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Duke Power Company

OCONEE NUCLEAR STATION

UNITS 1, 2 AND 3

Final Safety Analysis Report

Volume 1

DUKE POWER COMPANY

POWER BUILDING

422 SOUTH CHURCH STREET, CHARLOTTE, N. C.

A. C. THIES
VICE PRESIDENT
PRODUCTION AND OPERATION

P. O. BOX 2178
28201

May 29, 1969

Dr. Peter Morris, Director
Division of Reactor Licensing
Atomic Energy Commission
Washington, D.C. 20545

Re: Oconee Units 1, 2, 3
Dockets Nos. 50-269, -270 and -287

Dear Dr. Morris:

Duke Power Company is filing herewith Amendment No. 7 to its application for licenses for the Oconee Nuclear Station, which is under construction pursuant to provisional construction permits CPPR-33, -34, and -35 issued by the Commission on November 6, 1967. This filing includes three signed original copies of the Amendment and 70 additional copies of the Final Safety Analysis Report required by 10 C.F.R. § 50.34(b).

Sincerely,

A. C. Thies

A. C. Thies

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1. INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This Final Safety Analysis Report is submitted in support of Duke Power Company's application for a license to operate the three-unit Oconee Nuclear Station located on the shore of Lake Keowee in Oconee County, South Carolina. The station location is shown on Duke's Service Area Map, Figure 1-1.

The organization of this report is in accordance with the AEC's "Guide for the Organization and Contents of Safety Analysis Reports." Every attempt has been made to be responsive to the format and intent of that guide and to the intent as understood by Duke of the proposed AEC 70 design criteria. The latest revisions of 10CFR50 as published in the December 31, 1968 Federal Register have been considered in organizing this report.

Construction of Units 1, 2, and 3 was authorized by the United States Atomic Energy Commission by issuance of construction permits CPPR-33, 34 and 35, on November 6, 1967, in Dockets 50-269, 270 and 287. Construction of Unit No. 1 is scheduled for completion in time for fuel loading on January 1973 and
26. for commercial operation by May 1973 Commercial operation of Units 2 and 3 is scheduled for September 1973 and June 1974 respectively.

Since the issuance of the construction permit and during the design of the plant, there have been no major deviations from the intent of the PSAR. Changes in design features that have been found desirable during the design are covered in detail in the appropriate sections of this report.

The three units are identical except for certain auxiliary systems which are shared. Sharing of these systems and components is not detrimental to the safe operation of any unit. General arrangements of major equipment and structures, including the Reactor, Auxiliary, and Turbine Buildings, are shown in Figures 1-2 through 1-9.

The Oconee units are generally similar to those of other current pressurized water reactors. Differences include the generation of superheated steam in once-through steam generators, the use of Keowee Hydro Station as an emergency power source, the use of gravity flow for emergency condenser cooling, and the use of a penetration room ventilation system for collection and treatment of penetration leakage following a loss-of-coolant accident.

The nuclear steam supply system is a pressurized water type using chemical shim and control rods for reactivity control. The Babcock & Wilcox Company (B&W) is supplying the nuclear steam supply system and the initial fuel cores and reloads for each of the three units.

The first generating unit will operate initially at core power levels up to 2452 MWt which corresponds to a net electrical output of 847 MWe. All physics and core thermal hydraulics information in this report is based upon a reference core design of 2568 MWt. The first unit will be capable of an output of 2584 MWt (including 16 MWt contribution from reactor coolant pumps), corresponding to a net electrical capacity of about 886 MWe, shortly after startup and confirmation of core and coolant flow parameters. Units 2 and 3 will operate initially at 2584 MWt. Site parameters, principal structures, engineered safeguards, and accidents are evaluated for a core output of 2568 MWt.

Duke is fully responsible for the complete safety and adequacy of the station; and, consistent with long-standing practice, Company personnel are designing, constructing, testing, starting, and operating the station. Assistance in performing these functions is rendered principally by B&W and by Duke's general consultant, Bechtel Corporation. Technical qualifications of key personnel are given in Section 12 and Appendix 12A.

1.2 SUMMARY PLANT DESCRIPTION

1.2.1 SITE CHARACTERISTICS

The site is characterized by a one-mile exclusion radius; remoteness from population centers; sound, hard rock foundation for structures; freedom from flooding; an abundant supply of cooling water; an on-site hydroelectric station capable of supplying ample emergency power; and favorable conditions of hydrology, geology, seismology, and meterology. The proximity of the hydro tailrace offers the unusual capability of providing emergency, powerless water flow by gravity through the Oconee condensers. This reliable heat sink is available for rejection of decay heat conveyed by natural circulation in the reactor coolant system and steam-driven pumps in the secondary system.

1.2.2 STATION DESCRIPTION

1.2.2.1 General Arrangement

The general arrangement of the major equipment and structures is shown on Figures 1-2 through 1-9.

1.2.2.2 Nuclear Steam System

Each nuclear steam system consists of a pressurized water reactor and two-loop reactor coolant system. The mechanical, thermal-hydraulic, and nuclear design of the reactor core is similar to other systems operating or under construction.

The reactor core is composed of uranium dioxide pellets enclosed in Zircaloy tubes with welded end plugs. The tubes are supported in assemblies by spacer grid assemblies and the upper and lower end fitting assemblies. The reactor core is initially loaded in three regions of different enrichments. The control rod assemblies consist of clusters of stainless steel clad Ag-In-Cd absorber rods and guide tubes located within the fuel assembly.

The two steam generators are vertical, straight tube units producing superheated steam at constant pressure. With the once-through design, natural circulation flow is adequate to remove full decay heat without the use of reactor coolant pumps. Thus, with total loss of pumps, departure from nucleate boiling will not occur in the core.

An electrically heated pressurizer establishes and maintains the reactor coolant pressure and provides a surge chamber and a water reserve to accommodate reactor coolant volume changes during operation.

The reactor coolant pumps (two in each loop) are vertical, single speed, centrifugal units equipped with controlled leakage shaft seals.

1.2.2.3 Containment System

The prestressed, post-tensioned, steel lined, concrete Reactor Building is designed to withstand the maximum internal pressure resulting from an analysis of a spectrum of reactor coolant system leaks.

Isolation valves are provided on fluid piping penetrating the Reactor Building to provide containment integrity when required. These valves are actuated automatically by signals received from the Engineered Safeguards Protective System.

All electrical and fluid penetrations except the main steam lines, component drain line for Units 2 and 3, two emergency recirculation lines, emergency sump drain and normal sump drain are grouped in a penetration room. Any leakage that might occur from any of these penetrations (except the noted lines) will be filtered and exhausted through a unit vent. Access hatches are provided with double seals, and the volume between the seals is piped to the penetration room. Provision is made to leak test all the access hatch closures.

1.2.2.4 Engineered Safeguards Systems

Engineered safeguards systems reduce the potential radiation dose to the general public from the Maximum Hypothetical Accident to less than the guideline values of 10CFR100. Automatic isolation of Reactor Building fluid penetrations that are not required for limiting the consequences of the accident reduces potential leakage paths. Long term potential releases following the accident are reduced by rapidly decreasing the Reactor Building pressure to near atmospheric, thereby reducing the driving potential for fission product escape.

In addition, the engineered safeguards system will provide ample core cooling following the worst postulated loss-of-coolant accident. This is accomplished by large capacity, injection core flooding systems. These systems, coupled with the thermal, hydraulic, and blowdown characteristics of the reactors, will reliably prevent metal-water reactions.

Each reactor unit will have the following engineered safeguards equipment, with the normal operating mode of each as indicated:

(a) High pressure injection system - a portion is used in normal reactor operation.

(b) Low pressure injection system - operates for shutdown cooling.

(c) Core flooding tanks - normally ready for operation.

(d) Reactor Building spray system - normally shutdown.

(e) Reactor Building emergency coolers - operate for Reactor Building cooling during normal operation.

(f) Penetration room ventilation system - test operation during normal operation.

(g) Reactor Building isolation system - normally ready for operation and testable.

The engineered safeguards systems are independent for each unit. The following Table 1-1 lists the major equipment in each system:

Table 1-1
Engineered Safeguards Equipment

System	Total Equipment Installed/Unit
High Pressure Injection System	3 pumps 1 storage tank
Low Pressure Injection System	2 pumps * 2 heat exchangers
Core Flooding Tanks	2 tanks
Reactor Building Spray System	2 pumps 2 spray headers
Reactor Building Coolers	3 coolers 3 fans

* plus one installed spare pump

1.2.2.5 Unit Control

The reactor is controlled by control rod movement and regulation of the boric acid concentration in the reactor coolant. Between 15 percent and 100 percent full power the integrated control system maintains constant average reactor coolant temperature. Constant steam pressure is maintained over the full power range. The combined actions of the control system, steam bypass to the condenser and steam relief valves are designed to maintain station auxiliary load on separation from the transmission system.

The Reactor Protective System and the Engineered Safeguards System automatically initiate appropriate action whenever the parameters monitored by these systems reach pre-established set-points. These systems act to trip the reactor, provide core cooling, close isolation valves and initiate the operation of standby systems as required.

1.2.2.6 Electrical Systems and Emergency Power

36. Each of the three nuclear units at Oconee have up to six available sources of electrical power:

(a) Six 230 kV transmission lines from three directions and two 500 kV transmission lines from two directions serve Oconee.

(b) The other two nuclear units.

36. (c) One 100 kV transmission line. (Does not come through switching station)

(d) One of the quick-starting on-site Keowee Hydroelectric 87,500 KVA Generating Units connected to Oconee by an underground 13.8 kV cable.

(e) The other Keowee Hydroelectric Generating Unit connected to Oconee by an overhead 230 kV transmission line.

Oconee has multiple redundant buses and tie buses supplying power to loads, instruments, and controls. The engineered safeguards for each unit are generally arranged on a three-component basis and supplied from three separate auxiliary power buses, each of which can be supplied from any of the seven principal sources of power.

The sources of power and associated electrical equipment will insure safe functioning of the station and its engineered safeguards.

1.2.2.7 Steam and Power Conversion System

The steam and power conversion system for each unit is designed to remove heat energy from the reactor coolant in the two steam generators and convert it to electrical energy. The closed feedwater cycle will condense the steam and heat feedwater for return to the steam generators.

1.2.2.8 Fuel Handling and Storage

Spent fuel is handled and stored underwater at all times. Both new and spent fuel are stored in the spent fuel pool and transferred to and from the reactor building via the fuel transfer tubes. One spent fuel pool is shared between Units 1 and 2, and a separate spent fuel pool is provided for Unit 3. The system is designed to minimize the possibility of mishandling or maloperations that could cause fuel assembly damage and/or potential fission product release.

1.2.2.9 Radioactive Waste Control

Gaseous waste disposal systems collect, holdup as necessary, filter, monitor, release, and record the gaseous effluent from the station. Liquid waste disposal systems provide for collection, storage, treatment, monitoring, disposal, and recording of liquid wastes. Solid radioactive wastes are stored, packaged, and shipped off-site.

1.2.3 DESIGN CHARACTERISTICS

The important design and operating characteristics of the nuclear steam supply systems for the Oconee Station are summarized in Table 1-2.

The more significant design revisions made to the units since the Preliminary Safety Analysis Report are listed below:

19. Control Rod Drives

Each of the units will utilize sealed roller-nut and leadscrew type control rod drives rather than shaft seal rack and pinion drives.

Fuel Assembly

The fuel assembly utilizes Inconel spacer grids supported by the control rod guide tubes and center instrument tube rather than stainless steel grids supported by an external stainless steel perforated can.
9. All fuel rods except those in the low burnup region of Core 1, Unit 1, are internally prepressurized with helium to minimize clad fatigue due to power and pressure cycling.

Axial Power Shaping Rod Assemblies

Eight (8) of the 69 control rods contain neutron absorber for a portion of their length to aid in controlling Xenon oscillations.

Burnable Poison Rod Assembly

Burnable poison rod assemblies have been added to the first cycle of Unit 2 to reduce the magnitude of the beginning of life positive moderator temperature coefficient.

Emergency Core Cooling System

The Emergency Core Cooling System piping and equipment have been arranged to meet the intent of Criterion 44.

Reactor Building Coolers

The Reactor Building coolers are designed to serve both the normal and emergency building cooling functions.

In-Core Instrumentation Readout

Auxiliary readout of selected in-core detectors is included as a part of the control room instrumentation.

Control Rod Withdrawal Stop and Trip Signals

An intermediate range high start-up rate signal prevents further withdrawal of control rods. Reactor Trip during startup is furnished by a high power range flux level setting.

Emergency Core Cooling System Actuation

The Emergency core cooling system is actuated on high reactor building pressure as well as low reactor coolant system pressure.

1.3 PRINCIPAL DESIGN CRITERIA

The principal design criteria for Oconee Units 1, 2 and 3 were developed in consideration of the 70 General Design Criteria for Nuclear Power Plant Construction Permits proposed by the AEC in a proposed rule-making published for 10CFR50 in the Federal Register of July 11, 1967.

Appendix 1A of this report lists the 70 criteria together with applicant's response indicating the applicant's interpretation of and agreement with the intent of each criterion. In the discussion of each criterion reference is made to sections of this report where more information is presented.

1.4 IDENTIFICATION OF CONTRACTORS

Duke Power Company is responsible for the design, purchasing, construction, and operation of Oconee, a practice successfully followed for all of the Company's major generating facilities now in service or planned.

The Engineering Department has the responsibility for specification of materials and equipment, design of structures and systems, and preparation of installation drawings. Procurement is the responsibility of the Purchasing Department. The Construction Department has the responsibility for all site construction activities. Shop inspections and witness testing are the responsibility of the Engineering Department while field quality control and testing are the responsibility of the Construction Department. The Steam Production Department has the responsibility for preoperational testing and initial startup as well as operation and maintenance of the station. All other departments of Duke are available as needed to assist in the design, construction, or operation of the station.

Duke contracted with B&W to design, manufacture, and deliver to the site three complete nuclear steam supply systems and fuel. In addition, B&W is supplying technical direction of erection; and consultation for initial fuel loading, testing, and initial startup of the complete nuclear steam supply system with coordination, scheduling, and administrative direction by Duke.

The Bechtel Corporation was retained by Duke as a general consultant to provide such engineering assistance as needed during the design and construction of the station. Layout, engineering, and design of the Reactor Buildings were assigned to Bechtel.

Duke retained Pittsburg Testing Laboratory for shop inspection of valves and piping as required. As consultants on seismology and meteorology, the firm of Dames & Moore was retained. Duke also retained Mr. William V. Conn from Atlanta, Georgia, for geology studies and the Law Engineering Testing Company for subsurface investigations under the direction of Dr. George F. Sowers.

1.5 QUALITY ASSURANCE

As designer, constructor, operator, and owner of its power plants, Duke has long recognized the importance of being "quality conscious." The results of emphasis on achieving quality are demonstrated by the record of safety, reliability, and performance of the 26 generating units placed in service by this team since 1946. Building on this experience as a base, the quality efforts at Oconee have been expanded in two principal areas:

(1) The program is formalized by written procedures to effect a broad and consistent understanding of quality objectives.

(2) Documentation is being developed and reviewed to give assurance that the necessary quality standards have, in fact, been achieved.

Responsibility for establishing and implementing the quality assurance program is vested in Duke's Engineering, Construction, and Steam Production Departments. The specific responsibility of each department is clearly delineated, and each has closeness of communications and technical support from the other two. Engineering is responsible for assuring the quality of design and of procured equipment and materials through job site delivery. Construction assures quality of field work from receipt of equipment and materials until systems are ready for start. Quality assurance of pre-operational testing, operation, and maintenance is the responsibility of the Steam Production Department.

A summary description of the Oconee quality assurance program is included as Appendix 1B.

1.6 CONCLUSIONS

On the basis of the information presented in this Final Safety Analysis Report and referenced material, Duke Power Company concludes that the three units of the Oconee Nuclear Station are designed and constructed and will be operated without undue risk to the health and safety of the public.

Table 1-2
Design Parameters - Oconee Nuclear Station
Unit 1, 2, or 3

1. Hydraulic and Thermal Design Parameters

Rated Heat Output (core), MWt	2,568
Rated Heat Output (core), Btu/h	$8{,}765 \times 10^6$
Design Overpower, %	114
System Pressure (nominal), psia	2,200
System Pressure (minimum steady state), psia	2,135
Power Distribution Factors	
Heat Generated in Fuel and Cladding, %	97.3
$F_{\Delta h}$ (nuclear)	1.78
F_q (nuclear)	3.03
Hot Channel Factors	
F_q (nuc. and mech.)	3.12
DNB Ratio at Rated Conditions	2.0
Minimum DNB Ratio at Design Overpower	1.55
Coolant Flow	
Total Flow Rate, lb/h	131.3×10^6
Effective Flow Rate for Heat Transfer, lb/h	124.2×10^6
Effective Flow Area for Heat Transfer, ft^2	49.19
Average Velocity Along Fuel Rods, ft/s	15.73
Coolant Temperature, F	
Nominal Inlet	554
Average Rise in Vessel	50.7
Average Rise in Core	51.5
Average in Core	579.3
Average in Vessel	578.9
Nominal Outlet of Hot Channel	642.8
Average Film Coefficient, Btu/h-ft^2-F	5,000
Average Film Temperature Difference, F	31
Heat Transfer at 100% Power	
Total Heat Transfer Surface Area, ft^2	49,734
Average Heat Flux, Btu/h-ft^2	171,470
Maximum Heat Flux, Btu/h-ft^2	534,440

Table 1-2 (Cont'd)

Average Thermal Output, kW/ft	5.656
Maximum Thermal Output, kW/ft	17.63
Maximum Clad Surface Temperature at Nominal Pressure, F	654
Fuel Central Temperature, F	
Maximum at 100% Power at Hot Spot	4,250
Maximum at 114% Overpower	4,650
Thermal Output, kW/ft at Maximum Overpower	20.1

2. Core Mechanical Design Parameters

Fuel Assemblies	
Number	177
Rod Pitch, in.	0.568
Overall Dimensions, in.	8.536 x 8.536
Total Weight, lb	274,350
Number of Spacer Grids per Assembly	8
Fuel Rods	
Number	36,816
Outside Diameter, in.	0.430
Diametral Gap, in.	0.007
Clad Thickness, in.	0.0265
Clad Material	Zircaloy-4
Fuel Pellets	
Material	UO_2 sintered
Density, % of theoretical	93.5
Diameter, in.	0.370
Length, in.	0.7
Control Rod Assemblies (CRA)	
Neutron Absorber	5% Cd-15% In-80% Ag
Length of Poison Section, in.	134
Cladding Material	304 SS, cold worked
Clad Thickness, in.	0.021
Number of Assemblies	61
Number of Control Rods per Assembly	16

Table 1-2 (Cont'd)

Axial Power Shaping Rod Assemblies (APSRA)	
Neutron Absorber	5% Cd-15% In-80% Ag
Length of Poison Section, in.	36
Cladding Material (Poison Section)	304 SS, cold worked
Clad Thickness, in.	0.021
Number of Assemblies	8
Number of Control Rods per Assembly	16
Burnable Poison Rod Assemblies (BPRA) Unit 2 - First Cycle	
Neutron Absorber	Boron
Length of Poison Section, in.	134
Cladding Material	Zircaloy-4, cold worked
Cladding Thickness, in.	0.0265
Number of Assemblies	68
Number of Rods per Assembly	16
Orifice Rod Assemblies (ORA)	
Rod Material	304 SS, annealed
Number of Orifice Rods per Assembly	16
Core Structure	
Core Barrel ID/OD, in.	141/145
Thermal Shield ID/OD, in.	147/151

3. Preliminary Nuclear Design Data

Structural Characteristics	
Fuel Weight (as UO_2), lb	207,486
Clad Weight (active zone), lb	42,200
Core Diameter, in. (equivalent)	128.9
Core Height, in. (active fuel)	144
Reflector Thickness and Composition	
Top (water plus steel), in.	12
Bottom (water plus steel), in.	12
Side (water plus steel), in.	18
Metal/Water (unit cell - volume basis)	0.82
Fuel Rods/Fuel Assembly	208

Table 1-2 (Cont'd)

Performance Characteristics			
Loading Technique	3 region		
	Unit 1	Unit 2	Unit 3
Fuel Discharge Burnup, MWd/MTU (Average-First Cycle)	9,600	14,250	9,600
Feed Enrichments, wt % U-235 (Average-First Cycle)	2.10	2.65	2.25
Control Characteristics			
Effective Multiplication (Beginning of Life)			
Cold, No Power, Clean	1.248	1.244	1.188
Hot, No Power, Clean	1.198	1.170	1.142
Hot, Rated Power, Xe Equilibrium	1.134	1.109	1.082
Control Rod Worth (Δk/k), %	10.6	10.6	7.4
Boron Concentrations			
To Shut Reactor Down With All Rods Inserted (clean), cold/hot ppm	864/587	992/493	955/665
Boron Worth (hot), % (Δk/k) / ppm	1/85	1/100	1/110
Boron Worth (cold), % (Δk/k) / ppm	1/64	1/75	1/82
Kinetic Characteristics (Range During Life Cycle) - Unit 1			
Moderator Temperature Coefficient, (Δk/k)/F	$+0.5 \times 10^{-4}$ to -3.0×10^{-4}		
Moderator Pressure Coefficient, (Δk/k)/psi	-5.0×10^{-7} to $+3.0 \times 10^{-6}$		
Moderator Void Coefficient, (Δk/k)/% void	$+4.0 \times 10^{-4}$ to -1.6×10^{-3}		
Doppler Coefficient, (Δk/k)/F	-1.1×10^{-5} to -1.7×10^{-5}		

4. Principal Design Parameters of the Reactor Coolant System

System Heat Output, MWt	2,584
System Heat Output, Btu/h	$8{,}819 \times 10^{6}$
Operating Pressure, psig	2,185
Reactor Inlet Temperature, F	554
Reactor Outlet Temperature, F	604
Number of Loops	2
Design Pressure, psig	2,500
Design Temperature, F	650

Table 1-2 (Cont'd)

Hydrostatic Test Pressure (cold), psig	3,125
Coolant Volume, including pressurizer, ft^3	11,478
Total Reactor Flow, gpm	352,000

5. Reactor Coolant System Code Requirements

Reactor Vessel and Closure Head	ASME III, Class A
Steam Generator	
Tube Side	ASME III, Class A
Shell Side	ASME III, Class A
Pressurizer	ASME III, Class A
Pressurizer Safety Valves	ASME III, Art. 9
Reactor Coolant Piping	USAS B31.7
Reactor Coolant Pump Casing	ASME III, Class A (Not Code Stamped)

6. Principal Design Parameters of the Reactor Vessel

Material	SA-533, Grade B, clad with 18-8 Stainless Steel
Design Pressure, psig	2,500
Design Temperature, F	650
Operating Pressure, psig	2,185
Inside Diameter of Shell, in.	171
Outside Diameter Across Nozzles, in.	249
Overall Height of Vessel and Closure Head (over CRD and instrument nozzles), ft-in.	40-8 3/4
Minimum Clad Thickness, in.	1/8

7. Principal Design Parameters of the Steam Generators

Number of Units	2
Type	Vertical, once-through with integral superheater
Tube Material	Inconel
Shell Material	Carbon Steel
Tube Side Design Pressure, psig	2,500
Tube Side Design Temperature, F	650

Table 1-2 (Cont'd)

Tube Side Design Flow, lb/h	65.66×10^6
Shell Side Design Pressure, psig	1,050
Shell Side Design Temperature, F	600
Operating Pressure, Tube Side, Nominal, psig	2,185
Operating Pressure, Shell Side, Nominal, psig	910
Superheat at Outlet at Rated Load	35 F
Hydrostatic Test Pressure (tube-side cold), psig	3,125

8. Principal Design Parameters of the Reactor
16. Coolant Pumps (Reference Supplement 6 Revisions for Oconee 1)

Number of Units	4
Type	Vertical, single stage
Design Pressure, psig	2,500
Design Temperature, F	650
Operating Pressure, Nominal, psig	2,185
Suction Temperature, F	554
Design Capacity, gpm	88,000
9. Total Developed Head, ft	396
Hydrostatic Test Pressure (cold), psig	3,750
Motor Type	a-c induction, single speed
Motor Rating (nameplate), hp	9,000

9. Principal Design Parameters of the Reactor Coolant Piping

Material	Carbon steel clad with SS
Hot Leg (ID), in.	36
Cold Leg (ID), in.	28

10. Engineered Safeguards

Safety Injection System

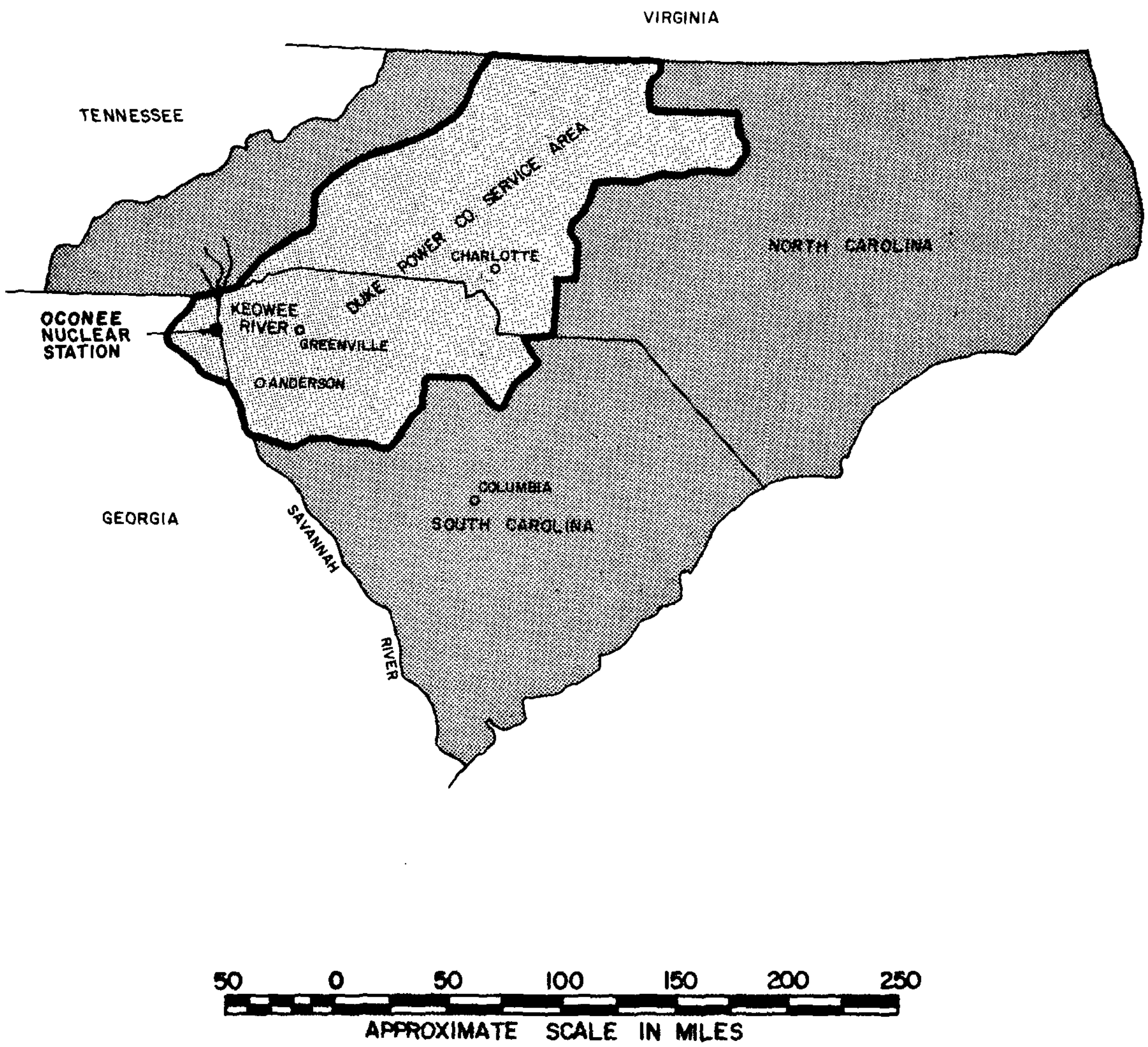
Number of High Pressure Injection Pumps	3
Number of Low Pressure Injection Pumps	2*

*plus one installed spare pump

Table 1-2 (Cont'd)

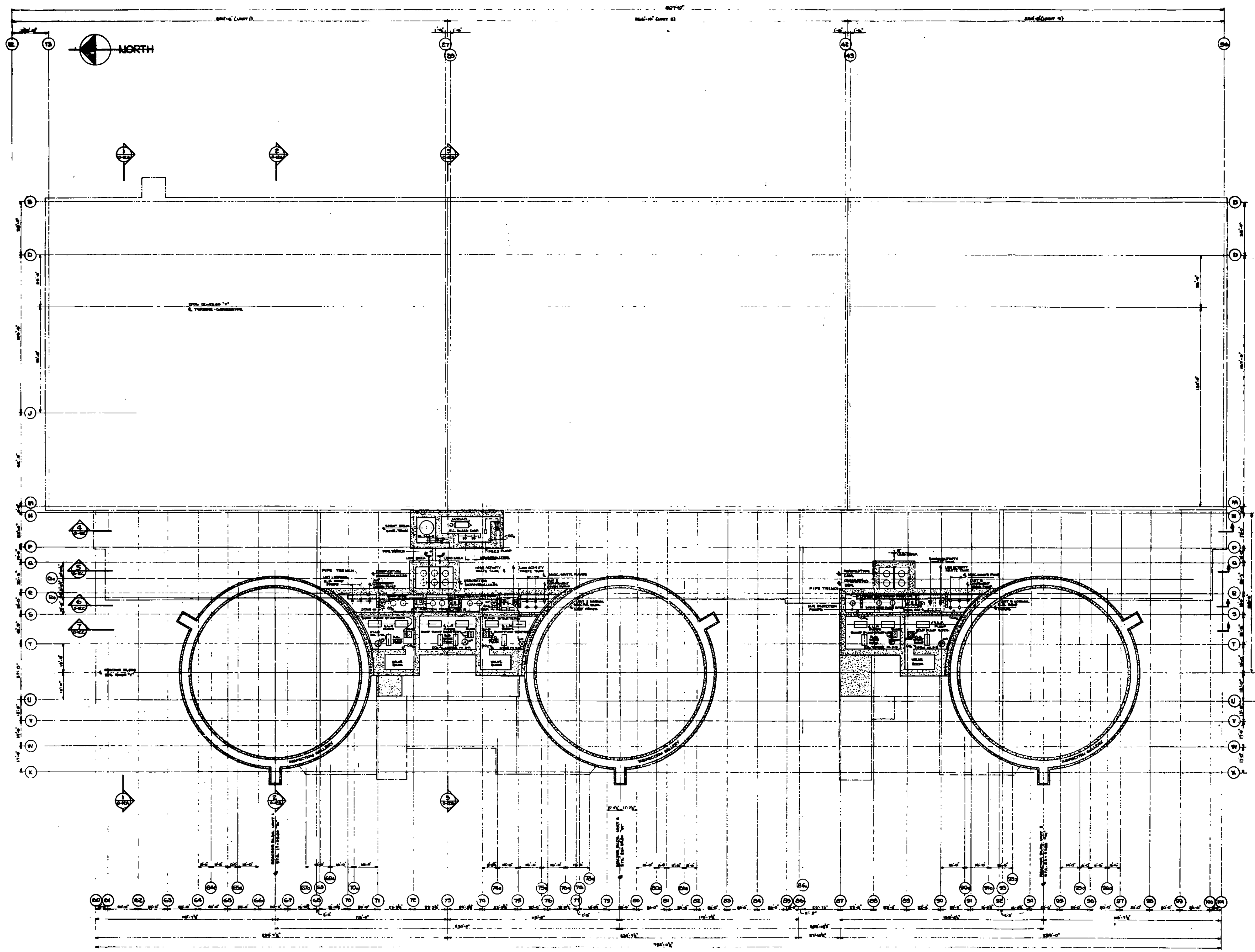
Reactor Building Coolers	
Number of Units	3
Rated Capacity, Each, at Accident Condition, Btu/h	80×10^6
Core Flooding System	
Number of Tanks	2
Total Volume, Each, ft^3	1,410
Reactor Building Spray	
Number of Pumps	2

DUKE POWER COMPANY SERVICE AREA



OCONEE NUCLEAR STATION

FIGURE 1-1



GENERAL ARRANGEMENT; FLOOR PLAN
ELEVATION 758+0

OCONEE NUCLEAR STATION
Figure 1 - 2
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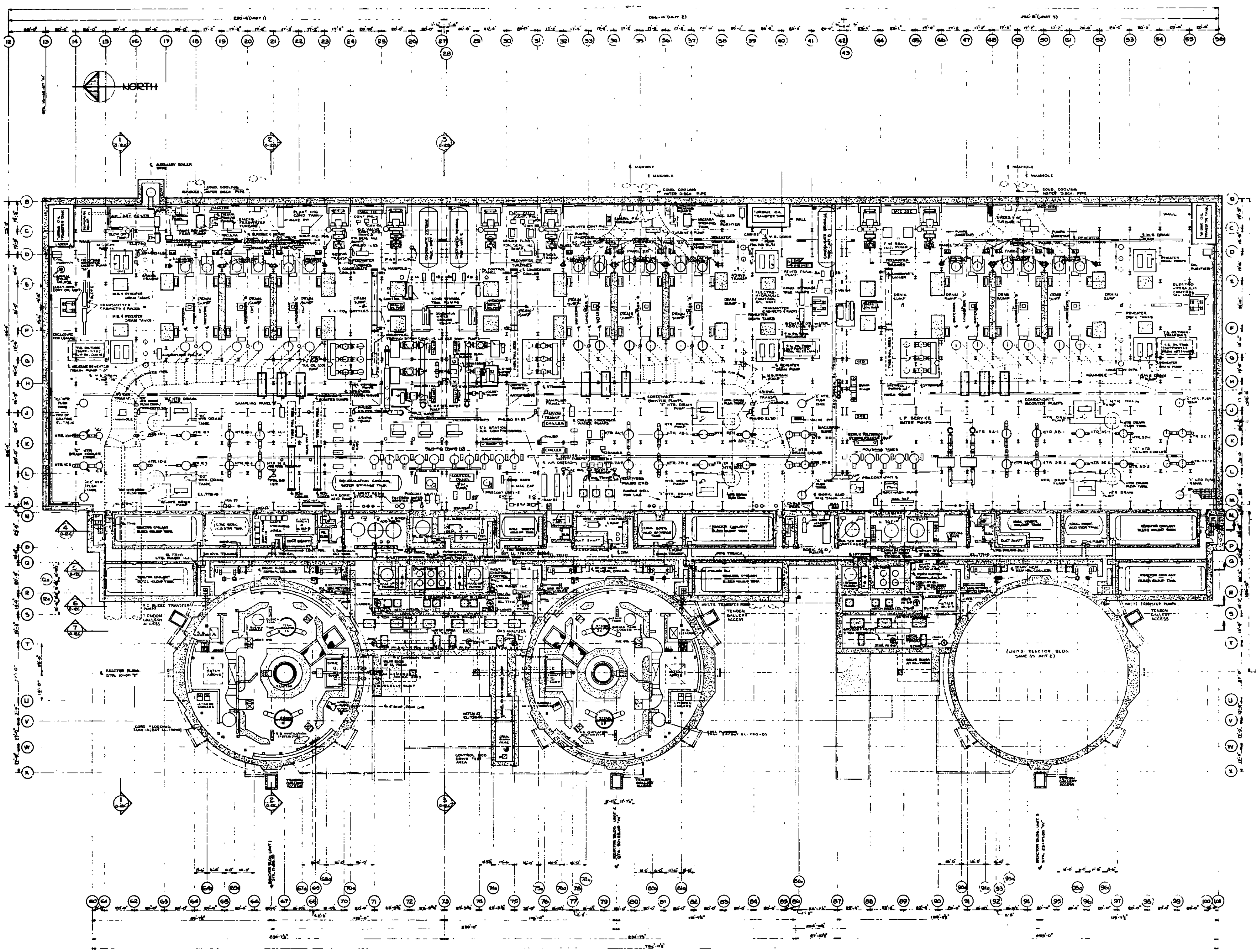

GENERAL ARRANGEMENT, FLOOR PLAN
ELEVATION 771+0 AND ELEVATION 775+0

CITIZENSHIP DUKE POWER SERVICE

OCONEE NUCLEAR STATION
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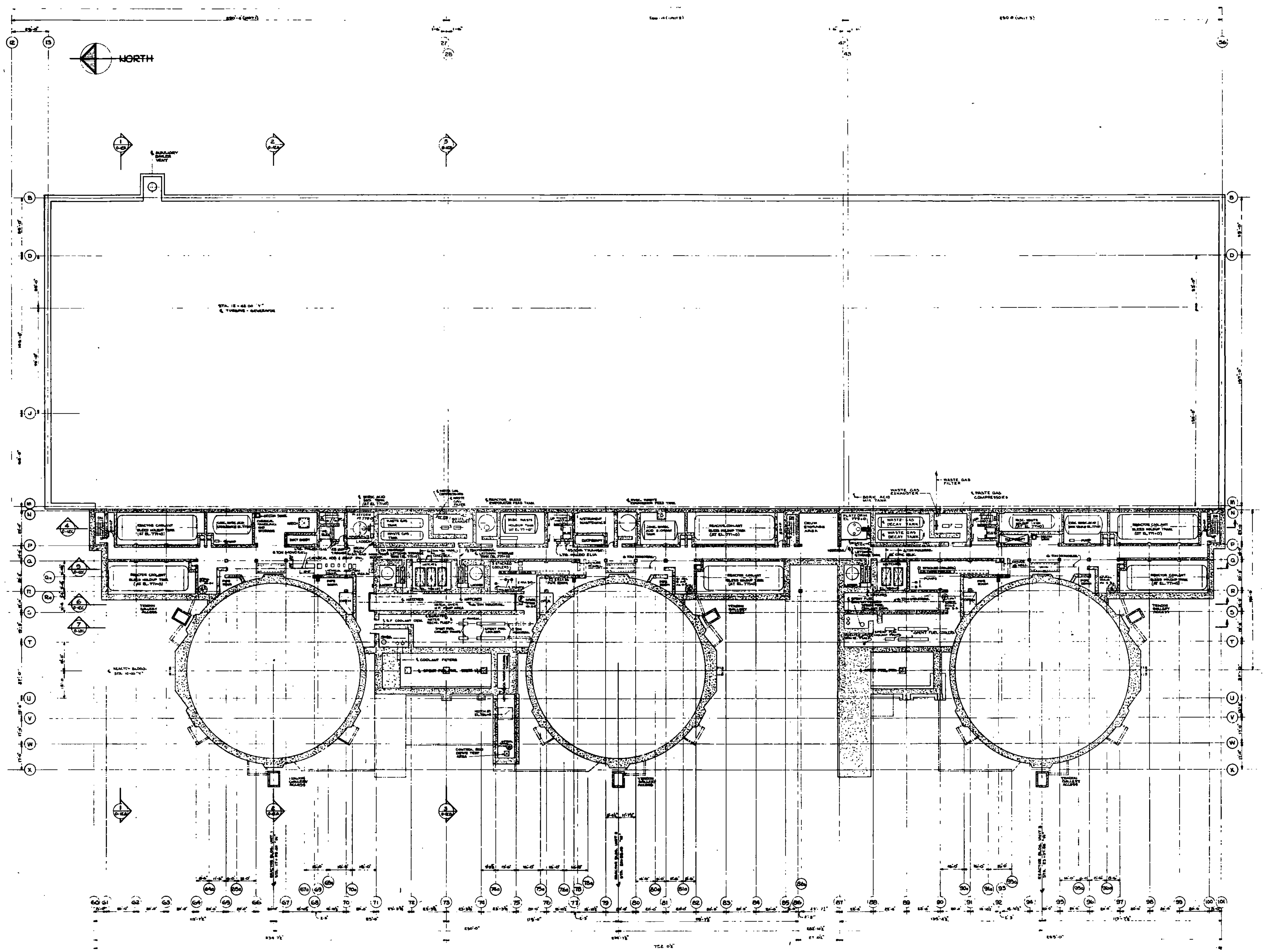

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OCONEE NUCLEAR STATION
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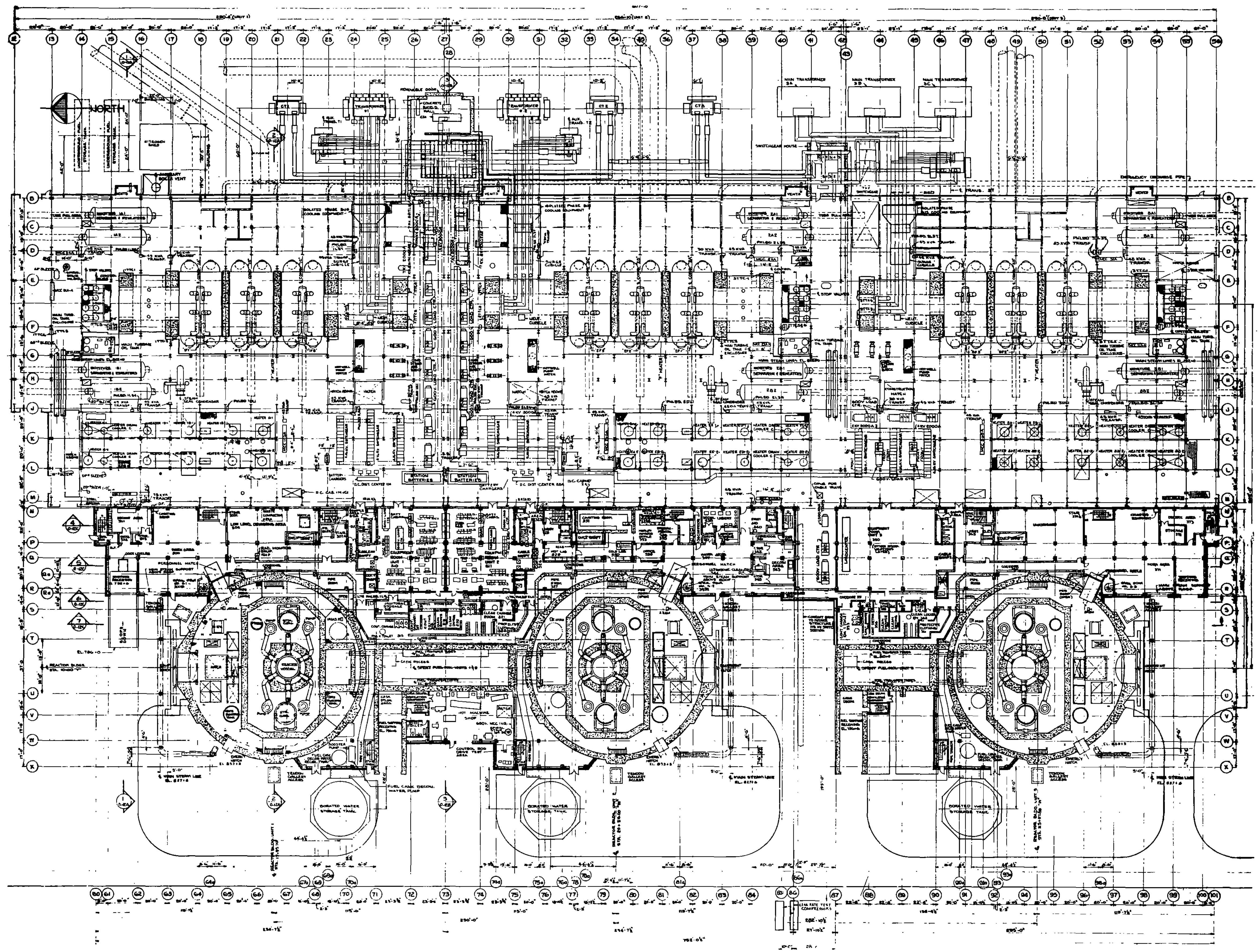

GENERAL ARRANGEMENT, FLOOR PLAN
ELEVATION 796+6

OCONEE NUCLEAR STATION

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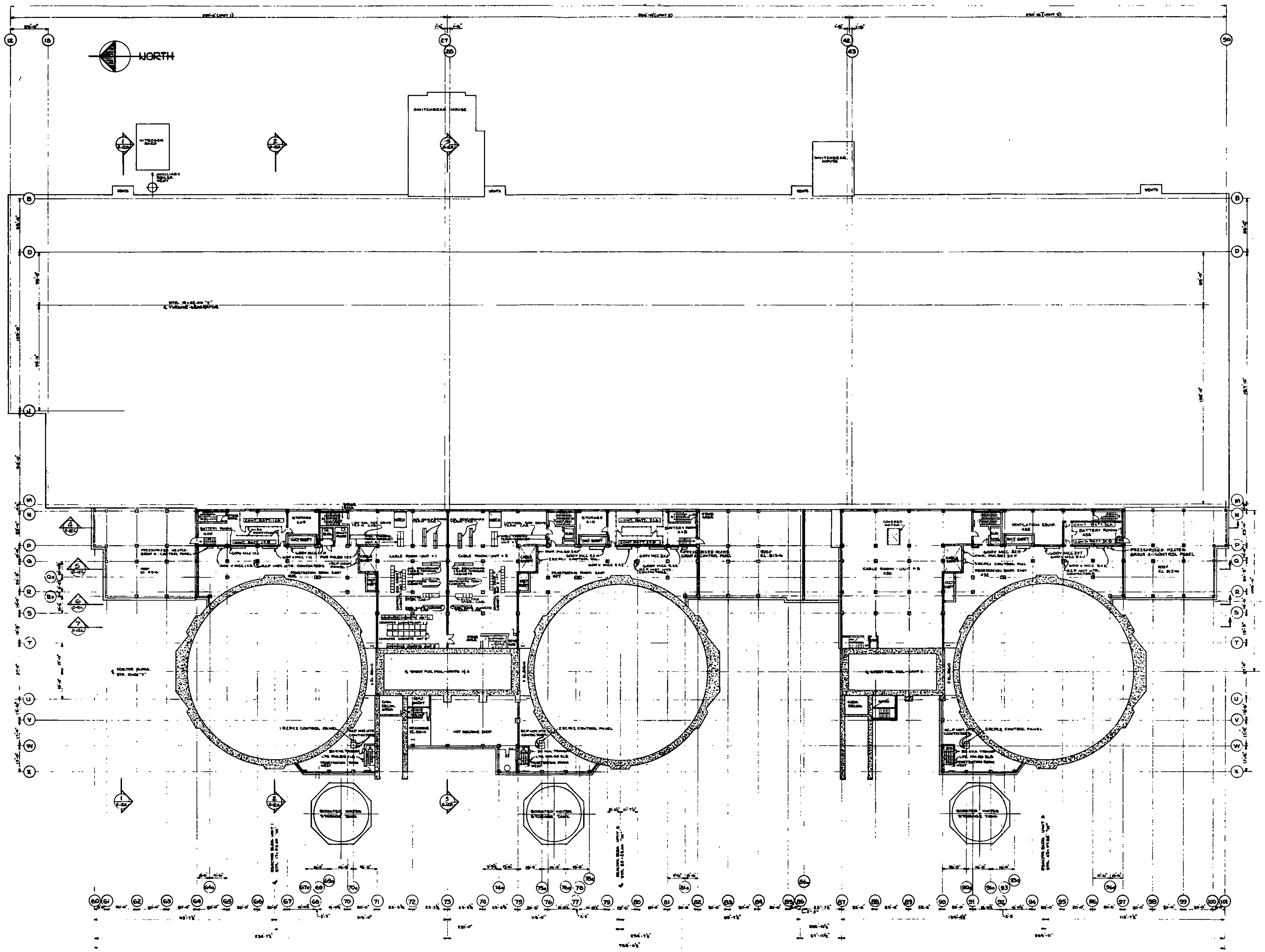

GENERAL ARRANGEMENT, FLOOR PLAN
ELEVATION 809+3

OCONEE NUCLEAR STATION
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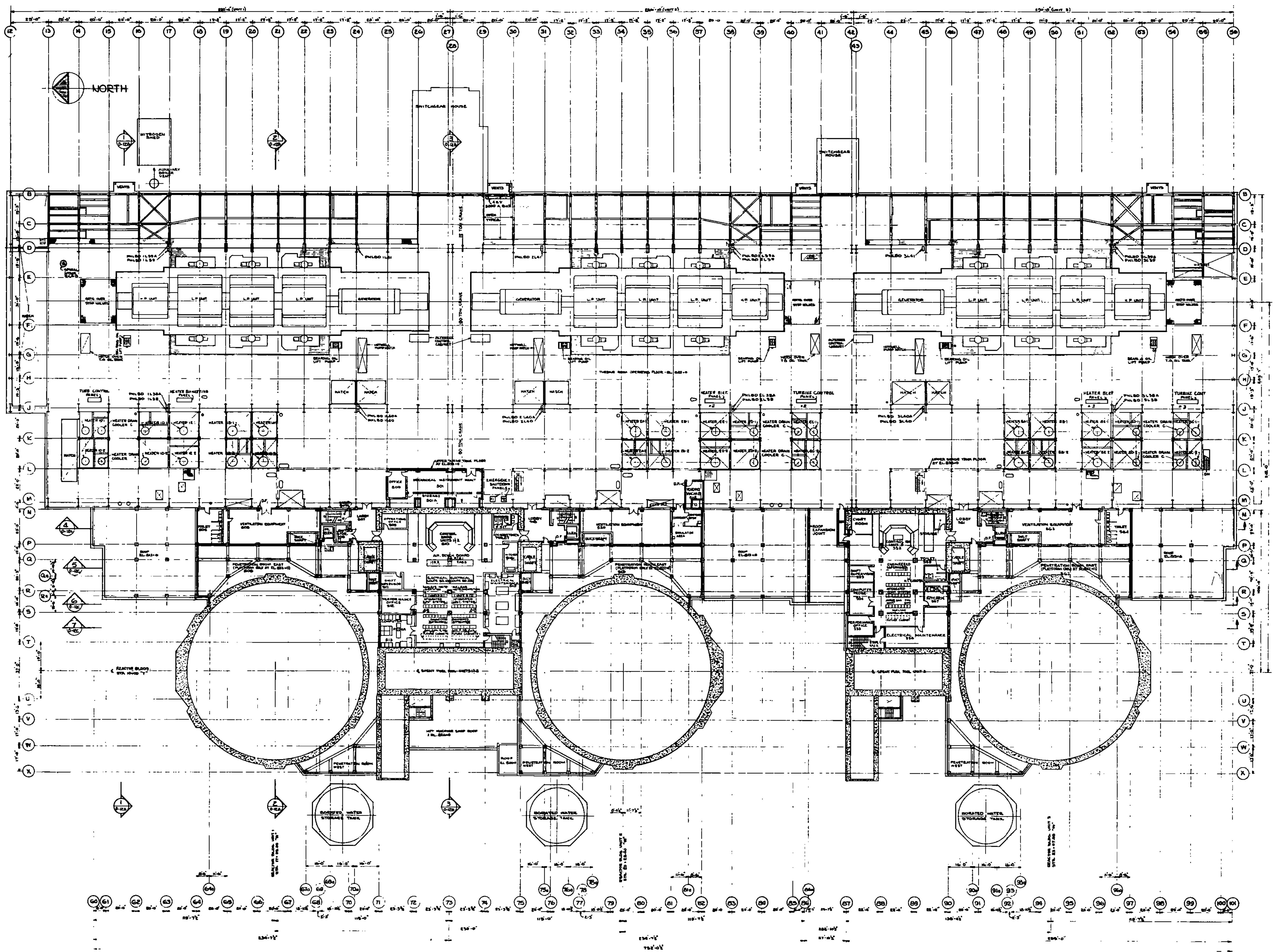

GENERAL ARRANGEMENT, FLOOR PLAN
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OCONEE NUCLEAR STATION

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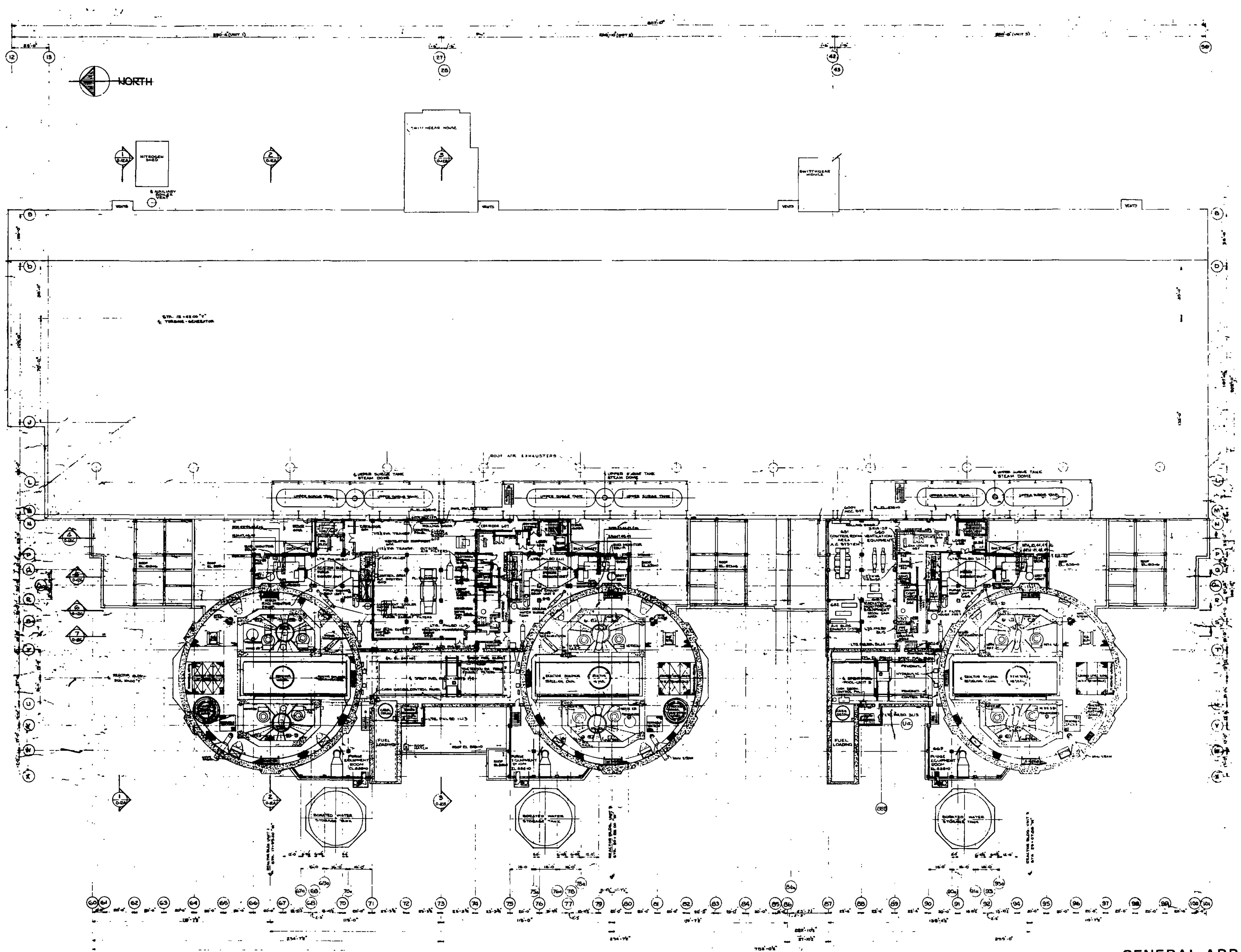

GENERAL ARRANGEMENT, FLOOR PLAN
ELEVATION 838+0 AND ELEVATION 844+0

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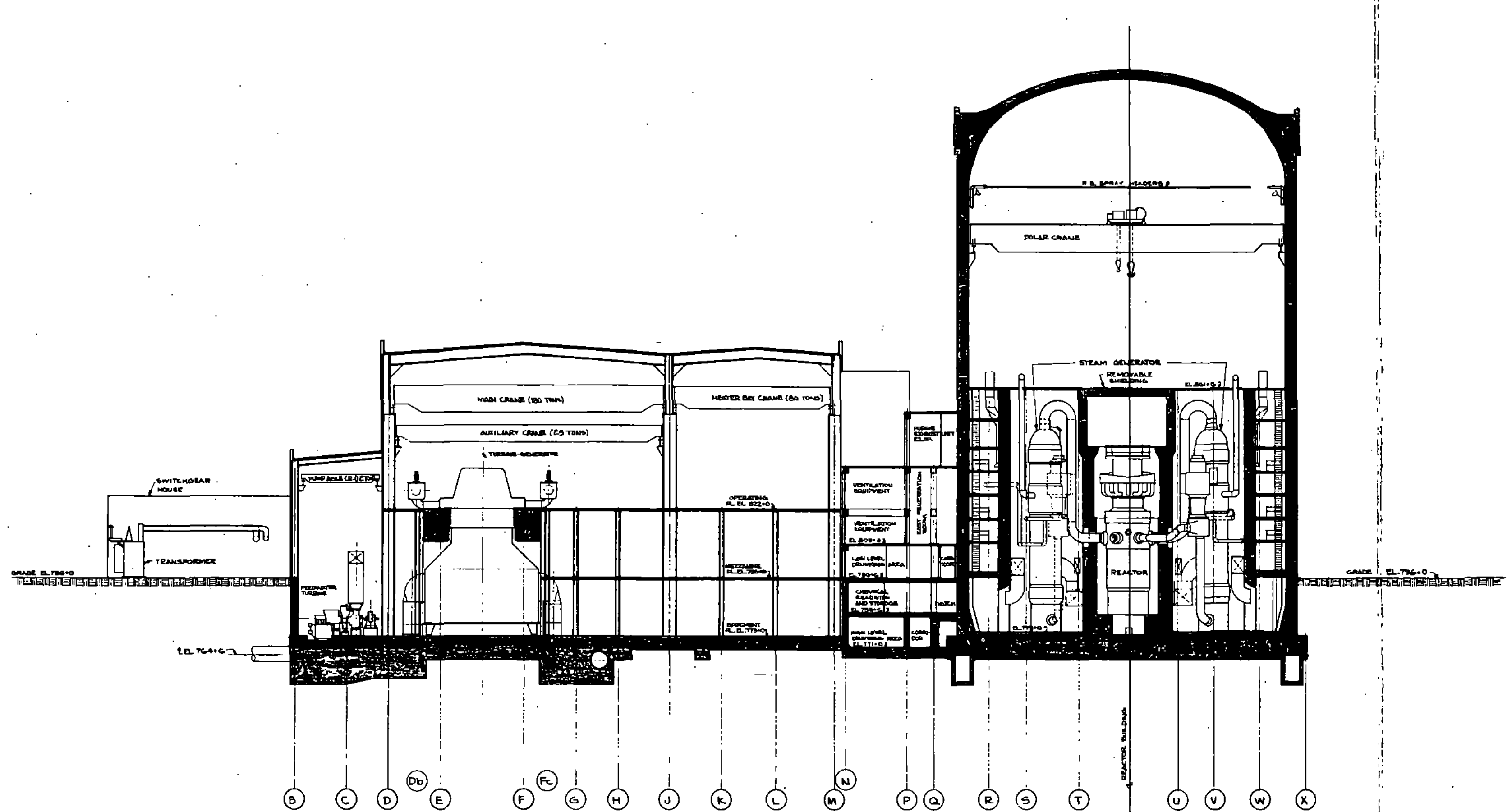
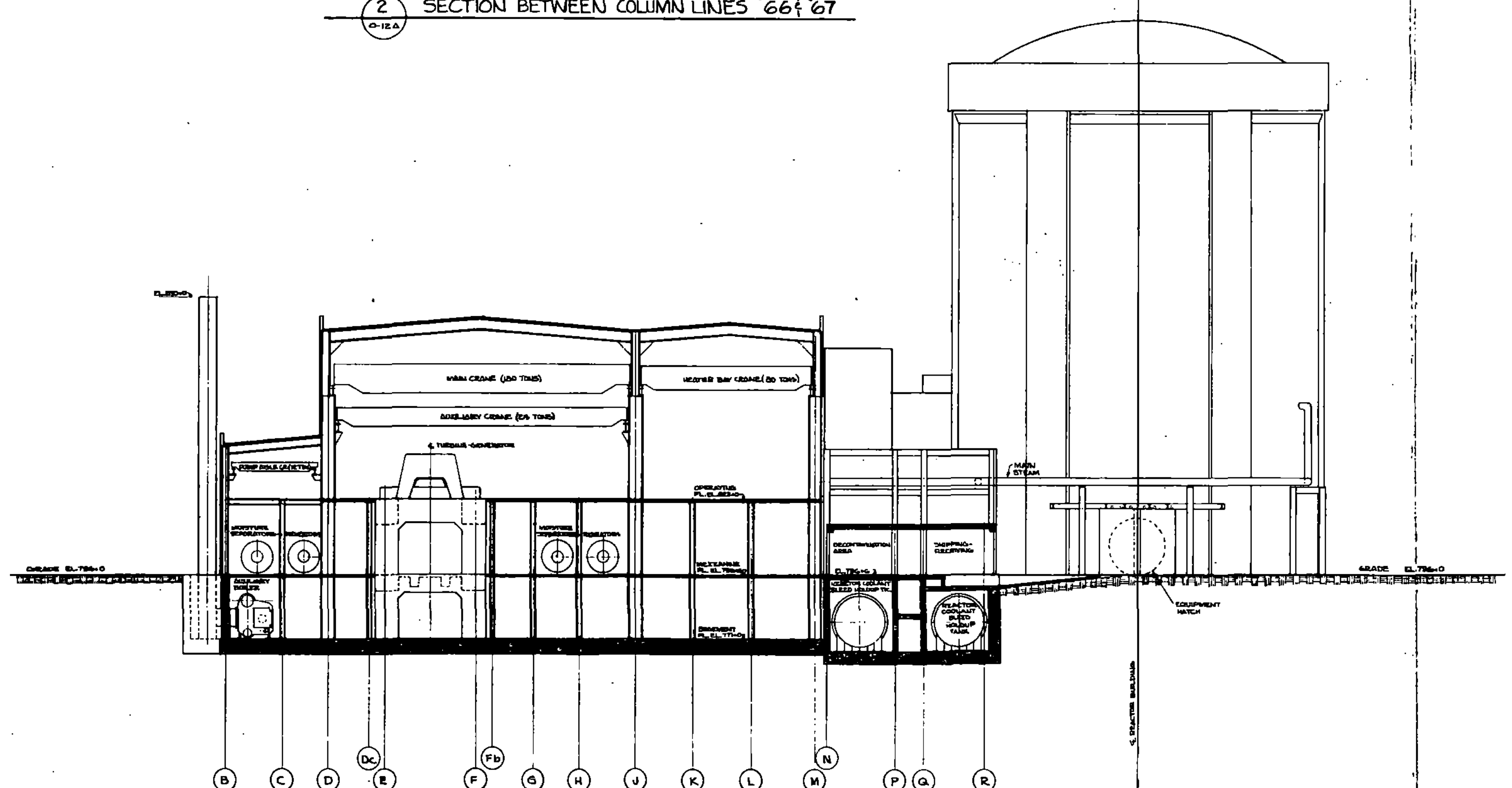


2 / O-12A SECTION BETWEEN COLUMN LINES "66" & "67"



1 / O-12A SECTION BETWEEN COLUMN LINES "61" & "62"

GENERAL ARRANGEMENT, SECTIONS

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APPENDIX 1A

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APPENDIX 1A
PRINCIPAL DESIGN CRITERIA

1A PRINCIPAL DESIGN CRITERIA

The principal design criteria for Oconee Units 1, 2 and 3 were developed in consideration of the 70 General Design Criteria for Nuclear Power Plant Construction Permits proposed by the AEC in a proposed rule-making published for 10CFR Part 50 in the Federal Register of July 11, 1967. Listed below are the 70 criteria proposed by the AEC, together with the applicant's response indicating the applicant's interpretation of and agreement with the intent of each criterion. In the discussion of each criterion, sections of the report containing more detailed information are referenced.

1A.1 CRITERION 1 - QUALITY STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

Discussion

(a) Essential Systems and Components

The integrity of systems, structures, and components essential to accident prevention and to mitigation of accident consequences has been included in the reactor design evaluations. These systems, structures, and components are:

1. Reactor coolant system
2. Reactor vessel internals
3. Reactor building
4. Engineered safeguards system
5. Electric emergency power sources.

(b) Codes and Standards

The following table references applicable sections where codes, quality control, and testing are included in the FSAR. The Quality Assurance program is discussed in detail in Appendix 1B.

Item	Codes	Quality Control	Testing
Reactor Coolant System	4.1	4.3.11 Table 4-12	4.3.11 4.4
Reactor Vessel Internals	3.1.2.4.1 3.2.4.1	3.2.4.1	3.3.4
Reactor Building	5.1.2; 5.6 5.1.4; 5.1.5; Appendix 5A	5.6	5.6
Engineered Safeguards Systems	Table 6.5 6.3.2.2 6.4.2.2	Table 6.5	6.1.4; 6.2.4 6.3.4; 6.4.4
Electric Emergency Power Sources			8.3

1A.2 CRITERION 2 - PERFORMANCE STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and, (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

Discussion

(a) Essential Systems and Components

The integrity of systems, structures and components essential to accident prevention and to mitigation of accident consequences has been included in the reactor design evaluations. These systems, structures, and components are:

1. Reactor coolant system
2. Reactor vessel internals
3. Reactor building
4. Engineered safeguards systems
5. Electric emergency power sources.

(b) These essential systems and components have been designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena. The designs are based upon the most severe of the natural phenomena recorded for the vicinity of the site, with an appropriate margin to account for uncertainties in the historical data.

These natural phenomena are listed below. The design bases for these forces are included in Appendix 5 of the PSAR.

1. Earthquake
2. Tornado
3. Ground Water and Flood
4. Wind and Hurricane
5. Snow and Ice
6. Other Local Site Effects.

1A.3 CRITERION 3 - FIRE PROTECTION (Category A)

The reactor facility shall be designed: (1) to minimize the probability of events such as fires and explosions and, (2) to minimize the potential effects of such events to safety. Noncombustible and fire-resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

Discussion

The reactor facility is designed to minimize the probability of fire and explosion. Noncombustibles and fire-resistant materials were used whenever practical throughout the facility.

The control rooms are constructed and furnished with non-flammable equipment. Adequate fire estinguishers are supplied, and combustible materials, such as records, will be kept to a minimum as indicated in 7.4.5. The control rooms are equipped with emergency breathing apparatus to permit continuous occupancy in the unlikely event of a fire.

Electrical distribution equipment will be physically located to reduce vulnerability of vital circuits to physical damage as a result of accidents. Locations to achieve this result are described in 8.2.2.12.

1A.4 CRITERION 4 - SHARING OF SYSTEMS (Category A)

Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

Discussion

Portions of the following systems are shared as indicated. Where sharing between Units 1 and 2 is indicated, a separate system is provided for Unit 3. Safety is not impaired by the sharing.

System	Shared by Units	Reference
Chemical Addition and Sampling	1, 2	9.2
Spent Fuel Cooling	1, 2	9.4
Liquid Waste Disposal	1, 2, 3	11.1.2
Gaseous Waste Disposal	1, 2	11.1.2
Solid Waste Disposal	1, 2, 3	11.1.2
Coolant Treatment	1, 2, 3	9.10
Recirculated Cooling Water	1, 2, 3	9.6
Low Pressure Service Water	1, 2, 3	9.6
High Pressure Service Water	1, 2, 3	9.6
Control Room Ventilation	1, 2	9.8
Auxiliary Building Ventilation	1, 2	9.8
Turbine Building Ventilation	1, 2, 3	9.8
Area Radiation Monitoring	1, 2	11.2.2
Process Radiation Monitoring	1, 2	11.1.2.4
4.16 Kv Standby Power Buses	1, 2, 3	8.2.2.4
125/250 Volt DC Power System	1, 2, 3	8.2.2.7.2
120 Volt AC Vital Power System	1, 2, 3	8.2.2.8
120 Volt Regulated Power System	1, 2, 3	8.2.2.10

1A.5 CRITERION 5 - RECORDS REQUIREMENTS (Category A)

Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under his control throughout the life of the reactor.

Discussion

Duke will have under its control or will have access to all records of major essential components for the life of the plant. Records maintained by Duke Power Company will include:

(a) A complete set of as-built facility plans and system diagrams which will include arrangement plans, system diagrams, major structural plans, and technical manuals of major installed equipment.

(b) A set of completed test procedures as associated data for all plant testing outlined in Section 13.

(c) Quality assurance data generated during fabrication and erection of the essential components of the plant as defined by the quality assurance program within the scope of Section 1A.1.

1A.6 CRITERION 6 - REACTOR CORE DESIGN (Category A)

The reactor core shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all off-site power.

Discussion

The reactor is designed with the necessary margins to accommodate, without fuel
damage, expected transients from steady-state operation including the transients
given in the criterion. Fuel clad integrity is ensured under all normal and
abnormal modes of anticipated operation by avoiding clad overstressing and over-
heating. The evaluation of clad stresses includes the effects of internal and
external pressures, temperature gradients and changes, clad-fuel interactions,
vibrations, and earthquake effects. Clad fatigue due to power and pressure
9. cycling is minimized by pre-pressurizing with helium all fuel rods except those
in the low burnup region of Core 1, Unit 1. The free-standing clad design
prevents collapse at the end volume region of the fuel rod and provides suf-
ficient radial and end void volume to accommodate clad-fuel interactions and
internal gas pressures (3.2.4.2).

Clad overheating is prevented by satisfying the core thermal and hydraulic criteria (3.1.2.3 and 3.2.3.1.1):

(a) At the design overpower, no fuel melting will occur.

(b) A 99 per cent confidence exists that at least 99.5 per cent of the fuel rods in the core will be in no jeopardy of experiencing a DNB during continuous operation at the design overpower of 114 per cent.

The design margins allow for deviations of temperature, pressure, flow, reactor power, and reactor-turbine power mismatch. Above 15 per cent power, the reactor is operated at a constant average coolant temperature and has a negative power coefficient to damp the effects of power transients. The reactor control system will maintain the reactor operating parameters within preset limits, and the reactor protection system will shut down the reactor if normal operating limits are exceeded by preset amounts (7.1 and 14.1).

Reactor decay heat will be removed through the steam generators until the reactor coolant system is cooled to 250 F. Steam generated by decay heat will supply the steam-driven main feedwater pump turbine and can also be vented to atmosphere and/or bypassed to the condenser. The steam generators are supplied feedwater from either the main steam-driven feedwater pumps, or from a steam-driven emergency feed pump, sized at 7.5 per cent of full feedwater flow.

The main feedwater pumps supply the steam generators with water contained in the feedwater train and the condensate storage tank. The emergency feed pump takes suction from the upper surge tank or from the condenser hotwell. These sources provide sufficient coolant to remove decay heat for about one day after reactor shutdown with the primary heat sink (condenser) isolated. The condenser is normally available so that water inventory is not depleted (10.2.2), even in the event of loss of electrical power.

The reactor coolant pumps are provided with sufficient inertia to maintain adequate flow to prevent fuel damage if power to all pumps is lost. Natural circulation coolant flow will provide adequate core cooling after the pump energy has been dissipated (14.1.2.6).

1A.7 CRITERION 7 - SUPPRESSION OF POWER OSCILLATIONS (Category B)

The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

Discussion

Power oscillations resulting from variations of coolant temperature are minimized by constant average coolant temperature when the reactor is operated above 15 per cent power. Power oscillations from spatial xenon effects are minimized by the large negative power coefficient and axial power shaping rod assemblies.

The ability of the reactor control and protection system to control the oscillations resulting from variation of coolant temperature within the control system dead band and from spatial xenon oscillations has been analyzed. Variations in average coolant temperature provide negative feedback and enhance reactor stability during that portion of core life in which the moderator temperature coefficient is negative. When the moderator temperature coefficient is positive, rod motion will compensate for the positive feedback. The maximum rate of power change resulting from temperature oscillations within the control system dead band has been calculated to be less than 1 per cent/minute. Since the unit has been designed to follow ramp load changes of 10 per cent/minute, this is well within the capability of the control system. (7.2.3)

Control flexibility, with respect to xenon transients, is provided by the combination of control rods and nuclear instrumentation. Axial, radial, or azimuthal neutron flux changes will be detected by the nuclear instrumentation. Individual control rods or groups of control rods can be positioned to suppress and/or correct flux changes. (3.2.2.2.2) The analysis of xenon-related power effects is presented in BAW-10010, "Stability Margin For Xenon Oscillation."

1A.8 CRITERION 8 - OVERALL POWER COEFFICIENT (Category B)

The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

Discussion

The overall power coefficient is negative in the power operating range. (3.1.2)

1A.9 CRITERION 9 - REACTOR COOLANT PRESSURE BOUNDARY (Category A)

The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

Discussion

The reactor coolant system pressure boundary meets the criterion through the following:

(a) Material selection, design, fabrication, inspection, testing, and certification in accordance with ASME codes.

(b) Manufacture and erection in accordance with approved procedures.

(c) Inspection in accordance with ASME code requirements plus additional requirements imposed by the manufacturer.

(d) System analysis to account for cyclic effects of thermal transients, mechanical shock, seismic loadings, and vibratory loadings.

(e) Selection of reactor vessel material properties to give due consideration to neutron flux effects and the resultant increase of the nil ductility transition temperature.

The materials, codes, cyclic loadings, and non-destructive testing are discussed further in Section 4.

1A.10 CRITERION 10 - CONTAINMENT (Category A)

Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity, and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

Discussion

Containment is provided by the Reactor Building. The Reactor Building has the capability to sustain, without loss of integrity, the effects of gross equipment failures, including the transient peak pressure associated with a hypothetical rupture of any pipe in the reactor coolant system including the effects of metal-water reactions described in 14.2.2.3.

The design parameters for the Reactor Building are tabulated in Section 5, and engineered safety systems have been evaluated for various combinations of credible energy releases as discussed in 14.2.2.3. Sufficient redundancy is provided both in equipment and control to ensure the functional availability and capability of systems required to protect the public.

1A.11 CRITERION 11 - CONTROL ROOM (Category B)

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10CFR20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

Discussion

The reactors and associated equipment are controlled from panels located in the control rooms. The control rooms are designed to permit continuous occupancy following a maximum hypothetical accident (MHA) (7.4.5).

All controls and instrumentation required to monitor and operate the reactors and electric power generating equipment are located within the control rooms. This includes indication of power level; process variables such as temperatures, pressures, and flows; valve positions; and control rod positions.

All engineered safety systems equipment are controlled and monitored from the control rooms. The status of all dynamic equipment (pumps, valves, etc.)--as well as pertinent pressures, temperatures, and flows--is displayed. The radiation monitoring system has provisions for alarms and for display of instrumentation readouts in the control room.

The concrete Reactor Buildings and control room walls and roofs are designed to provide adequate protection against direct radiation to control room personnel at all times. Control room personnel on eight-hour shifts during a 90-day period following the MHA would not receive an integrated whole body dose in excess of 3 rem from all sources of direct radiation, including exposure during egress and ingress for shift changes.

The control rooms are provided with independent ventilation and filtration systems to minimize ingress of airborne radioactive contaminants escaping from the Reactor Building. The details of the control room ventilation system and its operation following an accident are described in 9.8.2.

The control rooms are constructed and furnished with non-flammable equipment. Adequate fire extinguishers are supplied and combustible materials, such as records, are kept to a minimum as per 7.4.5. Emergency breathing apparatus is provided in the control room to permit occupancy in the unlikely event of a fire.

Adequate instrumentation and controls are provided to maintain the reactor in a safe hot shutdown condition from outside the control room if access to the control room is lost or if the room must be evacuated temporarily in the unlikely event of a fire or other causes.

1A.12 CRITERION 12 - INSTRUMENTATION AND CONTROL SYSTEMS (Category B)

Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

Discussion

Reactor regulation is based upon the use of movable control rods and a chemical neutron absorber (boron in the form of boric acid) dissolved in the reactor coolant. Input signals to the reactor controls include reactor coolant average temperature, megawatt demand, and reactor power. The reactor controls are designed to maintain a constant average reactor coolant temperature over the load range from 15 to 100 per cent of rated power. The steam system operates at constant pressure for all loads. Adequate instrumentation and controls are provided to maintain operating variables within their prescribed ranges (7.2.3).

The non-nuclear instrumentation measures temperatures, pressures, flows, and levels in the reactor coolant system, steam system, and auxiliary reactor systems, and maintains these variables within prescribed limits (7.3.2).

1A.13 CRITERION 13 - FISSION PROCESS MONITORS AND CONTROLS (Category B)

Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

Discussion

This criterion is met by reactivity control means and control room display. Reactivity control is by movable control rods and by chemical neutron absorber (in the form of boric acid) dissolved in the reactor coolant. The position of each control rod will be displayed in the control room. Changes in the reactivity status due to soluble boron will be indicated by changes in the position of the control rods. Actual boron concentration in the reactor coolant is determined periodically by sampling and analysis (7.2 and 9.1.2).

1A.14 CRITERION 14 - CORE PROTECTION SYSTEMS (Category B)

Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

Discussion

The reactor design meets this criterion by reactor trip provisions and engineered safety features. The reactor protection system is designed to limit reactor power which might result from unexpected reactivity changes, and provides an automatic reactor trip to prevent exceeding acceptable fuel damage limits. In a loss-of-coolant accident, the Engineered Safeguards System automatically actuates the high-pressure and low-pressure injection systems. The core flooding tanks are self-actuating. Certain long-term operations in the emergency core cooling systems which do not require immediate actuation are performed manually by the operator, such as remote switching of the low-pressure injection pumps to the recirculation mode and sampling of the recirculated coolant (7.1.2 and 7.1.3).

1A.15 CRITERION 15 - ENGINEERED SAFETY FEATURES PROTECTION SYSTEMS (Category B)

Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineeres safety features.

Discussion

The Engineered Safeguards Actuation System senses reactor coolant system pressure and reactor building pressure and initiates Emergency Core Cooling, reactor building isolation and reactor building cooling at the appropriate levels. It also initiates starting of the Standby Emergency Power Sources (6.1.2 and 8.2.3.3).

1A.16 CRITERION 16 - MONITORING REACTOR COOLANT PRESSURE BOUNDARY (Category B)

Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

Discussion

Reactor coolant pressure boundary integrity can be continuously monitored in the control room by surveillance of variation from normal conditions for the following:

(a) Reactor building temperature and sump level.

(b) Reactor building radioactivity levels.

(c) Condenser off-gas radioactivity levels and main steam line N-16 monitors (to detect steam generator tube leakage).

(d) Decreasing letdown storage tank water level (indicating system leakage).

Gross leakage from the reactor coolant boundary will also be indicated by a decrease in pressurizer water level and a rapid increase in the reactor building sump water level. (4.2.3.8)

1A.17 CRITERION 17 - MONITORING RADIOACTIVITY RELEASES (Category B)

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths and the facility environs for radioactivity that could be released from normal operations, from anticipated transients and from accident conditions.

Discussion

Various process radiation monitoring system detectors are used to measure airborne gaseous and particulate radioactivity, including iodine, in the Reactor Buildings; in releases from Waste Gas Tanks; and in effluent activity in the vent stacks. (11.1.2.4) These detectors have extended ranges to cover anticipated levels during normal operation, transient and accident conditions. They are also shielded against the background radiation levels expected to exist during an accident so that their readings will be valid under these conditions. Detectors are also located on the radioactive liquid waste discharge line which are interlocked to close the discharge valve on high activity. These instruments have been calibrated and have individual built-in secondary calibration sources of long half-life. Batch samples can also be collected for laboratory analysis and counting prior to the release of liquid and gaseous effluents. Service water, main steam lines, and turbine air ejector off-gas are also monitored to detect leakage of radioactivity in operation.

An area monitor is remotely located at the Visitors Center and serves to measure environmental gamma radiation at that location. It will also measure radiation levels in that area due to any reactor transients or accident conditions. (11.2.2)

As a part of the Environmental Radioactivity Monitoring Program, several stations will be located within the Exclusion Area. One of these is located near the Visitors Center and others in locations where the highest annual ground level concentrations of radioactivity from unit vent releases are expected to exist based on site meteorological studies. Airborne particulates, rain and settled dust, and integrated dose are monitored. (2.7)

In addition, environmental monitoring stations have been established in various populated areas and towns surrounding the site at distances up to 13 miles.

1A.18 CRITERION 18 - MONITORING FUEL AND WASTE STORAGE (Category B)

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

Discussion

All refueling operations will be carried out with the fuel under borated water to provide cooling for fuel assemblies and shielding for personnel.

Level indicators are provided to alarm low water level in the spent fuel storage pool. Penetrations of the pool liner are arranged to prevent accidental drainage of the pool. (9.7.2.3)

Temperature sensors and flow monitors in the spent fuel pool cooling loop alarm on high temperature or loss of flow. (9.4)

Radiation monitors and alarms are provided in the Reactor Building, in all refueling areas, and in the waste storage and processing areas to warn operating personnel of excessive radiation levels. (11.2.2)

1A.19 CRITERION 19 - PROTECTION SYSTEMS RELIABILITY (Category B)

Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

Discussion

The protection systems design meets this criterion by specific instrument location, component redundancy, and in-service testing capability. The major design criteria stated below have been applied to the design of the instrumentation.

(a) No single component failure shall prevent the protection systems from fulfilling their protective function when action is required.

(b) No single component failure shall initiate unnecessary protection system action, provided implementation does not conflict with the criterion above.

Test connections and capabilities are built into the protection systems to provide for:

(a) Pre-operational testing to give assurance that the protection systems can fulfill their required functions.

(b) On-line testing to assure availability and operability. (7.1.1)

1A.20 CRITERION 20 - PROTECTION SYSTEMS REDUNDANCY AND INDEPENDENCE (Category B)

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. Different principles shall be used where necessary to achieve true independence of redundant instrumentation components.

Discussion

Reactor protection is by four channels with 2/4 coincidence, and engineered safeguards features are by three channels with 2/3 coincidence. All protection system functions are implemented by redundant sensors, instrument strings, logic, and action devices that combine to form the protection channels. Redundant protection channels and their associated elements are electrically independent and packaged to provide physical separation. The reactor protection system initiates a trip of the channel involved when modules or equipment are removed. (7.1.1)

1A.21 CRITERION 21 - SINGLE FAILURE DEFINITION (Category B)

Multiple failures resulting from a single event shall be treated as a single failure.

Discussion

The protection systems meet this criterion in that the instrumentation is designed so that a single event cannot result in multiple failures that would prevent the required protective action (7.1.3).

1A.22 CRITERION 22 - SEPARATION OF PROTECTION AND CONTROL INSTRUMENTATION SYSTEMS (Category B)

Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

Discussion

The protection systems' input channels are electrically and physically independent. Shared instrumentation for protection and control functions satisfies the single failure criteria by the employment of isolation techniques to the multiple outputs of various instrument strings.

1A.23 CRITERION 23 - PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTION SYSTEMS (Category B)

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in a loss of the protection function.

Discussion

The protection systems are designed to extreme ambient conditions. The protection systems' instrumentation will operate from 40 to 140 F and sustain the loss-of-coolant building environmental conditions, including 100 per cent relative humidity, without loss of operability. Out-of-core neutron detectors however, will withstand 90 per cent relative humidity. The protection systems' instrumentation will be subject to environmental (qualification) testing as required by the proposed IEEE "Criteria for Nuclear Power Plant Protection Systems," IEEE No. 279, dated August, 1968. Protective equipment outside the reactor building (control room and relay room) is designed for continuous operation in an ambient temperature and relative humidity representative of loss-of-coolant accident conditions (7.1.1).

1A.24 CRITERION 24 - EMERGENCY POWER FOR PROTECTION SYSTEMS (Category B)

In the event of loss of all off-site power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems.

Discussion

In the event of loss of all off-site power to all units at Oconee or to any unit alone, sufficient power for operation of the protection systems of any unit will be available from either of two on-site independent hydroelectric generators. Details of the emergency power generation system are described in 8.2.3.

Redundant battery power is provided for vital instrumentation and control.

1A.25 CRITERION 25 - DEMONSTRATION OF FUNCTIONAL OPERABILITY OF PROTECTION SYSTEMS (Category B)

Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

Discussion

Test circuits are supplied which utilize the redundant, independent, and coincidence features of the protection systems. This makes it possible to manually initiate on-line trip signals in any single protection channel in order to test trip capability in each channel without affecting the other channels (7.1.3).

1A.26 CRITERION 26 - PROTECTION SYSTEMS FAIL-SAFE DESIGN (Category B)

The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined bases if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water), are experienced.

Discussion

The reactor protection system will trip the reactor on loss of power. The engineered safeguards systems are supplied with multiple sources of electric power for control and valve action. A total loss of electrical power to the engineered safeguards actuation system will cause it to assume a tripped position with the exception of the control relays. These relays require power to trip. However, since the engineered safeguards equipment also requires power to operate, this relay need not assume the tripped position upon a total loss of power.

The system is designed for continuous operation under adverse environments, as described in the discussion of Criterion 23 (7.1.1 and 7.1.2).

Redundant instrument channels are provided for the reactor protection and engineered safeguards actuation systems. Loss of power to each individual reactor protection channel will trip that individual channel. Loss of all instrument power will trip the reactor protection system and activate the engineered safeguards system instrumentation (with the exception of the reactor building spray valves).

Manual reactor trip is designed so that failure of the automatic reactor trip circuitry will not prohibit or negate the manual trip. The same is true with respect to manual operation of the engineered safeguards equipment.

1A.27 CRITERION 27 - REDUNDANCY OF REACTIVITY CONTROL (Category A)

At least two independent reactivity control systems, preferably of different principles, shall be provided.

Discussion

This criterion is met by movable control rods and soluble boron poison (7.2.2.1).

1A.28 CRITERION 28 - REACTIVITY HOT SHUTDOWN CAPABILITY (Category A)

At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

Discussion

A single reactivity control system consisting of 61 control rods is provided to rapidly make the core subcritical upon a trip signal. Trip levels are set to protect the core from damage due to the effects of any operating transient. The soluble absorber reactivity control system can add negative reactivity to make the reactor subcritical. However, its action is slow and its ability to protect the core from damage, which might result from rapid load changes such as a full load turbine trip, is not a design criterion for this system. The high degree of redundancy in the control rod system is considered sufficient to meet the intent of this criterion (3.2.2.1).

1A.29 CRITERION 29 - REACTIVITY SHUTDOWN CAPABILITY (Category A)

At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients), sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

Discussion

The reactor design meets this criterion both under normal operating conditions and under the accident conditions set forth in Section 14. The reactor is designed with the capability of providing a shutdown margin of at least 1 per cent $\Delta k/k$ with the single most reactive control rod fully withdrawn at any point in core life with the reactor at a hot, zero power condition. The minimum hot shutdown margin for Oconee 1 of 5.5 per cent $\Delta k/k$ occurs at the end of life (3.2.2.1.3)

1A.30 CRITERION 30 - REACTIVITY HOLDDOWN CAPABILITY (Category B)

At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

Discussion

The reactor meets this criterion with control rods for hot shutdown under normal operating conditions and for shutdown under the accident conditions set forth in Section 14. Reactor subcritical margin is maintained during cooldown by changes in soluble boron concentration. The rate of reactivity compensation from boron addition is greater than the reactivity change associated with the reactor cooldown rate of 100 F/hour. Thus, subcriticality is assured during cooldown with the most reactive control rod totally unavailable (3.2.2.1).

1A.31 CRITERION 31 - REACTIVITY CONTROL SYSTEMS MALFUNCTION (Category B)

The reactivity control systems shall be capable of sustaining any single malfunction, such as unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

Discussion

The reactor design meets this criterion. A reactor trip will protect against continuous withdrawal of a control rod (14.1.2.3).

1A.32 CRITERION 32 - MAXIMUM REACTIVITY WORTH OF CONTROL RODS (Category A)

Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements, and on rates at which reactivity can be increased to insure that the potential effects of a sudden or large change of reactivity cannot: (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

Discussion

The reactor design meets this criterion by safety features which limit the maximum reactivity insertion rate. These include rod-group withdrawal interlocks, soluble boron concentration reduction interlock, maximum rate of dilution water addition, and dilution-time cutoff (14.1.2.4). In addition, the rod drives and their controls have an inherent feature that limits overspeed in the event of malfunctions (3.2.4.3). Ejection of the maximum-worth control rod will not lead to further coolant boundary rupture or to internals damage which would interfere with emergency core cooling (14.2.2.2).

1A.33 CRITERION 33 - REACTOR COOLANT PRESSURE BOUNDARY CAPABILITY (Category A)

The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

Discussion

The reactor design meets this criterion. There are no credible mechanisms whereby damaging energy releases are liberated to the reactor coolant. Ejection of the maximum worth control rod will not lead to further coolant boundary rupture. (14.2.2.2).

1A.34 CRITERION 34 - REACTOR COOLANT PRESSURE BOUNDARY RAPID PROPAGATION FAILURE PREVENTION (Category A)

The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

Discussion

The reactor coolant pressure boundary design meets this criterion by the following:

(a) Development of reactor vessel plate material properties opposite the core to a specified Charpy-V-notch test result of 30 ft/lb or greater at a nominal low NDTT.

(b) Determination of the fatigue usage factor resulting from expected static and transient loading during detailed design and stress analysis.

(c) Quality control procedures including permanent identification of materials and non-destructive testing.

(d) Operating restrictions to prevent failure towards the end of design vessel life resulting from increase in the nil-ductility transition temperature (NDTT) due to neutron irradiation, as predicted by a material irradiation surveillance program (4.4.6).

1A.35 CRITERION 35 - REACTOR COOLANT PRESSURE BOUNDARY BRITTLE FRACTURE PREVENTION (Category A)

Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperature shall be at least 120 F above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60 F above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

Discussion

The reactor vessel is the only reactor coolant system component exposed to a significant level of neutron irradiation and is, therefore, the only component subject to material irradiation damage. Unit operating procedures will limit the operating pressure to 20 per cent of the design pressure when the reactor coolant system temperature is below NDTT +60 F throughout unit life. Analysis has shown no potential reactivity-induced conditions which will result in energy release to the primary system in the range expected to be absorbed by plastic deformation (4.3.3).

1A.36 CRITERION 36 - REACTOR COOLANT PRESSURE BOUNDARY SURVEILLANCE (Category A)

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leak-tight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

Discussion

The reactor coolant pressure boundary components meet this criterion. Space is provided for non-destructive testing during plant shutdown. A reactor pressure vessel material surveillance program conforming to ASTM-E-185-66 has been established (4.4.6).

1A.37 CRITERION 37 - ENGINEERED SAFETY FEATURES BASIS FOR DESIGN (Category A)

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

Discussion

The reactor design meets this criterion. The emergency core cooling systems can protect the reactor for any size leak up to and including the circumferential rupture of the largest reactor coolant pipe (14.2.2.3).

1A.38 CRITERION 38 - RELIABILITY AND TESTABILITY OF ENGINEERED SAFETY FEATURES (Category A)

All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

Discussion

All engineered safeguards systems are designed so that a single failure of an active component in a system will not prevent operation of that system or reduce its capacity below that required to maintain a safe condition. Two independent reactor building cooling systems, each having full heat removal capacity, are provided to prevent overpressurization. (7.1.3)

The high-pressure injection, core-flooding, and low-pressure injection systems have separate equipment and instrumentation strings to ensure availability of capacity.

Some portions of the engineered safeguards systems have both a normal and an emergency function, thereby providing nearly continuous demonstration of operability. During normal operation, the standby and operating units will be rotated into service on a scheduled basis.

Engineered safeguards systems equipment piping that is not fully protected against LOCA missle damage utilizes dual lines to preclude loss of the protective function as a result of the secondary failure.

Testing and inspection of the engineered safeguards systems is further described in Section 6.

1A.39 CRITERION 39 - EMERGENCY POWER FOR ENGINEERED SAFETY FEATURES (Category A)

Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the on-site power system and the off-site power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

Discussion

The electrical systems meet the intent of the criterion as discussed in 8.2.1, 8.2.3 and 8.3.

