company and the Rochester Gas and Electric Corporation, and with the Sequoyah units.

1.3.1.3 Comparison of Containment Concepts

Table 1.3-3 is a summary of the design and operational data on the subatmospheric containment for North Anna Units 1 and 2, together with comparable data derived from the FSARs for Surry Units 1 and 2 and Beaver Valley Unit 1. These references were selected because all of these units use the Stone & Webster subatmospheric containment design. Surry Units 1 and 2 received operating licenses before the North Anna units.

1.3.1.4 Comparison of Instrumentation and Electrical Systems

1.3.1.4.1 Comparison of Instrumentation and Control Systems

Table 1.3-4 provides a comparison of containment pressure data.

Table 1.3-5 provides a comparison of data on reactor coolant pump protection.

Table 1.3-6 provides a comparison of engineered safety features actuation signals.

Table 1.3-7 provides a comparison of the emergency diesel generator and steam generator auxiliary feedwater pump start signals.

Tables 1.3-8 through 1.3-10 provide a comparison of radiation monitoring sampling locations.

Data for North Anna Units 1 and 2 are compared with similar data for Surry Units 1 and 2 and Beaver Valley Unit 1. The latter units were chosen because they are similar to North Anna Units 1 and 2 and because each had undergone an AEC licensing review.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

1.3.1.4.2 Comparison of Electrical Systems

Table 1.3-11 provides a comparison of electrical parameters for North Anna Units 1 and 2 with similar data from Surry Power Station Units 1 and 2, and Maine Yankee Power Station. These units were chosen for the comparison because they have electrical systems similar in design to North Anna Units 1 and 2.

In addition, Surry and Maine Yankee were selected because they received operating licenses before the North Anna units.

1.3.1.5 **Comparison of Waste Systems**

Table 1.3-12 presents a comparison of radioactive waste disposal equipment used at North Anna Power Station Units 1 and 2 and that used at Surry Power Station Units 1 and 2 and Beaver Valley Nuclear Station Unit 1. Where decontamination factors are given, they are for all radionuclides unless specifically stated otherwise.

Surry Units 1 and 2 (Docket Nos. 50-280 and 50-281) and Beaver Valley Unit 1 (Docket No. 50-334) were selected for this comparison because these units are technically similar to those of the North Anna Power Station. The Surry and Beaver Valley units are examples of Stone & Webster facilities that had received operating licenses before the North Anna units.

1.3.1.6 **Comparison of Other Nuclear Plant Systems**

Table 1.3-13 presents a summary of the major design data on various nuclear plant systems for North Anna Units 1 and 2, Surry Units 1 and 2 (Docket Nos. 50-280 and 50-281), and Beaver Valley Unit 1 (Docket No. 50-334).

The Surry and Beaver Valley units were chosen for comparison because they are closely related technically to the North Anna units and had received operating licenses before the North Anna units.

1.3.2 Comparison of Final and Preliminary Designs

The significant changes that were made in station design between the submittal of the PSAR and the submittal of the original FSAR are listed below:

1. Technological advances permitted increasing the ultimate core rating to 2900 MWt with a minimum of changes in plant design. With this increase in ultimate core rating, the guaranteed core rating was increased to 2775 MWt. In 1986, both units were uprated to a core power of 2893 MWt. This power corresponds to the maximum calculated turbine rating, defined in Section 15.1.
2. All systems and components were designed and evaluated at the increased design power level of 2900 MWt. To accommodate the increase in design power level to 2900 MWt, the following changes were made to components and systems:

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

a. The containment (Section 3.8.2) was increased in height by 5 feet to ensure that the peak pressure reached during a postulated LOCA at the increased design power level would be within acceptable limits.

b. The refueling water storage tank (Section 6.2.2) was increased in size from the usable volume of 350,000 gallons to 450,000 gallons to provide the additional heat removal capacity required by the increase in energy release during a postulated LOCA.

c. Various components in the steam and power conversion system (Chapter 10) were increased in capacity to accommodate the increased design steam flow.

d. The reactor coolant pump (Section 5.5) motor horsepower was increased from 6000 to 7000 to accommodate the design reactor coolant flow, which was increased from 100.7×10^6 to 105.1×10^6 lb/hr commensurate with the increase in power level.

e. The centrifugal charging pump motor horsepower was changed from 600 to 900 in order to provide a greater flow during the injection flow operation of the emergency core cooling system.

3. The quality assurance program during design and construction was modified to conform to the requirements of 10 CFR 50, Appendix B (Chapter 17).
4. The service water reservoir (Section 3.8.4) was increased in size to provide a 30-day supply of water to conform to Regulatory Guide 1.27.
5. A containment atmosphere cleanup system (Section 6.2.5) was added to conform to Regulatory Guide 1.7.
6. The design of nuclear piping systems was changed to generally meet the requirements of ANSI B31.7, the Code for Nuclear Power Piping (see Section 3.2.2), to ensure that the quality of these piping systems is in accordance with 10 CFR 50.55a (PSAR Section 3.2 and Chapters 5, 6, 9, and 11).
7. A vent condenser was added to the low-capacity steam generator blowdown system tank to further reduce the potential release of radioactive effluents to the atmosphere.
8. The emergency core cooling system was revised to meet the AEC’s interim criteria. The injection paths were altered to provide cold-leg injection with cold- or hot-leg recirculation. This change was made in order to counteract potential steam binding above the core and the subsequent bypass of safety injection flow around the upper plenum and out through a broken hot leg (see Section 6.3).
9. The online testing capability for final actuator devices was added to conform to Regulatory Guide 1.22 (PSAR Chapter 7).
10. The normal source of water for the service water system (Section 9.2.1) was changed from the North Anna Reservoir (Lake Anna) to the service water reservoir so that no transfer from one source of cooling water to the other is required to meet the single-failure criterion.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

11. Reactor protection systems and engineered safety features actuation circuits were changed from a relay to a solid-state logic system. This change was made to provide a simpler and faster method of performing online testing; to reduce the amount of fuel wiring required; to provide a more compact system that requires less space and is easier to maintain; to achieve a standardized and more flexible system; and to provide a system requiring less power to operate and less (electrical) heat removal (see Sections 7.2 and 7.3).
12. A safety injection permissive block was replaced with a reactor trip permissive block in the safety injection reset logic. This change was made in order to prevent a safety injection signal, if one occurred several hours after an accident, from realigning the emergency core cooling system from recirculation to injection (Section 7.3).
13. The B_4C control rods were changed to Ag-In-Cd control rods. This change was made to minimize tritium releases from the control rods to the primary coolant (see Section 4.2.3).
14. Pellet density and fuel-rod pressure were changed to reflect the evolution of the design as core performance and safety requirements were met. The initial backfill pressure of the helium in the fuel was increased to offset densification effects. The pellet densities were changed from a variation by fuel-rod region to a constant density for all regions. This change was made because operating experience has shown that fuel pellet swelling is not a strong function of burnup, as previously believed, so that a uniform core pellet density can be employed (see Section 4.3).
15. The burnable poison loading pattern was changed to reflect more detailed design calculations (see Section 4.3).
16. The reactor vessel top and bottom head penetration and the control-rod drive mechanism were redesigned to meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code (see PSAR Section 5.4).
17. Removable insulation on the closure and lower reactor vessel heads was added to provide access to those areas for inspection purposes (see Section 5.4).
18. The rod withdrawal stop from the rod drop signal and the automatic turbine load cutback initiated by rod drop were replaced by the power range neutron flux rate trips. The positive neutron flux rate trip ensures that the criteria appropriate for an ANS Condition IV event are met even for rod ejections from partial power. The negative neutron flux rate trip will ensure that the departure from nucleate boiling ration (DNBR) remains above 1.30 for all multiple rod drop accidents (see Section 7.2).
19. The recirculation spray casing cooling subsystem was added to increase the available net positive suction head for the outside recirculation spray pumps. The available net positive suction head for the inside recirculation spray pumps was increased by diverting water from the quench spray system header to the inside recirculation spray pump suction (see Section 6.2).

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-1

DESIGN OF NORTH ANNA NUCLEAR STEAM SUPPLY SYSTEM: COMPARISON WITH SYSTEMS AT BEAVER VALLEY UNIT 1 AND SURRY UNITS 1 AND 2

Chapter	Chapter Title, System/Component	Section[a]	Significant Similarities	Significant Differences
4.0	Reactor			
	Fuel	Section 4.2.1	Similar to Beaver Valley Unit 1.	North Anna units have 17 x 17 fuel assemblies. Surry units have 15 x 15 fuel assemblies.
	Reactor vessel internals	Section 4.2.2	Similar to Beaver Valley Unit 1 and Surry Units 1 and 2.	None.
	Reactivity control systems	Section 4.2.3	Similar to Beaver Valley Unit 1 and Surry Units 1 and 2.	None.
	Nuclear design	Section 4.3	Similar to Beaver Valley Unit 1 and Surry Units 1 and 2.	None.
	Thermal-hydraulic design	Section 4.4	Similar to Beaver Valley Unit 1 and Surry Units 1 and 2.	Surry has a core thermal output of 2441 MW. Small variations in thermal-hydraulic and heat transfer parameters.
5.0	Reactor coolant system	Sections 5.1, 5.2		
	Reactor vessel	Section 5.4	Similar to Beaver Valley Unit 1 and Surry Units 1 and 2.	No significant differences.
	Reactor coolant pumps	Section 5.5.1	Similar to Beaver Valley Unit 1 and Surry Units 1 and 2.	The pump motors for North Anna Units 1 and 2 are 7000 hp; those for Surry and Beaver Valley Unit 1 are 6000 hp.

a. Final safety analysis report for North Anna Units 1 and 2.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-1 (continued)

DESIGN OF NORTH ANNA NUCLEAR STEAM SUPPLY SYSTEM: COMPARISON WITH SYSTEMS AT BEAVER VALLEY UNIT 1 AND SURRY UNITS 1 AND 2

Chapter	Chapter Title, System/Component	Section[a]	Significant Similarities	Significant Differences
5.0 (continued)			The Beaver Valley and North Anna units have pump casings that employ three integrally cast feet and are supported by box-frame structures.	Surry Units 1 and 2 pumps are supported by hangers to four feet welded to each pump casing.
	Steam generators	Section 5.5.2	Similar to Beaver Valley Unit 1 and Surry Units 1 and 2.	No significant differences.
	Piping	Section 5.5.3	Similar to Beaver Valley Unit 1. The piping and fittings are cast; and the 90-degree elbows are cast with longitudinal electroslag welds.	Surry Units 1 and 2 employ wrought seamless pipe. The cast 90-degree elbows have longitudinal submerged arc welds.
	Residual heat removal system	Section 5.5.4	Similar to Beaver Valley Unit 1 and Surry Units 1 and 2.	No significant differences.
	Pressurizer	Section 5.5.5	Similar to Beaver Valley Unit 1 and Surry Units 1 and 2.	The head material for North Anna is fabricated plate; that for Beaver Valley and Surry is cast. The surge safety and relief spray for North Anna are forgings; those for Beaver Valley and Surry are cast.

a. Final safety analysis report for North Anna Units 1 and 2.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-1 (continued)

DESIGN OF NORTH ANNA NUCLEAR STEAM SUPPLY SYSTEM: COMPARISON WITH SYSTEMS AT BEAVER VALLEY UNIT 1 AND SURRY UNITS 1 AND 2

Chapter	Chapter Title, System/Component	Section[a]	Significant Similarities	Significant Differences
5.0 (continued)	Loop stop valves	Section 5.5.7	Similar to Beaver Valley Unit 1 and Surry Units 1 and 2. Westinghouse manufactures the valves for Beaver Valley Unit 1 and the North Anna units.	Darling Valve manufactured the valves for Surry Units 1 and 2.
6.0	Engineered safety features			
	Emergency core cooling system	Section 6.3	Similar to Beaver Valley and Surry.	North Anna's low-head safety injection pumps and high-head safety injection/charging pumps have a higher maximum flow rate than those at Beaver Valley and Surry. Surry locks power off of the accumulator isolation valve; North Anna and Beaver Valley provide "S" signal.
7.0	Instrumentation and controls			
	Reactor trip system	Section 7.2	Similar to Beaver Valley and Surry.	Surry has relay logic; North Anna and Beaver Valley have solid state.
	Engineered safety features systems	Section 7.3	North Anna, Surry, and Beaver Valley all have extended engineered safety features testability.	No significant differences.

a. Final safety analysis report for North Anna Units 1 and 2.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-1 (continued)

DESIGN OF NORTH ANNA NUCLEAR STEAM SUPPLY SYSTEM: COMPARISON WITH SYSTEMS AT BEAVER VALLEY UNIT 1 AND SURRY UNITS 1 AND 2

Chapter	Chapter Title, System/Component	Section[a]	Significant Similarities	Significant Differences
7.0 (continued)	System required for safe shutdown	Section 7.4	Similar to Beaver Valley and Surry.	No significant differences
	Safety-related display instrumentation	Section 7.5	Similar to Beaver Valley and Surry.	No significant differences
	Other safety systems	Section 7.6	Similar to Beaver Valley and Surry.	Surry locks power off one residual heat removal isolation valve and interlocks the other; administrative control of closure.
	Control systems	Section 7.7	Similar to Beaver Valley and Surry.	North Anna and Surry have a 50% load rejection capability without reactor trip; Beaver Valley has a 95% capability. Surry has turbine runback and rod withdrawal block on rod drop; North Anna and Beaver Valley do not.
9.0	Auxiliary systems			
	Fuel handling system	Section 9.1.4	Similar to Beaver Valley and Surry.	The Beaver Valley system is sized to store spent fuel from only one unit, while the Surry and North Anna systems are sized to handle fuel for two units.

a. Final safety analysis report for North Anna Units 1 and 2.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-1 (continued)

DESIGN OF NORTH ANNA NUCLEAR STEAM SUPPLY SYSTEM: COMPARISON WITH SYSTEMS AT BEAVER VALLEY UNIT 1 AND SURRY UNITS 1 AND 2

Chapter	Chapter Title, System/Component	Section[a]	Significant Similarities	Significant Differences
9.0 (continued)	Chemical and volume control system	Section 9.3.4	Similar to Surry.	The Beaver Valley differs from the North Anna Chemical and Volume Control Systems as follows: Boric acid system uses 4% boric acid instead of 12%. System has no deborating demineralizers. The system serves one reactor unit and does not have equipment shared between units as do North Anna and Surry. The system does not contain a letdown filter.
14.0	Initial tests and inspections		North Anna is the same as Beaver Valley.	North Anna and Beaver Valley are organized to reflect AEC guideline. Surry precedes issuance of the guide.
15.0	Accident analysis		Similar to Beaver Valley and Surry (Chapter 14).	The spectrum for small reactor coolant system break sizes for North Anna includes the 0.35 ft^2 and 0.5 ft^2 break sizes, while Beaver Valley and Surry do not. Surry differs from North Anna and Beaver Valley in that:

a. Final safety analysis report for North Anna Units 1 and 2.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-1 (continued)

DESIGN OF NORTH ANNA NUCLEAR STEAM SUPPLY SYSTEM: COMPARISON WITH SYSTEMS AT BEAVER VALLEY UNIT 1 AND SURRY UNITS 1 AND 2

Chapter	Chapter Title, System/Component	Section[a]	Significant Similarities	Significant Differences
15.0 (continued)	Accident analysis			Surry does not analyze for an accidental depressurization of the reactor coolant system. Surry does not analyze for an inadvertent loading of fuel assembly into an improper position. North Anna and Beaver Valley contain analyses of both split and guillotine break modes for break sizes of 0.6 x DE and larger. North Anna and Beaver Valley have format revision based on ANS classification of plant conditions. North Anna and Beaver Valley have expanded content based on the requirement to meet the AEC "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," February 1972.

a. Final safety analysis report for North Anna Units 1 and 2.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-2

COMPARISON OF ENGINEERED SAFETY FEATURES[a]

Parameter	North Anna Units 1 & 2	Beaver Valley Unit 1	Surry Units 1 & 2
Emergency Core Cooling System			
High-head safety injection pumps (charging pump)			
Number per unit	3	3	3
Design capacity gpm	150	150	150
Design total developed head, ft	5800	5800	5800
Low-head safety injection pumps			
Number per unit	2	2	2
Design capacity, gpm	3000	3000	3000
Design total developed head, ft	225	225	225
Accumulator			
Number per unit	3	3	3
Total volume, ft^3	1450	1450	1450
Water volume, ft^3 min.	925	925	925
Operating pressure, psia min.	600	600	600
Boron injection tanks			
Number per unit	1	1	1
Volume, gal	900	900	900
Boron concentration, ppm	20,000	21,000	20,000
Refueling water storage tank			
Number per unit	1	1	1
Usable volume, gal	450,000	425,000	350,000
Temperature, °F	40-55	45	45
Containment Heat Removal Systems			
Quench spray pumps			
Number per unit	2	2	2
Design capacity, gpm	2000	2000	3200
Design total developed head, ft water	240	285	225
Containment Heat Removal Systems (continued)			
Recirculation spray pumps			

a. The values presented in this table are the original values provided to the NRC for comparison prior to initial licensing. These values may vary from the current design values for the installed equipment. The actual design values for the installed equipment can be found in other sections of the UFSARs for North Anna and Surry.

b. The UA values for North Anna and Surry are the original values provided to the NRC. However, if the UA values are calculated using data from the original heat exchanger data sheets of NAS-160 and NUS-85, the resultant UA values for North Anna and Surry are 3.79×10^6 Btu/hr-°F and 3.8×10^6 Btu/hr-°F, respectively.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-2 (continued)

COMPARISON OF ENGINEERED SAFETY FEATURES[a]

Parameter	North Anna Units 1 & 2	Beaver Valley Unit 1	Surry Units 1 & 2
Number inside or outside of containment	2 inside, 2 outside	2 inside, 2 outside	2 inside, 2 outside
Design capacity, gpm	3300 inside, 3700 outside	3500	3500
Design total developed head, ft water	269 inside, 286 outside	245	230
Recirculation heat exchangers			
Number per unit	4	4	4
UA, Btu/hr-°F (per exchanger)	3.65×10^6 [b]	2.75×10^6	3.5×10^6 [b]
Recirculation flow, gpm	3500	3500	3500
Service water flow, gpm	4500	4000	6000
Post-design basis accident (DBA) hydrogen control systems			
Number of recombiners per unit	2	2	2
Type of recombiner	External thermal-electric	External thermal-electric	Internal thermal-electric
Recombiner flow rate (each), scfm	50	50	50
Number of purge blowers per unit	2	2	2
Purge blow flow rate (each), scfm	50	50	45

a. The values presented in this table are the original values provided to the NRC for comparison prior to initial licensing. These values may vary from the current design values for the installed equipment. The actual design values for the installed equipment can be found in other sections of the UFSARs for North Anna and Surry.

b. The UA values for North Anna and Surry are the original values provided to the NRC. However, if the UA values are calculated using data from the original heat exchanger data sheets of NAS-160 and NUS-85, the resultant UA values for North Anna and Surry are 3.79×10^6 Btu/hr-°F and 3.8×10^6 Btu/hr-°F, respectively.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-3

COMPARISON OF CONTAINMENT CONCEPTS

Parameter	North Anna Units 1 & 2	Beaver Valley Unit 1	Surry Units 1 & 2
Type	Subatmospheric	Subatmospheric	Subatmospheric
Internal diameter, ft.	126	126	126
Overall height, ft.	≈191	185	185
Free volume, 10^6 ft^3	1.825	1.8	1.8
Design pressure, psig	45	45	45
Calculated peak pressure for a LOCA, psig	40.6	42.7	39.2
Concrete thickness			
Vertical wall, ft-in.	4-6	4-6	4-6
Dome, ft-in.	2-6	2-6	2-6
Containment leak rate, percent per day of containment free volume	0.1	0.1	0.1
Reactor coolant system			
Volume (including pressurizer), ft^3	9438	9388	9455
Temperature, °F (coolant mass average)	574.5	569	572

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-4

COMPARISON OF CONTAINMENT ATMOSPHERE PRESSURE SENSORS

Parameter	Surry Units 1 & 2	North Anna Units 1 & 2	Beaver Valley Unit 1
Containment Atmosphere High-Pressure Transmitter			
Number of channels	4	3	3
Logic matrix	3/4	2/3	2/3
Approximate setpoint, psia	16.2	15.0	15.8
Containment Atmosphere High-High-Pressure Transmitter			
Number of channels	4	4	4
Logic matrix	3/4	2/4	2/4
Approximate setpoint, psia	25.0	25.0	24.3
Containment Atmosphere Intermediate-High-High-Pressure Transmitter			
Number of channels	NA[a]	3	NA
Logic matrix	NA	2/3	NA
Approximate setpoint, psia	NA	20	NA

a. NA = not applicable.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-5

COMPARISON OF RCP BUS PROTECTION

Parameter	Surry Units 1 & 2	North Anna Units 1 & 2	Beaver Valley Unit 1
Undervoltage RCP Buses			
Number of channels	3	3	3
Logic matrix	2/3	2/3	2/3
Approximate setting, V	2912 (70% of 4160V)	2912 (70% of 4160V)	2912 (70% of 4160V)
Underfrequency RCP Buses			
Number of channels	3	3	3
Logic matrix	2/3	2/3	2/3
Approximate setting, Hz	57.8	54-59	54-59

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-6

COMPARISON OF ACTUATION SIGNALS OF ENGINEERED SAFETY FEATURES

Parameter	Surry Units 1 & 2	North Anna Units 1 & 2	Beaver Valley Unit 1
Safety Injection Signal (SIS)			
Low pressurizer pressure coincident with low pressurizer level	Yes	Yes	Yes
High main steam line differential pressure	Yes	Yes	Yes
High main steam flow coincident with low main steam pressure or low-temperature average	Yes	Yes	Yes
High containment atmosphere pressure	Yes	Yes	Yes
Manual initiation	Yes	Yes	Yes
Containment Isolation Phase A (CIA) Signal			
Safety injection signal (SIS)	Yes	Yes	Yes
Manual initiation	Yes	Yes	Yes
Containment Isolation Phase B (CIB) Signal or Containment Depressurization Actuation (CDA) Signal			
High-high containment atmosphere pressure	Yes	Yes	Yes
Manual initiation	Yes	Yes	Yes
Steam Line Isolation Signal			
High main steam flow coincident with low main steam pressure or low reactor coolant temperature average	Yes	Yes	Yes
High-high containment pressure	Yes	No	Yes
Intermediate-high-high containment atmosphere pressure	No	Yes	Yes
Manual initiation	Yes	Yes	Yes

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-7

COMPARISON OF EMERGENCY GENERATOR AND STEAM GENERATOR AUXILIARY FEEDWATER PUMP START SIGNALS

Component	Surry Units 1 & 2	Beaver Valley Unit 1	North Anna Units 1 & 2
Emergency generator auto start signals	Emergency-bus undervoltage	Emergency-bus undervoltage	Emergency-bus undervoltage
	Safety injection actuation	Safety injection actuation	Safety injection actuation
	Containment high-high pressure signal	Emergency-bus supply breakers tripped	
Steam generator auxiliary feedwater pump (motor-driven) auto start signals	All steam generator feedwater pumps tripped	All steam generator feedwater pumps tripped	All steam generator feedwater pumps tripped
	2/3 low-low-level trip, any steam generator [a]	2/3 low-low-level trip, any steam generator [a]	2/3 low-low-level trip, any steam generator [a]
	Safety injection actuation	Safety injection actuation	Safety injection actuation
	Undervoltage reserve station service power		Undervoltage-reserve station service power
Steam generator auxiliary feedwater pump (turbine-driven) auto start signals	2/3 undervoltage on each station service bus [c]	Undervoltage on steam generator feedwater pump buses	2/3 undervoltage on station service bus [b]
	2/3 steam generator low-low-level trip (2/3 matrix) [a]	2/3 low-low-level trip, any steam generator [a]	2/3 steam generator low-low-level trip (2/3 matrix) [a, b]

a. Coincident with reactor coolant loop hot-leg stop valve open or reactor coolant loop cold-leg stop valve open.

b. These are the Unit 1 turbine-driven pump auto start signals at the time Unit 1 Operating License was issued. Before issuance of Unit 2 Operating License, the auto starts for the Unit 1 and Unit 2 turbine-driven pumps were changed to be the same as the auto starts for the motor-driven pumps.

c. A 1997 review of this table found this entry to be inaccurate. Surry's auto start should read, "2/3 undervoltage on station service busses." This table represents historical information as provided to the NRC during license application and will not be updated.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-8

COMPARISON OF PROCESS AND EFFLUENT RADIATION MONITORING SYSTEMS

Monitor	Surry[a]	Beaver Valley Unit 1	North Anna[a]
Aerated vent particulate	1	1	1
Aerated vent gas	1	1	1
Ventilation vent particulate	1	1	1
Ventilation vent gas	1	1	1
Multisampler ventilation sample particulate	1	1	1
Multisampler ventilation sample gas	1	1	1
Containment purge vent	1	2	1
Auxiliary building lower level vents	1	1	1
Auxiliary building upper level vents	1	1	1
Contiguous areas, vent	1	1	1
Component cooling water	2	1	1
Liquid wastes	1	1	2
Steam generator blowdown sample	2	1	2
Reactor coolant letdown gross activity	2	2	2
Containment recirculation cooler service water outlet	4	4	4
Service water discharge	2	0	2
Circulating water discharge	1	0	1
Condenser air ejection	1	1	1
Spares	2	4	2

a. North Anna Unit 1 and Surry Unit 1 are shown; North Anna Unit 2 is similar.

Note: This table is HISTORICAL and is not intended or expected to be updated for the life of the plant. However, for informational accuracy, it is noted that North Anna revised the number of blowdown sample monitors to 3 (one per steam generator) prior to initial startup.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-9

COMPARISON OF AREA RADIATION MONITORING SYSTEMS

Monitor	Surry[a]	Beaver Valley Unit 1	North Anna[a]
Containment structure low range	1	1	1
Containment structure high range	1	1	1
Manipulator crane	1	1	1
Incore instrumentation transfer device area	1	1	1
Decontamination area	1	1	1
New-fuel storage area	1	1	1
Fuel pit bridge	1	1	1
Auxiliary building control area	1	4	1
Sample room	1	0	1
Waste solidification and shipping area	1	1	1
Control room	1	1	1
Laboratory	1	1	1
Spares	2	2	2

a. North Anna Unit 1 and Surry Unit 1 are shown; North Anna Unit 2 is similar.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-10

COMPARISON OF AIRBORNE RADIATION MONITORING SYSTEMS

Monitor	Surry[a]	Beaver Valley Unit 1	North Anna[a]
Containment structure particulate	1	1	1
Containment structure gas	1	1	1

a. North Anna Unit 1 and Surry Unit 1 are shown; North Anna Unit 2 is similar.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-11

COMPARISON OF ELECTRICAL SYSTEMS

Parameter	North Anna Units 1 & 2	Surry Units 1 & 2	Maine Yankee
	Transmission System (Section 8.1.1)		
Transmission line	3 at 500 kV	4 - 2 at 500 kV and 2 at 230 kV	4 - 2 at 345 kV and 2 at 115 kV
	AC Power Systems (Section 8.3.1)		
Main transformers	6 at 330 mVA FOA	6 at 180/240/300 mVA OA/FOA/FOA	2 at 430 mVA
Unit station service transformer	6 at 15/20 mVA OA/FA	6 at 15/20 mVA OA/FA	2 - 1 at 30 mVA and 1 at 20 mVA
Reserve station service transformer	3 at 18/24/30 mVA OA/FA/FOA	3 at 18/24/30 mVA OA/FA/FOA	2 - 1 at 30 mVA and 1 at 20 mVA
	Emergency Power System (Section 8.3.1)		
Emergency diesel generators	4 at 3000 kW/2000 hr	3 at 2850 kW	2 at 2500 kW
Emergency 4.16 kV buses	4 at 1200 A	4 at 1200 A	2 at 1200 A
Emergency 4.16 kV tie buses	2 at 1200 A	2 at 1200 A	None
	AC Vital-Bus System		
Distribution cabinets	8	4	4
Inverters	8 - 2 at 20 kVA and 6 at 15 kVA	4 at 10 kVA	4 at 10 kVA
	125-V DC System (Section 8.3.2)		
Unit batteries (125V)	8 - 4 at 900 A-h, 2 at 800 A-h, and 2 at 1500 A-h	4 at 1500 A-h	4 - 2 at 1800 A-h and 2 at 480 A-h
Battery chargers	12 at 250 A	4 at 200 A	4 at 250 A