Three alternate emergency electric power supplies are provided for the station from which power to the engineered safety feature buses of each unit can be supplied. These are the 230 KV switching station with multiple off-site interconnections and two on-site independent 87,500 KVA hydroelectric generating units. Each nuclear unit can receive emergency power from the 230 KV switching station through its start-up transformer as a preferred source. Each unit can receive emergency power from one hydroelectric generating unit through a 13.8 KV underground connection to standby transformer CT4. The other hydroelectric generating unit serves as a standby emergency power source and can supply power to each unit's startup transformer when required. Both on-site hydroelectric generating units will start automatically upon loss of all normal power or upon an engineered safety feature action.

Upon completion of Units 2 and 3, two additional sources of alternate power will be available, as each nuclear unit is capable of supplying any other unit through the 230 KV switching station. In addition, a connection to the 100 KV transmission network is provided as an alternate source of emergency power whenever both hydroelectric generating units are unavailable.

1A.40 CRITERION 40 - MISSILE PROTECTION (Category A)

Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

Discussion

Engineered safety features are redundant and either physically separated or shielded to provide protection against dynamic effects and missiles resulting from hypothesized plant equipment failure. (5.1.5.3)

1A.41 CRITERION 41 - ENGINEERED SAFETY FEATURES PERFORMANCE CAPABILITY (Category A)

Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

Discussion

All engineered safeguards systems are designed so that a single failure of an active component will not prevent operation of that system or reduce the system capacity below that required to maintain a safe condition. Redundancy is provided in equipment and piping so that the failure of a single active component of any system will not impair the required safety function of that system. (7.1.3)

1A.42 CRITERION 42 - ENGINEERED SAFETY FEATURES COMPONENTS CAPABILITY (Category A)

Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

Discussion

The engineered safeguards system design meets this criterion. A single-failure analysis of the emergency core cooling systems (6.1.2.9) and reactor building heat removal systems (6.2; 6.3) demonstrates that these systems have sufficient redundancy to perform their design functions.

The core flooding tanks contain check valves which operate to permit flow of emergency coolant from the tanks to the reactor vessel. These valves are self-actuating and need no external signal or external supplied energy to make them operate. Accordingly, it is not considered credible that they would fail to operate when needed.

The engineered safeguards features are designed to function in the unlikely event of a loss of coolant accident with no impairment of function due to the effects of the accident.

1A.43 CRITERION 43 - ACCIDENT AGGRAVATION PREVENTION (Category A)

Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided.

Discussion

The engineered safeguards systems are designed to meet this criterion. The water injected to ensure core cooling is sufficiently borated to ensure core subcriticality. Non-essential sources of water inside the Reactor Building are automatically isolated to prevent dilution of the borated coolant. Essential sources of post-accident cooling waters are monitored to detect leakage which may lead to dilution of boron content. An analysis has been made to demonstrate that the injection of cold water on the hot reactor coolant system surfaces will not lead to further failure. The design of the equipment and its actuating system ensures that water injection will occur in a sufficiently short time period to preclude significant metal-water reactions and consequent energy release to the Reactor Building (14.2.2.3).

1A.44 CRITERION 44 - EMERGENCY CORE COOLING SYSTEMS CAPABILITY (Category A)

At least two emergency core cooling systems, preferable of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that: (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

Discussion

Emergency core cooling is provided by pumped injection and pressurized core flooding tanks. Pumped injection is subdivided in such a way that there are two separate and independent strings, each including both high pressure and low pressure coolant injection, and each capable of providing 100 per cent of the necessary core injection with the core flooding tanks. There is no sharing of active components between the two subsystems in the post-accident operating mode. The core flooding tanks are passive components which are needed for only a short period of time after the accident, thereby assuring 100 per cent availability when needed. This equipment prevents clad melting for the entire spectrum of reactor coolant system failures ranging from the smallest leak to the complete severance of the largest reactor coolant pipe. (14.2.2.3)

1A.45 CRITERION 45 - INSPECTION OF EMERGENCY CORE COOLING SYSTEMS (Category A)

Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling system including reactor vessel internals and water injection nozzles.

Discussion

All critical parts of the emergency core cooling systems, including the reactor vessel internals, can be inspected during plant shutdown (4.4).

1A.46 CRITERION 46 - TESTING OF EMERGENCY CORE COOLING SYSTEMS COMPONENTS (Category A)

Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance.

Discussion

The design of emergency core cooling systems and components has incorporated adequate test and operational features to permit periodic testing of active components to assure operability and functional capability. Core flooding tank functional performance will be demonstrated only in pre-operational testing.

1A.47 CRITERION 47 - TESTING OF EMERGENCY CORE COOLING SYSTEMS (Category A)

A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

Discussion

The high-pressure (makeup water) and low-pressure (decay-heat removal) injection systems are included as part of normal service systems. Consequently, the active components can be tested periodically for delivery capability. The core flooding system delivery capability will be demonstrated during startup testing. In addition, all valves will be periodically cycled to ensure operability. With these provisions, the delivery capability of the emergency core cooling systems can be periodically demonstrated (6.1.4).

1A.48 CRITERION 48 - TESTING OF OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEMS (Category A)

A capability shall be provided to test, under conditions as close to design as practical, the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.

Discussion

The operational sequence that would bring the emergency core cooling systems into action, including transfer to alternate power sources, can be tested in parts (6.1.4 and 7.1.3).

1A.49 CRITERION 49 - CONTAINMENT DESIGN BASIS (Category A)

The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

Discussion

The Reactor Building, access openings and penetrations, have been designed to accommodate a pressure of 59 psig at 286 F (5.1.1). As described in 14.2.2.3,these conditions exceed the greatest transient peak pressure associated with a hypothetical rupture of a pipe in the reactor coolant system, including the margin for the effects of metal-water reactions. The capacity of each Reactor Building cooling system (6.2 and 6.3) is designed to remove sufficient heat from the Reactor Building to prevent exceeding the design pressure at any time following a loss-of-coolant accident.

Components of the Reactor Building cooling system--including electric motors, valves, and damper operators, which function within the Reactor Building during accident conditions--are capable of operation as required to accomplish the safeguards function.

1A.50 CRITERION 50 - NDT REQUIREMENT FOR CONTAINMENT MATERIAL (Category A)

Principal load-carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30 F above nil-ductility transition (NDT) temperature.

Discussion

The Reactor Building liner has been designed so that it is not susceptible to a low temperature brittle fracture.

All principal load-carrying components of ferritic materials for the containment vessel exposed to the external environment have been selected and tested to confirm that their ductile-to-brittle-transition (NDT) temperature is at least 30 F below the minimum service metal temperature. The ferritic materials exposed to the external environment consist of the penetrations and large openings (equipment access hatch and personnel locks), for which materials have been selected to conform with the ASME Boiler and Pressure Vessel Code, Section III, for Class "B" Vessels. Material specifications for the penetrations are more completely described in 5.1.2.1.

1A.51 CRITERION 51 - REACTOR COOLANT PRESSURE BOUNDARY OUTSIDE CONTAINMENT (Category A)

If part of the reactor coolant pressure boundary is outside the containment, appropriate features, as necessary, shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features, such as isolation valves and additional containment, shall include consideration of the environmental and population conditions surrounding the site.

Discussion

The reactor coolant pressure boundary is defined as those piping systems or components which contain reactor coolant at high pressure and temperature. With the exception of the reactor coolant sampling line, the reactor coolant pressure boundary, as defined above, is located entirely within the Reactor Building. The sampling line is provided with remotely operated valves for isolation in the unlikely event of a failure. This line is normally isolated and is used only during actual sampling operations. All other piping and components which may contain reactor coolant are at low temperatures such that any leakage would be collected by the waste disposal system. No significant environmental dose would result from these sources. (5.2 and 11.1.2)

1A.52 CRITERION 52 - CONTAINMENT HEAT REMOVAL SYSTEMS (Category A)

Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

Discussion

Two systems of different principles are provided to remove heat from each Reactor Building following an accident in order to maintain the pressure below the containment design pressure. The systems discussed in 6.2 and 6.3 are capable of removing sufficient energy to maintain the pressure below the containment design pressure.

The Reactor Building Cooling System removes heat by circulating building atmosphere over cooling coils.

The Reactor Building Spray System supplies droplets of cool, borated water which absorb sensible and latent heat from the containment atmosphere.

1A.53 CRITERION 53 - CONTAINMENT ISOLATION VALVES (Category A)

Penetrations that require closure for the containment function shall be protected with redundant valving and associated apparatus.

Discussion

Piping penetrations that require closure under accident conditions are provided with double valves so that no single credible failure or malfunction could result in a loss of isolation. Valves are manually, electrically or pneumatically operated. All isolation valves inside the Reactor Building requiring remote operation are electrically operated.

1A.54 CRITERION 54 - CONTAINMENT LEAKAGE RATE TESTING (Category A)

Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period to verify its conformance with required performance.

Discussion

The Reactor Buildings are designed so that leakage rate can be determined at design pressure after completion and installation of all penetrations. The leak-rate test will verify that the maximum integrated leak rate does not exceed the design leakage rate. (5.6.1.3)

1A.55 CRITERION 55 - CONTAINMENT PERIODIC LEAKAGE RATE TESTING (Category A)

The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

Discussion

The Reactor Building has been structurally designed to permit integrated leakage rate testing at design pressure (5.6.2.1), but retesting is contemplated at lesser pressures.

1A.56 CRITERION 56 - PROVISIONS FOR TESTING OF PENETRATIONS (Category A)

Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at design pressure at any time.

Discussion

All Reactor Building penetrations with resilient seals or expansion bellows are constructed so that they may be pressurized to design pressure for leak tests at any time. (5.1.4.4 and 5.6.2.1)

1A.57 CRITERION 57 - PROVISIONS FOR TESTING OF ISOLATION VALVES (Category A)

Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valves leakage does not exceed acceptable limits.

Discussion

All remotely operated valves serving an Engineered Safeguards function will have the capability for testing their functional operability. These tests can be conducted from the control rooms.

Isolation valves that are required to be closed from an Engineered Safeguards signal will have test provisions for leak testing (Table 5-4).

1A.58 CRITERION 58 - INSPECTION OF CONTAINMENT PRESSURE - REDUCING SYSTEMS (Category A)

Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as pumps, valves, spray nozzles, torus, and sumps.

Discussion

Provision is made to permit periodic physical inspection of components of the two containment pressure-reducing systems, the Reactor Building Spray System and the Reactor Building Cooling System. The Reactor Building spray pumps and the valves and operators associated with piping in each of these systems are located outside the Reactor Building, permitting the inspection of these components. The fan units of the Reactor Building cooling units are located so that physical inspection is possible during normal operation.

The cooling coils of the Reactor Building cooling units can be inspected during shutdown. The spray header and nozzles of the Reactor Building Spray System, located in the dome of the Reactor Building, can be inspected visually during shutdown. The sumps can be inspected and the screens cleaned during shutdown.

1A.59 CRITERION 59 - TESTING OF CONTAINMENT PRESSURE-REDUCING SYSTEM COMPONENTS (Category A)

The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves can be tested periodically for operability and required functional performance.

Discussion

The containment pressure-reducing systems have the capability of being periodically tested as follows:

1. Reactor Building Cooling Units

   (a) The air fans can be individually tested for low speed operation.

   (b) The cooling coil low pressure service water valves can be operated through their full travel with resulting flow alarm indication.

   (c) The stand-by low pressure service water pumps can be tested for automatic starting.

2. Reactor Building Spray System

   (a) The operation of the spray pumps can be tested by recirculating to the borated water storage tank through a test line.

   (b) The building spray isolation valves can be operated through their full travel.

1A.60 CRITERION 60 - TESTING OF CONTAINMENT SPRAY SYSTEMS (Category A)

A capability shall be provided to periodically test the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical.

Discussion

The delivery capability of the spray nozzles will be tested by blowing low pressure air through the system and verifying flow through the nozzles.

The delivery capability of the pumps will be tested by recirculating to the borated water storage tank and monitoring the resultant flow.

1A.61 CRITERION 61 - TESTING OF OPERATIONAL SEQUENCE OF CONTAINMENT PRESSURE REDUCING SYSTEMS

A capability shall be provided to test, under conditions as close to the design as practical, the full operational sequence that would bring the containment pressure-reducing systems into action including the transfer to alternate power sources.

Discussion

Each of the three redundant 4KV switchgear buses supplying power to essential loads receives its power from two 4KV main feeder buses. These main feeder buses are supplied by (1) the main unit auxiliary transformer, (2) the start-up transformer, and (3) the underground feeder from Keowee hydro plant. Each main feeder bus is fed from each of the three sources above. In normal operation the two main feeders will be supplied through breakers from the unit auxiliary transformer and the breakers from the start-up transformer and the underground feeder will be open.

To test the transfer to alternate power source, the three breakers associated with one of the main feeders will be placed in test position with the normal breaker closed and the two alternate power source breakers open. A low voltage simulation will be used to trip the normal breaker and close the start-up breaker. A low voltage and an ESG simulation will be used to trip the start-up breaker and close the underground feeder breaker. In making these tests, the automatic dropping of load will not take place.

Testing the two independent channels for the Reactor Building Cooling System and the Building Spray System by inserting an analog signal can be accomplished without placing the systems in operation.

1A.62 CRITERION 62 - INSPECTION OF AIR CLEANUP SYSTEMS

Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems such as ducts, filters, fans, and dampers.

Discussion

The penetration room ventilation system is designed to collect and process the leakage from penetrations as noted in 6.4. All components of this system are located in the Auxiliary Building, permitting periodic physical inspection.

1A.63 CRITERION 63 - TESTING OF AIR CLEANUP SYSTEM COMPONENTS

Design provisions shall be made so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

Discussion

The penetration room ventilation system (6.4.2) is designed so that active components can be tested periodically. Each penetration room fan can be manually started periodically to demonstrate operability. Provision is made for pressure testing the valves for leak tightness.

1A.64 CRITERION 64 - TESTING OF AIR CLEANUP SYSTEMS

A capability shall be provided for in situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and, (b) filter and trapping materials have not deteriorated beyond acceptable limits.

Discussion

The design of the penetration room ventilation system (6.4.2) incorporates provisions for testing and surveillance of the filters. Connections and instrumentation for each filter bank allow in situ testing to ensure that filter performance has not deteriorated beyond acceptable limits.

1A.65 CRITERION 65 - TESTING OF OPERATIONAL SEQUENCE OF AIR CLEANUP SYSTEMS (Category A)

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the air cleanup systems into action including the transfer to alternate power sources and the design air flow delivery capability.

Discussion

The penetration room ventilation system is designed to allow testing the operational sequences required to bring the system into operation including the transfer to alternate power sources and the design air flow delivery capability.

Actuation of the penetration room ventilation system and the transfer to alternate power sources are identical to the description provided in 1A.61 for the containment pressure reducing systems.

1A.66 CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY (Category B)

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

Discussion

Criticality in new and spent fuel storage is prevented by designing storage facilities to maintain an eversafe geometric spacing of 21 inches between assemblies. Fuel assemblies cannot be placed in other than the prescribed locations (9.7.2.3).

1A.67 CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT (Category B)

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

Discussion

This criterion is met by the Spent Fuel Cooling System which incorporates provisions to maintain water cleanliness, temperature, and water level. Two pumps and two coolers will be adequate to maintain the spent fuel pool temperature within acceptable limits. The pumps in this system can be operated from the standby bus in case of loss of outside power to provide continuous cooling capability in the fuel storage facility (9.4).

1A.68 CRITERION 68 - FUEL AND WASTE STORAGE RADIATION SHIELDING (Category B)

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities to meet the requirements of 10CFR20.

Discussion

Shielding meeting the requirements of 10CFR20 is provided for protection of operating personnel:

(a) During all phases of spent fuel removal and storage (11.2.1)

(b) From radioactive waste holdup tanks and other containers containing potentially radioactive solutions, resins, or gases (11.2.1).

1A.69 CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (Category B)

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

Discussion

Analyses in Section 14 have demonstrated that accidental release of the maximum activity content of a tank containing waste gases or liquids will not cause excessive off-site doses. The fuel handling accident, analyzed in Section 14, does not result in excessive off-site doses.

1A.70 CRITERION 70 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT (Category B)

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified: (a) on the basis of 10CFR20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10CFR100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

Discussion

The waste disposal system is designed to insure that station personnel and the general public are protected against excessive exposure to radioactive material in accordance with the regulations of 10CFR20.

The gaseous, liquid, and solid waste storage facilities are discussed in Section 11 where it is demonstrated that adequate holdup capacity is provided. Gaseous and liquid wastes will be sampled before release and will be monitored for activity level at all times during release.

Control of leakage following a reactor accident is accomplished by the containment system. This system consists of the Reactor Building and the Reactor Building Penetration Room Ventilation System. Experience has shown that Reactor Building leakage is more likely at penetrations than in liner plates or weld joints. Any potential penetration leakage will be into the Penetration Room which has a ventilation system designed to collect and filter leakage from penetrations. The ventilation system maintains a slightly negative pressure within the Penetration Room. Discharge from the Penetration Room ventilating system is to the unit vent through a series filter system consisting of a prefilter, an absolute filter and a charcoal filter.

The release of radioactive materials produced by a reactor accident or waste gas tank failure are within the guidelines set by 10CFR100.

LIST OF EFFECTIVE PAGES
FSAR APPENDIX 1B

Quality Assurance

APPENDIX 1B
QUALITY ASSURANCE PROGRAM

A discussion of the Duke Power Company Operational Quality Assurance Program is presented in Duke Power Company Topical Report DUKE-1, Section 17.2.

LIST OF EFFECTIVE PAGES
FSAR APPENDIX 1C

Systems Design Criteria

Page	Revision	Page	Revision
LOEP--1 of 2	Rev. 26	1C-4j	Rev. 6
LOEP--2 of 2	Rev. 26	1C-4k	Rev. 6
1C-i	Rev. 21	1C-4l	Rev. 6
1C-ia	Rev. 21	1C-4m	Rev. 21
1C-ii	Rev. 21	1C-4mi	Rev. 21
1C-iii	Rev. 16	1C-4n	Rev. 20
1C-iv	Rev. 20	1C-4o	Rev. 20
1C-1	Original	1C-4p	Rev. 24
1C-2	Rev. 24	1C-5	Rev. 16
1C-3	Original	1C-5a	Rev. 16
1C-4	Rev. 20	1C-5b	Rev. 16
1C-4a	Rev. 20	1C-6	Rev. 6
1C-4a-i	Rev. 6	1C-7	Rev. 5
1C-4b	Rev. 6	1C-8	Rev. 5
1C-4c	Rev. 24	1C-9	Rev. 5
1C-4c-i	Rev. 21	1C-10	Rev. 5
1C-4c-ii	Rev. 7	1C-11	Rev. 6
1C-4d	Rev. 6	1C-12	Rev. 21
1C-4e	Rev. 6	1C-13	Rev. 21
1C-4f	Rev. 6	1C-14	Rev. 21
1C-4g	Rev. 6	Fig. 1C-1	Rev. 16
1C-4h	Rev. 6	Fig. 1C-2, 1C-19	Rev. 16
1C-4i	Rev. 6	Fig. 1C-20	Rev. 6

LIST OF TABLES

LIST OF FIGURES

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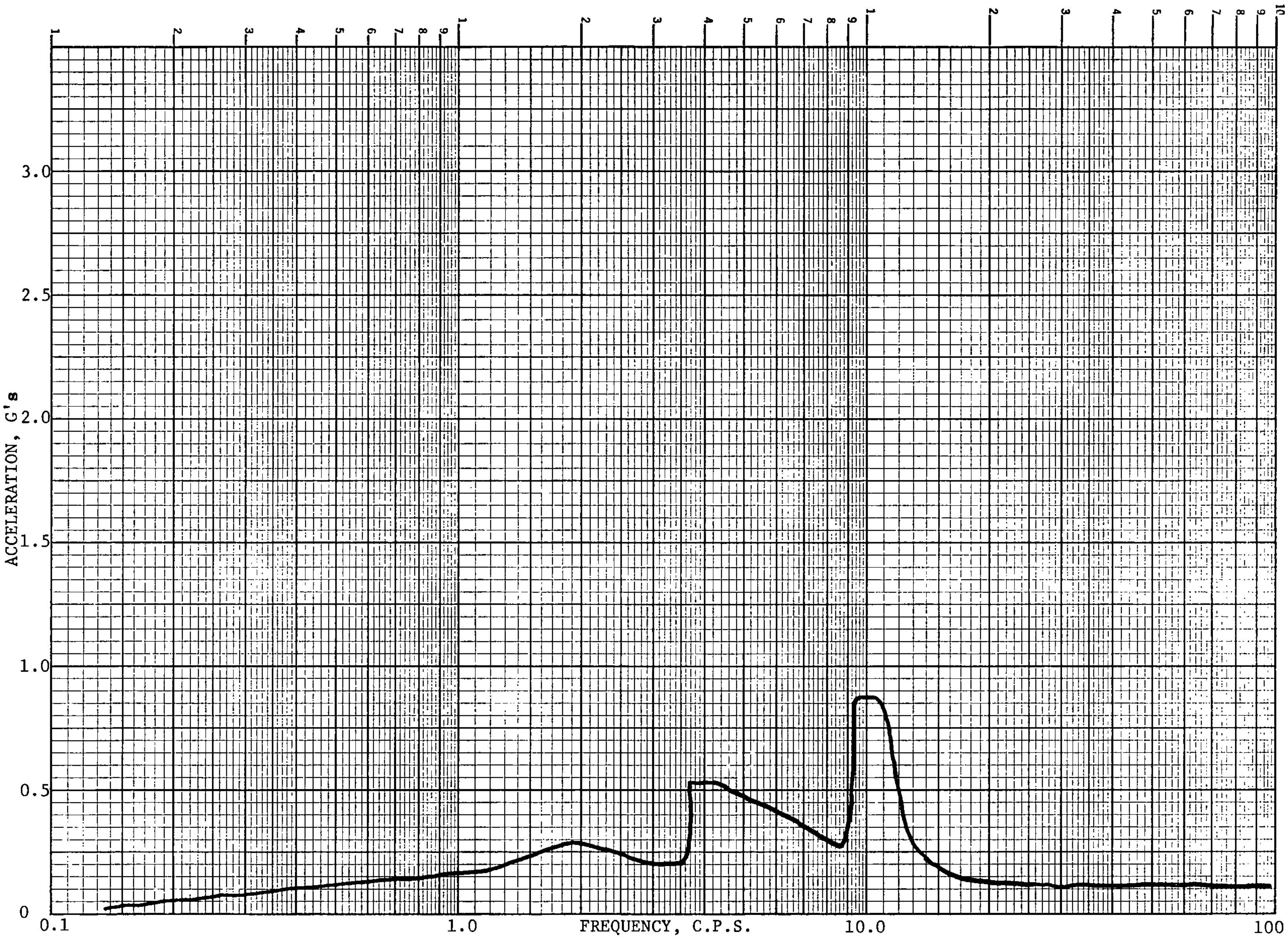
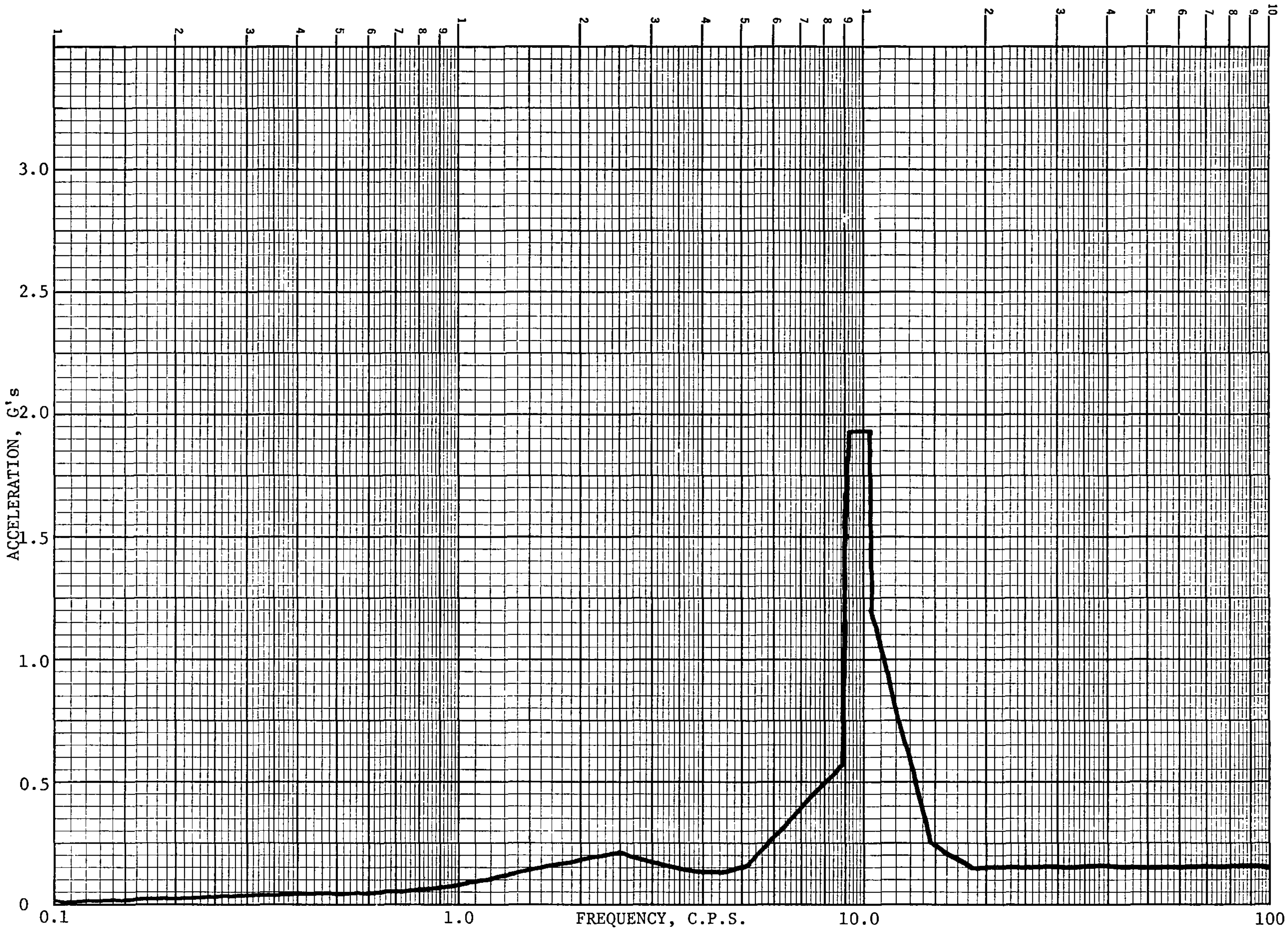
6.



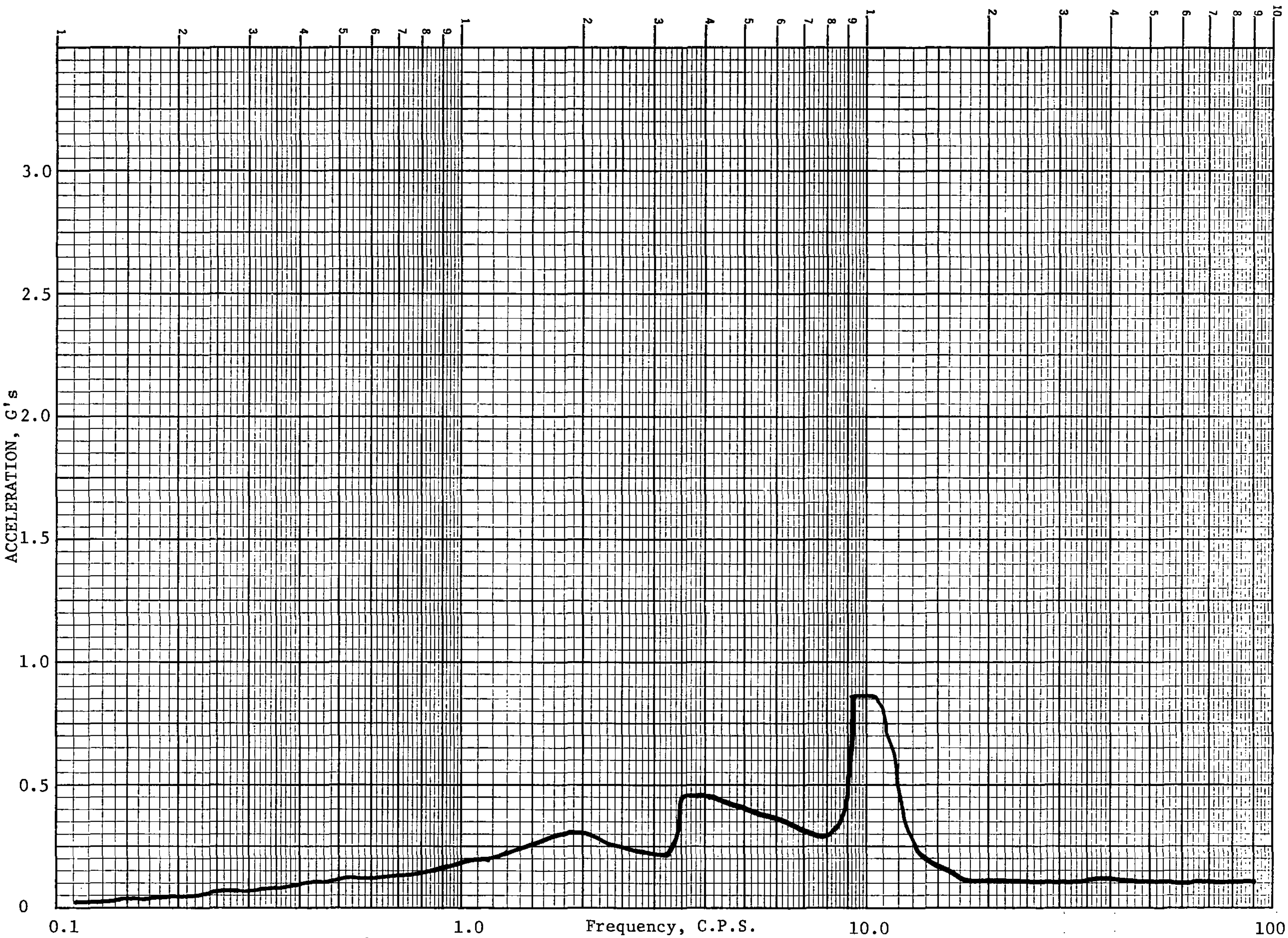
Envelope Response Curve, Component Cooling System

OCONEE NUCLEAR STATION
Figure 1C - 22
(New) Rev. 6 6/22/70

OCONEE NUCLEAR STATION
Figure 1C - 23
(New) Rev. 6 6/22/70



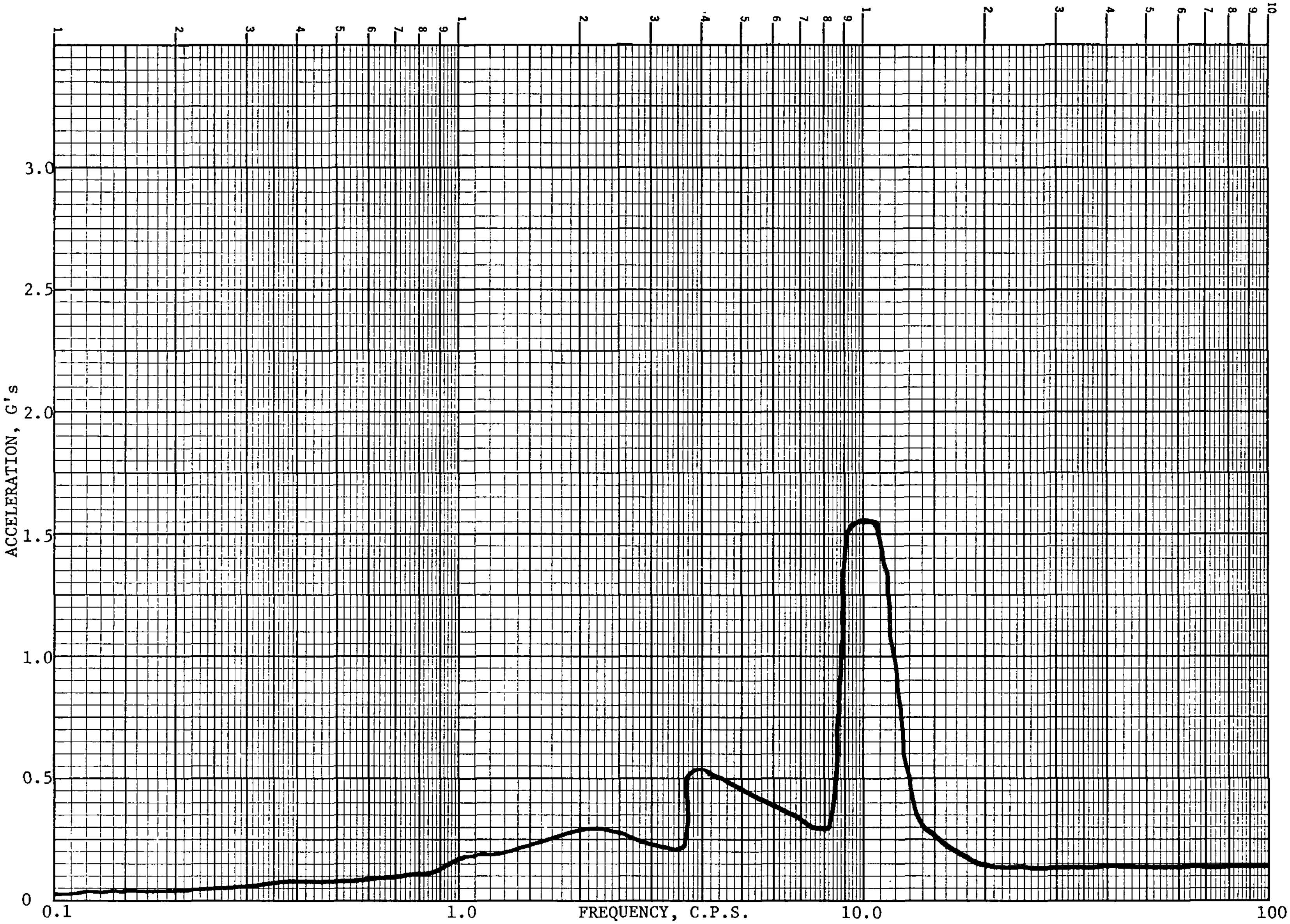
Envelope Response Curve, Pressurizer Spray



Envelope Response Curve - High Pressure Injection, East Coolant Loop

OCONEE NUCLEAR STATION
Figure 1C - 20
(New) Rev. 6 6/22/70

OCONEE NUCLEAR STATION
Figure 1C - 21
(New) Rev. 6 6/22/70



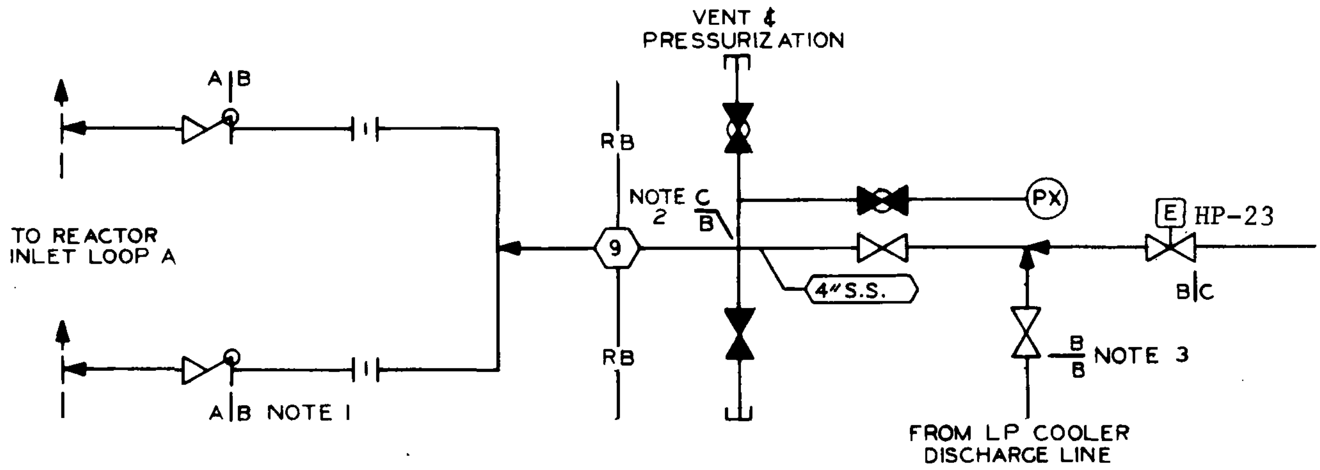
Envelope Response Curve - Low Pressure Service Water

FIGURES 1C-2 THROUGH 1C-19

(Deleted)

Piping classifications are shown on the following figures.

System	Deleted Figure	Classifications Included in Figure
Reactor Coolant	1C-2	4-1
H.P. Inj.	1C-3	9-2, 9-2A
Chem. Add. & Samp.	1C-4	9-3
Comp. Cooling	1C-5	9-4
Spent Fuel Cool.	1C-6	9-5
L.P. Inj.	1C-7	9-6, 9-6A
R.B. Spray	1C-8	6-3
Penet. Room Vent.	1C-9	6-5
HP & LP Ser. Water	1C-10, 1C-11	9-8, 9-9
Recirc. Cool. Water	1C-12	9-10
Coolant Storage	1C-13	9-14
Coolant Treatment	1C-14	9-15
Steam & Power Conv.	1C-15	10-1
Aux. Feedwater Supply	1C-16	10-2
Emerg. Feedwater Pump Turbine Steam	1C-17	10-3
Liq. Waste Disposal	1C-18	11-2
Gas Waste Disposal	1C-19	11-3



16.

NOTES:

1) This notation indicates that piping downstream of check valve and the check valve are "Class A" (See Table 1C-1). Piping and valves upstream of check valve are "Class B" until a change in classification is noted (For Example, change to "Class C" at inlet of Motor-operated valve HP-23). Generally, change in piping classification is made at a valve, with valve included in higher class. 16.

2) Vent & pressurization line is an example of piping one inch and smaller which is excluded from Class 1 and Class 2 (See 1C.3). Vent & pressurization line is "Class C" from connection to 4 inch line until a change in classification is noted (No such change shown here).

3) This notation does not indicate a change in classification, but is included to assure that classification of connection to another system is not omitted.

PIPING AND VALVE CLASSIFICATION LEGEND

OCONEE NUCLEAR STATION

Figure 1C - 1

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TABLE 1C-4 (CONT)

Equipment	Test Description
10. Reactor Bldg. Cooling Fans (Cont)	Motor was disassembled and inspected for deterioration and electrical characteristics in accordance with IEEE bulletin and witnessed by members thereof. Fan considered prototype and representative in design to the fans installed in Duke system.
11. Reactor Bldg. Cooling Coils	Coils design qualified for post loss-of-coolant accident environment as documented by American Air Filter Report RS-1007 per B&W scope of supply
12. Spent Resin Transfer Pump	Performance tests (capacity vs head, efficiency, horsepower)
13. Condensate Test Tank Pump	Performance tests (capacity vs head, efficiency, horsepower)
14. Emergency Feedwater Pumps	Performance tests (capacity vs head, efficiency, horsepower)
15. Emergency Feedwater Pump Turbines	Checkout of actual turbine running including oil temperature and pressure tolerance and vibration measurement. At increased inlet pressure, check for emergency trip (rpm), sentinel valve set (over pressure), low oil pressure alarm, low and high pressure stop speed, maximum bearing temperature, steam leaks, maximum vibration level, and general visual checkout. Each machine received inspection in this manner.
16. Component Drain Pump	Performance tests (capacity vs head, efficiency, horsepower)
17. Forged Steel Valves (Velan)	Seismic qualification of EMO 4" steel gate valve. Pre- and post vibration hydrostatic and operation tests were performed with vibration through varying cycle frequencies and magnitude conducted to insure functional ability to perform. Tests performed for Duke Power with report and summary received.
18. Reactor Coolant Bleed Evaporator Unit Waste Evaporator Unit	Units were operational tested at manufacturer's site at installed conditions. Using anticipated chemical solutions, concentration function and levels were checked to specifications. In all cases, the concentration capability and limits of design solution in the effluent were met.

NOTES TO TABLE 1C-4

1. Above tests do not list NDT type testing such as hydrostatic tests, ultrasonic, dye penetration examinations, etc. Only when these tests were part of the final performance testing of the completely assembled component were they mentioned.

2. Tests for items such #10, #15, #17 are only briefly described. Specific number for seismic loadings, speed, pressure, and temperature are listed in the various reports, B&W file documents, or memos to file concerning witness of these tests, as applicable.

3. Items not noted as qualification or prototype test received testing described for each machine delivered. Qualification tests were for representative or identical machines.

4. Major components in primary system, such as reactor coolant pumps, control rod drives, are not listed due to large amount of testing and qualification data available and presented elsewhere.

5. All testing performed prior to receipt by Duke Power. Does not include on-site testing of such item as filter systems or performance runs of pump-turbine combinations.

Table 1C-3

Summary of Dynamic Analysis of
Letdown Storage Tank

A dynamic modal analysis was performed for the letdown storage tank. A planar model representing the physical characteristics of the tank was constructed with the mass considered to be acting at five discrete points. In addition, the masses were connected by eight elastic members representing the stiffness of the tank and support legs.

The seismic forces were defined in terms of the ground response acceleration spectra for the Oconee site. For components, the spectrum curve of interest is associated with 1% of critical damping. The dynamic response of the letdown tank was then calculated with the resulting inertia loads applied to the structure as concentrated static loads acting at the mass points.

Results of the analysis indicate that the structural integrity of the tank and supporting structure is not jeopardized by seismic loadings.

TABLE 1C-4
PREOPERATIONAL TESTING OF ESSENTIAL SYSTEMS EQUIPMENT

Equipment	Test Description
1. HP Injection Pumps	Performance tests (capacity vs head, efficiency, horsepower, NPSH tests)
2. Component Cooling Pumps	Performance tests (capacity vs head, efficiency, horsepower)
3. LP Injection Pumps	Performance tests (capacity vs head, efficiency, horsepower, NPSH tests)
4. Spent Fuel Pumps	Performance tests (capacity vs head, efficiency, horsepower, NPSH tests)
5. Reactor Building Spray Pumps	Performance tests (capacity vs head, efficiency, horsepower, NPSH tests)
6. High Activity Waste Tank Pumps	Performance tests (capacity vs head, efficiency, horsepower)
7. Low Activity Waste Tank Pumps	Performance tests (capacity vs head, efficiency, horsepower)
8. LP Service Water Pumps	Performance tests (capacity vs head, efficiency, horsepower)
9. Fenetration Room Vent Fans	Performance tests (capacity vs static pressure, horsepower) Overspeed rotor to 130% design operating speed
10. Reactor Building Cooling Fans	Performance test (each fan) covering capacity vs total pressure, blade setpoint, brake horsepower, sound levels Fan design qualification test for operation of fans following an MHA in the containment. Motor for fan was seismic vibrated, checked for electrical characteristics, heat aged, again seismic vibrated, then rechecked for electrical characteristics. After heat aging, motor and fan installed in simulated containment environment including heat shock to 280°F simultaneously with pressure rise condition to 60 psig and introduction of chemical sprays anticipated following an MHA. Test continued for cooldown of seven days. Fan was completely disassembled and inspected.

NEW PAGE
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Table 1C-2 - (Continued)
System Component Classification

	Design Code	Designed For Seismic Loading (D=Dynamic Analysis) (S=Static Analysis)
Steam & Power Conversion System (Pertinent Components Only)		
Condenser	See 10.2.3	Yes - S
Upper Surge Tank	ASME VIII	Yes - S
Emergency Feedwater Pump	-	Yes - Note 1
Emergency Feedwater Pump Turbine	-	Yes - Note 1
Liquid Waste Disposal System		
High Activity Waste Tank	*	Yes - Note 5
High Activity Waste Tank Pump	-	No
Low Activity Waste Tank	*	Yes - Note 5
Low Activity Waste Tank Pump	-	No
Waste Holdup Tank	AWWA D-100	Yes - S
Waste Holdup Transfer Pump	-	Yes - Note 1
Spent Resin Storage Tank	AWWA D-100	Yes - S
Spent Resin Transfer Pump	-	Yes - Note 1
Spent Resin Sluicing Pump	-	Yes - Note 1
Waste Evaporator Feed Tank	AWWA D-100	Yes - S
Waste Evaporator	ASME VIII (lethal)	Yes - S
Recirculating Pump	-	Yes - S
Concentrate Cooler	ASME VIII (lethal)	Yes - S
Separator	ASME VIII (lethal)	Yes - S
Vapor Condenser	ASME VIII (lethal)	Yes - S
Distillate Pump	-	Yes - S
Distillate Cooler	ASME VIII (lethal)	Yes - S
Reactor Building Sump Pump	-	Yes - Note 1
Waste Evaporator Feed Pump	-	Yes - S
Gaseous Waste Disposal System		
Waste Gas Compressor	-	Yes - S
Waste Gas Separator	ASME VIII	Yes - S

* Stainless Steel Lining for Concrete Sump

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(Entire Page)

Table 1C-2 - (Continued)
System Component Classification

	Design Code	Designed For Seismic Loading (D=Dynamic Analysis) (S=Static Analysis)
Gaseous Waste Disposal System (Cont'd)		
Seal Water Cooler	-	Yes - S
Waste Gas Tank	ASME III-C	Yes - S
Waste Gas Filter	-	Yes - S
Waste Gas Exhauster	-	No

Notes

1. Vendor certification that component will meet seismic loading requirement.

2. Static and dynamic analyses performed.

3. Shock tested in lieu of analysis.

4. Vendor certification that component will meet seismic loading requirement will be furnished.

5. Tank meets loading requirement by its location in Auxiliary Building basement floor.

Table 1C-2 - (Continued)
System Component Classification

	Design Code	Designed For Seismic Loading (D=Dynamic Analysis) (S=Static Analysis)
Spent Fuel Cooling System		
Spent Fuel Cooler	ASME III-C & VIII	Yes - Note 2
Spent Fuel Pump	-	Yes - Note 1
Spent Fuel Filter	ASME III-C	Yes - S
Borated Water Recirculation Pump	-	Yes - Note 1
Spent Fuel Demineralizer	ASME III-C	Yes - Note 2
Fuel Transfer Tube	ASME III-B	Yes - D
Incore Instrument Handling Tank	AWWA D-100	Yes - D
Low Pressure Injection System		
LP Injection Pump	See Table 6-5	Yes - Note 1
LP Injection Cooler	ASME III-C & VIII	Yes - Note 2
Borated Water Storage Tank	AWWA D-100	Yes - S
Core Flooding Tank	ASME III-C	Yes - D
Reactor Building Spray System		
Reactor Building Spray Pump	See Table 6-5	Yes - Note 1
Reactor Building Penetration Room Ventilation System		
Penetration Room Filter	See 6.4.2.2	Yes - S
Penetration Room Fan	See 6.4.2.2	Yes - Note 4
LP Service Water System		
LP Service Water Pump	-	Yes - Note 1
Reactor Building Cooling System		
Reactor Building Coolers	See 6.3.2.2	Yes - D

Table 1C-2 - (Continued)
System Component Classification

	Design Code	Designed For Seismic Loading (D=Dynamic Analysis) (S=Static Analysis)
Recirculated Cooling Water System		
RCW Pump	-	No
RCW Heat Exchanger	-	No
RCW Surge Tank	ASME VIII (not code stamped)	No
Coolant Storage System		
Quench Tank	ASME III-C	Yes - Note 2
Quench Tank Cooler	ASME III-C & VIII	yes - Note 2
Component Drain Pump	-	Yes - Note 1
Coolant Bleed Holdup Tank	ASME VIII (not code stamped)	Yes - S
Bleed Transfer Pump	-	Yes - Note 1
Deborating Demineralizer	ASME III-C	Yes - Note 2
Concentrated Boric Acid Storage Tank	USAS B96.1	Yes - S
Concentrate Transfer Pump	-	No
Coolant Treatment System		
Coolant Bleed Evaporator Demineralizer	ASME III-C	Yes - Note 2
Coolant Bleed Evaporator Feed Tank	AWWA D-100	Yes - S
Coolant Bleed Evaporator	ASME VIII (lethal)	Yes - S
Recirculating Pump	-	Yes - S
Concentrate Cooler	ASME VIII (lethal)	Yes - S
Separator	ASME VIII (lethal)	Yes - S
Vapor Condenser	ASME VIII (lethal)	Yes - S
Distillate Pump	-	Yes - S
Distillate Cooler	ASME VIII (lethal)	Yes - S
Condensate Test Tank	USAS B96.1	Yes - S
Condensate Test Tank Pump	-	Yes - Note 1
Condensate Demineralizer	ASME III-C	Yes - S
Coolant Bleed Evaporator Feed Pump	-	Yes - S

Table 1C-2 - (Continued)
System Component Classification

	Design Code	Designed For Seismic Loading (D=Dynamic Analysis) (S=Static Analysis)
Spent Fuel Cooling System		
Spent Fuel Cooler	ASME III-C & VIII	Yes - Note 2
Spent Fuel Pump	-	Yes - Note 1
Spent Fuel Filter	ASME III-C	Yes - S
Borated Water Recirculation Pump	-	Yes - Note 1
Spent Fuel Demineralizer	ASME III-C	Yes - Note 2
Fuel Transfer Tube	ASME III-B	Yes - D
Incore Instrument Handling Tank	AWWA D-100	Yes - D
Low Pressure Injection System		
LP Injection Pump	See Table 6-5	Yes - Note 1
LP Injection Cooler	ASME III-C & VIII	Yes - Note 2
Borated Water Storage Tank	AWWA D-100	Yes - S
Core Flooding Tank	ASME III-C	Yes - D
Reactor Building Spray System		
Reactor Building Spray Pump	See Table 6-5	Yes - Note 1
Reactor Building Penetration Room Ventilation System		
Penetration Room Filter	See 6.4.2.2	Yes - S
Penetration Room Fan	See 6.4.2.2	Yes - Note 4
LP Service Water System		
LP Service Water Pump	-	Yes - Note 1
Reactor Building Cooling System		
Reactor Building Coolers	See 6.3.2.2	Yes - D

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(Entire Page)

Table 1C-2 - (Continued)
System Component Classification

	Design Code	Designed For Seismic Loading (D=Dynamic Analysis) (S=Static Analysis)
Recirculated Cooling Water System		
RCW Pump	-	No
RCW Heat Exchanger	-	No
RCW Surge Tank	ASME VIII (not code stamped)	No
Coolant Storage System		
Quench Tank	ASME III-C	Yes - Note 2
Quench Tank Cooler	ASME III-C & VIII	yes - Note 2
Component Drain Pump	-	Yes - Note 1
Coolant Bleed Holdup Tank	ASME VIII (not code stamped)	Yes - S
Bleed Transfer Pump	-	Yes - Note 1
Deborating Demineralizer	ASME III-C	Yes - Note 2
Concentrated Boric Acid Storage Tank	USAS B96.1	Yes - S
Concentrate Transfer Pump	-	No
Coolant Treatment System		
Coolant Bleed Evaporator Demineralizer	ASME III-C	Yes - Note 2
Coolant Bleed Evaporator Feed Tank	AWWA D-100	Yes - S
Coolant Bleed Evaporator	ASME VIII (lethal)	Yes - S
Recirculating Pump	-	Yes - S
Concentrate Cooler	ASME VIII (lethal)	Yes - S
Separator	ASME VIII (lethal)	Yes - S
Vapor Condenser	ASME VIII (lethal)	Yes - S
Distillate Pump	-	Yes - S
Distillate Cooler	ASME VIII (lethal)	Yes - S
Condensate Test Tank	USAS B96.1	Yes - S
Condensate Test Tank Pump	-	Yes - Note 1
Condensate Demineralizer	ASME III-C	Yes - S
Coolant Bleed Evaporator Feed Pump	-	Yes - S

E. Coolant Boundary Valves

All reactor coolant boundary valves listed above with the exception of those listed under Exceptions below were inspected in accordance with the non-destructive testing requirements of Par. 1-724 of B31.7 - February 1968 Draft for Class 1 or Class II piping.

Where valves had cast discs, these discs were radiographed.

F. Auxiliary System Piping

Portions of the high pressure injection system, low pressure injection system, and core flooding system piping are included within the reactor coolant boundary, as identified by the above listing of valves. This piping is in accordance with Classes II and III of B31.7 - February 1968 draft.

G. Exceptions

Valve	Size	Material Form	Class
HP-5	2-1/2	Forged	C
HP-21	4	Forged	C
HP-20	4	Forged	B
HP-26	4	Forged	B
HP-27	4	Forged	B

16. Valves HP-5, and HP-21 as they are the third and second valves in their respective lines, have been specified as Class C valves as defined in Section 1C.3.

Valves HP-20, HP-26, and HP-27 have been specified as Class B valves as defined in Section 1C.3.

The inspection requirements for these five valves were determined by the requirements of Par. 1-724 of B31.7 for the specified class. As the valves have forged bodies, non-destructive testing was not required. As an added check, the proposed pump and valve code was reviewed for possible additional requirements. This code also does not require non-destructive testing for forged valves for Class II and III service and further excludes valves with inlet piping connections four inches or less in nominal pipe size. It is our conclusion that these valves meet the intent of the proposed changes to Part 50.

H. Non-destructive Testing

Where non-destructive testing was required on valves, acceptance standards for radiographic, ultra-sonic, magnetic particle, and liquid penetrant testing were based on B31.7 standards.

Table 1C-2
System Component Classification

	Design Code	Designed For Seismic Loading (D=Dynamic Analysis) (S=Static Analysis)
Reactor Coolant System		
Reactor Vessel	ASME III, Class A	Yes - D
Pressurizer	ASME III, Class A	Yes - D
Reactor Coolant Pump Casing	ASME III, Class A (not code stamped)	Yes - D
Steam Generator	ASME III, Class A	Yes - D
High Pressure Injection System		
HP Injection Pump	See Table 6-5	Yes - Note 1
Letdown Cooler	ASME III-C & VIII	Yes - D
Seal Return Cooler	ASME III-C & VIII	Yes - Note 2
6. Letdown Storage Tank	ASME III-C	Yes - Note 2
Purification Demineralizer	ASME III-C	Yes - Note 2
Letdown Filter	ASME III-C	Yes - Note 2
2. RC Pump Seal Filter	USAS B31.7, Paragraph 2-724, Class II	Yes - Note 3
Chemical Addition and Sampling System		
Boric Acid Mix Tank	USAS B96.1	No
Lithium Hydroxide Mix Tank	-	No
Caustic Mix Tank	-	No
Boric Acid Pump	-	No
Lithium Hydroxide Pump	-	No
Hydrazine Pump	-	No
Caustic Pump	-	No
Pressurizer Sample Cooler	ASME III-C & VIII	No
Steam Generator Sample Cooler	ASME VIII	No
Component Cooling System		
Component Cooling Pump	-	Yes - Note 1
Component Cooler	ASME VIII	Yes - Note 2
Component Cooling Surge Tank	AWWA D-100	Yes - S
CRD Cooling Coil Filter	ASME VIII	Yes - S

4. (Moved to Page 1C-4a)

1C.4 SYSTEM DIAGRAMS SHOWING PIPING AND VALVE CLASSIFICATION

The criteria, for normal and unusual conditions, which are applied in design of systems are discussed in appropriate sections of the FSAR. Diagrams for each system, reflecting the application of these criteria in piping and valve
16. classification, are included in the FSAR sections where each system is described (i.e., Sections 4, 6, 9, 10, and 11). Piping and valve classifications are indicated on the figures as explained in Figure 1C-1.

A comparison between the codes and standards required by the proposed change to the AEC regulations published in Volume 34, No. 226, of the Federal Register dated November 25, 1969, and those used on Oconee 1, 2, and 3 have been made for the reactor coolant pressure boundary. The reactor coolant boundary, as defined by the aforementioned proposed rule making includes the system defined as the "reactor coolant system" and connecting auxiliary piping including or terminating as described by the valves below.

The system used on Oconee 1-3 and described in this section for classifying piping and valves, while not in strict accordance with the proposed rule making, is a consistent system which meets or exceeds the requirements of the proposed changes to Part 50.

2.

16.

Figure	System	Valve
4-1	Reactor Coolant System	RC-67
		RC-68
		RC-4
		RC-66
		RC-1
		RC-3
9-2 9-2A	HP Injection System	HP-26
		HP-27
		HP-120
		HP-20
		HP-21
		HP-2
		HP-3
		HP-4
		HP-5
		HP-31
		HP-1

Figure	System	Valve
4-1	Core Flooding System	Two Check Valves
9-6 9-6A	Low Pressure Injection System	LP-17 LP-18 LP-1 LP-2

16.

Piping and fittings, 1-1/2" nominal pipe size and below have not been considered in the comparison.

Valves, 2" nominal pipe size and below have not been considered in the comparison. The results of the comparison are summarized below:

A. Pressure Vessels

The codes and addenda used for pressure vessels in the reactor coolant pressure boundary are equivalent to or later than those specified by the proposed rule making.

Vessel	Code and Addenda Used
Reactor Vessel	
Unit 1 2 3	ASME III, Class A, Summer 1967
Steam Generator	
Unit 1 2 3	ASME III, Class A, Summer 1967
Pressurizer	
Unit 1 2 3	ASME III, Class A, Summer 1967

B. Reactor Coolant Piping

The reactor coolant system piping is in accordance with B31.7 including errata through June 1968.

C. Reactor Coolant Pumps

The pressure boundary portions of the pump and pump casing were designed and inspected in accordance with ASME Section III for Class "A" Vessels including the Summer 1967 Addenda.

D. Letdown Cooler

The reactor coolant side of the letdown cooler is in accordance with ASME Section III for Class "C" vessels, including the Winter 1966 Addenda.

(5) Instrument impulse line installation is inspected by an independent group on site and stamped approval and signoff are required before the system can be turned over to operating personnel.

(6) Instrumentation testing programs are well defined by Duke test procedures. These tests document conclusively that the instrument loops are correctly installed and operate properly.

(7) Engineering frequently visits the site and inspects instrumentation installations with corrective measures as necessary originating with the designer.

(8) Both "field run" piping and impulse line installation work are scrutinized by three independent parties other than the erection party. They are the Construction QC Group, Operating personnel and Design Engineering.

This practice of "controlled field routing" of small piping and instrument impulse lines produces the best possible overall results. It is not practical to limit "field run" piping to a greater extent.

1C.3.6 SAFETY AND RELIEF VALVES

Design analysis and installation criteria for safety and relief valves located within the reactor coolant and main steam (thru main stop valves) pressure boundaries are as follows:

a. Piping and its Support-Restraint System are designed to accommodate and/or restrain the piping for both dynamic and static loadings as applicable such that stresses produced are within code allowables for the following:

   (1) Dead weight effect
   (2) Thermal loads and movements
   (3) Seismic loads and deflections - movements
   (4) Safety valve thrust and moment
   (5) Maximum absolute differential movement between structures

   Applicable loadings are combined and considered as described in 1C.3.4.

b. Nozzles are analyzed and appropriate reinforcement added such that code allowable stresses are maintained for:

   (1) Internal pressure
   (2) Safety valve thrust
   (3) Safety valve moment

In particular, for the main steam lines outside the Containment, pressure relief is accomplished through the use of a single relief valve and sufficient safety valves to meet code requirements. The safety valves are set for progressive relief in intermediate steps of pressure within the allowed range of pressure settings to prevent more than two valves actuating simultaneously. Valves are located on a horizontal run of pipe and are oriented in a manner that will produce torsion and bending in the main pipe during operation of the valves. The valves are staggered on opposite sides of the main steam line and set to relieve progressively to counterbalance the torque produced so that the maximum net torque on the piping system is essentially only that which results from the discharge of two safety valves. See Figure 1C-31 for additional clarification of this arrangement. For conservatism, the piping system is designed to accept the net torque resulting from three safety valves operating simultaneously on the same side of the line. The piping support and restraint system is designed using shock suppressors and rigid stops to limit piping system stresses within code allowables as discussed above.

Dynamic thrust effects were analyzed for the Reactor Coolant System pressurizer relief discharge line to the Quench Tank, constituting a closed system. Stresses produced by the thrust effects were within the established Code allowables for the station. No other safety-related closed systems exist for Oconee.

1C.3.7 SYSTEM VIBRATION OPERATIONAL TEST PROGRAM

It is Duke's normal practice and a startup procedure consideration to put essential and safety related systems thru all of their normal and emergency modes of operation, visually observing the system for excessive movement and/or vibration. Based on operational reports indicating possible excessive movement and/or vibration, the Steam Production Department requests Engineering review of each case. Engineering observes the system making necessary measurements, readings, etc., as required to analyze the problem against existing design stress analysis and design criteria. Based on this analysis, Engineering either approves the system as satisfactory or requiring additional design consideration. Additional supports or suppressors are designed to accommodate the effects of valve closures, pump trips, safety valve operations, and operational vibrations as required. Any problems defined for any unit are reviewed and corrected for all three units as required.

Although not required for the Oconee project, Duke will conduct prior to station
24. startup the following monitoring programs which are typical of Engineering reviews as discussed above for the purpose of comparing results with design analysis.

a. Thermal Movement Monitoring Program for the Reactor Coolant System Piping (Data will be taken on Unit 1 only; however, the report will be qualified for all three units.)

b. Thrust Movement Monitoring Program for the Main Steam Bypass to Condenser Piping (Data will be taken on Unit 1 only; however, the report will be qualified for all three units.)

c. Hanger and Restraint Setting Monitoring Program for the LP Injection System (Data will be taken on Unit 1 only; however, the report will be qualified for all three units.)

1C.3.8 EQUIPMENT PRE-OPERATIONAL TEST PROGRAM

Selected system equipment for the Oconee station has undergone some meaningful and related testing in manufacturer's shops, in special testing laboratories, or in the installed position at the station. Table 1C-4 is a tabulation of the specific equipment tested and a description of the type test conducted. Tests described in Table 1C-4 were conducted prior to Duke's acceptance of the equipment and further performance type tests required of equipment in installed system are not described in Table 1C-4. Tests conducted on the installed equipment are described in the station test procedures.

1C.3.5 FIELD-ROUTED PIPING AND INSTRUMENTATION

Duke's practice is to detail the routine of all safety-related process lines regardless of size, except as follows:

a. Process piping - All main run process piping in Duke System Classifications A, B, C, D, E, & F is detailed on engineering drawings; however, items such as vents, drains, valve bypass warming lines and pump seal water for all systems are "field run".

   Instrument impulse lines - end points and specific routing requirements of any safety-related instrument impulse line are established and defined by Engineering, and the actual path is established in the field by Construction. At Oconee, the one exception and improvement is that the reactor coolant flow impulse lines were detailed-routed by Design Engineering.

b. It is not practical to limit "field run" piping to an extent greater than this for the following reasons:

   (1) Obstructions to desirable routing would be difficult to determine and documentation of a precisely designed path would be lengthy, difficult to prepare and difficult to follow.

   (2) Revision to major process piping would cause changes in routing of small lines, resulting in many drawing changes without significant improvement in the final result.

   (3) Sloping of impulse lines would be difficult to accomplish and document.

   Thus, field routing of small lines results in a superior job since obstructions and other design revisions are clearly visible and easier to consider while meeting design requirements as established by Engineering.

c. The special rigorous quality assurance measures and performance tests that will be conducted to assure satisfactory installation of field run piping and instrument impulse lines are as follows:

   (1) All field engineered lines are schematically shown either on a diagrammatic, an instrumentation detail or a piping drawing such that mistakes in valving, connection termination points and materials are virtually eliminated.

   (2) All field run piping requiring seismic design is reviewed after erection by appropriate Engineering Department personnel and applicable seismic controls are detailed and forwarded to the Construction Department for installation.

   (3) Except for very low pressure lines downstream of vent and drain valves and instrument impulse lines, all "field run" piping is hydrostatically tested in accordance with the requirements of the main process system.

   (4) An engineering surveillance program is conducted after erection to review all safety-related piping as well as non-safety related piping in the area to assure that appropriate criteria have been followed.

1C.3.4.4 Component Seismic Design Assurance

When the response spectra at each elevation in the building has been determined, the G-loadings imposed on a component may then be determined. These loads are evaluated by the equipment supplier and in the case of complex components such as a heat exchanger, the design calculations performed by the supplier are reviewed by B&W Engineering or Duke, as applicable. The supplier has the freedom to use either of two alternate analytical methods to evaluate the equipment or he may choose to test it. Components may be tested by either shaker or impact tests and a certification of the test results are required. In a few cases, a manufacturer's certification that the equipment would withstand seismic conditions is acceptable based on tests of similar equipment, an example of this would be similar type pumps. Analytically the evaluation can be made by calculating the natural frequency of the component, entering the appropriate damping curve and determining the amplification factor from the response spectrum curve. The equipment is then evaluated using these G-loadings. As an alternate, the component may be evaluated without calculating the natural frequency by using the peak amplification factor from the appropriate damping curve to determine the equipment loads. This latter approach is conservative.

Special attention is given to foundation and nozzle loadings for equipment such as tanks, pumps, heat exchangers, demineralizers and filters. Loads imposed by connecting piping on a given component are included and in some cases, component nozzles have had to be reinforced to accommodate these loads. Components which are most likely to require special reinforcement due to seismic loads are long horizontal, saddle mounted tanks, vertical tanks mounted on legs, and stacked heat exchangers. These have all been evaluated and appropriately designed for the seismic conditions.

1C.3.4.5 Piping Seismic Restraint Location Quality Assurance

The location, type, and design loads of restraints as determined by system design analysis are indicated on prints of plan drawings and forwarded to the hanger designer for his use in the design of the restraints. The hanger designer produces the necessary sketches describing the restraints and sends them to Duke for approval. Duke checks the sketches against the design information for location, type, and design loads. Approved sketches are forwarded to the field for erection.
21. For the Oconee Project after the piping is erected, appropriate erection and function of seismic restraints are assured by several means:

a. The Construction Department checks erection against design sketches.

b. For hangers inside the Containment, an additional QA surveillance of the system and supports is made for proper functional aspects by the analytical engineer responsible for the design. This surveillance is documented.

c. A QA surveillance by the Hanger-Contractor is conducted of all safety-related systems with respect to correct location, direction of action, and hardware being properly installed. This surveillance is documented.

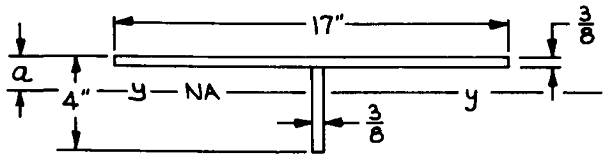
1C.3.4.6 Seismic/Non-Seismic Piping Interactions

For the Oconee Project, Duke has not distinguished piping as Category 1 or 11 but rather as seismic or non-seismically designed. The interaction between seismic/non-seismic lines has been considered and safety system integrity is assured by the following methods:

a. Seismic/non-seismic lines are physically separated insofar as possible such that failure of a non-seismic line has no effect on safety-related piping.

b. Seismic/non-seismic boundaries are established by valves which are designed to meet the seismic design criteria. Failure in the non-seismic portion of the system cannot cause loss of function to the safety system in that automatic or remote manual-operated valves are used for valves normally open during Reactor Operation.

c. The seismic/non-seismic boundary valve is protected from seismic effects by restraining or anchoring the non-seismic portion of the system downstream of the valve.



AREA

$17 \times \frac{3}{8} + 3.625 \times \frac{3}{8}$

$= 6.4 + 1.34 = 7.74 \text{ IN.}^2$

$6.4 \times .186 = 1.19$

$1.34 \times 2.188 = \underline{2.93}$

4.12

$a = 4.12/774 = .534 \text{ IN.}$

$I_{y-y} = 6.4(.534 - .186)^2 + 17 \times .375^3/12 + 1.34 \times (2.188 - .534)^2 + .375 \times 3.625^3/12$

$= .775 + .075 + 3.66 + 1.49 = 6.00 \text{ IN}^4 \quad S.M._{y-y} = \frac{6}{4 - .534} = 1.73 \text{ IN}^3$

STRESSES IN SUPPORTS

COMPRESSIVE	15450/7.74	= 2000 PSI
MAX SHEAR	$\sqrt{1195^2 + 4050^2}/7.74$	= 545 PSI
BENDING STRESS	20250/.173	= 11700 PSI

MAX STRESS FROM SIMULTANEOUS HORIZONTAL AND VERTICAL ACCEL

2000 + 11700 = 13700 PSI

LOADS ON SHELL AT LOWER SUPPORTS

DIRECT SUPPORT LOAD	15450 LBS
LONGITUDINAL MOVEMENT	4050 X 6 = 24300 IN/LBS
INTERNAL PRESSURE	100 PSI

EVALUATION OF LOCALIZED STRESSES IN SHELL AT LOWER SUPPORT USING O ROURKE FORMULAS FOR STRESS AND STRAIN - EDITION 1954 - PAGE 282

$K = .02 - .00012(150 - 90) = .0128$

EQUIVALENT DIRECT LOAD ON SHELL

$15450 + 24300/\frac{2}{3} \times 10 = 19090 \text{ LBS}$

DURING SIMULTANEOUS HORIZONTAL AND VERTICAL ACCELERATION

$.0128 \times \frac{19090}{.25^2} \text{ Ly } \frac{20}{.25} = 17100 \text{ PSI}$

STRESS IN SHELL FROM PRESSURE

$\frac{100 \times 9.75}{.25} + .6 \times 100 = 3960 \text{ PSI}$

THE SHELL WILL NOT COLLAPSE

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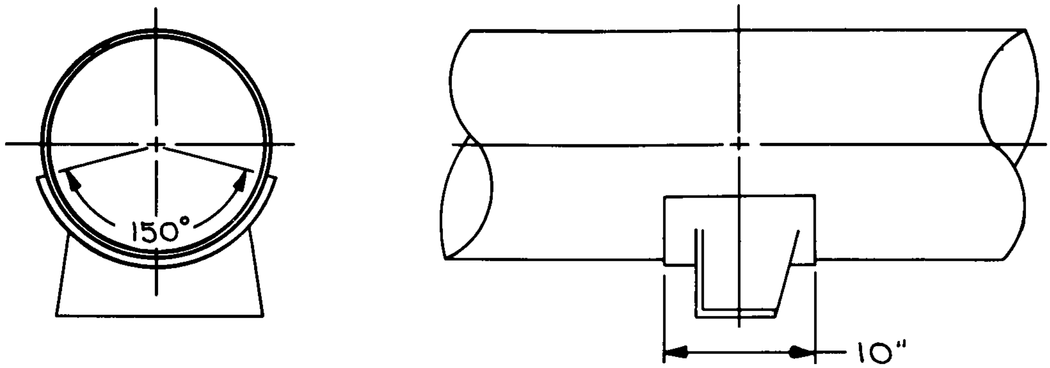
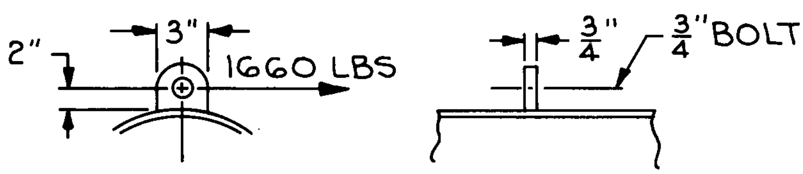
TYPICAL SUPPORT SADDLE



SEISMIC LUG



MOMENT $2 \times 1660 = 3320$ IN LBS
SM OF LUG $\frac{3}{4} \times 3^2/6 = 1.13$ IN^2
STRESS IN LUG $3320/1.13 = 2930$ PSI

EQUIVALENT DIRECT LOAD ON SHELL
$2 \times 3320/\frac{2}{3} \times 3 = 3320$ LBS

MAX. LOCAL STRESS IN SHELL
$.03 \times \frac{3320}{.25^2}$ $L_y \frac{20}{.25} = 7160$ PSI

STRESS IN SHELL FROM PRESSURE 3960 PSI
NO PAD REQUIRED

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3) Systems 51A and 59 Problem #020 — Fig. 1C-29 High Pressure Injection System - Letdown Cooler Inlet

Fig. 1C-30 Reactor Building Component Drains - West Coolant Loop

(c) General Input

The input to this program consists of the input required for ME 553 - the Piping Flexibility Analysis Program (Option 000), additional data describing the choice of mass points, data cards requesting dynamic analysis, and the floor response spectrums.

The user has the option of either allowing the program to compute the weight at each mass point or inputing the weight at each mass point. Relation between mass point input and mass matrix output: the sequence of the resulting mass matrix is in reverse order of the mass input as assigned in data and proceeds from the last (BINP) branch intersection point to the first. The first third of the mass matrix corresponds to the X directions in global cartesian co-ordinates, the second third corresponds to the Y directions, and the last third to the Z directions.

1C.3.4.2.4 List of Figures

The following list of figures represent typical seismic analysis models utilized in connection with the foregoing sections of 1C.3.4.2 for the dynamic analysis of class A, B, C, and F piping systems within the Reactor building.

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1C.3.4.3 Static Seismic Analysis Examples

1C.3.4.3.1 Systems

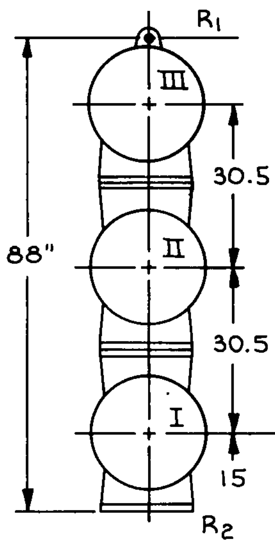
Duke Engineering Design Report, "Static Method of Seismic Analysis of Piping Systems for Oconee 1, 2, 3," File #OS-27-B, dated June 6, 1970, describes the approach and a sample problem for nuclear piping outside the Reactor Building.

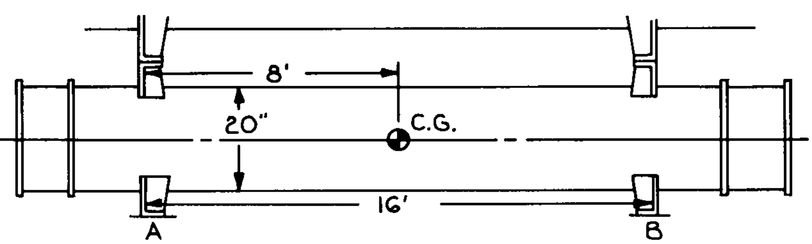
1C.3.4.3.2 Components

The following seismic analysis of the component coolers is an example of the static analysis applied to components in Class B, C, and F Systems outside the Reactor Building.

SEISMIC ANALYSIS OF COMPONENT COOLERS

WEIGHT FULL OF WATER 9,000 LBS EACH UNIT
VERTICAL SEISMIC LOAD 1.15 X 9000 = 10,300 LBS/IN
HORIZONAL SEISMIC LOAD .15 X 9000 = 1350 LBS/IN





FOUNDATION LOADS

VERTICAL (DOWNWARDS): 10300 X 3/2 = 15,450 LBS AT EACH SUPPORT
LONGITUDINAL: 1350 X 3 = 4050 LBS AT SUPPORT "A"
0 LBS AT SUPPORT "B" (SLOTTED HOLES)

LATERAL

$R_1 = 2 \times 675/2 \times 88^3 \times (15^2 [73 + 2 \times 88] + 45.5^2 [42.5 + 2 \times 88] + 76^2 [12 + 2 \times 88]) = 1660$ LBS = TOTAL LOAD ON SEISMIC LUG

$R_2 = 3 \times 1350/2 - 1660/2 = 1195$ LBS EACH SUPPORT ON FOUNDATION

LATERAL MOVEMENT

$675/2 \times 88^2 ([73 + 88] + [42.5 + 88] + [12 + 88]) = 16.7$ IN/LBS
(NEGLIGIBLE)

LOADS ON SUPPORTS OF UNIT I

VERTICAL (COMPRESSIVE)	15450 LBS
LATERAL (SHEAR)	1195 LBS
LONGITUDINAL (SHEAR)	4050 LBS
LONGITUDINAL MOMENT	4050 (15-10) = 20250 IN/LBS
LATERAL MOMENT	NEGLIGIBLE

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It is a program that consists of two other programs. These are:

ME 553 - Piping Flexibility Analysis (Option 000), and
CE 548 - Symbolic Matrix Interpretive System

ME 601 provides the user with the flexibility, stiffness and mass matrices, and mode shapes and frequencies.

The piping system is idealized into a number of lumped masses connected by elastic members. These elastic members will include the effects of torsion, bending, shear, and axial deformation. Bends and elbows are assigned a flexibility factor based on USAS B 31.1.0 - 1967.

The program utilizes ME 553, Piping Flexibility Analysis, to obtain the flexibility matrix [F] and the mass matrix [Mi]. The flexibility matrix is then inverted to produce the stiffness matrix [K]. The stiffness and mass matrices are utilized to solve the characteristic value problem for the natural frequencies and mode shapes of the system by the equation

$$[K_i]\,[\vec{\Phi}_{ij}] - \omega_j^2\,[M_i]\,[\vec{\Phi}_{ij}] = 0$$

The spectrum response curves of Saj for both the vertical and horizontal directions are employed in the program. After the natural frequencies ω have been calculated, the corresponding spectral acceleration is interpolated from the given floor spectrum response curves for the piping system. ME 601 uses the concept of modal analysis and spectrum response curves for earthquake analysis.

The equations of motion as formulated and solved for the mode shapes and frequencies, yielding $\ddot{q}_j = \omega_j^2\, q_j$ where:

ω_j is the circular frequency of the j^{th} mode

q_j is the generalized coordinate of the j^{th} mode

$\ddot{q}_j$ is the generalized acceleration of the j^{th} mode

The maximum generalized acceleration is calculated from the expression:

$$\ddot{q}_{j\,max} = \frac{\left[S_{ajx}\sum_{(i=1)}^{N}\phi_{ij}(x)M_i\right] + \left[S_{ajy}\sum_{1}^{N}\phi_{ij}(y)M_i\right] + \left[S_{ajz}\sum_{1}^{N}\phi_{ij}(z)M_i\right]}{\sum_{i=1}^{N} M_i\,\phi_{ij}^2(x) + \sum_{i=1}^{N} M_i\,\phi_{ij}^2(y) + \sum_{i=1}^{N} M_i\,\phi_{ij}^2(z)}$$

M_i is the mass lumped at each point

$\dfrac{\sum M_i\,\vec{\phi}_{ij}}{\sum M_i\,\vec{\phi}_{ij}^{\,2}}$ is the participation factor

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$\phi_{ij}(x), \phi_{ij}(y), \phi_{ij}(z)$ are the column vectors for the j^{th} mode shapes in X, Y, Z directions

S_{aj} is the acceleration from the spectrum curve at the natural frequency for the j^{th} mode

N is the number of modes used

The accelerations at each point are calculated by:

$$\ddot{e}_{ix} = \sqrt{\sum_{j=1}^{N} [\phi_{ij}(x) q_j]^2} \quad \ddot{e}_{iy} = \sqrt{\sum_{j=1}^{N} [\phi_{ij}(y) q_j]^2} \quad \ddot{e}_{iz} = \sqrt{\sum_{j=1}^{N} [\phi_{ij}(z) q_j]^2}$$

where: $\ddot{e}_{ix}, \ddot{e}_{iy}, \ddot{e}_{iz}$ are the accelerations in each direction at mass

The seismic forces are computed by the equations

$$P_{ix} = M_i \ddot{e}_{ix}, \quad P_{iy} = M_i \ddot{e}_{iy}, \quad P_{iz} = M_i \ddot{e}_{iz}$$

where: P_{ix}, P_{iy}, P_{iz} are the X, Y, and Z force components at the i^{th} mass.

These seismic forces are calculated in each directional axis of the global coordinates of the pipe. The final step of computing the piping system deflections, rotations, and stresses is then completed by calling the ME 553 program and providing it with the seismic forces just calculated. For this analysis, the three force components are assumed to act simultaneously each in its own positive direction.

(b) Problem Definition

A problem is described to the program in accordance with the input description of program ME 553. First, an isometric sketch of the system, in cartesian coordinates, is prepared, and elements are assigned data points. Lumped mass points are then assigned at selected points to adequately describe the system. This is done by assigning a mass point in each change of direction and near weight concentrations to develop a mathematical model. One (1) copy of each of the following three (3) mathematical models are attached for purposes of example.

1)	System 01A Problem #002	Fig. 1C-24	Main Steam System - West Generator (Thermal and Weight)
		Fig. 1C-25	Main Steam System - West Generator (Seismic)
2)	Systems 53A and 59 Problem #003	Fig. 1C-26	Core Flooding Tank 1A
		Fig. 1C-27	Low Pressure Injection System - West
		Fig. 1C-28	Reactor Building Component Drains West Coolant Loop

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1C.3.4.2.2 Seismic Analysis of Piping Systems

(a) Section 1, Rigid Systems

A piping system can be classified as rigid if its first mode natural frequency is higher than 30 cps.

(b) Section 2, Flexible Systems

The response of a flexible system to seismic forces depends upon its natural frequencies and the frequencies of excitation. For these systems, it is necessary to know the natural frequencies and the seismic excitation usually defined as acceleration response spectrum.

Essentially, an acceleration response spectrum is a curve representing G levels over a range of natural frequencies. The curve is obtained by the following procedure. An acceleration time-history is applied at the base of a single degree of freedom system. Initially the system is set with specific values for its natural frequency and damping. The time-history response of the mass is determined and examined for the value of maximum acceleration. The same process is repeated over a range of natural frequencies. The resulting maximum G levels and frequencies are tabulated and plotted into the spectrum curve for a single structure elevation. The resulting curve is labeled with the damping value. The process is repeated for required structure elevations and damping values. For piping systems, the spectrum curve used is associated with 1/2 percent of critical damping. The range of natural frequencies is chosen to correspond with the range of seismic excitation frequencies, 0.1 to 30 cps.

A sample of the acceleration spectrum curves at different floor levels of a building is shown on page 1C-4f. For these curves, the horizontal axis is logarithmic in cycles per second and the vertical axis is linear in G's. The curves are for 1/2 percent of critical damping. The building has natural frequencies of 4.8 cps at the first mode and 10 cps at the second mode. Thus maximum accelerations occur between 1.0 cps and 10.0 cps. At the far right end, the curve converges on the peak value of the input earthquake as the single degree of freedom system becomes rigid, relative to the seismic excitation. At progressively higher locations, the building amplifies the input earthquake, especially in the vicinity of its natural frequencies. Note the sharp peak in each curve at the natural frequency of the building.

When using response curves for piping systems which are located at different elevations, it is necessary to superimpose several curves and plot the envelope curve for the system inputs. At the maximum acceleration peak of each specific curve used for the envelope curve, the envelope has a plateau of approximately ± 10 percent to avoid the condition where a small change in frequency could result in a significant change in acceleration. Through the ME 601 program, the natural frequency and mode shapes of the pipe are found and combined with the spectrum curves to find the seismic forces on the pipe.

1C.3.4.2.3 Technical Description of Piping System Seismic Analysis Program

(a) Introduction

ME 601 is a computer program for performing seismic analysis of piping systems.

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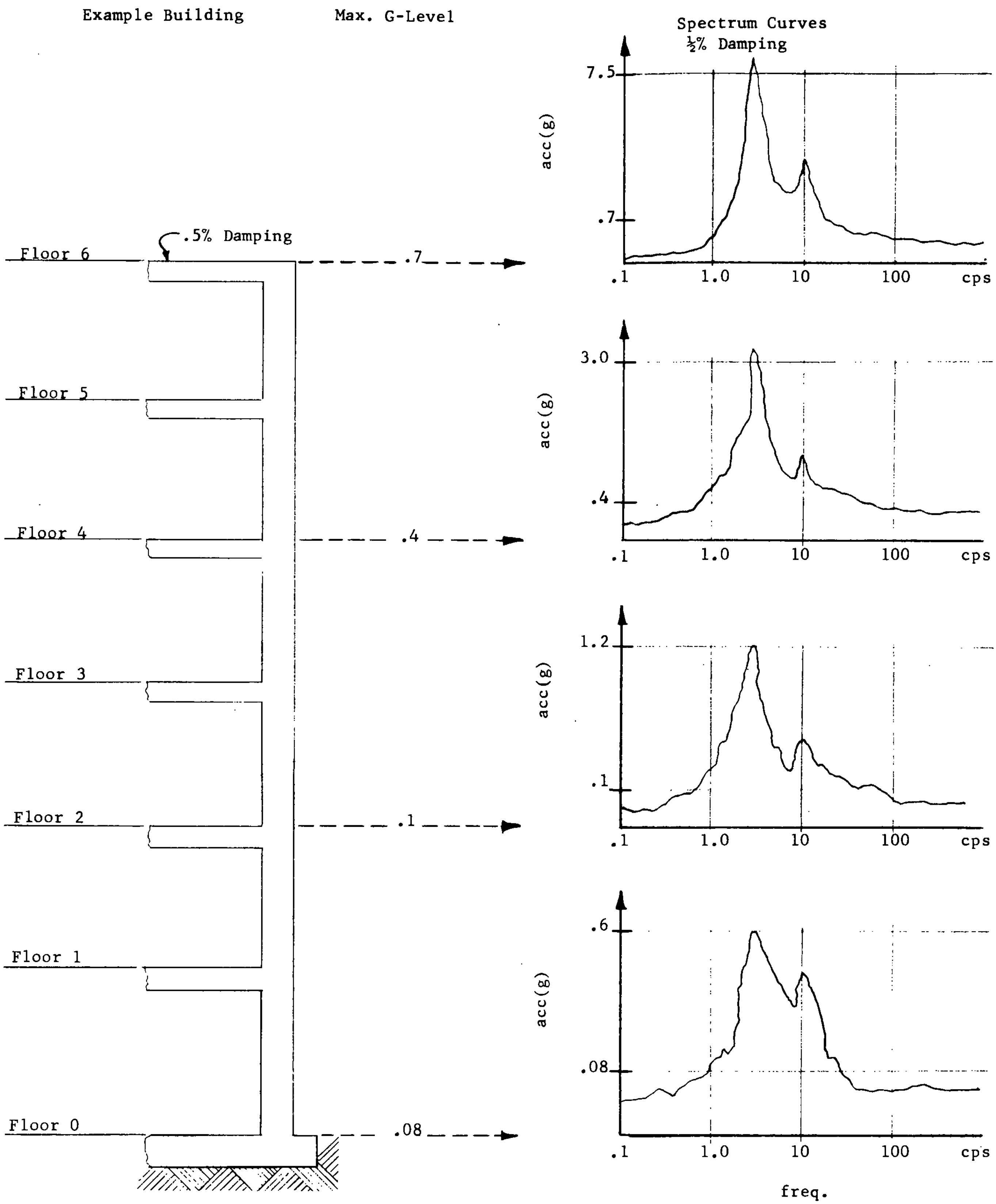
EXAMPLE SPECTRUM CURVES



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in which:

$Y_n max$ = response of the n^{th} mode
R_n = participation factor for the n^{th} mode = $\Sigma M_i \phi_{in}$
Sa_n = spectral acceleration for the n^{th} mode
D = earthquake direction matrix
M = generalized mass matrix for the n^{th} mode = $\Sigma M_i \phi_{in}^2$

Using these results, the maximum displacements for each mode are calculated for each mass point in accordance with the following equation:

$$V_{in} = \phi_{in} Y_n \max$$

in which:

V_{in} = maximum displacement of mass i for mode n

The total displacement for each mass is determined by taking the square root of the sum of the squares of the maximum deflection for each mode:

$$V_i = \sqrt{\Sigma V_{in}^2}$$

in which:

V_i = maximum displacement of mass i due to all modes calculated

The inertia forces for each direction of earthquake for each mode are then determined from:

$$Q_n = K V$$

in which:

Q_n = inertia force matrix for mode n
V = displacement matrix corresponding to Q_n

Each mode's contribution to the total displacements, internal forces, moments and stresses are determined from standard structural analysis methods using the inertia forces for each mode as an external loading condition. The total combined results are obtained by taking the square root of the sum of the squares of each parameter under consideration, in a manner similar to that done for displacements.

1C.3.4.2 Dynamic Seismic Analysis Method of Reactor Building Piping

As discussed in 1C.3.4.1, the following discussion represents the methods utilized to determine the input for piping, the analytical procedures used and sample seismic analysis methods.

1C.3.4.2.1 Response Spectra Determination

The analysis procedures that were used to determine that the Class I piping within the containment structure would satisfy the seismic criteria were based on the time-history method for constructing response spectra curves. A response spectrum can be defined as a plot of the maximum response experienced by a single degree of freedom system subject to varying natural periods of vibrations.

The initial step for developing response-spectra is to idealize the actual structural system as a mathematical model in the form of a lumped mass system interconnected by elastic members. Lumped masses, which are a summation of structure and equipment masses, are located at pertinent floor levels and at other levels where response spectra are desired. These other levels would include equipment support elevations, pipe support elevations, etc.

In the case of the Duke reactor building, two mathematical models were generated to describe the complete reactor building. The first model represented the reactor building shell, and the second model represented the internal structure. Modifications to each model were required to determine response from ground motions. After these models were developed, the following procedure was employed for all the models.

The flexibility matrix of the structural system was determined by using Bechtel program CE309. The procedure for each loading condition was to apply a unit load at each mass point and determine the deflection at all mass points.

The mode shapes and frequencies for the lumped mass systems were obtained by means of either of two Bechtel computer programs; CE548, "Symbolic Matric Interpretive System" or CE617, "Diagonalization Method for Eigenvalues and Eigenvectors".

The Time-History record of the N-S, May 1940 El Centro earthquake was used (vertical and N-S horizontal components) at 0.01 second intervals for the first 30 seconds of duration. Using the mode shapes and frequencies and the time history (time vs. acceleration record), properly scaled, the time history of accelerations, velocities and displacements of the lumped masses were obtained. Bechtel program CE611 was utilized for this computation.

The acceleration time history at each level was used as input data for the Bechtel computer program CE591, "Spectral Analysis", to obtain the acceleration and velocity response spectra at each floor for each percentage of damping required.

The response spectra of the structure supporting the Class I piping system is required in the seismic analysis for the piping.

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All Class B, C, and F systems outside the Reactor Building are analyzed by static methods as described below, Class D, E, G, and H Systems are not analyzed for earthquake.

The design of the B, C, and F Systems has been based on a static analysis using a 0.5g design acceleration. However, subsequent floor response spectra presented in Bechtel "Seismic Analysis Auxiliary Building" report dated January 1970 and subsequent floor response spectra for the Turbine Building developed by Duke Power Company show that there are peak accelerations greater than 0.5g. Consequently, additional analysis will be made to ensure that either (1) span lengths are reduced to avoid fundamental frequencies corresponding to accelerations above 0.5g or (2) piping stresses and restraint load capabilities are reviewed for adequacy for the appropriate accelerations. Conservative manual methods will be used to determine span frequencies.

Also, piping spans will be kept simple to avoid the necessity for modal analysis. Where this technique cannot be applied with confidence, a dynamic analysis will be performed. See example in 1C.3.4.3 for further clarification.

The spectra for the different buildings are not the same, and the buildings exhibit different amplifications for different frequencies. For piping passing from one building into another building, the maximum movements of the two buildings (deflections produced by earthquake) are summed absolutely and the piping system is subjected to these movements through the piping system restraints. The stresses produced in the piping by the building movements are considered additive to the stresses resulting from accelerations.

The CCW intake piping furnishing water to the LPSW pumps is designed for seismic
conditions as described in 1C.2.3. These lines are partially embedded in con-
crete and rest on bed rock at the point of entry into the station. There will
not be any differential movement at the piping-structural interface with the rock
24. base, thereby precluding any stress problems. Except for the CCW piping described
above, there are no other seismically designed safety-related buried lines for
the Oconee Project.