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-12

COMPARISON OF WASTE SYSTEMS

Parameter	North Anna Units 1 & 2	Surry Units 1 & 2	Beaver Valley Unit 1
Clarifier-demineralizer filter	(Not used except for demineralizer)		
Number	2	None	None
Capacity, maximum gpm	300		
Clarifier equipment	(Not used except for demineralizer)		
Number	2 sets	None	None
Capacity, gpm	300		
Design decontamination factor (DF) (including demineralizer)	10 for Mo, I_2, and Te 100 for all others		
Flat-bed filter	(Not used)		
Number	1	None	None
Capacity, maximum gpm	75		
Liquid waste demineralizer filter			
Number	1	1	1
Capacity, maximum gpm	75	75	50
Liquid waste demineralizer (IEFS)			
Number	1	1	1
Capacity, maximum gpm	50	50	50
Design decontamination factor	(a)	(a)	10
Liquid waste effluent filters (IEFS)			
Number	2	1	2
Capacity, maximum gpm	75	75	50
Process vent charcoal filters			
Number	2	2	2
Capacity, maximum operating, scfm	1000/300	1000/300	1250/1250
Design decontamination factor	100 for Iodine	100 for I	100 for I

a. See design decontamination factor for waste disposal evaporator.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-12 (continued)

COMPARISON OF WASTE SYSTEMS

Parameter	North Anna Units 1 & 2	Surry Units 1 & 2	Beaver Valley Unit 1
Spent-resin dewatering filter			
Number	1	1	2
Capacity, maximum gpm	150	150	-
Waste gas catalytic recombiner	(Permanently removed)		
Number	1	1	5
Capacity, scfm	1.5	1.5	None
Waste disposal evaporator, number	(Not used)		
Number	1	1	1
Capacity, net gpm	6	6	6
Trays, number	0	0	5
Design decontamination factor (including demineralizer)	10^4	10^4	10^5
Waste disposal evaporator reboiler	(Not used)		
Number	2 in series	1	1
Total duty, Btu/hr	4.0×10^6	4.0×10^6	3.9×10^6
Waste gas decay tank			
Number	2	2	3
Capacity, gal	3400	3250	743
Decay time, days	60	60	30
Operating pressure, maximum psig	145	150	100

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-13

COMPARISON OF NUCLEAR PLANT SYSTEMS

Parameter	North Anna Units 1 & 2	Beaver Valley Unit 1	Surry Units 1 & 2
Fuel Pit Cooling System			
Fuel pit cooling pump			
Number	2	2	2
Capacity, gpm	2750	750	4200
Design TDH, ft.	79.4	74	62
Fuel pit coolers			
Number	2	2	2
Duty, Btu/hr (each)	28,400,000	11,400,000	34,750,000
Fuel pit water flow, gpm	2600	650	4200
Component cooling flow, gpm	2000	1100	1322
Component Cooling System			
Component cooling pumps	Dual rated		
Number per unit	2	3	2
Capacity, gpm	8000/12,000	4700	9000
Design, TDH, ft.	191.7/140.0	250	200
Component cooling heat exchangers			
Number per unit	2	3	2
Duty, Btu/hr (each)	52,000,000	33,000,000	50,300,000
Component cooling water flow, gpm	9000	4620	6830
Service water flow, gpm	10,500	6060	9000
Service Water System			
Service water pumps			
Number per unit	3	3	None
Capacity, gpm	11,500	9000	Gravity
Design TDH, ft	127	155	Flow
Boron Recovery System			
Gas stripper			
Number	2	2	1
Capacity, gpm (each)	135	75	240

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.3-13 (continued)
COMPARISON OF NUCLEAR PLANT SYSTEMS

Parameter	North Anna Units 1 & 2	Beaver Valley Unit 1	Surry Units 1 & 2
Boron Recovery System (continued)			
Boron recovery tanks			
Number	3	2	3
Capacity, gal (each)	120,000	195,000	120,000
Boron evaporator			
Number	2	2	2
Capacity, gpm (each)	20	15	20
Primary water tanks			
Number	2	2	2
Capacity, gal (each)	180,000	75,000	180,000
Evaporator bottoms tank			
Number	1	1	1
Capacity, gal	4000	2000	4000
Gas stripper surge tank			
Number	1	None	1
Capacity, ft^3	80	None	70
Design pressure, psig	200		200
Test tanks			
Number	2	2	2
Capacity, gal (each)	20,000	12,000	30,000
Primary water pumps			
Number	Service - 2 Standby - 2	2	2
Capacity, gpm (each)	Service - 120 Standby - 200	200	350
Design TDH, ft	Service - 270 Standby - 285	310	255

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

1.4.1 Introduction

Virginia Electric and Power Company has contracted with Westinghouse Electric Corporation for the purchase of (1) the nuclear steam supply system for each nuclear unit, including its turbine generator, and (2) uranium dioxide and its fabrication into fuel for each reactor. The balance of plant was designed and constructed by VEPCO with architectural, engineering, and construction services from Stone & Webster Engineering Corporation. Consulting services also were received from NUS Corporation and Dames & Moore, Inc. NUS Corporation assisted VEPCO in areas of site meteorology and climatology and radiation dose assessment; Dames & Moore, Inc., assisted VEPCO in areas of site geology, hydrology, and seismology.

The contractors involved in the day-to-day design and construction of North Anna Units 1 and 2 were Westinghouse and Stone & Webster. Each of these parties, by contract, was assigned responsibility for the design of certain systems, structures, and components. The assignment of responsibility for major structures, systems, and components is shown in Table 3.2-1. Chapter 17 describes the quality assurance programs that were used to ensure that these responsibilities were carried out in accordance with applicable codes, rules, and regulations.

The following subsections provide information on the experience and qualifications of the aforementioned agents and contractors.

1.4.2 Stone & Webster Engineering Corporation

Stone & Webster Engineering Corporation is a Massachusetts corporation with its main office in Boston, Massachusetts. The home office staff of engineers, designers, construction specialists, and clerical personnel during the North Anna project numbered 4000.

Before the advent of commercial nuclear power, Stone & Webster was engaged in the engineering, design, and construction of hydroelectric and fossil-fueled power plants and chemical process plants. During the past 25 years, Stone & Webster has engineered, designed, and/or constructed 176 hydroelectric and fossil-fueled power plants for a total electric power output of about 33,000,000 kW.

Stone & Webster has been actively engaged in nuclear engineering and the construction of nuclear plants since 1954. They have participated in the design and/or construction of the Shippingport atomic power plant, the Army Package Power Reactor, the Yankee-Rowe nuclear power station, the Carolinas-Virginia nuclear station, the Haddam Neck plant of the Connecticut Yankee Atomic Power Company, Surry Units 1 and 2 of VEPCO, the Nine Mile Point power station of the Niagara Mohawk Power Corporation, the Maine Yankee atomic power station, Beaver Valley Unit 1 of the Duquesne Light Company et al., and the James A. FitzPatrick nuclear power station of the Power Authority of the State of New York et al., all of which are operating or have operated successfully.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

In addition to North Anna Units 1 and 2, they have under design or construction at this time the Greene County nuclear power station of the Power Authority of the State of New York, North Anna Units 3 and 4 of VEPCO, the Shoreham nuclear power station and Jamesport Units 1 and 2 of the Long Island Lighting Company, the Beaver Valley power station of the Duquesne Light Company et al., Millstone Unit 3 and Montague Units 1 and 2 of the Northeast Utilities Service Company, Nine Mile Point Unit 2 of the Niagara Mohawk Power Corporation, River Bend Unit 1 of the Gulf States Utilities Company, and Sundesert Units 1 and 2 of the San Diego Gas and Electric Company.

1.4.3 Westinghouse Qualifications and Experience as a Supplier of Nuclear Steam Supply Systems

The experience of the Westinghouse Electric Corporation in nuclear plants for the electrical utility industry is demonstrated by the PWR plants that it has designed, developed, and manufactured. Table 1.4-1 lists all Westinghouse PWR plants, including plants under construction or on order at the time of the original FSAR submitted for the North Anna units.

Westinghouse has long held a position of leadership in the electrical products manufacturing industry. This leadership is based on a tradition of standard and new product reliability and quality. Nowhere is this leadership displayed more vividly than in nuclear power. Through early participation in basic research and engineering developments, Westinghouse has established a broad technological foundation in nuclear power applications. It has also established a continuing program of sound technological development that enables the corporation to offer to the electrical utility industry a reliable and safe source of power from the atom.

The experience of Westinghouse in nuclear activity is evident in numerous atomic power projects—completed, soon to go into operation, or being developed. The following paragraphs describe Westinghouse-designed PWR plants that were in operation at the time of the original FSAR submittal for the North Anna units.

1.4.3.1 **Plants in Operation**

Westinghouse PWR nuclear power plants in operation are as follows:

1. Shippingport

   Shippingport was the world's first large central station nuclear power plant. The reactor plant was designed by the Bettis Atomic Power Laboratory, which is operated by Westinghouse under AEC contract. Shippingport's pressurized water reactor has produced steam for the Duquesne Light Company's turbine-generator plant since December 1957.

2. Yankee-Rowe

   Singled out by the AEC as a "nuclear success story," Yankee went online in November 1960. Owned and operated by the Yankee Atomic Electric Company, Yankee has progressed from an initial rating of 120 MWe to its present 176 MWe. Westinghouse supplied the nuclear steam supply system and the turbine generator.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

3. Enrico Fermi

   The Enrico Fermi nuclear plant was one of the first Westinghouse-designed plants to incorporate the chemical shim control of reactivity. Chemical shim has since become a standard feature of Westinghouse PWR control. Enrico Fermi achieved initial criticality in June 1964 and began power operation in October 1964. The plant is rated at 256 MWe.

4. Ardennes

   The Ardennes plant is unique in that the Westinghouse pressurized water reactor and its auxiliaries are housed in man-made caverns. Ardennes, a Franco-Belgian undertaking owned and operated by the Societe d'Energie Nucleaire Franco-Belge des Ardennes (SENA), is located in France near the France-Belgium border. Ardennes achieved initial criticality in October 1966 and began power operation in 1967.

5. San Onofre

   San Onofre employs the Westinghouse-developed rod cluster control that has become a standard feature of the Westinghouse pressurized water reactor. Owned by the Southern California Edison Company and the San Diego Gas and Electric Company, the 430-MWe plant is located near San Clemente. Westinghouse supplied the nuclear steam supply system and the turbine generator. Initial criticality was achieved in June 1967, and power operation began in January 1968.

6. Connecticut Yankee

   Owned and operated by the Connecticut Yankee Atomic Power Company, this plant went critical in mid-1967 and attained full power operation in December 1967. Like San Onofre, the plant employs rod cluster control in conjunction with chemical shim control. Westinghouse supplied the nuclear steam supply system and the turbine generator. In March 1969, Connecticut Yankee received AEC approval to uprate the plant from its initial rating of 462 MWe to 567 MWe.

7. Jose Cabrera

   The Jose Cabrera station is located near Zorita, Spain. The 153-MWe plant employs rod cluster control, chemical shim control, and a Zircaloy-clad core. Construction began in mid-1965, and power operation began in 1968. Jose Cabrera is owned and operated by Union Electrica Madrilena, a Spanish utility.

8. Beznau 1

   Beznau 1, Switzerland's first commercial nuclear power plant, achieved initial criticality on June 30, 1969, and supplied power to the system on July 17, 1969. The 350-MWe plant was designed and constructed by the Westinghouse-Brown Boveri Consortium for the owner/operator utility, Nordostschweizerische Kraftwerke A.G. The plant started producing power less than 4 years after the award of the plant contract.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

9. Robert Emmett Ginna

   The Robert Emmett Ginna Plant, owned and operated by Rochester Gas and Electric Corporation, is located on the south shore of Lake Ontario. Westinghouse supplied the 420-MWe plant on a turnkey basis. Construction began in April 1966, and initial criticality was achieved on November 9, 1969—just 42 months after the start of construction. Power was supplied to the system on December 2, 1969.

10. Mihama 1

    Mihama 1, a 2-loop, 320-MWe unit, is owned and operated by the Kansai Electric Power Company. Mihama 1 marks the beginning of a line of Westinghouse pressurized water reactors supplying the generation needs of the Far East. Westinghouse International Company was the prime contractor for Unit 1, supplying the nuclear steam supply system engineering, nuclear fuel, and some major system components. The plant required only 44 months from the start of site construction to first power production in August 1970.

11. H. B. Robinson 2

    This plant is a 3-loop, 700-MWe unit that was built on a turnkey basis for the Carolina Power & Light Company. The plant is located at a site near Hartsville, South Carolina, on a man-made cooling lake. The construction permit was granted in April 1967. The plant achieved criticality and first power to system in October 1970.

12. Point Beach 1

    Point Beach 1 is a 454-MWe Westinghouse PWR unit built on a turnkey basis for the Wisconsin Michigan Power Company and the Wisconsin Electric Power Company. The plant is located near Two Creeks, Wisconsin, 90 miles north of Milwaukee on Lake Michigan. This is the first of two units at the station that will share many facilities and auxiliary systems. The construction permit for Unit 1 was granted in July 1967, and initial criticality and first power to system were achieved in November 1970.

13. Surry 1

    This plant is a 3-loop, 823-MWe PWR unit built for VEPCO. The plant is located in Surry County, Virginia, on a point of land called Gravel Neck, which juts into the James River. This is the first unit of a twin-unit station. Westinghouse supplied the nuclear steam supply system and the turbine generator. The construction permit for Unit 1 was granted on June 26, 1968, and the unit achieved initial criticality on July 1, 1972.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

1.4.3.2 Westinghouse Facilities

Westinghouse, in its effort to plan for the future, has developed a broad range of facilities to satisfy the needs of the nuclear industry. The facilities are as follows:

1. Columbia Plant, Nuclear Fuel Division

   The Columbia Plant is capable of performing all the operations necessary to manufacture finished nuclear fuel assemblies. These operations include the conversion of uranium hexafluoride to uranium dioxide powder, the fabrication of fuel assembly grids, complete pellet loading, and the final fabrication of assemblies. The plant, located at Columbia, South Carolina, began full production in early 1970. The Columbia plant is the largest commercial nuclear fuel fabrication facility in the world.

2. Tampa Division

   The Tampa Division plant is the world's most modern heat transfer equipment production facility. The plant has 236,000 ft^2 of working space with 2 manufacturing aisles for the production of steam generators and pressurizers. Transportation facilities include four railroad spurs and a complete barge slip and dock facility for water shipment to all parts of the world. The Tampa plant made its first steam generator and pressurizer shipment in September 1969.