Rocking of the turbine support structure has been considered with respect to the main steam system analysis and movements of the turbine support are negligible as compared to other design movements of the main turbine piping leads attached to the main steam stop valve and control valve assembly.

Because of the costs involved in meeting the conservative design criteria above
for the main steam system, a dynamic analysis was performed by Engineering Data
Systems (EDS) of California for the main steam piping between the Reactor Build-
ing and the turbine. The response spectra recommended by EDS and used in this
analysis was equal to 1.25 times the recommended ground response spectra hori-
zontally and 1.0 times the recommended ground response spectrum vertically for
7. 0.05g ground acceleration and 0.005 damping. Most of the mass of pipe is sup-
ported by rigid stanchions from the ground with a short section at each end
connecting to the reactor and turbine buildings. Therefore, although ground
motion could be used for most of the pipe, the horizontal spectrum was increased
by 25 percent to include some conservatism and allow for the end connection to
the two buildings. Using the described spectrum resulted in a total seismic
stress of 2.0 ksi. Considering all other required load conditions, the allowable

stress left for seismic is 8.0 ksi. Since the restraints are also equally conservatively designed, the seismic input could be increased by as much as four times and the pipe and its supports would still be within their allowables. This
7. analysis confirms the conservatism of stresses obtained from static analyses and proves the adequacy of design loads used for all restraints.

The EDS Program description used for the outside main steam piping is as follows:

Each pipe loop is idealized as a mathematical model consisting of lumped masses connected by elastic members. Lumped masses are located at carefully selected points in order to adequately represent the dynamic and elastic characteristics of the pipe system. Using the elastic properties of the pipe, the flexibility matrix for the pipe is determined. The flexibility calculations include the effects of torsional, bending, shear, and axial deformations. In addition, for curved members, the stiffness is decreased in accordance with USAS B 31.1.0-1967 Code for Pressure Piping, Power Piping.

Once the flexibility and mass matrices of the mathematical model are calculated, the frequencies and mode shapes for all significant modes of vibration are determined. All modes having a period greater than 0.05 seconds are used in the analysis. The mode shapes and frequencies are solved in accordance with the following equation:

$$(K - w_n^2 M)\phi_n = 0$$

in which:

K = square stiffness matrix of the pipe loop
M = mass matrix for the pipe loop
w_n= frequency for the n^{th} mode
ϕ_n= mode shape matrix of the n^{th} mode

After the frequency is determined for each mode, the corresponding spectral acceleration is read from the appropriate response spectrum for the pipe. Using these spectral accelerations, the response for each mode is found by solving the following equation:

$$Y_n \max = \frac{R_n Sa_n D}{M_n w_n^2}$$

Static loads at points of support are determined by utilizing the computer program ME553-Piping Flexibility Analysis - to perform a modified weight analysis which is based on applying the maximum horizontal forces in the positive X and Y directions simultaneously with the maximum, upward directed, vertical force.

The horizontal forces are obtained by using the maximum acceleration peak from the appropriate envelope curves as the multiplier to convert uniform pipe weight into forces. The vertical force is obtained from the pipe weight density multiplied by the vertical peak acceleration.

The valves and special fittings on the system are mathematically expressed in the analysis as equivalent pipe of the same weight as the valve or fitting.

The combination of all maximum forces in the positive directions produces resulting static loads of greater magnitude than the dynamic analysis.

Systems containing seismic classification interfaces between dynamic and static analyses are as follows: This interface appears at the Reactor Building penetration proper. All penetrations serve as anchors to the piping systems, and the interface allows the dynamic - static analyses to be made independent of each other.

High Pressure Injection System
Low Pressure Injection System
Reactor Building Spray System
Spent Fuel Cooling System
Component Cooling System
Chemical Addition and Sampling System
LP Service Water System
Steam and Power Conversion System
Auxiliary Feedwater Supply System
Gaseous Waste Disposal System

Other seismically designed systems which penetrate the Reactor Building with a very minor portion of the system inside the Reactor Building (ie, from the penetration point to the inside isolation valve) are statically analyzed. These systems are as follows:

Reactor Building Penetration Room Ventilation System
Coolant Storage System
Liquid Waste Disposal System
Miscellaneous Non-Nuclear Service Systems; i.e., Service Air, Nitrogen, Demineralized Water, Filtered Water, etc.

Although there is not a seismic classification type interface, the Reactor Coolant System is a B&W Duke system interface.

All seismically designed systems penetrating the Reactor Building wall are designed as follows. Within the Reactor Building a dynamic analysis is performed as mentioned above utilizing the Bechtel "Seismic Analysis, Reactor Building" report dated February, 1970. As each penetration serves as an anchor to the system passing thru the Reactor Building wall, a separate analysis can be and is run on the piping outside the Reactor Building.

1C.3.2 System Valve Classification

In the absence of definitive codes, the non-destructive testing criteria applied to system valves are consistent with the intent of Par. 1-724 of USAS B31.7 Nuclear Power Piping Code (Feb. 1968) and the piping classification applicable to that portion of the system which includes the valve. On this basis, valves
2. are grouped into the same eight classes as shown for piping in Table 1C-1, and a valve is in the same class as the portion of system piping which includes the valve.

1C.3.3 System Component Classification

In the absence of definitive codes, the design criteria applied to pressure retaining system components are generally consistent with the intent of Sections III and VIII of the ASME Boiler and Pressure Vessel Code, the piping system classification applicable to that portion of the system which includes the component, and the required function of the component. Atmospheric water storage tanks important to safety conform to American Waterworks Association Standard for

Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage, D100, or equivalent.

Components are listed by system in Table 1C-2. This tabulation shows the code to which the component was designed, whether the component was designed to with-
5. stand the seismic load imposed by the maximum hypothetical earthquake, and the analytical technique employed in seismic analysis (1C.3.4).

1C.3.4 Application of Seismic Design Analyses

1C.3.4.1 Analytical Techniques

Two analytical techniques were employed in the seismic analyses: dynamic and static methods. In general, components in a Class A System have been thoroughly analyzed by dynamic techniques (Appendix 4B) &(1C.3.4.2), Table 1C-3 and components in Class B, C, and F Systems are analyzed by static methods.

All Class A, B, C, and F Systems inside the Reactor Building are analyzed by dynamic methods except for systems or portions of systems analyzed statically as follows:

System	Envelope Response Curve Figure No.
High pressure injection, east coolant loop, north & south legs and 1" makeup lines for pumps 1A1 & 1A2. (Only the 1" makeup lines)	1C-20
Low pressure service water coolant motor 1B2 inlet	1C-21
Low pressure service water coolant motor 1B1 inlet	1C-21
Low pressure service water east & west inlet headers & coolant motors inlets	1C-21
Component cooling outlet & pumps outlets	1C-22
Component cooling inlet & pumps inlets	1C-22
Pressurizer spray & low pressure injection supply to pressurizer spray	1C-23

The experience and judgment acquired in performing the dynamic seismic analyses on piping systems with varying complexities provided the basis for selecting the above systems analyzed by a static analysis. The results obtained by this method are more conservative than the results calculated by the dynamic analysis.

The use of the static analysis procedure is limited to piping systems which are not considered complex and where the anticipated seismic effects are minimal.

The envelope of response curve(s) developed for the dynamic analysis are used for the static analysis which is based on the assumption that the natural frequency of the piping system is at the critical frequency.

Protected or physically separated lines are used to supply cooling water to each steam generator. One of the six sources of electric power for the pump is supplied from Keowee Hydro Station.

An external source of cooling water is not immediately required due to the large quantities of water stored underground in the intake and discharge CCW piping. The stored volumes of water would provide sufficient cooling water for all three units for these approximate times after trip of the three reactors.

Intake and discharge lines below elevation 791 ft	37 days
Intake lines only below elevation 791 ft	17 days
Intake and discharge lines below elevation 775 ft	78 hours
Intake lines only below elevation 775 ft	51 hours

1C.3 SYSTEM CLASSIFICATIONS

Plant piping systems, or portions of systems, are classified according to their function in meeting design objectives. The systems are further segregated depending on the nature of the contained fluid. For those systems which normally contain radioactive fluids or gases, the Nuclear Power Piping Code, USAS B31.7, is used to define material, fabrication, and inspection requirements. Definitions of these classes are listed below:

Class I

This class is limited to the reactor coolant system. The class includes connecting piping out to and including the first isolation valve. Isolation valves can be either stop, relief, or check valves. Piping 1 inch and less is excluded from Class I.

Class II

Class II systems, or portions of systems, are those whose loss or failure could cause a hazard to plant personnel but would represent no hazard to the public. Class II systems normally contain radioactive fluid whose temperature is above 212 F, and in addition, those portions of Engineered Safeguards Systems which may see recirculated Reactor Building sump water following a LOCA. Piping 1 inch and less is excluded.

Class III

Class III systems, or portions of systems, are those which would normally be Class II except that the contained fluid is less than 212 F. Valves, piping, instrument fittings and thermowells with a penetration area equal to or less than a 1 inch ID pipe or less (all schedules) are placed in Class III regardless of system temperature or pressure, when such equipment is connected to Class I, II, or III systems.

In-line instrument components such as turbine meters, flow nozzle assemblies, and control valves, etc., are classified with their associated piping unless their penetration area is equal to or less than that of a 1 inch ID pipe of appropriate schedule for the system design temperature and pressure, in which case they are placed in Class III.

Fabrication and erection of piping, fittings, and valves are in accordance with the rules for their respective classes. Welds between classes of systems (Class I to II, I to III, or II to III) are performed and inspected in accordance with the rules for the higher class.

1C.3.1 System Piping Classification

System piping is divided into eight classes, depending on the required function of the system or portion of a system. These eight piping classes result from the combination of the preceding system classifications with and without design for seismic loading, as indicated in Table 1C-1. Piping classes A through E meet the intent of USAS B31.7 Nuclear Power Piping Code (February 1968) and
20. Addenda (June 1968) with the exception of those portions of the code which lack adequate definition for complete application.

Table 1C-1
System Piping Classification

Piping Class	Design Criteria	Designed For Seismic Loading
A	Class I, USAS B31.7	Yes
B	Class II, USAS B31.7	Yes
C	Class III, USAS B31.7	Yes
D	Class II, USAS B31.7	No
E	Class III, USAS B31.7	No
F	USAS B31.1.0	Yes
G	USAS B31.1.0	No
H	Good Industry Practice	No

Code Applicability: Due to the numerous code references located throughout this FSAR, no attempt is made to revise these references as Codes are amended, superseded or substituted. The existing Code references are the basis for design and materials; however, it is Duke Power Company's intent to comply
20. with portions of, or all of, the latest versions of existing Codes unless material and/or design commitments have progressed to a stage of completion such that it is not practical to make a change. When only portions of Code Addenda are utilized, the appropriate engineering review of the entire addenda will be made to assure that the overall intent of the Code is still maintained. Detailed information for each station unit and code applicability with respect to design, material procurement, fabrication techniques, NDT requirements and material traceability for each piping system class is described in the station piping specifications.

This table applied uniformly to all piping except that for auxiliary systems in the reactor building. Due to schedule commitments, and concern over lack of definitive design guidance in B31.7, it was decided to use B31.1 and applicable nuclear cases in the reactor building, but the materials were bought, erected, and inspected to the standards set down in B31.7. The reactor coolant system was designed to B31.7, Class I.

APPENDIX 1C
SYSTEMS DESIGN CRITERIA

1C SYSTEMS DESIGN CRITERIA

1C.1 INTRODUCTION

The design bases for all safety related systems are discussed in the appropriate sections of the FSAR. This appendix defines the design criteria used with respect to the loss-of-coolant accident (LOCA) and natural phenomena and also explains the division of components and piping into classifications related to design and function. Structural design bases are given in Appendix 5A.

Appropriate codes or standards were employed as design guides for systems where possible. In some cases, tentative or proposed codes are referenced. The new, and untried, nature of these codes makes it difficult, and sometimes impossible, to comply fully, but in all cases an effort was made to conform to the intent as the applicable sections were interpreted. In the absence of definitive guides, a definite attempt was made to apply good judgement and sound engineering practices.

The criteria with respect to natural phenomena, loss of coolant accidents and turbine missiles are as follows:

(1) A maximum hypothetical earthquake will not result in a loss-of-coolant accident (LOCA), but the simultaneous occurrence of these events will not result in loss of function to vital safety related components or systems.

(2) A tornado will not be allowed to cause a LOCA.

(3) A tornado does not occur simultaneously with or following a LOCA.

(4) A tornado and earthquake do not occur simultaneously.

(5) A turbine missile will not be allowed to cause a LOCA.

(6) A turbine missile does not occur simultaneously with a LOCA.

1C.2 DESIGN OBJECTIVES

The following design objectives result from consideration of the design criteria listed above.

1C.2.1 Loss-of-Coolant Accident

Capability is provided to assure necessary protective actions, including reactor trip and operation of the emergency core cooling system, to protect the public during a LOCA, even in the event of a simultaneously occurring maximum hypothetical earthquake.

1C.2.2 Turbine Missile Accident

The reactor coolant system will not be damaged by a turbine missile. Capability is provided to safely shut down the affected units.

1C.2.3 Earthquake

Capability is provided to shut down safely all three units in the event of a maximum hypothetical earthquake. The following equipment and portions of systems can withstand the maximum hypothetical earthquake:

(1) Reactor coolant system.

(2) Borated water storage tank and piping to high pressure and low pressure injection pumps and reactor building spray pumps.

(3) HP injection pumps and piping to reactor coolant system.

(4) LP injection pumps, LP injection coolers and piping to both reactor coolant system and reactor building spray pumps.

(5) Core flood tanks and piping to reactor coolant system.

(6) Reactor building spray pumps, piping to spray headers, and the spray headers.

(7) Reactor Building coolers.

(8) Low pressure service water (LPSW) pumps, LPSW piping to LP injection coolers and Reactor Building coolers and LPSW piping from these coolers to the condenser circulating water (CCW) discharge.

(9) CCW pumps and intake piping to the LPSW pumps.

24. (10) Upper surge tanks, and piping to the emergency feedwater pump.

(11) Emergency feedwater pump and turbine and auxiliary feedwater piping to the steam generators.

24. (12) Main steam lines to and including turbine stop valves. Turbine bypass system up thru main steam system isolation valves, and steam supply lines to the emergency feedwater pump turbine.

(13) Penetration room ventilation system.

(14) Reactor Building penetrations and piping through isolation valves.

(15) Electric power for above.

1C.2.4 Tornado

The reactor coolant system will not be damaged by a tornado. Capability is provided to shut down safely all three units.

The reactor coolant system, by virtue of its location within the reactor building, is protected from tornado damage. A sufficient supply of cooling water for safe shutdown is assured by an auxiliary service water pump located in the auxiliary building and taking suction from Unit 2 CCW intake piping.

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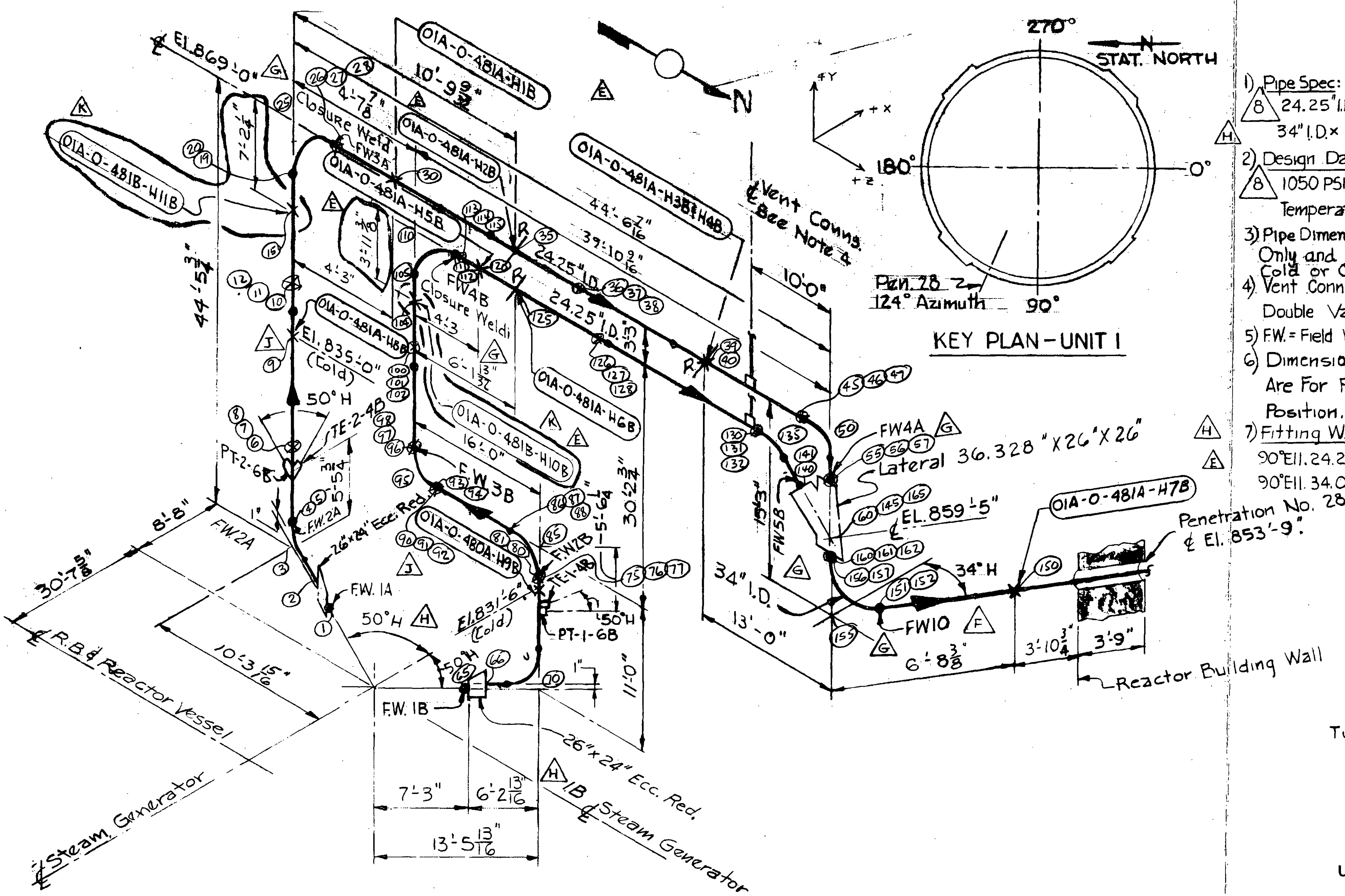
LIST OF FIGURES



FOR SEISMIC FORCE & SEISMIC STRESS ANALYSES

⊗ = MASS POINT

SYSTEM NO. 01A.

GENERAL NOTES

1) Pipe Spec:
   8 24.25" I.D. x .9375" Wall A-155 KC-70 Class-1
   H 34" I.D. x 1.25" Wall Plate Pipe.
2) Design Data:
   8 1050 PSIG 650°F Max. Operating Temperature 590°F Max.
3) Pipe Dimensions Shown are for Design Purposes Only and do not Necessarily Reflect Either Cold or Operating Position of Pipe.
4) Vent Connections To Be 1" In Size And Double Valved.
5) F.W. = Field Weld
6) Dimensions For Hangers And Restraints Are For Pipe In the Normal Operating Position.
7) H Fitting Wall Thickness
   E 90° Ell. 24.25" I.D. x 1.250" Rad.-39" Wt. 1738 lbs.
   90° Ell. 34.0" I.D. x 1.687" Rad.-54" Wt. 4545 lbs.

Typical Hanger Number *

01-0-481A-H6

UNIT NO. 1 ONLY; UNITS 2 & 3 SIMILAR

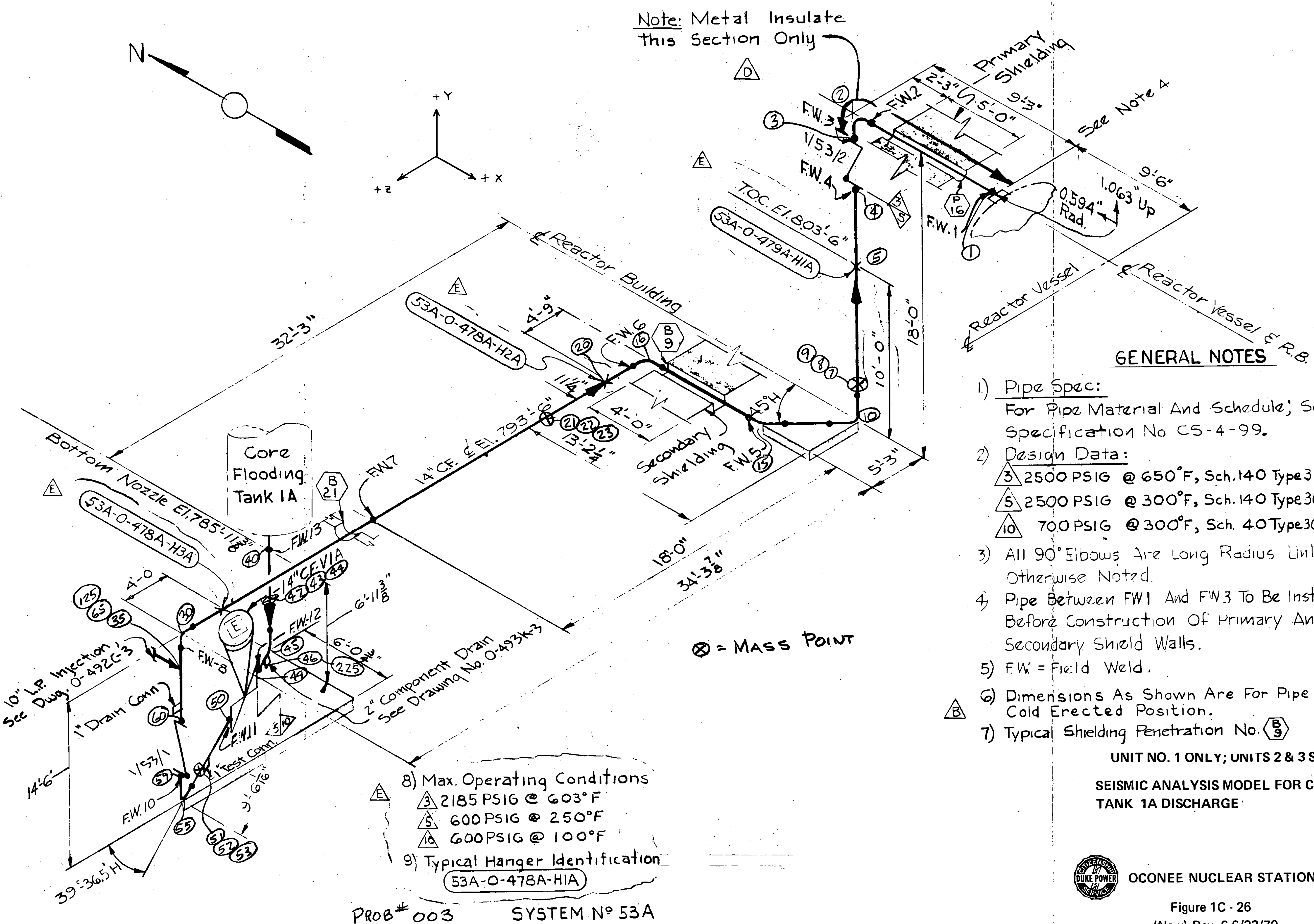
SEISMIC ANALYSIS MODEL FOR MAIN STEAM SYSTEM—WEST GENERATOR, SEISMIC FORCE AND SEISMIC STRESS ANALYSIS

DUKE POWER

OCONEE NUCLEAR STATION

Figure 1C - 25

(New) Rev. 6 6/22/70



GENERAL NOTES

1) Pipe Spec:
For Pipe Material And Schedule; See B&W Specification No CS-4-99.

2) Design Data:
3 2500 PSIG @ 650°F, Sch. 140 Type 316H S.Stl.
5 2500 PSIG @ 300°F, Sch. 140 Type 304H S.Stl.
10 700 PSIG @ 300°F, Sch. 40 Type 304H S.Stl.

3) All 90° Elbows Are Long Radius Unless Otherwise Noted.

4) Pipe Between FW1 And FW3 To Be Installed Before Construction Of Primary And Secondary Shield Walls.

5) F.W. = Field Weld.

B 6) Dimensions As Shown Are For Pipe In The Cold Erected Position.

7) Typical Shielding Penetration No. B 9

E 8) Max. Operating Conditions
3 2185 PSIG @ 603°F
5 600 PSIG @ 250°F
10 600 PSIG @ 100°F

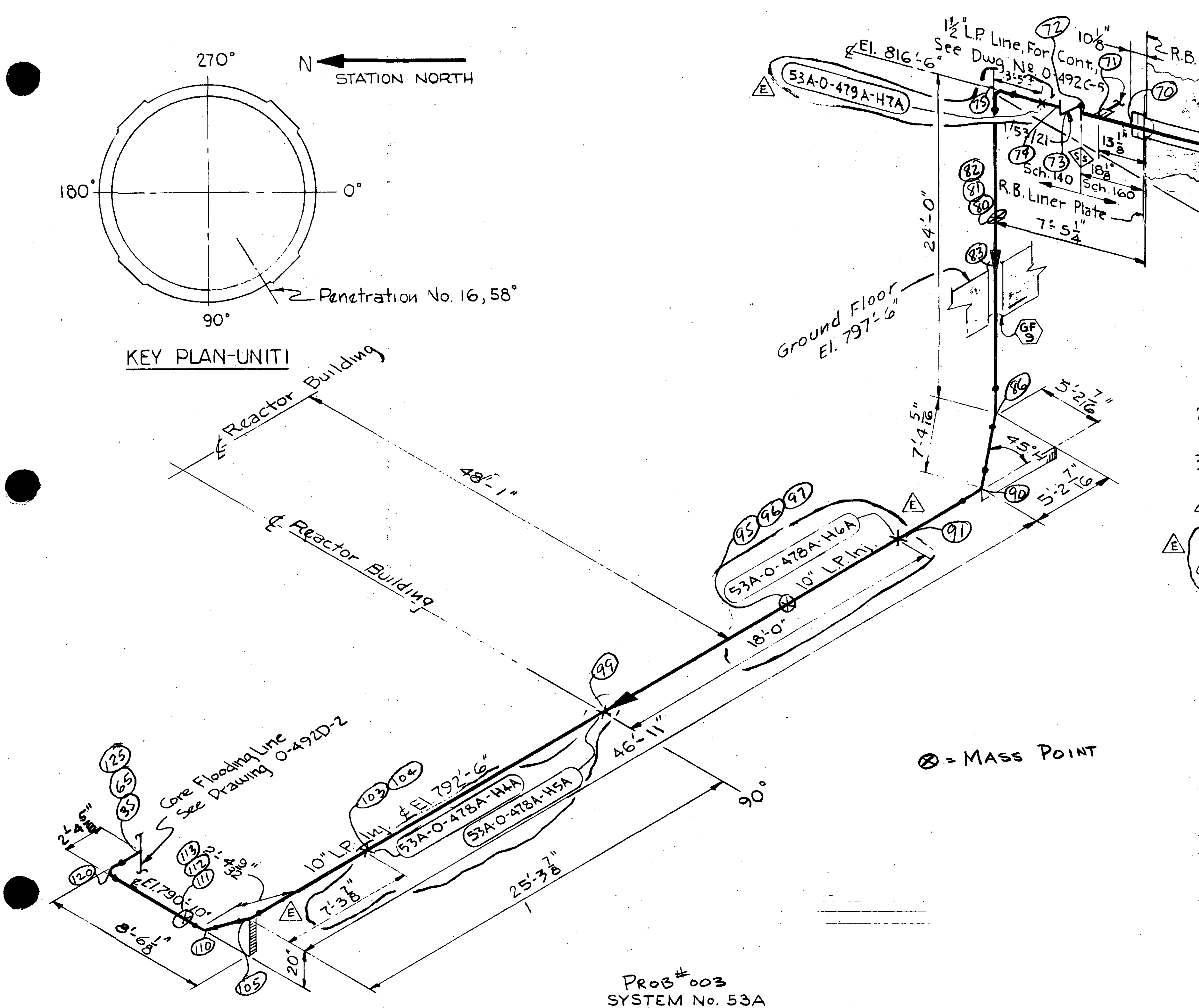
9) Typical Hanger Identification
53A-0-478A-H1A

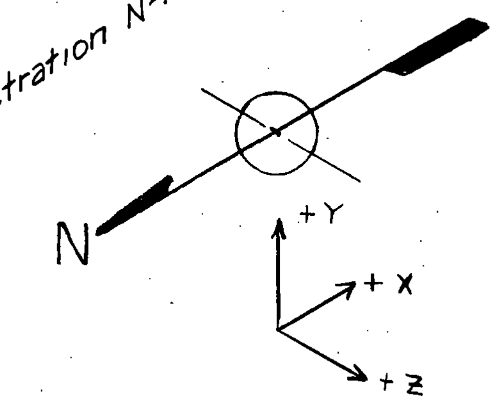
UNIT NO. 1 ONLY; UNITS 2 & 3 SIMILAR

SEISMIC ANALYSIS MODEL FOR CORE FLOODING TANK 1A DISCHARGE

OCONEE NUCLEAR STATION

Figure 1C - 26
(New) Rev. 6 6/22/70





GENERAL NOTES

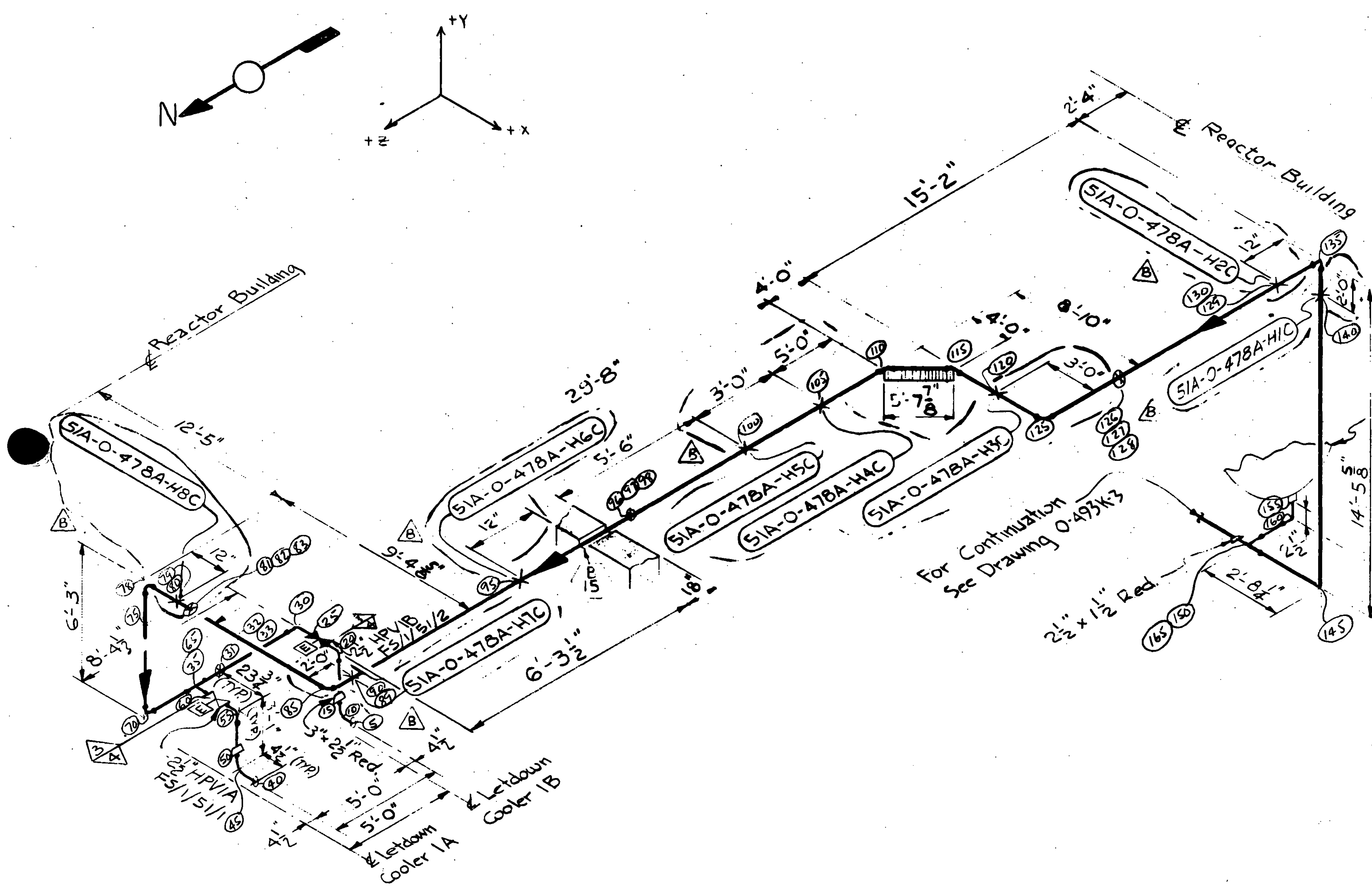
1) Pipe Spec:
   5 10" Sch. 140 ASTM A-376, Type 304H.
   5 10" Sch 160 ASTM A-376, Type 304H.
2) Design Data:
   5 2500 PSIG @ 300°F Max.
3) Dimensions As Shown Are For Pipe In The Cold Erected Position.
4) Typical Shielding Penetration No. GF 9
E 5) Oper. Temp. 250°F
6) Typical Hanger Identification
   53A-0-478A-H4A

UNIT NO. 1 ONLY; UNITS 2 & 3 SIMILAR

SEISMIC ANALYSIS MODEL FOR LOW PRESSURE INJECTION SYSTEM, WEST ISOMETRIC

OCONEE NUCLEAR STATION

Figure 1C - 27
(New) Rev. 6 6/22/70



GENERAL NOTES

1) Pipe Spec:
   (3) 2½" Sch. 160 Stainless Steel, Type 316, ASTM A-376.
   (4) 2½" & 3" Sch. 160 Stainless Steel, Type 316, ASTM A-376.
2) Design Data:
   (3) 2500 PSIG @ 650°F.
   (4) 2500 PSIG @ 600°F.
3) Moments Of Primary Coolant System, See B&W Drawings 128625E & 128626E.
4) Dimensions Shown Are For Pipe In The Cold Erected Position.
5) Typical Shielding Penetration No (B/15)
6) Typical Hanger Identification No (51A-0-478A-H1C) (B)

⊗ = MASS POINT

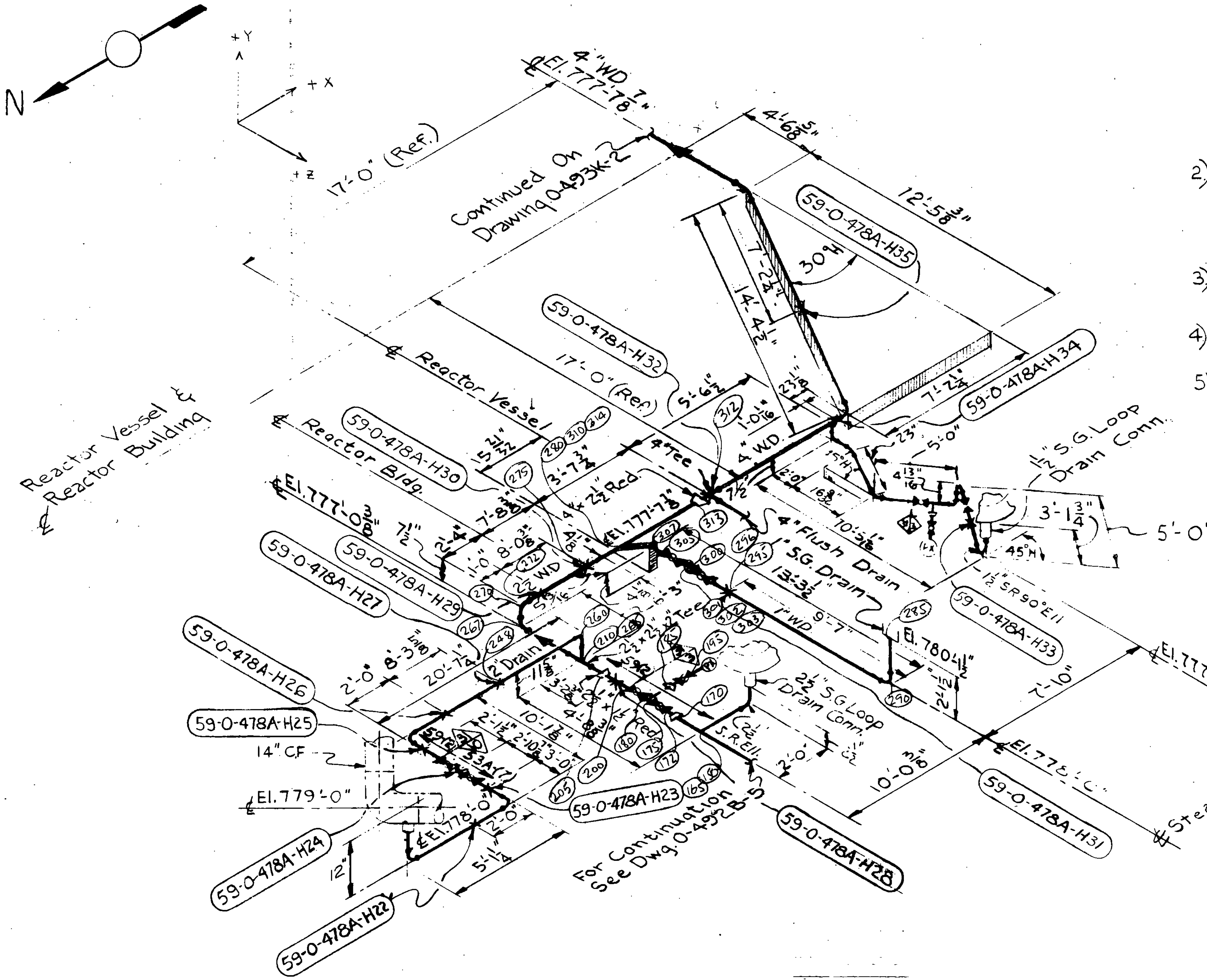
UNIT 1 ONLY; UNITS 2 & 3 SIMILAR

SEISMIC ANALYSIS MODEL FOR HIGH PRESSURE INJECTION SYSTEM, LET DOWN COOLER INLET

OCONEE NUCLEAR STATION

Figure 1C - 28
(New) Rev. 6 6/22/70

PROB #020
SYSTEM No. 51A



GENERAL NOTES

1) Pipe Spec:
   - (3) 1½" Seamless S.Stl., ASTM A-376, Type 316H, Sch. 160
   - (10) 2" Seamless S.Stl., ASTM A-312, Type 304H, Sch. 40S
   - (31) 4" Seamless S.Stl., ASTM A-312, Type 304H, Sch. 10S
   - (31) 2½" Seamless Stainless Steel ASTM A-312 Type 304H, Sch. 10S.
   - (31) 1½" Seamless Stainless Steel ASTM A-312 Type 304H, Sch. 40S.
2) Design Data:
   - (3) 2500 PSIG @ 650°F Max., Oper. Temp. 580°F.
   - (31) 150 PSIG @ 200°F Max., Oper. Temp. 175°F.
   - (10) 700 PSIG @ 300°F Max., Oper. Temp. 100°F.
3) Moments Of Primary Coolant System See B & W Drawings 128625E & 128626E.
4) Dimensions As Shown Are For Pipe In The Cold Erected Position.
5) Typical Hanger Identification (59-0-478A-H22)

⊗ = MASS POINT

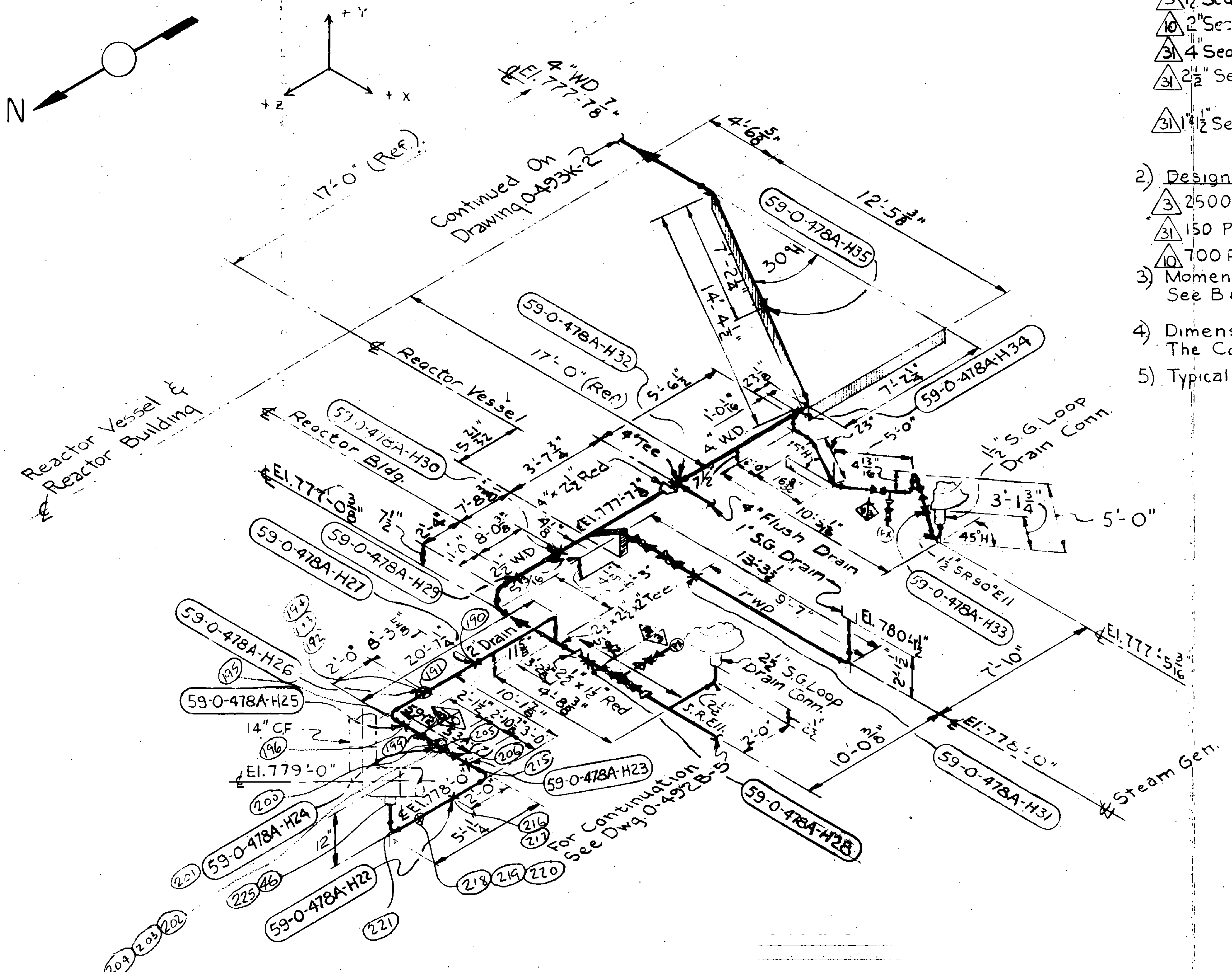
UNIT NO. 1 ONLY; UNITS 2 & 3 SIMILAR

SEISMIC ANALYSIS MODEL FOR REACTOR BUILDING COMPONENT DRAINS, WEST COOLANT LOOP, PROB. 020

OCONEE NUCLEAR STATION

Figure 1C - 29
(New) Rev. 6 6/22/70

PROB # 020
SYSTEM No. 59



GENERAL NOTES

1) Pipe Spec:
   - △3 1 1/2" Seamless S.Stl., ASTM A-376, Type 316H, Sch. 160
   - △10 2" Seamless S.Stl., ASTM A-312, Type 304H, Sch 40S
   - △31 4" Seamless S.Stl, ASTM A-312, Type 304H, Sch. 10S
   - △31 2 1/2" Seamless Stainless Steel ASTM A-312 Type 304H, Sch. 10S.
   - △31 1" 1 1/2" Seamless Stainless Steel ASTM A-312 Type 304H, Sch. 40S.
2) Design Data:
   - △3 2500 PSIG @ 650°F Max., Oper. Temp. 580°F.
   - △31 150 PSIG @ 200°F Max., Oper. Temp. 175°F.
   - △10 700 PSIG @ 300°F Max., Oper. Temp. 100°F.
3) Moments Of Primary Coolant System See B & W Drawings 128625E & 128626E.
4) Dimensions As Shown Are For Pipe In The Cold Erected Position.
5) Typical Hanger Identification (59-0-478A-H22)

⊗ = MASS POINT

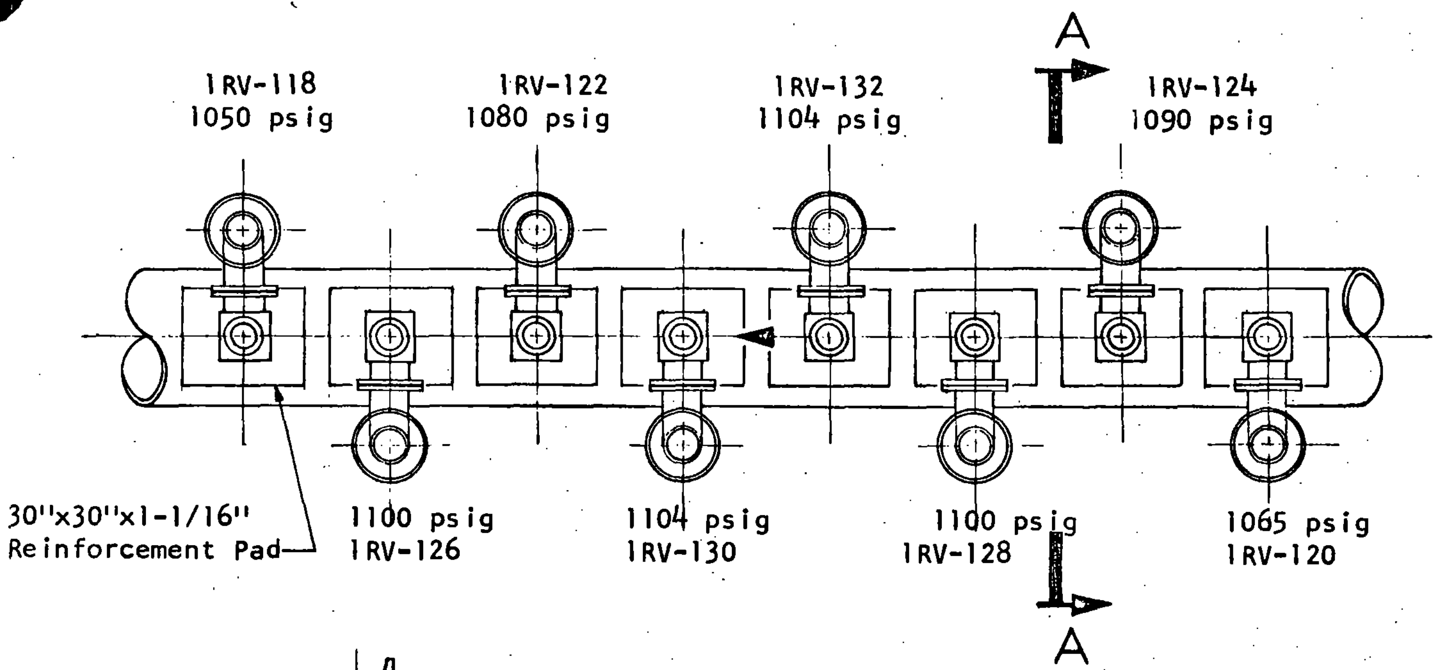
PROB #003
SYSTEM No. 59

UNIT NO. 1 ONLY; UNITS 2 & 3 SIMILAR

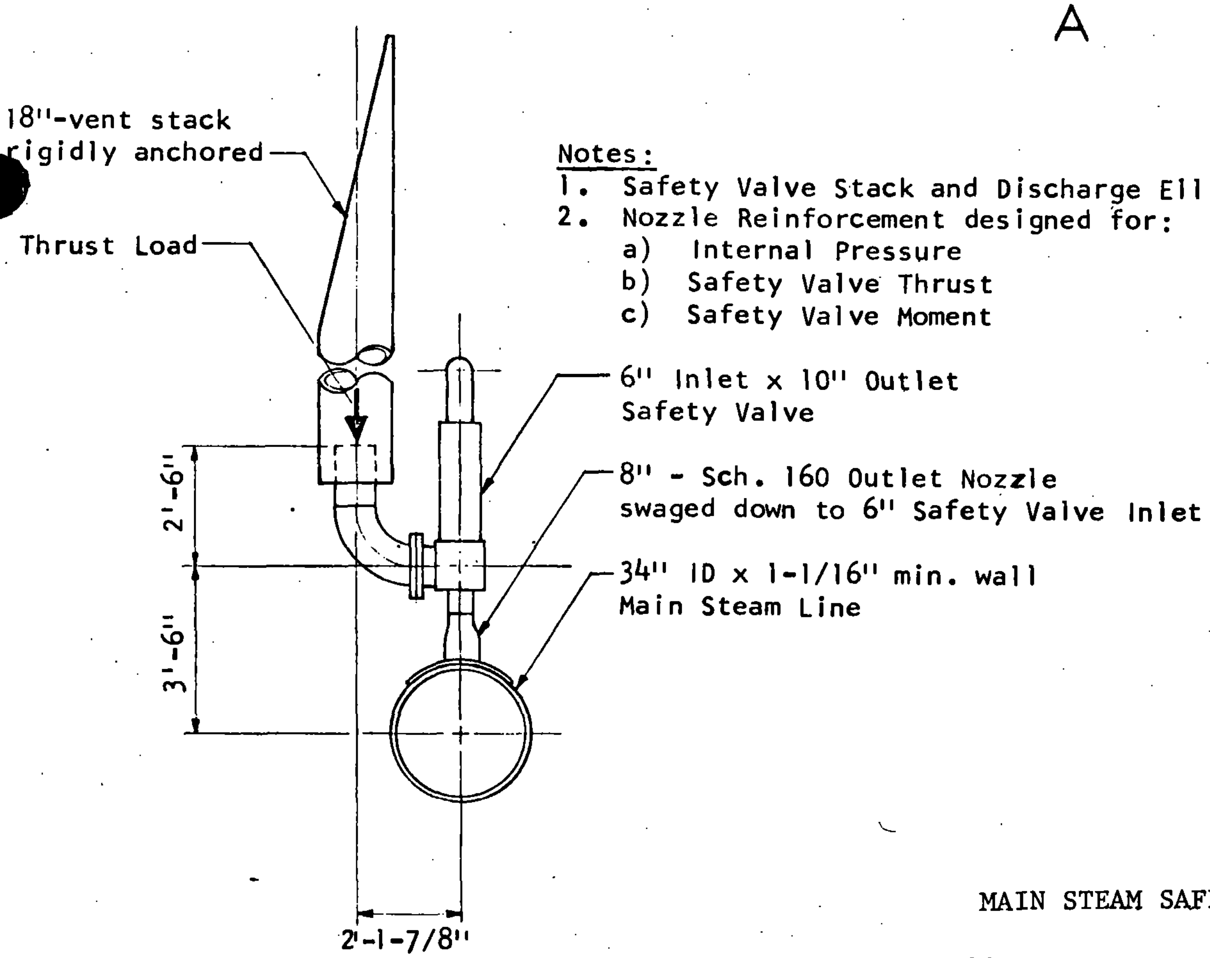
SEISMIC ANALYSIS MODEL FOR REACTOR BUILDING COMPONENT DRAINS, WEST COOLANT LOOP, PROB. 003

OCONEE NUCLEAR STATION

Figure 1C - 30
(New) Rev. 6 6/22/70



Notes:
1. Safety Valve Stack and Discharge Ell are not connected.
2. Nozzle Reinforcement designed for:
   a) Internal Pressure
   b) Safety Valve Thrust
   c) Safety Valve Moment

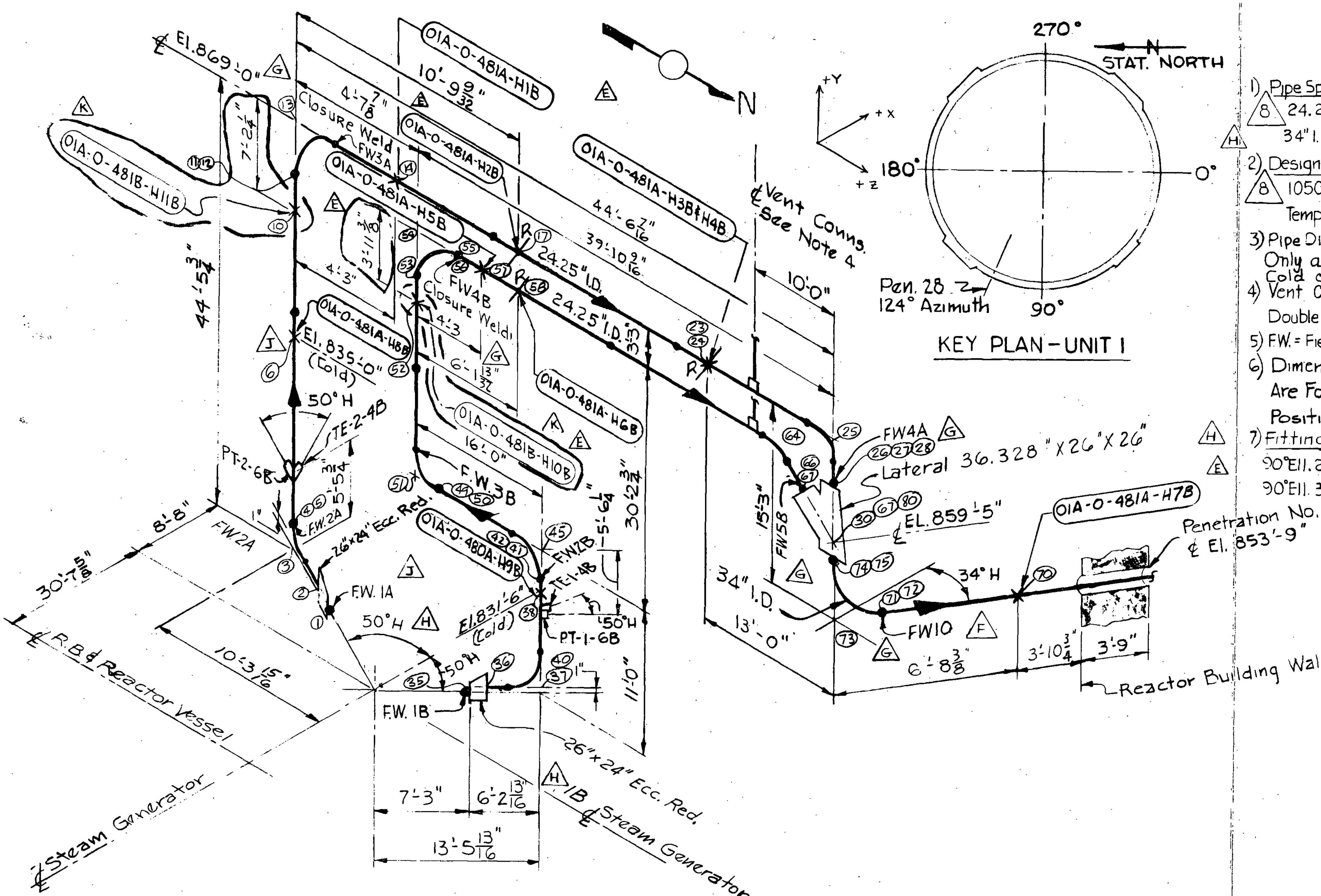


SECTION "A-A"

MAIN STEAM SAFETY VALVES

OCONEE NUCLEAR STATION

Figure 1C-31
Rev. 20 5/25/72
(New Page)



GENERAL NOTES

1) Pipe Spec:
   (8) 24.25" I.D. x .9375" Wall A-155 KC-70 Class-1
   (H) 34" I.D. x 1.25" Wall Plate Pipe.
2) Design Data:
   (8) 1050 PSIG 650°F Max. Operating Temperature 590°F Max.
3) Pipe Dimensions Shown are for Design Purposes Only and do not Necessarily Reflect Either Cold or Operating Position of Pipe.
4) Vent Connections To Be 1" In Size And Double Valved.
5) F.W. = Field Weld
6) Dimensions For Hangers And Restraints Are For Pipe In the Normal Operating Position.
7) Fitting Wall Thickness
   (H) 90° Ell. 24.25" I.D. x 1.250" Rad.-39" Wt. 1738 lbs.
   (E) 90° Ell. 34.0" I.D. x 1.687" Rad.-54" Wt. 4545 lbs.

Typical Hanger Number ✱
01-0-481A-H6

FOR THERMAL & WEIGHT ANALYSES

UNIT NO. 1 ONLY; UNITS 2 & 3 SIMILAR

SEISMIC ANALYSIS MODEL FOR MAIN STEAM SYSTEM– WEST GENERATOR, THERMAL AND WEIGHT ANALYSES

DUKE POWER

OCONEE NUCLEAR STATION

Figure 1C - 24

(New) Rev. 6 6/22/70

PROB # 002

SYSTEM NO. 01A.

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2-4	Original
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2-6	Rev. 19
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2-9	Original
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2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in Figure 2.1-1A - Unit 1. If the actual pressure/temperature point is below
2.1-1B - Unit 2

and to the right of the line, the safety limit is exceeded.

The combination of reactor thermal power and reactor power imbalance (power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the safety limit as defined by the locus of points (solid line) for the specified flow set forth in Figure 2.1-2A - Unit 1. If the actual-reactor-thermal-power/
2.1-2B - Unit 2

power-imbalance point is above the line for the specified flow, the safety limit is exceeded.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of the W-3 correlation.(1) The W-3 correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.3. A DNBR of 1.3 corresponds to a 94.3% probability at a 99% confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip set points to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1A represents the conditions at which a
2.1-1B

minimum DNBR of 1.3 is predicted for the maximum possible thermal power (112%)
when four reactor coolant pumps are operating(minimum reactor coolant flow is
30. 131.3 x 10^6 lbs/hr. This curve is based on the following nuclear power peaking
factors(2) with potential fuel densification effects;

$$F_q^N = 2.67; \; F_{\Delta H}^N = 1.78; \; F_z^N = 1.50$$

The design peaking combination results in a more conservative DNBR than any other shape that exists during normal operation.

The curves of Figure 2.1-2A are based on the more restrictive of two thermal
2.1-2B

limits and include the effects of potential fuel densification:

1. The 1.3 DNBR limit produced by a nuclear power peaking factor of F_q^N = 2.67 or the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.3 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.1 kw/ft - Unit 1
   19.8 kw/ft - Unit 2

Power peaking is not a directly observable quantity and therefore limits have been established on the bases of the reactor power imbalance produced by the power peaking.

The specified flow rates for Curves 1, 2, 3, and 4 of Figure 2.1-2A correspond
2.1-2B

to the expected minimum flow rates with four pumps, three pumps, one pump in each loop and two pumps in one loop, respectively.

The curve of Figure 2.1-1A is the most restrictive of all possible reactor
2.1-1B

coolant pump-maximum thermal power combinations shown in Figure 2.1-3A.
2.1-3B

The curves of Figure 2.1-3A represent the conditions at which a minimum DNBR
2.1-3B

2 SITE AND ENVIRONMENT

2.1 GENERAL DESCRIPTION

The site is located in Oconee County, South Carolina, approximately 8 miles northeast of Seneca, South Carolina. In the area immediately north and west of the site is Duke Power Company's Lake Keowee. The Government's Hartwell Reservoir is south of the site.

Cooling water for Oconee is drawn from and returned to Lake Keowee. Limnological and thermal analyses of the lake have established that the Oconee nuclear capacity of 2658 MWe is fully compatible with ecological and recreation aspects of the lake. These analyses have been approved by applicable State and Federal agencies, and Duke's license from the Federal Power Commission for the Keowee Development authorizes the use of cooling water for up to 3000 MW of steam generated electric power at Oconee.

The exclusion area around the site has a radius of 1 mile. The boundary of the low population zone lies at a 6-mile radius around the site. There are only six population centers within a 100 mile radius of the site. Anderson,
19. South Carolina, with a 1970 population of 27,556 is the nearest and is located 21 miles southeast of the site.

The structures are founded on normal Piedmont granite gneiss rock, and the foundation materials presented no special problems in design or construction. Foundations have been designed in accordance with the recommendations in the Report on Subsurface and Foundations Investigations attached as Appendix 2A in the Preliminary Safety Analysis Report.

2.2 SITE AND ADJACENT AREAS

2.2.1 LOCATION

The Oconee Nuclear Station site is located in eastern Oconee County, South Carolina, as shown in Figure 2-1, at latitude 34°-47'-38.2" N and longitude 82°-53'-55.4" W.

2.2.2 LAND OWNERSHIP

All property within a 1 mile radius of the nuclear station is owned in fee except for a small rural church plot, highway right-of-way and approximately 9.8 acres of United States of America Hartwell Project property. The Hartwell property is either a portion of the Hartwell Reservoir or subject to flooding and not suitable for other uses. Duke has obtained from the owners of the church plot and from the United States the right to restrict activities on these properties and to evacuate them of all persons at any time without prior notice if, in its opinion, such evacuation is necessary or desirable in the interest of public health and safety.

The Duke owned single family residence now remaining in the Exclusion Area will be removed by December 1, 1970. This residence, which was formerly leased to
8. the occupant with provisions for immediate evacuation in the event Duke felt that such action should be necessary in the interest of public health and safety, has been evacuated by the leaseholder and the lease terminated.

26.

7. A fence will be erected around the security area as shown in Figure 2-2. The nuclear station site area, within the security boundary, is as shown in Figure 2-2.

2. 2.2.3 ACTIVITIES WITHIN EXCLUSION AREA

The activities will be limited to the highways through the Exclusion Area,
27. Duke's Visitors Center, recreation on the lakes, and the Old Pickens Church and Cemetery which are historical landmarks and will not be used for regular services. The only commercial enterprises within the Exclusion Area will be Duke's Keowee Hydroelectric Station and the Oconee Nuclear Station. There are
2. two bachelor camps now located inside the Exclusion Area which will continue to be occupied by Construction personnel until the project is completed. Only male employees of Duke Power Company or their contractors live in these quarters. These camps can be evacuated or their use as living quarters discontinued anytime it is considered necessary. Both camps are 2,000 ft. or more from the center line of the Unit No. 1 Reactor Building. Arrangements have been made with the South Carolina State Highway Department to control and limit traffic on public highways in the Exclusion Area should it become necessary in the interest of public health and safety. Refer to Section 12.3.2.

2.2.4 VICINITY

North and west of the site will be the 18,372 acre Lake Keowee to be formed by Duke's Keowee and Little River Dams, as shown in Figure 2-3. Lake Jocassee, with a surface area of 7,565 acres, is located approximately 11 miles north of the site. Hartwell Reservoir, a United States Government-owned lake, is located south of the site, and the Blue Ridge Mountains of the Appalachian chain lie northwest of the site. Figure 2.4 shows general geographical and topographical features within 50 miles of the site.

Oconee Nuclear Station is a part of Duke's Keowee-Toxaway Project combining hydroelectric, pumped-storage, and nuclear power generation. Duke now owns approximately 150,000 acres of land in and adjacent to this project, including the 18,372 acres for Lake Keowee and 7,565 acres for Lake Jocassee to the north. The bulk of Duke property outside these lakes lies in the sectors between NNW and NE from the Oconee Nuclear Station. This property is under Duke's forestry management and will also be used for controlled public hunting, fishing, camping, tourism, and recreation in cooperation with State agencies and conservation groups. There are no industrial activities within
20. five miles of the site.

2.2.5 POPULATION AND LAND USE

Population centers within a 100 mile radius are shown in Figure 2-12. The largest city, Knoxville, Tennessee, located 97 miles northwest of the site, had a 1970 population of 174,587. The nearest population center, Anderson, South
19. Carolina, is 21 miles southeast of the site and had a 1970 population of 27,556.

Figures 2-13, 2-14, and 2-15 show the actual 1970 and estimated 2010 population by sectors at varying radii from the site. The projected population estimates for 2010 have been projected upward over that submitted in the PSAR. This increase is due to plans for lake proximity developments extending out as much as 20 miles from

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site, particularly in NW and NNW sectors.

19. The actual permanent population within the low population boundary (6 miles from site) is 3620 for 1970 and estimated to be 8900 for 2010.

4. The estimated transient population within the low population boundary is 2000 for 1970 and 19000 for 2010.

It is expected that Lake Keowee's 300 mile shoreline will be fully developed by 1985 at which time the estimated transient population will be 36,000. This estimate is based on development of lakeside lots, public access areas, and ex-
2. panded commercial activities to take advantage of expanded recreational opportunities. There will not be any summer cottages within the Exclusion Area.

Land use is largely forest lands with some cultivation and pasture. Additional information on land use is found in the PSAR Sections 2.2.6 and 2.2.8.

2.2.6 KEOWEE RESERVOIR ELEVATIONS

7. All of the man-made dikes forming the Keowee Reservoir rise to an elevation of 815 feet including the Oconee Nuclear Station Intake Chanel Dike.

2.3 METEOROLOGY

The historical climatology of the area as presented in Appendix 2B to the FSAR remains applicable. Subsequently, no extraordinary meteorological phenomena have occurred.

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2.3.1 ON-SITE SURVEYS

The evaluations of two comprehensive meteorological surveys conducted on-site confirm that the meteorological characteristics of the site are favorable for the Oconee Nuclear Station.

2.3.1.1 Near Ground Study

The first survey, started in mid-October 1966, prior to filing the PSAR and extended until late October 1967, was a study of near-ground diffusion climatology. Wind data were continuously recorded by a Packard Bell Electronics Corporation (Beckman and Whitley, Inc.) WS-101 system mounted on a 14 meter pole located near mid-site (See Figure 2-5). Temperature gradients were determined by thermographs located in standard United States Weather Bureau Cotton Region Instrument Shelters stationed on the site at varying terrain elevations. A standard recording precipitation gage with wind shield was installed near the base of the 14 meter pole. After initial installation and check-out, the instrumentation was checked twice weekly until the performance of the instrumentation was verified. Thereafter a weekly check was maintained. A total of 8619 hourly observations were considered valid out of a possible 9058 or about 95 percent. The results of this study established the frequency of wind conditions with varying lapse rates near ground. These results confirm the Valley Drainage Model concept postulated in PSAR Supplement 2, dated April 18, 1968, as shown below:

		Results of One Year On-Site Studies	PSAR Values
(a)	Frequency of temperature inversions of total hourly observations	24%	35%
(b)	Direction of predominating inversion wind (Figure 2-6)	Northerly 11.6% of observations	Northerly
(c)	Inversion wind speed average, meters per second	1.40	1.0

(d) The minimum average standard deviation of inversion winds in any sector for the one year averaged 6.6 degrees which is about two and one-half times the 2.5 degree value assigned to Pasquill F conditions in the PSAR.

Wind Roses presenting near-ground data, Figures 2-6, 2-7, and 2-8 compared to PSAR submittal of Greenville-Spartanburg, South Carolina, Airport data, reflect wind reorientation by nearby mountain ranges and some channeling by the river valley.

For ground releases, actual on-site data results in greater dispersion than was calculated by the valley drainage model in Answer 2.5, Supplement 2 to the PSAR. For example, ground-release X/Q for the 30-day model at 6 1/2 miles from site, near boundary of low population zone, is 4.34×10^{-7} seconds m^{-3} from on-site data, but is 1.17×10^{-6} seconds m^{-3} from the valley drainage concept. Nevertheless, dispersions of ground level releases are based on the valley drainage concept in the safety analyses for Oconee.

Thus, the one-year near-ground, on-site, meteorological assessment confirmed that the site is meteorologically favorable for Oconee and that the valley drainage model is conservative.

2.3.1.2 Elevated Release Study

In June 1967, prior to receiving Construction Permit in November 1967, a second study was started to establish meteorological parameters related to elevated (vent) releases. Reference to Figure 2-5 of this section illustrates the arrangement of meteorological instrumentation required to initiate this study. Investigations of winds and atmospheric stability were made at vent effluent levels by wind and temperature gradient measuring systems mounted on a 46 meter (150 ft) tower. Wind sensors are Packard Bell Electronic Corporation (Beckman and Whitley, Inc.) WS-101 System. A Leeds & Northrup Company, four point, resistance thermometer system is being used continuously for temperature gradient measurements. The resistance thermometers were matched for accuracy and each is shrouded by a motor operated ventilating system equipped with radiation shield. The actual temperature is recorded at only the lowest elevation and three higher thermometers record a temperature departure (delta-t) from the lowest level value.

All instruments were carefully checked initially until performance warranted only the present weekly checking procedures. These systems were purchased, installed, and are being maintained as permanent station equipment. Appropriate transducers and telemetry equipment were included initially to enable functional use and display of these variables in the Units 1 and 2 control room.

In addition to tower meteorological instrumentation, a standard weather instrument shelter containing a thermograph and a mercury-in-glass dry bulb thermometer, for checking, was set up near the tower base. A standard recording precipitation gage with wind shield was also installed nearby.

A brief summary of data through the first year (June 19, 1967, through May 31, 1968,) shows the following:

(a) The average wind speed recorded by the anemometer at elevation 1028 ft (232 ft above plant yard level) was 6.5 miles per hour or about 3 meters per second for all conditions, and about 2 meters per second during inversions.

(b) The dominant all-wind direction was northerly which accounts for 10.98 percent of all observations.(See Figures 2-9 and 2-10)

(c) The average standard deviation associated with winds less than 1 meter per second was about 22 degrees. As expected, the standard deviations decreased generally as wind speeds increased.

(d) A frequency of inversions of approximately 40 percent was found for the one year of tower data compared to 24 percent for near-ground observations. Although the two periods of observations are not chronologically identical, one would expect the inversion duration time to be less near-ground due to more rapid inversion "burn-off"; however, it is also noteworthy that the frequency of inversions for the Greenville-Spartanburg Airport for Pasquill-Turner computations also increased for the year during tower observations compared to near-ground observation period.

(e) The maximum amount of rain was from the northeast where during the year 7.09 inches of rain fell in an aggregate of 71 hours. (See Figure 2-11)

2.3.2 DESCRIPTION OF DISPERSION FACTORS

Dispersion factors (x/Q, seconds m^{-3}) as tabulated in the following paragraph (2.3.2.1) are to be used for accident (10CFR100) and routine operational (10CFR20) analyses. Dispersion factors for elevated releases are based on analysis of on-site meteorological data. The factors given for ground releases were negotiated through discussions with the AEC/DRL staff during the early summer of 1970. These discussions were related to the additional meteorological studies submitted April 20, 1970 as Appendix 2A at the end of this section. During the negotiations, Duke agreed to reduce the Reactor
18. Building design leakage rate from 0.50 percent by volume in 24 hours to 0.25 percent and increase the atmospheric dispersion factors (x/Q seconds m^{-3}) for ground releases. It was agreed to depart from the dispersion factors for ground releases as submitted previously and supported by the Near Ground Study, The Valley Drainage Model (Answer 2.5, Supplement 2, PSAR) and Appendix 2A. The accepted ground release dispersion factor (x/Q seconds m^{-3}) at the Exclusion Area Boundary (one mile) for Oconee Units 1, 2, and 3 is 1.16×10^{-4} for the 0-2 hour analysis.

2.3.2.1 <u>Dispersion Factors</u> (x/Q seconds m^{-3})

2. (a) At Exclusion Area Boundary

1-mile (1609 m)	0-2 Hours	0-24 Hours	0-7 Days
18. Ground Releases	1.16×10^{-4}		
Elevated Releases	3.35×10^{-5}	9.73×10^{-6}	2.98×10^{-6}

(b) At Boundary of Low

Population Zone	0-24 Hours	0-30 Days
18. Ground Releases (2)	1.32×10^{-5}	7.2×10^{-7}
Elevated Releases (2)	3.90×10^{-6}	3.42×10^{-7}
Elevated Releases (3)	4.15×10^{-6}	3.70×10^{-7}

(c) Long Term (One Year) Exclusion Area Boundary

19. Ground Release 4.61×10^{-6} seconds m^{-3}
18. Elevated Release 8.74×10^{-7} seconds m^{-3}

Notes: (2) At valley construction 6 1/2 miles (10,464 m) from site near Boundary of Low Population Zone.
(3) Six miles (9,658 m) from site at Boundary of Low Population Zone.

2.3.2.2 Basis for Elevated Release Dispersion Factors

(a) Meteorological Conditions

0 - 2 Hours
Pasquill F with a wind speed of 1.0 meter per second with plume confined to "Sigma Y" dimensions.

2 - 24 Hours
Pasquill F with wind speed of 1.0 meter per second with plume confined to one sector.

1 - 7 Days
Pasquill F 50 percent of time, 1 meter per second wind speed.
Pasquill D 50 percent of time, 4 meters per second wind speed.
Wind confined to one sector 30 percent of time.

7 - 30 Days
Pasquill F 35 percent of time, 1 meter per second wind speed.
Pasquill E 5 percent of time, 2 meters per second wind speed.
Pasquill D 60 percent of time, 4 meters per second wind speed.
Wind confined to one section 12.5 percent of time.

2.3.2.3 Assignment of Dispersion Factors

	Conditions	Appropriate Dispersion Models
(a)	Fuel Handling Accident	0-2 Hours Elevated Release at Exclusion Area Boundary (8.)
(b)	Steam Line Failure	0-2 Hours Ground Release at Exclusion Area Boundary (8.)
(c)	Rod Ejection Accident	Same as (d) below
(d)	Loss-of-Coolant Accident (Assume 50 percent Ground Release and 50 percent Elevated Release after 90 percent Iodine Removal by Filtration)	0-2 Hours at Site Boundary; 0-24 Hours and 0-30 Days at Boundary of Low Population Zone
(e)	Maximum Hypothetical Accident (MHA)	Same as (d) above
(f)	Loss of Electric Power	Same as (a) above
(g)	Loss-of-Coolant Accident during purge	Same as (a) above

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	(h)	Engineered Safeguards Leakage	Same as (a) above
8.	(i)	Lifetime Shim Bleed (Continuous Release)	Long Term Elevated Releases at Exclusion Area Boundary
8.	(j)	Start-up Expansion (7-Day Release)	0-7 Days Elevated Releases at Exclusion Area Boundary
8.	(k)	Reactor Building Purge	0-24 Hours Elevated Release at Exclusion Area Boundary
	(l)	Steam Generator Tube Failure	Same as (a) above
8.	(m)	Steam Generator Tube Leakage	Long Term Elevated Releases at Exclusion Area Boundary
8.	(n)	Pressurizer and Letdown Storage Tank Venting	0-7 Days Elevated Releases at Exclusion Area Boundary
8.	(o)	Waste-Gas-Tank Rupture	0-2 Hours Elevated Release at Exclusion Area Boundary

2.3.3 FUTURE

A meteorological program is continuing at the Oconee Nuclear Station to serve as a guide for waste management.

2.4 HYDROLOGY AND GROUNDWATER

2.4.1 CHARACTERISTICS OF STREAMS IN VICINITY

The major streams which drain the site are the Keowee River and the Little River. These are joined near Newry, South Carolina, to form the Seneca River which is a major tributary to the Savannah River. Dams are constructed on the Little and Keowee Rivers and the two reservoirs are connected by a canal to form Lake Keowee. Additional information on streams is furnished in 2.4.1 of PSAR.

2.4.2 WATER USAGE

Facilities have been constructed for the Town of Seneca to take its raw water supply from Lake Keowee at a point 6 miles from Oconee Nuclear Station and to discharge its treated waste into another tributary to the Seneca River watershed.

Several industrial plants, Clemson University, the Town of Clemson, the Town of Pendleton, and the Town of Anderson, South Carolina, take their raw water supplies
4. from Hartwell Reservoir.

Anderson raw water intake which is owned by Duke Power Company is located 40.4 stream miles downstream of Keowee tailrace. The Clemson-Pendleton raw water intake which is owned by Clemson University is located 13.7 stream miles downstream of Keowee tailrace.

The following table lists daily average raw water withdrawals from Hartwell Reservoir.

Year	Duke Anderson Intake	Clemson-Pendleton Intake
1968	0.734 mgd**	1.437 mgd*
1969	3.339 mgd	1.585 mgd*

* Based on finished water deliveries, it is estimated that 65 percent of water treated was used by Clenson University and 35 percent was delivered to Duke for resale to customers in Clemson, S. C., Pendleton, S. C. and adjacent areas.

** Intake was completed in 1968 and consumption reflects partial year of operation.

Additional information is furnished on municipal water supplies, including streamflows and dilution in answer to Question 2.3 in PSAR Supplement 1.

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Liquid effluent from Oconee's waste treatment facilities (Section 11) will be discharged to the tailrace of the Keowee Hydroelectric Station. Although not required by the design of Oconee, dilution can be increased by operation of the hydro units if necessary.

Additional information is furnished on water usage, including transit times and dilution for discharges into the condenser cooling water discharge canal and Keowee tailrace in PSAR 2.4.1 and in Answer to Question 2.4 in PSAR Supplement 1.

The Keowee tailrace minimum dilution flow with no hydro units operating will be
18. 4. measured during a low water period in Hartwell Reservoir in accordance with
U.S.G.S. Publication "Water Supply Paper 888."

2.4.3 FLOOD STUDIES

Flood studies documented in PSAR 2.4.2 show that Lake Keowee and Jocassee are designed with adequate margins to contain and control floods such as to pose no risk to the nuclear site.

2.4.4 DESIGN OF KEOWEE AND JOCASSEE DAMS

Duke Power Company has designed these dams based on sound Civil Engineering methods and criteria. These designs have been reviewed by a board of consultants and reviewed and approved by the Federal Power Commission in accordance with the license issued by that agency. Keowee Dam, Little River Dam, Oconee Intake Canal Dike, and Jocassee Dam have been designed to have an adequate factor of safety under the same conditions of seismic loading as used for design of the Oconee Nuclear Station.

Additional information has been furnished in PSAR 2.4.3 and as tabulated below:

Dam Seismic Loadings..........	Question 8.6 - PSAR Supplement 1
Dam Foundation Investigation...	Question 12.1 - PSAR Supplement 4
Intake Canal Dike..............	Question 12.2 - PSAR Supplement 4
Submerged Weir.................	Item 11 - PSAR Supplement 5
Submerged Weir.................	Item 1 - PSAR Supplement 6

The dams will be constructed, maintained and inspected consistent with their function in major hydro projects. The safety of such structures is the major objective of Duke's designers and builders, with or without the presence of the nuclear station.

2.4.5 GROUNDWATER

Groundwater at the site presents no special problems as documented in PSAR 2.4.4.

2.5 GEOLOGY

Site geology covered in Section 2.5 of PSAR has been confirmed by visual observations during construction and has presented no unusual design or construction problems.

2.6 SEISMOLOGY

No active or recent faulting has been recognized in the area of the proposed site. The closest known fault is the Brevard Zone, approximately 11 miles northwest of the site.

The Reactor Buildings' foundations are located on rock which has excellent strength properties and relatively small amplification of ground motion resulting from an earthquake.

The structural design criteria for the maximum hypothetical earthquake are 0.10g and 0.15g for Class 1 structures founded on bedrock and overburden, respectively. The structural design criteria for the design earthquake is 0.05g.

A detailed Seismology Study is included as a part of Appendix 2B in the PSAR. Additional information on seismology is included in Answers to Questions 2.7 and 8.5 of PSAR Supplement 1 and Question 11.3 of PSAR Supplement 4.

2.7 OCONEE ENVIRONMENTAL RADIOACTIVITY MONITORING PROGRAM

2.7.1 INTRODUCTION

The purpose of the environmental radioactivity monitoring program is to measure and evaluate the significance of contributions to the existing environmental radioactivity levels from plant operations.

The program is divided into pre-operational and operational phases, with the assumption that pre-operational levels may provide a base line to which operational levels can be compared. Such comparisons are complicated by additional nuclear weapons testing, seasonal and annual variations in fallout level, and discharges of radioactive material from other installations. However, pre-operational monitoring does document the existing radioactivity levels and their variability. Also the use of control locations well out of the influence of the plant can serve as a means of comparison for evaluating the plant's contribution to the environment during the operational phase.

During the operation of a nuclear power plant, the only contribution of radioactive materials to the environment will be due to the release of low level radioactive wastes; that is, from controlled releases of radioactive gases, airborne particulates, and liquid wastes. This activity, if it can be detected at all, is most likely to be found in the air and water beyond the plant where these materials are dispersed and diluted by stream flow and wind. Air and water are therefore considered as primary samples. They serve as one of the earliest indicators of change in environmental radioactivity levels. Samples of secondary significance, in this regard, include river bottom and lake sediment, vegetation, fish and animals, and milk.

2.7.2 THE PRE-OPERATIONAL PROGRAM

Table 2.1 describes the pre-operational environmental radioactivity monitoring program for the Oconee Nuclear Station. It lists the type samples, sampling locations, and the collection frequency. In general, the samples are counted for gross alpha and gross beta radioactivity. Gamma spectral analyses are also performed to identify the radionuclides involved. Specific analyses such as for tritium in water, cesium-137 and strontium-90 in milk, water, fish, and animal samples are also performed. The measurement of gamma dose and dose rate is considered as an appropriate sample to determine the radiation background of an area as well as to measure the effects of gaseous activity released during the operating period. Thermoluminescent dosimeters, environmental film badges, and beta-gamma survey instruments (geiger counters) are used for this measurement.

Criteria for the selection of the various sampling locations were as follows:

1. Water

   For comparison purposes water samples are collected:

   (a) Upstream
   (b) Near liquid effluent release point
   (c) Downstream of Site and Exclusion Area

   Particular emphasis has also been given to water sampling to evaluate the effect of the filling of Lake Keowee.

2. Airborne Particulates, Rain and Settled Dust

   Comparison of on-site vs. off-site locations near towns and populated areas; consideration given to prevailing wind direction.

3. Radiation Dose and Dose Rate

   Comparison of on-site vs. off-site locations near towns and populated areas; consideration given to prevailing wind direction.

4. Silt (River and lake bottom sediment; filtered solids from municipal drinking water supplies)

   For comparison purposes, silt samples are collected:

   (a) Upstream
   (b) Near liquid effluent release point
   (c) Downstream of Site and Exclusion Area

5. Terrestrial Vegetation

   Comparison of on-site vs. off-site locations; consideration given to prevailing wind direction.

6. Aquatic Vegetation, and/or Plankton, Bottom organisms
6. and Crustaceans

   For comparison purposes, samples are collected, depending on availability:
   (a) Upstream, from Lake Keowee
   (b) Downstream, from Hartwell Reservoir as close to liquid effluent release point as they can be obtained

36. 6. 7. Milk — From local area dairies within 10 miles of site and nearby farms in prevailing wind directions.

8. Fish and Animals — Fish samples of Carp, Shad and Bass, as well as rabbit, squirrel, and other mammals are collected in accordance with the recommendations of, and in coorporation with, the S. C. Wildlife
6. Resources Commission from:

	Fish	Animals
(a)	Lake Keowee	Exclusion Area
(b)	Hartwell Reservoir	

9. Miscellaneous — Investigation of special situations made to provide program flexibility and extended coverage; such as may be required due to nuclear weapons testing or unusual fallout conditions. Includes study of Lake Keowee tributary streams and modification as lake fills. Investigation of reported deposits of uranium or thorium in area of plant.

The sampling stations were established in the Oconee environs at the end of 1968, and a laboratory for the analysis and counting of the samples was also established at that time. The laboratory equipment includes a low background gas flow porportional counter for measuring gross alpha and gross beta radioactivity and a 400 channel gamma scintillation spectrometer (multi-channel analyzer). In addition, some samples are sent to commercial laboratories for analysis.

The full scale environmental sampling program was begun in January, 1969. Thus, two years of pre-operational monitoring data will be obtained prior to the operation of Unit 1. Water samples from the Keowee and Little Rivers which flow by the site and well water from private residences in the area have been collected, analyzed, and counted since late 1966. The results of this early sampling as well as the results for the first quarter, 1969, are shown in Table 2-2.

The pre-operational environmental radioactivity program for Oconee has been discussed with the South Carolina State Board of Health, Division of Radiological Health, and the South Carolina Pollution Control Authority. The U. S. Government Fish and Wildlife Service has also been advised of the program through their district office in Atlanta, Georgia. In addition, the program was discussed with the South Carolina Wildlife Resources Department. This latter department is cooperating with Duke Power Company in regard to the collection of fish and animal samples. They have made recommendations as to what specimens should be collected and are supplying fish samples from the Hartwell Reservoir and Lake Keowee. They have also issued a special research permit to Duke Power Company for the collection of animal samples.

The results of the environmental radioactivity monitoring program to date are comparable to those reported from throughout the country by the U. S. Department of Health, Education, and Welfare (Public Health Service) in their "Radiological Health Data and Reports," for the same period. It is of interest also to note that radium daughter products have been observed, as a result of gamma analysis, to exist in considerable amounts in deep well water. Further investigation has shown that this condition seems to be peculiar to the Piedmont area of the Carolinas.

2.7.3 THE OPERATIONAL PROGRAM

The Environmental Radioactivity Monitoring Program will continue during the operating period with the counting and analysis of samples essentially as outlined in 2.7.2. Table 2-1a describes what is generally intended as the operational program for the Oconee Nuclear Station with one, two, or three units in operation.

The Operational Program was put into effect prior to the operation of Unit 1, so that background data, reflecting changes made in the program, was available for comparison purposes after operation commenced.

Minor reductions in the frequency of collection of a given type sample may be made in the operational program if the radioactive liquid and airborne wastes released remain within the annual average limits for the station with three units in operation. If the radioactive wastes released increase beyond the annual average limits for the station, the monitoring program may be expanded. This may take the form of additional monitoring stations on-site, or off-site, or an increase in the frequency of collection of existing samples or a change in the types of samples collected.

2.7.4 CONCLUSION

The environmental radioactivity monitoring program for the Oconee Nuclear Station is conducted by the station Health Physics Supervisor. The program is directed and reviewed by the Duke Power Company System Health Physicist.

Environmental monitoring results will be correlated with information on station radioactive waste releases, site meteorological data, and radiological controls and with information obtained from the installed process radiation monitoring system. Results will also be compared with published information from the national radiological surveillance programs reported by the Environmental Protection Agency and with environmental monitoring reports of other nuclear installations in the area.

Results of the Oconee Environmental Radioactivity Monitoring Program will be made available to the State of South Carolina and to the Federal agencies mentioned above who have a direct interest and concern in these matters.

It is expected that the results of the Environmental Radioactivity Monitoring Program for the Oconee Nuclear Station will demonstrate the effectiveness of plant control over radioactive waste disposal operations and of compliance with Federal and State regulations for the disposal of these materials.

It is expected that the results of the Environmental Radioactivity Monitoring Program for the Oconee Nuclear Station will demonstrate the effectiveness of plant control over radioactive waste disposal operations and of compliance with Federal and State regulations for the disposal of these materials.

Table 2-1

OCONEE PRE-OPERATIONAL ENVIRONMENTAL RADIOACTIVITY MONITORING PROGRAM

CODE	Frequency
Monthly - M	
Quarterly - Q	
Annually - A	
Type of Sample - (A) thru (M)	

Code No.	Location	(A) H_2O Well - Residence	(B) H_2O Finished - Water Supply	(C) H_2O Raw - Water Supply	(D) H_2O Surface - River, Lakes	(E) Rain, Settled Dust - Fallout	(F) Air - Particulate	(G) Vegetation - Terrestrial	(H) Vegetation - Aquatic	(I) Bottom Sediment Water Supply & Lakes	(J) Radiation Dose & Rate TLD & Film, Instrument	(K) Animals (3) 1 Mile Radius	(L) Fish (3) Lakes	(M) Milk - Local Daries
000	Site: Visitors Center					M	M	Q			Q			
000.3	1st Bridge North of Site on New 183 Connecting Canal				M+A				Q	Q				
000.4	2nd Bridge North of Site on New 183				M									
000.5	1 Mile Radius of Site - Specify N, S, E, W											Q		
000.6	Keowee Lake												Q	
000.7	@ Bridge on 183 Existing				M					Q				
000.8	Residence within Exclusion Area	Q												
000.9														
000.10														
001	Salem: Vol. Fire Dept. Lot										Q			
001.3	4.5 Mi. N.E. of Salem on Hwy. 11 @ Bridge (Cedar Creek)				A									
001.4	8.0 Mi. E. of Salem @ Bridge (Crow Creek)				A									
001.5														
001.6														
002	Walhalla: Branch Rd. Sub Station					M					Q			
002.1	7.5 Miles West of Site on Hwy. 183													Q
002.2														
003	Keowee: High School Hwy. 16 (Opposite Side)										Q			
003.1														
003.2														
004	Seneca: Oconee Memorial Hospital										Q			
004.1	Water Supply, Lake Keowee Intake, (When Completed)		M	M						Q				
004.2														
005	Newry: Abandoned High School on S. C. 130							Q			Q			
005.1	Spill Dam (L.R. & Keowee Spill)				M					Q				
005.3	Hwy. 27 at Bridge				M					Q				
005.4	3.75 Mi. W. of Newry on Keowee Hwy. @ Bridge (Cain Creek)				A									
005.5	3.25 Mi. N.W. of Newry on Keowee Hwy. @ Bridge (Crooked Creek)				A									
005.6														
005.7														
006	Clemson: Meteorology Plot					M	M	Q			Q			
006.1	Water Supply		M	M										
006.2	Intake Hartwell Reservoir K-3				M+A				Q	Q				
006.3														
006.4														
006.5														
007	Central, S. C.: Joint Sub Station Hwy. 93										Q			
007.1														
007.2														
008	Liberty, S. C.: Branch Office Yard										Q			
008.1														
008.2														

(Cont'd)

Table 2-1

OCONEE PRE-OPERATIONAL ENVIRONMENTAL RADIOACTIVITY MONITORING PROGRAM

CODE	
Monthly - M	Frequency
Quarterly - Q	
Annually - A	
Type of Sample - (A) thru (M)	

Code No.	Location	H_2O Well - Residence (A)	H_2O Finished - Water Supply (B)	H_2O Raw - Water Supply (C)	H_2O Surface - River, Lakes (D)	Rain, Settled Dust - Fallout (E)	Air - Particulate (F)	Vegetation - Terrestrial (G)	Vegetation - Aquatic (H)	Bottom Sediment Water Supply & Lakes (I)	Radiation Dose & Rate TLD, Film, Instrument (J)	Animals (3) 1 Mile Radius (K)	Fish (3) Lakes (L)	Milk - Local Dairies (M)
009	Six Mile, S. C.: Microwave Tower Hwy. 137										Q			
009.1														
009.2														
010	Pickens, S. C.: Branch Office Yard					M					Q			
010.1														
010.2														
011	Floating Station: Subject to Change with Conditions										Q			
011.1														
011.2														
012	Anderson, S. C.: Water Supply		M	M						Q				
012.1														
012.2														
013	Hartwell Reservoir: 5.8 Mi. South of Keowee Dam												Q	
013.1														
013.2														

Note: 1. 000.3 and 006.2 will be sent to outside services for analysis for ^{3}H and ^{90}Sr (2 gals. each location).
2. Fish speciments will be collected alternately from Lake Keowee and Hartwell.
3. 001.3, 001.4, 005.4, and 005.5 will be collected once per year during rainy season.

Note: Location numbers that appear in Table 2-2 which are not shown above are results of special investigations at the general location indicated.