3. Pensacola Division

   The Pensacola Division plant, located on Escambia Bay on the northwest coast of Florida, is a 140,000-ft^2 manufacturing plant committed to the production of precision reactor vessel internals. Contributing to the precision manufacturing capability is an environmental control system that minimizes annual temperature variations throughout the shop area. Transportation facilities for the plant include a railroad spur that permits loading and unloading inside the shop, as well as access to barge-loading facilities on Escambia Bay. Pensacola shipped its first package of reactor internals in July 1970.

4. Cheswick Plant, Electro-Mechanical Division

   The Electro-Mechanical Division was established in 1953 in Cheswick, Pennsylvania, to manufacture canned motor primary coolant pumps for nuclear reactors. The product line was expanded to include shaft seal pumps (reactor coolant pumps), valves, and control-rod drive mechanisms—essential components of the Westinghouse pressurized water reactor. The facility occupies 404,000 ft^2 and contains the most modern facilities available for the production and testing of nuclear plant components.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

5. Speciality Metals Division

   The Specialty Metals facility is located in Blairsville, Pennsylvania. Several essential parts of PWR components are manufactured at Blairsville, including the Inconel tubing for steam generators and the Zircaloy seamless tubing for nuclear fuel cladding. At Blairsville, complete quality control facilities are used for the analysis and evaluation of all specialty metal products used in Westinghouse nuclear systems.

6. Westinghouse Nuclear Center

   The headquarters of Westinghouse Nuclear Energy Systems is located just east of Pittsburgh in Monroeville, Pennsylvania. Operating primarily as a headquarters and engineering facility, the complex houses many of the divisions involved in Westinghouse's nuclear activities associated with the electrical utility industry.

7. Zion Nuclear Training Center

   The Westinghouse Electric Corporation and the Commonwealth Edison Company of Chicago have built and are operating a nuclear training center at Zion, Illinois. The 28,000-ft^2 training center contains classrooms, a training reactor, a training material center, video recording facilities, and multiplant nuclear power plant simulators. Westinghouse staffs and operates the center, supplies all the equipment required, and is responsible for the development and presentation of all training programs. Commonwealth Edison provided the building and access to the Zion nuclear units for in-plant observation training, and advises and assists Westinghouse in developing training programs.

1.4.4 Consultants

1.4.4.1 Dames & Moore, Inc.

Dames & Moore is a nationwide consulting firm in the field of soil mechanics and building foundations. The firm has developed recognized competence in the areas of geology and geophysical problems in the nuclear industry. The firm was retained to provide consulting services for the North Anna project in the areas of geology, hydrology, and seismology. Dames & Moore has provided similar services for numerous nuclear projects, including VEPCO's Surry Power Station.

1.4.4.2 NUS Corporation

NUS Corporation of Rockville, Maryland, was retained to provide general consulting services on the North Anna project in the areas of general site climatology and diffusion meteorology, demographic and land use studies, and radiation dose assessment. NUS has extensive experience in the nuclear industry in a broad spectrum of activities, including reactor engineering safeguards, radiation safety and shielding, site environmental studies, reactor siting, and reactor design services.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.4-1

WESTINGHOUSE PWR NUCLEAR POWER PLANTS

Plant	Operator Utility	Location	Scheduled Commercial Operation	Nominal MWe Net	Number of Loops
Shippingport	Duquesne Light Company	Pennsylvania	1957	157	4
Yankee-Rowe	Yankee Atomic Electric Company	Massachusetts	1960	176	4
Enrico Fermi	Ente Nazionale per L'Energia Elletrica (ENEL)	Italy	1964	256	4
Ardennes	Societe d'Energie Nucleaire Franco-Belge des Ardennes (SENA)	France	1967	250	4
San Onofre	Southern California Edison Company San Diego Gas & Electric Company	California	1968	430	3
Connecticut Yankee	Atomic Power Company	Connecticut	1967	567	4
Jose Cabrera	Union Electrica Madrilena	Spain	1968	153	1
Beznau 1	Nordostschweizerische Kraftwerke AG (NOK)	Switzerland	1969	350	2
Robert Emmett Ginna	Rochester Gas and Electric Corporation	New York	1969	420	2
Turkey Point 3	Florida Power and Light Company	Florida	1972	721	3
Indian Point 2	Consolidated Edison Company of New York	New York	1972	873	4
H. B. Robinson 2	Carolina Power and Light Company	South Carolina	1970	700	3
Point Beach 1	Wisconsin Michigan Power Company Wisconsin Electric Power Company	Wisconsin	1970	454	2
Mihama 1	Kansai Electric Power Company	Japan	1970	320	2
Salem 1	Public Service Electric and Gas Company	New Jersey	1973	1050	4
Surry 1	Virginia Electric and Power Company	Virginia	1972	823	3
Surry 2	Virginia Electric and Power Company	Virginia	1973	823	3
Diablo Canyon 1	Pacific Gas and Electric Company	California	1973	1090	4

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.4-1 (continued)

WESTINGHOUSE PWR NUCLEAR POWER PLANTS

Plant	Operator Utility	Location	Scheduled Commercial Operation	Nominal MWe Net	Number of Loops
Point Beach 2	Wisconsin Michigan Power Company Wisconsin Electric Power Company	Wisconsin	1972	454	2
Turkey Point 4	Florida Power and Light Company	Florida	1972	721	3
Prairie Island 1	Northern States Power Company	Minnesota	1972	527	2
Zion 1	Commonwealth Edison Company	Illinois	1972	1050	4
Kewaunee	Wisconsin Public Service Corporation Wisconsin Power and Light Company Madison Gas and Electric Company	Wisconsin	1972	527	2
Indian Point 3	Consolidated Edison Company of New York	New York	1973	965	4
Salem 2	Public Service Electric and Gas Company	New Jersey	1973	1050	4
Prairie Island 2	Northern States Power Company	Minnesota	1974	527	2
Zion 2	Commonwealth Edison Company	Illinois	1973	1050	4
Donald C. Cook 1	Indiana and Michigan Electric Company	Michigan	1973	1100	4
Donald C. Cook 2	Indiana and Michigan Electric Company	Michigan	1974	1100	4
North Anna 1	Virginia Electric and Power Company	Virginia	1974	875	3
Beaver Valley 1	Duquesne Light Company Pennsylvania Power Company Ohio Edison Company	Pennsylvania	1973	847	3
Mihama 2	Kansai Electric Power Company	Japan	1972	500	2
Beznau 2	Nordostschweizerische Kraftwerke AG (NOK)	Switzerland	1972	350	2
Sequoyah 1	Tennessee Valley Authority	Tennessee	1973	1124	4
Sequoyah 2	Tennessee Valley Authority	Tennessee	1974	1124	4

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.4-1 (continued)

WESTINGHOUSE PWR NUCLEAR POWER PLANTS

Plant	Operator Utility	Location	Scheduled Commercial Operation	Nominal MWe Net	Number of Loops
Diablo Canyon 2	Pacific Gas and Electric Company	California	1974	1110	4
Ringhals	Swedish State Power Board	Sweden	1974	809	3
Trojan 1	Portland General Electric Company	Oregon	1974	1100	4
Ko-Ri	Korea Electric Company	Korea	1975	564	2
Joseph M. Farley 1	Alabama Power Company	Alabama	1975	829	3
Takahama 1	Kansai Electric Power Company	Japan	1974	826	3
William B. McGuire 1	Duke Power Company	North Carolina	1975	1150	4
William B. McGuire 2	Duke Power Company	North Carolina	1976	1150	4
North Anna 2	Virginia Electric and Power Company	Virginia	1975	875	3
Aguirre	Puerto Rico Water Resources Authority	Puerto Rico	1976	600	2
Watts Bar 1	Tennessee Valley Authority	Tennessee	1976	1140	4
Watts Bar 2	Tennessee Valley Authority	Tennessee	1977	1140	4
Ohi 1	Kansai Electric Power Company	Japan	1976	1125	4
Joseph M. Farley 2	Alabama Power Company	Alabama	1977	829	3
Virgil C. Summer 1	South Carolina Electric and Gas Company	South Carolina	1977	918	3
Ohi 2	Kansai Electric Power Company	Japan	1976	1125	4
Commonwealth Edison Unit 10	Commonwealth Edison Company	Illinois	1978	1100	4
Commonwealth Edison Unit 11	Commonwealth Edison Company	Illinois	1979	1100	4
Shearon Harris 1	Carolina Power and Light Company	North Carolina	1977	918	3
Shearon Harris 2	Carolina Power and Light Company	North Carolina	1978	918	3

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.4-1 (continued)

WESTINGHOUSE PWR NUCLEAR POWER PLANTS

Plant	Operator Utility	Location	Scheduled Commercial Operation	Nominal MWe Net	Number of Loops
Shearon Harris 3	Carolina Power and Light Company	North Carolina	1979	918	3
Shearon Harris 4	Carolina Power and Light Company	North Carolina	1980	918	3
Angra dos Reis	Central Electrica de Furnas	Brazil	1976	600	2
Crystal River 4	Florida Power Corporation	Florida	1977	897	3
Alvin W. Vogtle 1	Georgia Power Corporation	Georgia	1978	1100	4
Alvin W. Vogtle 2	Georgia Power Corporation	Georgia	1979	1100	4
Beaver Valley 2	Duquesne Light Company Pennsylvania Power Company Ohio Edison Company	Pennsylvania	1978	847	3
Ringhals 3	Swedish State Power Board	Sweden	1977	900	3
Ringhals 4	Swedish State Power Board	Sweden	1979	900	3
Almaraz 1	Hidroelectrica Espanola Union Electrica Madrilena Compania Sevillana de Electricidad	Spain	1976	900	3
Almaraz 2	Hidroelectrica Espanola Union Electrica Madrilena Compania Sevillana de Electricidad	Spain	1977	900	3
Lemoniz 1	Iberduero, S. A.	Spain	1976	900	3
Lemoniz 2	Iberduero, S. A.	Spain	1978	900	3

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

Reference 1 presents descriptions of the safety-related research and development programs that were being carried out for, by, or in conjunction with Westinghouse Nuclear Energy Systems, and were applicable to Westinghouse pressurized water reactors at the time of the original FSAR submittal for the North Anna units.

For each safety-related program then in progress, the program is first introduced, followed, where appropriate, by background information. This is followed by a description of the program that gives the program objectives and presents pertinent recent results. Finally, a backup position may be given for those generally experimental rather than analytical programs that had not yet reached a stage where it was reasonably certain that the results confirm the expectation. The backup position is one that might be used if the results are unfavorable; it is not necessarily the only course that might be taken.

The term "research and development," as used in this report, is the same as that used by the NRC in Section 50.2 of its regulations:

> (n) "research and development" means (1) theoretical analysis, exploration or experimentation; or (2) the extension of investigative findings and theories of a scientific nature into practical application for experimental and demonstration purposes including the experimental production and testing of models, devices, equipment, materials and processes.

The technical information generated by these research and development programs will demonstrate the safety of the design and more sharply define the margins of conservatism, and it may lead to design improvements.

Included in the overall research and development effort are the programs described below. These programs are applicable to the 17 x 17 fuel assembly.

1.5.1 Verification Test (17 x 17)

The design of the reactor uses a 17 x 17 array of fuel rods and thimbles in a fuel assembly. This array is conceptually similar to but geometrically different from the 15 x 15 array used in previous designs. The 17 x 17 design is considered to be a relatively small extrapolation of the 15 x 15 design. Comprehensive testing has been planned, however, to verify that the extrapolation is sufficiently conservative. A preliminary evaluation of the data obtained at the time of the original FSAR submittal for the North Anna units did not reveal any anomalies.

Design changes, if necessary, will be made to the reference 17 x 17 hardware in the unlikely event that any of the experimental results fall outside the conservative design values used in analysis.

Westinghouse maintains that no plant need be designated a prototype and instrumented to verify the 17 x 17 fuel design. The change in the flow-induced vibration response of the internals attributable to the change from a 15 x 15 to a 17 x 17 fuel design is minimal for the following reasons:

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

1. The only internals that change as a result of the design change from the 15 x 15 to the 17 x 17 fuel assembly are the guide tube and the control-rod drive line.
2. The guide tube is rigidly attached at the upper core support plate only. The upper core plate serves only to align the guide tubes. Because of this type of support arrangement, the guide tube makes a minimal contribution to the vibration response of the core barrel and other internals.
3. The effective flow area of the guide tube for the 17 x 17 fuel assembly is essentially the same as that for the 15 x 15 array and therefore there are no significant differences in the flow distribution in the upper plenum.
4. The differences in mass and spring rate between the 15 x 15 and 17 x 17 fuel assemblies are very small (approximately 3%). This ensures that the effects of the fuel on the vibration response of the reactor internals will remain essentially unchanged. The preoperational hot functional flow testing presented in Chapter 14 is considered the most conservative test condition, since higher flow rates exist.

More adequate and meaningful tests to verify the change from the 15 x 15 to the 17 x 17 array would be to test the new guide tube and fuel assembly designs individually in a special test facility, such as the loop test facilities at the Westinghouse Forest Hills site. This type of program was in fact conducted and is discussed below.

Some of the verification work described herein was conducted using 17 x 17 assemblies of a 7-grid design, whereas the selected 17 x 17 assembly design has 8 grids. Tabulated below are those 17 x 17 tests that used a 7-grid geometry and the effect of adding an eighth grid.

Test	Parameter	Effect
Fuel assembly structural test	Axial stiffness	Negligible effect from at blowdown impact forces (Reference 2)
	Lateral impact	Additional grid shares impact load (Reference 2)
Prototype assembly test	Pressure drop	Margin between 7-grid design delta P and D-loop results (Reference 3) adequate to cover the additional delta P resulting from the additional grid (less than 5% increase in delta P)
	Lift force	Margin between 7-grid design lift force and D-loop results (Reference 3) adequate to cover the additional lift force resulting from the additional grid
	Rod vibration	Decreased span length results in improved vibratory amplitude and reduced rod wear
Departure from nucleate boiling (DNB)	DNB correlation	Addition of a grid increases mixing, which increases DNB margin
Incore flow mixing	TDC	TDC increases as grid spacing decreases (Reference 4)

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

The above tabulation shows (1) that additional design changes are not required (e.g., no new fuel assembly hold-down spring is needed) as a result of the addition of a grid, and (2) that 7-grid test information can be used to assess the adequacy of the 8-grid design. Additional testing to specifically investigate the 8-grid assembly is not required.