TABLE 2-1a

OCONEE OPERATIONAL ENVIRONMENTAL RADIOACTIVITY MONITORING PROGRAM

COLLECTION FREQUENCY

Weekly	W
Monthly	M
Quarterly	Q
Semiannually	S
Triennially	T

		TYPE OF SAMPLE													
		WELL WATER Residence	FINISHED WATER Water supply	RAW WATER Water supply	SURFACE WATER River, Lakes	RAIN, SETTLED DUST Fallout	AIR Particulates, Iodine	VEGETATION Pasture grass, forage	VEGETATION Commercial crops	VEGETATION Aquatic	BOTTOM SEDIMENT Water supply & Lakes	RADIATION DOSE & RATE TLD & Instrument	FISH Lakes	MILK Local dairies	SOIL
000	Site: Visitors Center, Station #1					M	W	Q				Q			T
000.1	Station #2					M						Q			
000.2	Station #3					M						Q			
000.3	Bridge N of Site on Hwy 183 connecting canal				M										
000.4	Near liquid effluent release point									S	S				
000.5	1-mile radius of Site (including Lake Keowee)				M					S	S		S		
000.6	Lake Keowee cooling water discharge				M						S	Q			
000.7	At bridge on Hwy 183 existing				M						S				
000.9	NW Hwy 183											Q			
000.10	Skimmer wall											Q			
000.11	E Hwy 183											Q			
000.12	Construction living quarters											Q			
000.13	Boat Dock - Visitors Center											Q			
000.14	Keowee Hydro intake											Q			
000.15	Site fence, North											Q			
000.16	Site fence, North											Q			
000.17	Site fence, West											Q			
000.18	Site fence, West											Q			
000.19	Site fence, South											Q			
001	SALEM: Volunteer Fire Department lot											Q			
002	WALHALLA: Branch Road Substation					M						Q			
002.1	5 miles W of Site on Hwy 183							Q						W	T
003	KEOWEE: High School Hwy 188 (Opposite side)											Q			
004	SENECA: Oconee Memorial Hospital											Q			
004.1	Water supply, Lake Keowee intake		M	M											
005	NEWRY: Abandoned High School off SC 130											Q			
005.2	Hwy 27 at bridge				M					S	S				
006	CLEMSON: Meteorology plot					M	W	Q				Q			T
006.1	Water supply		M	M											
006.2	Intake Hartwell Reservoir K-3										S				
006.3	Dairy													Q	
007	CENTRAL: Joint Substation Hwy 93											Q			
008	LIBERTY: Branch Office yard											Q			
009	SIX MILE: Microwave tower Hwy 137						W					Q			
010	PICKENS: Branch Office yard					M						Q			
011	FLOATING STATION: Warpath Boat Landing											Q			
012	ANDERSON: Water supply		M	M											
013	HARTWELL RESERVOIR: 5.8 miles S of Keowee Dam				M								S		
014	Old Hwy 183 at Lake	Q					W								
015	Farms within a 5-mile radius of Site	Q						Q	Q*					W*	

*If sufficient quantities are available for sampling

TABLE 2-2
SUMMARY OF PRE-OPERATIONAL MONITORING RESULTS

1. Water	Suspended Solids pCi/l alpha	Suspended Solids pCi/l beta	Dissolved Solids pCi/l alpha	Dissolved Solids pCi/l beta	Total Activity pCi/l alpha	Total Activity pCi/l beta		Tritium pCi/l
								(< 2.7×10^{-3} = N.D)
A. Rivers								
1967 Keowee River At Site	N.D. - 4.73	1.16 - 24.09	N.D. - 2.17	N.D. - 7.67	N.D. - 5.03	5.58 - 25.57	range	
	2.02	8.21	0.75	3.20	2.78	11.41	average	< 2.7×10^{-3}
1967 Little River At Newry, S.C.	N.D. - 3.19	N.D. - 6.80	N.D. - 5.36	4.59 - 16.73	0.34 - 6.74	6.11 - 20.18	range	
	1.03	2.52	1.85	10.17	2.88	12.68	average	< 2.7×10^{-3}
1968 Keowee River At Site	N.D. - 5.42	N.D. - 9.36	0.96 - 21.70	8.00 - 38.20	1.45 - 23.78	11.19 - 43.57	range	
	1.31	4.52	5.75	17.75	7.04	23.33	average	< 2.7×10^{-3}
1968 Little River At Newry, S.C.	N.D. - 2.17	N.D. - 12.07	0.28 - 3.29	6.25 - 30.14	0.28 - 8.68	7.28 - 31.81	range	
	0.82	5.22	2.30	13.30	3.82	19.70	average	< 2.7×10^{-3}
B. Open and Deep Wells at Residences On-Site and in Surrounding Area			Designation	Year				
			On-Site (3 locations, from residences then existing)	1966	3.30 - 5.92	11.86 - 28.58	range	
					4.16	17.74	average	
			Surrounding Area (3 locations, from residences still existing)	1966	2.17 - 4.96	10.26 - 25.70	range	
					3.72	16.20	average	
			On-Site 1	1967	N.D. - 11.94	2.69 - 26.97	range	
					4.57	16.17	average	< 2.7×10^{-3}
			On-Site 1	1968	N.D. - 33.64	9.85 - 62.46	range	
					7.82	25.00	average	
			On-Site 2	1967	N.D. - 7.60	1.36 - 19.96	range	
					2.70	10.40	average	< 2.7×10^{-3}
			On-Site 2	1968	N.D. - 4.91	10.17 - 34.78	range	
					3.00	20.00	average	

Note: 1. Above measurements at 90% confidence level, based on natural uranium alpha and "apparent Cesium-137 beta activity," calibration standards.
2. N. D. = non-detectable

TABLE 2-2 (Continued)
SUMMARY OF PRE-OPERATIONAL MONITORING RESULTS
Averages For First Quarter 1969

1. Water	Suspended Solids pCi/l		Dissolved Solids pCi/l		Total Activity pCi/l		Gamma Analysis Results
	alpha	beta	alpha	beta	alpha	beta	
Residences							
000.1	-	-	-	-	N.D.	3.55	Indications of Radium daughter products
000.2	-	-	-	-	N.D.	2.86	
Municipal Water Supplies							
006.1 Clemson-Pendleton							
raw water	0.12 (2)	0.64 (2)	0.08 (2)	3.80 (2)	0.16 (3)	4.78 (3)	background
finished water	-	-	-	-	N.D.	2.62	
012 Anderson							
raw water	0.08 (2)	0.49 (2)	N.D. (2)	4.04 (2)	0.05 (3)	4.30 (3)	background
finished water	-	-	-	-	N.D.	3.44	
Rivers and Lakes							
000.3 Site	0.11	0.83 (2)	0.08	3.20 (2)	0.13	3.56 (3)	background
001.1 Salem	0.07	0.30 (1)	0.08	1.98 (1)	0.21	4.96 (2)	background
001.2	0.15	N.D. (1)	0.08	1.90 (1)	0.29	2.63 (2)	background
005.1 Newry	0.52	2.59 (2)	0.17	2.89 (2)	0.81	6.87 (3)	background
005.2	0.16	0.98 (1)	0.08	1.90 (1)	0.64	4.93 (2)	background
005.3	0.20	0.56 (2)	0.08	3.10 (2)	0.29	3.81 (3)	background
006.2 Clemson	0.24	0.45 (2)	0.21	2.75 (2)	0.44	4.19 (3)	background

2. Rain and Settled Dust	Total Activity nCi/m^2 alpha	beta	
002 Walhalla	0.02	5.81	Indications of fission products, possibly Ce^{144} - Pr^{144}
010 Pickens	0.03	6.73	

3. Terrestrial Vegetation	pCi/g alpha	beta	
000 Site	2.38	558.6	Appears to be K^{40}
005 Newry	N.D.	1006.5	
006 Clemson	3.06	647.2	

TABLE 2-2 (Continued)
SUMMARY OF PRE-OPERATIONAL MONITORING RESULTS
Averages for First Quarter 1969

	Total Activity pCi/g alpha	Total Activity pCi/g beta	Gamma Analysis Results
4. Plankton			
005.3 Newry	3.70	22.86	Appears to be K^{40}
006.2 Clemson	9.27	38.33	
5. Bottom Sediment			
000.3 Site	0.34	0.64	
001.1 Salem	0.46	1.90	
005.1 Newry	0.23	0.91	
005.3	1.36	7.12	
006.1 Clemson	1.04	3.80	
012 Anderson	0.61	2.49	

6. Radiation Dose & Dose Rate

000 Site
001 Salem
002 Walhalla
003 Keowee (High School)
004 Seneca (Mem. Hosp.)
005 Newry
006 Clemson
007 Central
008 Liberty
010 Pickens
011 Lake Jocassee Area

Initial data indicates a gamma dose rate of 0.01 mR/h and a dose of approximately 21mRem/90 days for all locations,measured at three feet above the ground.

7. Animals

			pCi/g
000 Site (Rabbit)	Sr^{90}	bone	22.52 ± 0.45
	Cs^{137}	muscle	0.50 ± 0.01

8. Fish

	pCi/g Sr^{90}	pCi/g Cs^{137}
Lake Hartwell		
Carp (Adult)	1.52 ± 0.13	0.08 ± 0.02
Largemouth Bass (Fingerlings)	1.19 ± 0.07	0.36 ± 0.04
Gizzard Shad (Fingerlings)	0.35 ± 0.11	0.39 ± 0.02

9. Milk

	pCi/l Cs^{137}	pCi/l Sr^{90}	pCi/gCa	
002 Walhalla	11.4 ± 1.9	13.5 ± 1.7	Sr^{90}	11.9 ± 1.5

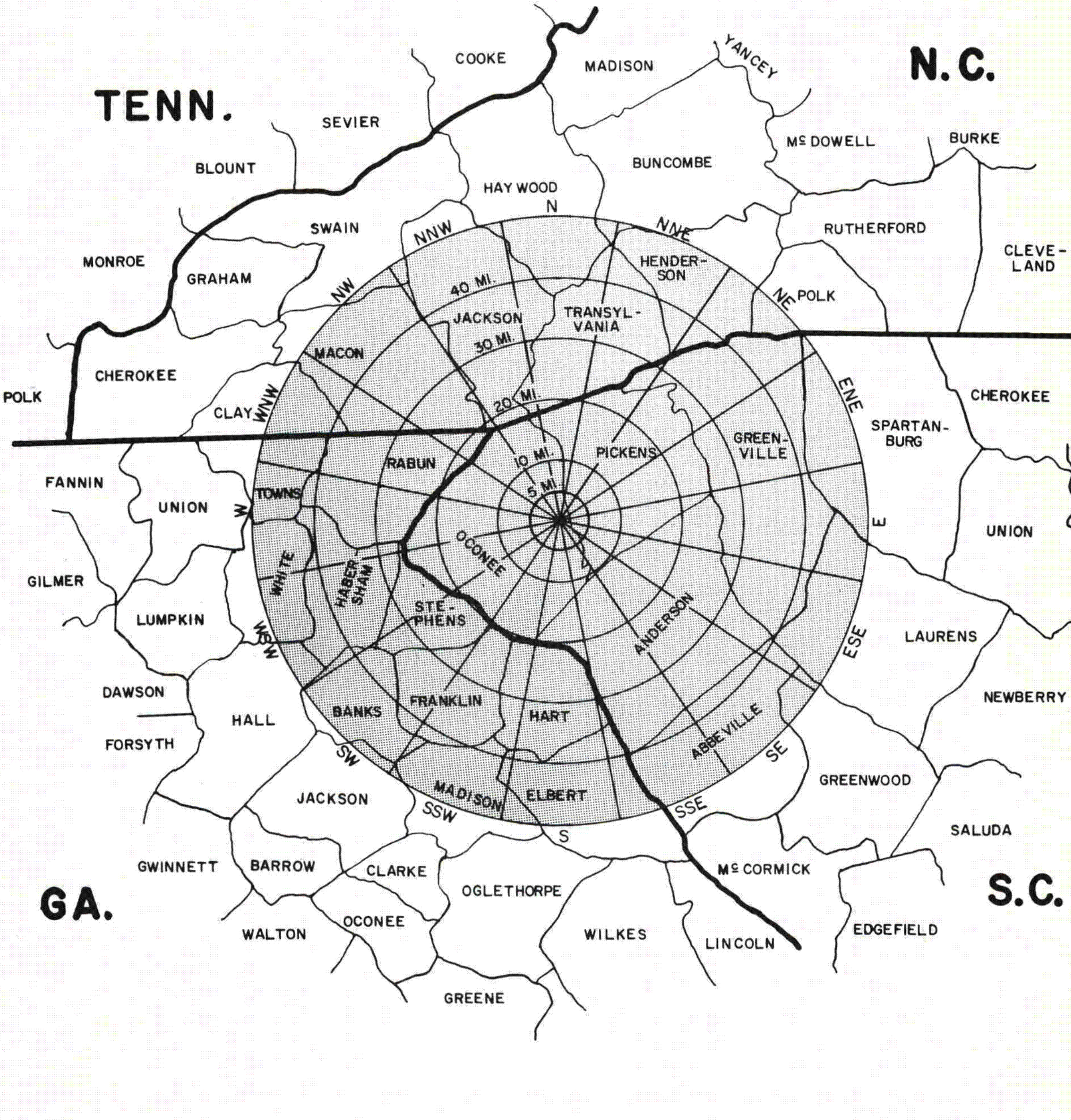
Results of gamma analysis show natural K^{40} as the predominant activity.

NOTE: Samples in categories 1 through 5 were measured at the 90% confidence level, based on RaD+E alpha, and "apparent Cesium^{137}beta activity," calibration standards.

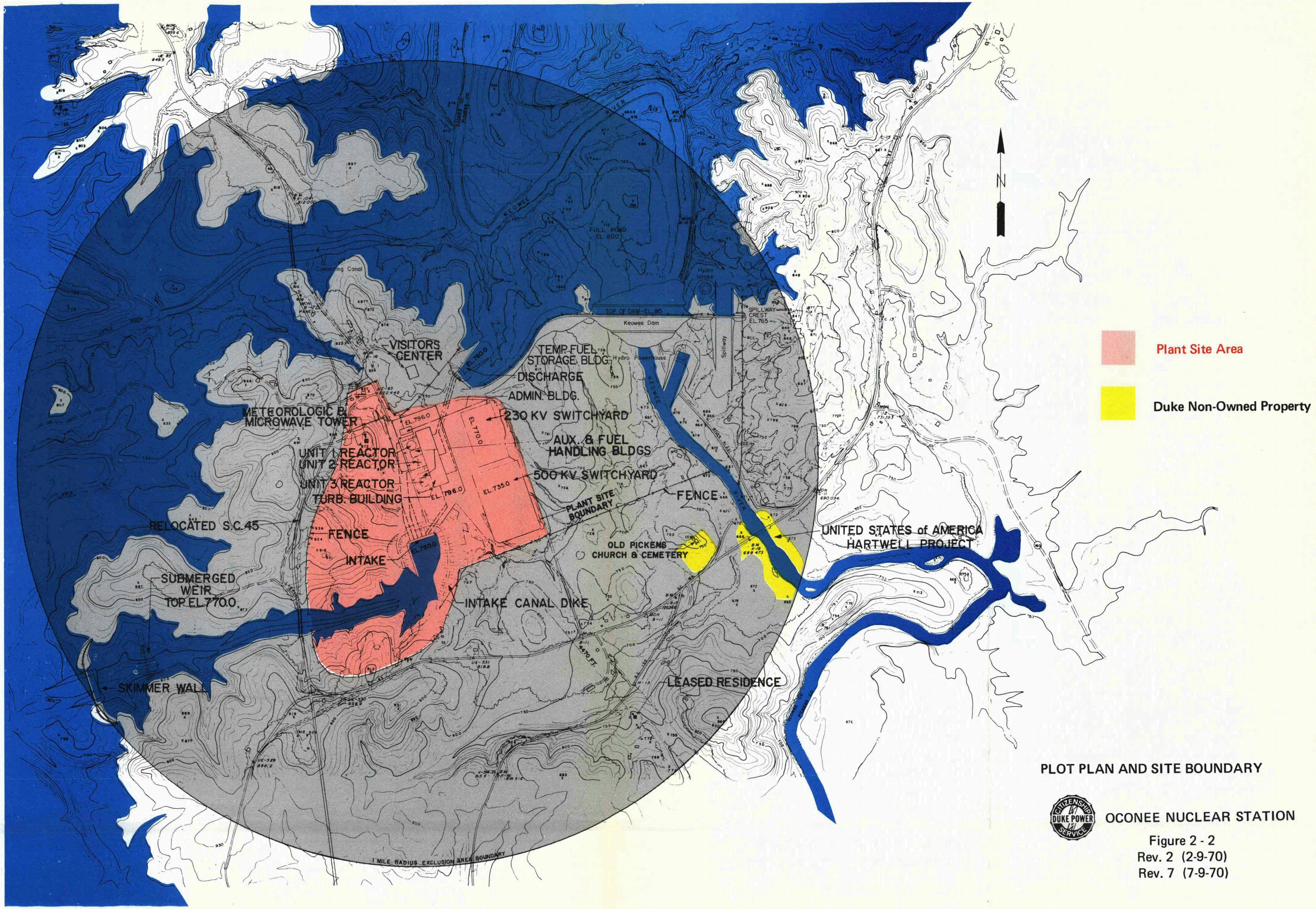
N.D. = non-detectable

(1), (2), (3) refer to numbers of samples averaged.

COUNTIES WITHIN
A 50 MILE RADIUS



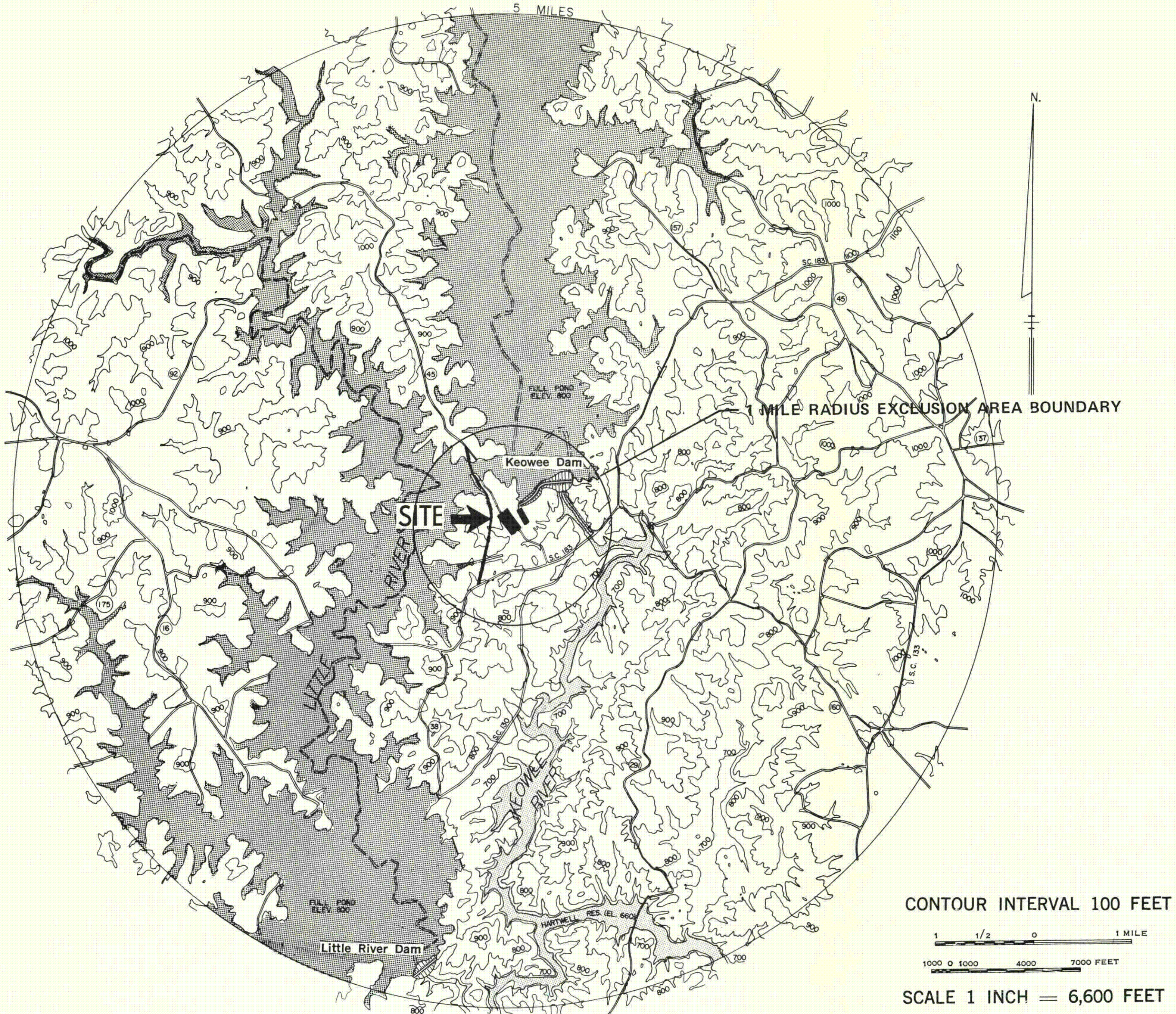
OCONEE NUCLEAR STATION
FIGURE 2-1



PLOT PLAN AND SITE BOUNDARY

OCONEE NUCLEAR STATION

Figure 2 - 2
Rev. 2 (2-9-70)
Rev. 7 (7-9-70)

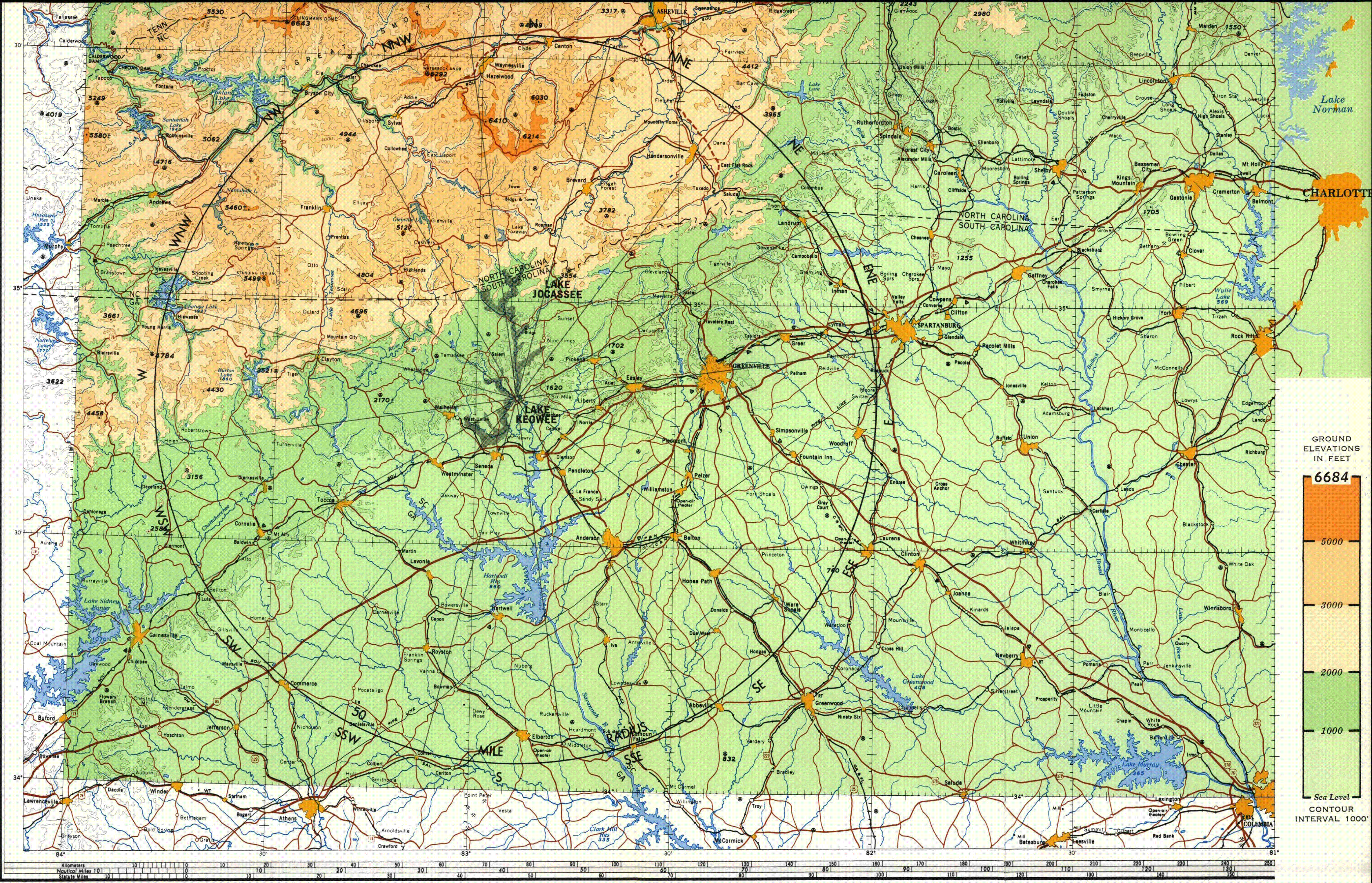


SITE TOPOGRAPHY
5 MILE RADIUS

OCONEE NUCLEAR STATION

Figure 2 - 3

Rev. 2 2/9/70



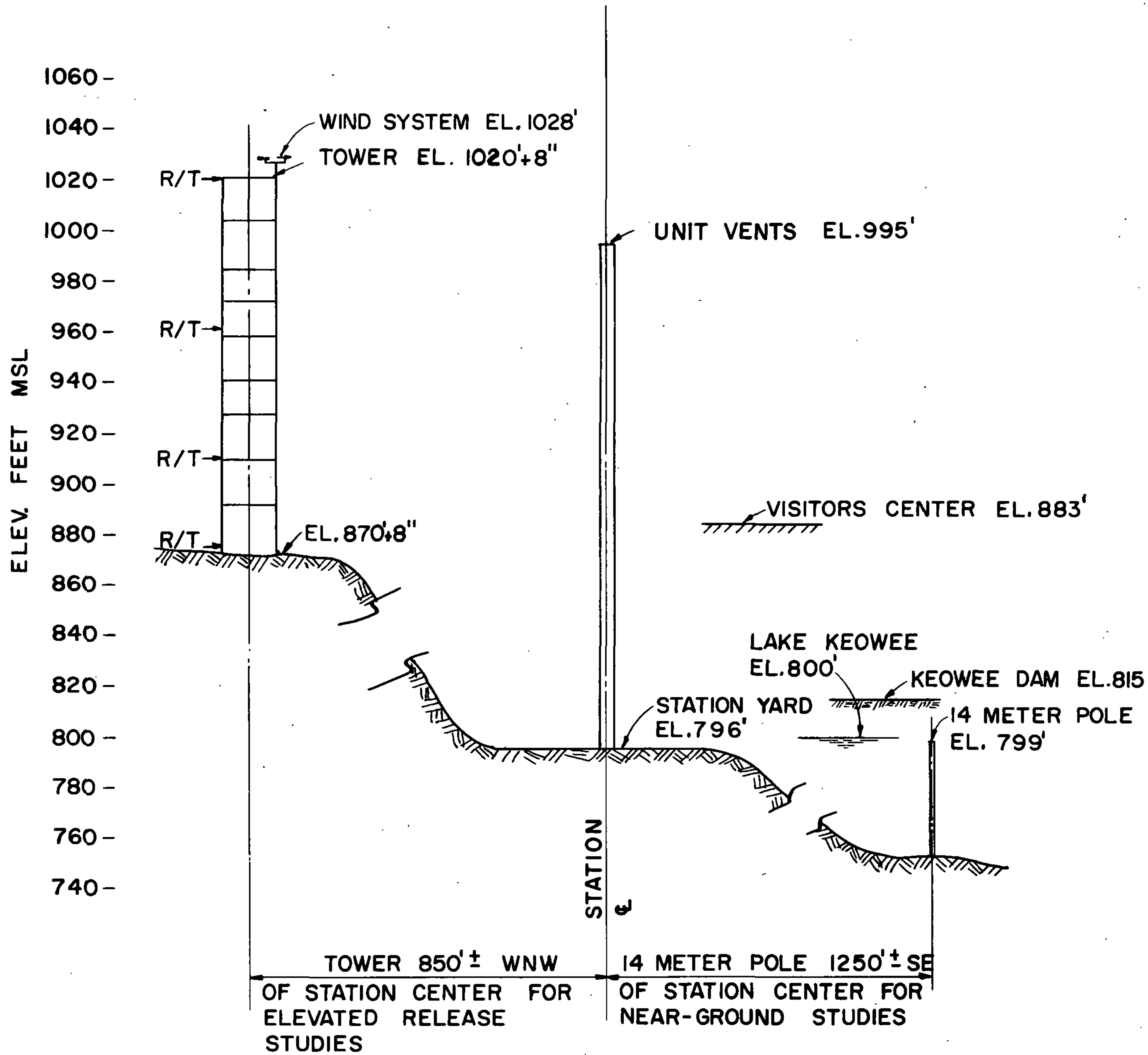
CHARLOTTE

ENVIRONMENTAL SCIENCE SERVICES ADMINISTRATION
COAST AND GEODETIC SURVEY
Base: Edition of Jan. 1961 Revised July 1966

GENERAL AREA MAP

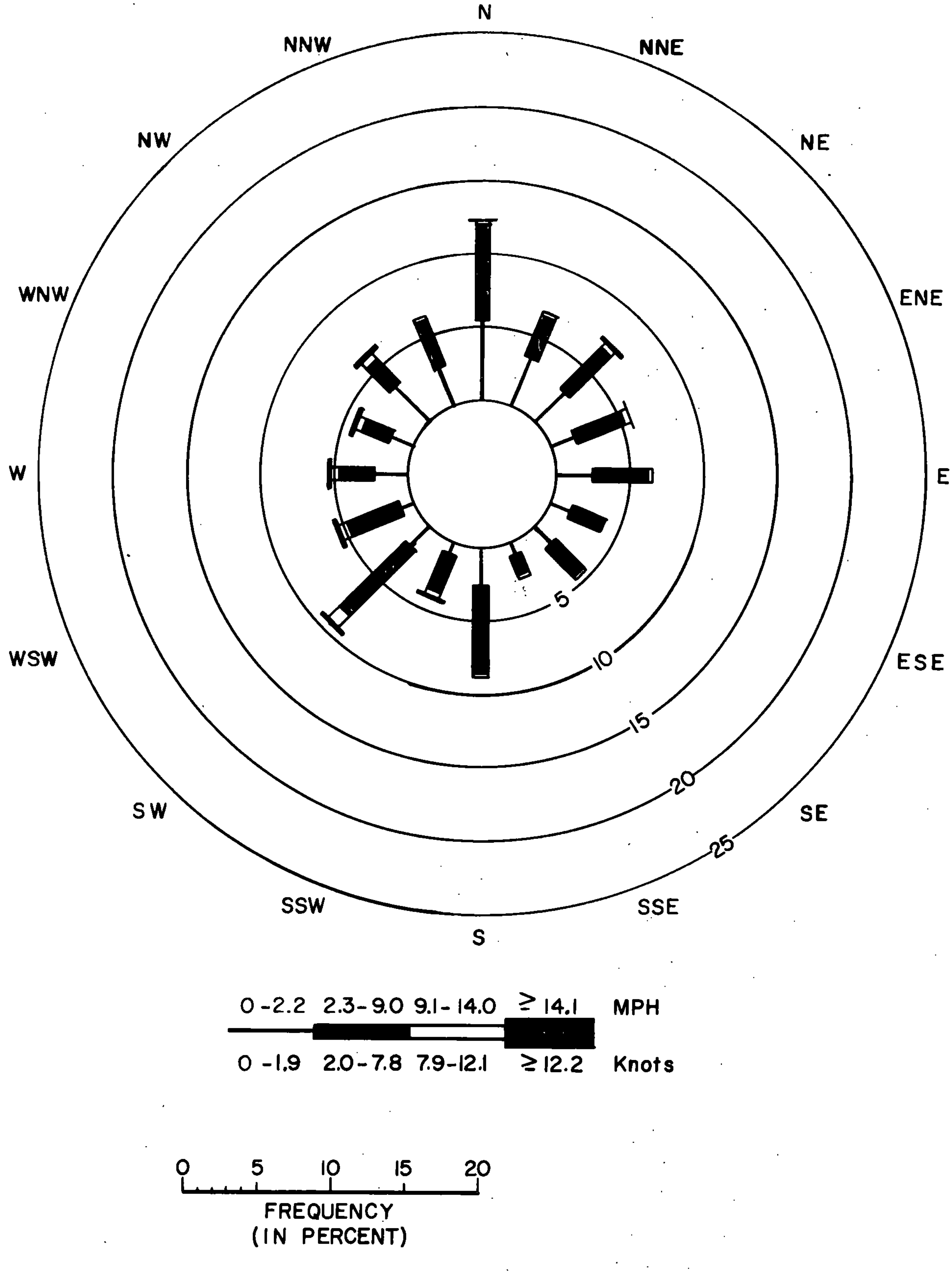
OCONEE NUCLEAR STATION

FIGURE 2-4



R/T = RESISTANCE THERMOMETER

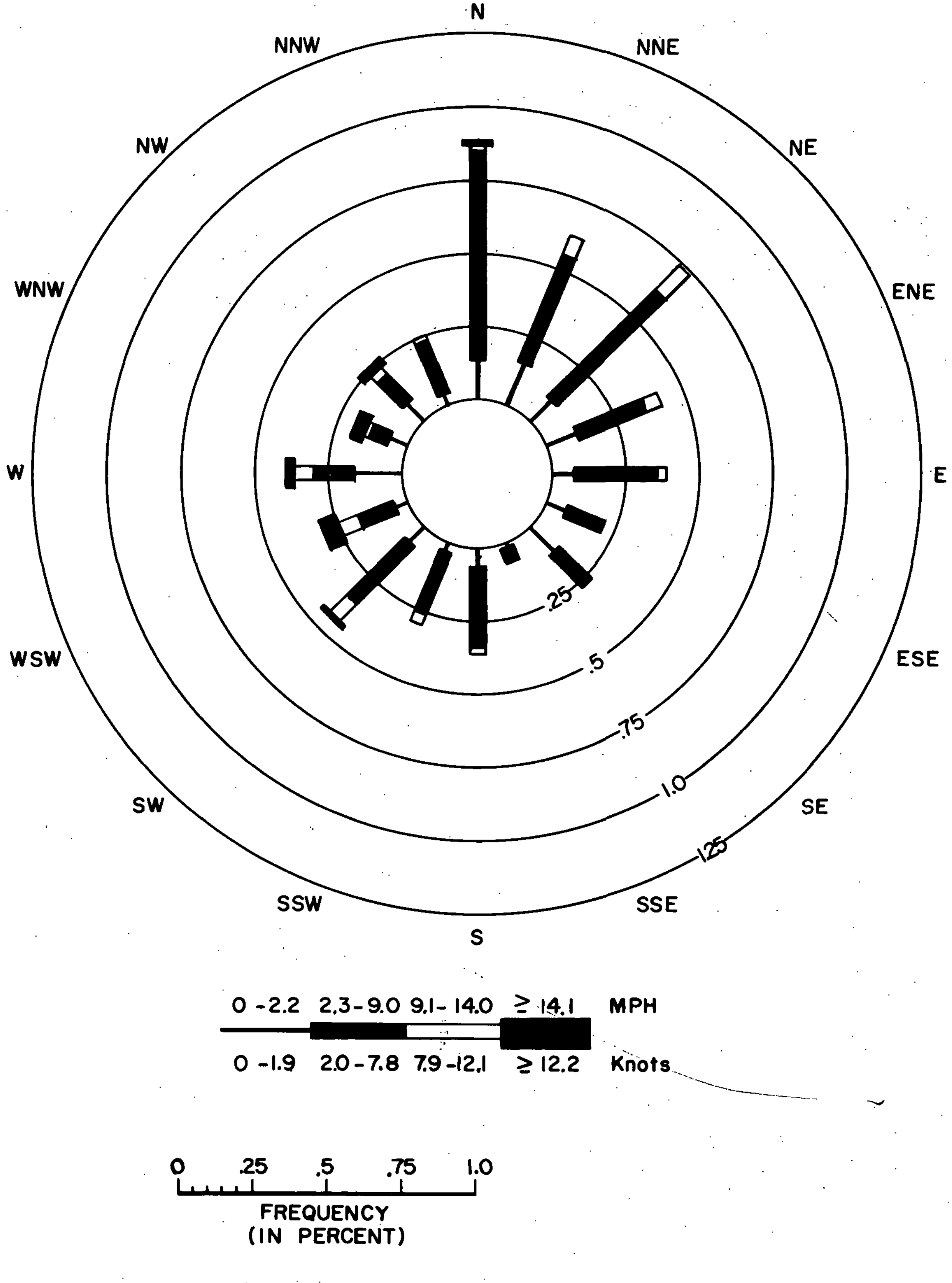
RELATIVE ELEVATIONS OF METEOROLOGICAL INSTRUMENTS



ANNUAL SURFACE WIND ROSE
(1 YEAR RECORD)
OCT. 19, 1966 - OCT. 31, 1967

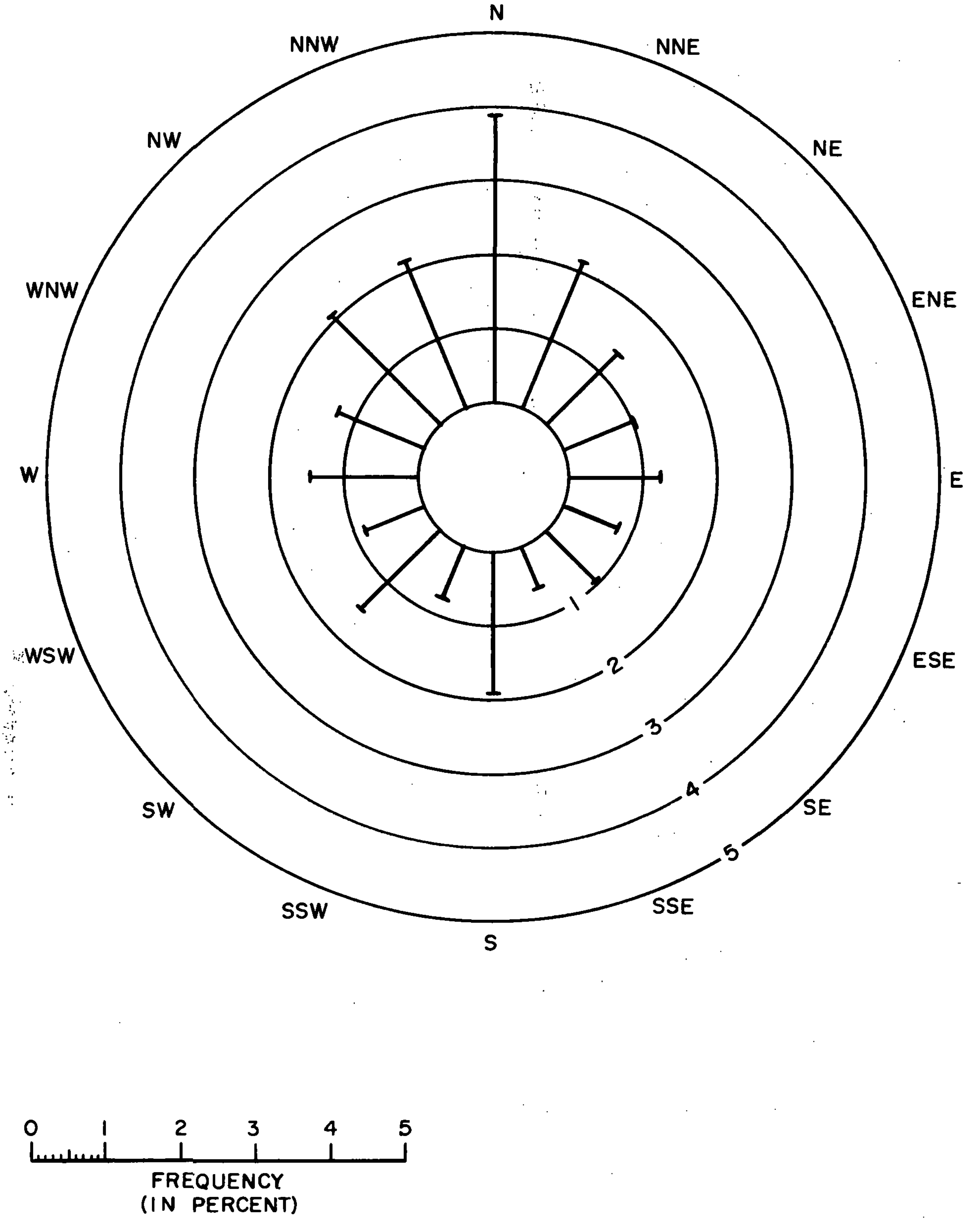
OCONEE NUCLEAR STATION

Figure 2 - 6



PRECIPITATION SURFACE WIND ROSE
(1 YEAR RECORD)
OCT. 19, 1966 - OCT. 31, 1967

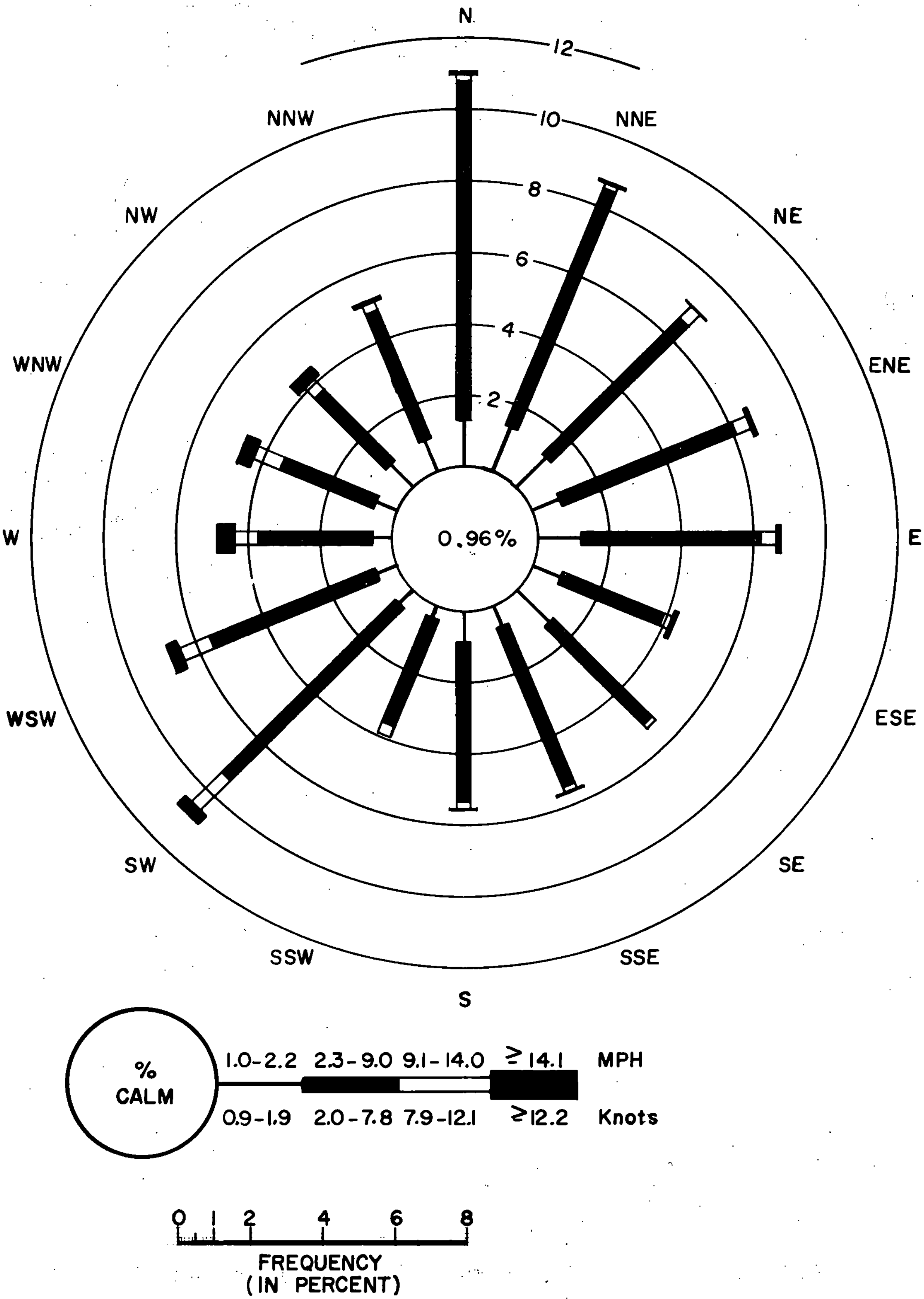
OCONEE NUCLEAR STATION
Figure 2 - 7



SURFACE WIND FREQUENCY DISTRIBUTION
DURING LOW-LEVEL
TEMPERATURE INVERSION CONDITIONS
(1 YEAR RECORD OF % OF TOTAL WIND)
OCT. 19, 1966 - OCT. 31, 1967

OCONEE NUCLEAR STATION

Figure 2 - 8

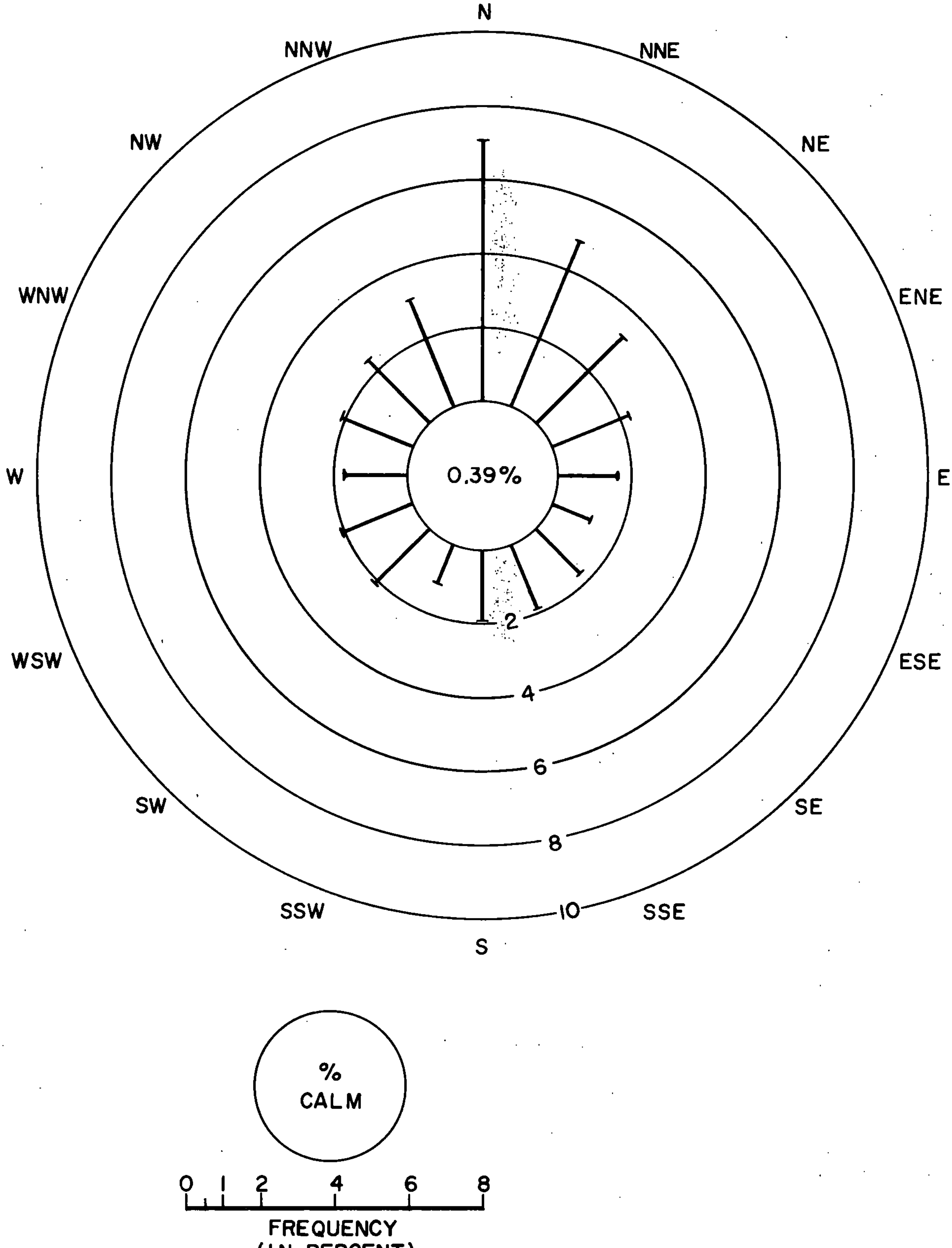


WIND ROSE FOR TOWER WINDS

FOR PERIOD JUNE 19, 1967 - MAY 31, 1968

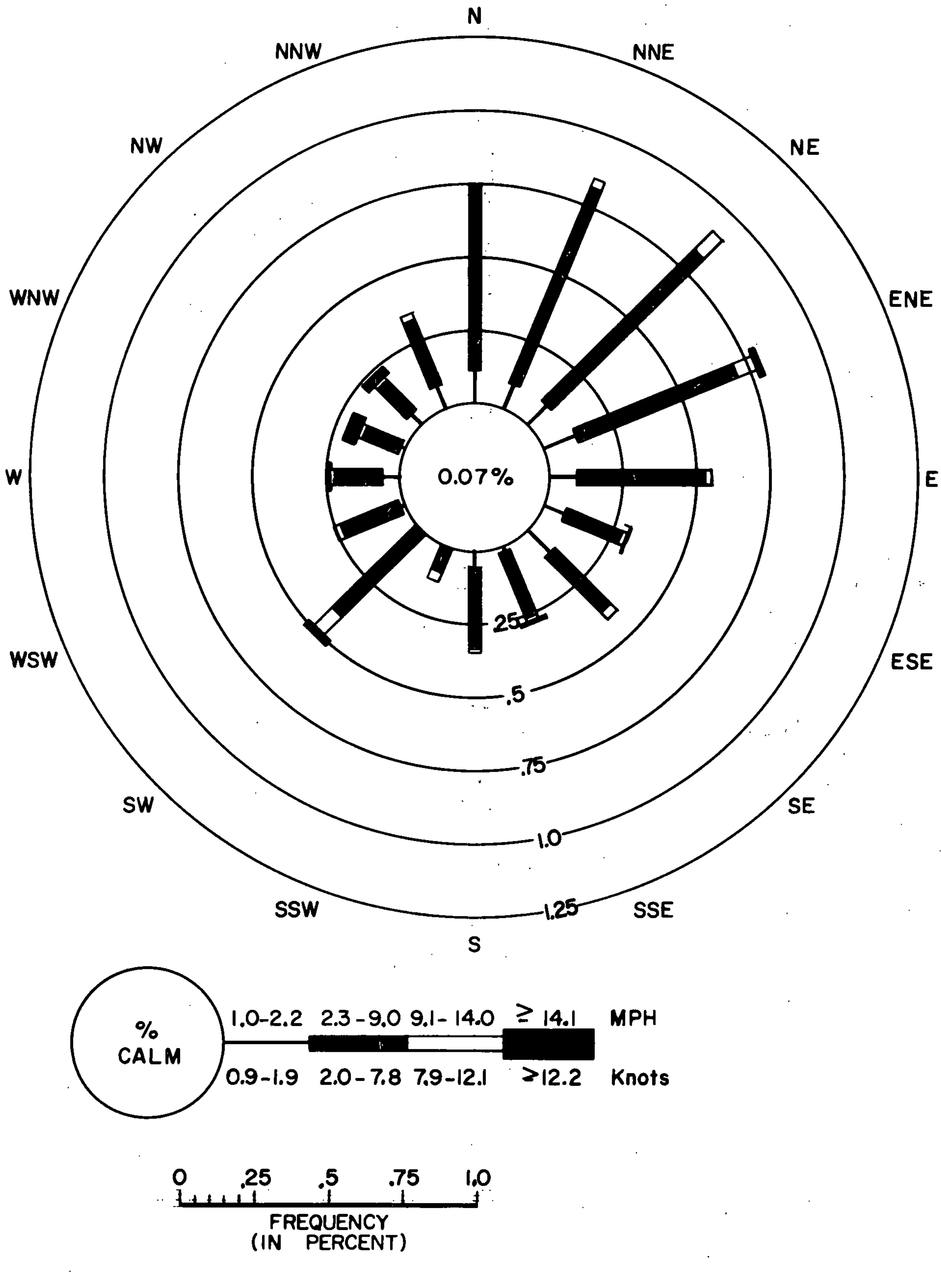
OCONEE NUCLEAR STATION

Figure 2 - 9



FREQUENCY DISTRIBUTION FOR TOWER WINDS DURING LOW-LEVEL TEMPERATURE INVERSION CONDITIONS
FOR PERIOD JUNE 19, 1967 - MAY 31, 1968

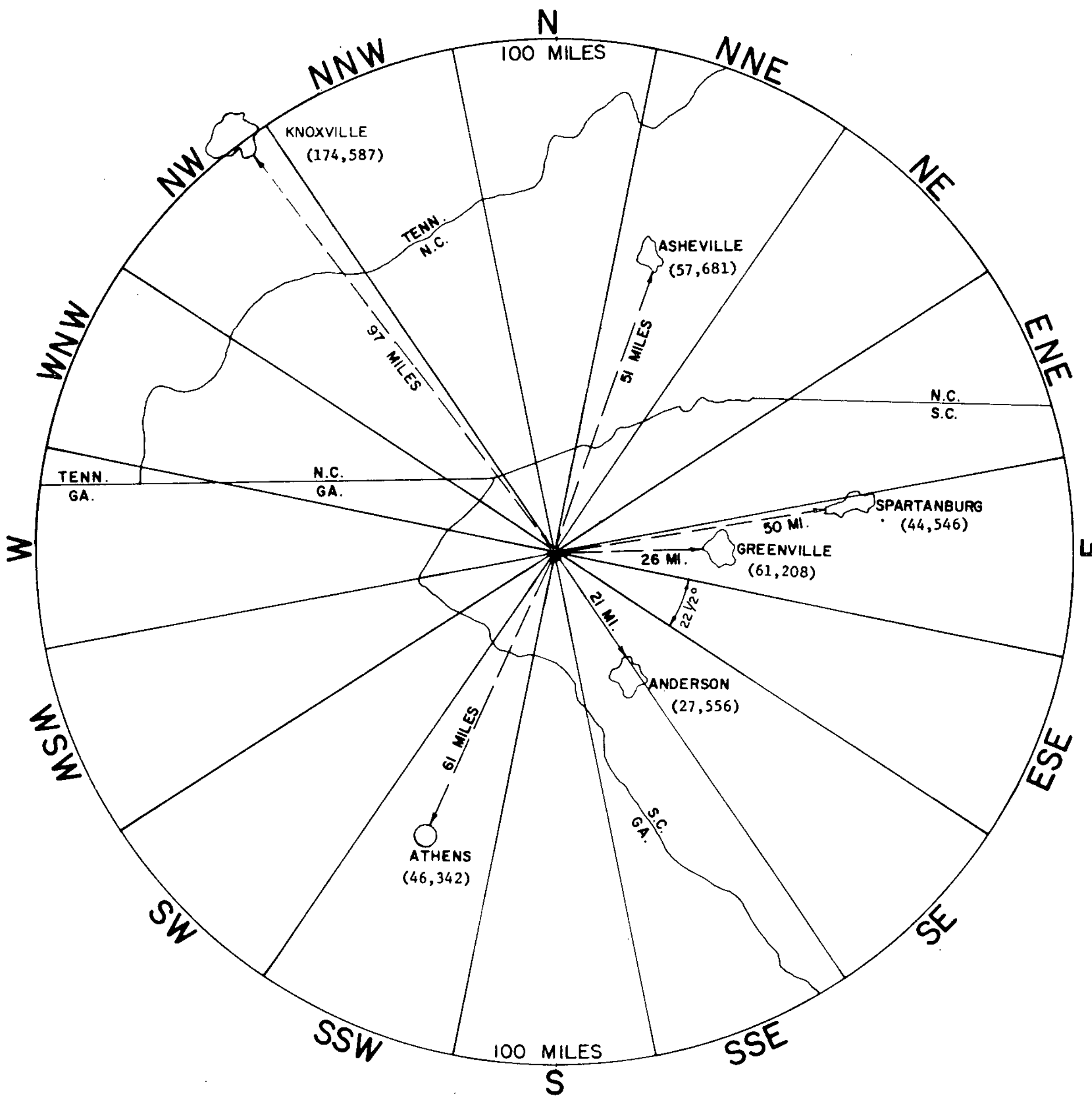
OCONEE NUCLEAR STATION
Figure 2 - 10



PRECIPITATION WIND ROSE FOR TOWER WINDS

FOR PERIOD JUNE 19, 1967-MAY 31, 1968

POPULATION CENTER DISTANCES
WITHIN
A 100 MILE RADIUS

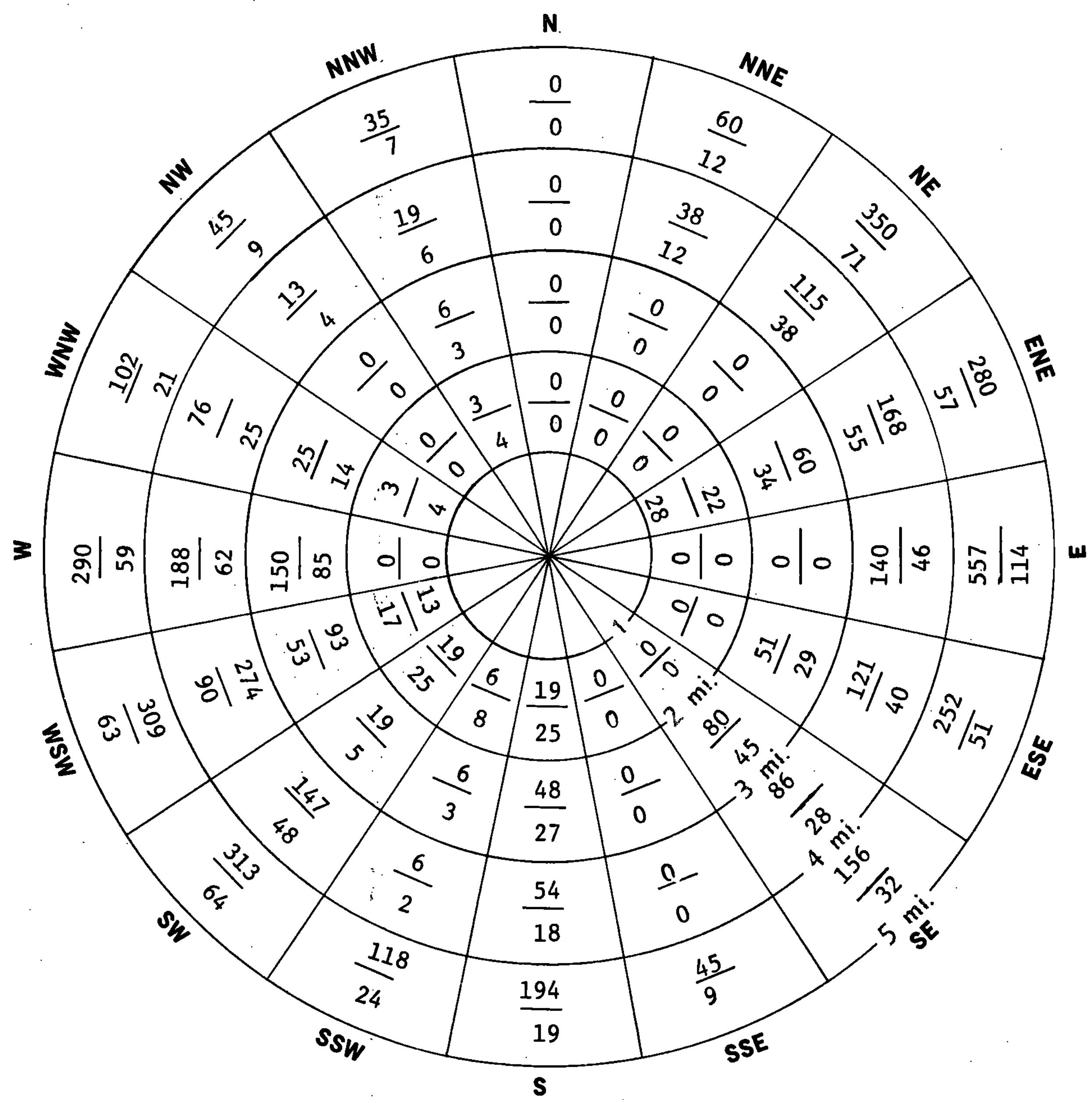


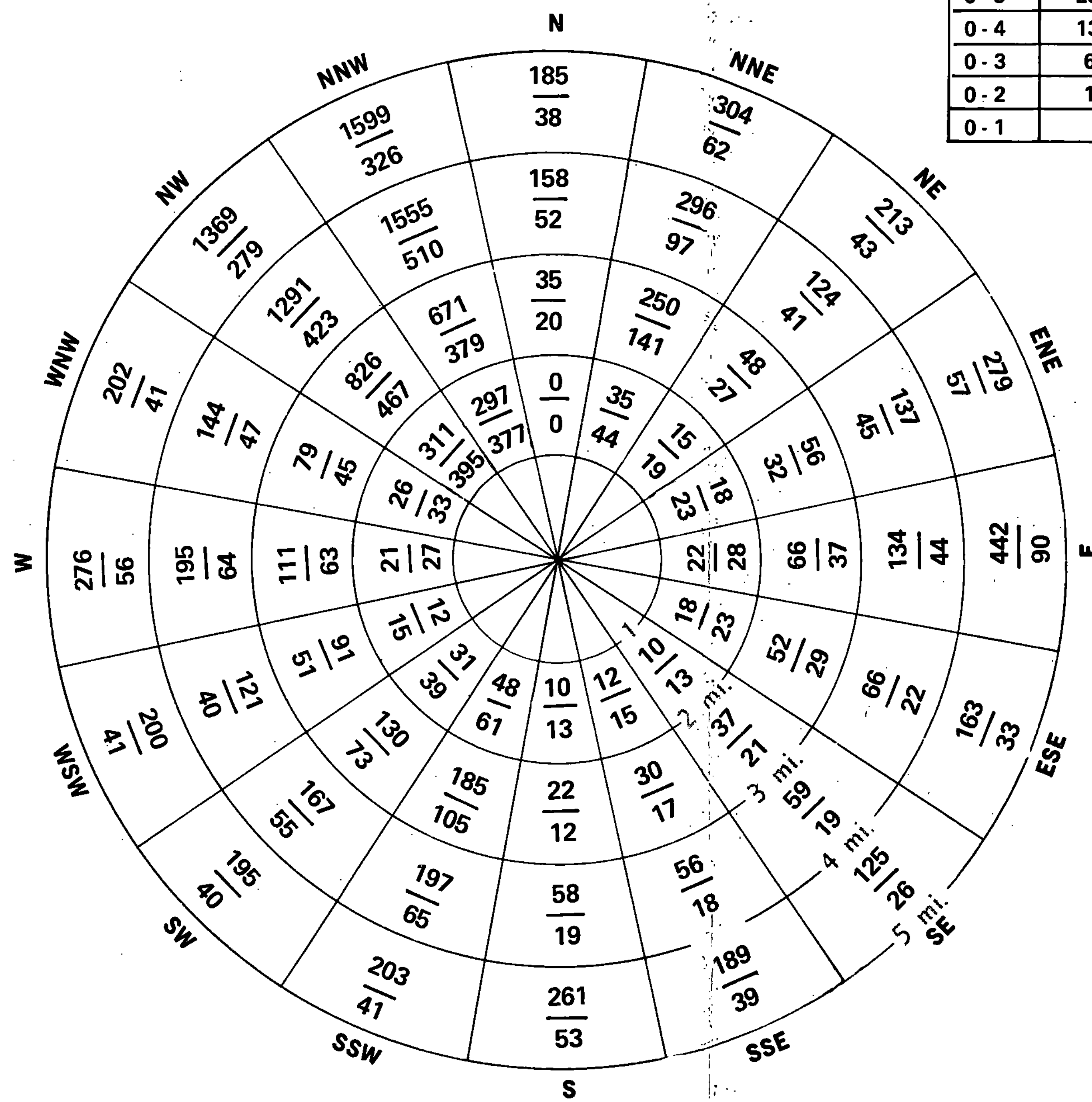
OCONEE NUCLEAR STATION

Figure 2 - 12

Source: U. S. Census 1970

Rev. 2 2/9/70
Rev. 18 3/10/72





2010 (1)
0 - 5 miles

Radius (Miles)	Total - 360° (1) 1966	Total - 360° Rev. 1969 (2)
0 - 5	2966	6205
0 - 4	1372	4758
0 - 3	611	2689
0 - 2	193	886
0 - 1	0	0

TOTAL POPULATION
PERSONS/SQ. MILE

(1) Source: U. S. Census 1910-1960. Extrapolation (for 2010) by Dr. C. Horace Hamilton, Department of Rural Sociology, North Carolina State University, Raleigh, N. C.

(2) November 1969. Projection for the year 2010 revised upward over that submitted in 1966. This is due to plans in progress for lake proximity developments extending outward to the 20 - mile radius. These plans reflect a higher rate of growth than that initially extrapolated for the counties involved. Particularly in the NW & NNW Sectors.

(3) Source: 1970 U.S. Census

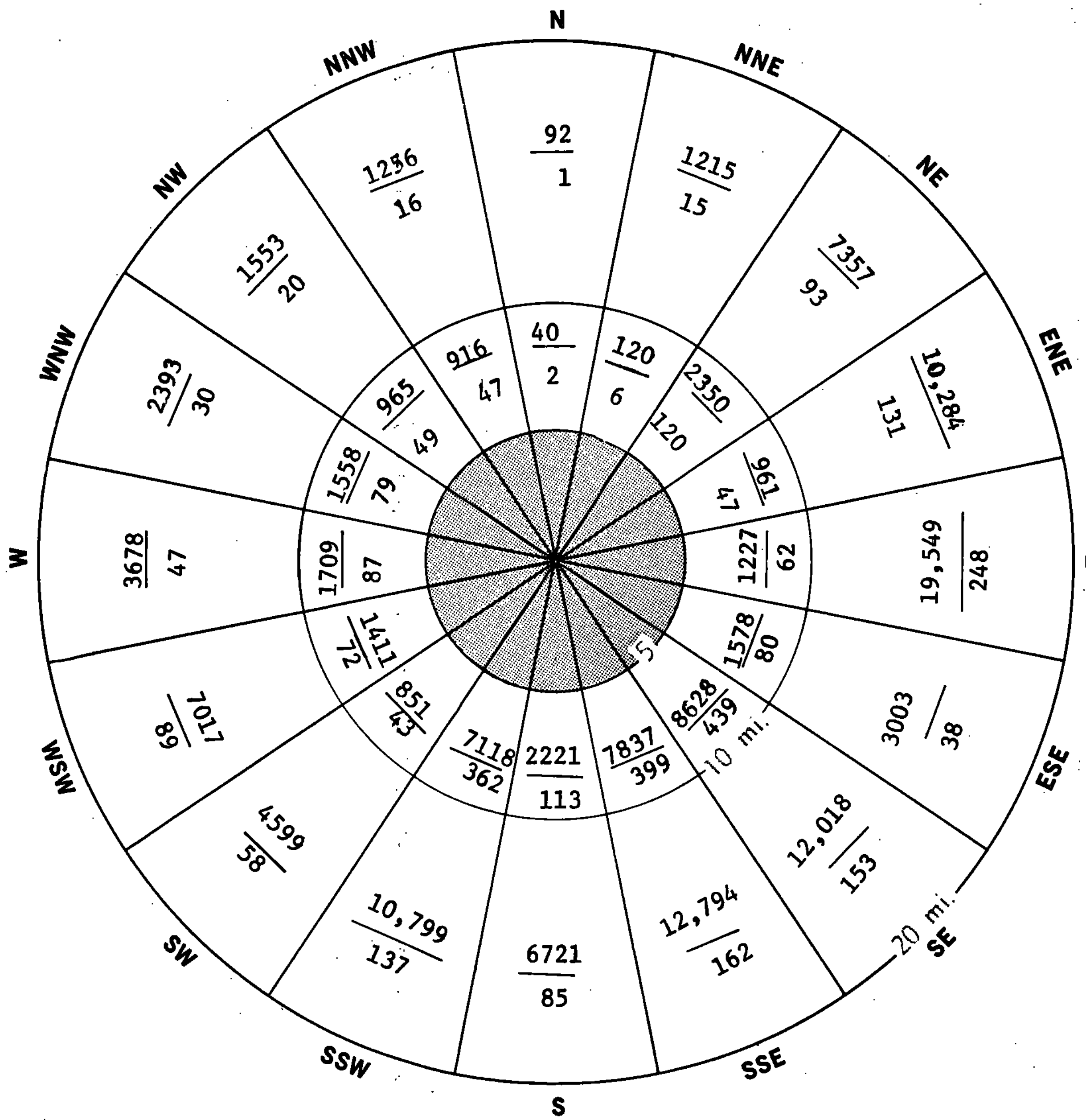
TOTAL POPULATION IS CUMULATIVE FROM THE CENTER.

POPULATION DISTRIBUTION (1970 - 2010)

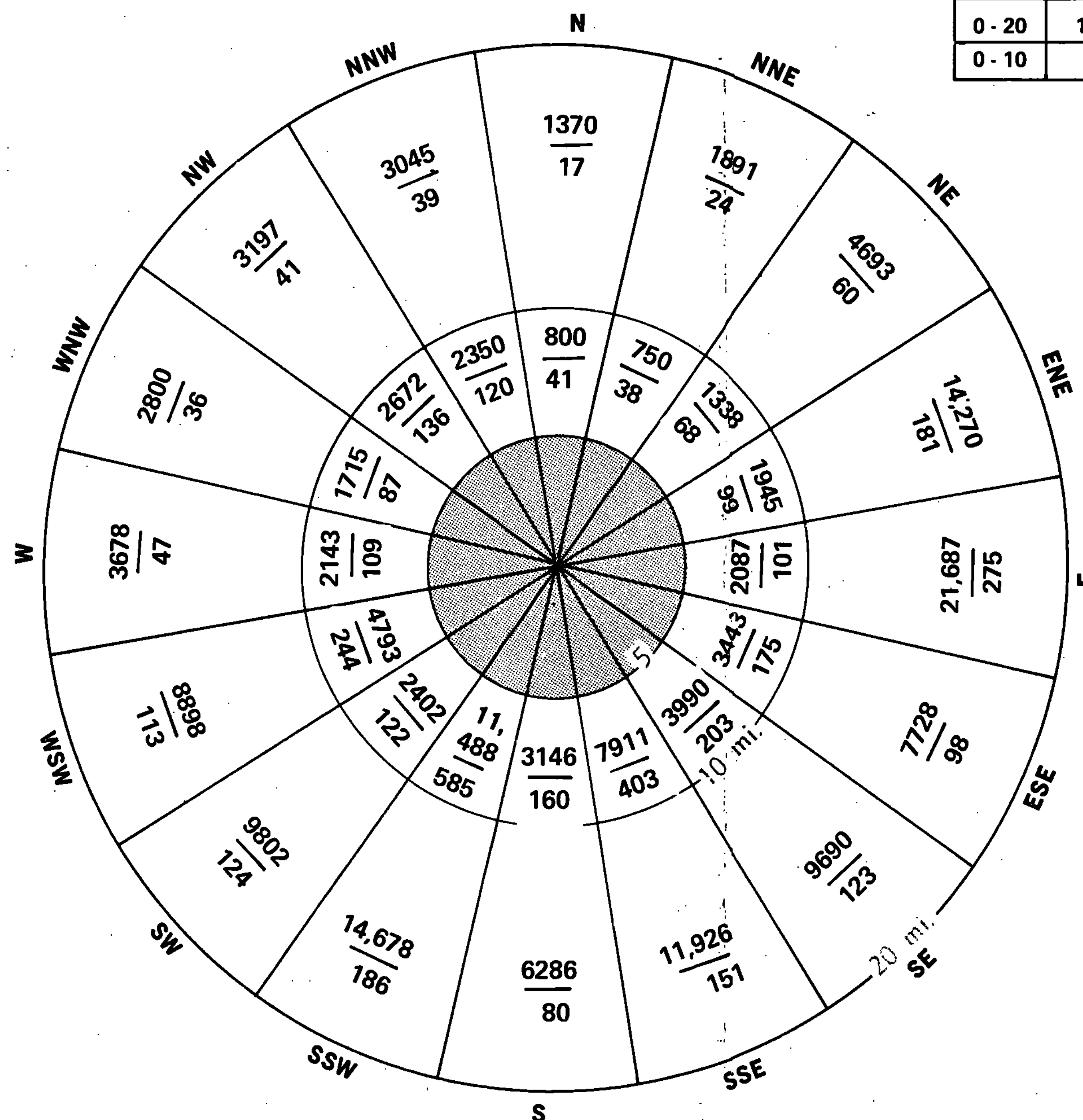
OCONEE NUCLEAR STATION
FIGURE 2-13

Rev. 2 2/9/70
Rev. 20 5/25/72

1970
5 - 20 MILES (3)



2010
5 - 20 MILES (1)



2010 (1)
5 - 20 Miles

Radius (Miles)	Total - 360° 1966 (1)	Total - 360° Rev. 1969 (2)
0 - 20	121,845	125,639
0 - 10	49,326	52,973

TOTAL POPULATION
PERSONS/SQ. MILE

TOTAL POPULATION IS CUMULATIVE FROM THE CENTER.

(1) Source: U. S. Census 1910-1960. Extrapolation (for 2010) by Dr. C. Horace Hamilton, Department of Rural Sociology, North Carolina State University, Raleigh, N. C.

(2) November 1969. Projection for the year 2010 revised upward over that submitted in 1966. This is due to plans in progress for lake proximity developments extending outward to the 20 - mile radius. These plans reflect a higher rate of growth than that initially extrapolated for the counties involved, Particularly in the NW & NNW Sectors.

(3) Source: 1970 U. S. Census

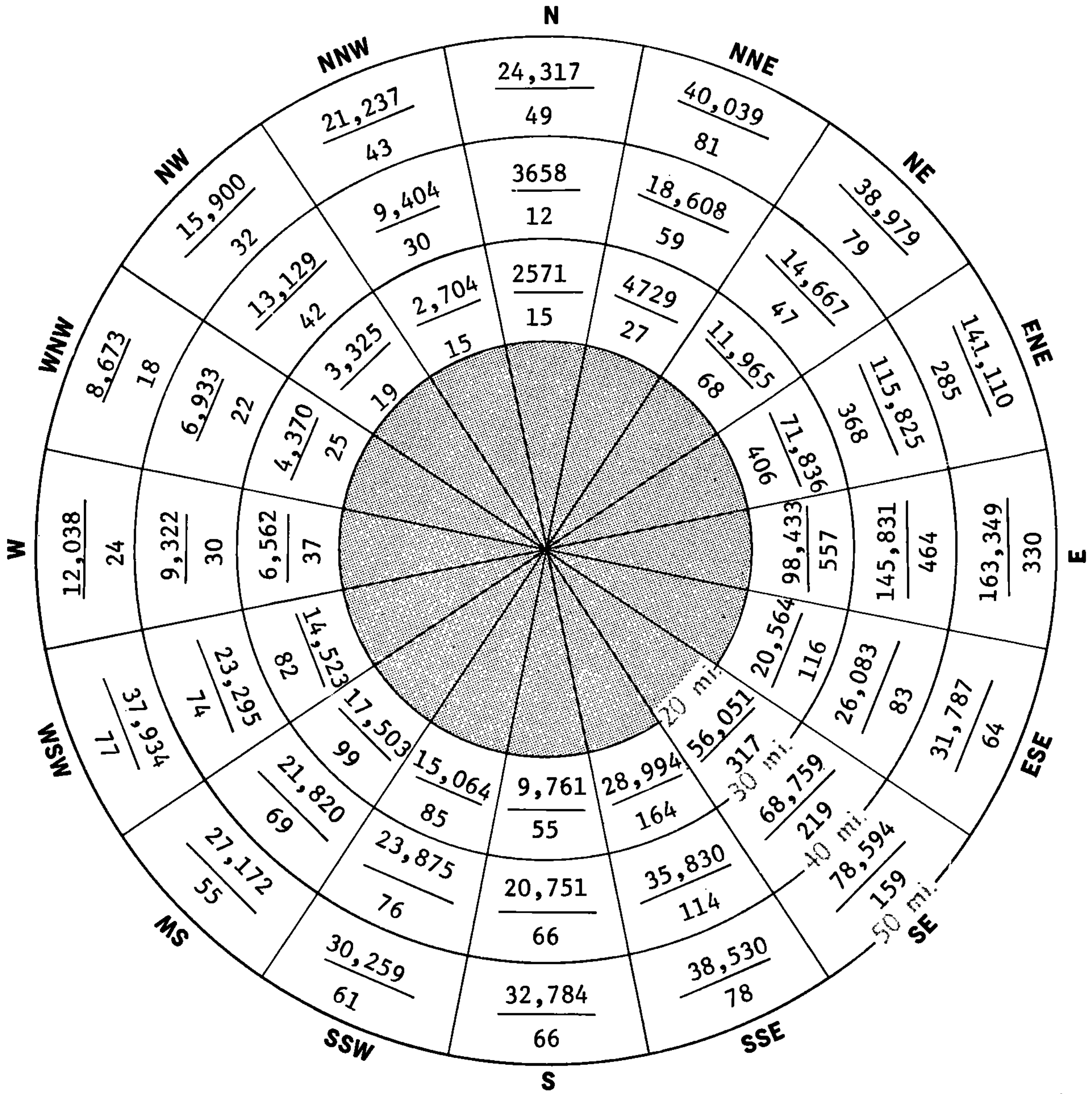
POPULATION DISTRIBUTION (1970 - 2010)

OCONEE NUCLEAR STATION
FIGURE 2-14

Rev. 2 2/9/70
Rev. 20. 5/25/72

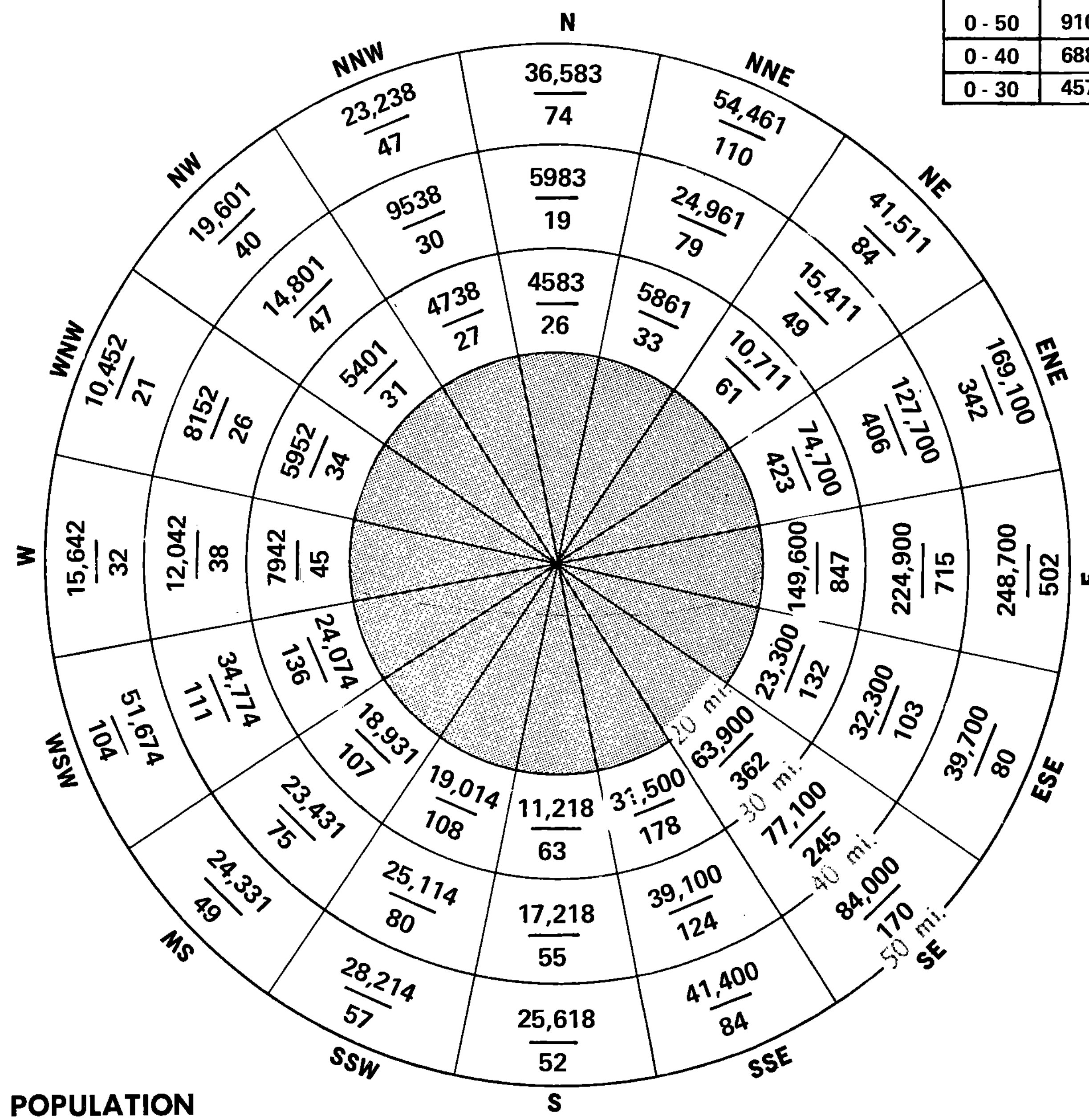
1970

20 - 50 MILES (3)



2010

20 - 50 MILES (1)



2010

20 - 50 Miles

Radius (Miles)	Total - 360° 1966(1)	Total - 360° Rev. 1969 (2)
0 - 50	910,300	914,225
0 - 40	688,600	692,525
0 - 30	457,500	461,425

TOTAL POPULATION

PERSONS/SQ. MILE

TOTAL POPULATION IS CUMULATIVE FROM THE CENTER.

(1) **Source: U. S. Census 1910-1960. Extrapolation (for 2010) by Dr. C. Horace Hamilton, Department of Rural Sociology, North Carolina State University, Raleigh, N. C.**

(2) November 1969. Projection for the year 2010 revised upward over that submitted in 1966. This is due to plans in progress for lake proximity developments extending outward to the 20 - mile radius. These plans reflect a higher rate of growth than that initially extrapolated for the counties involved, Particularly in the NW & NNW Sectors.

(3) Source: 1970 U.S.Census

POPULATION DISTRIBUTION (1970 2010)

OCONEE NUCLEAR STATION

FIGURE 2-15

Rev. 2 2/9/70
Rev. 20 5/25/72

LIST OF EFFECTIVE PAGES
FSAR APPENDIX 2A

Meteorology

LIST OF EFFECTIVE PAGES
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Technical Specifications

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Fig. 2A-2	Rev. 4
Fig. 2A-3	Rev. 4
Fig. 2A-4	Rev. 4
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Fig. 2A-6 (1/15/70)	Rev. 7
Fig. 2A-5 (1/28/70)	Rev. 7
Fig. 2A-6 (1/28/70)	Rev. 7
Fig. 2A-5 (1/31/70)	Rev. 7
Fig. 2A-6 (1/31/70)	Rev. 7
Fig. 2A-5 (2/5/70)	Rev. 7
Fig. 2A-6 (2/5/70)	Rev. 7
Fig. 2A-5 (2/6/70)	Rev. 7

Technical Specifications

Meteorology

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Technical Specifications

APPENDIX 2A

METEOROLOGY

Submitted with FSAR Revision No. 4

April 20, 1970

acknowledged
6-1-70 Jm

APPENDIX 2A

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7.

APPENDIX 2A
METEOROLOGY

2A METEOROLOGY

2A.1 ADDITIONAL METEOROLOGICAL STUDIES IN SUPPORT OF THE 0 TO 2 HOUR VALLEY DRAINAGE MODEL

For the 0 to 2 hour accident relative concentration, X/Q, a value of 7.41×10^{-5} was submitted based on the valley drainage concept. Additional meteorological studies have been performed subsequent to this submittal which give evidence that the valley drainage model is conservative. These studies show a X/Q value of 6.12×10^{-5} as being descriptive of the 0 to 2 hour accident relative concentration; therefore, the relative concentration value of 7.41×10^{-5} will not be changed. The following is a description of additional meteorological studies supporting this conclusion.

2A.1.1 Object

Meteorological studies were undertaken as a refinement to the valley drainage model. Site complexities due to terrain necessarily warranted further analyses.

2A.1.2 Scope

The site dispersion characteristics were investigated with five (5) instruments (see Figure 2A-1) indicating and recording wind direction and speed, two of which were elevated. During these studies, vertical temperature gradients were measured at two (2) locations. Fifteen SF_6 (Sulfur Hexafluoride) gas-tracer experiments were conducted under poor diffusion conditions, during periods with a temperature inversion.

2A.1.3 General Methods

The 0 to 2 hour accident relative concentration was recalculated using the equation $X/Q = 1/\bar{\mu}\pi\sigma_y\sigma_z$. Wind speed was obtained from the microwave tower instrument. Standard deviations of the lateral concentration distribution (Sigma Y) were computed from Pasquill assignments for standard deviations of the horizontal wind azimuth (Sigma Theta). Standard deviation of the horizontal wind was derived from wind range on the microwave tower instrument. Standard deviations of the vertical concentration distribution (Sigma Z) were determined by vertical temperature gradients for the following class intervals.

Pasquill Categories	Vertical Temp. Gradient Class Intervals
F	> 2.0°F in 150 feet
E	2.0 to 0.1°F in 150 ft.
D	0.0 to - 1.4°F in 150 ft.
C	- 1.5 to - 2.9°F in 150 ft.
B	- 3.0 to - 4.5°F in 150 ft.
A	< - 4.5°F in 150 ft.

Pasquill assignments for Sigma Z were again made for categories A, B, and C; however, for D, E, and F gas-tracer test values were substituted. Test Sigma Y values, although larger than Pasquill values, were not used because analysis for given stabilities and wind speeds showed horizontal dispersion too directionally dependent. It is noteworthy that Sigma Y was computed and used without a building effect term. Gas-tracer test results implied that Pasquill Sigma Z values for D, E, and F were too low. A reasonable representation for standard deviation of the vertical concentration distribution was sought for these class intervals, and based on test results, redefined as follows:

Pasquill F = 40 meters, Pasquill E = 50 meters,
Pasquill D = 50 meters.

A relative concentration calculation was made for each pair of valid consecutive observations from the microwave tower wind and temperature data. Relative concentration was computed as the average of the two one hour concentrations, if in successive hours there was an overlap in plume widths defined as 4.30 Sigma Y. Relative concentration was computed from the highest one hour concentration averaged with ten percent of the lowest one hour concentration, if successive hours showed no overlap as above, but did give an overlap of wind range sectors. Finally, relative concentration was computed from the highest one hour concentration averaged with 0, if successive hours showed no overlap of wind range sectors. A relative concentration frequency distribution was determined for the period June 1, 1968 to May 31, 1969. (See Table 2A-1). A hand calculation check on the relative concentration program ascertained its validity.

Wind speed for each hour was read as the average speed in the preceding 30 minute period. Wind speeds less than or equal to 0.9 miles per hour were read as 1.0 miles per hour. Wind range read for each hour also covered the preceding 30 minute period. Vertical temperature differentials read for each hour covered a period of 30 minutes before and after the hour. Further, vertical temperature differentials for each hour were read: (a) as highest value if all readings positive, (b) as highest value if both positive and negative readings occurred the same hour, (c) as 0 if both 0 and negative readings occurred the same hour, and (d) as the lowest value if all readings were negative during the same hour.

Data from the five (5) wind instruments were evaluated simultaneously and classified into five (5) flow patterns. Comparisons were made of flow patterns during gas-tracer test (January 15, 1970 to March 11, 1970) with those during temperature inversions from available data of an earlier period (October 13, 1969 to November 23, 1969). The most frequent test flow pattern was also the most frequent configuration during the earlier period. All five (5) patterns occurred in both periods.

Sample calculation at 1 mile (1609 meters): $X/Q = 1/\bar{\mu}\pi\sigma_y\sigma_z$

Input Parameters: $\bar{\mu}$ = 2.5 meters per second
wind range = 15°
vertical temperature differential = 3°F in 150 feet
$\sigma_\theta = 15/6 = 2.5°$
σ_y = 57 meters
σ_z = 40 meters

$X/Q = 1/(3.1416)(2.5)(57)(40)$
$X/Q = 5.58 \times 10^{-5}$ seconds per meter^3

2A.1.4 Gas-Tracer Test Procedures and Results

A. Calibration of Balance and Gas Chromatograph

The balance used in gas release rates, "Q" was calibrated by certified weights. Calibration of gas chromatograph was performed by the instrument manufacturer prior to testing. The instrument was also checked for calibration drift with a known gas mixture certified by an independent analytical laboratory.

B. SF_6 Field Procedures

1. Personnel required

   a. One (1) test supervisor
   b. One (1) release man
   c. One (1) vehicle driver
   d. Two (2) instrument operators

2. Test equipment

   a. Vehicle with test mount
   b. Utility vehicle
   c. Balance
   d. Gas bottle filled with SF_6
   e. SF_6 detector with recorder
   f. Flow meter
   g. Wooden stakes
   h. Weather instruments as noted
   i. Steel tape measure

3. Limitations and precautions

   Prevent sampling vehicle and its occupants from being contaminated with SF_6.

4. Prerequisite weather conditions

   a. Temperature inversion should exist as determined from the temperature gradient on the microwave tower.
   b. The test should not be run if fog or precipitation is occurring.

5. Data required

   a. Locations and non-zero readings of background points (Figure 2A-2)
   b. Locations, readings, and times of sampling points (Figures 2A-3 & 2A-4)
   c. Release rate (Figure 2A-4)
   d. Weather data (Figure 2A-4)
   e. Plume size data (Figure 2A-4)

6. Test procedure

   a. Check the weather instruments on the microwave tower and record the data on Attachment 5.
   b. Set up release equipment about 200 feet east of auxiliary boiler vent between the turbine and the switchyard.
   c. Take background readings at points marked on Attachment 1. Note any indications of SF_6 on Attachment 1.
   d. Prepare a smoky fire near the SF_6 release point and observe the general plume drift.
   e. When the background check has been made start the release of SF_6. Record data on Attachment 4 every 20 minutes with release rate of four (4) to six (6) grams per minute.
   f. Begin sampling for SF_6 around the site perimeter downwind from the release point. Number each sample point consecutively; however, if two (2) samples are taken at one point, only one (1) number needs to be used.
   g. When SF_6 gas is found, mark the point with a numbered wooden stake. By choosing appropriate sampling points attempt to define the width of any plume found.
   h. Continue sampling around the release point during the test period.
   i. Read the instruments at the utility pole weather station, near the Keowee River, at least once during the test (Attachment 5).
   j. Continue to sample and record data until good plume definition can be established.
   k. After sampling has ended, record the distances between sample stakes.

C. Analysis Procedure

1. Note each pass through a detection area and approximate time of the pass. Place data points marking positions where SF_6 is detected in a sequential space order (not time).

2. From map of area, determine the average distance from the source to the source to the detection stations.

3. Convert the source strength, Q, to micrograms per second from the release rate data.

4. Convert the detector scale readings to micrograms per cubic meter.

5. Estimate the average wind speed from surface instrumentation, and when applicable, microwave tower winds.

6. Utilize computer program to fit a Gaussian curve to the spatially ordered data points.

7. Find the first and second moment arms of the distribution of concentration. From the first moment arm note the center line position; from the second moment arm note the variance of the horizontal dispersion of the concentration.