1.5.1.1 **Rod Cluster Control Spider Tests**

Test Purpose and Parameters

The 17 x 17 RCC spider is conceptually similar to but geometrically different from the 15 x 15 spider. The 17 x 17 spider supports 24 rodlets (the 15 x 15 design supports 20) with no vane supporting more than two rodlets (same as the 15 x 15 design). The RCC spider tests verified the structural adequacy of the design.

The spider vane-to-hub joint was tested for structural adequacy by (1) a vertical static-load test to failure and (2) a vertical fatigue test to approximately 3 million steps. The static-load test was performed by applying tensile and compressive loads to the spider. The load was applied parallel to the spider hub and reacted between the spider hub and fingers. The spider fingers shared the load equally. The number of cycles for the fatigue test was determined from the expected number of steps a control-rod drive mechanism would experience during 20 years in a load-follow reactor (1.5×10^6 steps). The test met the recommended cyclic test requirements of the ASME Code, Section III, Appendix II, Paragraph 1520.

The spring pack within the spider hub was tested to determine the spring load deflection characteristic as a function of the loading cycles seen by the spring. The test was terminated after 1000 cycles compared to a 400-cycle (rod drop) design value. The test loads were equal to or greater than that predicted to result in a yielding of the spring material. These loads were in excess of the design values. The test acceptance criterion was for the spring to retain adequate preload after the repeated cycling.

Facility

The 17 x 17 spider tests were performed at the Westinghouse Engineering Mechanics Laboratory (see Section 1.5.3.2.15).

Status

The spider tests have been completed. A vertical static-load test approximately seven times the design dynamic load did not result in spider vane-to-hub joint failure. A spider was tested to 2.8×10^6 steps without failure. The spider loading was 110% of the design value for 1.8×10^6 cycles and 220% of the design loading for 1×10^6 cycles. The design load is 3600 lb compression and 1800 lb tension. The spring test resulted in negligible preload loss.

1.5.1.2 **Grid Tests**

Test Purpose and Parameters

The 17 x 17 grid is conceptually similar but geometrically different from the 15 x 15 grid. The purpose of the grid tests was to verify the structural adequacy of the grid design.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Load deflection tests have been made on the grid spring and dimple. The grid spring radial (normal) stiffness and grid dimple radial and tangential stiffnesses were obtained. This information was used to verify that the fuel-rod clad wear evaluation has been based on conservative values of these parameters. The fuel-rod evaluation is conservative, as shown by the flow test results presented in Reference 3.

The grid-buckling strength has been determined from tests. The grid test specimens had short sections of fuel tubing inserted in the cell in place of fuel rods. These tests were used to verify that grid buckling during a postulated seismic occurrence does not interfere with control-rod insertion.

The grid-buckling strength is defined as the maximum load that can be applied without failure. In the case of static tests, the applied load, which is deflection controlled, results in an elastic buckling failure, since no permanent deformation is experienced on removing the load. The static test established the lower limit for grid failure.

The grid dynamic buckling strength is similarly defined as the maximum load resulting from an impact that can be applied without failure; however, some localized permanent deformation occurs before the maximum load is attained.

The grids were tested under both static and dynamic loads. The loads were applied uniformly to the face of the outside strap, transmitted directly through the grid, and reacted at the grid face opposite the input. A description of the grid impact test is given in Reference 2. A description of the analytical use of the test parameters is also given in Reference 2.

Facility

The grid tests were conducted in the Westinghouse Forest Hills Engineering Mechanics Laboratory (see Section 1.5.3.2.15).

Status

The grid tests have been completed. Test results are in agreement with pretest design values. The test results, along with fuel assembly structural test results, were factored into the seismic analysis (Reference 2).

1.5.1.3 Fuel Assembly Structural Tests

Test Purpose and Parameters

The 17 x 17 fuel assembly tests were performed to determine mechanical strength and properties. The fuel assembly parameters obtained were as follows: lateral and axial stiffnesses, impact and internal structural damping coefficients, vibration characteristics, and lateral and axial impact responses to postulated accident loads. The parameters obtained from the lateral dynamic tests are used for seismic analysis, while those obtained from the axial tests are incorporated in the LOCA (blowdown) accident analysis. The remaining tests are primarily to demonstrate that the assembly has sufficient mechanical strength to avoid damage during shipment, normal handling, and normal operation.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

The fuel assembly is subjected to both lateral and axial loads to obtain the respective static axial and lateral stiffnesses. The information obtained from these tests is used to establish parameters primarily for accident analysis, since these conditions appear limiting. The axially applied loads, which were well in excess of shipment, normal handling, and normal operational design loads, did not result in permanent deformation or damage to fuel assemblies.

Lateral tests were accomplished with both nozzles fixed in place and forces applied to various grids. The lateral stiffness was found by incrementally increasing and decreasing the static load.

The fuel assembly was tested in a vertical position using core pins to simulate reactor support conditions. An electrodynamic shaker was attached to the center (fourth) grid to provide excitation. The fuel assembly mode shapes and corresponding natural frequencies were obtained from displacement transducers. A comparison of analytical and experimental results is given in Reference 2. Experimental vibration studies of individual fuel rods were also performed. The rods were tested under simulated fuel assembly support conditions and as assembled in a prototype fuel assembly. The information obtained from these tests included the fundamental frequencies and mode shapes. A general test description and a summary of the results are presented in Reference 3.

The fuel assembly axial stiffness was found by incrementally increasing the static load (compressive) and then incrementally decreasing the static load.

Lateral impact tests were performed by displacing the center of the assembly with the nozzles fixed in place. The assembly was released and allowed to impact on lateral restraints at each of the five center grid locations.

The axial impact response and damping were found by dropping the fuel assembly from various heights. The axial impact test was performed with the fuel assembly in the upright position.

The relevant parameters measured during the lateral and axial impact tests were as follows:

1. Impact duration versus impact load.
2. Impact force versus drop height or initial displacement.
3. Impact during damping or restitution as a function of impact force.

A general description of the test procedure, including a description of the use of the parameters as related to accident analysis, is presented in Reference 2.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

There is a general axial test buckling criterion that does not allow the local buckling of components that could prevent control-rod insertion during an accident. The overall fuel assembly buckling and local component buckling is checked during the axial static and dynamic tests. The lateral displacement associated with the overall (beam-type) fuel assembly buckling is constrained by the reactor internals and therefore does not reduce the ultimate strength of the fuel assembly. Local component buckling was not experienced during either the static or dynamic tests for loads well in excess of the design values. The general acceptance criteria were not violated.

Facility

These tests were conducted at the Westinghouse Engineering Mechanics Laboratory (see Section 1.5.3.2.15).

Status

The fuel assembly structural tests have been completed. The fuel assembly structural test results are factored into the seismic and blowdown analyses (Reference 2).

1.5.1.4 **Guide Tube Tests**

Test Purpose and Parameters

A new guide tube was designed to accommodate the 24-rodlet pattern adopted for the 17 x 17 cores. This guide tube is sufficiently strong to provide increased margins of safety over present guide tubes. A high degree of interchangeability of parts has been designed into the guide tube. The main features of the design are full-length enclosures and cylindrical upper guide tubes. The 17 x 17 rodlet pattern reduced the central area available for driveline passage significantly, thus necessitating a generally tighter design of the rod guidance elements.

The following guide tube tests are considered engineering tests:

1. Engineering prototype assembly tests.
2. The guide tube drop and deflection test.

These tests are used as design tools and are not specifically required for the demonstration of plant safety.

The engineering prototype assembly tests are described below.

Engineering Prototype Assembly Tests. The purpose of these tests was to demonstrate that the 17 x 17 fuel assembly and driveline hardware designs perform as predicted. The tests were run before the required plant functional tests and are used as engineering information tests to obtain experimental data. A single set of driveline hardware, including control rods, was used in the tests. The fuel assemblies and driveline were subjected to flow and system conditions covering those mostly likely to occur in a plant during normal operations or during a pump overspeed transient.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

These tests were used to verify, from an engineering confidence standpoint, the integrated fuel assembly and rod cluster control performance in several areas. Data obtained included pressures and pressure drops throughout the system, hydraulic loadings on the fuel assembly and driveline, control-rod drop time and stall velocity, fuel-rod vibration, and control-rod, driveline, guide tube, and guide thimble wear during a lifetime of operation. None of this information is considered to be safety related.

Specifically, two full-size 17 x 17 fuel assemblies (one for each phase of testing) and one control rod, drive shaft, and control-rod drive mechanism were installed and tested in the 24-inch-diameter by 40-foot-high D-loop at the Westinghouse Test Engineering Laboratory Facility.

Fuel Assembly Life Test (Phase I). The first fuel assembly was subjected to the maximum expected control-rod travel during one fuel assembly lifetime. The nominal test conditions were a flow velocity based on the design flow rate, a temperature of 585°F, and a pressure of 2000 psig. These conditions represent an extreme set of conditions.

Using a fully instrumented 17 x 17 prototype fuel assembly, guide tube, and rod cluster control drive assembly, the test conducted in the D-loop produced information on the following:

1. Mechanical integrity and performance.
2. Drop time.
3. Fuel-rod vibration.
4. Control-rod velocity.
5. Hydraulic lift force.
6. Guide thimble dashpot pressure.

Following this, the prototype fuel assembly underwent a complete post-test evaluation, and the guide tubes and driveline were inspected for any abnormal wear conditions. The purpose of this test was basically to determine the effect of the 17 x 17 fuel assembly and the control-rod configuration on Items 1 through 6 in Phase I and Items 1 and 2 in Phases II and III. The effect on control-rod drop due to a seismic disturbance is evaluated analytically.

The test procedures, conditions, and results for Phase I are described in Reference 3.

Guide Tube and Rod Cluster Control Life Test (Phases II and III). The second fuel assembly was then installed to continue the test at the same flow and temperature until 3 million total steps of the driveline were accumulated. For Phases II and III, the testing was run at temperatures between 250°F and 585°F and at flow rates from 110 to 150% of the design flow rate.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

The test included a program of control-rod drops and mechanism stepping that approximated the driveline duty for the design lifetime of an operating plant. Specifically, in Phase II, approximately 1,275,000 mechanism steps and approximately 170 control-rod drops were accumulated. The components were then inspected. Following inspection, the testing was continued until, at the end of Phase III testing, a total of over 3 x 10^6 mechanism steps and approximately 500 control-rod drops were accumulated.

These tests were directed toward:

1. Lifetime wear evaluation.
2. Drop time.
3. Rod stall characteristics.

These were not safety-related tests; they were used as engineering tools. Final verification of the fuel assembly and control rod hardware designs was demonstrated during the rod drop tests performed after the initial loading of the core.

At the completion of the Phase II tests, the test assembly was inspected to determine guide tube and driveline wear characteristics. This inspection was repeated at the end of the test (Phase III).

Facility

The above testing was conducted in the Westinghouse Test Engineering Laboratory Facility (see Section 1.5.3).

Status

The D-loop testing has been completed. The results of the testing are given in References 3 and 5.

1.5.1.5 **Departure from Nucleate Boiling**

Test Purpose and Parameters

The effect of the 17 x 17 fuel assembly geometry on the departure from nucleate boiling (DNB) heat flux has been determined experimentally and has been incorporated into a modified spacer factor for use with the W-3 correlation. The effect of cold-wall thimble cells in the 17 x 17 geometry has also been quantified.

A similar program was conducted to quantify the DNB performance of the R-type mixing vane grid as developed for the 15 x 15 fuel assembly design (References 6 & 7). The results of that program were used to develop a modified spacer factor that quantifies the power capability associated with the use of the R mixing vane grid as well as the change in power capability due to the axial spacing of the grids. The modified spacer factor, along with the W-3 correlation with the cold-wall factor, was shown to be applicable to cold-wall thimble cells in the 15 x 15 geometry (Reference 7).

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

The experimental program consisted of three test series employing rod bundles that are representative of the 17 x 17 fuel assembly geometry. Two of the tests employed all heated rods, one test section being 8 feet long and the other being 14 feet long. The third test had one simulated cold-wall thimble tube. All three tests employed a uniform axial heat flux. The applicability of DNB data obtained using a uniform heat flux to a nonuniform heat flux has been well established by the use of an axial flux shape factor. Tong (Reference 8) first developed the form of the factor. This same form with some minor change in the empirical constants has been confirmed by Wilson (Reference 9). This method of analysis has proven correct for nonuniform rod bundle data, as shown by Rosal (Reference 10), Motley (Reference 6), and Wilson (Reference 9).

The concern over uniform versus nonuniform axial heat flux in long bundles is addressed in Reference 11, where comparative data on bundles of 0.422-inch-o.d. rods are presented. This provided a suitable basis for 17 x 17 DNB evaluation for all axial heat fluxes.

The final 17 x 17 fuel assembly design incorporated an additional grid and had a grid spacing less than 22 inches. An additional test program was conducted to provide data applicable to the 17 x 17 fuel assembly with 22-inch axial grid spacing (Reference 17). Four geometries were tested to vary axial heat flux shape (uniform vs. chopped cosine), heated length (8 vs. 14 feet), and cell type (typical vs. thimble coldwall), over a wide range of inlet fluid parameters. The results demonstrated the applicability of the R grid CHF correlation multiplied by a factor of 0.88 in predicting CHF for uniform and non-uniform axial heat flux. The results also verified the use of the F-factor for non-uniform data, the heated length effect incorporated into the modified spacer factor, and the cold wall factor.

Facility

These tests were conducted in the high-temperature and high-pressure loop that was constructed by Westinghouse at the Columbia University Heat Transfer Laboratories. The loop characteristics of this facility are as follows:

Flow rate	400 gpm maximum
	40 gpm minimum
Working pressure	3500 psia maximum
Test section inlet temperature	650°F maximum
Test section outlet temperature	700°F maximum
Test section heated length	16 ft maximum
Power input to test section	7.5 MWe maximum

The 17 x 17 DNB tests were performed parametrically for various combinations of inlet temperature and flow rate by increasing the bundle power incrementally until a DNB occurs.

Status

The original DNB test program is complete and the results are reported in Reference 11.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

1.5.1.6 **Incore Flow Mixing**

Test Purpose and Parameters

In the thermal-hydraulic design of a reactor core, the effect of mixing or turbulent energy transfer within the hot assembly was evaluated using the THINC code. The rate of turbulent energy transfer is formulated in the THINC analysis in terms of a thermal diffusion coefficient (TDC).

A program (Reference 4) to determine the proper value of the TDC for the R grid vane, as used in the 15 x 15 fuel assembly design, has been completed, and the results show that a design value of 0.038 (for 26-inch spacing) can be used for the TDC. The results also show that the TDC is independent of Reynold’s number, mass velocity, pressure, and quality over the ranges tested.

A test program was conducted to determine the effects of the 17 x 17 fuel assembly geometry on mixing and to determine an appropriate value for the TDC (Reference 13). A uniform axial heat flux was used. There was no analytical reason to expect that the mixing coefficient would be affected by a nonuniform axial heat flux. The THINC computer code considers the mixing in each increment along the heated length, and within that increment the heat flux is considered uniform. The tests reported by Cadek (Reference 12) indicate that there is no difference, within the limits of experimental accuracy, between a test section with a uniform flux (Pitt) and one with one-half of a cosine flux (Columbia). The heat flux in the test program for the 17 x 17 fuel assembly geometry was varied between the simulated fuel rods in the test section to create a thermal gradient in the radial direction (Reference 13). Using different flow rates and inlet temperatures, the TDC for the 17 x 17 geometry was determined.

Facility

These tests were conducted at the Columbia University Heat Transfer Laboratories.

Status

The TDC tests are complete and the results are reported in Reference 13.

1.5.2 LOCA Heat Transfer Tests (17 x 17)

1.5.2.1 **Reflood Heat Transfer Tests**

Extensive experimental programs have been performed with a simulated 17 x 17 assembly to determine its behavior under LOCA conditions. The 17 x 17 tests were conducted in the G-loop facility at the Westinghouse Forest Hills Laboratory.

Results form the 17 x 17 programs were compared with data from the 15 x 15 assembly test programs and were used to confirm predictions made by correlations and codes using the 15 x 15 test results (see Reference 16).

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

1.5.2.2 **Delayed Departure from Nucleate Boiling Testing**

1.5.2.2.1 Introduction

The NRC acceptance criteria for emergency core cooling systems for light water powered reactors were issued in Section 50.46 of 10 CFR 50 on December 28, 1973. They define the basis and conservative assumptions to be used in the evaluation of the performance of emergency core cooling systems. Westinghouse believes that some of the conservatism of the criteria is associated with the manner in which transient DNB phenomena are treated in the evaluation models. Transient critical heat flux data presented at the 1972 specialists' meeting of the Committee on Reactor Safety Technology indicated that the time to a DNB can be delayed under transient conditions. To demonstrate the conservatism of the models for evaluating emergency core cooling systems, Westinghouse initiated a program to experimentally simulate the blowdown phase of a loss-of-coolant accident (LOCA). This testing was part of the Electric Power Research Institute's Blowdown Heat Transfer Program, which was started early in 1976. The testing was completed confirming model conservatism and Westinghouse did not change the LBLOCA methodology.

1.5.2.2.2 Objective

The objective of the delayed departure from nucleate boiling (DDNB) test was to determine the time that a DNB occurs under LOCA conditions. This information would be used to confirm the existing Westinghouse transient DNB correlation or to develop a new one. The steady-state DNB data obtained from 15 x 15 and 17 x 17 test programs can be used to ensure that the geometrical differences between the two fuel arrays can be correctly treated in the transient correlations.

1.5.2.2.3 Program

The program was divided into two phases. The Phase I tests started from steady-state conditions; sufficient power was available to maintain nucleate boiling throughout the bundle. Controlled ramps of decreasing test section pressure or flow initiated a DNB. Through the application of a series of controlled conditions, the DNB was studied over a range of qualities and flows, and at pressures relevant to a PWR blowdown.

Typical parameters used for Phase I testing are shown in Table 1.5-1.

Phase I provided separate-effects data to permit heat transfer correlation development.

Phase II simulated PWR behavior during a LOCA and thus permitted the definition of the time delay associated with the onset of a DNB. Tests in this phase cover the large double-ended guillotine cold-leg break. All tests in Phase II were started after establishment of typical steady-state operating conditions. The fluid transient was then initiated, and the rod power decay was programmed in such a manner as to simulate the actual heat input of fuel rods. The test was terminated when the heater rod temperatures reached a predetermined limit.

Typical parameters used for Phase II testing are shown in Table 1.5-2.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

1.5.2.2.4 Test Description

The experimental program was conducted in the J-loop at the Westinghouse Forest Hills Facility. A full-length 5 x 5 rod bundle simulated one section of a 15 x 15 assembly subjected to a DNB under LOCA conditions.

The heater rod bundle used in this program consisted of internally heated rods capable of a maximum power of 18.8 kw/ft, with a total power of 136 kW (for extended periods), over the 12-foot heated length of the rod. Heat is generated internally by means of a varying cross-sectional resistor that approximates a chopped cosine power distribution. The rods were adequately instrumented, each having a total of 12 clad thermocouples.

1.5.2.2.5 Results

The experiments in the DDNB Facility resulted in cladding temperature and fluid properties measured as a function of time throughout a blowdown range of 0 to 20 seconds.

Facility modifications and the installation of the initial test bundle have been completed. A series of shakedown tests in the J-loop have been performed. These tests provided data for instrumentation calibration and checkout, as well as information regarding facility control and performance. Initial program tests were performed during the first half of 1975. Under the sponsorship of the Electric Power Research Institute, testing was reinitiated during 1976 on the same test bundle. The testing was terminated in November 1976, and a new test bundle was installed prior to further testing during 1977-1978. A DNB correlation was developed from the test results and compared to the use of the steady state correlations. Since only a minimal improvement was noted, the Westinghouse LBLOCA methodology continued to use the steady state DNB correlations.

1.5.2.3 **Single-Rod Burst Test**

The single-rod burst test results were used to quantify the maximum assembly flow blockage that is assumed in LOCA analyses.

The 15 x 15 fuel assembly rods have already been subjected to single-rod and multi-rod burst tests under LOCA conditions. The results of these tests indicated that fuel rods burst in a staggered manner, so that maximum average assembly-wide flow area blockage is 55% during blowdown and 65% during reflood, based on the characteristics of the PWR rod and the conservative peak clad temperature predicted for the period of the LOCA transient.

The single-rod burst test program for the 17 x 17 fuel assembly rods involved bursting specimens at various internal pressures and heating rates in a steam atmosphere.

In addition, tests were run on 15 x 15 fuel assembly rods to ensure the reproducibility of the 1972 single-rod burst test results. Results of the program are documented in References 14 and 15.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

The single-rod burst tests and evaluation have been completed and are reported in References 14 and 15. Results of the tests show that the 17 x 17 and 15 x 15 clads do not differ significantly in failure ductility under LOCA conditions. Because of the results and the geometric scaling, the flow blockage (percentage) as determined in the 15 x 15 multi-rod burst test simulation can be used for the 17 x 17 fuel geometry.

1.5.3 Westinghouse Test Engineering Laboratory Facility

1.5.3.1 Introduction

The Test Engineering Laboratory at Forest Hills, Pennsylvania, has long been the major Westinghouse center for nuclear research and development. The Test Engineering Laboratory is totally involved with the design and implementation of facilities and programs to prove the reliability of Westinghouse PWR concepts and components.

The laboratory has full in-house capabilities to design and construct PWR loops for both hydraulic and heat transfer testing programs. The most vital current project is the analysis of emergency core cooling systems by means of scale-model tests conducted on three separate facilities.

The G-loop, a test vessel that contains a bundle of 480 heater rods, is the largest such test facility in the world. The G-loop has a steam supply to provide the proper environment during system blowdown, as well as the capability to test high-pressure and low-pressure emergency core cooling systems. The G-loop operates at pressures up to 2000 psi and temperatures up to 650°F. It is designed to start operation at 8 seconds after a LOCA and is capable of investigating the upper-head injection and other spray systems of the current emergency core cooling system.

The J-loop is a test vessel containing an array of 25 heater rods, a broken-loop simulation, and an unbroken-loop simulation. The loop is designed to operate at 2500 psi and 650°F, and is capable of simulating the first 20 seconds of a LOCA with primary emphasis on a DDNB.

FLECHT-SET, a test vessel containing an array of 100 heater rods and thimbles, was used to investigate the reflood phase of the emergency core cooling system, plant system effects being measured with scaled piping and two scale-model steam generators. The facility is designed to operate at up to 100 psia.

Five general-purpose hydraulic loops were also involved in the development of improved water reactor components, as well as the reliability testing of current and prototype PWR components.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Historically, the services of the Test Engineering Laboratory have reflected the prevailing need. At the time of the North Anna FSAR submittal it was needed for the development of data on emergency core cooling systems and for the verification of many new PWR system components. Other past needs and accomplishments have included the development of supercritical heat transfer once-through loops, rod cluster control drive mechanisms, fuel assemblies, underwater handling tools, and fuel assembly grid designs. Testing at the laboratory has included air filter tests, water chemistry tests, in-pile tests of fuel rods, single fuel-rod burst tests, studies of fuel assembly hydraulics, and corrosion tests of Zircaloy and other PWR components and materials, with and without heat transfer.

The Test Engineering Laboratory is a very flexible installation, one that will continue to expand and develop as future needs for its services arise. Its staff, too, varies according to requirements. At the time of the North Anna FSAR submittal, there were more than 100 persons involved in laboratory projects, including 12 electrical and mechanical engineers, more than 75 highly skilled technicians, and some 30 specialists from other divisions of Westinghouse. The laboratory had the option of obtaining personnel from other divisions of the corporation, depending on the need for specific skills, knowledge, and experience.

Ongoing research performed at the Test Engineering Laboratory continues to demonstrate the reliability of Westinghouse PWR plant components and greatly facilitates the development of improved reactor system components. As the test center for Westinghouse Nuclear Energy Systems, the laboratory is totally committed to the advancement of the nuclear energy industry.

1.5.3.2 **Test Loops and Equipment**

This section contains a brief description of the major test loops and test equipment at the Westinghouse Test Engineering Laboratory Facility.

1.5.3.2.1 A- and B-Loops: Low-Flow/High-Pressure Hydraulic Facilities

These loops are small, high-pressure, stainless steel facilities used for testing small components and individual parts of larger components under normal working conditions. A canned motor pump circulates water in both the A- and B-loops at 150 gpm. Operating temperatures are obtained from the conversion of the pumping power into heat, as well as from external heaters. Typical tests run in these loops include (1) full-scale gate and check valve tests, (2) material corrosion-erosion tests, with variable water chemistry, and (3) tests of corrosion product release and the transport properties of crud. The following are the characteristics of the A- and B-loops:

Maximum flow rate at 300 ft	150 gpm
Maximum pump head at 60 gpm	335 ft
Maximum allowable temperature	650°F
Normal working pressure	2000 psi
Normal working temperature	600°F

1.5.3.2.2 D-Loop: Medium-Flow/High-Pressure Hydraulic Facility

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

The D-Loop is a flexible test facility used for demonstrating the interplay of reactor subsystems and evaluating component design concepts. It contains a canned-motor pump, which produced a 290-foot head at 3000 gpm. All piping (10-inch Schedule 160) in contact with the primary water is stainless steel. The loop pressure is established and maintained by an air-driven charging pump operating in conjunction with a gas-loaded backpressure valve. Most of the power required to establish and maintain loop temperature is derived from the circulating pump operation, and 75 kW of heat is available from electric strip heaters.

The D-loop services a 24-inch-i.d. x 40-foot-long test vessel, which accommodates full-scale models of large PWR core components for operational studies. The characteristics of the D-loop are as follows:

Maximum flow rate	4400 gpm
Maximum allowable pressure	2400 psi
Maximum allowable temperature	650°F
Normal working pressure	2000 psi
Normal working temperature	600°F
Pump head at 3000 gpm	290 ft
Maximum pump head at 1500 gpm	340 ft
Main loop flow measurement	10-in. venturi
Auxiliary flow measurements	6-in. venturis (2-in. branch lines)

1.5.3.2.3 E-Loop: Low-Flow/Low-Pressure Hydraulic Facility

The E-loop is a low-pressure, 6-inch, stainless steel loop, with two circulating pumps. These pumps may be connected in parallel, giving 2000 gpm at 130-foot head, or in series, giving 1000 gpm at 260-foot head. Flow and vibration studies are conducted with this loop, and, because of its low pressure, plastic models for visual observation or photography may be used. In addition, a 4-inch Rockwell water meter in a branch line permits the calibration of flowmeters up to 800 gpm. The characteristics of the E-loop are as follows:

Maximum flow rate, gpm	
At 130 ft	2000
At 360 ft	1000
Maximum working pressure	Pump head

1.5.3.2.4 G-Loop: Emergency Core Cooling System Facility

The G-loop is a high-pressure emergency core cooling system test facility designed and fabricated for 2000 psi and 650°F in accordance with ASME Section I. It consists of a main test section and vessel, an exhaust system, piping, separators and a muffler, a flash chamber steam supply system, and high-pressure/low-pressure cooling systems.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

This loop is basically designed to obtain test data for the analysis of LOCA-related breaks up to and including double-ended pipe breaks. Tests are initiated at simulated conditions existing 8 seconds after the start of a LOCA. A typical run consists of constant power and pressure, followed by pressure blowdown, power decay, and emergency core cooling system activation.

The G-loop is capable of performing the following methods of emergency core cooling system: current, upper-head injection, upper-head injection with current, and other core spray systems. It may also be used for constant temperature/pressure small-leg break tests (core-uncovering tests). These tests involve boiling off water at a constant bundle power input until the rods can no longer be cooled.