8. Take positive square root of the variance to get a standard deviation in the horizontal, Sigma Y.

9. Obtain center line concentration by $X = A\sigma_y^{-1}(2\pi)^{-1/2}$ where A is the area under the distribution curve.

10. Calculate the standard deviation in the vertical, Sigma Z, by $\sigma_z = Q(\pi\bar{\mu}\sigma_y X)^{-1}$ which is applicable for a ground release.

11. Determine the stability category by the temperature differential on the microwave tower.

12. Using graphs of Sigma Y and Sigma Z as functions of stability and distance from a source, locate test values.

13. Following the curvature of the Pasquill curves for the stability found in Number 11 above, read Sigma Y and Sigma Z values for one (1) mile from the graph.

14. Compute the center line values X/Q at one (1) mile by $X/Q = [\pi\bar{\mu}\sigma_y \text{ (1 mi.) } \sigma_z \text{ (1 mi.)}]^{-1}$.

D. Gas-Tracer Experiment Results (See Table 2A-2)

2A.1.5 Other Considerations

A 1.4 wind speed correction factor for the period June 1968 to September 1969 may be warranted based on a calibration check made October 1, 1969, and comparative wind speed data at Greenville-Spartanburg and Oconee. A relative concentration frequency distribution was determined with a 1.4 wind speed correction factor for the period June 1, 1968 to May 31, 1969, (see Table 2A-3). No wind speed correction was factored into the 0 to 2 hour accident relative concentration value of 6.12×10^{-5}.

Table 2A-4 displays comparative wind speed data for Greenville-Spartanburg and Oconee from June, 1968 to January, 1970. Comparisons were made at 13:00 EST for wind speeds equal to or greater than eight (8) knots at Greenville-Spartanburg.

2A.1.6 Consultants

These analyses, tests, and the resulting X/Q values have been performed by Duke Power with guidance from and review by Messrs. Harry L. Hamilton, Jr. & Walter D. Bach, Jr. of the Research Triangle Institute, Research Triangle Park, North Carolina. Comments of these consultants are included as page 2A-6.

2A.1.7 Supplemental Data

Supplemental data includes an all occurrence annual wind rose, a Pasquill F
9. 7. annual wind rose, a Pasquill E annual wind rose, a Pasquill A, B, C, and D annual
wind rose, a relative concentration frequency distribution based on single hour
calculations, and SF_6 field data. This material is presented in Table 2A-5 and
Figures 2A-5 and 2A-6.

RESEARCH TRIANGLE INSTITUTE
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ENGINEERING AND ENVIRONMENTAL SCIENCES DIVISION

April 10, 1970

Mr. W. S. Lee
Vice President for Engineering
Duke Power Company
P. O. Box 2178
Charlotte, North Carolina 28201

Dear Mr. Lee:

Messrs. Harry L. Hamilton, Jr. and Walter D. Bach, Jr. of the Research Triangle Institute (RTI) have assisted Messrs. Charles A. Dewey and Stephen T. Apple of Duke Power Company in studying the meteorological factors affecting the transport and dispersions of airborne material at the Oconee, South Carolina, nuclear generating facility of Duke Power Company.

Examination of the topography of the Oconee Site and consideration of the meteorological data collected prior to RTI association with the study made obvious the fact that conventional evaluation of dispersion parameters from temperature lapse rate and wind velocity variations measurements would not suffice. Accordingly, dispersion experiments using a tracer (SF_6) released from a ground-level point source near the reactor buildings were planned to provide measurements of contaminant transport and dilution. RTI personnel assisted in planning these experiments and evaluated the results in terms of characteristic cloud dimensions [$\sigma_y\{x\}$, $\sigma_y\{x\}$, and $\frac{\chi}{Q}\{x\}$]. From these characteristic dimensions, estimates of dilution rates under worst dispersion conditions were made by the Duke Power Company environmental engineers. The RTI meteorologists concur with these estimates.

We believe that relative concentrations estimated at the exclusion area boundary using these dispersion rates are conservative--that is, relative concentrations in all probability would be lower. On the basis of available data and professional judgement, the values quoted in the ammended FSAR are reasonable for use in an evaluation of potential hazard.

Very truly yours,

Harry L. Hamilton, Jr.

Harry L. Hamilton, Jr.
Environmental Sciences Department

HLH:bah

TABLE 2A-1
RELATIVE CONCENTRATION, X/Q, FREQUENCY DISTRIBUTION
WITHOUT WIND SPEED CORRECTION

Relative Concentration		Frequency (No. of Obs.)	Percentage	Cumulative Per Cent
>=4.0	X10-4	0	0.00	0.00
3.0-3.99	X10-4	0	0.00	0.00
2.0-2.99		8	0.09	0.09
1.0-1.99		35	0.41	0.51
9.0-9.99	X10-5	20	0.24	0.74
8.0-8.99		53	0.62	1.37
7.0-7.99		106	1.25	2.62
6.0-6.99		229	2.70	5.32
5.0-5.99		506	5.97	11.28
4.0-4.99		838	9.88	21.16
3.0-3.99		1484	17.50	38.66
2.0-2.99		2313	27.27	65.93
1.0-1.99		2307	27.20	93.13
9.0-9.99	X10-6	167	1.97	95.10
8.0-8.99		134	1.58	96.68
7.0-7.99		87	1.03	97.70
6.0-6.99		88	1.04	98.74
5.0-5.99		53	0.62	99.36
4.0-4.99		27	0.32	99.68
<=3.99	X10-6	27	0.32	100.00
TOTALS		8482	100.00	------

Percentage of Valid Observations: 96.82

Average Relative Concentration = 2.92960 X 10^{-5}

TABLE 2A-2
GAS-TRACER EXPERIMENT RESULTS

Test Date	Test Number	Time (hours)	Release Rate (micrograms per second)	$\bar{u}$ (meters per second)	Stability Catagory	Source to Receptor Distance (meters)	At Receptor: Center Line Concentration (micrograms per meter^3)	At Receptor: Sigma Y (meters)	At Receptor: Sigma Z (meters)	At One Mile: Sigma Y (meters)	At One Mile: Sigma Z (meters)	At One Mile: PI, Sigma Y, Sigma Z, $\bar{u}$ (meters^3 per second)	At One Mile: Relative Concentration (seconds per meter^3)
Jan. 15, 1970	1a	2100	90×10^3	5.36	F	176	9.40	38.6	14.70	270	74	3.36×10^5	2.97×10^{-6}
	1b	2200	90×10^3	5.36	F	680	3.59	145	10.24	225	19	7.19×10^4	1.38×10^{-5}
Jan. 28, 1970	Plume Measurements Indeterminable												
Jan. 31, 1970	Plume Measurements Indeterminable												
Feb. 5, 1970	2	2100	91×10^3	0.89	F	630	1.58	104	197	280	380	2.97×10^5	3.36×10^{-6}
Feb. 6, 1970	3	2040	85.8×10^3	0.89	F	835	3.49	28	313	52	490	7.12×10^4	1.40×10^{-5}
Feb. 10, 1970	4a	2156	83.3×10^3	1.34	E	190	9.02	38	57	260	280	3.06×10^5	3.26×10^{-6}
	4b	2210	91.6×10^3	1.79	E	352	9.30	26	67	108	186	1.12×10^5	8.85×10^{-6}
	4c	2250	86.7×10^3	1.79	E	411	3.68	62	67	210	172	2.03×10^5	4.92×10^{-6}
Feb. 11, 1970	Plume Measurements Indeterminable												
Feb. 12, 1970	5a	2055	85.5×10^3	1.56	E	530	7.54	55	44	152	100	7.44×10^4	1.34×10^{-5}
	5b	2115	88.5×10^3	1.34	E	530	3.79	77	72	205	168	1.44×10^5	6.89×10^{-6}
Feb. 17, 1970	6a	2210	89.7×10^3	3.13	E	399	8.29	74	14.9	260	38	9.71×10^4	1.02×10^{-5}
	6b	2250	88.0×10^3	1.79	E	426	16.6	34	27.5	115	82	5.30×10^4	1.88×10^{-5} *
Feb. 19, 1970	Plume Measurements Indeterminable												
Mar. 2, 1970	7	2240	89.2×10^3	0.89	F	451	1.54	45	461	138	920	3.55×10^5	2.81×10^{-6}
Mar. 3, 1970	8a	2018	85.0×10^3	0.89	E-F	450	3.63	43	193	138	500	1.92×10^5	5.18×10^{-6}
	8b	2110	83.3×10^3	0.89	E-F	450	6.21	38	126	125	300	1.04×10^5	9.53×10^{-6}
	8c	2200	86.6×10^3	0.89	E-F	450	2.18	73	195	220	500	3.07×10^5	3.25×10^{-6}
Mar. 5, 1970	Plume Measurements Indeterminable												
Mar. 10, 1970	9a	2045	91.4×10^3	0.67	E-F	120	9.38	32	145	300	1050	6.63×10^5	1.50×10^{-6}
	9b	2205	91.4×10^3	0.67	E-F	120	6.44	52	129	500	910	9.57×10^5	1.04×10^{-6}
	9c	2315	91.4×10^3	0.67	E-F	120	2.28	43	467	450	3500	3.31×10^6	5.01×10^{-7}
Mar. 11, 1970	Plume Measurements Indeterminable												

*Highest test relative concentration at one mile = 1.88×10^{-5} seconds per meter^3

TABLE 2A-3
RELATIVE CONCENTRATION, X/Q, FREQUENCY DISTRIBUTION
WITH WIND SPEED CORRECTION

Relative Concentration		Frequency (No. of Obs.)	Percentage	Cumulative Per Cent
>=4.0	X10-4	0	0.00	0.00
3.0-3.99	X10-4	0	0.00	0.00
2.0-2.99	"	0	0.00	0.00
1.0-1.99	"	18	0.21	0.21
9.0-9.99	X10-5	6	0.07	0.28
8.0-8.99	"	6	0.07	0.35
7.0-7.99	"	15	0.18	0.53
6.0-6.99	"	40	0.47	1.00
5.0-5.99	"	137	1.62	2.62
4.0-4.99	"	391	4.61	7.23
3.0-3.99	"	957	11.28	18.51
2.0-2.99	"	2085	24.58	43.09
1.0-1.99	"	3407	40.17	83.26
9.0-9.99	X10-6	313	3.69	86.95
8.0-8.99	"	298	3.51	90.46
7.0-7.99	"	260	3.07	93.53
6.0-6.99	"	218	2.57	96.10
5.0-5.99	"	136	1.60	97.70
4.0-4.99	"	113	1.33	99.03
<=3.99	X10-6	82	0.97	100.00
TOTALS		8482	100.00	------

Percentage of Valid Observations: 96.82

Average Relative Concentration = 2.09257 X 10^{-5}

TABLE 2A-4
COMPARATIVE WIND SPEED DATA

DATE	GREENVILLE-SPARTANBURG* (AVERAGE)	OCONEE (AVERAGE)	OCONEE TO GREENVILLE-SPARTANBURG (RATIO)	OCONEE TO GREENVILLE-SPARTANBURG (RATIO X 1.4)
June, 1968	13.9 mph	7.6 mph	0.54	0.76
July, 1968	11.2 "	6.3 "	0.56	0.79
August, 1968	11.3 "	6.8 "	0.60	0.84
September, 1968	10.9 "	5.6 "	0.52	0.72
October, 1968	12.3 "	8.1 "	0.65	0.92
November, 1968	13.1 "	7.4 "	0.56	0.78
December, 1968	15.6 "	9.3 "	0.59	0.83
January, 1969	14.6 "	8.1 "	0.55	0.77
February, 1969	15.4 "	11.0 "	0.72	1.02
March, 1969	11.8 "	7.7 "	0.66	0.94
April, 1969	11.6 "	7.8 "	0.68	0.96
May, 1969	11.9 "	6.8 "	0.57	0.81
June, 1969	11.6 "	6.5 "	0.56	0.80
July, 1969	11.1 "	5.5 "	0.50	0.70
August, 1969	11.0 "	8.2 "	0.74	1.06
September, 1969	11.3 "	7.3 "	0.65	0.91
** October, 1969	12.1 "	11.2 "	0.92	----
November, 1969	12.5 "	12.3 "	0.97	----
December, 1969	12.6 "	10.5 "	0.83	----
January, 1970	13.0 "	14.1 "	1.08	----

* Greenville-Spartanburg, S.C. Airport ESSA Station

** Calibration Check - October 1, 1969

TABLE 2A-5 SUPPLEMENTAL DATA
OCONEE METEOROLOGICAL SURVEY (TOWER DATA)
FOR PERIOD OF JUNE 1, 1968 THRU MAY 31, 1969

FREQUENCY OF TOTAL RELATIVE CONCENTRATION FOR ALL OBSERVATIONS

Relative Concentration	Frequency No. of Obs.	Percentage	Cumulative Per Cent
$\geq 4.0 \times 10^{-4}$	20	0.24	0.24
$3.0 - 3.99 \times 10^{-4}$	4	0.05	0.28
2.0 - 2.99	1	0.01	0.29
1.0 - 1.99	52	0.61	0.91
$9.0 - 9.99 \times 10^{-5}$	20	0.24	1.14
8.0 - 8.99	71	0.84	1.98
7.0 - 7.99	86	1.01	2.99
6.0 - 6.99	194	2.28	5.27
5.0 - 5.99	407	4.79	10.06
4.0 - 4.99	783	9.22	19.28
3.0 - 3.99	1288	15.16	34.44
2.0 - 2.99	1961	23.08	57.52
1.0 - 1.99	2604	30.65	88.17
$9.0 - 9.99 \times 10^{-6}$	256	3.01	91.18
8.0 - 8.99	205	2.41	93.60
7.0 - 7.99	214	2.52	96.12
6.0 - 6.99	129	1.52	97.63
5.0 - 5.99	78	0.92	98.55
4.0 - 4.99	78	0.92	99.47
$\leq 3.99 \times 10^{-6}$	45	0.53	100.00
TOTALS	8496	100.00	------

Percentage of Valid Observat ions - 96.98

Average Relative Concentration 3.11000×10^{-5}

TABLE 2A-5 SUPPLEMENTAL DATA

OCONEE METEOROLOGICAL SURVEY (TOWER DATA) FOR PERIOD OF JUNE 19, 1968 THRU JUNE 19, 1969

SUMMARY OF WIND OCCURRENCES BY SECTOR & SPEED CLASS (NO. OCCUR, PERCENT, STANDARD DEVIATION)

Wind Sector	Item	Sector Total	1.0-3.2 .45-1.49	3.3-5.5 1.5-2.49	5.6-7.8 2.5-3.49	7.9-10.0 3.5-4.49	10.1-12.3 4.5-5.49	12.4-14.5 5.5-6.49	14.6-16.7 6.5-7.49	16.8-19.0 7.5-8.49	19.1-21.2 8.5-9.49	>21.2 MPH >=9.5 M/S
360.0	No	1472	465	698	247	40	16	3	3	0	0	0
-N-	Pct	17.24	5.45	8.18	2.89	0.47	0.19	0.04	0.04	0.00	0.00	0.00
	Sd		16.0	7.5	6.1	9.1	11.8	12.5	6.4	0.0	0.0	0.0
22.5	No.	708	261	312	94	20	9	4	4	3	1	0
-NNE-	Pct	8.29	3.06	3.65	1.10	0.23	0.11	0.05	0.05	0.04	0.01	0.00
	Sd		16.5	9.5	7.0	11.5	5.0	5.6	5.4	11.7	5.0	0.0
45.0	No	842	224	281	185	85	35	15	14	3	0	0
-NE-	Pct	9.86	2.62	3.29	2.17	1.00	0.41	0.18	0.16	0.04	0.00	0.00
	Sd		17.9	10.1	6.8	6.1	6.4	5.6	6.2	6.4	0.0	0.0
67.5	No.	493	134	143	96	83	24	10	3	0	0	0
-ENE-	Pct	5.77	1.57	1.68	1.12	0.97	0.28	0.12	0.04	0.00	0.00	0.00
	Sd		18.7	10.7	8.0	7.8	6.6	6.3	4.7	0.0	0.0	0.0
90.0	No	508	177	195	74	47	8	6	1	0	0	0
-E-	Pct	5.95	2.07	2.28	0.87	0.55	0.09	0.07	0.01	0.00	0.00	0.00
	Sd		17.3	11.6	9.0	8.0	6.0	7.5	9.2	0.0	0.0	0.0
112.5	No	318	131	141	33	8	4	1	0	0	0	0
-ESE-	Pct	3.72	1.53	1.65	0.39	0.09	0.05	0.01	0.00	0.00	0.00	0.00
	Sd		16.6	11.0	11.7	9.7	11.0	10.8	0.0	0.0	0.0	0.0
135.0	No	307	87	154	47	18	1	0	0	0	0	0
-SE-	Pct	3.60	1.02	1.80	0.55	0.21	0.01	0.00	0.00	0.00	0.00	0.00
	Sd		14.5	11.6	11.4	8.1	25.0	0.0	0.0	0.0	0.0	0.0
157.5	No	161	52	74	27	6	2	0	0	0	0	0
-SSE-	Pct	1.89	0.61	0.87	0.32	0.07	0.02	0.00	0.00	0.00	0.00	0.00
	Sd		13.9	9.1	7.7	8.9	15.4	0.0	0.0	0.0	0.0	0.0
180.0	No	173	46	100	15	7	5	0	0	0	0	0
-S-	Pct	2.03	0.54	1.17	0.18	0.08	0.06	0.00	0.00	0.00	0.00	0.00
	Sd		11.7	6.7	4.4	2.6	2.5	0.0	0.0	0.0	0.0	0.0
202.5	No	304	49	110	59	55	20	10	1	0	0	0
-SSW-	Pct	3.56	0.57	1.29	0.69	0.64	0.23	0.12	0.01	0.00	0.00	0.00
	Sd		14.3	8.4	8.5	5.9	4.9	6.4	4.2	0.0	0.0	0.0
225.0	No	631	129	218	126	89	41	27	1	0	0	0
-SW-	Pct	7.39	1.51	2.55	1.48	1.04	0.48	0.32	0.01	0.00	0.00	0.00
	Sd		15.5	10.6	7.2	6.5	5.9	5.8	7.5	0.0	0.0	0.0
247.5	No	434	106	112	98	36	34	27	13	3	2	3
-WSW	Pct	5.08	1.24	1.31	1.15	0.42	0.40	0.32	0.15	0.04	0.02	0.04
	Sd		17.1	11.2	9.0	5.5	5.0	4.9	4.4	4.4	4.6	3.9
270.0	No	524	131	125	91	52	50	39	21	12	0	3
-W-	Pct	6.14	1.53	1.46	1.07	0.61	0.59	0.46	0.25	0.14	0.00	0.04
	Sd		18.5	12.4	9.0	6.5	5.4	4.0	4.6	4.6	0.0	4.2
292.5	No	364	117	114	46	39	25	9	7	5	1	1
-WNW-	Pct	4.26	1.37	1.34	0.54	0.46	0.29	0.11	0.08	0.06	0.01	0.01
	Sd		17.5	11.2	9.2	7.2	7.7	8.6	7.4	5.8	4.2	6.7
315.0	No	515	204	199	55	33	17	3	3	0	1	0
-NW-	Pct	6.03	2.39	2.33	0.64	0.39	0.20	0.04	0.04	0.00	0.01	0.00
	Sd		15.5	9.5	7.6	9.3	5.2	6.9	6.4	0.0	7.5	0.0
337.5	No	684	268	303	92	14	4	3	0	0	0	0
-NNW-	Pct	8.01	3.14	3.55	1.08	0.16	0.05	0.04	0.00	0.00	0.00	0.00
	Sd		15.4	7.0	6.2	8.7	11.3	13.3	0.0	0.0	0.0	0.0
Calm	No	99										
	Pct	1.16	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----
Total	No	8537	2581	3279	1385	632	295	157	71	26	5	7
	Pct	100.0	30.23	38.41	16.22	7.40	3.46	1.84	0.83	0.30	0.06	0.08

Total Valid Observations: 8537

OCONEE METEOROLOGICAL SURVEY (TOWER DATA) TABLE 2A-5 SUPPLEMENTAL DATA FOR PERIOD JUNE 19, 1968 THRU JUNE 19, 1969

SUMMARY OF PASQUILL F WIND OCCURRENCES BY SECTOR & SPEED CLASS (NO. OCCUR, PERCENT, STANDARD DEVIATION)

Wind Sector	Item	Sector Total	1.0-3.2 .45-1.49	3.3-5.5 1.5-2.49	5.6-7.8 2.5-3.49	7.9-10.0 3.5-4.49	10.1-12.3 4.5-5.49	12.4-14.5 5.5-6.49	14.6-16.7 6.5-7.49	16.8-19.0 7.5-8.49	19.1-21.2 8.5-9.49	>21.2 MPH >=9.5 M/S
-N-	No.	499	131	260	95	12	1	0	0	0	0	0
	Pct	5.76%	1.51%	3.00%	1.10%	0.14%	0.01%	0.00%	0.00%	0.00%	0.00%	0.00%
-NNE-	No.	166	68	66	29	3	0	0	0	0	0	0
	Pct	1.92%	0.79%	0.76%	0.33%	0.03%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
-NE-	No.	135	61	57	13	3	0	0	1	0	0	0
	Pct	1.56%	0.70%	0.66%	0.15%	0.03%	0.00%	0.00%	0.01%	0.00%	0.00%	0.00%
-ENE-	No.	57	36	20	0	1	0	0	0	0	0	0
	Pct	0.66%	0.42%	0.23%	0.00%	0.01%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
-E-	No.	116	55	55	4	1	1	0	0	0	0	0
	Pct	1.34%	0.63%	0.64%	0.05%	0.01%	0.01%	0.00%	0.00%	0.00%	0.00%	0.00%
-ESE-	No.	65	30	32	3	0	0	0	0	0	0	0
	Pct	0.75%	0.35%	0.37%	0.03%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
-SE-	No.	41	18	19	2	2	0	0	0	0	0	0
	Pct	0.47%	0.21%	0.22%	0.02%	0.02%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
-SSE-	No.	23	10	11	2	0	0	0	0	0	0	0
	Pct	0.27%	0.12%	0.13%	0.02%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
-S-	No.	19	6	10	2	1	0	0	0	0	0	0
	Pct	0.18%	0.07%	0.12%	0.02%	0.01%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
-SSW-	No.	39	16	18	4	0	0	1	0	0	0	0
	Pct	0.45%	0.18%	0.21%	0.05%	0.00%	0.00%	0.01	0.00	0.00%	0.00%	0.00%
-SW-	No.	95	29	40	15	10	1	0	0	0	0	0
	Pct	1.10%	0.33%	0.46%	0.17%	0.12%	0.01%	0.00%	0.00%	0.00%	0.00%	0.00%
-WSW-	No.	75	31	23	17	3	0	1	0	0	0	0
	Pct	0.87%	0.36%	0.27%	0.20%	0.03%	0.00%	0.01%	0.00%	0.00%	0.00%	0.00%
-W-	No.	102	43	28	23	5	2	0	0	1	0	0
	Pct	1.18%	0.50%	0.32%	0.27%	0.06%	0.02%	0.00%	0.00%	0.01%	0.00%	0.00%
-WNW-	No.	101	40	42	10	8	1	0	0	0	0	0
	Pct	1.17%	0.46%	0.48%	0.12%	0.09%	0.01%	0.00%	0.00%	0.00%	0.00%	0.00%
-NW-	No.	222	87	105	21	9	0	0	0	0	0	0
	Pct	2.56%	1.00%	1.21%	0.24%	0.10%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
-NNW-	No.	352	110	188	52	2	0	0	0	0	0	0
	Pct	4.06%	1.27%	2.17%	0.60%	0.02%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
Calm	No.	27	---	---	---	---	---	---	---	---	---	---
	Pct	0.31%	---	---	---	---	---	---	---	---	---	---
Total	No.	2134	771	974	292	60	6	2	1	1	0	0
	Pct.	24.64%	8.90%	11.25%	3.37%	0.69%	0.07%	0.02%	0.01%	0.01%	0.00%	0.00%

Total Valid Observations: 8661

OCONEE METEOROLOGICAL SURVEY (TOWER DATA) TABLE 2A-5 SUPPLEMENTAL DATA FOR PERIOD JUNE 19, 1968 THRU JUNE 19, 1969

SUMMARY OF PASQUILL E WIND OCCURRENCES BY SECTOR & SPEED CLASS (NO. OCCUR, PERCENT, STANDARD DEVIATION)

Wind Sector	Item	Sector Total	1.0-3.2 .45-1.49	3.3-5.5 1.5-2.49	5.6-7.8 2.5-3.49	7.9-10.0 3.5-4.49	10.1-12.3 4.5-5.49	12.4-14.5 5.5-6.49	14.6-16.7 6.5-7.49	16.8-19.0 7.5-8.49	19.1-21.2 8.5-9.49	>21.2 MPH >=9.5 M/S
-N-	No.	458	118	247	77	12	4	0	0	0	0	0
	Pct	5.29%	1.36%	2.85%	0.89%	0.14%	0.05%	0.00%	0.00%	0.00%	0.00%	0.00%
-NNE-	No.	166	52	85	23	3	2	1	0	0	0	0
	Pct	1.92%	0.68%	0.98%	0.27%	0.03%	0.02%	0.01%	0.00%	0.00%	0.00%	0.00%
-NE-	No.	138	40	61	26	10	1	0	0	0	0	0
	Pct	1.59%	0.46%	0.70%	0.30%	0.12%	0.01%	0.00%	0.00%	0.00%	0.00%	0.00%
-ENE-	No.	55	18	23	9	4	1	0	0	0	0	0
	Pct	0.64%	0.21%	0.27%	0.10%	0.05%	0.01%	0.00%	0.00%	0.00%	0.00%	0.00%
-E-	No.	56	25	23	4	4	0	0	0	0	0	0
	Pct	0.65%	0.29%	0.27%	0.05%	0.05%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
-ESE-	No.	42	18	20	1	2	0	1	0	0	0	0
	Pct	0.49%	0.21%	0.23%	0.01%	0.02%	0.00%	0.01%	0.00%	0.00%	0.00%	0.00%
-SE-	No.	41	4	29	5	3	0	0	0	0	0	0
	Pct	0.47%	0.05%	0.34%	0.06%	0.03%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
-SSE-	No.	33	10	13	9	0	1	0	0	0	0	0
	Pct	0.38%	0.12%	0.15%	0.10%	0.00%	0.01%	0.00%	0.00%	0.00%	0.00%	0.00%
-S-	No.	32	9	14	3	2	4	0	0	0	0	0
	Pct	0.37%	0.10%	0.16%	0.03%	0.02%	0.05%	0.00%	0.00%	0.00%	0.00%	0.00%
-SSW-	No.	51	6	20	7	13	4	1	0	0	0	0
	Pct	0.59%	0.07%	0.23%	0.08%	0.15%	0.05%	0.01%	0.00%	0.00%	0.00%	0.00%
-SW-	No.	130	22	46	34	22	6	0	0	0	0	0
	Pct	1.50%	0.25%	0.53%	0.39%	0.25%	0.07%	0.00%	0.00%	0.00%	0.00%	0.00%
-WSW-	No.	103	18	27	28	16	11	3	0	0	0	0
	Pct	1.19%	0.21%	0.31%	0.32%	0.18%	0.13%	0.03%	0.00%	0.00%	0.00%	0.00%
-W-	No.	136	25	27	30	22	17	10	4	1	0	0
	Pct	1.57%	0.29%	0.31%	0.35%	0.25%	0.20%	0.12%	0.05%	0.01%	0.00%	0.00%
-WNW-	No.	82	24	28	10	14	4	1	1	0	0	0
	Pct	0.95%	0.28%	0.32%	0.12%	0.16%	0.05%	0.01%	0.01%	0.00%	0.00%	0.00%
-NW-	No.	89	36	31	8	6	8	0	0	0	0	0
	Pct	1.03%	0.42%	0.36%	0.09%	0.07%	0.09%	0.00%	0.00%	0.00%	0.00%	0.00%
-NNW-	No.	127	54	54	15	3	1	0	0	0	0	0
	Pct	1.47%	0.62%	0.62%	0.17%	0.03%	0.01%	0.00%	0.00%	0.00%	0.00%	0.00%
Calm	No.	14	---	---	---	---	---	---	---	---	---	---
	Pct	0.16%	---	---	---	---	---	---	---	---	---	---
Total	No.	1753	479	748	289	136	64	17	5	1	0	0
	Pct	20.25%	5.53%	8.64%	3.34%	1.57%	0.74%	0.20%	0.06%	0.01%	0.00%	0.00%

Total Valid Observations: 8656

OCONEE METEOROLOGICAL SURVEY (TOWER DATA) TABLE 2A-5 SUPPLEMENTAL DATA FOR PERIOD JUNE 19, 1968 THRU JUNE 19, 1969

SUMMARY OF PASQUILL D WIND OCCURRENCES BY SECTOR & SPEED CLASS (NO. OCCUR, PERCENT, STANDARD DEVIATION)

Wind Sector	Item	Sector Total	1.0-3.2 .45-1.49	3.3-5.5 1.5-2.49	5.6-7.8 2.5-3.49	7.9-10.0 3.5-4.49	10.1-12.3 4.5-5.49	12.4-14.5 5.5-6.49	14.6-16.7 6.5-7.49	16.8-19.0 7.5-8.49	19.2-21.2 8.5-9.49	>21.2 MPH >=9.5 M/S
-N-	No.	505	211	188	73	18	10	3	2	0	0	0
	Pct	5.86%	2.49%	2.18%	0.85%	0.21%	0.12%	0.03%	0.02%	0.00%	0.00%	0.00%
-NNE-	No.	371	138	161	40	14	7	3	4	3	1	0
	Pct	4.30%	1.60%	1.87%	0.46%	0.16%	0.08%	0.03%	0.05%	0.03%	0.01%	0.00%
-NE-	No.	566	121	163	145	72	34	15	13	3	0	0
	Pct	6.57%	1.40%	1.89%	1.68%	0.84%	0.39%	0.17%	0.15%	0.03%	0.00%	0.00%
-ENE-	No.	374	76	100	85	77	23	10	3	0	0	0
	Pct	4.34%	0.88%	1.16%	0.99%	0.89%	0.27%	0.12%	0.03%	0.00%	0.00%	0.00%
-E-	No.	336	97	117	66	41	8	6	1	0	0	0
	Pct	3.90%	1.13%	1.36%	0.77%	0.48%	0.09%	0.07%	0.01%	0.00%	0.00%	0.00%
-ESE-	No.	213	84	90	29	6	4	0	0	0	0	0
	Pct	2.45%	0.97%	1.04%	0.34%	0.07%	0.05%	0.00%	0.00%	0.00%	0.00%	0.00%
-SE-	No.	224	65	105	40	13	1	0	0	0	0	0
	Pct	2.60%	0.75%	1.22%	0.46%	0.15%	0.01%	0.00%	0.00%	0.00%	0.00%	0.00%
-SSE-	No.	104	32	50	15	6	1	0	0	0	0	0
	Pct	1.21%	0.37%	0.58%	0.17%	0.07	0.01%	0.00%	0.00%	0.00%	0.00%	0.00%
-S-	No.	122	28	79	10	4	1	0	0	0	0	0
	Pct	1.42%	0.32%	0.92%	0.12%	0.05%	0.01%	0.00%	0.00%	0.00%	0.00%	0.00%
-SSW-	No.	214	27	72	48	42	16	8	1	0	0	0
	Pct	2.48%	0.31%	0.84%	0.56%	0.49%	0.19%	0.09%	0.01%	0.00%	0.00%	0.00%
-SW-	No.	406	79	131	77	57	34	27	1	0	0	0
	Pct	4.71%	0.92%	1.52%	0.89%	0.66%	0.39%	0.31%	0.01%	0.00%	0.00%	0.00%
-WSW-	No.	254	71	54	50	17	27	20	7	3	2	3
	Pct	2.95%	0.82%	0.63%	0.58%	0.20%	0.31%	0.23%	0.08%	0.03%	0.02%	0.03%
-W-	No.	287	63	70	38	25	31	24	17	11	0	3
	Pct	3.33%	0.73%	0.81%	0.44%	0.29%	0.36%	0.34%	0.20%	0.13%	0.00%	0.03%
-WNW-	No.	180	52	44	26	17	20	8	6	5	1	1
	Pct	2.09%	0.60%	0.51%	0.30%	0.20%	0.23%	0.09%	0.07%	0.06%	0.01%	0.01%
-NW-	No.	203	81	62	26	18	9	3	3	0	1	0
	Pct	2.36%	0.94%	0.72%	0.30%	0.21%	0.10%	0.03%	0.03%	0.00%	0.01%	0.00%
-NNW-	No.	200	102	58	25	9	3	3	0	0	0	0
	Pct	2.31%	1.18%	0.67%	0.29%	0.10%	0.03%	0.03%	0.00%	0.00%	0.00%	0.00%
Calm	No.	52	---	---	---	---	---	---	---	---	---	---
	Pct	0.60%	---	---	---	---	---	---	---	---	---	---
Total	No.	4611	1327	1544	793	436	229	135	58	25	5	7
	Pct	53.50%	15.40%	17.91%	9.20%	5.06%	2.66%	1.57%	0.67%	0.29%	0.06%	0.08%

Total Valid Observations: 8619

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Oconee Meteorology
Environmental SF_6 Test

Test Date January 15, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 RELEASE DATA

Bottle weight before release 9.526 kg.

Bottle weight at completion of release 8.338 kg.

Time (24 hr. clock)	Scale Reading (kg)	Total gm. per 20 min.	Total gm per min.	Flowmeter Reading	Smoke Drift Direction
2030	9.526				
2050	9.433	93	4.6	6.0	
2110	9.324	109	5.4	6.0	
2130	9.215	109	5.4	6.0	
2150	9.106	109	5.4	6.0	
2210	8.998	108	5.4	6.0	
2230	8.890	108	5.4	6.0	
2256	8.732	158*	6.0	6.0	
2310	8.657	75**	5.4	6.0	
2330	8.553	104	5.2	6.0	
2350	8.447	106	5.3	6.0	
2410	8.338	109	5.4	6.0	

*Total gm per 26 minutes
**Total gm per 14 minutes

Oconee Meteorology
Environmental SF_6 Test

Test Date January 15, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 DETECTOR READINGS

Point Number	Time (24 hr. clock)	Recorder Reading (%)
1	2044	2
2		2
3		0
4		39
5		100
6	2105	90
7		0
8		0
9		0
10		0
11		0
12		0
13		0
14	2135	16
15		29
16		21
17		27
18		22
19		24
20		21
21		19
22		24
23		20
24		21
25		6
26	2212	0
27		0
28		0
29		5
30		1
31		0
32	2309	3
33		0
34		0
35	2318	6
36		10
37		7
38		9
39		9
40		11
41		9
42		7
43		5

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 DETECTOR READINGS (CONTINUED)
January 15, 1970

Point Number	Time (24 hr. clock)	Recorder Reading (%)
44		5
45	2358	2
46		0
47		0

Oconee Meteorology
Environmental SF_6 Test

Test Date January 15, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
PLUME SIZE DATA

Distance (ft)	From Point No.	To Point No.
77	1	2
174	2	4
77	4	5
96	5	6
96	14	15
115	15	16
232	16	17
115	17	18
115	18	19
154	19	20
154	20	21
174	21	22
154	22	23
308	23	24
250	24	25
270	29	30
482	35	36
174	36	37
212	37	38
135	38	39
154	39	40
154	40	41
482	39	45

Oconee Meteorology
Environmental SF_6 Test

Test Date January 28, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 RELEASE DATA

Bottle weight before release 8.668 kg.

Bottle weight at completion of release 7.740 kg.

Time (24 hr. clock)	Scale Reading (kg)	Total gm. per 20 min.	Total gm. per min.	Flowmeter Reading	Smoke Drift Direction
2055	8.668				
2115	8.563	105	5.2	6.0	
2135	8.457	106	5.3	6.0	
2155	8.354	103	5.1	6.0	
2215	8.251	103	5.1	6.0	
2235	8.148	103	5.1	6.0	
2255	8.045	171*	5.7	6.0	
2325	7.874	134**	5.3	6.0	
2350	7.740	134**	5.3	6.0	

*Total gm. per 30 minutes
**Total gm. per 25 minutes

Oconee Meteorology
Environmental SF6 Test

Test Date January 28, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF6 DETECTOR READINGS

Point Number	Time (24 hr. clock)	Recorder Reading (%)
1	2111	56
2	2117	0
9A	2121	0
9B	2124	0
9C	2126	0
8	2131	0
1A	2134	0
1B	2136	0
1C	2138	0
1	2141	0
10A	2145	67
10B	2148	100
10C	2150	0
2A	2153	0
2B	2155	0
2C	2157	0
3	2159	2
3B	2202	0
3C	2205	0
3D	2207	0
3E	2210	0
3F	2213	0
4	2215	0
3	2221	0
2B	2223	0
2	2227	100
10A	2227	100
10C	2230	100
10D	2233	100
10E	2235	100
1D	2238	100
2D	2242	100
2E	2245	100
2F	2248	100
2G	2251	100
2H	2254	100
2I	2256	63
2J	2259	97
2K	2301	88
2L	2305	13
2M	2308	25
2N	2311	9

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 DETECTOR READINGS (CONTINUED)
January 28, 1970

Point Number	Time (24 hr. clock)	Recorder Reading (%)
2O	2313	26
2P	2320	81
2Q	2324	34
2R	2327	10
2S	2330	17
2T	2335	16
2U	2338	0
2D	2344	5
2A	2348	0
3F	2351	0
4	2353	0
6	2356	0
7	2359	0
8	2401	78
9	2403	35
7A	2408	17
7B	2410	42

Oconee Meteorology
Environmental SF_6 Test

Test Date January 28, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
PLUME SIZE DATA

Distance (ft)	From Point No.	To Point No.
138	10B	10A
116	10A	10C
110	10C	10D
92	10D	10E
305	10E	1D
1900	2D	2I
158	2I	2J
166	2J	2K
109	2K	2L
97	2L	2M
80	2M	2N
95	2N	20
113	20	2P
187	2P	2Q
176	2Q	2R
115	2R	2S
237	2S	2T
150	2T	2U
965	8	9
250	8	7A
350	7A	7B

Oconee Meteorology
Environmental SF6 Test

Test Data January 31, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF6 RELEASE DATA

Bottle weight before release 8.337 kg.

Bottle weight at completion of release 7.867 kg.

Time (24 hr. clock)	Scale Reading (kg)	Total gm. per 20 min.	Total gm. per min.	Flowmeter Reading	Smoke Drift Direction
0512	8.337				
0532	8.280	57	2.9		
0552	8.222	58	2.9		
0612	8.164	58	2.9		
0632	8.110	54	2.7		
0643	8.083	27*	2.4		
0703	7.969	114	5.7		
0723	7.867	102	5.1		

*Total gm per 11 minutes.

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Oconee Meteorology
Environmental SF6 Test

Test Date January 31, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF6 DETECTOR READINGS

Point Number	Time (24 hr. clock)	Recorder Reading (%)
1	0524	18
2	0530	0
3	0533	12
4	0536	0
5	0540	0
6	0545	1
7	0548	3
8	0551	0
9	0554	0
10	0557	0
11	0600	0
12	0603	0
2	0606	0
13	0610	0
14	0613	0
15	0615	0
16	0619	56
17	0622	96
18	0625	0
18	0628	61
19	0631	1
20	0634	0
21	0637	0
18	0640	53
17	0643	12
16	0646	47
22	0650	22
2	0652	0
6	0655	0
7	0657	0
23	0700	0
24	0703	0
25	0706	0
21	0709	1
26	0712	9
20	0715	0
19	0718	0
27	0721	0

Test Date January 31, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
MICROWAVE TOWER ATMOSPHERIC DATA

Time (24 hr. clock)	Ground	Temperature (°F) Δt1	Δt2	Δt3	Wind Direction	Wind Speed (mph)
0500	28	0.7	1.0	1.6	SE	3
0555	27	1.5	2.4	3.0	S	3
0635	27	1.2	2.0	2.5	S	2
0705	25	1.5	2.0	3.0	W	3

KEOWEE RIVER STATION NO. 4 DATA

Time (24 hr. clock)	Temperature (°F) Ground	Pole Top	Wind Direction	Wind Speed (mph)

Test Date February 5, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 RELEASE DATA

Bottle weight before release 8.892 kg.

Bottle weight at completion of release 7.580 kg.

Time (24 hr. clock)	Scale Reading (kg)	Total gm. per 20 min.	Total gm. per min.	Flowmeter Reading	Smoke Drift Direction
1945	8.892				
2005	8.784	108	5.4	6.0	
2025	8.675	109	5.4	6.0	
2045	8.566	109	5.4	6.0	
2105	8.456	110	5.5	6.0	
2125	8.348	108	5.4	6.0	
2145	8.238	110	5.5	6.0	
2205	8.128	110	5.5	6.0	
2225	8.020	108	5.4	6.0	
2245	7.910	110	5.5	6.0	
2305	7.800	110	5.5	6.0	
2325	7.690	110	5.5	6.0	
2343	7.580	110	6.1	6.0	

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Oconee Meteorology
Environmental SF_6 Test

Test Date February 5, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 DETECTOR READINGS

Point Number	Time (24 hr. clock)	Recorder Reading (%)
8	2015	11
8	2036	10
15	2042	11
16	2050	7
17	2054	11
18	2058	13
19	2100	9
20	2105	5
21	2108	5
3	2138	5
24	2143	4
25	2145	3
26	2150	1
22	2247	1
21	2251	6
20	2254	17
19	2257	11
18	2300	6
17	2304	0
35	2315	100
36	2319	24
37	2322	4
38	2325	0
39	2328	0

Test Date February 5, 1970

TABLE 2A-5 SUPPLEMENTAL DATA

PLUME SIZE DATA

Distance (ft)	From Point No.	To Point No.
206	3	24
150	24	25
127	25	26
295	15	8
200	8	16
250	16	17
176	17	18
233	18	19
263	19	20
247	20	21
255	21	22
300	36	37
243	37	38
225	36	39
135	39	40
573	38	39

Oconee Meteorology
Environmental SF_6 Test

Test Date February 5, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
MICROWAVE TOWER ATMOSPHERIC DATA

Time (24 hr. clock)	Temperature (°F) Ground	Δt_1	Δt_2	Δt_3	Wind Direction	Wind Speed (mph)
1855	40	0.7	2.7	2.9		
1940	39	1.5	3.5	4.3	SSE	10
2016	35	0.0	3.1	5.8	WSW	3
2110	33	0.4	2.5	6.3	NNW	6
2150	33	0.2	3.4	6.6	NNW	7
2238	34	-0.5	0.5	2.3	N	7
2312	33	-0.1	1.5	4.1	ENE	5

KEOWEE RIVER STATION NO. 4 DATA

Time (24 hr. clock)	Temperature (°F) Ground	Pole Top	Wind Direction	Wind Speed (mph)
2037	31	32	SSE	3
2128	31	32	SSE	4

Oconee Meteorology
Environmental SF_6 Test

Test Date February 6, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 RELEASE DATA

Bottle weight before release 8.604 kg.

Bottle weight at completion of release 7.705 kg.

Time (24 hr. clock)	Scale Reading (kg)	Total gm. per 20 min.	Total gm. per min.	Flowmeter Reading	Smoke Drift Direction
2000	8.604				
2020	8.512	92	4.6	6.0	
2040	8.404	108	5.4	6.0	
2100	8.295	109	5.4	6.0	
2120	8.180	115	5.7	6.0	
2140	8.060	120	6.0	6.0	
2200	7.966	94	4.7	6.0	
2220	7.850	116	5.8	6.0	
2240	7.742	108	5.4	6.0	
2246	7.705	37*	6.2	6.0	

*Total gm. per 6 minutes

Oconee Meteorology
Environmental SF_6 Test

Test Date February 6, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 DETECTOR READINGS

Point Number	Time (24 hr. clock)	Recorder Reading (%)
1	2015	0
2	2018	0
3	2021	0
4	2025	0
5	2029	0
6	2032	0
7	2035	0
8	2038	8
9	2041	23
10	2043	21
11	2047	20
12	2052	2
13	2055	0
14	2058	0
15	2101	0
16	2105	0
17	2108	0
18	2112	0
19	2115	0
20	2118	0
21	2122	0
5	2129	0
6	2132	0
7	2135	3
8	2139	5
9	2144	0
9	2147	0
10	2147	0
11	2150	0
12	2153	4
13	2159	4
13A	2202	2
14	2205	0
15	2208	0
17	2211	0
18	2214	0
19	2219	0
20	2222	0
21	2226	0
22	2229	0
23	2232	0
24	2235	0

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 DETECTOR READINGS (CONTINUED)
February 6, 1970

Point Number	Time (24 hr. clock)	Recorder Reading (%)
25	2238	0
26	2240	11
27	2243	14

Test Date February 6, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
PLUME SIZE DATA

Distance (ft)	From Point No.	To Point No.
177	7	8
104	8	9
89	9	10
83	10	11
63	11	12
156	12	13
91	13	13A

Test Date February 6, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
MICROWAVE TOWER ATMOSPHERIC DATA

Time (24 hr. clock)	Temperature (°F) Ground	$\Delta t1$	$\Delta t2$	$\Delta t3$	Wind Direction	Wind Speed (mph)
1900	47	-0.3	1.8	2.0	E	2
2015	41	2.0	5.0	5.6	E	2
2055	39	1.0	4.5	7.0	N	2
2140	38	-0.7	4.5	8.0	N	1
2145	38	0.5	5.5	7.5	NE	1
2220	37	0.5	3.5	6.5	NW	4

KEOWEE RIVER STATION NO. 4 DATA

Time (24 hr. clock)	Temperature (°F) Ground	Pole Top	Wind Direction	Wind Speed (mph)

Oconee Meteorology
Environmental SF_6 Test

Test Date February 10, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 RELEASE DATA

Bottle weight before release 9.233 kg.

Bottle weight at completion of release 8.540 kg.

Time (24 hr. clock)	Scale Reading (kg)	Total gm per 20 min.	Total gm per min.	Flowmeter Reading	Smoke Drift Direction
2110	9.233				SE
2130	9.145	88	4.4	6.0	SE
2150	9.033	112	5.6	6.0	S
2210	8.925	108	5.4	6.0	SE
2230	8.810	115	5.7	6.0	SE
2250	8.710	100	5.0	6.0	SW
2310	8.595	115	5.7	6.0	SE
2320	8.540	55*	5.5	6.0	SW

*Total gm. per 10 minutes

Test Date February 10, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 DETECTOR READINGS

Point Number	Time (24 hr. clock)	Recorder Reading (%)
1	2124	0
2	2126	0
3	2129	0
4	2133	0
5	2138	100
6	2141	8
7	2144	0
8	2148	0
6	2150	0
9	2153	9
5	2156	48
10	2158	69
11	2202	46
12	2204	13
13	2207	96
14	2210	24
15	2213	6
16	2215	0
17	2220	31
18	2223	27
19	2225	0
19	2228	0
19	2234	21
19	2237	44
20	2240	23
21	2244	22
22	2247	23
23	2252	22
24	2255	19
25	2257	6
26	2300	5
18	2303	21
17	2305	6
17	2307	55
16	2309	0

Oconee Meteorology
Environmental SF_6 Test

Test Date February 10, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
PLUME SIZE DATA

Distance (ft)	From Point No.	To Point No.
165	6	9
139	9	5
135	5	10
105	10	11
154	11	12
102	17	13
74	13	14
145	14	15
60	17	18
123	18	19
97	19	20
79	20	21
83	21	22
59	22	23
55	23	24
75	24	25
71	25	26

Test Date February 10, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
MICROWAVE TOWER ATMOSPHERIC DATA

Time (24 hr. clock)	Temperature (°F) Ground	Δt_1	Δt_2	Δt_3	Wind Direction	Wind Speed (mph)
1908	40	0.7	0.2	0.0	SW	20
1958	40	0.7	0.2	0.0	SW	20
2028	39	0.7	0.2	0.1	SW	23
2056	40	0.7	0.2	0.1	SW	18
2145	39	0.8	0.4	0.2	SW	15
2218	38	0.8	0.5	0.4	SW	11

KEOWEE RIVER STATION NO. 4 DATA

Time (24 hr. clock)	Temperature (°F) Ground	Pole Top	Wind Direction	Wind Speed (mph)
2013	42	42	W	14
2130	40	40	E	8

Test Date February 11, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 RELEASE DATA

Bottle weight before release 8.511 kg.

Bottle weight at completion of release 7.574 kg.

Time (24 hr. clock)	Scale Reading (kg)	Total gm. per 20 min.	Total gm. per min.	Flowmeter Reading	Smoke Drift Direction
1955	8.511				S
2015	8.420	91	4.6	6.0	S
2035	8.315	105	5.3	6.0	SW
2055	8.215	100	5.0	6.0	SW
2115	8.090	125	6.3	6.0	SE
2135	7.975	115	5.8	6.0	SE
2155	7.870	105	5.3	6.0	E
2215	7.760	110	5.5	6.0	E
2235	7.660	100	5.0	6.0	SE
2255	7.574	86	4.3	6.0	SE

Oconee Meteorology
Environmental SF_6 Test

Test Date February 11, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 DETECTOR READINGS

Point Number	Time (24 hr. clock)	Recorder Reading (%)
1	2015	0
2	2018	0
3	2021	0
4	2024	0
5	2027	0
6	2032	0
7	2035	0
8	2038	0
9	2044	0
10	2048	0
11	2051	4
12	2054	4
13	2057	21
14	2100	100
15	2103	95
16	2106	100
17	2109	100
18	2112	2
18	2114	0
19	2117	100
20	2120	44
21	2124	100
22	2126	81
23	2129	50
24	2131	25
25	2134	63
26	2136	50
27	2140	44
28	2143	45
29	2145	33
30	2148	40
31	2153	7
32	2155	3
33	2158	0
34	2200	0
35	2203	0
36	2205	0
37	2208	0
38	2211	0
39	2214	0
40	2216	0
41	2223	6

TABLE 2A-5 SUPPLEMENTAL DATA

SF_6 DETECTOR READINGS (CONTINUED)

February 11, 1970

Point Number	Time (24 hr. clock)	Recorder Reading (%)
42	2226	9
43	2230	1
44	2233	0
45	2241	0
46	2245	0
47	2247	0
48	2250	0
49	2253	0
50	2256	0

Oconee Meteorology
Environmental SF_6 Test

Test Date February 11, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
PLUME SIZE DATA

Distance (ft)	From Point No.	To Point No.
85	11	12
130	12	13
85	13	14
106	14	15
192	15	16
163	16	17
117	17	18
222	18	19
111	19	20
110	20	21
102	21	22
133	22	23
140	23	24
135	24	25
108	25	26
104	26	27
154	27	28
124	28	29
163	29	30
182	30	31
125	41	42
138	42	43

Test Date February 11, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
MICROWAVE TOWER ATMOSPHERIC DATA

Time (24 hr. clock)	Temperature (°F) Ground	$\Delta t1$	$\Delta t2$	$\Delta t3$	Wind Direction	Wind Speed (mph)
1925	54	1.5	1.3	1.4	W	15
2005	53	1.3	1.2	1.2	W	8
2040	52	1.4	1.2	1.3	W	11
2125	50	1.5	1.5	2.0	W	15
2225	50	1.4	1.0	1.2	W	20

KEOWEE RIVER STATION NO. 4 DATA

Time (24 hr. clock)	Temperature (°F) Ground	Pole Top	Wind Direction	Wind Speed (mph)
2015	50	45	NW	2

Oconee Meteorology
Environmental SF_6 Test

Test Date February 12, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 RELEASE DATA

Bottle weight before release 9.260 kg.

Bottle weight at completion of release 8.565 kg.

Time (24 hr. clock)	Scale Reading (kg)	Total gm per 20 min.	Total gm per min.	Flowmeter Reading	Smoke Drift Direction
2005	9.260				W
2025	9.125	135	6.2	6.0	SSE
2045	9.042	83	4.3	6.0	NW
2108	8.917	125**	5.5	6.0	NW
2125	8.830	87***	5.1	6.0	SSE
2145	8.724	106	5.3	6.0	E
2205	8.620	104	5.2	6.0	S
2215	8.565	55*	5.5	6.0	S

*Total gm. per 10 minutes
**Total gm. per 23 minutes
***Total gm. per 17 minutes

Oconee Meteorology
Environmental SF6 Test

Test Date February 12, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF6 DETECTOR READINGS

Point Number	Time (24 hr. clock)	Recorder Reading (%)
1	2014	0
1	2016	0
2	2028	>100
3	2031	1
3	2033	0
4	2035	0
2	2037	0
5	2040	0
6	2043	36
7	2045	46
8	2048	68
9	2051	81
10	2053	44
11	2055	57
12	2057	23
13	2059	20
14	2101	2
15	2103	75
16	2105	0
15	2108	7
13	2112	16
11	2114	40
9	2116	25
7	2118	31
6	2120	11
5	2122	2
2A	2125	43
2	2128	13
4	2132	0
3	2135	0
17	2143	0
18	2145	0
19	2151	0
20	2154	0
21	2156	0
22	2159	0
23	2201	0
24	2203	0
25	2205	0
26	2208	0
27	2210	0
28	2212	4

Oconee Meteorology
Environmental SF_6 Test

Test Date February 12, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
PLUME SIZE DATA

Distance (ft)	From Point No.	To Point No.
87	2	2A
94	2A	5
89	5	6
78	6	7
79	7	8
59	8	9
57	9	10
69	10	11
71	11	12
71	12	13
64	13	14
37	14	15
42	15	16

Oconee Meteorology
Environmental SF_6 Test

Test Date February 12, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
MICROWAVE TOWER ATMOSPHERIC DATA

Time (24 hr. clock)	Temperature (°F) Ground	$\Delta t1$	$\Delta t2$	$\Delta t3$	Wind Direction	Wind Speed (mph)
1900	47	0.7	0.7	0.7	WSW	20
1940	46	0.5	0.4	-0.2	WSW	18
2000	44	0.8	0.5	0.5	SW	21
2045	42	0.7	0.4	0.5	WSW	15
2140	40	1.1	0.8	0.8	WSW	10

KEOWEE RIVER STATION NO. 4 DATA

Time (24 hr. clock)	Temperature (°F) Ground	Pole Top	Wind Direction	Wind Speed (mph)
2100	43	43	WSW	6
2135	43	42	W	3

Test Date February 17, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF6 RELEASE DATA

Bottle weight before release 8.549 kg.

Bottle weight at completion of release 7.830 kg.

Time (24 hr. clock)	Scale Reading (kg)	Total gm per 20 min.	Total gm per min.	Flowmeter Reading	Smoke Drift Direction
2050	8.549				
2110	8.439	110	5.5	6.0	
2130	8.330	109	5.5	6.0	
2150	8.222	108	5.4	6.0	
2210	8.116	106	5.3	6.0	
2230	8.011	105	5.3	6.0	
2250	7.906	105	5.3	6.0	
2304	7.830	76	5.4	6.0	

Oconee Meteorology
Environmental SF_6 Test

Test Date February 17, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 DETECTOR READINGS

Point Number	Time (24 hr. clock)	Recorder Reading (%)
1	2102	0
2	2106	58
3	2108	12
4	2110	10
5	2112	0
6	2120	0
7	2124	0
8	2140	0
9	2142	0
10	2144	>100
11	2152	0
12	2156	37
13	2200	27
14	2202	15
7	2205	9
6	2207	6
2	2209	5
3	2211	6
4	2213	7
5	2215	18
15	2217	47
16	2220	>100
17	2222	53
18	2224	12
19	2226	3
20	2229	0
18	2237	0
17	2239	0
15	2241	0
2	2244	0
13	2247	10
21	2249	>100
22	2251	>100
23	2254	21
24	2256	4

Oconee Meteorology
Environmental SF_6 Test

Test Date February 17, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
PLUME SIZE DATA

Distance (ft)	From Point No.	To Point No.
138	2	3
81	3	4
508	12	13
105	13	14
215	14	7
150	7	6
165	6	2
138	2	3
81	3	4
165	4	5
226	5	15
195	15	16
133	16	17
183	17	18
122	18	19
330	13	21
118	21	22
96	22	23
122	23	24

Oconee Meteorology
Environmental SF_6 Test

Test Date February 17, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
MICROWAVE TOWER ATMOSPHERIC DATA

Time (24 hr. clock)	Temperature (°F) Ground	Δt_1	Δt_2	Δt_3	Wind Direction	Wind Speed (mph)
2100	47	0.0	0.0	0.4	NE	5
2130	46	1.0	1.6	1.0	E	4
2200	47	0.4	-0.1	-0.1	SE	4
2230	47	0.2	-0.1	-0.2	SE	7

KEOWEE RIVER STATION NO. 4 DATA

Time (24 hr. clock)	Temperature (°F) Ground	Pole Top	Wind Direction	Wind Speed (mph)
2000	46	46	S	2

Test Date February 19, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 RELEASE DATA

Bottle weight before release 9.863 kg.

Bottle weight at completion of release 8.808 kg.

Time (24 hr. clock)	Scale Reading (kg)	Total gm. per 20 min.	Total gm per min.	Flowmeter Reading	Smoke Drift Direction
2318	9.863				N
2340	9.738	125	6.2	6.2	S
2400	9.632	106	5.3	6.0	N
0020	9.520	112	5.6	6.0	SE
0040	9.402	118	5.9	6.0	S
0100	9.300	102	5.1	6.1	SE
0120	9.182	118	5.9	6.0	S
0140	9.082	100	5.0	5.9	N
0200	8.968	114	5.7	5.9	NW
0220	8.865	103	5.1	6.0	NE
0230	8.808	57	5.7	6.0	S

Oconee Meteorology
Environmental SF_6 Test

Test Date February 19, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 DETECTOR READINGS

Point Number	Time (24 hr. clock)	Recorder Reading (%)
1	2330	0
2	2333	0
3	2336	0
4	2340	0
5	2343	100
6	2348	2
6	2350	0
7	2353	0
8	2359	0
9	2405	0
10	2407	0
9	2410	0
8	2413	0
11	2418	0
12	2424	0
5	2428	100
13	2432	15
14	2435	8
15	2438	6
16	2441	0
17	2444	1
7	2447	0
18	2455	0
18	0100	0
19	0106	0
20	0114	100
12	0116	2
21	0120	25
22	0124	0
23	0126	0
24	0129	0
25	0132	0
26	0134	0
27	0137	0
2	0140	0
28	0144	0
29	0148	0
30	0153	0
31	0155	0
32	0157	0
33	0159	0
34	0200	0
8	0207	0
35	0210	0

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 DETECTOR READINGS (CONTINUED)
February 19, 1970

Point Number	Time (24 hr. clock)	Recorder Reading (%)
9	0212	0
10	0214	0
36	0216	0
37	0218	0
38	0220	0
39	0222	0

Oconee Meteorology
Environmental SF_6 Test

Test Date March 2, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
PLUME SIZE DATA

Distance (ft)	From Point No.	To Point No.
365	13	15
78	15	16
32	16	17

Test Date March 2, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
MICROWAVE TOWER ATMOSPHERIC DATA

Time (24 hr. clock)	Temperature Ground	$\Delta t1$	$\Delta t2$	$\Delta t3$	Wind Direction	Wind Speed (mph)
1917	64	2.6	3.1	3.5	S	8
1953	63	2.1	2.6	3.1	S	9
2050	56	3.2	5.8	8.0	S	6
2138	54	2.2	4.8	6.0	NNW	4
2215	52	2.2	2.8	6.8	NNW	6
2250	50	2.5	3.4	8.7	N	5
2330	49	2.4	3.2	7.8	N	7

KEOWEE RIVER STATION NO. 4 DATA

Time (24 hr. clock)	Temperature (°F) Ground	Pole Top	Wind Direction	Wind Speed (mph)
1928	55	51	S	3
2100	51	47	N	1

Oconee Meteorology
Environmental SF_6 Test

Test Date March 2, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 RELEASE DATA

Bottle weight before release 8.779 kg.

Bottle weight at completion of release 8.185 kg.

Time (24 hr. clock)	Scale Reading (kg)	Total gm per 20 min.	Total gm per min.	Flowmeter Reading	Smoke Drift Direction
2150	8.779				S
2210	8.668	111	5.6	6.0	SE
2230	8.563	105	5.3	6.0	NW
2250	8.458	105	5.3	6.0	N
2315	8.320	138*	5.5	6.0	SW
2330	8.242	78**	5.2	6.0	SW
2339	8.185	57***	6.3	6.0	SW

*Total gm. per 25 min.
**Total gm. per 15 min.
***Total gm. per 9 min.

Oconee Meteorology
Environmental SF6 Test

Test Date March 2, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF6 DETECTOR READINGS

Point Number	Time (24 hr. clock)	Recorder Reading (%)
1	2157	0
2	2202	0
3	2206	0
4	2209	0
5	2212	0
6	2215	0
7	2219	0
8	2223	0
9	2225	0
10	2228	0
11	2230	0
12	2232	0
13	2235	11
14	2239	0
15	2242	10
16	2245	12
17	2249	4
18	2252	0
13	2255	0
19	2258	0
20	2301	0
15	2305	0
16	2307	0
21	2309	0
22	2312	0
23	2315	0
24	2318	0
25	2322	0
26	2327	0
27	2330	0
13	2333	0
14	2336	0

Oconee Meteorology
Environmental SF_6 Test

Test Date March 3, 1970

TABLE 2A-5 SUPPLEMENTAL DATA

PLUME SIZE DATA

Distance (ft)	From Point No.	To Point No.
324	1	4
156	4	5
92	5	6
105	6	7
216	7	9
103	9	14
424	14	6
572	6	1
160	1	15
76	15	16
48	16	17
47	17	18
57	18	4
200	14	10

Test Date March 3, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
MICROWAVE TOWER ATMOSPHERIC DATA

Time (24 hr. clock)	Temperature (°F) Ground	$\Delta t1$	$\Delta t2$	$\Delta t3$	Wind Direction	Wind Speed (mph)
1930	64	1.3	1.1	1.0	E	7
2020	62	1.6	1.8	2.0	E	5
2048	62	1.5	1.6	1.7	ESE	7
2120	60	1.5	2.2	2.2	ESE	3
2200	61	1.1	0.9	0.9	ESE	5
2223	60	2.0	2.0	2.0	ESE	5

KEOWEE RIVER STATION NO. 4 DATA

Time (24 hr. clock)	Temperature (°F) Ground	Pole Top	Wind Direction	Wind Speed (mph)
2010	59	55	N	3

Test Date March 3, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 RELEASE DATA

Bottle weight before release 9.886 kg.

Bottle weight at completion of release 9.125 kg.

Time (24 hr. clock)	Scale Reading (kg)	Total gm per 20 min.	Total gm per min.	Flowmeter Reading	Smoke Drift Direction
2000	9.886				W
2020	9.796	90	4.5	6.0	W
2040	9.682	114	5.7	6.0	E
2100	9.585	97	4.8	6.0	S
2120	9.480	105	5.2	6.0	S
2142	9.355	125	5.7	6.0	NE
2200	9.260	95	5.2	6.0	NE
2220	9.160	100	5.0	6.0	W
2227	9.125	35	5.0	6.0	W

Oconee Meteorology
Environmental SF_6 Test

Test Date March 3, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 DETECTOR READINGS

Point Number	Time (24 hr. clock)	Recorder Reading (%)
1	2015	3
2	2019	0
3	2023	0
4	2026	17
5	2029	21
6	2032	30
7	2046	4
8	2048	0
9	2050	0
10	2053	0
11	2056	0
12	2058	0
13	2101	0
7	2105	47
9	2108	40
10	2111	0
14	2114	1
6	2117	14
4	2121	0
1	2123	25
15	2127	15
16	2131	18
17	2134	6
18	2137	3
4	2140	3
2	2143	0
19	2146	0
5	2149	16
6	2152	17
7	2154	15
9	2157	11
14	2200	13
10	2202	9
11	2205	1
12	2208	0
13	2212	0
20	2215	0
21	2220	91

Oconee Meteorology
Environmental SF_6 Test

Test Date March 5, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 RELEASE DATA

Bottle weight before release 9.128 kg.

Bottle weight at completion of release 8.020 kg.

Time (24 hr. clock)	Scale Reading (kg)	Total gm per 20 min.	Total gm per min.	Flowmeter Reading	Smoke Drift Direction
2025	9.128				NW
2045	9.018	110	5.5	6.0	NW
2105	8.900	118	5.9	6.1	SE
2125	8.798	102	5.0	6.0	W
2145	8.689	109	5.4	6.1	W
2205	8,580	99	5.0	6.0	SE
2225	8.482	98	5.0	6.0	W
2245	8.376	106	5.3	6.0	SE
2305	8.269	107	5.3	6.0	W
2325	8.160	109	5.3	6.2	N
2345	8.049	111	5.5	6.3	NW
2355	8.020	29*	2.	1.8	N

*Total gm. per 10 minutes

Oconee Meteorology
Environmental SF_6 Test

Test Date March 5, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 DETECTOR READINGS

Point Number	Time (24 hr. clock)	Recorder Reading (%)
1	2038	0
2	2042	0
3	2044	0
4	2048	0
5	2053	0
6	2055	0
7	2100	0
8	2103	0
9	2105	0
10	2108	0
11	2110	0
12	2113	0
13	2116	0
14	2119	0
15	2121	0
16	2123	0
17	2129	0
18	2131	0
19	2135	0
20	2138	0
21	2142	0
22	2147	0
5	2152	2
23	2154	6
1	2157	13
24	2201	19
25	2204	14
2	2206	10
26	2209	7
3	2212	3
4	2217	8
27	2220	4
28	2223	3
29	2226	3
30	2229	3
7	2233	2
31	2238	2
8	2241	3
32	2244	3
9	2247	2
9	2250	2
10	2252	2
11	2255	0

TABLE 2A-5 SUPPLEMENTAL DATA

SF_6 DETECTOR READINGS

March 5, 1970

Point Number	Time (24 hr. clock)	Recorder Reading (%)
11	2257	2
12	2259	4
33	2303	2
33	2305	3
34	2307	2
35	2310	3
36	2313	2
17	2317	3
37	2320	2
37	2322	7
18	2326	28
38	2329	10
19	2332	32
20	2335	75
39	2337	7
40	2341	12
41	2344	16
42	2347	20
22	2350	>100
6	2353	65

Oconee Meteorology
Environmental SF6 Test

Test Date March 5, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
MICROWAVE TOWER ATMOSPHERIC DATA

Time (24 hr. clock)	Temperature (°F) Ground	$\Delta t1$	$\Delta t2$	$\Delta t3$	Wind Direction	Wind Speed (mph)
1955	53	1.2	1.5	2.0	N	4
2100	51	2.0	3.0	3.5	NE	2
2155	51	1.0	3.0	4.0	N	3
2305	50	1.4	2.5	4.0	N	5
2335	50	1.3	2.0	3.8	NE	3

KEOWEE RIVER STATION NO. 4 DATA

Time (24 hr. clock)	Temperature (°F) Ground	Pole Top	Wind Direction	Wind Speed (mph)
2050	51	50	SE	2

Oconee Meteorology
Environmental SF_6 Test

Test Date March 10, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 RELEASE DATA

Bottle weight before release 8.728 kg.

Bottle weight at completion of release 7.673 kg.

Time (24 hr. clock)	Scale Reading (kg)	Total gm per 20 min.	Total gm per min.	Flowmeter Reading	Smoke Drift Direction
2030	8.728				E
2050	8.618	110	5.5	6.2	W
2110	8.500	118	5.9	6.1	NW
2130	8.380	120	6.0	6.1	S
2150	8,260	120	6.0	6.2	SE
2210	8.135	125	6.2	6.0	NE
2230	8.020	115	5.7	6.1	E
2250	7.902	118	5.9	6.0	SW
2315	7.758	144*	5.7	6.0	NE
2330	7.673	85**	5.7	6.0	N

*Total gm. per 25 minutes
**Total gm. per 15 minutes

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Oconee Meteorology
Environmental SF_6 Test

Test Date March 10, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 DETECTOR READINGS

Point Number	Time (24 hr. clock)	Recorder Reading (%)
1	2040	0
2	2044	0
3	2047	0
4	2050	0
5	2053	0
6	2058	0
7	2101	0
8	2104	0
9	2107	88
10	2111	0
11	2114	0
12	2117	0
9	2120	0
13	2123	14
14	2127	4
15	2129	0
14	2131	0
8	2136	0
7	2139	11
16	2142	4
17	2144	47
18	2148	100
6	2151	61
19	2154	0
6	2157	0
18	2200	45
17	2203	43
16	2207	41
7	2210	47
20	2212	21
21	2217	0
20	2219	0
7	2222	0
16	2224	0
17	2226	0
18	2230	0
6	2233	0
22	2238	0
5	2244	100
23	2247	27
24	2251	0
25	2253	100

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 DETECTOR READINGS (CONTINUED)
March 10, 1970

Point Number	Time (24 hr. clock)	Recorder Reading (%)
26	2258	7
27	2302	0
25	2305	0
5	2308	4
6	2312	18
18	2316	10
17	2318	18
16	2320	12
7	2322	7
20	2325	0
17	2328	100

Oconee Meteorology
Environmental SF_6 Test

Test Date March 10, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
PLUME SIZE DATA

Distance (ft)	From Point No.	To Point No.
504	9	13
98	13	14
193	7	16
94	16	17
92	17	18
82	18	6
127	7	20
98	5	23
285	23	25
68	25	26
276	5	6
82	6	18

Oconee Meteorology
Environmental SF6 Test

Test Date March 10, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
MICROWAVE TOWER ATMOSPHERIC DATA

Time (24 hr. clock)	Temperature (°F) Ground	$\Delta t1$	$\Delta t2$	$\Delta t3$	Wind Direction	Wind Speed (mph)
1945	66	1.0	1.0	1.0	S	12
2035	63	1.5	1.5	1.8	SSW	6
2105	61	2.0	2.3	2.4	SSW	7
2130	62	1.0	1.1	1.5	SSW	10
2220	61	0.9	0.9	1.2	SSW	8
2245	55	3.7	4.0	4.4	SSW	6
2322	56	2.5	2.6	3.0	SSW	6

KEOWEE RIVER STATION NO. 4 DATA

Time (24 hr. clock)	Temperature (°F) Ground	Pole Top	Wind Direction	Wind Speed (mph)
2050	57	50	S	1

Test Date March 11, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 RELEASE DATA

Bottle weight before release 10.112 kg.

Bottle weight at completion of release 9.020 kg.

Time (24 hr. clock)	Scale Reading (kg)	Total gm. per 20 min.	Total gm. per min.	Flowmeter Reading	Smoke Drift Direction
2015	10.112				E
2035	10.000	112	5.6	6.0	E
2057	9.855	145	7.3	6.1	E
2115	9.750	105	5.3	6.0	E
2135	9.620	130	6.5	6.0	NE
2155	9.500	120	6.0	6.0	E
2215	9.380	120	6.0	6.0	E
2235	9.260	120	6.0	6.0	S
2255	9.140	120	6.0	6.0	S
2315	9.020	120	6.0	6.0	SE

Oconee Meteorology
Environmental SF6 Test

Test Date March 11, 1970

TABLE 2A-5 SUPPLEMENTAL DATA
SF6 DETECTOR READINGS

Point Number	Time (24 hr. clock)	Recorder Reading (%)
1	2034	0
2	2036	0
3	2038	0
4	2041	0
5	2044	2
6	2048	4
7	2051	1
8	2054	
8	2056	5
9	2059	8
10	2102	2
11	2106	9
12	2109	5
13	2111	6
14	2114	16
15	2116	17
16	2120	14
17	2123	15
18	2127	14
19	2130	28
20	2133	2
21	2136	8
22	2139	9
23	2144	31
24	2146	>100
25	2149	81
26	2152	84
27	2154	14
28	2158	23
29	2201	3
30	2204	10
30	2206	7
31	2209	18
32	2212	15
33	2216	0
33	2222	0
32	2226	0
30	2228	0
29	2230	9
27	2233	6
25	2236	4
23	2239	2

TABLE 2A-5 SUPPLEMENTAL DATA
SF_6 DETECTOR READINGS (CONTINUED)
March 11, 1970

Point Number	Time (24 hr. clock)	Recorder Reading (%)
21	2241	32
19	2244	23
17	2246	4
15	2249	3
13	2251	0
11	2253	0
5	2257	2
4	2300	6
3	2303	4
2	2306	38
1	2309	35
34	2311	44
35	2314	>100
36	2318	0

Oconee Meteorology
Environmental SF_6 Test

Test Date March 11, 1970

TABLE 2A-5 SUPPLEMENTAL DATA

MICROWAVE TOWER ATMOSPHERIC DATA

Time (24 hr. clock)	Temperature Ground	Δt_1	Δt_2	Δt_3	Wind Direction	Wind Speed (mph)
1910	64	1.8	2.2	2.2	SE	8
1930	64	1.6	1.7	1.8	SE	8
2000	63	1.2	1.7	1.8	E	5
2040	62	1.2	1.3	2.3	E	7
2110	61	1.2	1.9	2.4	E	6
2150	60	1.8	2.5	3.0	E	6
2240	60	1.4	1.6	1.8	E	8

KEOWEE RIVER STATION NO. 4 DATA

Time (24 hr. clock)	Temperature (°F) Ground	Pole Top	Wind Direction	Wind Speed (mph)
1920	60	56	variable	3
2005	60	56	S	1

Rev. 7. 7/9/70

LEGEND – METEOROLOGICAL INSTRUMENT SITING

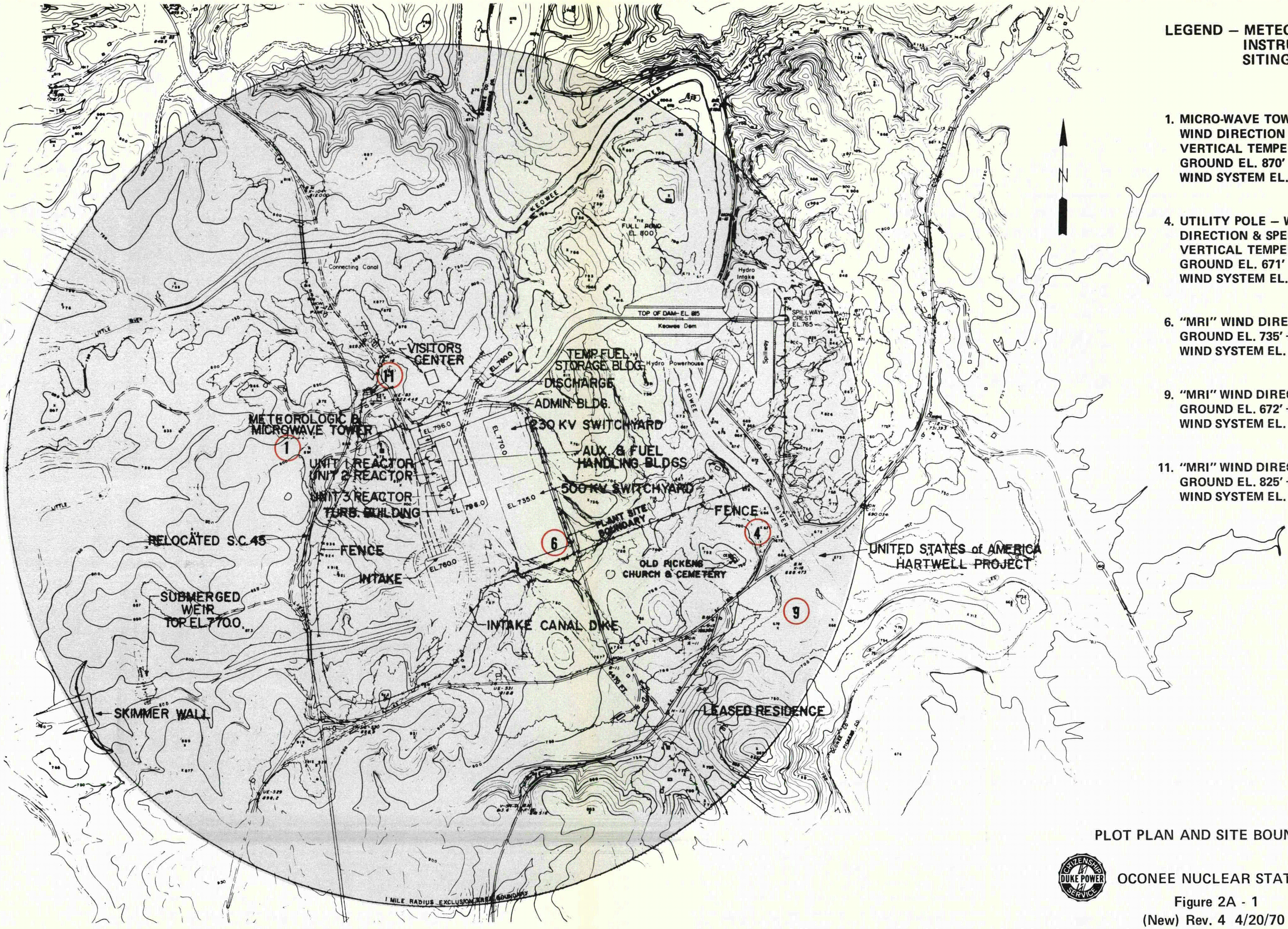
1. MICRO-WAVE TOWER
   WIND DIRECTION & SPEED
   VERTICAL TEMPERATURE
   GROUND EL. 870' + 8"
   WIND SYSTEM EL. 1028' + 8"

4. UTILITY POLE – WIND DIRECTION & SPEED
   VERTICAL TEMPERATURE
   GROUND EL. 671' + 0"
   WIND SYSTEM EL. 717' + 0"

6. "MRI" WIND DIRECTION & SPEED
   GROUND EL. 735' + 0"
   WIND SYSTEM EL. 741' + 0"

9. "MRI" WIND DIRECTION & SPEED
   GROUND EL. 672' + 0"
   WIND SYSTEM EL. 678' + 0"

11. "MRI" WIND DIRECTION & SPEED
   GROUND EL. 825' + 0"
   WIND SYSTEM EL. 835' + 0"



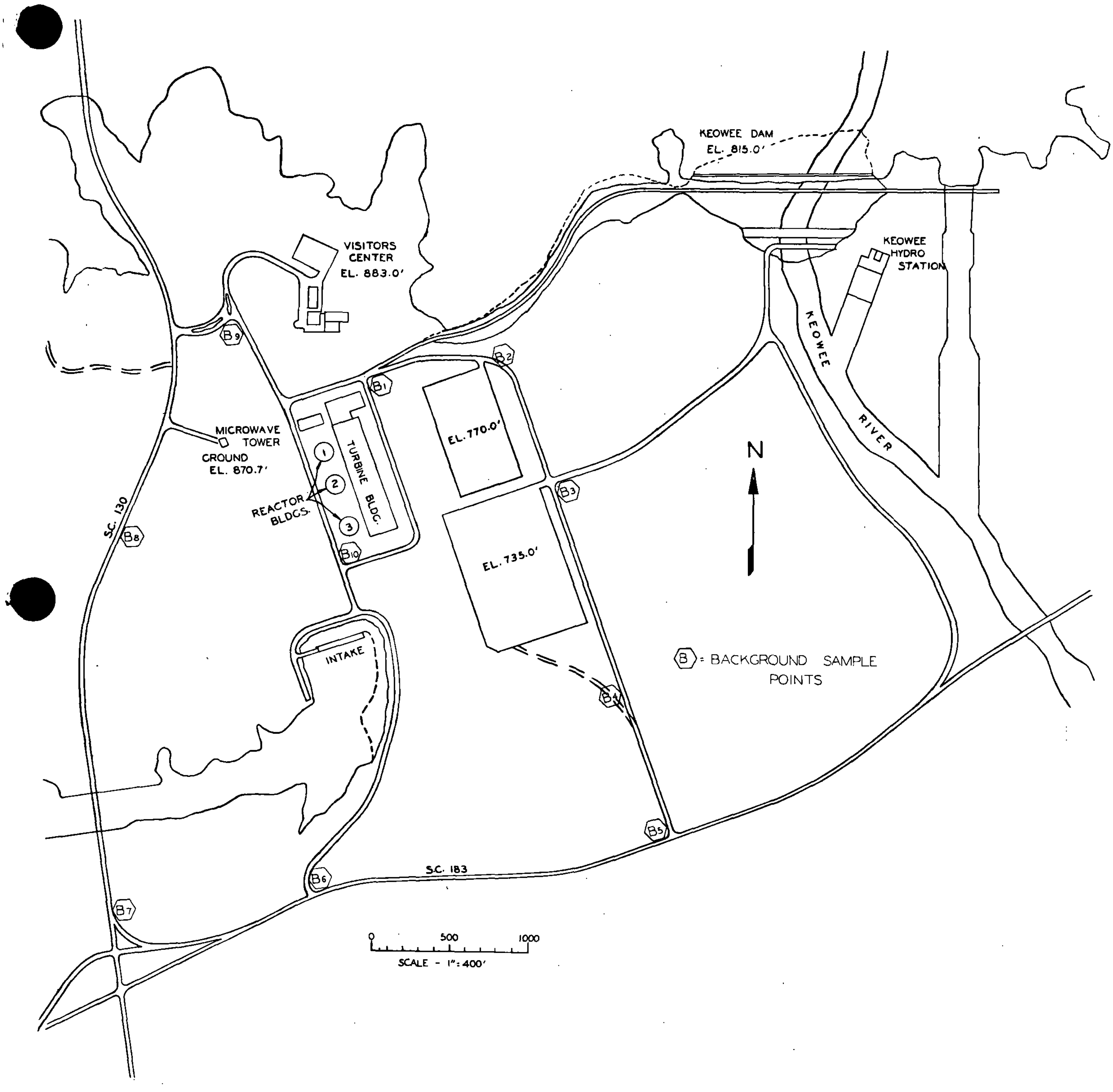
PLOT PLAN AND SITE BOUNDARY

DUKE POWER

OCONEE NUCLEAR STATION

Figure 2A - 1

(New) Rev. 4 4/20/70

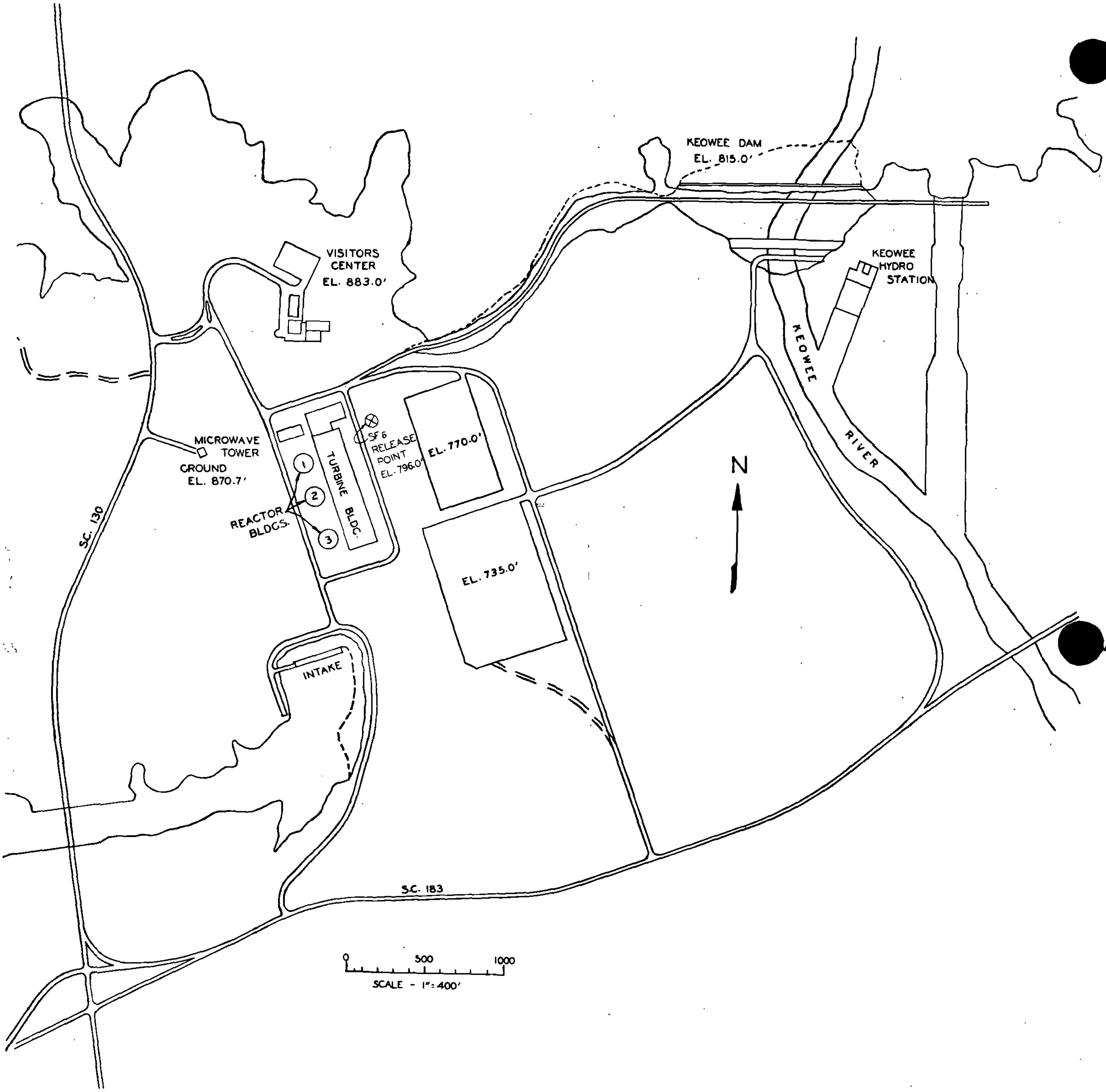


SF6 GAS TRACER TEST
BACKGROUND SAMPLE POINTS

OCONEE NUCLEAR STATION

Figure 2A - 2

(New) Rev. 4 4/20/70



SF6 GAS TRACER TEST

SF6 RELEASE POINT

OCONEE NUCLEAR STATION

Figure 2A - 3

(New) Rev. 4 4/20/70

Oconee Meteorology
Environmental SF_6 Test

Test Date ________________

SF_6 RELEASE DATA

Bottle weight before release _____kg.
Bottle weight at completion of release _____kg.