The G-loop test bundle consists of 480 electrically heated rods, 16 grid support thimbles, and 33 spray thimbles, all bounded by an octagonal stainless steel baffle and arranged in a 4-loop, 15 x 15 rod bundle configuration. The loop is controlled (fully automated during transients) through a PDP-II-DEC-16k computer with a 600-point Computer Products Analog-to-Digital converter operating at a sweep rate of 40,000 points per second for data acquisition. Figure 1.5-1 is a schematic of the G-loop emergency core cooling system test facility. The G-loop system components and characteristics are as follows:

		Rated Conditions		Typical Operating Conditions	
Component	Material	Pressure (psi)	Temperature (°F)	Pressure (psi)	Temperature (°F)
Test vessel	Carbon steel	2000	650	1000	545
Downcomer side tank	Carbon steel	2000	650	1000	545
In-line mixer	Carbon steel	2000	650	1000	545
Mixer-accumulator	Stainless steel	2500	650	1800	100
Flash chamber	Carbon steel	3000	700	2800	660
Separators 1 and 2	Carbon steel	2000	650	1000	545
Spray accumulators 1 and 2	Carbon steel	2000	650	1800	150
Spray accumulator 3	Stainless steel	2500	650	1800	150
Reflood tank	Stainless steel	Atmospheric	212	Atmospheric	150
Primary piping	Carbon steel	2000	650	1000	545

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

1.5.3.2.5 H-Loop: High-Flow Hydraulic Facility

The H-loop is a versatile hydraulic facility that is capable of supplying 14,000 gpm of water at a developed head of 600 feet and at temperatures as high as 200°F. This 4-loop system can simultaneously handle either full-scale prototype test assemblies or one large-scale reactor model. The major purpose of H-loop is to permit the use of 1/7-scale reactor models and full-scale fuel assemblies for conducting mixing studies, flow distribution studies, and similar low-temperature/low-pressure hydraulic tests. The characteristics of H-loop are as follows:

Maximum flow rate	14,000 gpm
Pressure drop across vessel model	120 psi
Minimum vessel outlet pressure	10 psig
Flow accuracy	0.5%
Water temperature range	70-200°F
Maximum loop-to-loop temperature variation	2°F
Maximum loop-to-loop flow rate variation	3%

1.5.3.2.6 J-Loop: Delayed Departure from Nucleate Boiling (DDNB) Heat Transfer Facility

The J-loop is a completely instrumented pressurized-water test facility for verifying DDNB phenomena during a LOCA and for conducting steady-state heat transfer studies. This test loop is a full-size, single-loop simulation of a typical 4-loop reactor system; it will accept a full-length 5 x 5 bundle of internally heated "fuel rods." The J-loop is designed to operate at 2500 psia at 650°F, and at variable flow rates up to 450 gpm.

During LOCA tests, fluid input to the "reactor vessel" is closely controlled by two servo-controlled mixers, which inject a two-phase water/steam mixture into the test vessel to simulate flow from the unbroken loops. Figure 1.5-2 is a schematic of the J-loop test facility. The characteristics of J-loop are as follows:

Test fluid	Demineralized water
Design pressure	2500 psia
Design temperature	650°F
Maximum flow rate (hot)	450 gpm
Power input to test vessel (maximum)	3,500,000W
Primary test heat exchanger rating	11,400,000 Btu/hr

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

1.5.3.2.7 K-Loop: Boron Thermal Regeneration Test

The K-loop, the boron thermal regeneration system test facility, is used to study the performance and to verify the component sizing of both the currently available THERM I and the improved THERM II boron thermal regeneration systems. The function of this system is to process boron-containing effluents from the reactor coolant system to yield a high-boron-concentration fraction, which can be used to borate the reactor coolant system. It also processes a relatively boron-free fraction, such as that required in load-follow operations, which can be used to dilute the reactor coolant system. The characteristics of K-loop are as follows:

Total tank capacity	30,000 gal
Chiller capacity, of ice	48 tons
Maximum ion exchange resin test volume	75 ft^3
Maximum test process rate capability, bed area	10 gpm/ft^2
Maximum flow test capability	200 gpm
Minimum boron storage mode fluid temperature	50°F
Maximum boron release mode fluid temperature	160°F

1.5.3.2.8 FLECHT-SET: Emergency Core Cooling System Facility

The FLECHT-SET is a low-pressure facility that is designed to provide experimental data on the influence of system effects on the emergency core cooling system during the reflood phase of a LOCA.

The facility consists of a once-through system, including an electrically heated test section ("fuel rods" and housing), an accumulator, steam generator simulators, a pressurizer, catch vessels, instrumentation, and the piping necessary to simulate the reactor primary coolant loop. Data acquisition is accomplished through a PDP-II-DEC-16K computer with a 256-point Computer Products A-D converter, operating at a sweep rate of 1200 points per second. The FLECHT-SET has the following characteristics:

100-rod bundle maximum power	1000 kW
Maximum bundle flooding rate	86 gpm
Water temperature range	100-200°F
System pressure	0-60 psia

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

1.5.3.2.9 Single-Rod Loop: Heater Rod Development Facility

The single-rod test loop is used for the evaluation of prototype heater rods and for the in-depth study of existing rods in pressurized-water systems. The test section of the loop is easily replaced to facilitate the installation of heater rods of varying length and diameter. The single-rod loop is electrically controlled and operated by one person. Steady-state conditions and blowdown at various conditions can be simulated in the loop. The main test section can be replaced with a quartz tube, and the DNB phenomenon can be observed on a single rod with a remotely operated camera. The characteristics of the single-rod test loop are as follows:

Maximum operating pressure	2250 psia
Maximum operating temperature	650°F
Maximum flow rate	10 gpm
System capacity	5 gal
Maximum power available	200 kW
Piping size	1 and 3 in.

1.5.3.2.10 Hydraulic Model Testing

Miscellaneous hydraulic tests on mock-ups of reactor system parts and components are routinely performed at the test engineering laboratory. Typical of this type of testing are the two discussed below, which have been completed.

Emergency Core Cooling Flow Distribution

As shown above, a 10 x 10 rod bundle was installed in a plastic housing with a water supply at the top. A grid collection unit at the bottom of the bundle collected the water as it flowed through the model and diverted it to the measuring tubes at the base. Knowledge of the flow distribution in the bundle was obtained in this manner.

Sample System Mixing Test

This test used one thermocouple to measure the temperature of water from four locations in a reactor. The purpose of the procedure was to determine whether the indication from the single thermocouple was representative of the average temperature of the four water supplies. A mock-up of the mixing chamber was constructed so that hot or cold water—under closely controlled pressure—could be supplied to any of the four inlets. By running combinations of hot and cold inlets and making simultaneous recordings of the various temperatures, highly useful information was obtained.

1.5.3.2.11 Autoclave Testing

The testing engineering laboratory is equipped with autoclaves ranging in size from 0.5 to 100 gallons. These devices are in constant use to determine the effects of various water chemistries on core components, and to perform corrosion tests. The units have also been used as boilers to provide steam for miscellaneous development tests, including acoustic leak detection.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

1.5.3.2.12 Mechanical Component and Vibration Tests

Full-scale mechanical and vibration tests of plant and reactor components are performed at the test engineering laboratory to prove the reliability of equipment design. Vibration testing of reactor components is also performed in this laboratory, using electronically excited shaker heads. Three sizes are available (2, 50 and 150 lb) for regular scale model testing at frequencies from 5 to 50 Hz.

1.5.3.2.13 Electronic Component Assembly

Highly skilled technicians are available at the test engineering laboratory to construct complex control and instrumentation systems. The work, which is initiated with engineering ideas and sketches, includes the mounting of process controllers, recorders, meters, relay logic, protection circuits, switches, and indicators.

Point-to-point wiring or printed circuit boards are used, as required. Final "as-built" drawings are prepared, and an inspection and a thorough electrical checkout are performed before installation.

1.5.3.2.14 Surveillance System Development

Surveillance systems are used for the online monitoring of pressure vessels for flaws. Under development at the test engineering laboratory are electronic components for an acoustic emission monitoring system for the inservice inspection of operating plant vessels and piping. This system is designed to detect the initiation and propagation of cracks at welds, stress risers, and other locations. Vessel flaw growth and rupture data have been obtained through joint programs at the National Reactor Testing Station in Idaho and the Oak Ridge National Laboratory. Pipe rupture data have been obtained from NRC-sponsored tests, and hydrostatic test data have been collected and operational noise and attenuation characteristics have been measured at various operating Westinghouse plants.

1.5.3.2.15 Engineering Mechanics Laboratory

Bench tests are performed in fixtures designed for the particular test using standard test equipment and techniques.

1.5 REFERENCES

1. F. T. Eggleston, *Safety Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries*, WCAP-8768, Revision 1, 1976.
2. L. Gesinski, D. Chiang, and S. Nakazato, *Safety Analysis of the 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident*, WCAP-8288, 1973.
3. E. E. De Mario and S. Nakazato, *Hydraulic Flow Test of the 17 x 17 Fuel Assembly*, WCAP-8279, 1974.
4. F. F. Cadek, F. E. Motley, and D. P. Dominicis, *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), Westinghouse Nuclear Energy Systems Proprietary, 1972, and WCAP-7959-A, January 1975.
5. F. W. Cooper, Jr., *17 x 17 Driveline Components Tests - Phase 1B, II, III D-Loop Drop and Deflection*, WCAP-8446, proprietary, and WCAP-8449, non proprietary, 1974.
6. F. F. Cadek and F. E. Motley, *DNB Results for New Mixing Vane Grid (R),* WCAP-7695-P-A (Proprietary) and WCAP-7958-A (Non-proprietary), Westinghouse Nuclear Energy Systems, January 1975.
7. F. F. Cadek and F. E. Motley, *DNB Test Results for R Grids with Thimble Cold Wall Cells*, WCAP-7695, Addendum 1-P-A (Proprietary) and WCAP-7958, Addendum 1-A (Non-proprietary), Westinghouse Nuclear Energy Systems, January 1975.
8. L. S. Tong, *Prediction of Departure From Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution*, *Journal of Nuclear Energy*, Vol. 21, PP-241-248, 1967.
9. R. H. Wilson, L. J. Stanek, J. S. Gellerstedt, and R. A. Lee, *Critical Heat Flux in a Non-Uniformly Heated Rod Bundle*, in *Two-Phase Flow and Heat Transfer in Rod Bundles*, American Society of Mechanical Engineers, New York, 1969, pp. 56-62.
10. E. R. Rosal, et al., *Rod Bundle Axial Non-Uniform Heat Flux Tests and Data*, WCAP-7411, Revision 1-P-A (Proprietary) and WCAP-7813, Revision 1-A (Non-proprietary), Westinghouse Nuclear Energy Systems, January 1975.
11. K. W. Hill, F. E. Motley, F. F. Cadek, and A. H. Wenzel, *Effect of 17 x 17 Fuel Assembly Geometry on DNB*, WCAP-8296-P-A (Proprietary) and WCAP-8297-A (Non-proprietary), Westinghouse Nuclear Energy Systems, January 1975.
12. F. F. Cadek, *Interchannel Thermal Mixing with Mixing Vane Grids*, WCAP-7667-P-A (Proprietary) and WCAP-7755-A (Non-proprietary), Westinghouse Nuclear Energy Systems Proprietary, January 1975.
13. F. E. Motley, A. H. Wenzel, and F. F. Cadek, *The Effect of 17 x 17 Fuel Assembly Geometry on Interchannel Thermal Mixing*, WCAP-8298-P-A (Proprietary) and WCAP-8299-A (Non-proprietary), Westinghouse Nuclear Energy Systems Proprietary, January 1975.

14. *17 x 17 Design Fuel Rod Behavior During Simulated LOCA Conditions*, WCAP-8290, proprietary, and WCAP-8289, non proprietary, 1974.

15. *A Temperature Sensitivity Study of Single Rod Burst Tests*, WCAP-8290, Addendum 1, proprietary, and WCAP-8289, Addendum 1, non proprietary, 1975.

16. A. J. Burnett and S. C. Kopelic, *Westinghouse ECCS Evaluation Model - October 1975 Version*, WCAP-8622, proprietary, and WCAP-8623, non proprietary, 1975.

17. F. E. Motley, A. H. Wenzel and F. F. Cadek, WCAP-8536 (Proprietary) and WCAP-8537 (Non-proprietary), *Critical Heat Flux Testing of 17 x 17 Fuel Assembly Geometry with 22-inch Grid Spacing,* Westinghouse Nuclear Energy Systems, May 1975.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 1.5-1

DDNB PHASE I TEST PARAMETERS

Parameters	Nominal Value
Initial Steady-State Conditions	
Pressure	1750 to 2250 psia
Test section mass velocity	2.3 to 2.5 x 10^6 lb/hr-ft^2
Core inlet temperature	560 to 600°F
Maximum heat flux	306,000 to 531,000 Btu/hr-ft^2
Transient Ramp Conditions	
Pressure decrease	0 to 350 psia and subcooled depressurization psi/sec from 2250
Flow decrease,	0 to 100%/sec
Inlet enthalpy	Constant

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

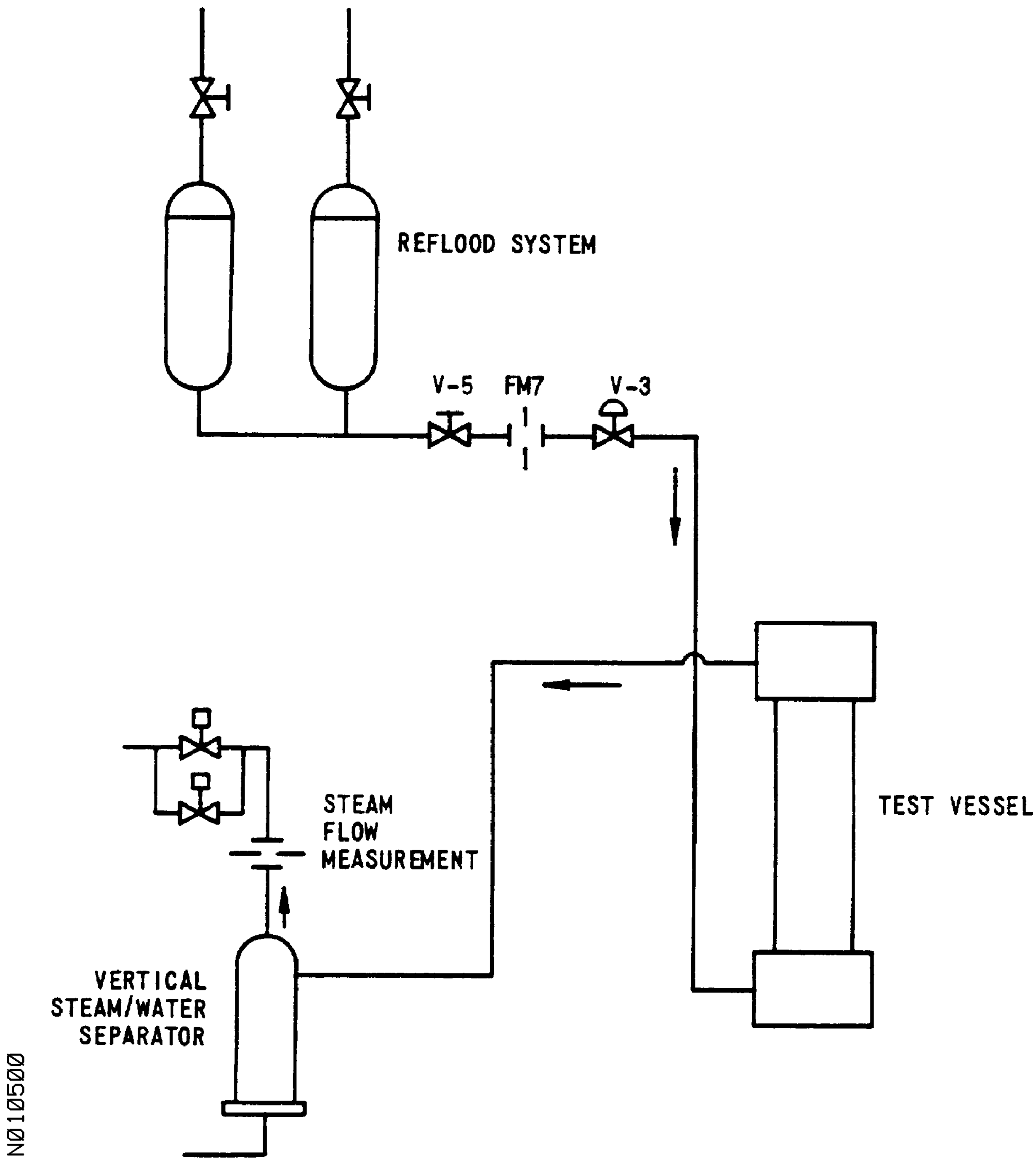
Table 1.5-2

DDNB PHASE II TEST PARAMETERS

Parameters	Nominal Value
Initial Steady-State Conditions	
Pressure	2250 psia
Test section mass velocity	2.5 x 10^6 lb/hr-ft^2
Inlet coolant temperature	560°F
Maximum heat flux	531,000 Btu/hr-ft^2
Transient Conditions	
Simulated break	Double-ended cold-leg guillotine breaks

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Figure 1.5-1
SCHEMATIC OF 17 X 17 REFLOOD TEST FACILITY



The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

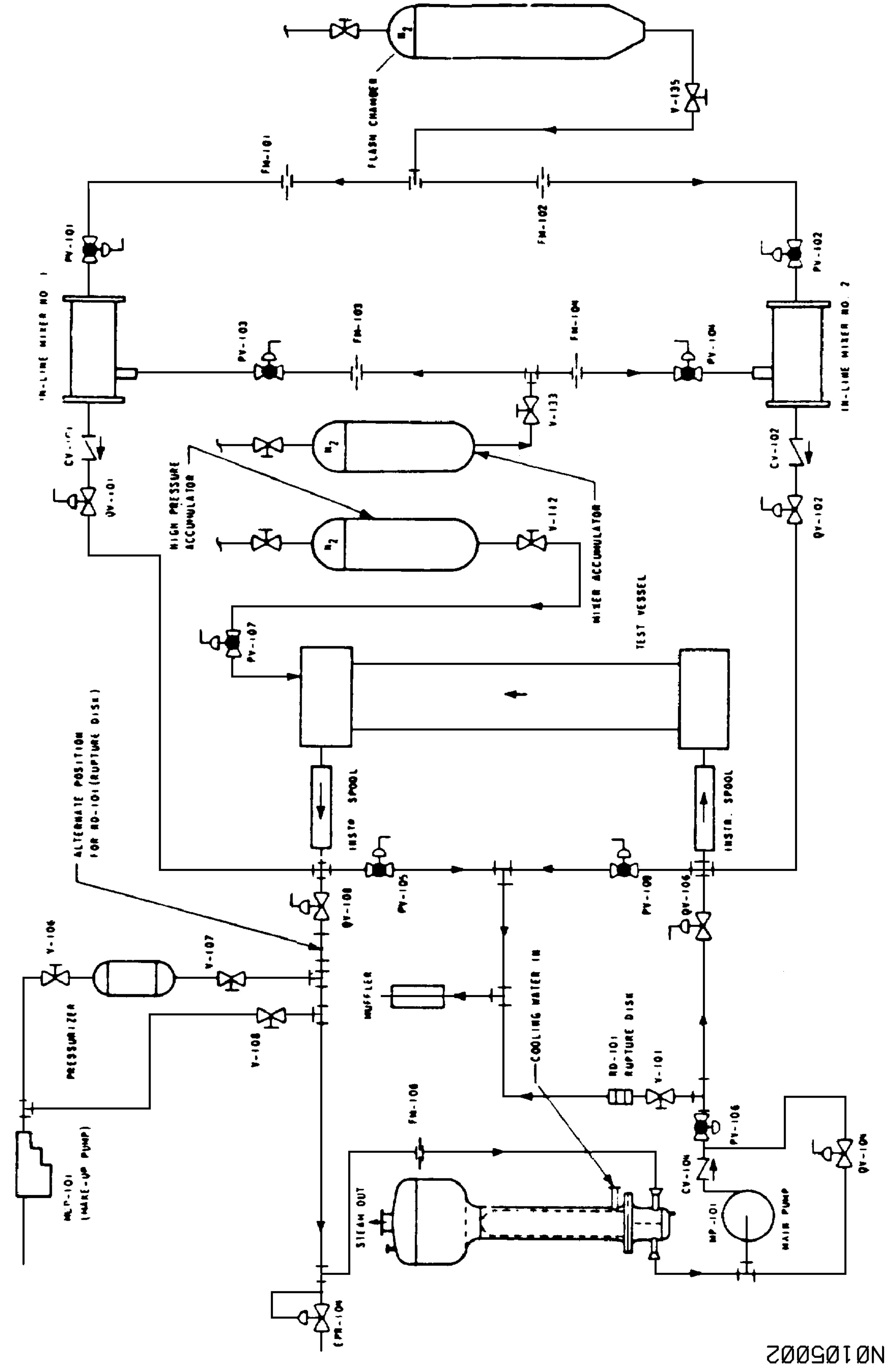


Figure 1.5-2
DNB TEST FACILITY SCHEMATIC

Intentionally Blank

1.6 GENERAL REFERENCES—WESTINGHOUSE TOPICAL REPORTS

Number	Report	Section
1.	R. F. Barry et al., *Topical Report - Power Distribution Monitoring in the R. E. Ginna PWR*, WCAP-7756, 1971.	3.9
2.	R. F. Barry and S. Altomare, *The TURTLE 24.0 Diffusion Depletion Code*, WCAP-7213, Westinghouse Nuclear Energy Systems Proprietary, 1968, WCAP-7758, 1971.	4.3, 15.2, 15.3
3.	R. F. Barry, *LEOPARD, a Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094*, WCAP-3269-26, 1963.	4.3, 15.3, 15.4
4.	R. Bartholomew and J. Lipchak, *Test Report, Nuclear Instrumentation System Isolation Amplifier*, WCAP-7819, Revision 1, 1972.	7.2
5.	A. E. Blanchard and D. N. Katz, *Solid State Rod Control System, Full Length*, WCAP-7778, 1971.	7.7
6.	A. E. Blanchard, *Rod Position Monitoring*, WCAP-7571, 1972.	7.7
7.	A. E. Blanchard, *Part Length Rod Control System*, WCAP-7406, 1971.	7.7
8.	G. J. Bohm, *Indian Point Unit No. 2 Internals Mechanical Analysis for Blowdown Excitation*, WCAP-7822, 1971.	3.9
9.	F. M. Bordelon, *A Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant (SATAN 4 Digital Code)*, WCAP-7750, 1971.	3.7, 6.2
10.	F. M. Bordelon, *Calculation of Flow Coastdown After Loss of Reactor Coolant Pump (PHOENIX Code)*, WCAP-7973, 1973.	15.2
11.	T. W. T. Burnett, *Reactor Protection System Diversity in Westinghouse Pressurized Water Reactor*, WCAP-7306, 1969.	7.2, 15.4
12.	T. W. T. Burnett, C. J. McIntyre, J. C. Buker, ant R. P. Rose. *LOFTRAN Code Description*, WCAP-7907, 1972.	15.1, 15.2, 15.3, 15.4
13.	F. F. Cadek, *Interchannel Thermal Mixing with Mixing Vane Grids*, WCAP-7755, 1971.	4.4
14.	K. Cooper, R. Starek, and V. Miselis, *Over-Pressure Protection for Westinghouse PWR*, WCAP-7769, Revision 1, 1972.	5.2, 15.2

Number	Report	Section
15.	S. Fabic, *Loss-of-Coolant Analysis: Comparison Between BLODWN-2 Code Results and Test Data*, WCAP-7401, 1969.	3.9
16.	S. Fabic, *Description of the BLODWN-2 Computer Code*, QCAP-7918, Revision 1, 1970.	3.9
17.	D. B. Fairbrother and H. G. Hargrove, *WIT-6 Reactor Transient Analysis Computer Program Description*, WCAP-7980, 1972.	15.2
18.	W. C. Gangloff, *An Evaluation of Anticipated Operational Transients in Westinghouse Pressurized Water Reactors*, WCAP-7486, 1971.	7.2, 15.2
19.	I. Garber, *Test Report of Isolation Amplifiers*, WCAP-7685, 1971.	7.2
20.	J. M. Geets and R. Salvatori. *Long Term Transient Analysis Program for PWR's (BLKOUT Code)*, WCAP-7898, 1972.	15.1
21.	J. M. Geets, *MARVEL - A Digital Computer Code for Transient Analysis of a Multiloop PWR System*, WCAP-7909, 1972.	15.2, 15.4
22.	L. T. Gesinski, *Fuel Assembly Safety Analysis for Combined Seismic and Loss-of-Coolant Accident*, WCAP-7950, 1972.	3.9
23.	W. S. Hazelton et al., *Basis for Heatup and Cooldown Limit Curves*, WCAP-7924, 1972.	Technical Specifications
24.	G. Hetsroni, *Hydraulic Tests of the San Onofre Reactor Model*, WCAP-3269-8, 1964.	4.4, 15.1
25.	L. E. Hochreiter, H. Chelemer, and P. T. Chu, *THINC-IV. An Improved Thermal-Hydraulic Analysis of Rod Bundle Cores*, WCAP-7956, 1972.	4.4
26.	C. Hunin, *FACTRAN, a Fortran IV Code for Thermal Transient in a* $U0_2$ *Fuel Rod*, WCAP-7908, 1972.	15.1, 15.2, 15.3, 15.4
27.	D. N. Katz, *Solid State Logic Protection System Description*, WCAP-7488-L, Westinghouse Nuclear Energy Systems Proprietary, 1971, and WCAP-7672, 1971.	7.1, 7.2, 7.3
28.	C. J. Kubit (ed.), *Safety Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries, Fall 1971-Spring 1972*, WCAP-7856, 1972.	4.2
29.	J. B. Lipchak and R. A. Stokes, *Nuclear Instrumentation System*, WCAP-7669, 1971.	7.2, 7.7

Number	Report	Section
30.	J. Locante and E. G. Igne, *Environmental Testing of Engineered Safety Features Related Equipment (NSSS - Standard Scope)*, WCAP-7744, Volume I, 1971.	7.3
31.	J. J. Loving, *In-Core Instrumentation (Flux- Mapping System and Thermocouples)*, WCAP-7607, 1971.	7.7
32.	A. F. McFarlane, *Power Peaking Factors*, WCAP-7912, 1972.	4.3, 4.4
33.	J. Molloy, E. A. Bassler, and W. P. Kaufmann, *Nuclear Fuel Division Reliability and Quality Assurance Program Plan*, WCAP-7800, 1971.	4.2
34.	J. S. Moore, *Westinghouse PWR Core Behavior Following a Loss-of-Coolant Accident*, WCAP-7422-L, Westinghouse Nuclear Energy Systems, 1970, and WCAP-7422, 1971.	3.9
35.	F E. Motley and P. F. Cadek, *DNB Test Results for New Mixing Vane Grids (R)*, WCAP-7695-L, W Proprietary Class II, 1971, and WCAP-7958, 1973.	4.4
36.	F. E. Motley and P. F. Cadek, *DNB Tests Results for R. Grid Thimble Cold Wall Cells*, WCAP-7695-L, Addendum I, W Proprietary Class II, and WCAP-7958, Addendum I, 1973.	4.4
37.	F. E. Motley and P. F. Cadek, *Application of Modified Spacer Factor to L Grid Typical and Cold Wall Cell DNB*, WCAP-8030, 1973.	4.4
38.	J. A. Nay, *Process Instrumentation for Westinghouse Nuclear Steam Supply Systems*, WCAP-7671, 1971.	5.2, 7.2, 7.3
39.	B. E. Olsen et al., *Indian Point No. 2 Primary Loop Vibration Test Program*, WCAP-7920.	3.9
40.	D. H. Risher Jr., *An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods*, WCAP-7588, Revision 1, 1971.	15.4
41.	D. H. Risher, Jr., and R. F. Barry, *TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code*, WCAP-7979, 1971.	15.4
42.	E. R. Rosal, J. O. Cermak, and L. S. Tong, *Rod Bundle Axial Non-Uniform Heat Flux DNB Tests and Data*, WCAP-7411-L, Revision 1, W Proprietary Class II, 1970, and WCAP-7813, 1971.	4.4

Number	Report	Section
43.	R. L. Rosenthal, *An Experimental Investigation of the Effect of Open Channel Flow on Thermal- Hydro-Dynamic Flow Instability*, WCAP-7240, Westinghouse Nuclear Energy Systems Proprietary, 1968, and WCAP-7966, 1972.	4.4
44.	J. J. Szyslowski and R. Salvatori, *Determination of Design Pipe Breaks for the Westinghouse Reactor Coolant System*, WCAP-7503, Revision 1, 1972.	5.2
45.	*Emergency Core Cooling Performance*, Westinghouse Electric Corporation, Westinghouse Nuclear Energy Systems Proprietary, 1971.	15.4
46.	*Indian Point Unit No. 2 Final Safety Analysis Report, Supplements 12 and 13*, AEC Docket Number 50-247, Consolidated Edison Company of New York.	15.4
47.	*Trojan Unit PSAR, Section 3.2.2*, AEC Docket Number 50-344, Portland General Electric Company.	4.4