Time (24 hr. clock)	Scale Reading (kg)	Total gm per 20 min.	Total gm per min.	Flowmeter Reading	Smoke Drift Direction

Oconee Meteorology
Environmental SF_6 Test

Test Date ________________

PLUME SIZE DATA

Distance From Point No. _____ To Point No. _____

Oconee Meteorology
Environmental SF_6 Test

Test Date ________________

SF_6 DETECTOR READINGS*

Point Number	Time (24 hr. clock)	Recorder Reading (%)

* Record all readings taken

Oconee Meteorology
Environmental SF_6 Test

Test Date ________________

MICROWAVE TOWER ATMOSPHERIC DATA

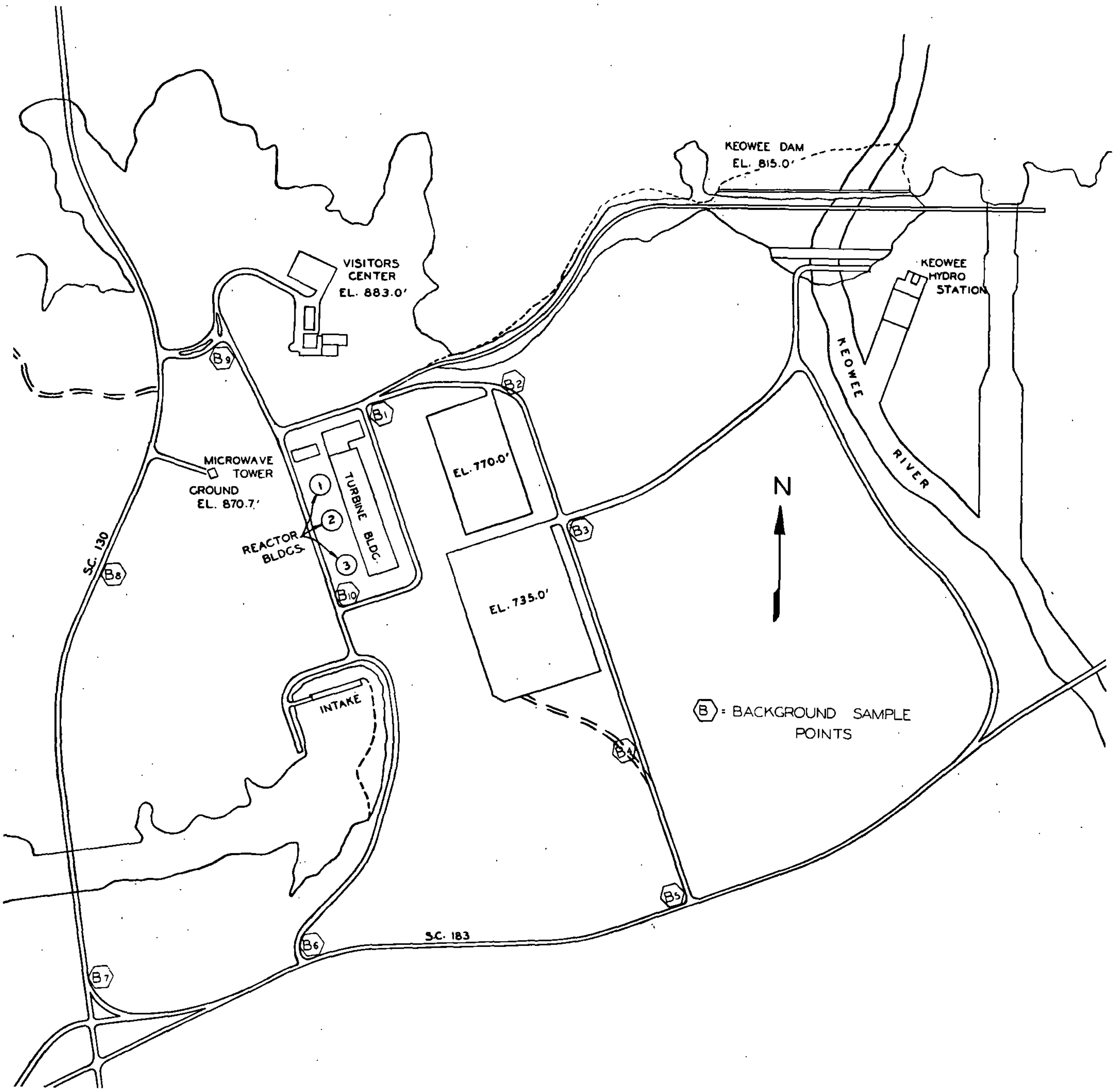
Time (24 hr. clock)	Temperature Ground	Δt_1	Δt_2	Δt_3	Wind Direction	Wind Speed

KEOWEE RIVER STATION NO. 4 DATA

Time (24 hr. clock)	Temperature Ground	Pole Top	Wind Direction	Wind Speed

SF_6 GAS TRACER TEST
TYPICAL LOG SHEETS

OCONEE NUCLEAR STATION
Figure 2A - 4
(New) Rev. 4 4/20/70

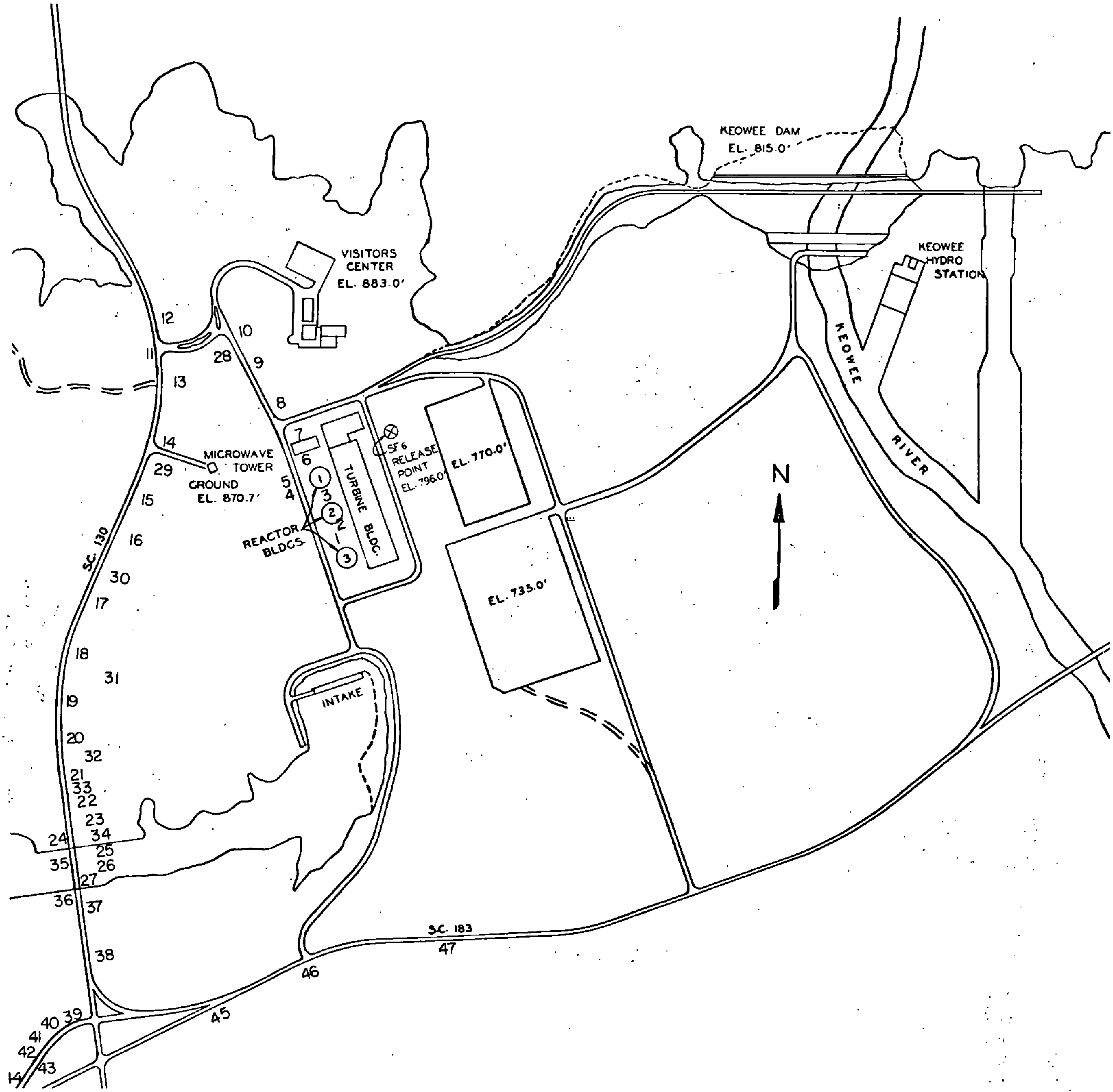


No SF_6 Detected

SF_6 GAS TRACER TEST
BACKGROUND SAMPLE POINTS

CITIZENSHIP DUKE POWER SERVICE

OCONEE NUCLEAR STATION
Figure 2A - 5 Test Date: 1/15/70
(New) Rev. 7 7/9/70



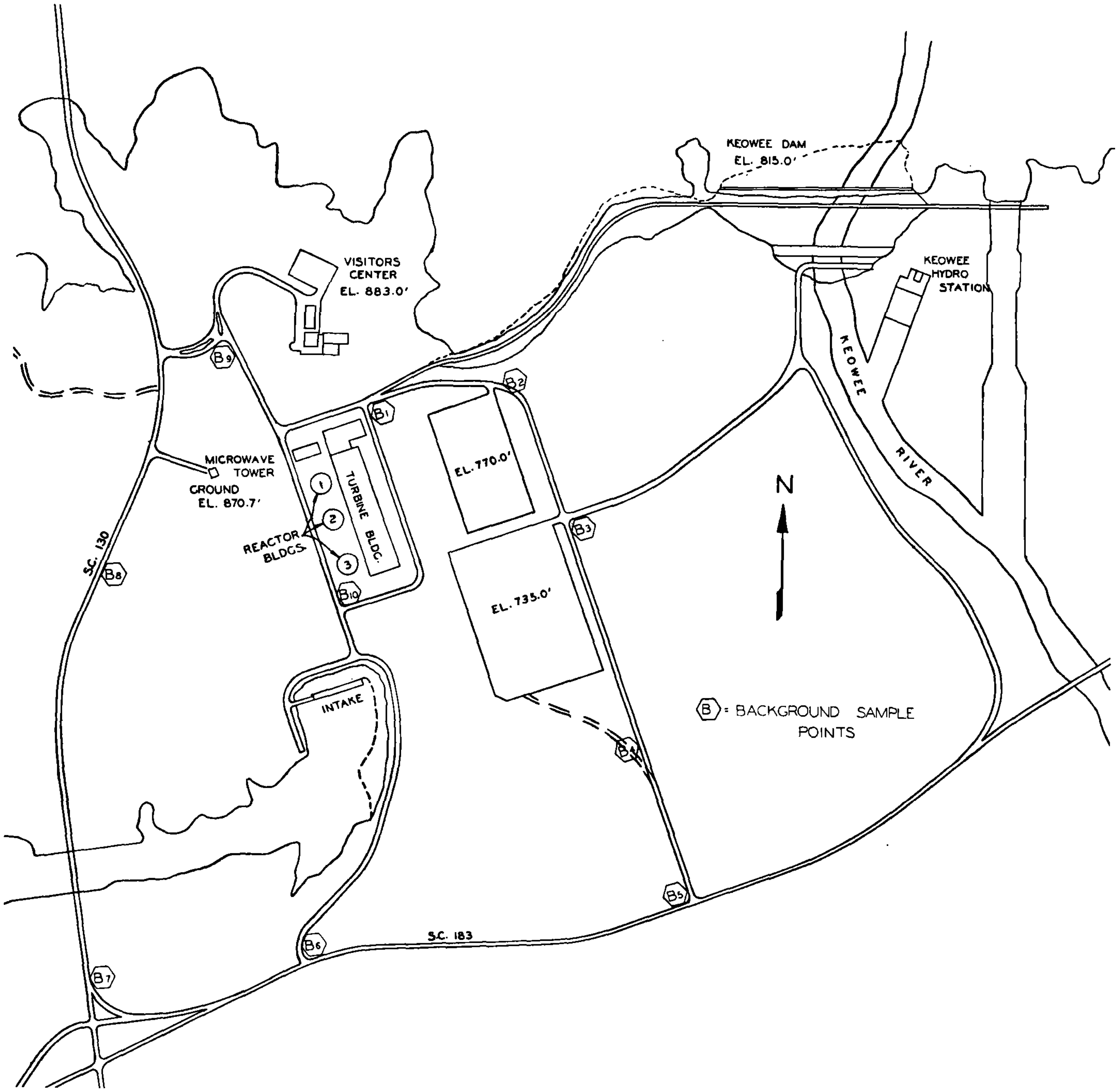
SF$_6$ GAS TRACER TEST

SF$_6$ RELEASE POINT AND SAMPLE STATIONS

OCONEE NUCLEAR STATION

Figure 2A - 6 Test Date: 1/15/70

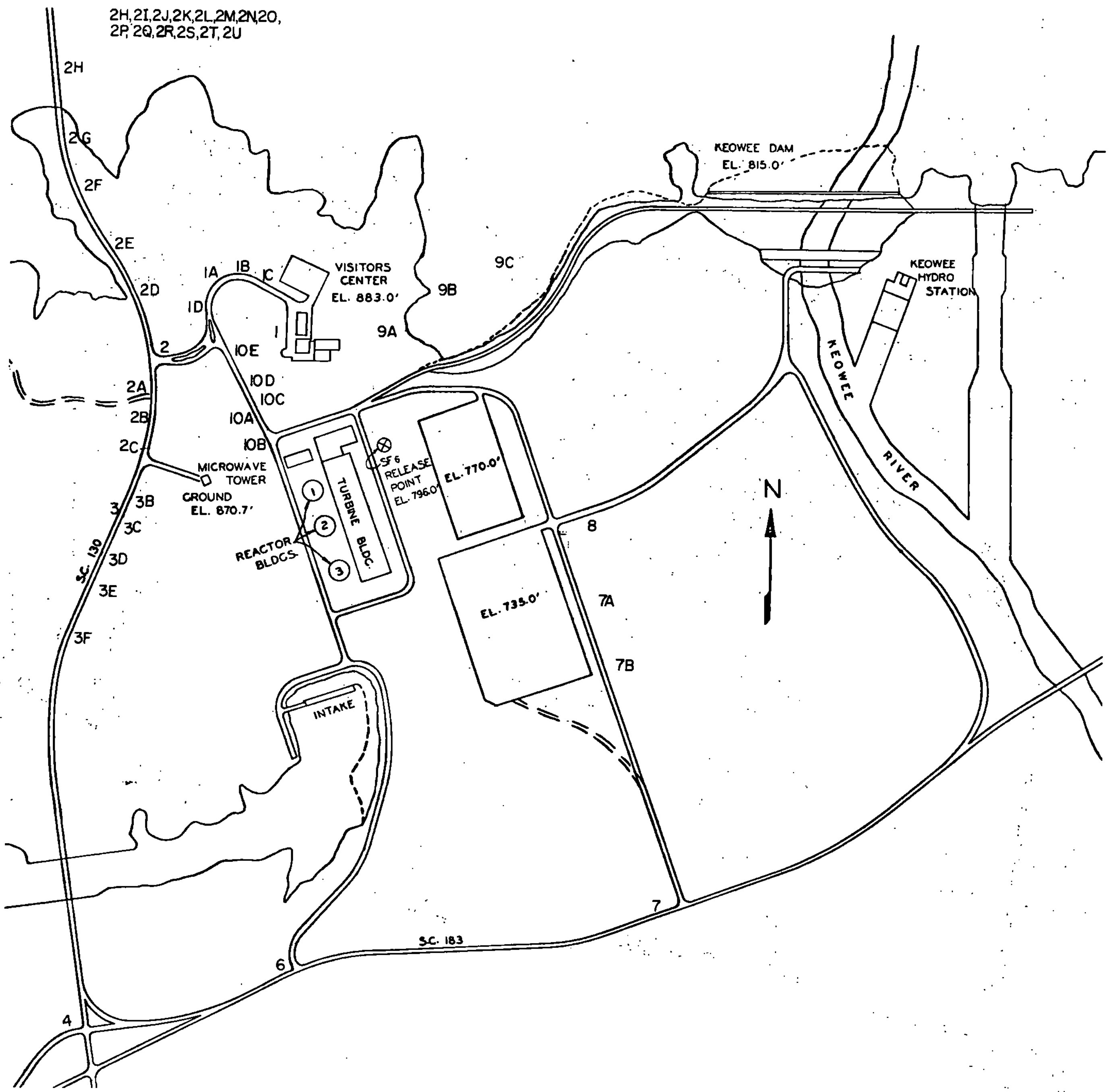
(New) Rev. 7 7/9/70



No SF_6 Detected

SF_6 GAS TRACER TEST
BACKGROUND SAMPLE POINTS

OCONEE NUCLEAR STATION
Figure 2A - 5 Test Date: 1/28/70
(New) Rev. 7 7/9/70



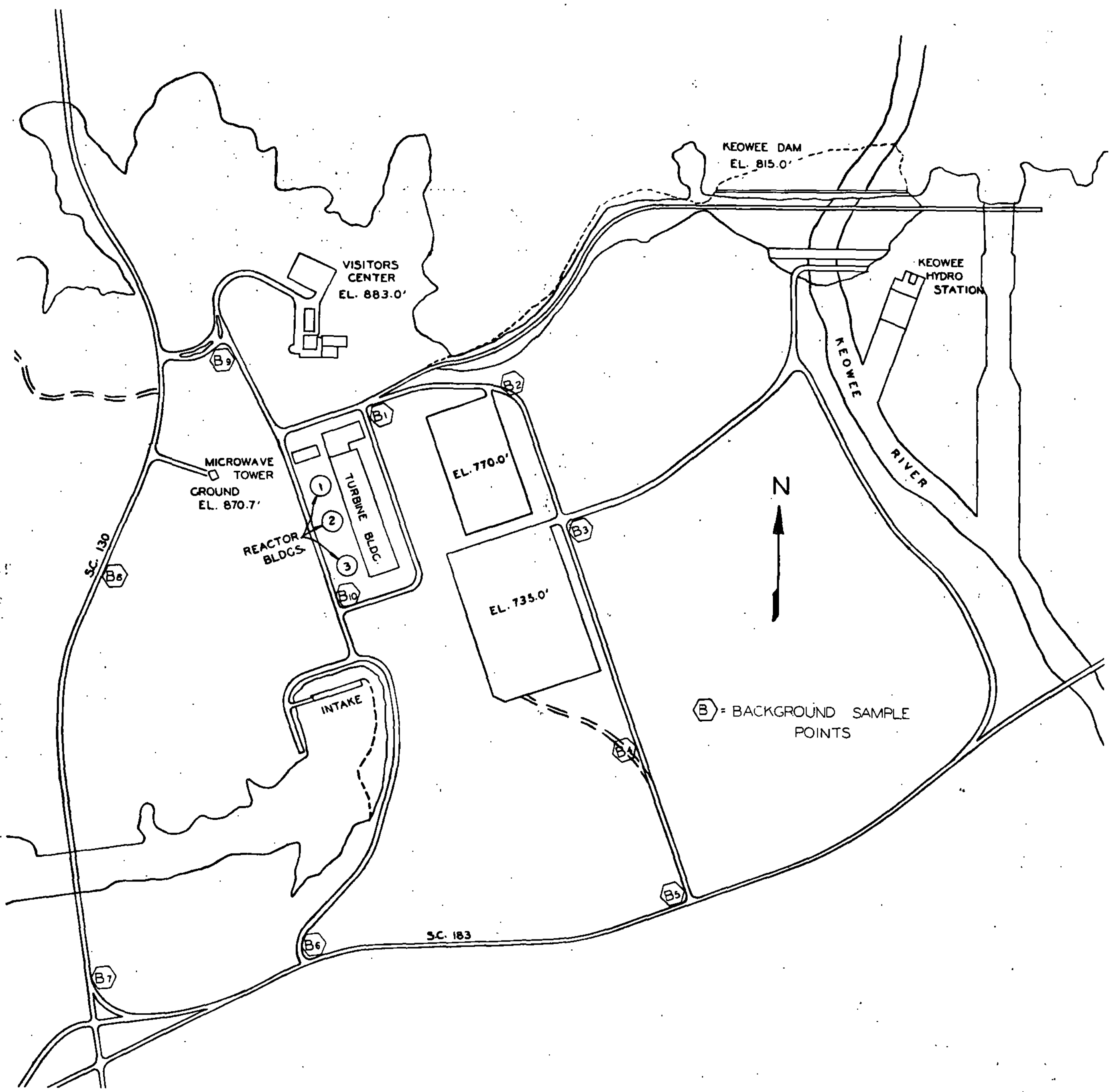
SF_6 GAS TRACER TEST

SF_6 RELEASE POINT AND SAMPLE STATIONS

OCONEE NUCLEAR STATION

Figure 2A - 6 Test Date: 1/28/70

(New) Rev. 7 7/9/70

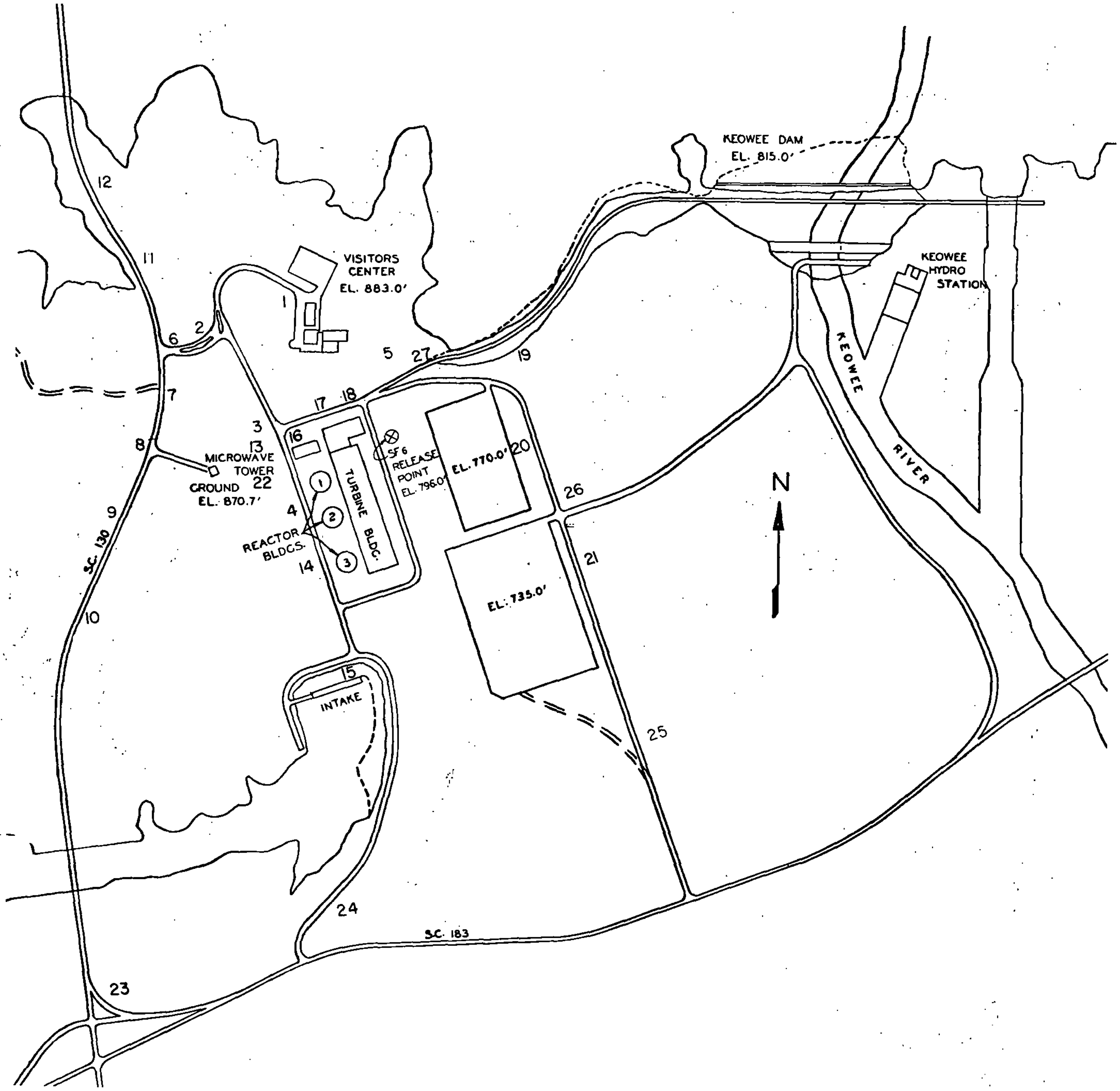


No SF_6 Detected

SF_6 GAS TRACER TEST
BACKGROUND SAMPLE POINTS

OCONEE NUCLEAR STATION

Figure 2A - 5 Test Date: 1/31/70
(New) Rev. 7 7/9/70



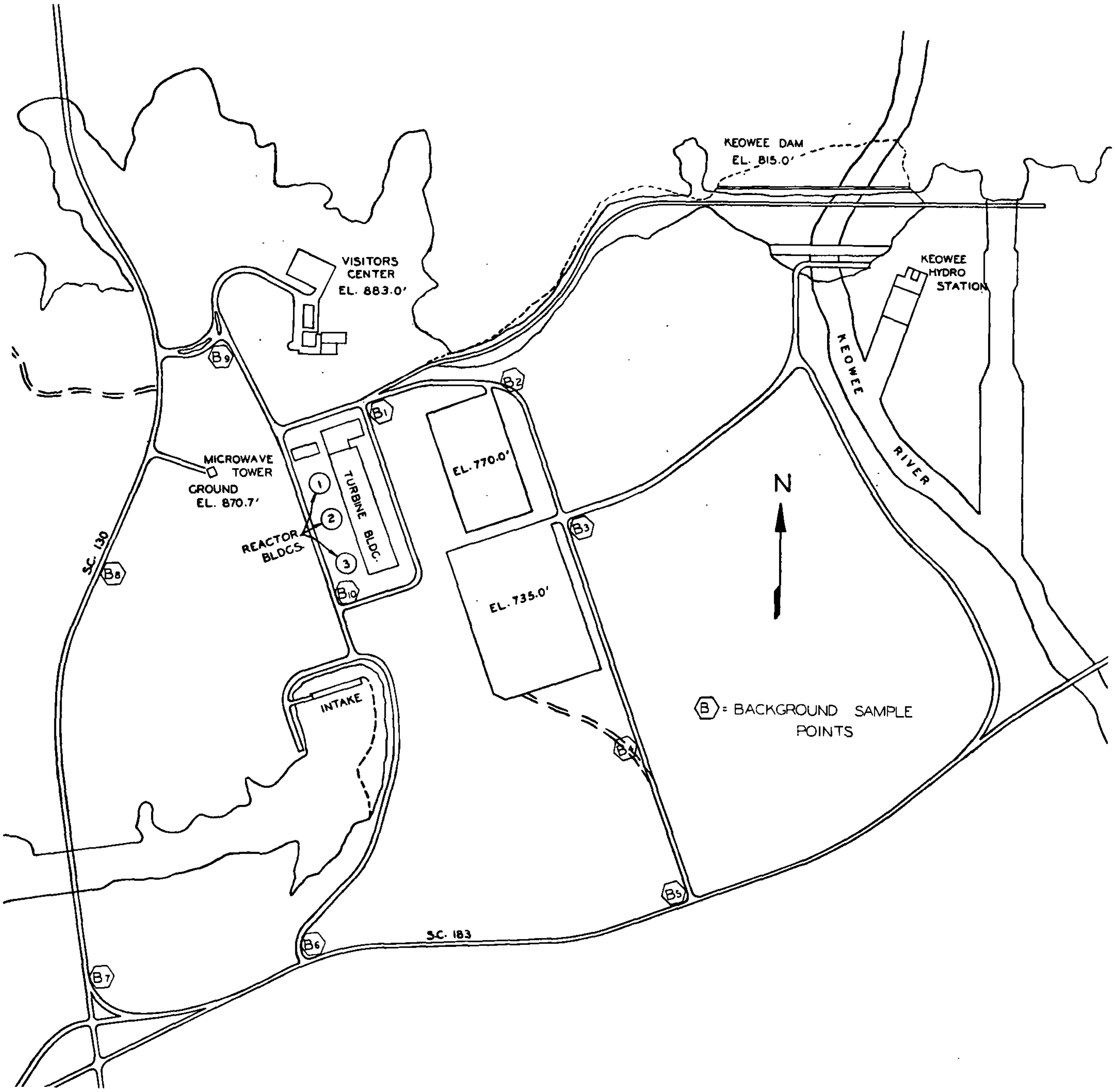
SF_6 GAS TRACER TEST

SF_6 RELEASE POINT AND SAMPLE STATIONS

OCONEE NUCLEAR STATION

Figure 2A - 6 Test Date: 1/31/70

(New) Rev. 7 7/9/70

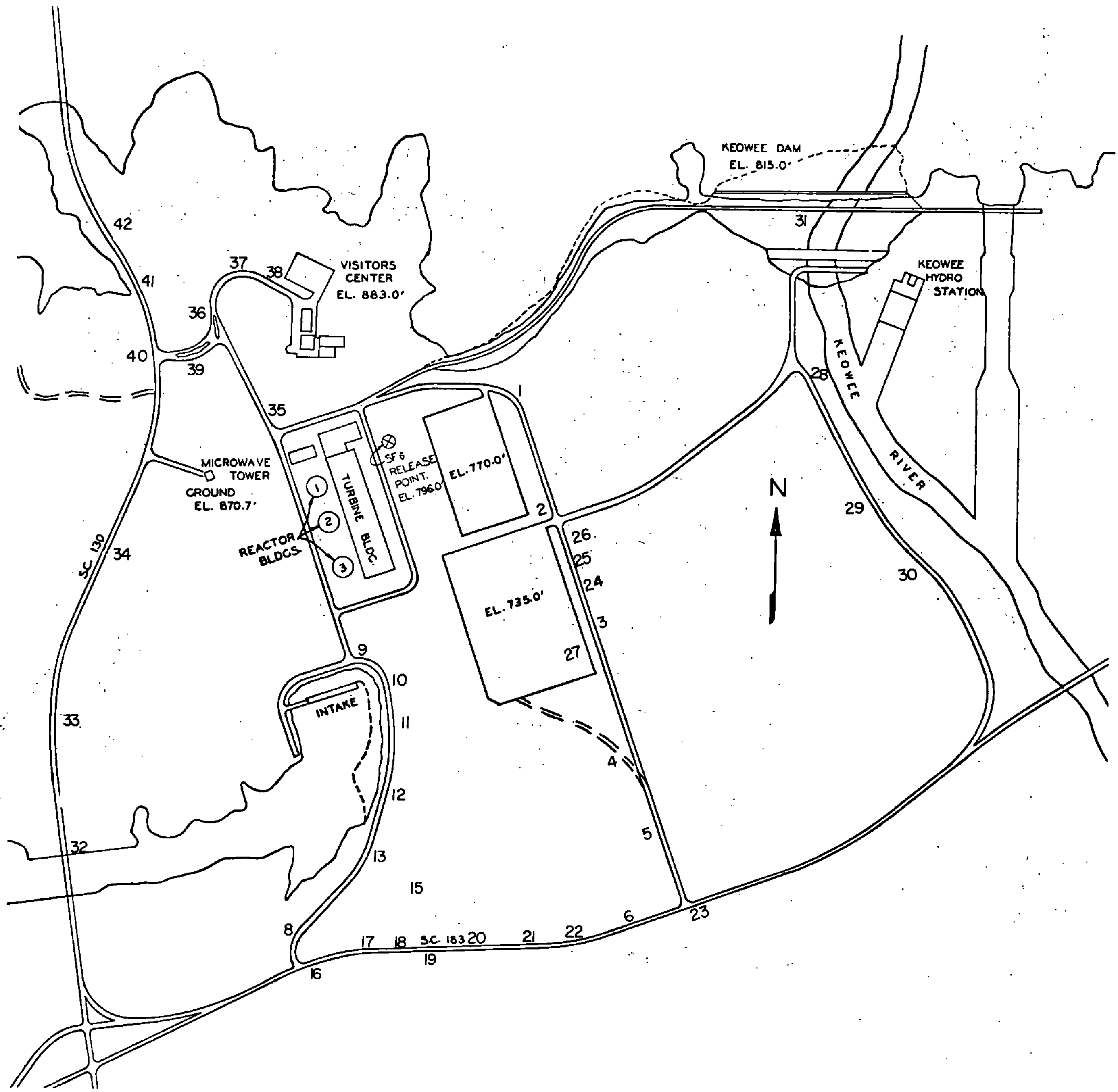


No SF_6 Detected

SF_6 GAS TRACER TEST
BACKGROUND SAMPLE POINTS

OCONEE NUCLEAR STATION

Figure 2A - 5 Test Date: 2/5/70
(New) Rev. 7 7/9/70



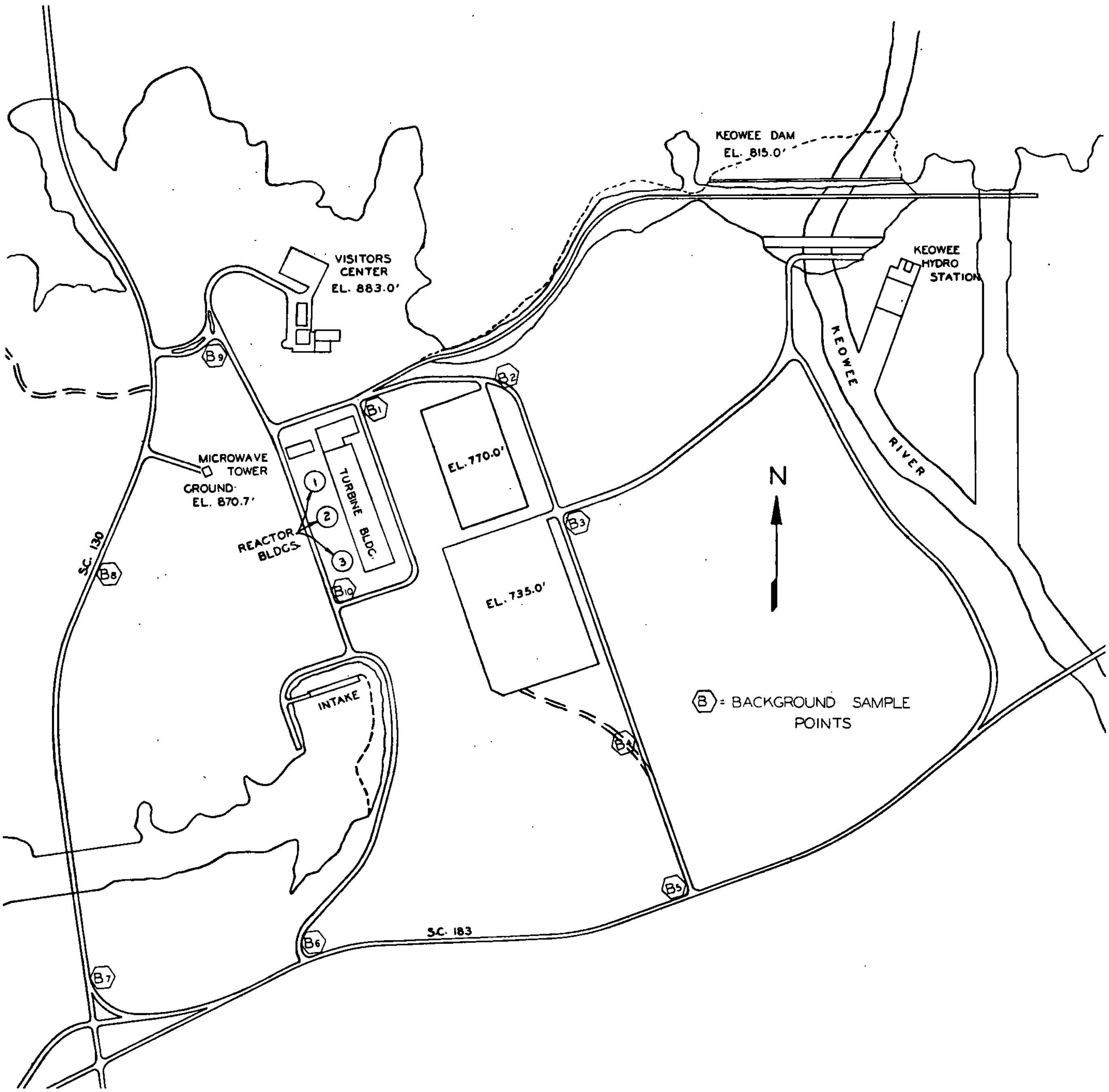
SF_6 GAS TRACER TEST

SF_6 RELEASE POINT AND SAMPLE STATIONS

OCONEE NUCLEAR STATION

Figure 2A - 6 Test Date: 2/5/70

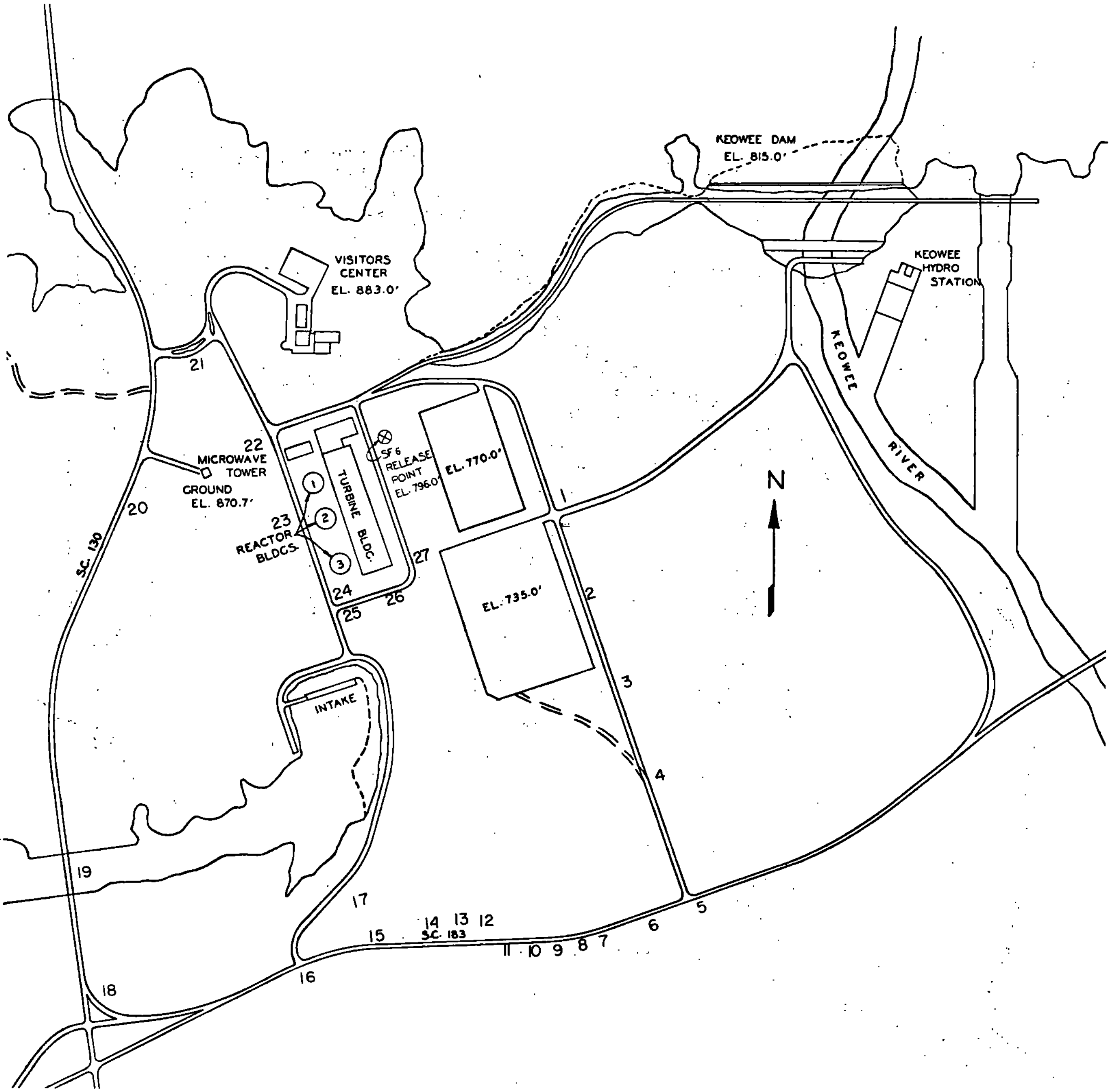
(New) Rev. 7 7/9/70



No SF_6 Detected

SF_6 GAS TRACER TEST
BACKGROUND SAMPLE POINTS

OCONEE NUCLEAR STATION
Figure 2A - 5 Test Date: 2/6/70
(New) Rev. 7 7/9/70



SF$_6$ GAS TRACER TEST

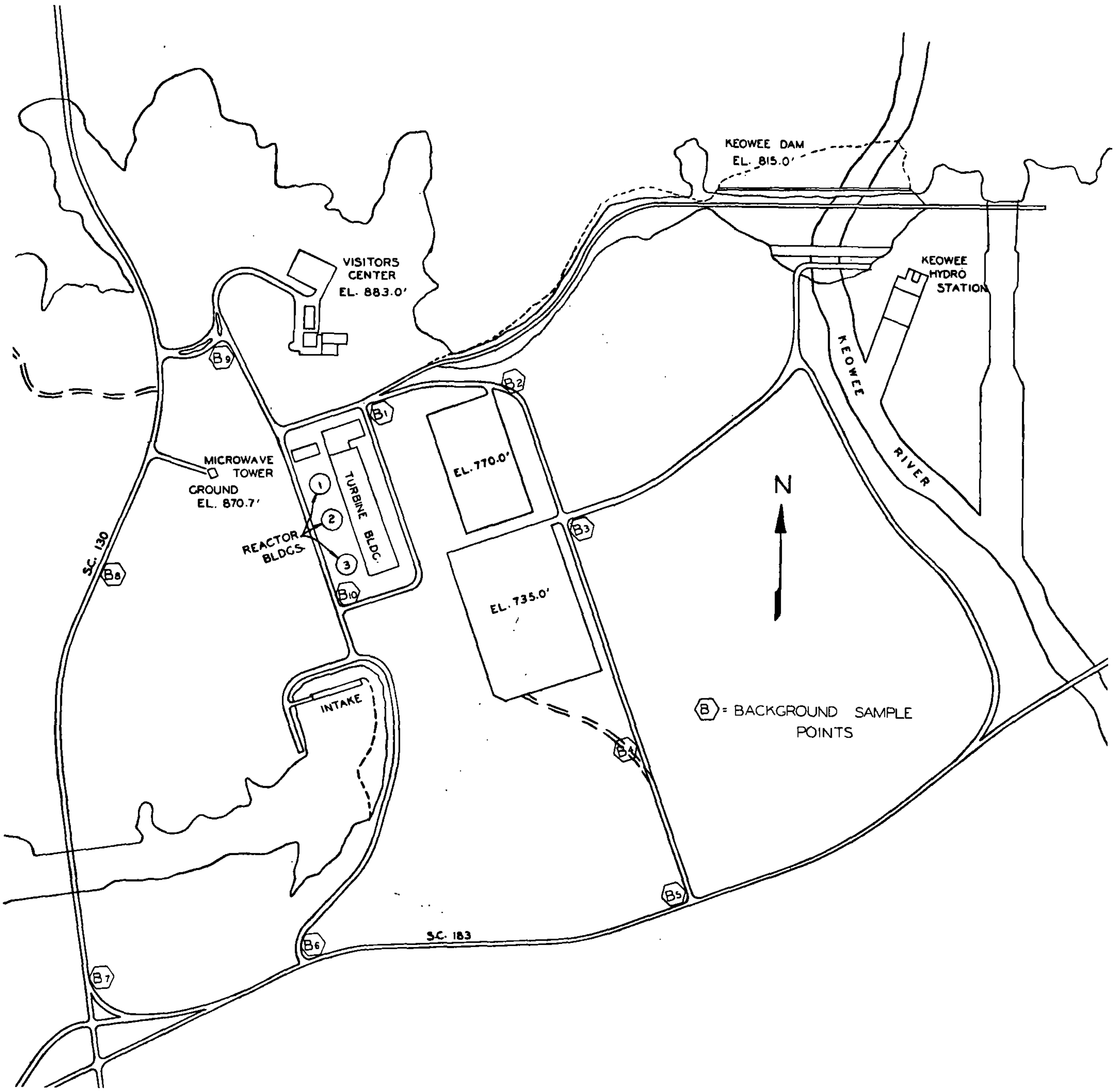
SF$_6$ RELEASE POINT AND SAMPLE STATIONS

DUKE POWER CITIZENSHIP SERVICE

OCONEE NUCLEAR STATION

Figure 2A - 6 Test Date: 2/6/70

(New) Rev. 7 7/9/70



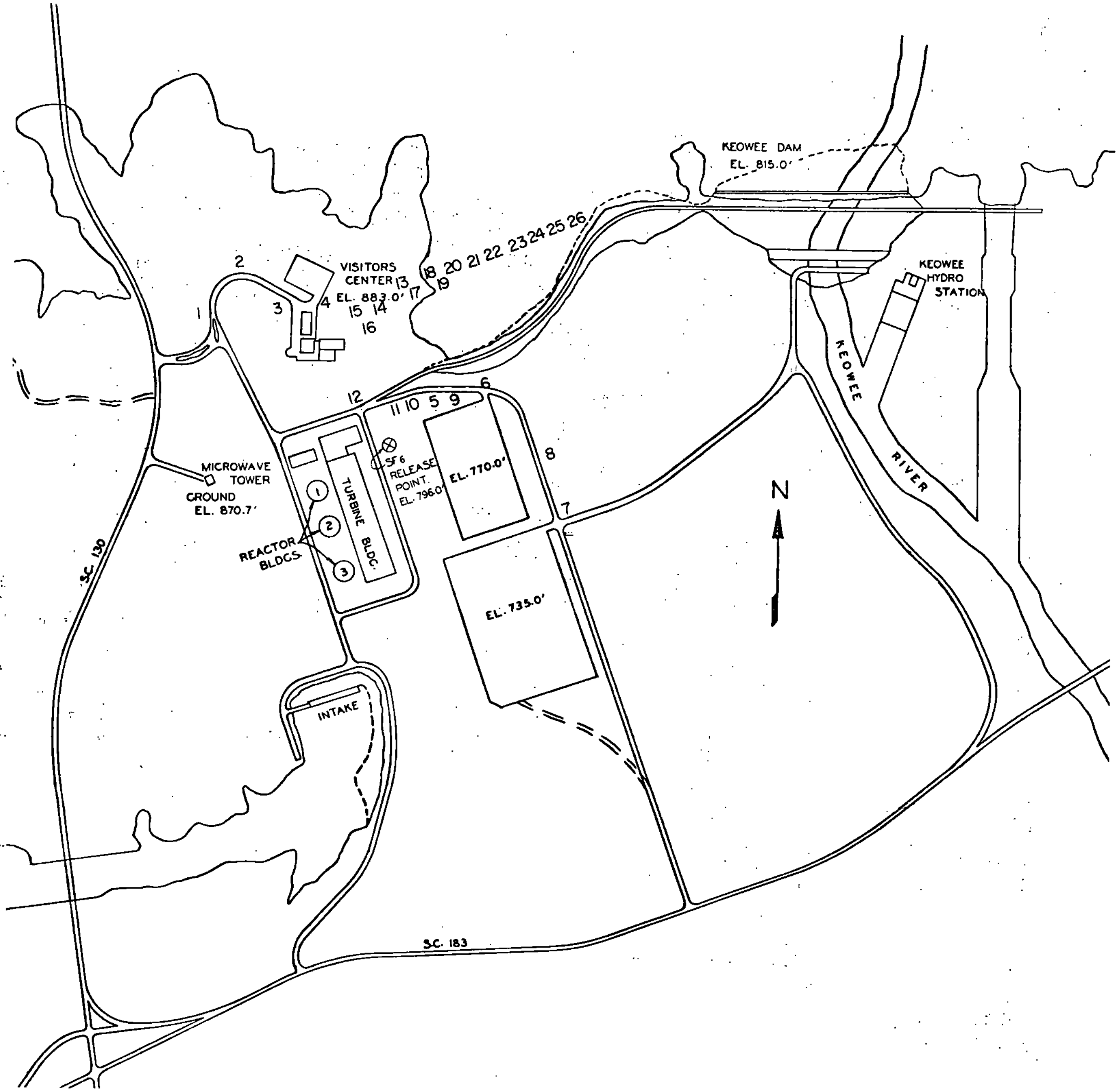
No SF_6 Detected

SF_6 GAS TRACER TEST
BACKGROUND SAMPLE POINTS

OCONEE NUCLEAR STATION

Figure 2A - 5 Test Date: 2/10/70

(New) Rev. 7 7/9/70



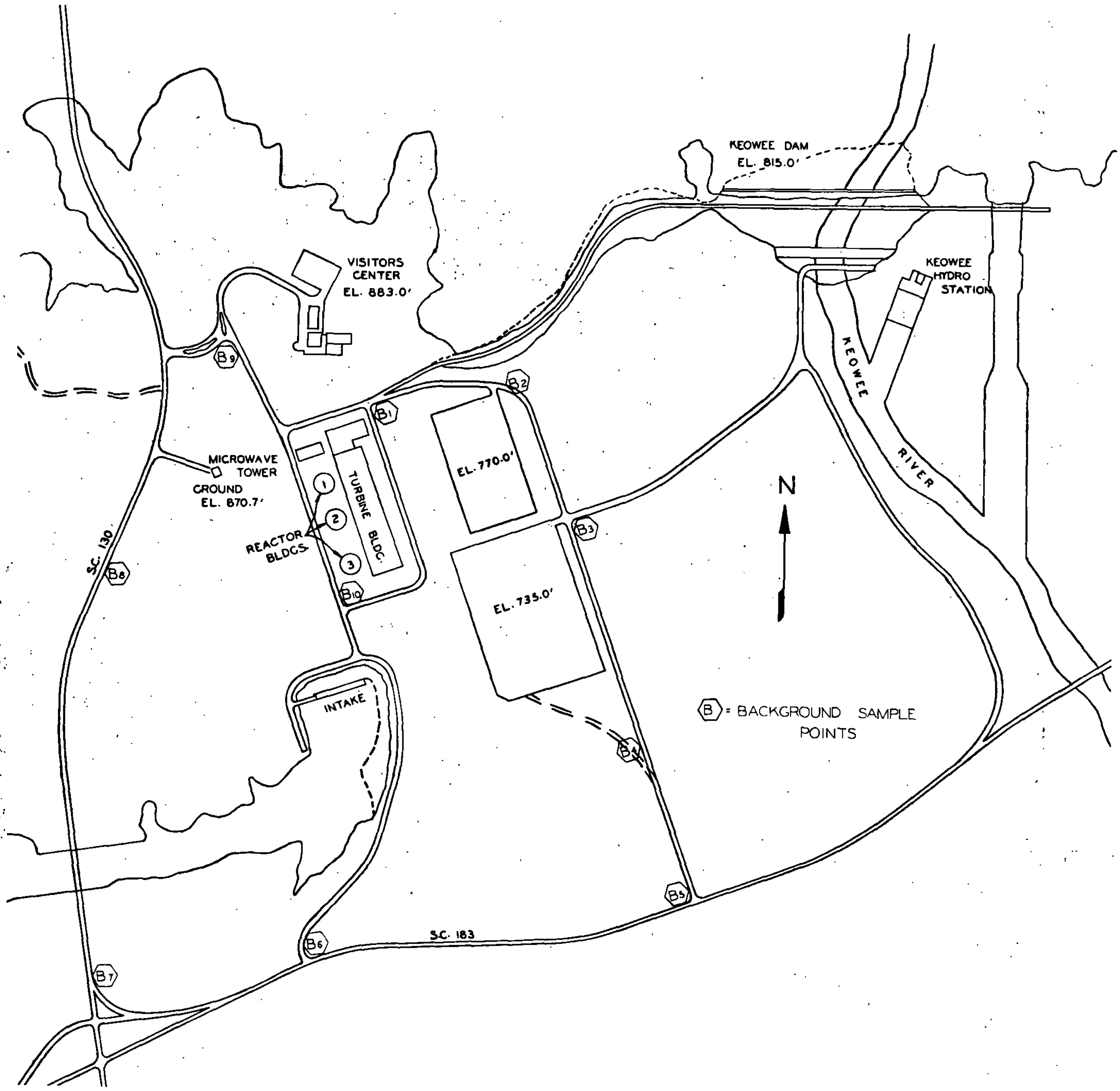
SF$_6$ GAS TRACER TEST

SF$_6$ RELEASE POINT AND SAMPLE STATIONS

OCONEE NUCLEAR STATION

Figure 2A - 6 Test Date: 2/10/70

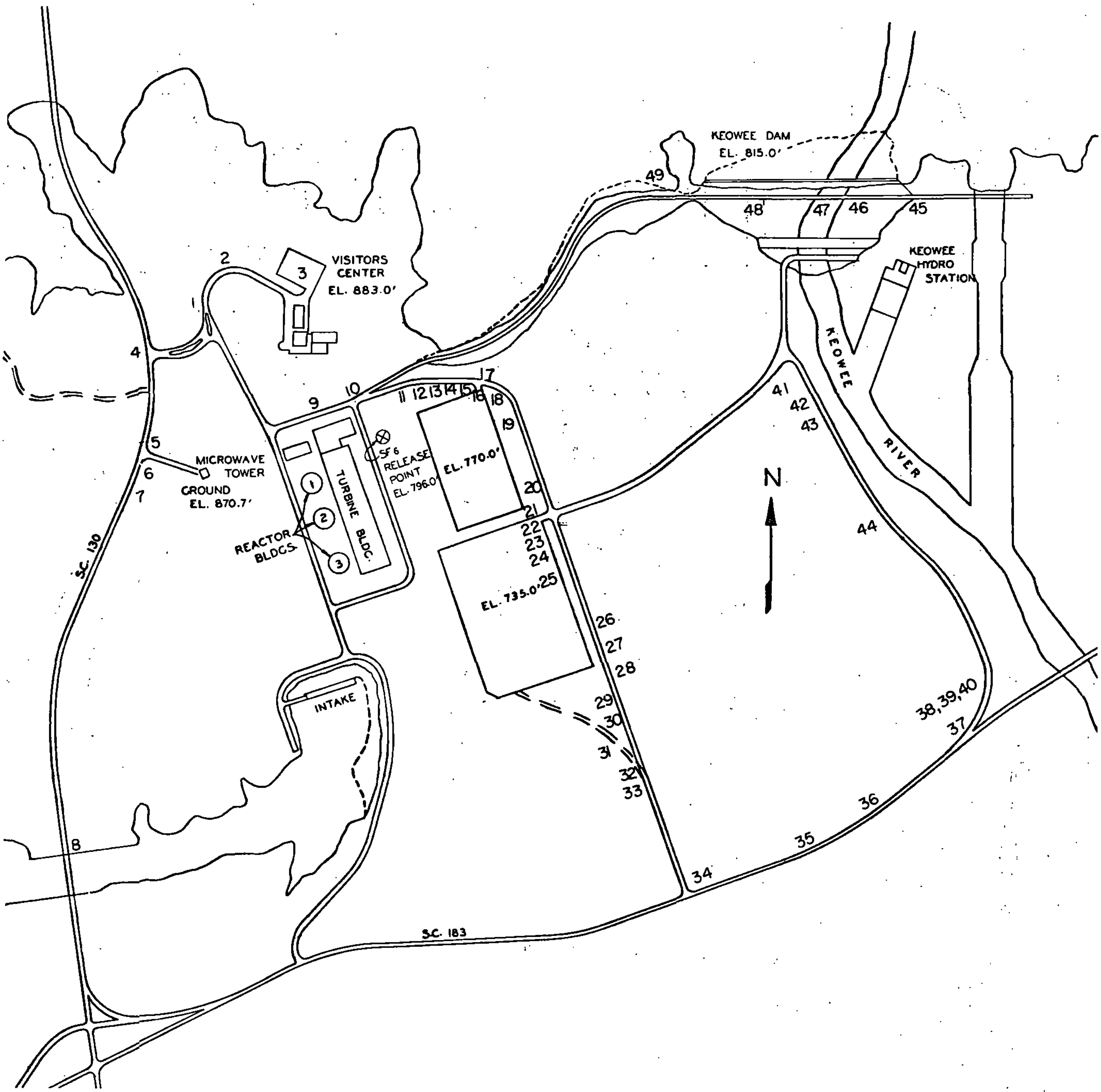
(New) Rev. 7 7/9/70



No SF_6 Detected

SF_6 GAS TRACER TEST
BACKGROUND SAMPLE POINTS

OCONEE NUCLEAR STATION
Figure 2A - 5 Test Date: 2/11/70
(New) Rev. 7 7/9/70



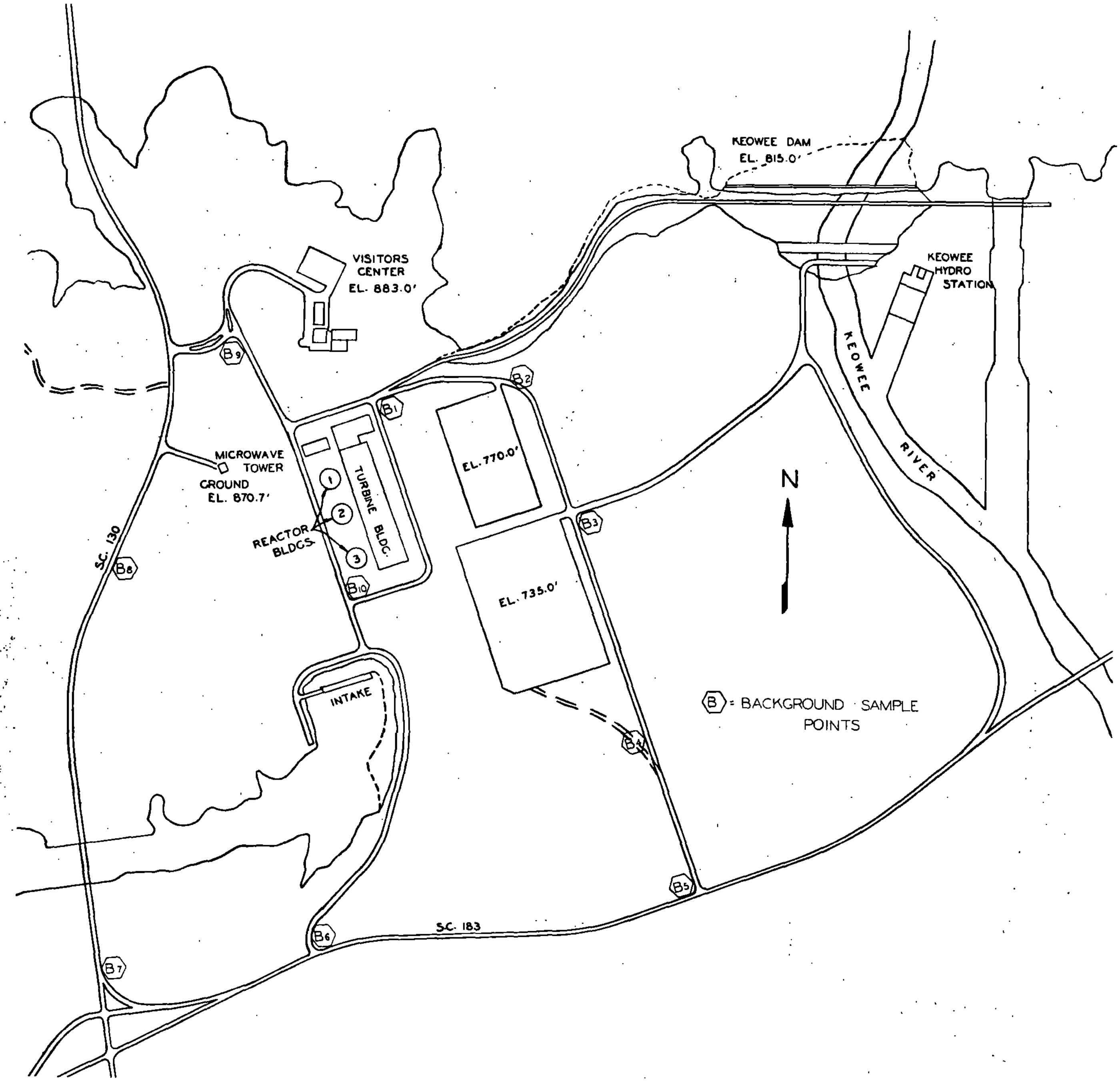
SF$_6$ GAS TRACER TEST

SF$_6$ RELEASE POINT AND SAMPLE STATIONS

OCONEE NUCLEAR STATION

Figure 2A - 6 Test Date: 2/11/70

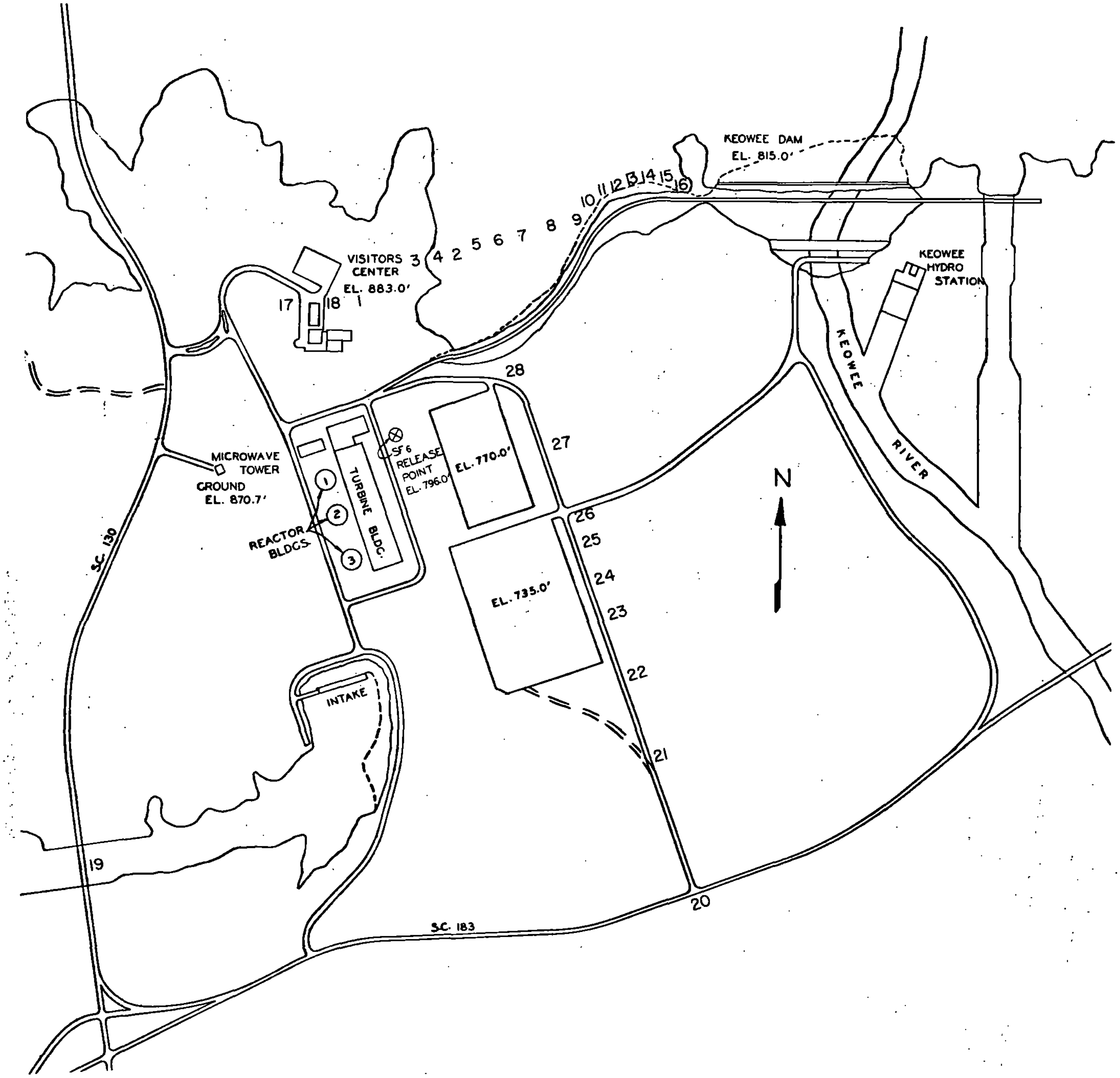
(New) Rev. 7 7/9/70



No SF_6 Detected

SF_6 GAS TRACER TEST
BACKGROUND SAMPLE POINTS

OCONEE NUCLEAR STATION
Figure 2A - 5 Test Date: 2/12/70
(New) Rev. 7 7/9/70



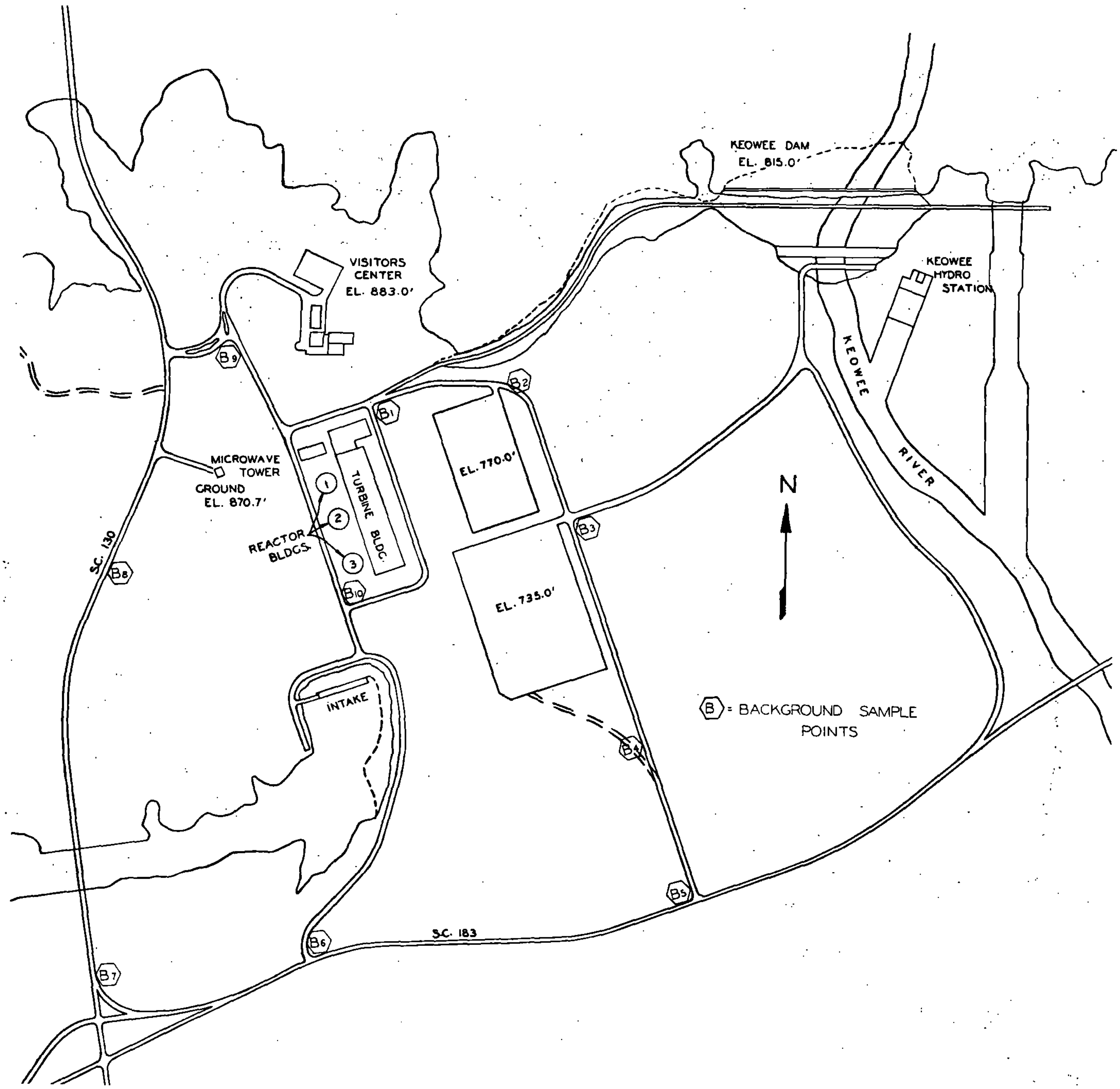
SF_6 GAS TRACER TEST

SF_6 RELEASE POINT AND SAMPLE STATIONS

OCONEE NUCLEAR STATION

Figure 2A - 6 Test Date: 2/12/70

(New) Rev. 7 7/9/70

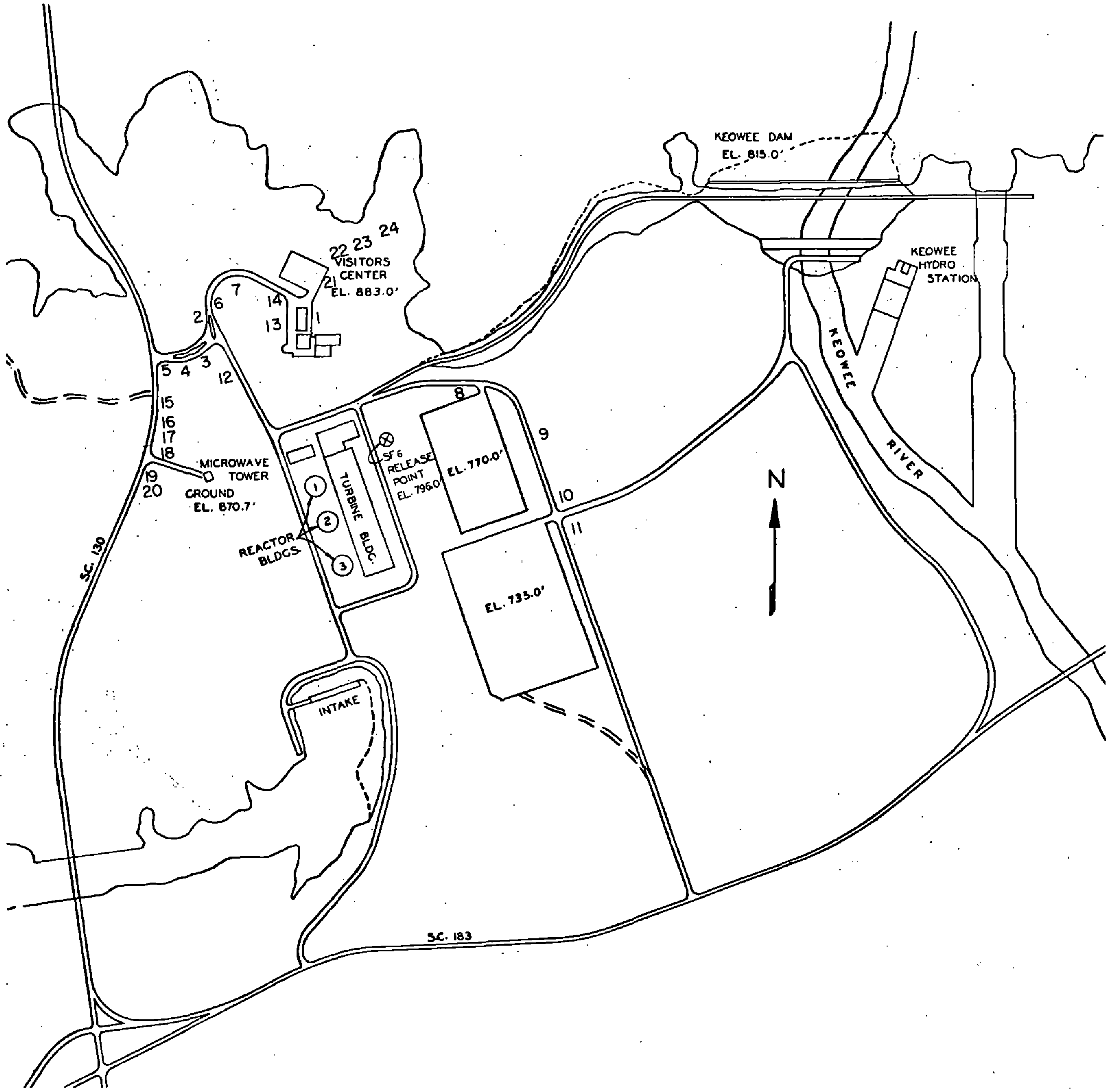


No SF_6 Detected

SF_6 GAS TRACER TEST
BACKGROUND SAMPLE POINTS

OCONEE NUCLEAR STATION

Figure 2A - 5 Test Date: 2/17/70
(New) Rev. 7 7/9/70



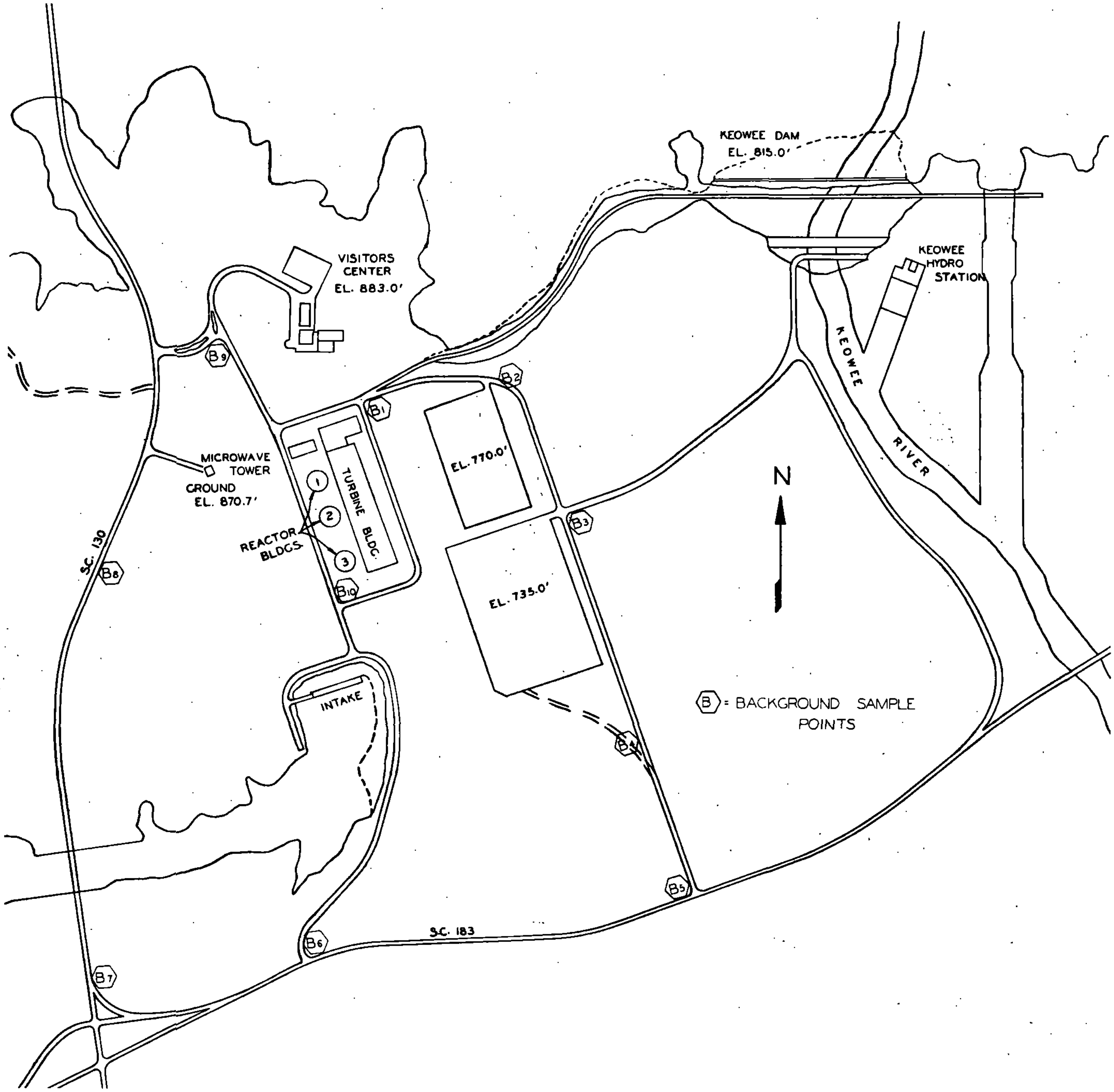
SF_6 GAS TRACER TEST

SF_6 RELEASE POINT AND SAMPLE STATIONS

OCONEE NUCLEAR STATION

Figure 2A - 6 Test Date: 2/17/70

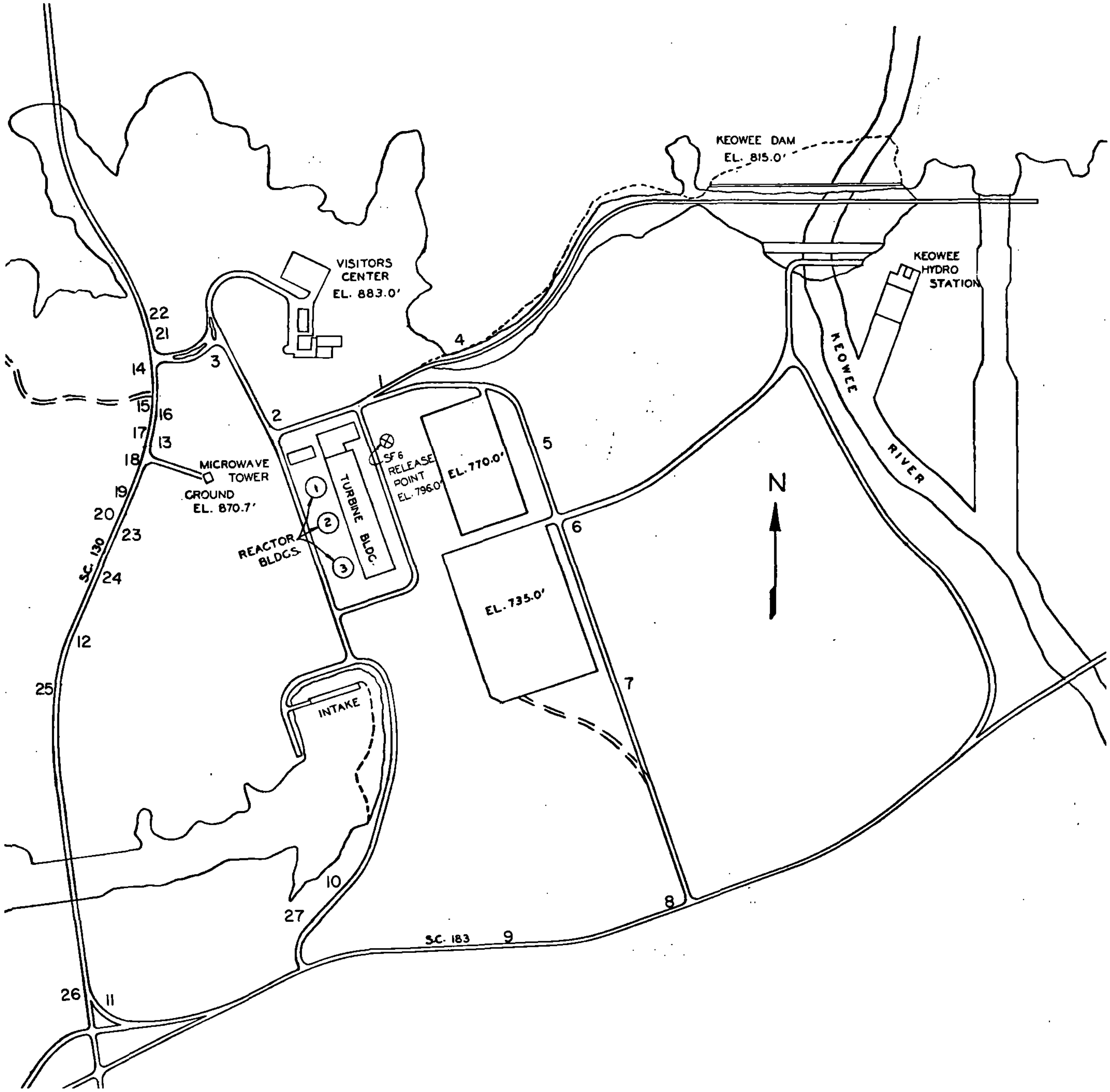
(New) Rev. 7 7/9/70



No SF_6 Detected

SF_6 GAS TRACER TEST
BACKGROUND SAMPLE POINTS

OCONEE NUCLEAR STATION
Figure 2A - 5 Test Date: 3/2/70
(New) Rev. 7 7/9/70



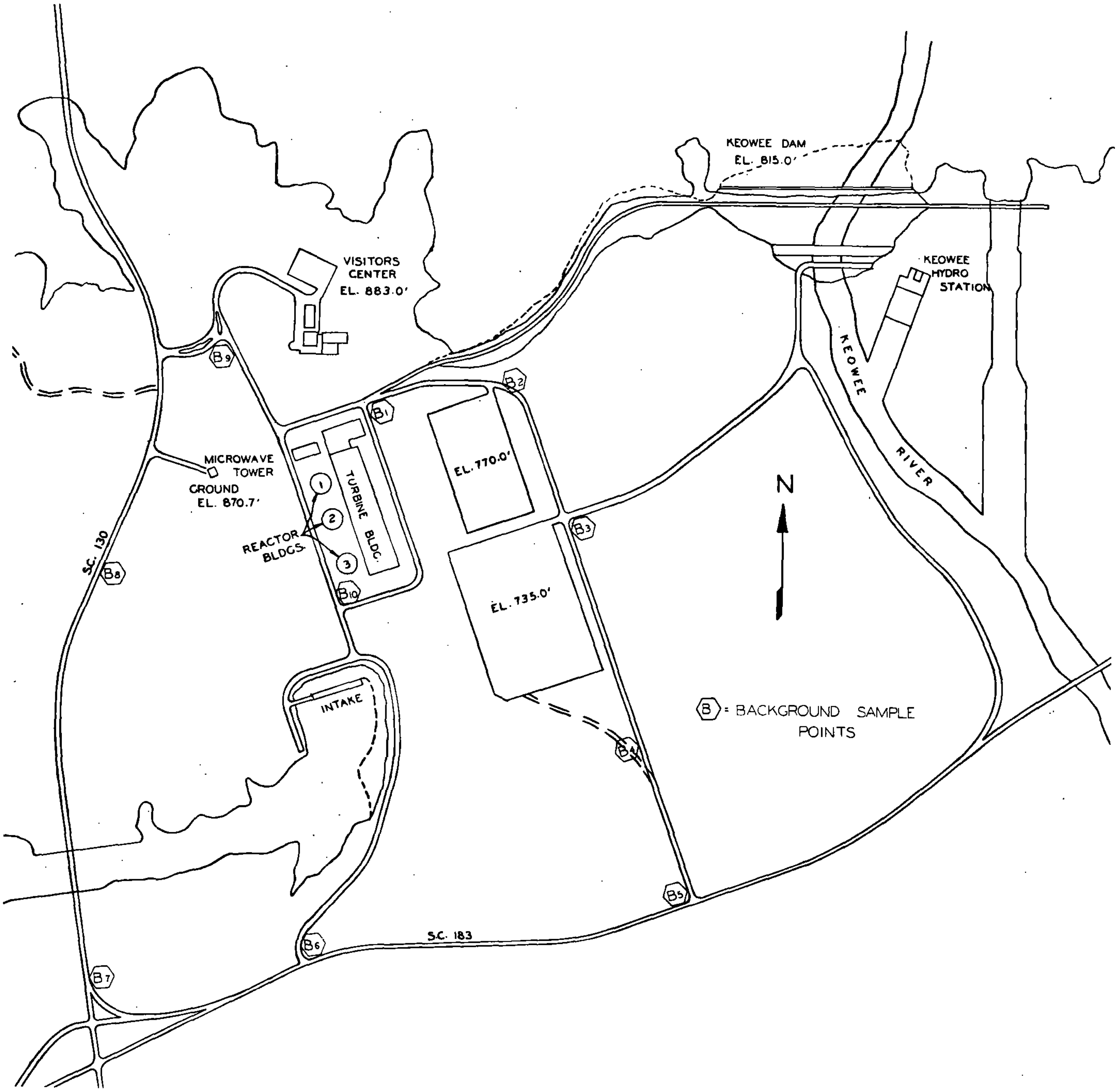
SF_6 GAS TRACER TEST

SF_6 RELEASE POINT AND SAMPLE STATIONS

OCONEE NUCLEAR STATION

Figure 2A - 6 Test Date: 3/2/70

(New) Rev. 7 7/9/70

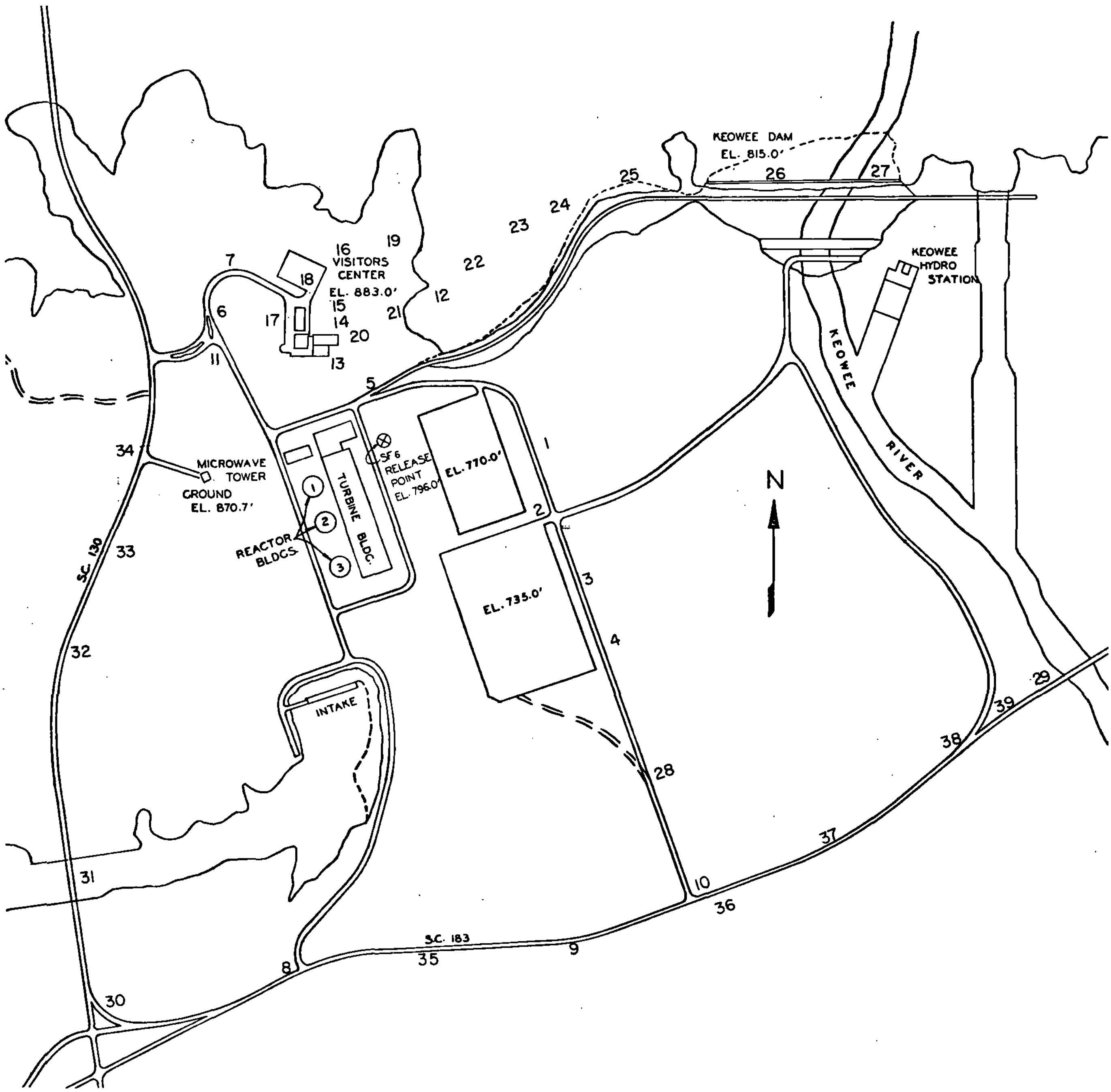


No SF_6 Detected

SF_6 GAS TRACER TEST
BACKGROUND SAMPLE POINTS

OCONEE NUCLEAR STATION

Figure 2A - 5 Test Date: 2/19/70
(New) Rev. 7 7/9/70



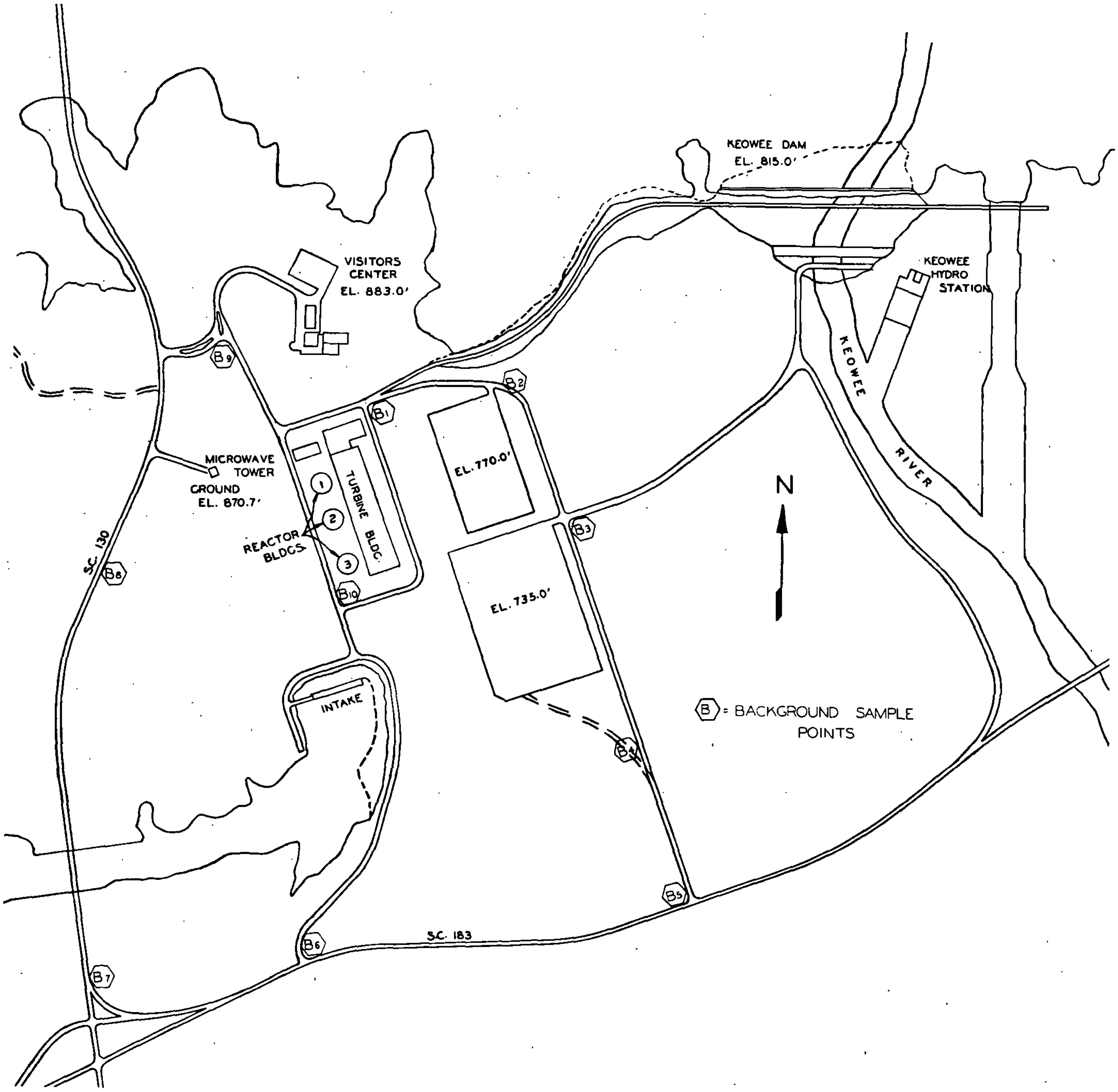
SF$_6$ GAS TRACER TEST

SF$_6$ RELEASE POINT AND SAMPLE STATIONS

OCONEE NUCLEAR STATION

Figure 2A - 6 Test Date: 2/19/70

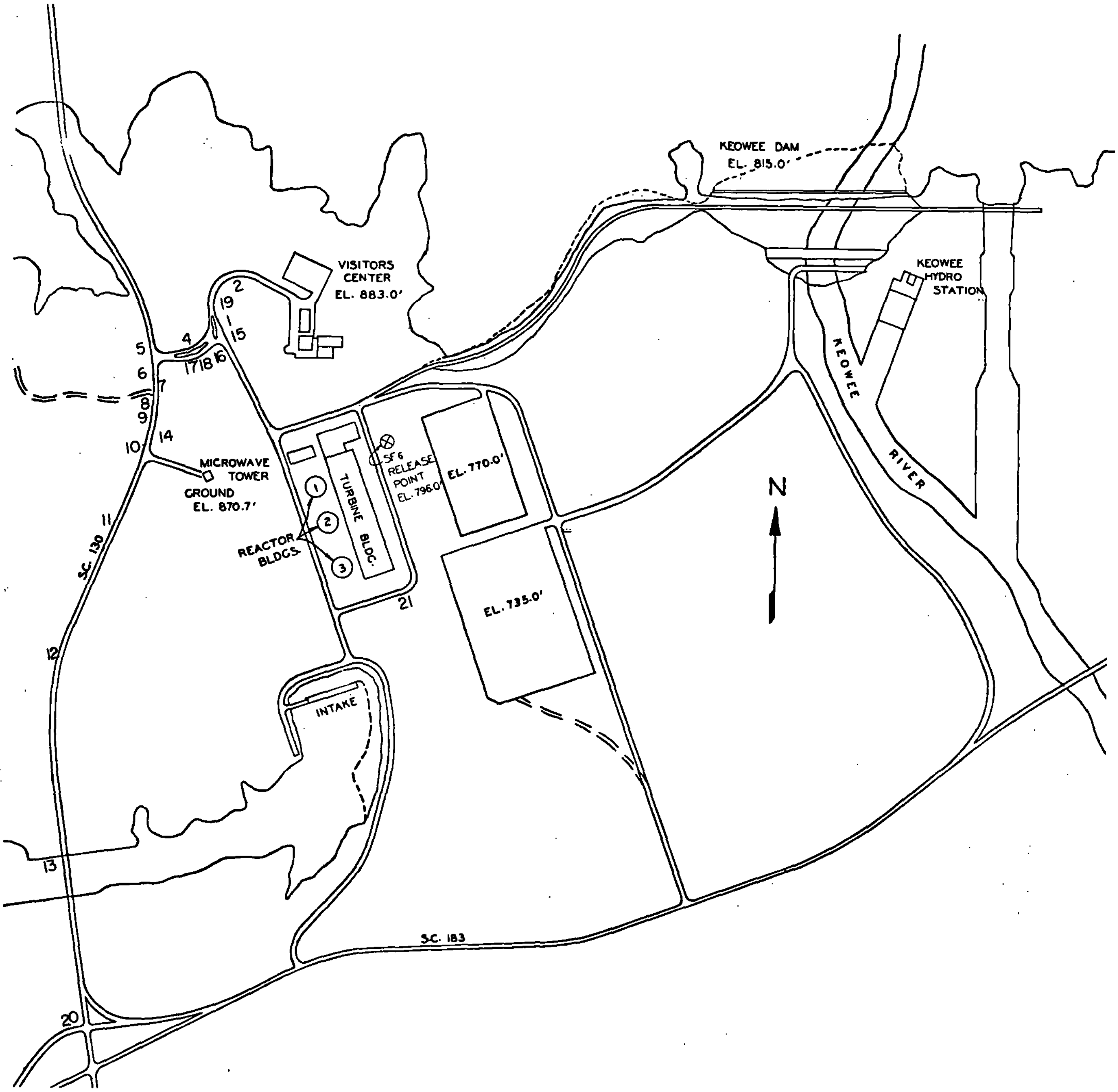
(New) Rev. 7 7/9/70



No SF_6 Detected

SF_6 GAS TRACER TEST
BACKGROUND SAMPLE POINTS

OCONEE NUCLEAR STATION
Figure 2A - 5 Test Date: 3/3/70
(New) Rev. 7 7/9/70



SF_6 GAS TRACER TEST

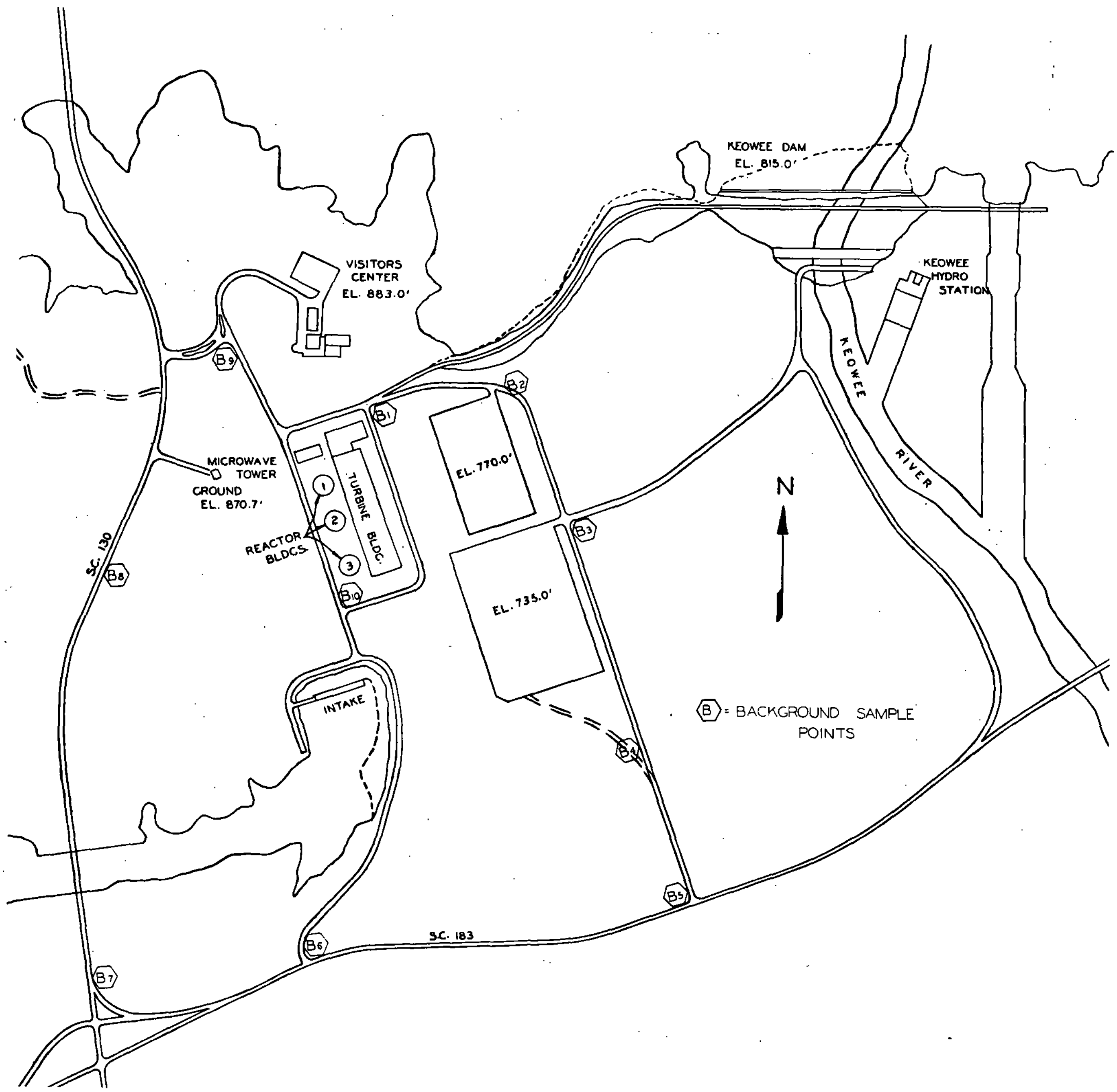
SF_6 RELEASE POINT AND SAMPLE STATIONS

CITIZENSHIP DUKE POWER SERVICE

OCONEE NUCLEAR STATION

Figure 2A - 6 Test Date: 3/3/70

(New) Rev. 7 7/9/70

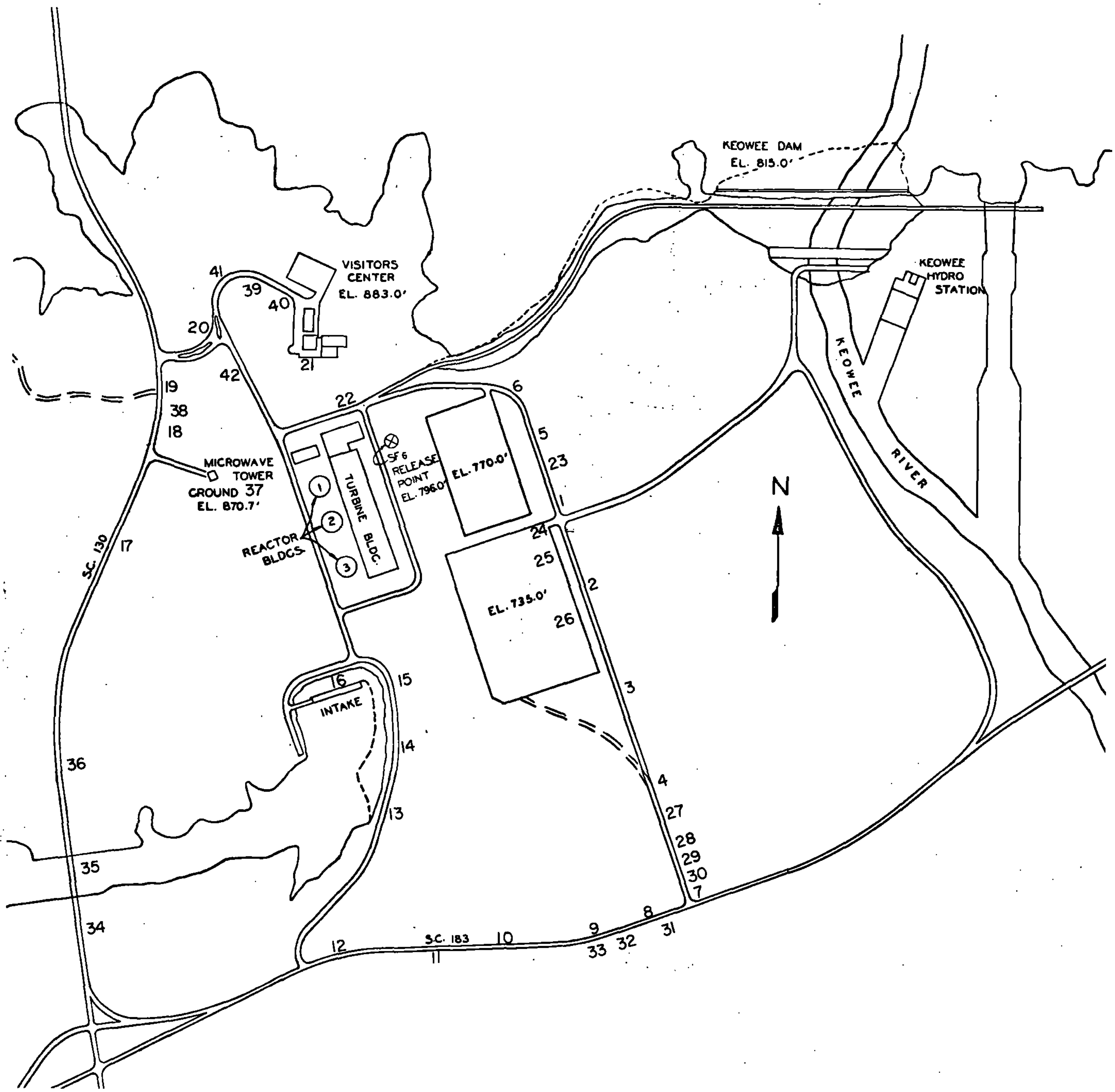


No SF_6 Detected

SF_6GAS TRACER TEST
BACKGROUND SAMPLE POINTS

OCONEE NUCLEAR STATION

Figure 2A - 5 Test Date: 3/5/70
(New) Rev. 7 7/9/70



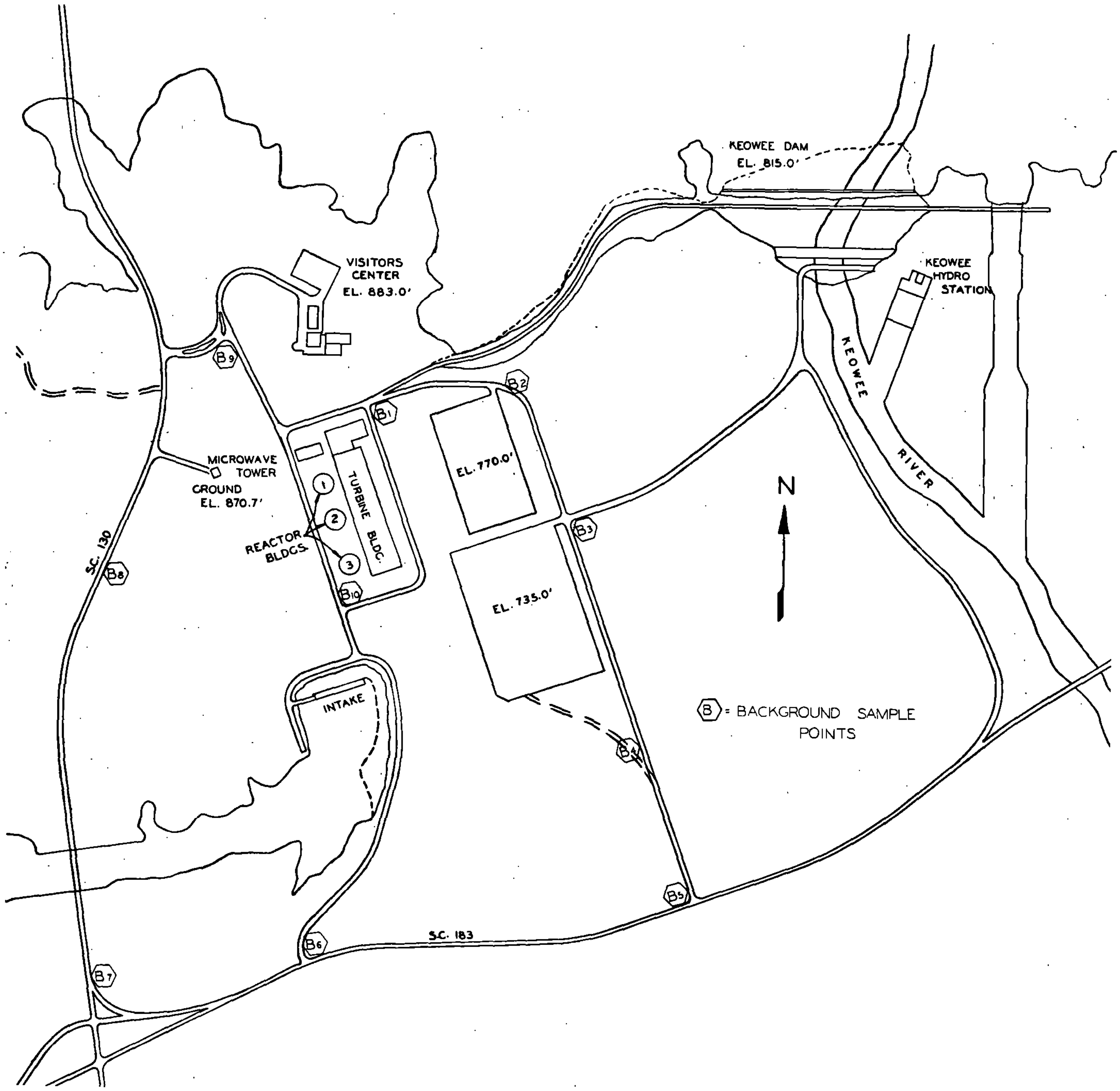
SF$_6$ GAS TRACER TEST

SF$_6$ RELEASE POINT AND SAMPLE STATIONS

OCONEE NUCLEAR STATION

Figure 2A - 6 Test Date: 3/5/70

(New) Rev. 7 7/9/70



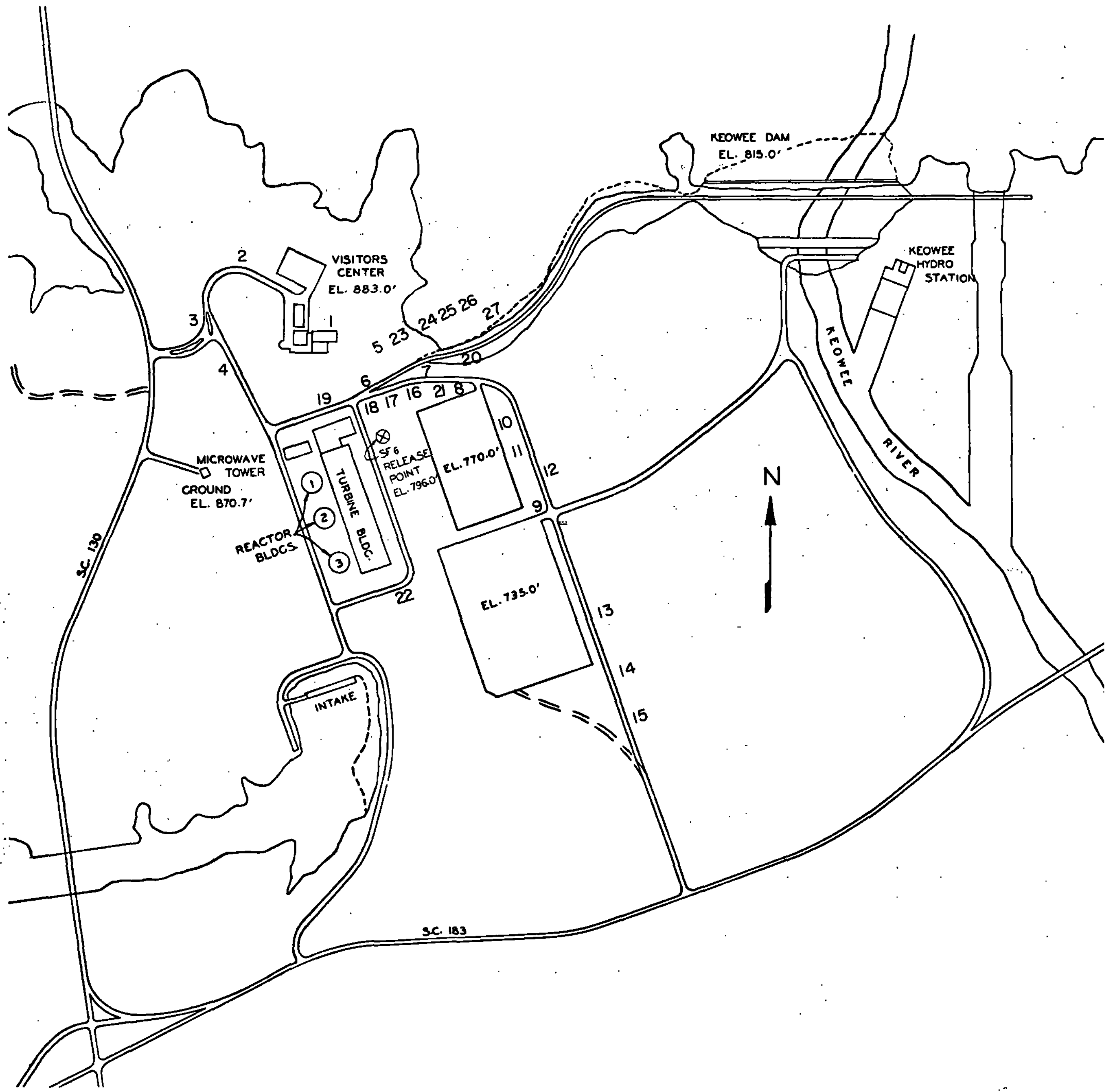
No SF_6 Detected

SF_6 GAS TRACER TEST
BACKGROUND SAMPLE POINTS

OCONEE NUCLEAR STATION

Figure 2A - 5 Test Date:3/10/70

(New) Rev. 7 7/9/70



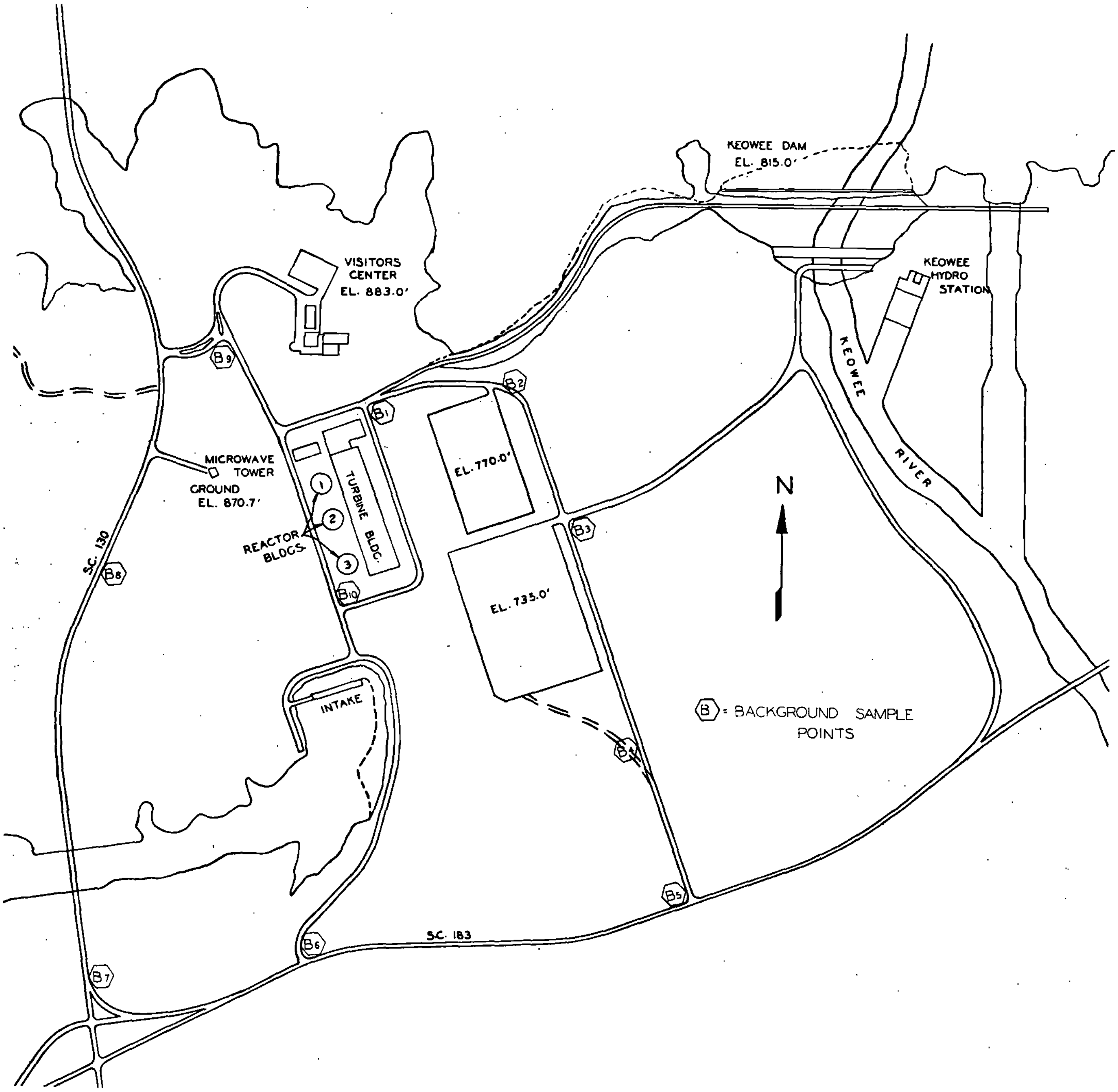
SF_6 GAS TRACER TEST

SF_6 RELEASE POINT AND SAMPLE STATIONS

OCONEE NUCLEAR STATION

Figure 2A - 6 Test Date: 3/10/70

(New) Rev. 7 7/9/70



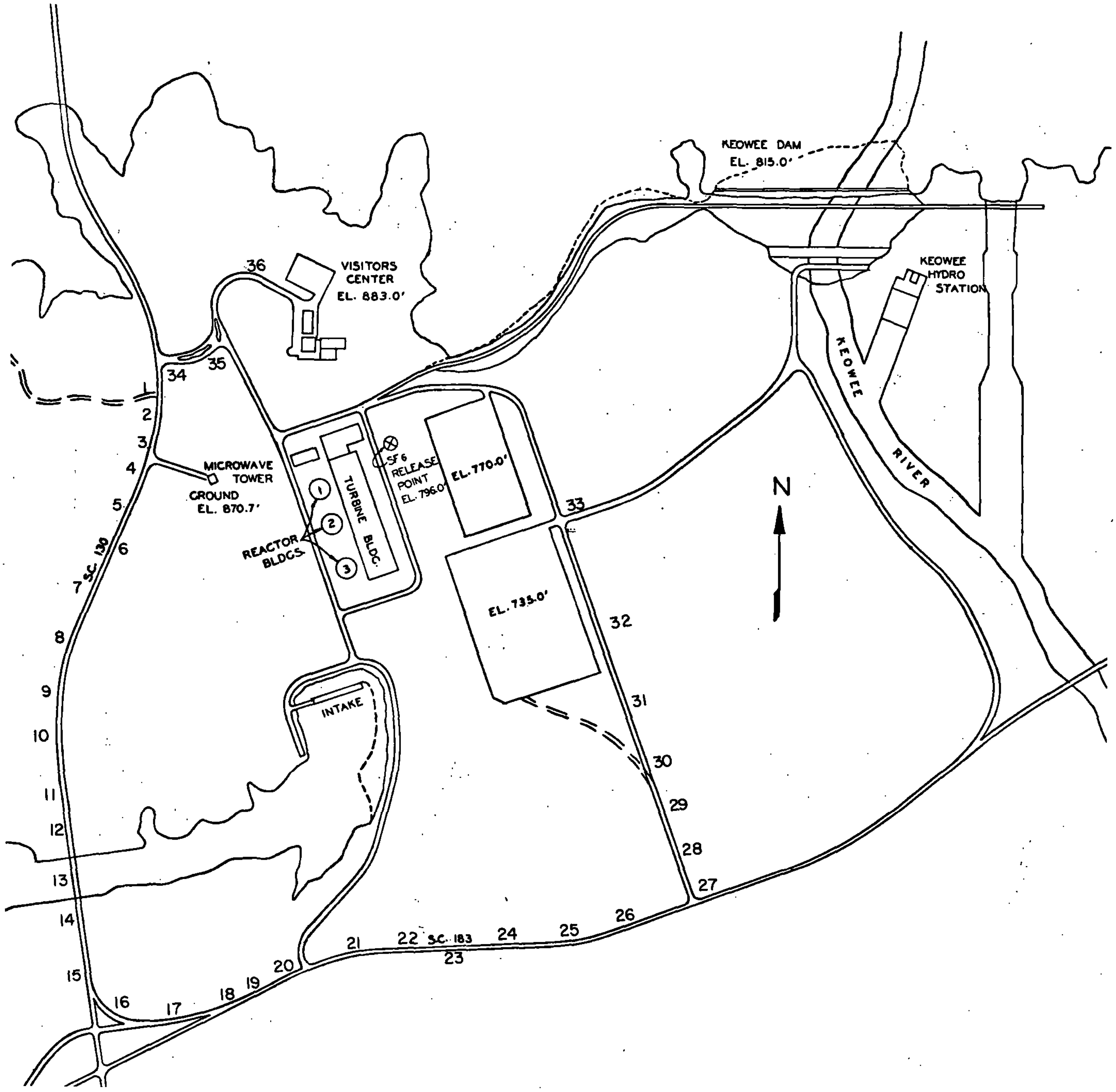
No SF_6 Detected

SF_6 GAS TRACER TEST
BACKGROUND SAMPLE POINTS

OCONEE NUCLEAR STATION

Figure 2A - 5 Test Date: 3/11/70

(New) Rev. 7 7/9/70



SF_6 GAS TRACER TEST

SF_6 RELEASE POINT AND SAMPLE STATIONS

OCONEE NUCLEAR STATION

Figure 2A - 6 Test Date: 3/11/70

(New) Rev. 7 7/9/70

LIST OF EFFECTIVE PAGES
FSAR SECTION 3

Reactor

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LOEP--2 of 5	Rev. 26	3-11	Rev. 22
LOEP--3 of 5	Rev. 30	3-12	Rev. 22
LOEP--4 of 5	Rev. 26	3-13	Rev. 22
LOEP--5 of 5	Rev. 26	3-14	Rev. 16
3-i	Original	3-14a	Rev. 16
3-ii	Rev. 4	3-14b	Rev. 4
3-iii	Original	3-15	Rev. 22
3-iv	Rev. 4	3-16	Rev. 4
3-v	Rev. 4	3-16a	Rev. 4
3-vi	Rev. 4	3-17	Rev. 22
3-vi-(a)	Rev. 16	3-17a	Rev. 4
3-vii	Rev. 16	3-17b	Rev. 4
3-viii	Rev. 16	3-17c	Rev. 4
3-1	Rev. 22	3-17d	Rev. 4
3-2	Original	3-18	Rev. 4
3-3	Rev. 23	3-19	Rev. 4
3-3a	Rev. 23	3-19a	Rev. 4
3-4	Original	3-19b	Rev. 4
3-5	Rev. 24	3-19c	Rev. 4
3-5a	Rev. 9	3-19d	Rev. 4
3-6	Rev. 9	3-19e	Rev. 4
3-7	Rev. 22	3-19f	Rev. 4
3-8	Rev.4	3-20	Rev. 4
3-9	Rev. 22	3-21	Rev. 4

Reactor

Pages	Revision	Pages	Revision
3-22	Rev. 16	3-45	Original
3-23	Rev. 3	3-46	Rev. 4
3-24	Rev. 4	3-46a	Rev. 4
3-24a	Rev. 4	3047	Rev. 1
3-25	Original	3-48	Rev. 1
3-26	Original	3-49	Rev. 1
3-27	Rev. 16	3-50	Rev. 1
3-28	Original	3-51	Rev. 18
3-29	Original	3-52	Rev. 16
3-30	Original	3-53	Original
3-31	Rev. 9	3-54	Rev. 1
3-32	Rev. 4	3-55	Rev. 16
3-33	Original	3-56	Rev. 9
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3 REACTOR

3.1 DESIGN BASES

The reactor is designed to meet the performance objectives specified in 3.1.1 without exceeding the limits of design and operation specified in 3.1.2.

3.1.1 PERFORMANCE OBJECTIVES

The reactor is designed to operate at 2,568 MWt (rated power) with sufficient design margins to accommodate transient operation and instrument error without damage to the core and without exceeding the pressure at the relief valve settings for the reactor coolant system.

The fuel rod cladding is designed to maintain its integrity for the anticipated operating transients throughout core life. The effects of gas release, fuel dimensional changes, and corrosion- or irradiation-induced changes in the mechanical properties of cladding are considered in the design of fuel assemblies.

22. Core reactivity is controlled by control rod assemblies (CRA) and soluble boron in the coolant. For the Oconee Units 2 and 3 core, it is also controlled with burnable poison rod assemblies (BPRA) added for the longer first fuel cycle. Sufficient CRA worth is available to shut the reactor down with at least a 1% $\Delta k/k$ subcritical margin in the hot condition at any time during the life cycle with the most reactive CRA stuck in the fully withdrawn position. Equipment is provided to add soluble boron to the reactor coolant to insure a similar shutdown capability when the reactor is cooled to ambient temperatures.

The reactivity worth of a CRA, and the rate at which reactivity can be added, is limited to insure that credible reactivity accidents cannot cause a transient capable of damaging the reactor coolant system or causing significant fuel failure.

3.1.2 LIMITS

3.1.2.1 Nuclear Limits

The core has been designed to the following nuclear limits:

a. Fuel has been designed for a maximum burnup of 55,000 MWd/MTU.

b. The power Doppler coefficient is negative. However, the control system is capable of compensating for reactivity changes resulting from either positive or negative nuclear coefficients.

c. A control system consisting of part length axial power shaping rods is provided to allow the shaping of power axially in the core, thereby thwarting any tendency towards axial instability resulting from a redistribution of xenon.

d. The core will have sufficient excess reactivity to produce the design power level and lifetime without exceeding the control capacity or shutdown margin.

e. Controlled reactivity insertion rates have been limited to a maximum value of 1.1×10^{-4} $(\Delta k/k)/s$ for a single regulating CRA group withdrawal, and 4.4×10^{-6} $(\Delta k/k)/s$ for soluble boron removal.

f. Reactor control and maneuvering procedures will not produce peak-to-average power distributions greater than those listed in Table 3-1. The low reactivity worth of CRA groups inserted during power operation limits power peaks to acceptable values.

3.1.2.2 Reactivity Control Limits

The control system and operational procedures will provide adequate control of the core reactivity and power distribution. The following control limits will be met:

a. Sufficient control will be available to produce an adequate shutdown margin.

b. The shutdown margin will be maintained throughout core life with the CRA of highest worth stuck out of the core.

c. CRA withdrawal rate limits the reactivity insertion rate to a maximum of 1.1×10^{-4} $(\Delta k/k)/s$ for a single regulating group. Boron dilution is limited to a reactivity insertion rate of 4.4×10^{-6} $(\Delta k/k)/s$.

3.1.2.3 Thermal and Hydraulic Limits

The reactor core is designed to meet the following limiting thermal and hydraulic conditions:

a. No central melting in the fuel at the design overpower (114 per cent).

b. A 99 per cent confidence that at least 99.5 per cent of the fuel rods in the core are in no jeopardy of experiencing a departure from nucleate boiling (DNB) during continuous operation at the design overpower.

c. Essentially 100 per cent confidence that at least 99.96 per cent of the fuel rods in the core are in no jeopardy of experiencing a DNB during continuous operation at rated power.

d. The generation of net steam in the hottest core channels is permissible, but steam voids will be low enough to prevent flow instabilities.

The design overpower is the highest credible reactor operating power permitted by the safety system. Normally, trip on overpower will occur at significantly lower power than the design overpower. Core rated power is 2,568 MWt.

3.1.2.4 Mechanical Limits

3.1.2.4.1 Reactor Internals

The reactor internal components are designed to withstand the stresses resulting from startup; steady state operation with one or more reactor coolant pumps running; and shutdown conditions. No damage to the reactor internals will occur as a result of loss of pumping power.

Reactor internals are fabricated primarily from SA-240 (Type 304) material and designed within the allowable stress levels permitted by the ASME Code, Section III, for normal reactor operation and transients. Structural integrity of all core support assembly circumferential welds is assured by compliance with ASME Code Sections III and IX, radiographic inspection acceptance standards, and welding qualification.

The core support structure is designed as a Class I structure, as defined in PSAR Appendix 5A to resist the effects of seismic disturbances. The basic design guide for the seismic analysis is AEC publication TID-7024, "Nuclear Reactors and Earthquakes."

Lateral deflection and torsional rotation of the lower end of the core support assembly is limited in order to prevent excessive deformation resulting from seismic disturbance thereby assuring insertion of control rod assemblies (CRA's). Core drop in the event of failure of the normal supports is limited by guide lugs so that the CRA's do not disengage from the fuel assembly guide tubes. (3.2.4.1)

The structural internals are designed to maintain their functional integrity in the event of any major loss-of-coolant accident. The dynamic loading resulting from the pressure oscillations because of a loss-of-coolant accident will not prevent CRA insertion.

Internals vent valves are provided to relieve pressure resulting from steam generation in the core following a postulated reactor coolant inlet pipe rupture, so that the core will be rapidly re-covered by coolant.

Allowable Stresses

The loading combinations and corresponding stress criteria, including the analytically predicted values of internals deflection for the combined maximum seismic and LOCA loadings, are given in B&W Topical Report BAW-10008; Part 1;
3. "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident (LOCA) and Maximum Hypothetical Earthquake." Additional criteria for stresses due to flow-induced vibratory loads are given in B&W Topical Report BAW-10051,
23. "Design of Reactor Internals and Incore Instrument Nozzles for Flow Induced Vibrations."

Methods of Load Analysis to be Employed for Reactor Internals and Core

3. In Part 1, static or dynamic analysis is used as appropriate. In general, dynamic analysis is used for earthquakes and the subcooled portion of the

loss-of-coolant accident (LOCA). For the relatively steady-state portion of the LOCA, a static analysis is used. BAW-10035 is analyzed in similar
21. fashion to Part 1 except the steady-state portion of LOCA is not a significant contributor.

Where it is indicated that substantial coupling, i.e., interrelationship, exists between major components of the Nuclear Steam System (NSS), such as the steam generator, the piping, and the vessel, the dynamic analysis includes the response of the entire coupled system. However, where coupling is found to be small, the component or groups of components are treated independently of the overall system.

The dynamic analysis for LOCA uses predicted pressure-time histories as input to a lumped-mass model. For earthquakes, actual earthquake records normalized to appropriate ground motion, are used as input to the model. The output from the analysis is in the form of internal motions (displacements, velocities, and accelerations), motions of individual fuel assemblies, impact loads between adjacent fuel assemblies, and impact loads between peripheral fuel assemblies and the core shroud. Motions of the reactor vessel, internals and core have been confirmed using a time history excited lumped mass solution.

In addition, seismic analysis is also performed using a modal superposition and response spectra approach.

For the simultaneous occurrence of LOCA and the maximum earthquake, both time-history excitations are input to the system simultaneously such that maximum structural motions, indicating maximum stresses, are obtained. Outputs are those mentioned above.

The output from the lumped-mass model and additional information such as pressure-time histories on separate internals and core components (including control rods) are used to calculate stresses and deflections. These stresses and deflections are compared to the allowable limits for the various loading combinations as established in Appendix 5A of the Preliminary Safety Analysis Report to insure that they are less than these allowables.

3.1.2.4.2 Core Components

Fuel Assemblies

Fuel assemblies are designed for structural adequacy and reliable performance during core operation, handling, and shipping. Design criteria for core operation include steady state and transient conditions under combined effects of flow induced vibration, temperature gradients, and seismic disturbances.

Spacer grids, located along the length of the fuel assembly, position fuel rods in a square array, and are designed to maintain fuel rod spacing during core operation, handling, and shipping. Spacer-grid to fuel-rod contact loads are established to minimize fretting, but to allow axial relative motion resulting from fuel rod irradiation growth and differential thermal expansion.

The fuel assembly upper end fitting is indexed to the plenum assembly by the grid immediately above the fuel assemblies to assure proper alignment of the fuel assembly guide tubes to the control rod guide tube. The guidance of the control rod assembly and axial power shaping rod assembly is designed such that these assemblies will never be disengaged from the fuel assembly guide tubes during operation.

The Zircaloy-4 cladding is designed to withstand strain resulting from combined effects of reactor pressure, fission gas pressure, fuel expansion, and thermal and irradiation growth. Clad strain resulting from normal and abnormal operating conditions is limited as follows:

a. Stresses not relieved by small material deformation are limited so as not to exceed either the yield strength of the material or 75 per cent of the stress rupture life of the material. An example of such a stress is the circumferential membrane stress in the clad due to internal or external pressure.

b. Stresses relieved by small material deformation are permitted to exceed the yield strength. Strain limits for this stress condition are established based on low-cycle fatigue techniques, not to exceed 90% of material fatigue life. Evaluation of cyclic loading is based on conservative estimates of the number of cycles to be expected. An example of this type of stress is the thermal stress resulting from thermal gradients across the clad thickness. The initial pressurization of fuel rods results in a change in fuel temperature as a
9. function of power level and fuel exposure. This change in fuel temperature will affect the power coefficient, the core power shape, and the critical boron concentration at the beginning of core life.
A design study has been completed that verifies that the following
24. physics parameters as they affect safety for the Oconee I core are within design limits: (See Appendix 3A)

   1. Initial fuel cycle reactivity and effects on the second cycle.

9. 2. Potential ejected and stuck rod worths.

   3. Moderator and power coefficient at BOL.

Pre-pressurizing fuel rods will result in lower fuel temperatures throughout core life. Internal pressures at beginning of life will will be significantly higher than for non-pressurized fuel rod. The reduction in fission gas released, as a result of lower fuel temperatures, tends to offset the increase in end-of-life internal pressure that one would normally expect with the initial gas backfill. The core design bases continue to be satisfied with pre-pressurization of the fuel for these units.

c. Combinations of these two types of stresses, in addition to the individual treatment outlined above, are evaluated on the low-cycle fatigue basis of item b. Clad plastic strain due to diameter increases resulting from fuel swelling, thermal ratcheting and creep, and the effects of internal gas pressure, is limited to about 1 per cent.

d. Minimum clad collapse pressure margins are as follows:

   1. 10 per cent margin over system design pressure, on short-time collapse, at fuel rod end voids.

2. End voids must not collapse (must be either freestanding or have adequate support) on a long-time basis.

3. 10 per cent margin over system operating pressure, on short time collapse, at hot spot average temperature of the clad wall.

4. Clad must be freestanding at design pressure on a short-time basis at approximately 725 F hot spot temperature averaged through the clad wall.

Control Rod Assemblies

Absorber material cladding used on the Control Rod Assembly (CRA), Axial Power Shaping Rod Assembly (APSRA), and the Burnable Poison Rod Assembly (BPRA), is designed to the same criteria as the fuel rod clad, as applicable to absorber material characteristics. Clearance is provided between the rods of each of these assemblies and fuel assembly guide tubes to permit coolant flow to limit operating temperature of the absorber materials. In addition, this clearance is designed to permit rod motion as required during reactor operation under any condition including seismic disturbances. Excessive stress in the CRA components during trip of the rod drive mechanism is prevented by use of conservative design stress limits and by hydraulic snubbing to minimize shock in the drive mechanism.

Orifice Rod Assembly (ORA)

The orifice rod assembly is designed to have adequate clearance when inserted into the fuel assembly guide tubes to permit coolant flow without unacceptable mechanical interference between the rod assembly and guide tubes under any operating condition.

3.1.2.4.3 Control Rod Drive Mechanisms

Shim Safety Drive

The shim safety control rod drives provide control rod assembly (CRA) insertion and withdrawal rates consistent with the required reactivity changes for reactor operational load changes. This rate is based on the worths of the various rod groups, which have been established to limit power-peaking flux patterns to design values. The maximum reactivity addition rate is specified to limit the magnitude of a possible nuclear excursion resulting from a control system or operator malfunction. The normal insertion and withdrawal velocity has been established as 30 in./min.

The drive provides a trip of the CRA which results in a rapid shutdown of the reactor for conditions that cannot be handled otherwise by the reactor control system. The trip set point is based on the results of various reactor emergency analyses, including instrument and control delay times and the amount of reactivity that must be inserted before deceleration of the CRA occurs. The maximum travel time for a 2/3 insertion on a trip command of a CRA has been established as 1.40 s.

The control rod drives can be coupled and uncoupled to their respective CRA's without any withdrawal movement of the CRA.

All pressure-containing components are designed to meet the requirements of the ASME Code, Section III, Nuclear Vessels, for Class A vessels (summer '67 addendum).

Materials selected for the control rod drive are capable of operating within the specified reactor environment for the life of the mechanism without any deleterious effects. Adequate clearances are provided between the stationary and moving parts of the control rod drives so that the CRA trip time to full insertion will not be adversely affected by mechanical interference under all operating conditions and seismic disturbances.

Axial Power Shaping Drive

The axial power shaping drives operate similarly to the shim safety drives, except that the trip function has been eliminated. They have the same insertion and withdrawal velocity of 30 in./min and can be coupled and uncoupled to their respective APSRA without any withdrawal movement of the assembly.

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The pressure-containing components are designed to meet the requirements of the ASME Code, Section III, Nuclear Vessels, for Class A vessels (summer '67 addendum). The material and structural design is the same as for the shim safety drive.

3.2 REACTOR DESIGN

3.2.1 GENERAL SUMMARY

The important core design, thermal, and hydraulic characteristics are tabulated in Table 3-1.

Table 3-1

Core Design, Thermal, and Hydraulic Data

Reactor	
Rated Heat Output, MWt	2,568
Vessel Coolant Inlet Temperature, F	554
4. Vessel Coolant Outlet Temperature, F	603.8
3. Core Outlet Temperature, F	606.2
Core Operating Pressure, psig	2,185
Core and Fuel Assemblies	
Total Number of Fuel Assemblies in Core	177
Number of Fuel Rods per Fuel Assembly	208
Number of Control Rod Guide Tubes per Assembly	16
Number of In-Core Instrumentation Positions per Fuel Assembly	1
Fuel Rod Outside Diameter, in.	0.430
Clad Thickness, in.	0.0265
Fuel Rod Pitch, in.	0.568
Fuel Assembly Pitch Spacing, in.	8.587
Unit Cell Metal/Water Ratio (Volume Basis)	0.82
Clad Material	Zircaloy-4 (Cold Worked)
Fuel	
Material	UO_2
Form	Dished-End, Cylindrical Pellets
Pellet Diameter, in.	0.370
Active Length, in.	144
Density, % of Theoretical - Unit I, Core 1	93.5
22. Density, % of Theoretical - All other Units and Cores	92.5
Heat Transfer and Fluid Flow at Rated Power	
Total Heat Transfer Surface in Core, ft^2	49,734
Average Heat Flux, Btu/h-ft^2	171,470
Maximum Heat Flux, Btu/h-ft^2	534,440

22.

Table 3-1 (Cont'd)

1.	Average Power Density in Core, kW/ℓ	83.38
	Average Thermal Output, kW/ft of Fuel Rod	5.656
	Maximum Thermal Output, kW/ft of Fuel Rod	17.63
	Maximum Clad Surface Temperature, F	654
	Average Core Fuel Temperature, F	1540
	Maximum Fuel Central Temperature at Hot Spot, F	4250
	Total Reactor Coolant Flow, lb/h	131.32×10^6
	Core Flow Area (Effective for Heat Transfer), ft^2	49.19
	Core Coolant Average Velocity, fps	15.73
1.	Coolant Outlet Temperature at Hot Channel, F	647.1
	Power Distribution	
	Maximum/Average Power Ratio, Radial x Local ($F_{\Delta h}$ Nuclear)	1.78
	Maximum/Average Power Ratio, Axial (F_z Nuclear)	1.70
	Overall Power Ratio (F_q Nuclear)	3.03
	Power Generated in Fuel and Cladding, %	97.3
	Hot Channel Factors	
	Power Peaking Factor (F_Q)	1.011
	Flow Area Reduction Factor (F_A)	
	Interior Bundle Cells	0.98
	Peripheral Bundle Cells	0.97
	Local Heat Flux Factor (F_Q'')	1.014
	Hot Spot Maximum/Average Heat Flux Ratio (F_q nuc and mech)	3.12
	DNB Data	
	Design Overpower (% Rated Power)	114
	DNB Ratio at Design Overpower (W-3)	1.55
	DNB Ratio at Rated Power (W-3)	2.0
4.	Limiting DNB Ratio at Design Overpower (W-3)	1.3

3.2.2 NUCLEAR DESIGN AND EVALUATION

This section presents the evaluation of significant core parameters and shows that the basic design of the core satisfies the performance limits and objectives of 3.1.2.1 and 3.1.2.2.

3.2.2.1 Nuclear Characteristics of the Design

A summary of the nuclear characteristics of the core is given in Table 3-2.

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Table 3-2

Nuclear Design Data

16. (Reference Supplement 9 Revisions for Oconee 3)

		Oconee I	Oconee II	Oconee III
	Fuel Assembly Volume Fractions			
	Fuel	0.303	0.303	0.303
	Moderator	0.580	0.580	0.580
	Zircaloy	0.102	0.102	0.102
	Stainless Steel	0.003	0.003	0.003
	Void	0.012	0.012	0.012
		1.000	1.000	1.000
	Total UO_2 (BOL, First Core),			
22.	Metric Tons	94.1	93.1	93.1
	Core Dimensions, in.			
	Equivalent Diameter	128.9	128.9	128.9
	Active Height	144.0	144.0	144.0
	Unit Cell H_2O to U Atomic Ratio (Fuel Assembly)			
22.	Cold	2.85	2.88	2.88
	Hot	2.04	2.06	2.06
	Full-Power Lifetime, Days			
	First Cycle	310	460	456
22.	Equilibrium Cycle	310	310	310
	Fuel Irradiation, MWD/MTU			
	First Cycle Average	9600	14,400	14,275
22.	Equilibrium Cycle Average	9700	9700	9700
	Fuel Loading, wt% U-235			
	Core Average First Cycle	2.10	2.62	2.56
22.	First Reload Average	2.98	3.23	3.23
	Core Average Equilibrium Cycle	3.06	3.06	3.06
	Control Data			
	Control Rod Material	Ag-In-Cd	Ag-In-Cd	Ag-In-Cd
	Number of Full Length CRA's	61	61	61
	Number of APSR's	8	8	8
22	Worth of 61 Full Length CRA's ($\Delta k/k$)%	10.9	10.0	10.0
	Control Rod Cladding Material	SS 304	SS 304	SS 304

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3.2.2.1.1 Excess Reactivity

The excess reactivity associated with various core conditions is tabulated in Table 3-3.

Table 3-3
Excess Reactivity Conditions

Effective Multiplication, k_{eff}[a]	Oconee I	Oconee II[b]	Oconee III[b]
1. Cold, 70 F, Clean	1.248	1.257	1.251
22. 2. Hot, 532 F, Clean, Zero Power	1.198	1.194	1.188
3. Hot, 580 F, Clean, Full Power	1.178	1.168	1.162
4. Hot, 580 F, F.P., Equilibrium Xenon and Samarium	1.132	1.122	1.116

Single Fuel Assembly[c]

Hot	0.77
Cold[d]	0.87

[a] First cycle at beginning of life (BOL).
[b] Reflects burnable poison holddown.
[c] Based on highest probable enrichment of 3.5 wt%.
[d] A center-to-center assembly pitch of 21 in. is required for this k_{eff} in cold, unborated water with no xenon or samarium.

22.

The minimum critical mass, with and without xenon and samarium poisoning, may be specified as a single assembly or as multiple assemblies in various geometric arrays. The unit fuel assembly has been investigated for comparative purposes. A single cold, clean assembly containing a maximum probable enrichment of 3.5 weight per cent is subcritical. Two assemblies side-by-side are supercritical under these conditions.

3.2.2.1.2 Reactivity Control Distribution

Control of excess reactivity is shown in Table 3-4.

4. Table 3-4

First Cycle Design Basis Reactivity Control Distribution

	Oconee I	Oconee II	Oconee III
1. Controlled by Soluble Boron (%Δk/k)			
a. Moderator Temperature Deficit (70 to 532 F)	3.4	3.4	3.4
b. Equilibrium Xenon and Samarium	3.5	3.5	3.5
22. c. Fuel Burnup and Fission Product Buildup	11.8	10.9	10.4
2. Controlled by Burnable Poison Rod Assemblies (BPRA)			
22. a. Fuel Burnup and Fission Product Buildup	0.0	4.0	4.0
3. Controlled by Movable Control Rod Assemblies (CRA)			
a. Transient Xenon	1.0	1.0	1.0
b. Doppler deficit (0 to 100% rated power)	1.5	1.5	1.5
c. Moderator Temperature Deficit (0 to 15% Power 532 to 579 F)	0.8	0.8	0.8
d. Dilution Control	0.2	0.2	0.2
e. Shutdown Margin	1.0	1.0	1.0
Total Movable CRA Required	4.5	4.5	4.5

4.

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Explanation of Table 3-4 Items

1. Control by Soluble Boron

   Boron in solution is used to control the following relatively slow-moving reactivity changes:

   a. The moderator deficit in going from ambient to operating temperatures. The value shown is for the maximum change which would occur toward the end of the cycle.

   b. Equilibrium xenon and samarium.

   c. The excess reactivity required for fuel burnup and fission product buildup throughout cycle life.

Figure 3-1 shows the typical variation in boron concentration with life for Cycle 1.

2. Control by Burnable Poison

   The 16 control rod positions in 68 of the fuel assemblies not equipped with control rod assemblies will be utilized as locations for burnable poison rods. The 68 assembly locations are shown in Figure 3-2. Burnable poison
22. will be utilized in Units 2 and 3, but not in Unit 1.

3. Control by Movable CRA

   a. Sufficient rod worth remains inserted in the core during normal operation for 90 per cent of the fuel cycle to overcome the peak xenon transient following a power reduction of 70 per cent of rated power. This override capability facilitates the return to normal operating conditions without extended delays. The presence of these rods in the core during operation does not produce power peaks above the design value, and the shutdown margin of the core is not adversely affected. Axial power peak variation, resulting from partial or full insertion
4. of xenon override rods, is described fully in Figures 3-3 and 3-4. The loss of movable reactivity control due to the insertion of this group
4. produces no shutdown difficulties and is reflected in Table 3-5 and 3-5a.

   b. Power level changes (Doppler) and regulation.

   c. Between 0 and 15 per cent of rated power, reactivity compensation by CRA may be required as a result of the linear increase of reactor coolant temperature from 532 F to the normal operating value.

   d. Additional reactivity is held by a group of partially inserted CRA (25 per cent insertion maximum) to allow periodic rather than continuous soluble boron dilution. Automatic withdrawal of these CRA during operation is allowed to a 5 per cent insertion limit where the dilution procedure is again initiated and this group of CRA is reinserted.

   e. A shutdown margin of 1 per cent $\Delta k/k$ below the hot critical condition also contributes to the reactivity controlled by CRA.

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4.

3.2.2.1.3 Reactivity Shutdown Analysis

The ability to shut down the core under hot conditions is illustrated below. In this tabulation both the first and equilibrium cycles are evaluated at the beginning of life (BOL) and the end of life (EOL) for shutdown capability.

TABLE 3-5

First Cycle Shutdown Reactivity Analysis

	Reactivity, %Δρ					
	Oconee I		Oconee II		Oconee III	
	BOL	EOL *	BOL	EOL	BOL	EOL
1. Required Movable Control Rod Worth						
(a) Doppler deficit, 0-100% power	1.4	1.9	1.1	1.3	1.1	1.3
(b) Moderator deficit (532°F to 580°F)	0.0	0.8	0.0	0.8	0.0	0.8
(c) Inserted transient control rod worth	1.2	1.3	1.0	1.4	1.0	1.4
(d) Possible reactivity feedback from xenon undershoot (below equilibrium)	0.4	0.4	0.4	0.4	0.4	0.4
22. (e) Possible dilution insertion	0.2	0.2	0.2	0.2	0.2	0.2
(f) Total required rod worth	3.2	4.6	2.7	4.1	2.7	4.1
2. Shutdown Analysis						
(a) Total calculated rod worth	12.1	11.2	11.1	10.2	11.1	10.2
(b) Rod model correction (10%)	10.9	10.1	10.0	9.2	10.0	9.2
(c) Rod nvt effect	10.9	10.0	10.0	9.1	10.0	9.1
(d) Stuck rod worth (not reduced)	-2.1	-2.8	-3.4	-2.1	-3.4	-2.1
(e) Reduction in rod worth (580°F - 532°F)	-0.2	-0.2	-0.2	-0.2	-0.2	-0.2
(f) Net rod worth available	8.6	7.0	6.4	6.8	6.4	6.8
3. Excess Control Rod Worth						
(a) Net rod worth available minus total required rod worth	+5.4	+2.4	+3.7	+2.7	+3.7	+2.7

* EOL in this table refers to about 30 days before the end of the fuel cycle. (This is the point in time when the transient rod bank begins to move out of the core.)

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TABLE 3-5a

Equilibrium Cycle Shutdown Reactivity Analysis

	Reactivity Effects	Reactivity, % Δρ Oconee I, II, and III BOL	EOL *
	1. Required Movable Control Rod Worth		
16.	(a) Doppler deficit, 0-100% power	1.3	1.4
	(b) Moderator deficit (532°F to 580°F)	0.2	1.0
16.	(c) Inserted transient control rod worth	1.0	1.0
	(d) Possible reactivity feedback from xenon undershoot (below equilibrium)	0.4	0.4
	(e) Possible dilution insertion	0.2	0.2
16.	(f) Total required rod worth	3.1	4.0
	2. Shutdown Analysis		
	(a) Total calculated rod worth	10.0	9.8
16.	(b) Rod model correction (10%)	9.0	8.9
	(c) Rod nvt effect (0.99 to 0.90)	8.9	8.0
	(d) Stuck rod worth (not reduced)	-2.8	-2.2
	(e) Reduction in rod worth (580-532°F)	-0.2	-0.2
	(f) Net rod worth available	5.9	5.6
16.	3. Excess Control Rod Worth		
	(a) Net rod worth available minus total required rod worth	+2.8	+1.6

* EOL in this table refers to about 30 days before the end of the fuel cycle. (This is the point in time when the transient rod bank begins to move out of the core.)

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An examination of Tables 3-5 and 3-5a shows that all three units can be shutdown with the most reactive control rod stuck in the full out position.

There are several conservatisms in the tables which should be understood.

1. Item 1(a) - the Doppler deficit shown here reflects the maximum expected fuel temperatures for the particular time in core life.

2. Item 2(b) - the calculated control rod worth has been reduced 10%.

3. Item 2(d) - the calculated worth of the stuck rod was not reduced.

4. Items 2(a) and 2(d) - two sets of calculations were performed in XY geometry. One set was with 69 full length control rods and the other was with 61 full length control rods. The minimum available rod worth accounting for the stuck rod case was evaluated for each case and the rod worths shown in 2(a) and 2(d) reflect the set resulting in this minimum available worth.

5. As shown on Figure 3-4a the 61 full length control rods are worth 12.8% $\Delta\rho$ with the APSR's in the core (as they will be during operation). Only 12.1% $\Delta\rho$ was used in the reactivity balance. This value is obtained when total worth is calculated with APSR's withdrawn. The two calculations referred to above were performed in three dimensions.

16. 3.2.2.1.3.1 Control Rod Groups for Operation (Reference Supplement 9 for Oconee 2 and 3)

Figure 3-4a shows the position, function, and reactivity worth of the control rod groups designated for Oconee I. These groupings will be used over the first 100 days of equivalent full power. At this time, group 4 will be swapped for group 7 and will be fully inserted into the core to function as a transient control rod bank. The control rod grouping at 100 days is shown on the top half of Figure 3-4b. At 200 days control rod groups 7 and 4 are again swapped. In addition, four of the group 7 rods are switched with four of the group 3 rods. The rod groupings for the remainder of the first cycle of Oconee I are shown on the bottom half of Figure 3-4b. For convenience, the transient control rods have been shown in the figures as solid black. For all groupings during the cycle, however, the control rod group number and the control rod function have the relationship tabulated in Figure 3-4a. The change in the number of control rods per group can be found from Figure 3-4b.

The transient rod worth for the patterns shown is as follows:

BOL transient rod worth = 1.2 % $\Delta\rho$
100 days transient rod worth = 1.0 % $\Delta\rho$
200 days transient rod worth = 1.2 % $\Delta\rho$
285 days transient rod worth = 1.3 % $\Delta\rho$

The above cases were examined to determine the maximum worth of an assumed ejected control rod. The results are shown in Figures 3-4c through 3-4f. No trends can be established from these numbers since the ejected worth is also a function of the rod pattern being utilized. These patterns were changed at 100 day intervals and only one time calculation was performed for each transient bank. Figure 3-4f shows the ejected rod worth at the hot zero power condition for the BOL transient rod pattern.

Ejected rod calculations were performed with PDQ07[1]. Input data was obtained from HARMONY[1a] fuel cycle calculations. The entire core was represented by a two-dimensional cross section of the core. Because of only two-dimensions, part-length rods could not be shown properly. For this reason, each calculation was first performed with only the transient bank inserted and next with the APSR's shown as full length control rods. The position of the maximum
4. worth rod was found by comparing power fractions in the rodded assemblies.

The maximum reactivity rod is one of the rods in an inserted control group for each case that is shown.

For safety analysis calculations the ejected rod worth is taken to be 0.65%$\Delta\rho$ Calculations show this value to be conservative. During startup physics tests the worth of single rods in rod groups will be evaluated so that these worths can be compared with the worths used in rod ejection analysis. The intent is to assure that no actual rod worths exceed the values used in safety analysis.

Under conditions where a cooldown to reactor building ambient temperature is required, concentrated soluble boron will be added to the reactor coolant to produce a shutdown margin of at least 1 percent $\Delta k/k$. Beginning-of-life boron levels for several core conditions are listed in Table 3-6 along with boron worth values. The conditions shown with no CRA illustrate the highest requirements.

Table 3-6
Soluble Boron Levels and Worth (First Cycle)

Core Conditions	Oconee I	Oconee II	Oconee III
	BOL Boron Levels, ppm[a]		
1. Cold, $k_{eff} = 0.99$			
a. No CRA in	1,338	1,609	1,571
b. All CRA in	864	1,094	1,056
c. One stuck CRA	973	1,283	1,245
2. Hot, Zero Power $k_{eff} = 0.99$			
a. No CRA in	1,488	1,728	1,678
b. All CRA in	587	748	698
c. One stuck CRA	800	1,088	1,038
3. Hot Rated Power, $k_{eff} = 1.00$			
a. No CRA in	1,284	1,441	1,391
b. Xenon transient rods in (1% $\Delta\rho$)	1,200	1,341	1,291
22. 4. Hot Rated Power with Equilibrium Xenon, $k_{eff} = 1.00$			
a. No CRA in	1,072	1,191	1,141
b. Xenon transient rods in (1% $\Delta\rho$)	987	1,091	1,041
5. Hot Rated Power with Equilibrium Xe & Sm, $k_{eff} = 1.00$			
a. No CRA in	987	1,091	1,041
b. Xenon transient rods in	902	991	941
6. Boron Worth ($\%\Delta k/k$)/ppm			
a. Hot	1/85	1/100	1/100
b. Cold	1/64	1/75	1/75

4. (a) Boron levels are calculated values. Measured values (titrametric method) can be obtained to $\pm$ 2 ppm.

3.2.2.1.4 Reactivity Coefficients

Reactivity coefficients form the basis for studies involving normal and abnormal reactor operating conditions. These coefficients have been investigated as part of the analysis of this core and are described below as to function and overall range of values.

a. Doppler Coefficient

The Doppler coefficient reflects the change in reactivity as a function of fuel temperature. A rise in fuel temperature results in an increase in the effective absorption cross section of the fuel and a corresponding reduction in neutron production. The range for the Doppler coefficient under operating conditions is expected to be -1.1×10^{-5} to -1.7×10^{-5} $(\Delta k/k)/F$.

The Doppler coefficient of reactivity is due primarily to Doppler broadening of the U-238 resonances with increasing fuel temperature. Temperature-dependent resonance integrals, which include Doppler broadening, are calculated by the B&W proprietary RIP program (See page 3-19a) and are based on the experimental work of Hellstrand, Blomberg, and Horner[1b]. A comparison of calculated to experimental resonance integrals for UO_2 rods of different radii is presented in Figure 3-4g. The curves for r = 0.5 are representative of the Oconee cores.

4. Uncertainties in the calculated values of the Doppler coefficient may be attributed in part to the slight mismatch between the RIP and Hellstrand calculations, but primarily to uncertainties involved in predicting fuel temperatures. Since the Doppler coefficient is a function of fuel temperature, a value calculated at a certain power level is dependent on the accuracy of predicting the fuel temperature at that power level.

In the nuclear calculations, this uncertainty due to fuel temperature is applied conservatively. That is, the highest expected fuel temperature is used to determine Doppler deficit for shutdown reactivity analysis and a lower fuel temperature is used for conservatism in the xenon stability analysis.

b. Moderator Void Coefficient

The moderator void coefficient relates the change in neutron multiplication to the presence of voids in the moderator. The expected range for the void coefficient is shown in Figure 3-5.

c. Moderator Pressure Coefficient

The moderator pressure coefficient relates the change in moderator density, resulting from a reactor coolant pressure change, to the corresponding effect on neutron production. This coefficient is opposite in sign and considerably smaller when compared to the moderator temperature coefficient. A typical range of pressure coefficients over a life cycle would by -5×10^{-7} to $+3 \times 10^{-6}$ $(\Delta k/k)$/psi.

d. Moderator Temperature Coefficient

The moderator temperature coefficient relates a change in neutron multiplication to the change in reactor coolant temperature. Reactors using soluble boron as a reactivity control have a less negative moderator temperature coefficient than do cores controlled solely by movable or fixed CRA. The major temperature effect on the coolant is a change in density. An increasing coolant temperature produces a decrease in water density and an equal percentage reduction in boron concentration. The concentration change results in a positive reactivity component by reducing the absorption in the coolant. The magnitude of this component is proportional to the total reactivity held by soluble boron. Distributed poisons (burnable poison rods or inserted control rods) have a negative effect on the moderator coefficient for a specified boron concentration. That is, the moderator coefficient for a system with 1200 ppm boron in the coolant and 1% rod worth inserted will be more negative than for a system with 1200 ppm boron and no rods inserted. Depending on the core size, core loading, and power density, a plant may or may not require additional distributed poisons to yield the appropriate moderator temperature coefficient as determined by the safety analysis and the stability analysis of the core. This is illustrated in Table 3-7.

Table 3-7

Moderator Temperature Coefficient

		Conditions	Oconee I	Oconee II	Oconee III
	1.	Core size, no. fuel assemblies	177	177	177
22.	2.	Core average enrichment w/o U-235	2.10	2.62	2.56
4.	3.	Avg Power density, MWt/assembly	14.508	14.508	14.508
	4.	Initial critical conditions (hot full power, clean)			
22.		a. Boron concentration, ppm	1200	1341	1291
		b. CRA inserted worth, % $\Delta k/k$	(a) 2.1	1.0	1.0
		c. Burnable poison worth, % $\Delta k/k$	0.0	4.0	4.0
		d. Moderator temperature coefficient, [10^{-4} $(\Delta k/k)/°F$]	+0.27	+0.03	-0.01
4.	5.	Value of Moderator temperature coefficient used in safety analysis, [10^{-4} $(\Delta k/k)/°F$]	+0.9	+0.9	+0.9
16.	6.	Threshold value of moderator temperature coefficient, [10^{-4} $(\Delta k/k)/°F$](b)	+1	>+1	>+1
22.	7.	Moderator temperature coefficient with equilibrium xenon, BOL, [10^{-4} $(\Delta k/k)/°F$]	-0.30	-0.50	-0.54

16.

(a) Inserted rod worth shown for Oconee 1 results from 3-D calculations and reflects transient group worth, APSR's, and partial Doppler insertion.

(b) See Section 3.2.2.2.2

Items 4d and 7 in Table 3-7 above reflect three dimensional calculations using thermal feedback. These coefficients are more negative than the two-dimensional isothermal values previously calculated and shown. It is seen from comparison (Tables 3-7, 3-7a and 3-7b) that three-dimensional spatially distributed effects are important in the determination of reactivity coefficients.

The three-dimensional PDQ07 calculation with thermal feedback was also used to calculate for Oconee I Cycle 1 the change in spatially dependent moderator coefficient for changes in inlet, outlet, and core average moderator temperature ($°F_m$), as shown in Table 3-7a below.

Table 3-7a

BOL Distributed-Temperature Moderator Coefficients, 100% Power, 1200 ppm Boron

Type of Temperature Change	T_{in} (°F)	T_{out} (°F)	α_m (x 10^{-4} $\frac{\Delta\rho}{°F_m}$)
1. T_{in} constant, T_{out} change	554.03	606.90	+0.14
	554.03	609.33	
2. T_{in} and T_{out} change	554.03	606.90	+0.27
	555.00	607.73	
3. T_{in} change T_{out} constant	554.03	606.90	+0.36
	551.20	606.79	

The Oconee reactors operate on a constant core average moderator temperature with both inlet and outlet temperature changing with power level. The core average moderator temperature as seen by the control system is defined to be

$$T_m = \frac{T_{in} + T_{out}}{2} \quad \text{(See Item 2, Table 3-7a)}$$

The BOL distributed temperature moderator coefficients for different reactor power levels are presented in Table 3-7b. These coefficients were found by changing both inlet and outlet temperatures. Criticality in each case was attained by appropriate control rod insertion.

Table 3-7b

BOL Distributed Temperature Moderator Coefficients vs. Power, 1200 ppm Boron, No Xenon

% Power	T_{in} (°F)	T_{out} (°F)	α_m (x 10^{-4} $\frac{\Delta\rho}{°F_m}$)
1. 15	575.00	583.01	+0.42
	576.00	584.00	
2. 60	563.88	595.78	+0.30
	565.00	596.94	
3. 100	554.03	606.90	+0.27
	555.00	607.73	

The moderator temperature coefficient was also calculated for the equilibrium xenon condition at the beginning of the fuel cycle. The calculation assumed 2.1% $\Delta\rho$ in control rods. The 100% power moderator coefficient varied in the manner shown below.

Table 3-7c

BOL Distributed-Temperature Moderator Coefficient, Equilibrium Xenon

Type of Temperature Change	$\alpha_m (\times 10^{-4} \frac{\Delta\rho}{°F_m})$ 0 Days (1200 ppm)	4 Days (920 ppm)
T_{in} and T_{out} change	+.27	-.30

The EOL coefficient was calculated for a change in both the inlet and outlet temperatures with a boron concentration of 17 ppm. The coefficient for 100% power was found to be

$$\alpha_m = -2.8 \times 10^{-4} \frac{\Delta\rho}{°F_m}$$

This, then, is the "rods out" moderator coefficient at the end of the first fuel cycle for Oconee I, Cycle 1.

The coefficients reported in Tables 3-7a and 3-7b above are for a core containing 2.1% $\Delta\rho$ in control rods. A "rods out" calculation for the beginning of life moderator conditions in Item 1, Table 3-7a, was performed as a basis for comparison and the result was

$$\alpha_m = +0.52 \times 10^{-4} \Delta\rho/°F_m$$

An examination of the data in Table 3-7 shows that the limiting factor on a moderator coefficient is the value used in safety analysis, i.e., $+0.9 \times 10^{-4} \Delta\rho/°F_m$. The margin between this value and the nominal calculated value of $+0.27 \times 10^{-4} \Delta\rho/°F_m$ is considered adequate to cover uncertainties. The physics startup program for Unit I includes measurements of the moderator temperature coefficient for several temperature plateaus (isotherms) at zero power and also for several power plateaus (distributed effects). This data will allow an extrapolation to the full power moderator coefficient to assure that the +0.9 α_m will not be exceeded. Previously calculated data will be available during the test period for comparison with the measured values. Two dimensional (XY) calculations will be used for the isothermal coefficients (zero power) and three dimension calculations will be performed to predict the distributed temperature coefficients.

Procedures for calculating and physically measuring (1) moderator coefficient at zero power and (2) moderator coefficients at power are similar. The former are obtained by raising or lowering the average temperature of the entire core and observing the corresponding change in reactivity. The change in reactivity divided by the change in T_{avg} (core) gives the temperature coefficient.

Measuring the moderator coefficients at power must be done in a less direct manner due to the inability to measure the fuel temperature. The moderator coefficient is measured by raising or lowering T_{avg} (moderator) without changing the power. The reactivity change divided by the change in T_{avg} (moderator) produces the measured moderator coefficient. In addition to the change in reactivity due to moderator temperature change, two other appreciable effects are included when measuring the moderator coefficient: (1) the Doppler effect due to increased fuel temperature, and (2) axial expansion of the fuel also due to an increase in fuel temperature. The first effect is incorporated in the calculational model. A correction for the second effect was obtained by separate analysis. This produced an axial-expansion reactivity coefficient of $-0.02 \times 10^{-4} \frac{\Delta\rho}{°F(\text{fuel})}$ which can be applied directly to the change in average fuel temperature when calculating moderator coefficients at rated power.

Comparison of moderator temperature coefficients at zero power and low power plateaus and extrapolating to full power should ensure that the limiting value of $+0.9 \times 10^{-4} \frac{\Delta k/k}{°F_m}$ is not exceeded at rated power. Applying the corrections mentioned above to the calculated moderator coefficients at power and comparing to measured moderator coefficients should give added assurance.

Inaccuracies during zero power tests will result primarily from uncertainties in system conditions and uncertainties in reactivity measurements. Uncertainties in temperature measurement and distribution (at power) will be relatively small. The zero power uncertainty will be approximately $+0.1 \times 10^{-4} \frac{\Delta k/k}{°F_m}$ in the measured values.

e. Power Coefficient

The power coefficient, α_p, is the fractional change in neutron multiplication per unit change in core power level. A number of factors contribute to α_p, but only the moderator temperature coefficient and the Doppler coefficient contributions are significant. The power coefficient can be written as

$$\alpha_p = \alpha_m \frac{\partial T_m}{\partial P} + \alpha_f \frac{\partial T_f}{\partial P}$$

where

α_m = moderator temperature coefficient

α_f = fuel Doppler coefficient

$\frac{\partial T_m}{\partial P}, \frac{\partial T_f}{\partial P}$ = change in moderator and fuel temperature per unit change in core power.

Power coefficients were calculated for Oconee I, Cycle 1 at BOL (time zero) at various power levels. A boron concentration of 1200 ppm was used for all power levels, and criticality was achieved with control rods. The three-dimensional PDQ07 Code with thermal feedback was used to include the effects of spatially distributed fuel and moderator temperatures. In calculating the power coefficients, the inlet temperatures were allowed to increase slightly, thus, providing a greater moderator effect (i.e., emphasizing the positive moderator coefficient - refer to Table 3-7a).

The results are presented in Table 3-17d and plotted in Figure 3-5a.

Table 3-7d

Power Coefficients of Reactivity

4.

Power (% Full Power)	α_p $\left(x10^{-6} \frac{\Delta\rho}{MWT}\right)$	α_p $\left(x10^{-4} \frac{\Delta\rho}{\%\Delta P}\right)$
15	-7.93	-2.04
60	-6.09	-1.56
100	-4.33	-1.11

f. pH Coefficient

Currently, there is no definite correlation which will permit prediction of pH reactivity effects. Some of the parameters needing correlation are the effects relating pH reactivity change for various operating reactors, pH effects versus reactor operating time at power, and changes in effects with varying clad, temperature, and water chemistry. Yankee, Saxton, and Indian Power Station 1 have experienced reactivity changes at the time of pH changes, but there is no clearcut evidence that pH is the direct reactivity influencing variable without considering other items such as clad materials, fuel assembly crud deposition, system average temperature, and prior system water chemistry.

The pH characteristic of this design is shown below in Table 3-8 where the cold values are measured and the hot values are calculated.

Table 3-8

pH Characteristics

	^{7}Li, ppm	T_{mid}, F	Boron Concen., ppm	pH Units
	0.5	70	1,800	5.0
	2.0	70	1,800	5.6
	0.5	580	1,200	7.0
	2.0	580	1,200	7.5
	0.5	580	17	7.2
	2.0	580	17	7.8
1.	0.5	70	17	7.9
	2.0	70	17	8.5

4. Saxton experiments[2] have indicated a pH reactivity effect of 0.0016 $\Delta\rho/\Delta$pH unit change with and without local boiling in the core. Considering the system makeup rate of 35,000 lb/h and the core in the hot condition with 1,200 ppm boron in the coolant, the corresponding changes in pH are 0.02 pH units per hour for boron dilution and 0.05 pH units per hour for ^{7}Li dilution (starting with 0.5 ppm ^{7}Li). Applying the pH worth value quoted above from Saxton, the total reactivity insertion rate for the hot condition is 3.1×10^{-8} $\Delta\rho$/s. This insertion rate of reactivity can be easily compensated by the operator or the automatic control system.

3.2.2.1.5 Reactivity Insertion Rates

Figure 7-5 displays the integrated rod worth of three overlapping rod banks as a function of distance withdrawn. The indicated groups are those used in the core during power operation. Using approximately 1.2 per cent $\Delta k/k$ CRA groups and a 30 in./min drive speed in conjunction with the reactivity response given in Figure 7-5 yields a maximum reactivity insertion rate of 1.1×10^{-4} $(\Delta k/k)$/s. The maximum reactivity insertion rate for soluble boron removal is 4.4×10^{-6} $(\Delta k/k)$/s.

3.2.2.1.6 Power Decay Curves

Figure 3-6 displays the beginning-of-life power decay curves for the CRA worths corresponding to the 1 per cent hot shutdown margin with and without a stuck rod. The power decay is initiated by the trip of the CRA with a 300 ms delay from initiation to start of CRA motion. The time required for insertion of a CRA 2/3 of the distance into the core is 1.4 s.

3.2.2.2 Nuclear Evaluation

Analytical models and their application are discussed in this section. Core
4. instabilities associated with xenon oscillation are also described.

3.2.2.2.1 Analytical Models

Reactor design calculations are made using a large number of computer codes. The choice of which code set or sets to use is set forth below by design area.

4. a. <u>Nuclear Calculation Model</u>

Calculation of the reactivity of a pressurized water reactor core is performed in one, two, or three dimensions. The geometric choice depends on the type of calculations to be made. In a "clean" type of calculation where there are no strong, localized absorbers of a type differing from the rest of the lattice, 1-dimensional analysis is satisfactory. This type of problem is handled quite well by the 1-dimensional B&W proprietary depletion package code LIFET.

4.

LIFET is a code developed by the Babcock & Wilcox Company. It utilizes the MUFT[3], WANDA[4], and the proprietary B&W RIP, TAME, and AMOP computer programs. A brief description of LIFET follows.

CALCULATION OF EPITHERMAL CROSS SECTIONS

The epithermal neutron energy spectrum for each composition region is calculated using the B_1 (or P_1) transport approximation with the Fermi age slowing-down treatment for heavy nuclides and the Grueling-Goertzel[4a] treatment for the light nuclides. For nuclides exhibiting resonances in the epithermal energy range, the effective resonance integral, corrected for shielding effects and Doppler broadening, is calculated by either the Narrow Resonance or Narrow Resonance-Infinite Absorber model for each epithermal multigroup. For tightly packed heterogeneous
4. lattices, the Dancoff factor is calculated by the method of Sauer.[4b,4c] In computing the epithermal spectrum, both inelastic scattering and (n, 2n) reactions are taken into account. After calculating the neutron spectrum, macrogroup (3 or less groups) diffusion constants are obtained by averaging over the spectrum.

The resonance integral calculation in the LIFET program is adapted from the RIP program. This program is based upon the work of Dresner[4d] and Adler, Hinmann and Nordheim.[5] Effective resonance integrals are calculated at each resolved and unresolved peak for each resonance nuclide. The effective resonance integral for each energy interval in the multigroup structure is then the sum of the integrals over all peaks in the multigroup. These multigroup resonance integrals are subsequently used in the epithermal spectrum calculation.

CALCULATION OF THERMAL CROSS SECTIONS

A thermal spectrum calculation is performed for each composition region of a reactor. Since these regions are not strictly homogeneous, a space-energy calculation should be performed to account for the heterogeneous configuration. For the case of uniformly distributed fuel rods in a moderating medium, the space-energy flux may be described in terms of separate space- and energy-dependent terms. The AMØP routine in the proprietary LIFET program describes the heterogeneous effects by calculating effective microscopic cross sections for each nuclide. These cross sections reflect the geometrical configuration of the fuel rods, clad and moderator. The TAME routine then calculates the energy dependence of the flux for an equivalent homogeneous region. Finally, the cross sections are averaged over the energy spectrum to obtain a single thermal cross section set for use in the multigroup diffusion calculation.

b. <u>Reactivity Analysis of Critical Experiments</u>

Thirty-one UO_2 and PuO_2-UO_2 critical assemblies were analyzed in order to verify the methods previously described. This verification program was undertaken for two purposes. The first objective was to determine the validity of these methods for predicting k_{eff} of critical assemblies by means of a one-dimensional model. The second purpose was to investigate the applicability of the present thermal and epithermal spectrum calculations for determining effective cross sections to be used in the analysis of large power reactors. The criticality portion of the LIFET program was used to determine k_{eff} for the critical assemblies. This one-dimensional (cylindrical) analysis was performed with four-group cross sections; one thermal group and three epithermal groups. The group structure which was used for the group collapse is given below.

Group	Energy Range
1	9.12 KeV - 10 MeV
2	61.4 eV - 9.12 KeV
3	1.85 eV - 61.4 eV
4 (thermal)	0.0 eV - 1.85 eV

The seventeen UO_2 criticals selected for analysis are described in references 6, 7, 8, 9 and 10. k_{eff} was calculated for each assembly assuming the "resonance" treatment for resonance absorption and using the four-group structure. These values are referred to as the reference values. Statistically, $k^r_{eff} = 0.9983 \pm 0.0047$, where 0.9983 is the arithmetic mean and the error corresponds to one standard deviation of a sample about the mean. All values of k^r_{eff} are between 0.9912 and 1.0068.

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A comparison was also made between k_{eff}^{r} and k_{eff} calculated using a two-group structure. Statistically, $k_{eff}^{(2\text{-}group)} = 1.0032 \pm 0.0080$ for the UO_2 criticals.

For all UO_2 criticals, the difference in k_{eff} between the 2-group and 4-group analysis was + .0049 ± 0.0040 with the 2-group being higher. The important feature is that the difference decreases rapidly as the assembly radius increases. Hence, for a PWR with a radius of about 150 cm., the difference between the two- and four-group treatments would diminish to less than 0.05%. This fact certainly justifies the use of a two-group structure for large power reactors. The increased accuracy of the four-group treatment appears in the calculation of the leakage, which is important only in small assemblies.

Fourteen PuO_2-UO_2 critical assemblies were studied, including three containing no PuO_2. These UO_2 assemblies were considered here because their lattice parameters were similar to the PuO_2-UO_2 assemblies and they contained a larger U-235 weight percent than the other UO_2 criticals. These criticals are described in detail in references 10a and 10b. A criticality calculation was performed on each PuO_2-UO_2 assembly using the "resonance" treatment for resonance absorption in the epithermal energy range and a four-group structure for the calculation of k_{eff}. These values are referred to as the reference values. Statistically, $k_{eff}^{r} = 1.0001 \pm 0.0048$. All values of k_{eff}^{r} are between 0.9889 and 1.0073. A second comparison was made between k_{eff}^{r} and $k_{eff}^{(2\text{-}group)}$ corresponding to a two-group structure. $k_{eff}^{(2\text{-}group)} = 1.0135 \pm 0.0070$ for the PuO_2-UO_2 critical assemblies. For the fourteen criticals the mean difference between the two-group k_{eff} and four-group k_{eff} was 0.0134 ± 0.0061.

It is apparent that the difference between the two- and four-group structures is essentially the same for the PuO_2-UO_2 and UO_2 criticals. The very small assemblies (e.g., 12.94 cm. radius) show errors in excess of 2 percent between the two-group structures.

For all UO_2 and PuO_2-UO_2 critical assemblies, $k_{eff}^{r} = 0.9991 \pm 0.0047$. This indicates that the present methods, employing the "resonance" treatment for resonance absorption and the four-group structure, predict k_{eff} for uniform lattices quite accurately. The one-dimensional model is adequate for assemblies which are nearly cylindrical. However, for square assemblies, k_{eff}^{r} may differ from the mean value by as much as one percent. A two-dimensional criticality calculation should be used for these assemblies.

The two-group treatment yields $k_{eff}^{(2\text{-}group)} = 1.0079 \pm 0.0074$ and $\Delta k_{eff}^{(2\text{-}group)} = 0.0088 \pm 0.0050$ for all UO_2 and PuO_2-UO_2 critical assemblies.

For small assemblies, $\Delta k_{eff}^{(2\text{-group})}$ is greater than two percent, indicating the importance of the group structure on the leakage calculation. For large assemblies, such as a PWR, the difference is less than 0.05%. For uniform lattices of fuel pins in a large reactor, the two-group cross sections calculated according to the methods described in this report should be adequate for design calculations.

c. Power Distribution Analysis of Critical Experiments

A series of detailed three-dimensional power distribution measurements have been performed at the Babcock & Wilcox Company Critical Experiments Laboratory. These experiments were designed to provide detailed power distribution measurements, with both part and full length control rods, against which analytical methods and results can be checked. A detailed description of the tests and results can be found in reference 10c.

The PDQ07 program was used to calculate the power distributions corresponding to the measured values. Since PDQ07 is a three-dimensional program, the core geometry can be described more exactly where necessary. The composition overlay accurately describes both fuel and poison pins in the control rod regions. Solution points are placed both along the boundary and in the center of the cells in the fuel rod regions. The thermal feedback option in PDQ07 is not exercised because the temperature variations are small. As in the actual measurements, the power densities are normalized to the value at mid-height of the twentieth pin (standard rod) on the west radius (shown in Figure 3-6a).

The two-group coefficients input to PDQ07 for fuel pins, borated water, and aluminum-water reflector are generated by the LIFET program. Homogenized poison pins are treated by using the method described in section 3.2.2.2.1-h to obtain absorption coefficients. Figure 3-6b shows graphs of the normalized relative power distributions at 35 cm., mid-plane and 105 cm. from core bottom for the east traverse from core center. These radial distributions correspond to the case where the control rod clusters are withdrawn 72.5 cm., 92.1 cm., and 72.5 cm. for the north, central, and south positions, respectively. The axial normalized power distributions at positions A, B, and C (Figure 3-6a) are shown in Figure 3-6c.

The radial and axial normalized power distributions calculated with the PDQ07 program agree quite well with the measured distributions. There are, however, deviations near control rod locations.

d. Application of Computer Codes to Nuclear Design

PDQ05[11] and PDQ07 in conjunction with HARMONY have been used in quarter core X-Y geometry to deplete the core through several cycles. This

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model was instrumental in choosing core enrichments, core loading schemes and shuffle patterns, and xenon-transient rod banks. End-of-life mass balances and megawatt days per tonne produced by each assembly were obtained from these cases. The same model was used to calculate rod bank worths, critical boron concentration, and total available rod worth, as well as isothermal moderator coefficients and other undistributed reactivity coefficients. Hot and cold ejected and stuck rod worths were calculated with two-dimensional PDQ, showing the full core.

In a depletion calculation, the time steps used are primarily 50 days in length except for the beginning and end of the fuel cycle. Boron reactivity worth is calculated by observing the k_{eff}'s associated with two boron concentrations. Since everything else is held constant, the change in reactivity divided by the change in boron concentration gives the boron worth. These worths are computed as a function of burnup and boron concentration.

The three-dimensional PDQ07 code is also an integral part of core design work. B&W has modified this program to include thermal feedback effects. This option is used in almost all three-dimensional calculations. Extensive analyses have been made of:

1) Control rod maneuvering

2) Xenon stability analysis and control

3) Reactivity coefficients

A three-dimensional analysis with feedback must be utilized for the above problems primarily to obtain an accurate description of the radial and axial power distributions. These distributions are necessary in the evaluation of thermal margin.

e. Fuel Cycle Analysis

The X-Y PDQ model with HARMONY was used to calculate fuel cycles for the three Oconee reactors. The fuel assembly arrangement in the core is shown in Figure 3-47. First cycle and succeeding cycle average burnups for all three units is given in Table 3-2, as is first cycle, second, and equilibrium core enrichment for all units.

f. Control of Power Distributions

The reactors are designed to permit power maneuvering on control rods. The reactors have an incore and an out-of-core detector system. During the Oconee Unit 1 start-up test program, the core power offset, for several rod combinations, as determined by these two systems will be compared. In this way, the alarm limits for unfavorable core offsets will be determined for the out-of-core instrumentation. The out-of-core

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instrumentation will be used to alarm for axial power tilts and provide trip protection for the overpower limit. The operator will maintain as feasible a balanced power split between the top and the bottom of the core throughout the cycle, on the basis of the signal from the out-of-core detectors.

The out-of-core neutron flux detectors each consist functionally of two six-foot sections of uncompensated ion chambers placed opposite the top and bottom halves of the core. Comparison of the signals from the two detectors gives an indication of the core axial offset or imbalance. This imbalance signal (top core power minus bottom core power) is monitored in the control room. When an imbalance is indicated, the operator will move the APSR's in the direction of the imbalance to reduce the axial offset, i.e.,

positive offset - move APSR's toward top;

negative offset - move APSR's toward bottom.

The integrated control system will automatically compensate for reactivity changes and consequent power swings caused by the part length control rod movement.

This concept is illustrated as follows. First, Figure 3-6d gives the radial and axial (hot assembly) power distribution in Oconee I cycle 1 at BOL with all rods out. It is noted on Figure 3-6d that a thermal margin of 17.7% exists for the worst peaking conditions in the core. This thermal margin is the difference between the maximum total hot channel peaking factor and the maximum allowable peaking for the design basis.

Next, Figure 3-6e shows the Oconee I rod pattern for the first 100 days of operation. This pattern was used to determine the worst combination of part and full length control rod positions for beginning of life operation. Figure 3-6f shows the radial and axial power shapes which yield the smallest thermal margin at BOL including transient xenon effects. These rod positions occur during xenon burnout following a power recovery. In the analysis, after equilibrium xenon has built up at 100% power, the power is reduced and a new equilibrium xenon distribution is attained. At this time power is recovered to 100% and the resulting xenon burnout yields the limiting rod positions and power shapes of Figure 3-6f.

Note the extreme inserted position of the APSR's to yield a power balance between top and bottom of the core. Also note that the thermal margin is 7.2%. The worst rod configuration (i.e., the rod configuration yielding the maximum core power imbalance) for this case would have been realized if the APSR's were moved in the wrong direction, i.e., towards the top of the core. The results for this case are shown on

Figure 3-6g. The power is peaked toward the bottom of the core and the thermal margin is increased to 7.8%. This "3-D" picture is necessary to evaluate the thermal margin of the core at the given set of conditions. The thermal margin concept is used to determine the worst case design configuration of part and full length control rods at all times during the fuel cycle for both steady state and transient conditions.

The out-of-core instrumentation system can stray from calibration in the event that the radial flux changes. However, the flux error occurring from these disturbances is expected to be within 4% of the actual flux. In addition, during steady state operation, a re-calibration is administratively required when it is determined that the flux error exceeds 2%. Consequently, the 4% flux error used in establishing the reactor trip setting is considered adequate. Furthermore, the start-up testing of Oconee I will include the verification of the uncertainties between radial flux variations and out-of-core detector response.

g. Power Maldistributions

MISALIGNED CONTROL RODS

The reactor has a control function to protect against a rod out of step with its group. The position of each rod is compared to the average of the group. If a fault is detected at power levels above 60% of rated power, a rod withdrawal inhibit is activated. If a rod is dropped, the ICS cannot maintain core power to match demand by withdrawal of other rods, and the plant is run back to 60% of rated power. Several cases were also analyzed for BOL for Oconee Unit I, Cycle 1, with single dropped rods. The calculations were performed with half-core X-Y geometry in PDQ07 at rated power without thermal feedback. The results are given in Figure 3-6h.

The maximum radial-local power peak is 1.92. The design limit is a 2.1 radial-local at rated power with a 1.5 cosine yielding a 1.3 DNBR. At the 114% overpower condition the design limit can also be expressed as a 1.9 radial-local with a 1.5 cosine yielding a 1.3 DNBR. The dropped rods illustrated in Figure 3-6h do not represent violations of the thermal limits of the design.

Radial power tilts can be detected with the out-of-core instrumentation, and the operator has the flexibility to monitor the upper or the lower detectors to determine the X-Y power symmetry condition at any time.

For the assumed case where one CRA is left out of the core while the remainder of the group is fully inserted - this condition would not occur except with regard to rod "swaps". Since rod swaps are performed at reduced power, and since the operator can monitor the out-of-core detectors, an X-Y tilt resulting from such a condition could be detected and appropriate action taken before the approach to thermal limits could be realized.

The APSR drives are also equipped with the position monitors and the alarm function for a rod out of step with the group average. These drives, however, do not permit rod drops. With the power removed from the rod drive windings of the APSR, the roller nut will not disengage and the rod remains in its position (See 3.2.4.3.2). Since the maneuvering range for these rods will be only three to four feet, it is not likely that thermal limits will be exceeded if one of the rods were stuck and the rest of the group were moved.

AZIMUTHAL XENON OSCILLATIONS

The Oconee Unit I reactor is predicted to have a substantial margin to threshhold for azimuthal xenon oscillations. Since Units II and III have more negative moderator temperature coefficients, they are expected to be at least as stable azimuthally as Unit I. Therefore, this mode is not considered to be likely to produce a power peaking problem.

FUEL MISLOADING

Misloading the fuel pins in an assembly is prevented by loading controls and procedures. Each fuel rod is identified by an enrichment code, and the design of the reactor is such that only one enrichment is used per assembly. The manufacturing process relies on administrative procedures and quality control checks to assure that fuel rods are placed in the proper assembly. One such administrative procedure which will be practiced to the extent practical is the "campaigning" of enrichments so that only a single enrichment is handled at a given time in fuel fabrication.

Gross fuel assembly misplacement in the core is prevented by administrative core loading procedures and the prominent display of fuel assembly identification markings on the upper end fitting of each assembly. A misplaced assembly would be detected with the incore detectors during startup tests. During this phase, the response of the incore detectors will be compared to calculations. Even if an assembly were out of position the operator can monitor the out-of-core detectors to determine if a trend is developing towards an X-Y tilt. Upon return to power operation after refueling, and to an even greater extent, upon initial increase to power, the operator will carefully monitor the out-of-core detectors to assure that core symmetry exists.

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h. Control Rod Analysis

B&W has developed a procedure for the analysis of the reactivity worth of small cylindrical Ag-In-Cd control rods. The procedure has been verified against a set of 14 critical experiments in which the variables included; number of rods per cluster, arrangement of rods within the cluster, number of clusters in the core, and soluble boron concentration. Approximately half of the experiments included water holes to simulate withdrawn rods. The comparison of calculated and experimental reactivity worths is shown in Table 3-9.

Table 3-9

Calculated and Experimental Rod and Rod Assembly Comparison

Core No.	Clusters Per Core	Ag-In-Cd Rods Per Cluster	H_2O Holes, Fraction of Core	Soluble Boron ppm	Control Rod Worth[a] % Δk/k Experimental	Calculated
5-B	4	4	0.051	1232	2.02 ± 0.09	2.13 ± 0.02
4-F	4	9	0.0	1219	3.36 ± 0.09	3.43 ± 0.02
5-C	2	12	0.056	1167	2.36 ± 0.09	2.44 ± 0.02
4-D	1	16	0.0	1390	1.45 ± 0.09	1.35 ± 0.02
5-D	2	16	0.058	1118	2.86 ± 0.09	2.78 ± 0.02
4-E	1	20	0.0	1365	1.52 ± 0.09	1.48 ± 0.02
5-E	2	20	0.059	1082	3.06 ± 0.09	3.02 ± 0.02

The experiments were performed at the Babcock & Wilcox Critical Experiment Laboratory with lattices of aluminum-clad uranium oxide fuel rods. Enrichment of the fuel was 2.46 weight percent Uranium-235. The Ag-In-Cd control rods used in the experiments had an absorber diameter of 0.400 inches. Geometrical arrangements of the control rods were chosen to encompass the reference design for the power cores. Experimental rod worths were determined by calibration against soluble boron concentration. The critical soluble poison concentration was determined for each configuration with rods in and again with rods out. (Soluble poison concentrations quoted in the table are for the rods in situation). Soluble poison concentrations were measured with an absolute accuracy of ±5 ppm and a precision of ±3 ppm. References 12 and 12a describe the experiments in detail.

The analytical method used in this analysis is based upon the PDQ code with coefficients generated by the B&W LIFET program. Key features include the allowance for interference and overlap effects between resonances and isotopes in the Ag-In-Cd rod, and calculation of the relative fluxes in the control rod and surrounding fuel in an 80 group thermal model.

(a) Variations in these data from those shown in the initial FSAR tabulation result from a change in the nuclear calculational model.

3.2.2.2.2 Xenon Stability Analysis and Control (Reference Supplement 9 Revisions for Oconee 3)

Modal and digital analysis of the Oconee Unit 1 core indicated that a tendency toward xenon instability in the axial mode would exist for a given combination of events (BOL, rodded core). Therefore, eight part-length Axial Power Shaping Rod Assemblies (APSRA's) have been included in the design. They will be positioned during operation to maintain an acceptable distribution of power for any particular operating condition in the core, thereby reducing the tendency for axial oscillations. Similar analysis which was performed on the Oconee 2 core indicated that it would be stable with regard to axial oscillations. Oconee Unit 3 should have characteristics similar to those of Unit 1.

The azimuthal stability of the cores are dependent upon core loadings, power densities, and moderator temperature coefficients. In any event, the cores will not be susceptible to diverging azimuthal oscillations. If the loadings and power densities are low enough, the core will be inherently stable (Oconee 1 & 3). If not, then burnable poison is added in the amount necessary to provide a moderator temperature coefficient that will result in azimuthal stability (Oconee 2). A detailed description of the xenon analyses performed on Unit 1 and 2 cores may be found in B&W Topical Report BAW-10010, "Stability Margin for Xenon Oscillations."

The first two parts of BAW-10010, which considered modal and one-dimensional digital analyses, pointed out the need for multi-dimensional calculations regarding xenon stability. These latter calculations are particularly important since there are no PWR's on line at the present time of the size and power rating of the plant described here. Therefore, the reactor core designs
16. for Oconee Units 1 and 2, Cycle 1, have been analyzed in three dimensions with thermal feedback. For the Unit 1 operating core at beginning of life, the predicted azimuthal stability index is -0.07 hr^{-1}. Using modal analysis with the three-dimensional results shows that the shape factor must be approximately 50% flat for the power coefficient of -5.05×10^{-6} as calculated by previously described methods. Since the curves in Part 1 of BAW-10010 were generated for a power coefficient of -3.92×10^{-6} $\Delta\rho$/MWt, it was necessary to generate two new curves for azimuthal stabilitv. These curves are shown in Figures 3-6i and 3-6j. From Figure 3-6i the threshold (i.e., stability index = 0) moderator coefficient for the nominal case is approximately $+3 \times 10^{-4}$ $\Delta\rho$/°Fm. Including compounded errors from Figures 3-6j, the threshold moderator coefficient is approximately $+1 \times 10^{-4}$ $\Delta\rho$/°F. Using the least favorable predictions of the Doppler and moderator coefficients, a stability index of -0.067 hr^{-1} is obtained. This corresponds to a power coefficient of -4.73×10^{-6} $\Delta\rho$/MWt. For the Unit 2 operating core at beginning of life (96 FPH), the predicted azimuthal stability index is -.085 hr^{-1}. Again, using modal analysis combined with three-dimensional results shows the shape factor to be approximately 40% flat for the calculated power coefficient of -4.67×10^{-6} $\Delta\rho$/MWt. Azimuthal stability curves for the nominal and compounded error cases are shown in Figures 3-6l and 3-6m, respectively. From Figure 3-6l the nominal threshold moderator coefficient extrapolates to approximately $+5 \times 10^{-4}$ $\Delta\rho$/°Fm. When compounded errors are considered as in Figure 3-6m, the threshold moderator coefficient is approximately $+2.5 \times 10^{-4}$ $\Delta\rho/°F_m$.

3.2.3 THERMAL AND HYDRAULIC DESIGN AND EVALUATION

3.2.3.1 Thermal and Hydraulic Characteristics

3.2.3.1.1 Fuel Assembly Heat Transfer Design

a. Design Criteria

The criterion for the heat transfer design is to be safely below Departure From Nucleate Boiling (DNB) at the design overpower (114 per cent of rated power). The analysis is described in detail in 3.2.3.2.2, Statistical Core Design Technique.

The input information for the statistical core design technique and for the evaluation of individual hot channels is as follows:

1. Heat transfer critical heat flux equations and data correlations.

2. Nuclear power factors.

3. Engineering hot channel factors.

4. Core flow distribution hot channel factors.

5. Maximum reactor overpower.

These inputs have been derived from test data, physical measurements, and calculations as outlined below.

b. Heat Transfer Equation and Data Correlation

The heat transfer relationship used to predict limiting heat transfer conditions is presented in References 13 and 14. The equations are as follows:

1. W-3 uniform flux DNB correlation for single channel with all walls heated:

$$\frac{Q''_{DNB,eu}}{10^6} = \{(2.022 - 0.0004302\ P)$$

$$+ (0.1722 - 0.0000984\ P) \exp [(18.177 - 0.004129\ P)\chi]\}$$

3.

$$\times \left[\left(0.1484 - 1.596\ \chi + 0.1729\ \chi\ |\chi|\right)\frac{G}{10^6} + 1.037\right]$$

$$\times [1.157 - 0.869\ \chi] \times [0.2664 + 0.8357 \exp (-3.151\ D_e)]$$

$$\times [0.8258 + 0.000784(H_{sat} - H_{in})]$$

where Q'' = flux, Btu/h-ft^2

P = pressure, psia

G = mass velocity, lb/h-ft^2

X = quality, expressed as fraction

D_e = equivalent diameter, in.

H = enthalpy, Btu/lb

2. W-3 nonuniform flux DNB correlation for single channel with all walls heated:

$$Q''_{DNB,N} = Q''_{DNB,eu}/F$$

where $Q''_{DNB,N}$ = DNB heat flux for the nonumiformly heated channel

$Q''_{DNB,eu}$ = equivalent uniform DNB flux

$$F = \left\{\frac{C}{Q''_{\text{local at } \ell_{DNB}}\left[1 - \exp\left(-C\,\ell_{DNB}\right)\right]}\right\}$$

$$x \left\{\int_0^{\ell_{DNB}} Q''(Z) \exp\left[-C(\ell_{DNB} - Z)\right] dZ\right\}$$

$$C = 0.44 \frac{(1 - \chi_{DNB})^{7.9}}{\left(\frac{G}{10^6}\right)^{1.72}} \text{ in.}^{-1}$$

where ℓ_{DNB} = Distance from the inception of local boiling to the point of DNB, in.

Z = Distance from the inception of local boiling measured in the direction of flow, in.

4. A new C-factor has been reported in the literature[14a] which is the following:

$$C = 0.15 \frac{(1 - \chi)^{4.31}}{(G/10^6)^{0.478}} \text{ in.}^{-1}$$

A comparison was made between the two C-factors for their effects on the DNB ratio. This is shown in Figure 3-6k for the design overpower condition.
4. Since the difference is small, and since all the DNBR analyses were completed when the new C-factor was published, the DNB ratios reported herein will be based on the original C-factor.

3. W-3 uniform flux DNB correlation for single channel with unheated walls:

$$\frac{Q''_{DNB,\ \text{with unheated wall}}}{Q''_{DNB,\ \text{using } D_h \text{ to replace } D_e}} = (1.36 + 0.12\ e^{9\ \chi}) \times (1.2 - 1.6\ e^{-1.92\ D_h}) \times (1.33 - 0.237\ e^{5.66\ \chi})$$

where D_e = equivalent diameter based on all the wetted perimeter, in.

D_h = equivalent diameter based on only the heated perimeter, in.

Individual channels are analyzed to determine a DNB ratio, i.e., the ratio of the heat flux at which a DNB is predicted to occur to the heat flux in the channel being investigated. This DNB ratio is related to the data correlation as shown in Figure 3-7. A confidence and population value is associated with every DNB ratio as described in the Statistical Core Design Technique (3.2.3.2.2). The plot of DNB versus population is for a confidence of 99 per cent. The criterion for evaluating the thermal design margin for individual channels or the total core is the confidence-population relationship. The DNB ratios required to meet the basic criteria or limits are a function of the experimental data and heat transfer correlation used, and vary with the quantity and quality of data. The recommended mimimum design DNB ratio for the W-3 correlation is 1.30.

The DNB and population relationship for a design limit of 1.30 in the hot channel corresponds to a 99 per cent confidence that at least 94.3 per cent of the population of all such hot channels is in no jeopardy of experiencing a DNB. The DNB ratios and the fraction of the core in no jeopardy of experiencing a DNB at design conditions are considerably higher than those given in the design limits outlined in 3.1.2.3.

The relationship of sustained DNB to core burnout conditions is discussed in B&W Company Topical Report BAW-10014, "Analysis of Sustained Departure From Nucleate Boiling Operation." The analysis shows that DNB conditions are very local and there is little likelihood of sudden fuel rod failures or a propagation of DNB conditions.

c. Nuclear Power Factors

The heated surfaces in every flow channel in the core are examined for heat flux limits. The heat input to the fuel rods in a coolant channel is determined from a nuclear analysis of the core and fuel assemblies. The results of this analysis are as follows:

1. The nominal nuclear peaking factors for the worst time in core life are

$$F\Delta h = 1.77$$

$$Fz = 1.70$$

$$Fq = 3.01$$

2. The design nuclear peaking factors for the worst time in core life are

$$F\Delta h = 1.78$$

$$Fz = 1.70$$

$$Fq = 3.03$$

where

$F\Delta h$ = max/avg total power ratio (radial x local nuclear)

Fz = max/avg axial power ratio (nuclear)

$Fq = F\Delta h \times Fz$ (nuclear total)

The nominal values are the maximum values calculated with nominal spacing of fuel assemblies. The design values are obtained by examining maximum, nominal, and minimum fuel assembly spacing and determining the worst values for the combined effect of flow and rod peaking.

The axial nuclear factor, Fz, is illustrated in Figure 3-8. The distribution of power expressed as $P/\overline{P}$ is shown for two conditions of reactor operation. The first condition is an inlet peak with a max/avg value of 1.70 resulting from partial insertion of a CRA group for transient control following a power level change. This condition results in the maximum local heat flux and maximum linear heat rate. The second power shape is a symmetrical cosine which is indicative of the power distribution with xenon override rods withdrawn. The flux peak max/avg value is 1.50 in the center of the active core. Both of these flux shapes have been evaluated for thermal DNB limitations. The limiting condition is the 1.50 cosine power distribution. The inlet peak shape has a larger maximum value. However, the position of the 1.50 cosine peak farther up the channel results in a less favorable flux to enthalpy relationship. This effect has been demonstrated in DNB tests of nonuniform flux shapes.[15] The 1.50 cosine axial shape has been used to determine individual channel DNB limits and to make the associated statistical analysis.

The nuclear factor for total radial x local rod power, FΔh, is calculated for each rod in the core. A distribution curve of the fraction of the core fuel rods operating above various peaking factors is shown in Figure 3-9 for a typical fuel cycle condition with the maximum fuel rod peaking factor of 1.78.

d. Engineering Hot Channel Factors

Power peaking factors obtained from the nuclear analysis are based on mechanically perfect fuel assemblies. Engineering hot channel factors are used to describe variations in fuel loading, fuel and clad dimensions, and flow channel geometry from perfect physical quantities and dimensions.

The application of hot channel factors is described in detail in 3.2.3.2.2, Statistical Core Design Technique. The factors are determined statistically from fuel assembly as-built or specified data where F_Q is a heat input factor, $F_{Q''}$ is a local heat flux factor at a hot spot, and F_A is a flow area reduction factor describing the variation in coolant channel flow area. Several subfactors are combined statistically to obtain the final values for F_Q, $F_{Q''}$, and F_A. These subfactors are shown in Table 3-10. The factor, the coefficient of variation, the standard deviation, and the mean value are tabulated.

Table 3-10
Coefficients of Variation

16. (Reference Supplement 9 Revisions for Oconee 2 and 3)

	CV No.	Description	Standard Deviation of Variable (σ)	Mean Value of Variable ($\bar{x}$)	Coefficient of Variation ($\sigma/\bar{x}$)
	1	Flow Area			
		Interior Bundle Cells	0.00190	0.17740	0.01071
		Peripheral Bundle Cells	0.00346	0.21546	0.01606
16.	2	Local Rod Diameter	0.000647	0.430	0.00150
	3	Average Rod Diameter (Die-Drawn, Local and Average Same)	0.000647	0.430	0.00150
	4	Local Fuel Loading			0.00698
		Subdensity	0.00647	0.935	0.00692
16.		Subfuel Area (Diameter Effect)	0.000094	0.1075	0.00087
	5	Average Fuel Loading			0.00557
		Subdensity	0.00485	0.935	0.00519
		Sublength	0.26294	144	0.00183
		Subfuel Area (Diameter Effect)	0.000094	0.1075	0.00087
16.	6	Local Enrichment	0.00421	2.00	0.00211
	7	Average Enrichment	0.00421	2.00	0.00211

Enrichment values are for worst case normal assay batch; maximum variation occurs for minimum enrichment.

e. Core Flow Distribution Hot Channel Factors

The physical arrangement of the reactor vessel internals and nozzles results in a nonuniform distribution of coolant flow to the various fuel assemblies. Reactor internal structures above and below the active core are designed to minimize unfavorable flow distribution. A 1/6 scale model test of the reactor and internals was performed to demonstrate the adequacy of the internal arrangements. The results of the test have confirmed the adequacy of the design values used.

A flow distribution factor is determined for each fuel assembly location in the core. The factor is expressed as the ratio of fuel assembly flow to average fuel assembly flow. The finite values of the ratio are greater or less than 1.0 depending on the position of the assembly being evaluated. The flow in the central fuel assemblies is in general larger than the flow in the outermost assemblies due to the inherent flow characteristics of the reactor vessel.

The flow distribution factor is related to a particular fuel assembly location and the quantity of heat being produced in the assembly. A flow-to-power comparison is made for all of the fuel assemblies. The worst condition in the hottest fuel assembly is determined by applying model test isothermal flow distribution data and heat input effects at power as outlined in 3.2.3.2.3. Two assumptions for flow distribution have been made in the thermal analysis of the core as follows:

1. For the maximum design condition and for the analysis of the hottest channel, all fuel assemblies receive minimum flow for the worst power condition.

2. For the most probable design conditions predicted, average flows have been assigned for each fuel assembly consistent with location and power.

f. Maximum Reactor Design Overpower

Core performance is assessed at the maximum design overpower. The selection of the design overpower is based on an analysis of the reactor protection system as described in Section 7. The reactor trip point is 107.5 per cent rated power, and the maximum overpower, which is 114 per cent, will not be exceeded under any conditions.

g. Maximum Design Conditions Analysis Summary

The Statistical Core Design Technique described in 3.2.3.2 was used to analyze the reactor at the maximum design conditions described previously. The total number of fuel rods in the core that have a possibility of reaching DNB is shown in Figure 3-10 for 100 to 122.5 per cent overpower for the maximum design conditions. Point B on Line 1 is the maximum design point for 114 per cent power with the design $F\Delta h$ nuclear of 1.78 and minimum flow to every channel in the core. This Point B forms the basis for this statistical statement:

There is a 99 per cent confidence that at least 99.96 per cent of the fuel rods in the core are in no jeopardy of experiencing a departure from nucleate boiling (DNB) during continuous operation at the design overpower of 114 per cent.

At 100 per cent power (2,568 MWt) as shown by Point A, the statistical number of fuel rods in jeopardy is less than one, resulting in a population protected in excess of 99.997 per cent. The limit imposed by a W-3 DNB ratio of 1.3 is 70 fuel rods in jeopardy or a population protected of 99.810 per cent.

An additional analysis of fuel rods in jeopardy for all the maximum design conditions except fuel assembly flow distribution is shown as Line 2 in Figure 3-10. Each assembly was assumed to receive average flow for the assembly power conditions. The number of rods that have a possibility of a DNB are within the bounds of the two lines. Statistical results for the maximum design condition calculation, shown by Figure 3-10, may be summarized as follows (see Table 3-11).

Table 3-11
DNB Results - Maximum Design Condition
(99% Confidence Level)

Point	Power % of 2,568 MWt	$F_{\Delta h}$	No. of Channels Possible DNB	Population Protected, %	Hot Channel DNB Ratio (W-3)
A	100	1.78	<1	>99.997	2.00
B	114	1.78	14.0	99.962	1.55
C	122.5	1.78	70.0	99.810	1.30

h. Most Probable Design Condition Analysis Summary

The previous maximum design calculation, as shown by Line 1, Figure 3.10, indicates the total number of rods that may be in jeopardy when it is conservatively assumed that every rod in the core has the mechanical and heat transfer characteristics of a hot channel as described in 3.2.3.2.2. For example, all channels are analyzed with F_A (flow area factor) less than 1.0, F_Q (heat input factor) greater than 1.0, and with minimum fuel assembly flow. It is physically impossible for all channels to have hot channel characteristics. A more relistic indication of the number of fuel rods in jeopardy may be obtained by the application of the statistical heat transfer data to average rod power and mechanical conditions.

An analysis for the most probable conditions has been made based on the average conditions described in 3.2.3.2.2. The results of this analysis are shown in Figure 3-11. The analysis may be summarized as follows (Table 3-12):

Table 3-12
DNB Results - Most Probable Condition

Point	Power % of 2,568 MWt	FΔh	No. of Channels Possible DNB	Population Protected, %	Hot Channel DNB Ratio (W-3)
D	100	1.78	<1	>99.997	2.25
E	114	1.78	2.5	99.993	1.82
F	122.5	1.78	9.0	99.976	1.54

The analysis was made from Point D at 100 per cent power to Point F at 122.5 per cent power to show the sensitivity of the analysis with power. The worst condition expected is indicated by Point E at 114 per cent power where it is shown statistically that there is a small possibility that 2.5 fuel rods may be subject to a departure from nucleate boiling (DNB). This result forms the basis for the following statistical statement for the most probable design conditions:

> There is at least a 99 per cent confidence that at least 99.993 per cent of the rods in the core are in no jeopardy of experiencing a DNB, even with continuous operation at the design overpower of 114 per cent.

i. Distribution of the Fraction of Fuel Rods Protected

The distribution of the fraction (P) of fuel rods that have been shown statistically to be in no jeopardy of a DNB has been calculated for the maximum design and most probable design conditions. The computer programs used provide an output of (N) number of rods and (P) fraction of rods that will not experience a DNB grouped for ranges of (P). The results for the most probable design condition are shown in Figure 3-12.

The population protected, (P), and the population in jeopardy, (1 - P), are both plotted. The integral of (1 - P) and the number of fuel rods gives the number of rods that are in jeopardy for given conditions as shown in Figures 3-10 and 3-11. The number of rods is obtained from the product of the percentage times the total number of rods being considered (36,816). Two typical distributions shown in Figure 3-12 are for the most probable condition analysis of Points E and F on Figure 3-11. The lower line of Figure 3-12 shows P and (1 - P) at the 114 per cent power condition represented by Point E of Figure 3-11. The upper curve shows P and (1 - P) at the 122.5 per cent power condition represented by Point F of Figure 3-11. The integral of N and (1-P) of the lower curve forms the basis for the statistical statement at the most probable design condition described in (h) above.

j. Hot Channel Performance for Four Pump Operation

The hottest unit cell with all surfaces heated has been examined for hot channel factors, DNB ratios, and quality for a range of reactor powers. The cell has been examined for the maximum value of $F\Delta h$ nuclear of 1.78. The hot channel was assumed to be located in a fuel assembly with 95 per cent of the average fuel assembly flow. The heat generated in the fuel is 97.3 per cent of the total nuclear heat. The remaining 2.7 per cent is assumed to be generated in the coolant as it proceeds up the channel within the core and is reflected as an increase in ΔT of the coolant.

Error bands of ±65 psi operating pressure and ±2 F are reflected in the total core and hot channel thermal margin calculations in the direction producing the lowest DNB ratios or highest qualities. The DNB ratio versus power is shown in Figure 3-13. The DNB ratio in the hot channel at the maximum overpower of 114 per cent is 1.55 which corresponds to a 99 per cent confidence that at least 99.34 per cent of the fuel channels of this type are in no jeopardy of experiencing a DNB. The engineering hot channel factors used in the design analysis are described in 3.2.3.2.2 and listed below:

$$F_Q = 1.011$$

$$F''_Q = 1.014$$

$$F_A = 0.98 \text{ (Interior Cells)}$$

$$F_A = 0.97 \text{ (Wall Cells)}$$

The hot channel exit quality for various powers is shown in Figure 3-14. The combined results may be summarized for 2,120 psig as follows:

Reactor Power, %	DNB Ratio (W-3)	Exit Quality, %
100	2.00	0.9
107.5 (Trip Setting)	1.75	3.4
114 (Maximum Power)	1.55	5.8
122.5	1.30	9.2

3.

9. k. Hot Channel Performance Summary for Partial Pump Flow
(Reference Supplement 6 statement for Oconee 1.)

The power limitations imposed on the reactor due to the loss of one or more pumps has been examined by studying DNB ratios and quality in the hot channel for a range of reactor powers. The system parameters used in the analysis are the same as the ones discussed in

3.2.3.1.1 j. A constant reactor vessel average temperature of 578.9 °F was used to determine inlet temperature. The DNB ratio versus power for the flow conditions caused by loss of pumps is illustrated in Figure 3-15. The hot channel quality at the minimum DNB ratio point (X_{DNB}) at the same conditions is shown in Figure 3-16.

Additional limits have been placed on the analysis. One is the rec-commended minimum DNB ratio of 1.3; the others are limitations on the W-3 critical heat flux correlation itself (i.e. ±15% on quality and $G \geq 1.0 \times 10^6$ lb/hr-ft^2). The first limitation reached is used to establish the maximum power. The limitation for the three pump case is the minimum DNBR of 1.3 while for both two pump cases the maximum power is limited to 15% quality. The analysis for the one pump case was not to obtain the maximum design power but rather to evaluate an es-
4. tablished power level of 30%. The DNBR was calculated for this case even though the mass velocity is below the W-3 correlation limit. The mass velocity for the one pump case is greater than 0.5×10^6 lb/hr-ft^2 and the coolant condition is highly subcooled. Table 3-13 below summarizes the power limitations on reactor operation at 2,120 psig as defined by the hot channel conditions. The overpower margins and system limitations are discussed in 4.1.1.3. The steady state rated power level is determined by dividing the maximum design overpower by the desired overpower margin.

Table 3-13
Hot Channel Performance Vs Pumps in Service

Reactor Coolant Pumps Operating	3 Pumps	2 Pumps (2 Loops)	2 Pumps (1 Loop)	1 Pump
Hot Channel DNBR at Max. Design Overpower	1.30	1.40	1.45	4.80
Hot Channel Quality at Min. DNBR Point, %	7.0	15.0	15.0	-8.0
Reactor Coolant Flow, % of Rated	74.7	49.0	45.8	21.9
4. Maximum Design Overpower, % of Rated Power	101.5	77.0	70.5	30.0

3.2.3.1.2 Fuel and Cladding Thermal Conditions

a. Fuel

A digital computer code is used to calculate the fuel temperature. The program uses nonuniform volumetric heat generation across the fuel diameter, and external coolant conditions and heat transfer coefficients determined for thermal-hydraulic channel solutions. The fuel thermal conductivity is varied in a radial direction as a function of the temperature variation.

Values for fuel conductivity were used as shown in Figure 3-17, a plot of fuel conductivity versus temperature. The heat transfer from the fuel to the clad is calculated with a fuel and clad expansion model proportional to temperature. The temperature drop is calculated using gas conductivity at the beginning-of-life conditions when the gas conductivity is 0.09 Btu-ft/h-F-ft^2. The gas conduction model is used in the calculation until the fuel thermal expansion relative to the clad closes the gap to a dimension equivalent to a contact coefficient. The contact coefficient is dependent upon pressure and gas conductivity.

A plot of fuel center temperature versus linear heat rate in kW/ft is shown in Figure 3-18 for beginning-of-life conditions for the maximum cold diametral clearance of 0.0085 in. The linear heat rate at the maximum overpower of 114 per cent is 20.1 kW/ft. The hot spot center and core average fuel temperatures at 100 per cent power are 4,250 and 1,540 as shown in Table 3-1.

The peaking factors used in the calculation are

$$F\Delta h = 1.78$$

$$F_z = 1.70$$

$$F''_Q = 1.03$$

$$F_q \text{ (nuc and mech)} = 3.12$$

A conservative value of 1.03 was assumed for the local hot spot heat flux peaking factor, F''_Q. The assigned value corresponds to a 99 per cent population-protected relationship as described in the statistical technique, and shown in Figure 3-19.

b. Clad

The assumptions in the preceding paragraph were applied in the calculation of the clad surface temperature at the maximum overpower. Boiling conditions prevail at the hot spot, and the Jens and Lottes relationship[16] for the coolant-to-clad ΔT for boiling was used to determine the clad temperature. The resulting maximum calculated clad surface temperature is 654 F at a system operating pressure of 2,185 psig.

3.2.3.1.3 End of Life Clad Transients

An investigation was carried out analyzing the ability of the cladding to withstand various end of life transients which, though not considered normal, could occur during the life of the plant. The specific transients examined were loss of flow at 108% power, and power excursions up to 114%. The latter value is the maximum power attainable with the upper trip point setting of 107.5%. The effects of internal cladding pressure and system pressure on the integrity of the cladding during the normal shutdown for refueling and due to depressurization transients were also examined.

For the flow coastdown analysis, temperatures for the fuel and cladding during a coastdown from 108% power were obtained. Even starting from this overpower condition, there was no rise in temperature in either the fuel or cladding following the loss of pumping power. The fact that the pumps have been designed to include a rotational inertial equivalent to 70,000 lb-ft^2 allows them to provide sufficient flow after the loss of power to avoid temperature increases and to maintain the DNB ratio for the hot channel at a value higher than the DNB ratio for continuous operation at the maximum overpower condition.

A power excursion transient to 114% was also considered. Since the DNB analysis has been done for steady-state operation at this power level, DNB was not a consideration. For this transient it was expected that a greater release rate of fission gases, and consequently a greater internal pressure and stress of the cladding during the excursion would occur. The analysis of the internal pressure buildup and stress and strain in the cladding at 114% overpower was carried out with conservative assumptions. For example, the design is based on a peak burnup of 55,000 megawatt days per ton whereas the maximum calculated burnup is approximately 42,000 megawatt days per ton. Other equally conservative assumptions were made in the analysis. At the end of life, the calculated internal gas pressure does not exceed system pressure. The maximum tensile stress due to the fission gas pressure is less than 10% of yield strength.

During a normal reduction in power, or a cooldown for refueling, the internal gas temperature and pressure decrease. At no time does the pressure differential exceed the external design pressure differential at beginning of life, or the internal design pressure at the end of life.

An investigation of depressurization transients indicates that in the limit, where coolant pressure is arbitrarily assumed to drop instantaneously to zero, the cladding stress due to internal pressure is less than yield strength. The design internal pressure in a hot fuel rod is 3,300 psi. (Calculated fission
3. gas pressure is less--about 2,200 psi at end of life in a high burnup rod as described in 3.2.3.2.3 h.) The internal pressure required to cause clad stress equal to the yield strength at the hot spot in the core at overpower conditions
3. (average clad temperature 703 °F) is 4,500 psi. It is concluded that any conceivable depressurization transient cannot damage the cladding.

3.2.3.2 Thermal and Hydraulic Evaluation

3.2.3.2.1 Introduction

Summary results for the characteristics of the reactor design are presented in 3.2.3.1. The Statistical Core Design Technique employed in the design represents a refinement in the methods for evaluating pressurized water reactors. Corresponding single hot channel DNB data were presented to relate the new method with previous criteria. A comprehensive description of the new technique is included in this section to permit a rapid evaluation of the methods used.

A detailed evaluation and sensitivity analysis of the design has been made by examining the hottest channel in the reactor for DNB ratio, quality, and fuel temperature. The W-3 correlation has been used in this analysis.

3.2.3.2.2 Statistical Core Design Technique

The core thermal design is based on a Statistical Core Design Technique developed by B&W. The technique offers many substantial improvements over older methods, particularly in design approach, reliability of the result, and mathematical treatment of the calculation. The method reflects the performance of the entire core in the resultant power rating and provides insight into the reliability of the calculation. This section discusses the technique in order to provide an understanding of its engineering merit.

The statistical core design technique considers all parameters that affect the safe and reliable operation of the reactor core. By considering each fuel rod, the method rates the reactor on the basis of the performance of the entire core. The result then will provide a good measure of the core safety and reliability since the method provides a statistical statement for the total core. This statement also reflects the conservatism or design margin in the calculation.

A reactor safe operating power has always been determined by the ability of the coolant to remove heat from the fuel material. The criterion that best measures this ability is the DNB, which involves the individual parameters of heat flux, coolant temperature rise, and flow area, and their inter-effects. The DNB criterion is commonly applied through the use of the departure from nucleate boiling ratio (DNBR). This is the minimum ratio of the DNB heat flux (as computed by the DNB correlation) to the surface heat flux. The ratio is a measure of the margin between the operating power and the power at which a DNB might be expected to occur in that channel. The DNBR varies over the channel length, and it is the minimum value of the ratio in the channel of interest that is used.

The calculation of DNB heat flux involves the coolant enthalpy rise and coolant flow rate. The coolant enthalpy rise is a function of both the heat input and the flow rate. It is possible to separate these two effects; the statistical hot channel factors required are a heat input factor, F_Q, and a flow area factor, F_A. In addition, a statistical heat flux factor, F''_Q is required; the heat flux factor statistically describes the variation in surface heat flux. The DNBR is most limiting when the burnout heat flux is based on minimum flow area (small F_A) and maximum heat input (large F_Q), and when the surface heat flux is large (large F''_Q). The DNB correlation is provided in a best-fit form, i.e., a form that best fits all of the data on which the correlation is based. To afford protection against DNB, the DNB heat flux computed by the best-fit correlation is divided by a DNB factor (BF) greater than 1.0 to yield the design DNB surface heat flux. The basic relationship

$$\mathrm{DNBR} = \frac{Q''_{\mathrm{DNB}}}{\mathrm{BF}} \times f(F_A, F_Q) \times \frac{1}{Q''_{\mathrm{surface}} \times F_{Q''}}$$

involves as parameters statistical hot channel and DNB factors. The DNB factor (BF) above is usually assigned a value of unity when reporting DNB ratios so that the margin at a given condition is shown directly by a DNBR greater than 1.0, i.e., 1.30 in the hot channel.

Selected heat transfer data are analyzed to obtain a correlation. Since thermal and hydraulic data generally are well represented with a Gaussian (normal) distribution (Figure 3-20), mathematical parameters that quantitatively rate the correlation can be easily obtained for the histogram. These same mathematical parameters are the basis for the statistical burnout factor (BF).

In analyzing a reactor core, the statistical information required to describe the hot channel subfactors may be obtained from data on the as-built core, from data on similar cores that have been constructed, or from the specified tolerances for the proposed core. The design factors can be shown graphically (Figures 3-19 and 3-21).

All the plots have the same characteristic shape whether they are for subfactors, hot channel factors, or burnout factor. The factor increases with either increasing population or confidence. The value used for the statistical hot channel and burnout factor is a function of the percentage of confidence desired in the result, and the portion of all possibilities desired, as well as the amount of data used in determining the statistical factor. A frequently used assumption in statistical analyses is that the data available represent an infinite sample of that data. The implications of this assumption should be noted. For instance, if limited data are available, such an assumption leads to a somewhat optimistic result. The assumption also implies that more information exists for a given sample than is indicated by the data; it implies 100 per cent confidence in the end result. The B&W calculational procedure does not make this assumption, but rather uses the specified sample size to yield a result that is much more meaningful and statistically rigorous. The influence of the amount of data for instance can be illustrated easily as follows: Consider the heat flux factor which has the form

$$F_{Q''} = 1 + K\sigma_{F_{Q''}}$$

where

$F_{Q''}$ = the statistical hot channel factor for heat flux

K = a statistical multiplying factor

$\sigma_{F_{Q''}}$ = the standard deviation of the heat flux factor, including the effects of all the subfactors

If $\sigma_{F_{Q''}} = 0.05$ for 300 data points, then a K factor of 2.608 is required to protect 99 per cent of the population. The value of the hot channel factor then is

$$F_{Q''} = 1 + (2.608 \times 0.050) = 1.1304$$

and will provide 99 per cent confidence for the calculation. If, instead of using the 300 data points, it is assumed that the data represent an infinite sample, then the K factor for 99 per cent of the population is 2.326. The value of the hot channel factor in this case is

$$F_{Q''} = 1 + (2.326 \times 0.050) = 1.1163$$

which implies 100 per cent confidence in the calculation. The values of the K factor used above are taken from SCR-607.[17] The same basic techniques can be used to handle any situation involving variable confidence, population, and number of points.

Having established statistical hot channel factors and statistical DNB factors, we can proceed with the calculation in the classical manner. The statistical factors are used to determine the minimum fraction of rods protected, or that are in no jeopardy of experiencing a DNB at each nuclear power peaking factor. Since this fraction is known, the maximum fraction in jeopardy is also known. It should be recognized that every rod in the core has an associative DNB ratio that is substantially greater than 1.0, even at the design overpower, and that theoretically no rod can have a statistical population factor of 100 per cent, no matter how large its DNB ratio.

Since both the fraction of rods in jeopardy at any particular nuclear power peaking factor and the number of rods operating at that peaking factor are known, the total number of rods in jeopardy in the whole core can be obtained by simple summation. The calculation is made as a function of power, and the plot of rods in jeopardy versus reactor overpower is obtained (Figure 3-11). The summation of the fraction of rods in jeopardy at each peaking factor summed over all peaking factors can be made in a statistically rigorous manner only if the confidence for all populations is identical. If an infinite sample is not assumed, the confidence varies with population. To form this summation then, a conservative assumption is required. B&W total core model assumes that the confidence for all rods is equal to that for the least-protected rod, i.e., the minimum possible confidence factor is associated with the entire calculation.

The result of the foregoing technique, based on the <u>maximum design</u> conditions (114 per cent power), is this statistical statement:

> There is at least a 99 per cent confidence that at least 99.96 per cent of the rods in the core are in no jeopardy of experiencing a DNB, even with continuous operation at the design overpower.

The maximum design conditions are represented by these assumptions:

a. The maximum design values of FΔh (nuclear max/avg total fuel rod heat input) are obtained by examining the maximum, nominal, and minimum fuel assembly spacing and determining the resulting maximum value for
1. fuel rod peaking.

b. The maximum value of F_z (nuclear max/avg axial fuel rod heat input) is determined for the limiting transient or steady-state condition.

c. Every coolant channel in the core is assigned to have less than the nominal flow area represented by engineering hot channel factors, F_A, less than 1.0.

d. Every channel is assumed to receive the minimum flow associated with core flow maldistribution.

e. Every fuel rod in the core is assumed to have a heat input greater than the maximum calculated value. This value is represented by engineering hot channel heat input factors, F_Q and F_Q'', which are greater than 1.0.

f. Every channel and associated fuel rod has a heat transfer margin above the experimental best-fit limits reflected in DNB ratios greater than 1.0 at maximum overpower conditions.

The statistical core design technique may also be used in a similar manner to evaluate the entire core at the most probable mechanical and nuclear conditions to give an indication of the most probable degree of fuel element jeopardy. The result of the technique based on the most probable design conditions leads to a statistical statement which is a corollary to the maximum design statement:

> There is at least a 99 per cent confidence that at least 99.993 per cent of the rods in the core are in no jeopardy of experiencing a DNB, even with continuous operation at the design overpower.

The most probable design conditions are assumed to be the same as the maximum design conditions with these exceptions:

a. Every coolant channel is assumed to have the nominal flow area ($F_A = 1.0$).

b. Every fuel rod is assumed to have (1) the maximum calculated value of heat input, and (2) F_Q and F_Q'' are assigned values of 1.0.

c. The flow in each coolant channel is based on a power analysis without flow maldistribution factors.

d. Every fuel rod is assumed to have a nominal value for $F\Delta h$ nuclear.

The full meaning of the maximum and most probable design statements requires additional comment. As to the 0.04 per cent or 0.007 per cent of the rods not included in the statements, statistically it can be said that no more than 0.04 per cent or 0.007 per cent of the rods will be in jeopardy, and that in general the number in jeopardy will be fewer than 0.04 per cent or 0.007 per cent. The statements do not mean to specify a given number of DNBs, but only acknowledge the possibility that a given number could occur for the 114 per cent overpower conditions assumed. Analyses for 100 per cent rated power conditions reported in Tables 3-11 and 3-12 show that essentially none of the fuel rods is subject to a DNB.

In summary, the calculational procedure outlined here represents a substantially improved design technique in two ways:

a. It reflects the performance and safety of the entire core in the resultant power rating by considering the effect of each rod on the power rating.

b. It provides information on the reliability of the calculation and, therefore, the core through the statistical statement.

3.2.3.2.3 Evaluation of the Thermal and Hydraulic Design

a. Hot Channel Coolant Quality and Void Fraction

An evaluation of the hot channel coolant conditions provides additional confidence in the thermal design. Sufficient coolant flow has been provided to insure low quality and void fractions. The quality in the hot channel versus reactor power is shown in Figure 3-14. The sensitivity of channel outlet quality with pressure and power level is shown by the 2,185 and 2,120 psig system pressure conditions examined. These calculations were made for an $F\Delta h$ of 1.78. Additional calculations for a 10 per cent increase in $F\Delta h$ to 1.96 were made at 114 per cent power. The significant results of both calculations are summarized in Table 3-14.

Table 3-14
Hot Channel Coolant Conditions

Power, %	$F\Delta h$	Exit Quality, %	Exit Void Fraction, %	Operating Pressure, psig
100	1.78	(-)1.3[b]	0.9[a]	2,185
114	1.78	3.4	12.8	2,185
128	1.78	8.7	31.2	2,185
114	1.96	7.1	32.2	2,185
100	1.78	0.9	2.8	2,120
114	1.78	5.8	26.3	2,120
128	1.78	11.3	39.8	2,120
114	1.96	9.5	41.1	2,120

[a] Subcooled voids.

[b] Negative indication of quality denotes subcooling.

The conditions of Table 3-14 were determined with all of the hot channel factors applied. Additional calculations were made for unit cell channels without engineering hot channel factors to show the coolant conditions more likely to occur in the reactor core. A value for $F\Delta h$ of 1.78 was examined with and without fuel assembly flow distribution hot channel factors at 2,185 psig as shown on Figure 3-22. These results show that the exit qualities from the hottest cells are lower than the maximum design conditions.

b. Core Void Fraction

The core void fractions were calculated at 100 per cent rated power for the normal operating pressure of 2,185 psig and for the minimum operating pressure of 2,120 psig. The influence of core fuel assembly flow distribution was checked by determining the total voids for both 100 and 95 per cent total core flow for the two pressure conditions. The results are as follows:

Flow, %	Pressure, psig	Core Void Fraction, %
100	2,185	0.06
100	2,120	0.194
95	2,185	0.201
95	2,120	0.770

The most conservative condition of 95 per cent flow at 2,120 psig re-
1. sults in no more than 0.77 per cent void volume in the core. Con-
servative maximum design values were used to make the calculation.

The void program uses a combination of Bowring's[18] model with Zuber's[19] correlation between void fraction and quality. The Bowring model considers three different regions of forced convection boiling. They are:

1. <u>Highly Subcooled Boiling</u>

   In this region, the bubbles adhere to the wall while moving upward through the channel. This region is terminated when the subcooling decreases to a point where the bubbles break through the laminar sublayer and depart from the surface. The highly subcooled region starts when the surface temperature of the clad reaches the surface temperature predicted by the Jens and Lottes equation. The highly subcooled region ends when

$$T_{sat} - T_{bulk} = \frac{\eta\phi}{V} \qquad \text{(A)}$$

where ϕ = local heat flux, Btu/hr-ft^2

$\eta = 1.863 \times 10^{-5} (14 + 0.0068p)$

V = velocity of coolant, ft/s

p = pressure, psia

The void fraction in this region is computed in the same manner as Maurer,[20] except that the end of the region is determined by Equation (A) rather than by a vapor layer thickness. The nonequilibrium quality at the end of the region is computed from the void fraction as follows:

$$x_d^* = \frac{1}{1 + \frac{\rho f}{\rho g}\left(\frac{1}{a_d} - 1\right)} \tag{B}$$

where

x_d^* = nonequilibrium quality at end of Region 1

a_d = void fraction at $T_{sat} - T_{bulk} = \frac{\eta\phi}{V}$

ρf = liquid component density, lb/ft^3

ρg = vapor component density, lb/ft^3

2. Slightly Subcooled Boiling

In this region, the bubbles depart from the wall and are transported along the channel (condensation of the bubbles is neglected). This region transcends to a point where the thermodynamic quality is equal to the apparent quality. In general, this is
1. the region of major concern in the design of pressurized water reactors.

The nonequilibrium quality in this region is computed from the following formula:

$$x^* = x_d^* + \frac{P_h}{\dot{m}\, h_{fg}(1 + \varepsilon)} \int_{z_d}^{z} (\phi - \phi_{SP})\,dz \tag{C}$$

where

x^* = nonequilibrium quality in Region 2

h_{fg} = latent heat of vaporization, Btu/lb

$\frac{1}{1 + \varepsilon}$ = fraction of the heat flux above the single phase heat flux that actually goes to producing voids

ϕ_{SP} = single phase heat flux, Btu/h-ft^2

$\dot{m}$ = mass flow rate, lb/h

P_h = heated perimeter, ft

z = channel distance, ft

The void fraction in this region is computed from

$$a = \frac{x^*}{C_o\left[x^* + \rho_g/\rho_f(1 - x^*)\right] + \frac{38.3\ A_f\rho_g}{\dot{m}}\left[\frac{\sigma g g_c(\rho_f - \rho_g)}{\rho_f^2}\right]^{1/4}} \qquad \text{(D)}$$

where

g = acceleration due to gravity, ft/s^2

g_c = constant in Newton's Second Law = 32.17 $\frac{\text{lb m ft}}{\text{lb f s}^2}$

C_o = Zuber's distribution parameter

A_f = flow area, in.^{2}

σ = surface tension

a = void fraction

Equation (D) results from rearranging equations found in Reference 24 and assuming bubbly turbulent flow in determining the relative velocity between the vapor and the fluid. Zuber has shown that Equation (D) results in a better prediction of the void fraction than earlier models based on empirical slip ratios.

3. Bulk Boiling

In this region, the bulk temperature is equal to the saturation temperature, and all the energy transferred to the fluid results in net vapor generation. Bulk boiling begins when the thermodynamic (heat balance) quality, x, is greater than the nonequilibrium quality, x^*. The void fraction in this region is computed using Equation (D) with the thermodynamic quality, x, replacing x^*.

c. Coolant Channel Hydraulic Stability

Flow regime maps of mass flow rate and quality were constructed in order to evaluate channel hydraulic stability. The confidence in the design is based on a review of both analytical evaluations[21-24] and experimental results obtained in multiple rod bundle burnout tests. Bubble-to-annular and bubble-to-slug flow limits proposed by Baker[21] are consistent with the B&W experimental data in the range of interest. The analytical limits and experimental data points have been plotted to obtain the maps for the four different types of cells in the reactor core. These are shown in Figures 3-23, 3-24, 3-25, and 3-26. The experimental data points represent the exit conditions in the various types of channels just previous to the burnout condition for a representative sample of the data points obtained at design operating conditions in the nine rod burnout test assemblies. In all of the bundle tests, the pressure drop, flow rate, and rod temperature traces were repeatable and steady, and did not exhibit any of the characteristics associated with flow instability.

Values of hot channel mass velocity and quality at 114 per cent and 130 per cent power for both nominal and design conditions are shown on the maps. The potential operating points are within the bounds suggested by Baker. Experimental data points for the reactor geometry with much higher qualities then the operating conditions have not exhibited unstable characteristics.

d. <u>Hot Channel DNB Comparisons</u>

DNB ratios for the hottest channel have been determined for the W-3 correlation, and the results are shown in Figure 3-13. DNB ratios are shown for the design 1.50 axial max/avg symmetrical cosine flux shape from 100 to 150 per cent power. The W-3 DNB ratio at the maximum design power of 114 per cent is 1.55. This compares with the suggested W-3 design value of 1.3. A ratio of 1.3 is reached at 122.5 per cent power at an exit quality of 9.2 per cent, which is within the prescribed quality limits of the correlation.

The sensitivity of the DNB ratio with Fz nuclear was examined from 100 to 150 per cent power. The detailed results are labeled in Figure 3-27. A cosine flux shape with an Fz of 1.80 and an FΔh of 1.78 results in a W-3 DNB ratio of 1.30 at 114 per cent power. Similar results are shown for a value of Fz of 1.65 and for the design value of 1.5.

The influence of a change in FΔh was determined by analyzing the hot channel for an FΔh of 1.96. This value is 10 per cent above the maximum design value of 1.78. The resulting W-3 DNB ratio is 1.23 at 114 per cent power. This value is well above the correlation best-fit values of 1.0 for the severe conditions assumed.

e. <u>Reactor Flow Effects</u>

Another significant variable to be considered in evaluating the design is the total system flow. Conservative values for system and reactor pressure drop have been determined to insure that the required system flow is obtained in the as-built plant. The reactor vessel model test and the production pump tests have confirmed the design conditions.

The difference between the reactor system flow and the reactor core
flow is the leakage flow. The leakage flow is defined as that part
of the flow that does not contact the active heat transfer surface
area. This part of the flow exists primarily through three dif-
ferent paths. These paths are (1) through the core shroud, (2)
through the control rod guide tubes and instrument tubes, and (3)
4. between all interfaces separating the inlet and outlet regions.

The leakage flow rates are determined through an iterative process. The total core flow is obtained by taking the difference between the total system flow and an assumed leakage value. Pressure drops are now calculated between the vessel inlet and outlet. These pressure drops are then used to calculate the flow rates through all the predefined leakage paths. The process is repeated until the calculated leakage value is equal to the assumed leakage value.

The flow through the control rod guide tubes is calculated with all
the control rods assumed to be in the fully withdrawn position. Flow
through the interfaces separating the inlet and outlet plenums is
calculated by assuming the maximum tolerances at hot conditions to
result in the minimum flow resistance. Additional conservatism is
applied to determining the leakage flow by adding an allowance of 50%
of the calculated value to account for any uncertainties. The result-
4. ing leakage through the various paths is shown below, based on the
percent of design system flow (131.3×10^6 lb/h).

Path	% of System Flow
1) Shroud	1.6
2) Control rod guide tubes & instrument guide tubes	1.7
3) Inlet to outlet interfaces	0.3
4) Assumed allowance	1.8
Total	5.4

The expected leakage flows have been recalculated and updated as of January 4, 1973 to reflect current core and system pressure drop information and reported as built dimensions.

Leakage flow is defined as that part of the system flow that does not
contact the active heat transfer surface. The reactor core flow is
26. the reactor system flow less the leakage flow. There are three
major leakage paths plus an additional leakage allowance to account
for calculational uncertainties. The three major leakage paths are
(1) through the core shroud, (2) through the control rod guide tubes
and instrument tubes, and (3) between all interfaces separating the
inlet and outlet regions.

The following table shows the breakdown of the updated leakage calculations:

Path	% of System Flow
1) Shroud	1.64
2) Control rod guide tubes & instrument guide tubes	1.67
3) Inlet to outlet interfaces	0.36
Total	3.67

The reactor core flow and power capability were evaluated by determining the steady-state power DNB ratios versus flow. Analyses were made for (2) variations of power capability with total reactor flow for a constant DNB ratio of 1.30, (b) DNB ratios for design flow with variations in hot channel mixing coefficients and (c) DNB ratios for gross flow variations of ± 10 percent. The results are shown in Figures 3-28 and 3-29. For the analysis shown in Figure 3-28 for design hot channel condition, the flow was determined that would give a DNB ratio of 1.30 for a range of reactor powers. This analysis shows, for example, that a DNB ratio of 1.30 can be maintained in the hot channel at 114 percent power with a total reactor flow of 118 x 10^6 lb/h as compared with the available design flow of 131.3 x 10^6 lb/h. The results shown by Line 2 in Figure 3-29 are the DNB ratios for rated flow of 131.3 x 10^6 lb/h versus power. The limiting condition is 122.5 per cent power for a DNB ratio of 1.30. Lines 1 and 3 show the DNB ratios versus power where the total system flow has been varied by ±10 per cent. Adequate DNB ratios can be maintained with a substantial reduction in reactor coolant system flow.

The foregoing sensitivity analyses were made using a fuel assembly design mixing coefficient of 0.02. A sensitivity analysis for a range of coefficients was made for the rated flow condition. The results are shown by Lines 4 and 5 of Figure 3-29 and discussed in more detail in 3.2.3.2.3 j.

f. Reactor Inlet Temperature Effects

The influence of reactor inlet temperature on power capability at design flow was evaluated. A variation of 1 F in reactor inlet temperature will result in a power capability change of 0.6 per cent at a given DNB ratio.

g. Fuel Temperature

1. Method of Calculation

A fuel temperature and gas pressure computer code was developed to calculate fuel temperatures, expansion, densification, equiaxed and columnar grain growth, center piping of fuel pellets, fission gas release, and fission gas pressure. Program and data comparisons were made on the basis of the fraction of the fuel diameter within these structural regions:

(a) Outer limit of equiaxed grain growth - 2,700 F.

(b) Outer limit of columnar grain growth - 3,200 F.

(c) Outer limit of molten fuel (UO_2) - 5,080 F.

Data from References 25 through 28 were used to compare calculated and experimental fractions of the rod in grain growth and central melting.

The radial expansion of the fuel pellet is computed from the mean fuel temperature and the average coefficient of linear expansion for the fuel over the temperature range considered. This model combined with the model for calculating the heat transfer coefficient was compared with the model developed by Notley et al.[29] of AECL. The difference in fuel growth for the two calculation models was less than the experimental scatter of data.

The fuel may be divided into as many as 30 radial and 70 axial increments for the analysis. An iterative solution for the temperature distribution is obtained, and the thermal conductivity of the fuel is input as a function of temperature. The relative thermal expansion of the fuel and cladding is taken into account when determining the temperature drop across the gap between the fuel and cladding surfaces.

The temperature drop across the gap is calculated using a gap conductance model based on the methods reported in Reference 55. The model which has the capability of calculating gap conductance before fuel to clad contact as well as after contact is an extension of the methods suggested by Ross and Stoute.[31] This fuel to clad heat transfer is a function of gap width, gas conductivity, mean conductivity of the interface materials, mean surface roughness, material hardness and fuel to clad contact pressure. Before total fuel to clad contact is made, a fraction of the fuel, based on fuel OD and gap size, is in contact with the clad. A constant contact pressure is applied to this fraction to simulate the effects of fuel cracking.[36-38]

The analytical model computes the amount of central void expected whenever the temperature approaches the threshold temperature for fuel migration, and readjusts the density according to the new geometry.

The program uses a polynomial fit relationship for fuel thermal conductivity. The B&W reference design[56] curve illustrated in Figure 3-17 is a modification of the relationship presented in GEAP-4624.[30] The curve yields a conservative integrated thermal conductivity of 93 w/cm with relatively little increase in conductivity beyond 3,000 °F.

2. Fuel Center Temperature Results at Beginning and End of Life

The results of the analysis for center temperatures with the methods described above are shown in Figures 3-30 and 3-31 for beginning- and end-of-life conditions. The beginning- and end-of-life gas conductivity values are 0.09 and 0.01 Btu/h-ft-F.

The design cold diametral clearance is 0.0085 inch which is based on the final design specified quality control measurements for fuel clad and UO_2 pellet tolerances for a 99 per cent confidence and 95 per cent population relationship. A sensitivity analysis was made for a range of cold diametral clearances of 0.007 inch (nominal) to 0.0095 inch to show the effect of clearance on center temperature. Temperatures for the design clearance and the sensitivity maximum and minimum values examined are shown on the figures mentioned above.

The calculated end-of-life center fuel temperatures are higher than the beginning-of-life values because of the reduction in the conductivity of the gas in the gap. The effect is apparent even though a contact condition prevails. The calculation includes the effect of fuel swelling due to irradiation and takes credit for the flux depression in the center of the rod because of the self-shielding effect of UO_2 (nonuniform power generation).

The most conservative assumptions using the B&W design curve with relatively little increase in thermal conductivity above 3,000 F, result in central fuel melting at about 23 kW/ft, which is 3 kW/ft higher than the maximum design value of 20.1 kW/ft at 114 per cent power as shown in Figure 3-31.

The transient analyses at accident and normal conditions have been made using the design fuel thermal conductivity curve (Figure 3-17) to reflect a conservative value for the maximum average temperature and stored energy in the fuel. Use of this curve results in a higher temperature and, therefore, a lower Doppler coefficient, since it decreases with temperature. Thus, the resultant Doppler effect is also conservative.

3. Fuel Center and Average Temperature Variations With Fuel Burnup

Maximum fuel temperature conditions are affected by the fuel to clad heat transfer coefficient. The coefficient is determined by fuel to clad clearance and gas conditions. Fuel swelling due to burnup decreases the clearance and results in improved heat transfer; however, the conductivity of the gas rapidly decreases with the addition of xenon and krypton gas to the helium fill gas. A combination of these effects for beginning and end of life conditions was described in the previous paragraph. It was conservatively assumed that the peak power could be obtained in a fuel rod with the maximum burnup. It is not likely that the peak power will be experienced in a rod with any significant fuel depletion; however, an additional sensitivity analysis of fission gas conductivity and fuel growth from zero to maximum burnup has been made for the design cold diametral clearance of 0.0085 inch. This analysis shows that the worst combination of gas conductivity and fuel to clad clearance occurs at a burnup of about 9,000 MWd/MTU. Center and average fuel temperature peaks higher than beginning and end of life conditions may occur if fuel rods with the maximum design cold diametral clearance, maximum enrichment, and maximum linear heat rate can be operated continuously at the maximum reactor design power. The center fuel temperature for these conditions will change with burnup as shown in the upper curve in Figure 3-32 for the maximum design linear heat rate of 17.63 kW/ft given in Table 3-1. The lower curve shows a comparison of the peak center temperature for 9,000 MWd/MTU with the beginning of life and the end of life temperature as a function of heat rate. A similar analysis of average fuel temperature was used to predict peak temperatures and conservative values for stored energy at the hot spot for the loss of coolant accident analysis outlined in Section 14.

4. Several experimental studies[57][58][59] have shown that the melting temperature of UO_2 decreases with burnup. The data of Christensen[58] shows the greatest decrease in melting temperature, a nearly linear decrease from 5080°F at the beginning of life to 4800°F at 40,000 MWD/MTU. A comparison of fuel melting temperature and maximum fuel temperatures with burnup is shown in Figure 3-32a.

4. Equilibrium Cycle Averaged Fuel Temperatures

An analysis has been made to show equilibrium average fuel conditions in the core. A typical fuel cycle, end-of-life, condition was used to determine the fraction of fuel at a given average condition. The results are shown in Figure 3-33 where the average temperature varies from 1,100 to 3,300 F, and the entire core average temperature is about 1,600 F. The bundle average powers shown in Figure 3-34 were used to obtain the fuel rod

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heat rates. A symmetrical cosine axial power distribution with a 1.50 max/avg value as shown in Figure 3-8 was used to predict the axial heat rate distribution. It was assumed that 97.3 per cent of the power is generated in the fuel. The fuel rods were divided into 14 axial and 10 radial segments to obtain the temperature distribution for this analysis. The heat rate for every fuel rod in the core was increased by a local peaking factor of 1.05 to account for uncertainties in the calculation of local peaks.

The B&W design thermal conductivity was used to provide conservative values for fuel conductivity. The maximum powers occurred in fuel assemblies with one and two cycles of operation as shown in Figure 3-34, and the assemblies with the highest burnup did not exceed 1.043 times the average power for the typical case analyzed. The results shown in Figure 3-33 were made by grouping all segments of fuel by temperature. Typical 6 and 10 kW/ft rod radial temperature profiles are shown in Figure 3-35. Typical fuel-to-clad heat transfer coefficients used were 380 and 640 Btu/hr-ft^2-F for 6 and 10-kW/ft heat rates, respectively. The corresponding beginning-of-life coefficients are about 500 and 700 Btu/hr-ft^2 for 6 and 10-kW/ft heat rates.

h. Fission Gas Release

The fission gas release is based on results reported in GEAP-4596.[32] Additional data from GEAP-4314,[33] AECL-603,[34] and CF-60-12-14[35] have been compared with the suggested release rate curve. The release rate curve[32] is representative of the upper limit of release data in the temperature region of most importance. An internal gas
1. pressure of 3,300 psi is used to determine the fuel clad internal design conditions reported in 3.2.4.2, Fuel Assemblies.

The design values for fission gas release from the fuel and for the maximum clad internal pressure were determined by analyzing various operating conditions and assigning suitable margins for possible increases in local or average burnup in the fuel. Adequate margins are provided without utilizing the initial porosity voids present in the UO_2 fuel. A detailed analysis of the design assumptions for fission gas release, and the relationship of burnup, fuel growth, and initial diametral clearance between the fuel and clad, are summarized in the following paragraphs. An evaluation of the effect of having the fuel pellet internal voids available as gas holders is also included.

1. Design Assumptions

(a) Fission Gas Release Rates

The fission gas release rate is calculated as a function of fuel temperature at 114 per cent of rated power. The procedures for calculating fuel temperatures are discussed in 3.2.3.2.3 g. The fission gas release curve and the supporting data are shown in Figure 3-36. Most of the data are on or below the design release rate curve. A release rate of 51 per cent is used for the portion of the fuel above 3,500 F. The fuel temperatures were calculated using the B&W design fuel thermal conductivity curve which yields conservatively high values for fuel temperatures.

(b) Axial Power and Burnup Assumptions

The temperature conditions in the fuel are determined for the most severe axial power peaking expected to occur. Two axial power shapes have been evaluated to determine the maximum release rates. These are 1.50 and 1.70 max/avg shapes as shown in Figure 3-8 and repeated as part of Figure 3-37 of this analysis. The quantity of gas released is found by applying the temperature-related release rates to the quantities of fission gas produced along the length of the hot fuel rod.

The quantity of fission gas produced in a given axial location is obtained from reactor core axial region equilibrium burnup studies. Three curves showing the axial distribution of burnup as a local-to-average ratio along the fuel rod are shown in Figure 3-37. Values of 100, 300, and 930 days of operation are shown.

The 930-day, or end-of-life condition, is the condition with the maximum fission gas inventory. The average burnup at the end of life in the hot fuel rod is 38,150 MWd/MTU which has been determined as follows:

Calculated Hot Bundle Average Burnup, MWd/MTU	33,000
Hot Fuel Rod Burnup Factor	1.05
Margin for Calculation Accuracy	1.10
Hot Rod Maximum Average Burnup, MWd/MTU	38,150

The local burnup along the length of the fuel rod is the product of the hot rod maximum average value above and the local-to-average ratio shown in Figure 3-37. The resulting hot rod local maximum burnup for the 930-day, end-of-life condition is about 42,000 MWd/MTU. This is the maximum calculated value. However, local values to 55,000 MWd/MTU have been evaluated to insure adequate local fuel cladding strength for possible increases in average or local burnup over the life of the fuel for various fuel management procedures.

(c) Hot Rod Power Assumptions

1. The maximum hot rod power as a function of life of the fuel has been used to calculate the temperature conditions. A study of

the power histories of all of the fuel assemblies through
5 cycles to equilibrium conditions shows that the powers
in the bundles varies from a maximum of 1.78 x the average
rod power to 1.22 x average power at the end of life. The
1. peak bundle ratio of 1.68 (1.78 ÷ hot rod ratio) will only
occur during the first two fuel cycles when the fission gas
inventory is less than the maximum value. This results in
a maximum linear heat rate of 20.1 kW/ft which corresponds
to 114 per cent of the maximum thermal output (17.63 kW/ft)
shown in Table 3-1.

(d) Fuel Growth Assumptions

The fuel growth was calculated as a function of burnup as indicated in 3.2.4.2.1. Fuel pellet dimensions in the thermal temperature and gas release models were increased to the end-of-life conditions as determined above.

(e) Gas Conductivity and Contact Heat Transfer Assumptions

The quantity of fission gas released is a function of fuel temperature. The temperatures are influenced by three factors: (a) the conductivity of the fission gas in the gap between the fuel and clad, (b) the diametral clearance between fuel and clad, and (c) the heat transfer conditions when the fuel expands enough to contact the clad.

A gas conductivity of 0.01 Btu/h-ft-F based on 35 per cent release of fission gas at the end-of-life condition was used in the analysis. Diametral clearances of 0.0025 to 0.0065 inch reflecting minimum and design clearances after fuel growth were analyzed. The contact heat transfer coefficients were calculated as suggested in Reference 55 and are illustrated in Figure 3-38. The gap conductance is plotted as a function of heat rate for two fuel to clad gap clearance conditions to show the dependence of fuel to clad heat transfer on this parameter. Heat transfer models presented in the literature[54,56] suggest that gap conductance is higher than the design values used in this analysis.

2. Summary of Results

The fission gas release rates were determined in the first evaluation. Rates were found for various cold diametral clearances and axial power peaking and burnup shapes. The results are shown in Figure 3-39. The lowest curve is the expected condition for a 1.70 axial power shape with a 930-day axial burnup distribution as shown in Figure 3-37. The increase in release rate with diametral clearance results from the fact that the fuel temperature

must be raised to higher values before contact with the fuel clad is made. The release rate at the minimum clearance of 0.0045 inch
1. is 15.5 per cent. This condition is equivalent to a 0.0025 inch gap after irradiation growth and produces the maximum clad stress (maximum sized pellets with minimum internal diameter cladding).
1. The release rate of 30.6 per cent for the maximum diametral clearance of 0.0085 inch will not occur with the maximum stress condition due to fuel growth, since the fuel has more room to grow into the clearance.

Two additional cases were examined to check the sensitivity of the calculations to axial power and burnup shapes. The results are shown by the upper two curves in Figure 3-39. The top curve is a plot of the release rates when it is assumed that both the axial power and burnup inventory of fission gas are distributed with a 1.50 max/avg ratio as shown on Figure 3-37. Similar results are shown for the 1.70 max/avg power ratio with a 1.50 max/avg burnup ratio. These curves show the release rates expected are not strongly influenced by the various power and burnup shapes.

The second evaluation shows the resulting internal pressures due to the release of fission product gases. Plots of internal clad pressures for the expected 930-day axial burnup distribution and a 1.70 max/avg axial power shape are shown in Figure 3-40. The lower curve is a plot of internal gas pressure assuming that 6.5 per cent of the fuel volume is available to hold the released
1. gas (open porosity). A corresponding closed pore case is also shown. The present design condition being used in clad-stress calculations assumes a closed pore condition with all released gas contained outside the fuel pellets in spaces between the expanded dished ends of the pellets, the radial gaps (if any), and the void spaces at the ends of the fuel rods. The effects of fuel densification and grain growth described in 3.2.3.2.3 g are included in the analysis. The calculation of maximum pressure is also relatively insensitive to the axial burnup distribution as shown by the line in Figure 3-40 for a 1.50 max/avg axial power and burnup shape. (This corresponds to a local burnup peak of 57,000 MWd/MTU.)

The allowable design internal pressure of 3,300 psi is well above the maximum values of internal pressures calculated for open or closed pores, and the maximum internal pressure should only occur with the maximum diametral clearance condition. An increase in average fuel burnup can be tolerated within the prescribed internal pressure design limits.

It has been indicated in Reference 29 and in AECL-1598 that the UO_2 fuel is plastic enough to flow under low stresses when the temperature is above 1,800 F. That fraction of the fuel below this temperature may retain a large portion of the original porosity and act as a fission gas holder. The hot-test axial locations producing the highest clad stresses will have little if any fuel below 1,800 F. However, the ends of the fuel rods will have some fuel below this temperature.

The approximate fraction of the fuel below 1,800 F at overpower for a 1.5 axial power shape is as follows for various cold diametral clearance. The bundle average powers shown in Figure 3-34 were used to determine the heat rates.

Clearance, in.	Per cent of Fuel Below 1,800 F, %
0.0045	78
0.0070	62
0.0085	48

The retention of fuel porosity in the low temperature and low burnup regions will result in modest reductions in internal gas pressure.

Gas pressures at rated and overpower conditions are shown in Figure
1. 3-40. The overpower condition is not expected to occur except for
brief periods during operating transients.

i. Hot Channel Factors Evaluation

1. Rod Pitch and Bowing

A flow area reduction factor is determined for the as-built fuel assembly by taking channel flow area measurements and statistically determining an equivalent hot channel flow area reduction factor. A fuel assembly has been measured, and the results are shown in Table 3-10. Interior channel measurements and measurements of the channels formed by the outermost fuel rods with adjacent assemblies have been analyzed. Coefficients of variation for each type of channel have been determined. In the analytical solution for a channel flow, each channel flow area is reduced over its entire length by the F_A factors shown in Figure 3-21 for the desired population protected at a 99 per cent confidence. The hot channels have been analyzed using values for 95 per cent population protected, or F_A in the interior cells of 0.98 and F_A in the wall cells of 0.97 as listed in 3.2.3.1.1 j.
A statistical analysis of coolant channel measurements made in
18. 72 Oconee 1 fuel assemblies indicates that at the 99% confidence
level, more than 98.51% of the coolant channels have a flow reduction factor (FA) greater than or equal to 0.98 in the interior cells and 0.97 in the wall cells.

Special attention is given to the influence of water gap variation between fuel assemblies when determining rod powers. Nuclear analyses have been made for the nominal, maximum, and minimum spacing between adjacent fuel assemblies. The nominal and maximum hot assembly fuel rod powers are shown in Figure 3-41 and 3-42. The hot channel nuclear power factor ($F\Delta h$ nuclear) of 1.78 shown in 3.2.3.1.1. is based on Figure 3-42 for the worse water gap between fuel assemblies. The factor of 1.783 is a product of the hot assembly factor of 1.68 times the 1.061 hot rod factor. This power factor is assigned to the hottest unit cell rod which is analyzed for burnout. Peaking factors for other channels are obtained in a similar manner. In all cases, the combined flow spacing and power peaking producing the lowest DNB ratio is used.

2. Fuel Pellet Diameter, Density, and Enrichment Factors

Variations in the pellet size, density, and enrichment are reflected in coefficients of variation Numbers 2 through 7 of Table 3-10. These variations have been obtained from the measured or specified tolerances and combined statistically as described in 3.2.3.2.2 to give a power factor on the hot rod. For 99 per cent confidence and 95 per cent population conditions, this factor, F_Q, is 1.011 and is applied as a power increase over the full length of the hot channel fuel rod. The local heat flux factor, F_Q'', for similar conditions is 1.014. These hot channel values are shown in Table 3-1. The corresponding values of F_Q and F_Q'' with 99.99 per cent population protected are 1.025 and 1.03, respectively. A conservative value of F_Q'' of 1.03 for 99 per cent confidence and 99.99 per cent population is used for finding the maximum fuel linear heat rates as shown in 3.2.3.1.2.

These factors are used in the direct solution for channel enthalpies and are not expressed as factors on enthalpy rise as is often done.

3. Flow Distribution Effects

Inlet Plenum Effects

The inlet plenum effects have been determined from the 1/6 scale model flow test. The isothermal flow test data has shown that the hot bundle positions receive average or better flow. It has been conservatively assumed that the flow in the hot bundle position is 5 per cent less than average bundle flow under isothermal conditions corresponding to the model flow test conditions. An additional reduction of flow due to hot assembly power is described below.

Redistribution in Adjacent Channels of Dissimilar Coolant Conditions

The hot fuel assembly flow is less than the flow through an average assembly at the same core pressure drop because of the increased pressure drop associated with a higher enthalpy and quality condition. This effect is allowed for by making a direct calculation for the hot assembly flow. The combined effects of upper and lower plenum flow conditions and heat input to the hot assemblies have been used to determine hot assembly flows. The worst flow maldistribution effect has been assumed in the design,
16. and the minimum hot assembly flow has been calculated to be 87 per cent of the average assembly flow at 114 per cent overpower. Actual hot assembly flows are calculated rather than applying an equivalent hot channel enthalpy rise factor.

Physical Mixing of Coolant Between Channels

The flow distribution within the hot assembly is calculated with a mixing code that allows an interchange of heat between channels.

Mixing coefficients have been determined from multirod mixing tests. The fuel assembly, consisting of a 15 x 15 array of fuel rods, is divided into unit, wall, control rod, and corner cells as shown by the heavy lines in Figure 3-41. The mixed enthalpy for every cell is determined simultaneously so that the ratio of cell to average assembly enthalpy rise (Enthalpy Rise Factor) and the corresponding local enthalpy are obtained for each cell. Typical enthalpy rise factors are shown in Figures 3-41 and 3-42 for the hot and surrounding cells. The assumptions used to describe the channels for the peaking and enthalpy rise factors shown are given in 3.2.3.2.3 j, below.

j. Evaluation of the DNB Ratios in the Unit, Wall, Control Rod, and Corner Cells

DNB Results at Rated Flow

The DNB ratios in the hot unit cell at the maximum design condition described in 3.2.3.1 are shown in Figure 3-13. The relationship shown is based on the application of the W-3 correlation. An additional sensitivity analysis of the assembly corner, wall, i.e., peripheral, and control rod cells has been made for the worst combination of fuel assembly spacing and power peaking.

The sensitivity of the assembly design with respect to variations of mass velocity (G), channel spacing, mixing intensity, and local peaking on the DNB ratios in the fuel assembly channels has been evaluated by analyzing the nominal conditions and a postulated worst case condition. The summary results are shown below in Tables 3-15 for the nominal case and 3-16 for the maximum design or postulated worst case. The unit cell DNB ratios are repeated for comparison. All of the DNB ratios are for 114 per cent overpower.

Table 3-15
DNB Ratios in the Fuel Assembly Channels (W-3)

Nominal Case

Cell Type	G, $lb/h\text{-}ft^2$ x 10^{-6}	DNBR (W-3) (114% Power)
Unit	2.51	1.82
Corner	2.58	1.86
Wall (peripheral)	2.56	1.90
Control Rod	2.40	1.97

Table 3-16

DNB Ratios in the Fuel Assembly Channels (W-3)

Postulated Worst Case (Design)

Cell Type	G, lb/h-ft^2 x 10^{-6}	DNBR (W-3) (114% Power)
Unit	2.26	1.55
Corner	2.14	1.66
Wall (peripheral)	2.20	1.65
Control Rod	2.16	1.69

The DNB ratios in all channels are high enough to insure a confidence-population relationship equal to or better than that outlined in 3.2.3.1.1 for the hot unit cell channel. All of the wall, corner, and control rod cells have DNB ratios equal to or greater than that of the unit cell hot channel. This results from a more favorable flow to power ratio in these cells associated with relatively larger flow areas.

The DNB ratios were obtained by comparing the fuel rod local heat fluxes and channel coolant conditions with the limitations predicted by the correlation. Typical results are shown in Figures 3-43 and 3-44 for the nominal and worst case conditions in the unit cell.

Fuel Rod Power Peaks and Cell Coolant Conditions

The nominal case local-to-average rod powers and the local-to-average exit enthalpy rise ratios are shown in Figure 3-41 for the hot corner, wall, control rod, and unit cells in the hot fuel assembly. Values shown are for nominal water gaps between the hot fuel assembly and adjacent fuel assemblies with nominal flow to the hot fuel assembly, and with a minimum intensity of turbulence, α,(*) equal to 0.02.

(*)The intensity of turbulence, α, is defined as

1.

$$\sqrt{V_t'^{2}}\,/\,V$$

where V_t' is the transverse component of the fluctuating turbulent velocity, and V is the coolant velocity in the axial direction. This method of computing mixing is described by Sandberg, R. O., and Bishop, A. A., CVTR Thermal-Hydraulic Design for 65 MW Gross Fission Power, CVNA-227.

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The postulated worst case local-to-average rod powers (nuclear peaking factor) and exit enthalpy rise factors in the hot fuel assembly are shown in Figure 3-42. The factors were determined for this case with the minimum water gap between the hot fuel assembly and adjacent fuel assemblies, with minimum flow to the hot fuel assembly, and with a minimum assumed intensity of turbulence, α, equal to 0.02. An evaluation of minimum, nominal, and maximum spacing between assemblies showed the minimum to have the lowest DNB ratios.

A mixing coefficient of 0.02 was used for both nominal and design worst case analyses. The influence of mixing coefficients is shown in Figure 3-29, which shows values ranging from 0.01 to 0.06. The value of 0.02 is sufficiently conservative for design evaluation. The conditions analyzed to obtain the DNB ratios for various values of the mixing coefficients shown in Figure 3-29 were outlined previously in 3.2.3.2.3 i.

Fuel Assembly Power and Rated Flow Conditions

The nominal and postulated worst cases were run at 114 per cent reactor power with the nominal and worst $F\Delta h$ factors shown in 3.2.3.1.1 c. The 1.50 modified cosine axial power shape of Figure 3-8 was used to describe the worst axial condition.

The hot assembly flow under nominal conditions without a flow maldistribution effect is 96 per cent of the average assembly flow, and the reduction in flow is due entirely to heat input effects. The hot assembly flow under the worst postulated conditions is
16. 87 per cent of the average assembly flow and considers the worst combined effects of heat input and flow maldistribution.

k. DNB Results for Postulated Loss of an Internals Vent Valve

The reactor arrangement includes vent valves above the core to equalize the pressure between inlet and outlet regions during a loss-of-coolant accident. The effective core flow will be reduced in the unlikely event that an internals vent valve is open. A DNB analysis was made to show the design margin for a postulated accidental failure of one valve disc.

1. Four Pump Operation

   In the event the disc from one of these valves is completely removed, a small reduction in effective core flow for heat removal will be experienced. Approximately 5.7 per cent of the incoming flow will bypass the core through the valve opening. However, the reduction of resistance results in an increase in total system flow of about 1.1 per cent. The net reduction of flow for core heat removal is 4.6 per cent.

   The minimum DNB ratios for the reduced effective core flow compare with the full flow ratios given in 3.2.3.1.1 j as follows:

Per Cent Rated Power	DNBR (Full Flow)	DNBR (Reduced Flow)
100	2.00	1.84
107.5[a]	1.75	1.59
114	1.55	1.4

[a]Trip set point.

DNB ratios were determined for the worst corner, control rod, wall, or unit cell. The DNB ratios in the hot unit cell were the lowest. The minimum DNB ratio at the trip set point of 107.5 per cent power is well above the minimum recommended value of 1.30. The DNB ratio of 1.30 is maintained up to 117 per cent power for the postulated worst case design conditions.

A complete sensitivity analysis has been made to determine the effects of design flow and unexpected core bypass flow from the inlet to the outlet chambers in the reactor vessel. The results are shown in Figure 3-45. Bypass flow was varied from 0 to 10 per cent while holding a constant reactor average temperature of 578.9 F. A design allowance of 2 per cent (2.63 x 10^6 lb/h) bypass flow for vent valve seat and fitup leakage is included in all calculations for nominal or maximum design DNB ratios. This design condition is indicated by Line 2 and is identical with rated conditions in Figure 3-13 as previously discussed. Line 5 shows the DNB ratios versus power for a condition of the loss of one vent valve disc.

9 2. Partial Pump Operation (Reference Supplement 6 statement.)

The power limitations imposed on the reactor due to the loss of one or more pumps has been examined for the full flow condition in 3.2.3.1.1 k. The results of a similar analysis for the reduced flow conditions caused by the loss of an internals vent valve are summarized below.

Reactor Coolant Pumps Operating	3 Pumps	2 Pumps 2 Loops	2 Pumps 1 Loop	1 Pump
Hot channel DNBR	1.3	1.45	1.70	4.7
Hot channel quality at min DNBR point, %	8.0	15.0	15.0	-7.0
Reactor coolant flow, % of rated	70.1	44.4	41.2	17.4
Max design overpower, % of rated power (2,568 MW)	97.5	73.0	65.0	30.0

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3.2.3.2.4 Evaluation of Internals Vent Valve

A vapor lock problem could arise if water is trapped in the steam generator blocking the flow of steam from the top of the reactor vessel to a cold leg leak. Under this condition, the steam pressure at the top of the reactor would rise and force the steam bubbles through the water leg in the bottom of the steam generator. This same differential pressure that develops a water leg in the steam generator will develop a water leg in the reactor vessel which could lead to uncovering of the core.

The most direct solution to this problem is to equalize the pressure across the core support shield, thus eliminating the depression of the water level in the core. This was accomplished by installing vent valves in the core support shield to provide direct communication between the top of the core and the coolant inlet annulus. These vent valves open on a very low-pressure differential to allow steam generated in the core to flow directly to the leak from the reactor vessel. Although the flow path in the steam generator is blocked, this is of no consequence since there is an adequate flow path to remove the steam being generated in the core.

During the vent valve conceptual design phase, criteria were established for valves for this service. The design criteria were (1) functional integrity, (2) structural integrity, (3) remote handling capability, (4) individual part-capture capability, (5) functional reliability, (6) structural reliability, and (7) leak integrity throughout the design life. The design criteria resulted in the selection of the hinged-disc (swing-disc) check valve, which was considered suitable for further development.

Because of the unique purpose and application of this valve, B&W recognized the need for a complete detailed design and development program to determine valve performance under nuclear service conditions. This program included both analytical and experimental methods of developing data. It was performed primarily by B&W and the selected valve vendor or his subcontractors.

Vent valve preliminary design drawings were prepared and analyzed both by B&W and the vendor/subcontractor. Specifications and drawings were prepared, and orders were placed with the vendor for the design, development, fabrication, and test of a full-size prototype vent valve. The prototype valve was completed and subjected to the tests described in 3.3.4. All testing was successfully completed and minor problems encountered during valve assembly handling or use were corrected to arrive at the final design for the production valve.

The only significant problem encountered during test was seizing of one jack-screw. This was attributable to an excessive thickness of "Electrolyze" which spalled off the screw threads. This problem was corrected by reducing the specified "Electrolyze" thickness from 0.0015" to 0.0004" max. and no further galling was encountered. To further enhance resistance to galling, the final design jackscrew has a 1-1/8"-8 Acme thread form instead of a 1"-12 UNF and the material is an age hardened corrosion resistant alloy instead of 410 S.S. No further jackscrew problems have occurred or are anticipated on the basis that the surfaces are separated by the low friction "Electrolyze", different materials of different hardnesses are used, loose fits are employed, and thread contact stresses are low (3775 psi).

The final design of this valve is shown in Figure 3-49a. The valve disc hangs closed in its natural position to seal against a flat, stainless steel seat inclined 5 degrees from vertical to prevent flow from the inlet coolant annulus to the plenum assembly above the core. In the event of LOCA, the reverse pressure differential will open the valve. At all times during normal reactor operation, the pressure in the coolant annulus on the outside of the core support shield is greater than the pressure in the plenum assembly on the inside of the core support shield. Accordingly, the vent valve will be held closed during normal operation. With four reactor coolant pumps operating, the pressure differential is 42 psi resulting in a several-thousand-pound closing force on the vent valve.

Under accident conditions, the valve will begin to open when a pressure differential of less than 0.15 psi develops in a direction opposite to the normal pressure differential. At this point, the opening force on the valve counteracts the natural closing force of the valve. With an opening pressure differential of no greater than 0.3 psi, the valve would be fully open. With this pressure differential, the water level in the core would be above the top of the core. In order for the core to be half uncovered, assuming solid water in the bottom half of the core, a pressure differential of 3.7 psi would have to be developed. This would provide an opening force of about 10 times that required to open the valve completely. This is a conservative limit since it assumes equal density in the core and the annulus surrounding the core. The hot, steam-water mixture in the core will have a density much less than that of the cold water in the annulus, and somewhat greater pressure differentials could be tolerated before the core is more than half uncovered.

An analog computer simulation was developed to evaluate the performance of the vent valves in the core support shield. This analysis demonstrated that adequate steam relief exists so that core cooling will be accomplished. This analysis is described in detail in Section 14.2.2.3.

The behavior of the valve disc during LOCA conditions was investigated and the rather complex dynamic behavior of the disc during LOCA was analyzed as a series of simpler models which provide conservative predictions of peak stresses and deflections.

The valve disc remains closed initially for the LOCA hot leg (36" pipe) case and the disc opening on subsequent differential pulses is less than one-half of the initial disc to vessel wall impact velocity for the LOCA cold leg (28" pipe) case. Therefore, the disc motion and initial impact with the vessel inside wall was chosen as the worst case and the only one requiring consideration. The cold-leg LOCA pressure time history acting on the disc was approximated by a piecewise linear time function. The moment due to pressure was equated to the rotary inertia of the disc to determine the velocity of impact with the vessel inside wall.

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The model chosen for the initial impact consisted of three effective springs and two masses to represent the disc with its lug, the compliance of the disc, and the vessel inside wall.

Loads generated on impact were based on the conservation of energy. The stresses obtained for these loads indicated that the elastic model assuming conservation of energy was not valid and that the impact must assume plastic deformation. The locations and modes of plastic deformation are illustrated in BAW-10005.

The plastic analysis provided the following information:

a. Crush deformation of lug after disc corner contacts the vessel wall is predicted to be 0.165 inches.

b. The total deformation of lug from contact with the vessel wall until disc assembly motion is arrested is predicted to be 0.483 inches.

c. The total angular deformation at the plastic hinge is predicted to be 0.016 radians.

d. An analysis was performed on the reactor vessel wall for disc assembly impact and the results indicate that while the stainless steel cladding is deformed locally, the reactor maintains its structural and pressure boundary integrity.

Because of conservative assumptions used in the plastic analysis, actual deformations will be considerably less than the above predicted values. Although plastic deformation may occur as predicted above on impact, the disc will retain its structural integrity. Plastic deformation of the disc dissipates the stored kinetic energy stored in the disc effectively; thus, the energy available for rebound is less than 1% of the initial impact energy and is too low to overcome the pressure differential and cause impact on the valve body. Disc and body hinge components were analyzed for worst case disc impact loadings and the resulting stresses were found to be less than the allowable limits; therefore, the valve disc free-motion (venting) function will be unaffected.

From the above, it is concluded that vent valve performance will not be impaired during the course of an accident because disc free-motion part stresses remain within allowable limits, disc structural integrity is maintained, vessel pressure boundary integrity is maintained, and plastic deformation of the disc seating surface improves the venting function.

With reference to Figure 3-49, each jackscrew assembly consists of a jackscrew, internally splined mating nut ring, nut ring spring, capture cover and cover attachment fasteners (socket head cap screws). In the figure, the splined nut ring and its spring are hidden from view by the capture cover. The potential for loss of jackscrew assembly parts during the plant lifetime is considered remote on the basis that the jackscrews and capture parts are accessible for visual inspection during scheduled refueling outages. A jackscrew loss is considered remote because a failure in service is highly improbable with the low compressive load (1000 psi) involved and the jackscrew is retained in the valve body by a central shoulder and the ends are threaded into the retaining rings. An in-service failure of the splined nut ring and its spring is remote because these parts are subjected to little or no load and even if they did fail all parts would be retained within the capture cover. Capture cover failure and loss is highly improbable on the same basis that it is not loaded in service. The capture cover is attached to the upper retaining ring by socket head cap screws which are lock welded to the cover at installation. By design, these screws are retention rather than structural devices and are not loaded in service. These screws do not require a pre-load to hold the formed cover in place; therefore, a loss of pre-load by lock welding would not jeopardize the cover or screw installation or structural integrity. Two fillet welds 180° apart are used to lock weld each screw head to the capture cover and in the absence of loads on both the cover and screws, the likelihood of lock weld failure and loss of screw heads is considered remote. With the capability to inventory these cap screw heads visually at scheduled refuelings, any problem related to the loss of these screws would be apparent early in the plant life and the valve assemblies could be removed for corrective action.

The internals vent valves are described, including materials and hinge part loose clearances in 3.2.4.1.2h.

The internals vent valves have been tested for ability to withstand the effects of vibratory excitations and for other functional characteristics as described in 3.3.4.

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3.2.4 MECHANICAL DESIGN

3.2.4.1 Reactor Internals

Reactor internal components include the plenum assembly and the core support assembly. The core support assembly consists of the core support shield, vent valves, core barrel, lower grid, flow distributor, incore instrument guide tubes, thermal shield, and surveillance holder tubes. The plenum assembly consists of the upper grid plate, the control rod guide assemblies, and a turnaround baffle for the outlet flow. Figure 3-46 shows the reactor vessel, reactor vessel internals arrangement, and the reactor coolant flow path. Figure 3-47 shows a cross section through the reactor vessel, and Figure 3-48 shows the core flooding arrangement.

Reactor internal components do not include fuel assemblies, control rod assemblies (CRA's), surveillance specimen assemblies, or incore instrumentation. Fuel assemblies and control rod assemblies are described in 3.2.4.2, control rod drives in 3.2.4.3, surveillance specimen assemblies in 4.4.6, and incore instrumentation in 7.3.3.

The reactor internals are designed to support the core, maintain fuel assembly alignment, limit fuel assembly movement, and maintain CRA guide tube alignment between fuel assemblies and control rod drives. They also direct the flow of reactor coolant, provide gamma and neutron shielding, provide guides for incore instrumentation between the reactor vessel lower head and the fuel assemblies, support the surveillance specimen assemblies in the annulus between the thermal shield and the reactor vessel wall, and support the internals vent valves. The vent valves are designed to vent the stream generated within the core, thereby permitting the rapid re-covering of the core by coolant following a reactor coolant inlet pipe rupture. All reactor internal components can be removed from the reactor vessel to allow inspection of the reactor internals and the reactor vessel internal surface.

A shop fitup and checkout of the internal components for Oconee I in an as-built reactor vessel mockup will insure proper alignment of mating parts before shipment. Dummy fuel assemblies and control rod assemblies will be used to check fuel assembly clearances and CRA free movement.

To minimize lateral deflection of the lower end of the core support assembly as a result of horizontal seismic loading, integral weld-attached, deflection-limiting guide lugs have been welded on the reactor vessel inside wall. These blocks will also limit the rotation of the lower end of the core support assembly which could result from flow-induced torsional loadings. The lugs allow free vertical movement of the lower end of the internals for thermal expansion throughout all ranges of reactor operating conditions. In the unlikely event that a flange, circumferential weld, or bolted joint might fail, the lugs will limit the possible core drop to 1/2 in. or less. The elevation plane of these lugs was established near the elevation of the vessel support skirt attachment to minimize dynamic loading effects on the vessel shell or bottom head. A 1/2 in. core drop will not allow the lower end of the CRA rods to disengage from their respective fuel assembly guide tubes, even if the CRA's are in the full-out position. In this rod position, approximately 6-1/2 in. of rod length remains in the fuel assembly guide tubes. A core drop of 1/2 in. will not result in a significant reactivity change. The core cannot rotate and bind the drive lines, because rotation of the core support assembly is prevented by the guide lugs.

9. The core internals are designed to meet the stress requirements of the ASME Code, Section III, during normal operation and transients. Additional criteria
23. and analysis are given in B&W Topical Report BAW-10051, "Design of Reactor Internals and Incore Instrument Nozzles for Flow Induced Vibrations." A detailed stress analysis of the internals under accident conditions has been completed and is reported in B&W Topical Report No. 10008, Part 1, "Reactor Internals Stress and Deflection Due to LOCA and Maximum Hypothetical Earthquake." This report analyzes the internals in the event of a major loss-of-
3. coolant accident (LOCA) and for the combination of LOCA and seismic loadings. It is shown that although there is some internals deflection, failure of the internals will not occur because the stresses are within established limits. These deflections would not prevent CRA insertion because the control rods are guided throughout their travel, and the guide-to-fuel assembly alignment cannot change because positive alignment features are provided between them and the deflections do not exceed allowable values. All core support circumferential weld joints in the internals shells are inspected to the requirements of the ASME Code, Section III.

3.2.4.1.1 Plenum Assembly

The plenum assembly is located directly above the reactor core and is removed as a single component before refueling. It consists of a plenum cover, upper grid, CRA guide tube assemblies, and a flanged plenum cylinder with openings for reactor coolant outlet flow. The plenum cover is constructed of a series of parallel flat plates intersecting to form square lattices and has a perforated top plate and an integral flange at its periphery. The cover assembly is attached to the plenum cylinder top flange. The perforated top plate has matching holes to position the upper end of the CRA guide tubes. The plenum cover is attached to the top flange of the plenum cylinder by a flange. Lifting lugs are provided for remote handling of the plenum assembly. These lifting lugs are welded to the cover grid. The CRA guide tubes are welded to the plenum cover top plate and bolted to the upper grid. CRA guide assemblies provide CRA guidance, protect the CRA from the effects of coolant cross-flow, and provide structural attachment of the grid assembly to the plenum cover.

Each CRA guide assembly consists of an outer tube housing, a mounting flange, 12 perforated slotted tubes and four sets of tube segments which are oriented and attached to a series of castings so as to provide continuous guidance for the CRA full stroke travel. The outer tube housing is welded to a mounting flange, which is bolted to the upper grid. Design clearances in the guide tube accommodate misalignment between the CRA guide tubes and the fuel assemblies. Final design clearances were established by tolerance studies and Control Rod Drive Line Facility (CRDL) prototype test results. The test results are described in 3.3.3.4.

The plenum cylinder consists of a large cylindrical section with flanges on both ends to connect the cylinder to the plenum cover and the upper grid. Holes in the plenum cylinder provide a flow path for the coolant water. The upper grid consists of a perforated plate which locates the lower end of the individual CRA guide tube assembly relative to the upper end of a corresponding fuel assembly. The grid is bolted to the plenum cylinder lower flange. Locating keyways in the plenum assembly cover flange engage the reactor vessel flange locating keys to align the plenum assembly with the reactor vessel, the reactor closure head control rod drive penetrations, and the core support assembly. The bottom of the plenum assembly is guided by the inside surface of the lower flange of the core support shield.

3.2.4.1.2 Core Support Assembly

The core support assembly consists of the core support shield, core barrel, lower grid assembly, flow distributor, thermal shield, incore instrument guide tubes, surveillance specimen holder tubes, and internals vent valves. Static loads from the assembled components and fuel assemblies, and dynamic loads from CRA trip, hydraulic flow, thermal expansion, seismic disturbances, and loss-of-coolant accident loads are all carried by the core support assembly.

The core support assembly components are described as follows:

a. Core Support Shield

The core support shield is a flanged cylinder which mates with the reactor vessel opening. The forged top flange rests on a circumferential ledge in the reactor vessel closure flange. The core support shield lower flange is bolted to the core barrel. The inside surface of the lower flange guides and aligns the plenum assembly relative to the core support shield. The cylinder wall has two nozzle openings for coolant flow. These openings are formed by two forged rings, which seal to the reactor vessel outlet nozzles by the differential thermal expansion between the stainless steel core support shield and the carbon steel reactor vessel. The nozzle seal surfaces are finished and fitted to a predetermined cold gap providing clearance for core support assembly installation and removal. At reactor operating temperature, the mating metal surfaces are in contact to make a seal without exceeding allowable stresses in either the reactor vessel or internals. Eight vent valve mounting rings are welded in the cylinder wall. Internals vent valves are installed in the core support shield cylinder wall to control steam flow from the core following a postulated cold leg (reactor coolant inlet) pipe rupture as described in 3.2.4.1.

b. Core Barrel

The core barrel supports the fuel assemblies, lower grid, flow distributor, and incore instrument guide tubes. The core barrel consists of a flanged cylinder, a series of internal horizontal former plates bolted to the cylinder, and a series of vertical baffle plates bolted to the inner surfaces of the horizontal formers to produce an inner wall enclosing the fuel assemblies. The core barrel cylinder is flanged on both ends. The upper flange of the core barrel cylinder is bolted to the mating lower flange of the core support shield assembly and the lower flange is bolted to the lower grid assembly. All bolts are lock welded after final assembly. Coolant flow is downward along the outside of the core barrel cylinder and upward through the fuel assemblies contained in the core barrel. A small portion of the coolant flows upward through the space between the core barrel outer cylinder and the inner baffle plate wall. Coolant pressure in this space is maintained lower than the core coolant pressure to avoid tension loads on the bolts attaching the plates to the horizontal formers.

c. Lower Grid Assembly

The lower grid assembly provides alignment and support for the fuel assemblies, supports the thermal shield and flow distributor, and aligns the incore instrument guide tubes with the fuel assembly instrument tubes. The lower grid consists of two lattice type grid structures, separated by short tubular columns, and surrounded by a forged flanged cylinder. The upper structure is a perforated plate, while the lower structure consists of intersecting plates welded to form a grid. The top flange of the forged cylinder is bolted to the lower flange of the core barrel.

A perforated flat plate located midway between the two lattice structures aids in distributing coolant flow prior to entrance into the core. Alignment between fuel assemblies and incore instruments is provided by pads bolted to the upper perforated plate.

d. Flow Distributor

The flow distributor is a perforated dished head with an external flange which is bolted to the bottom flange of the lower grid. The flow distributor supports the incore instrument guide tubes and distributes the inlet coolant entering the bottom of the core.

e. Thermal Shield

A cylindrical stainless steel thermal shield is installed in the annulus between the core barrel cylinder and reactor vessel inner wall. The thermal shield reduces the incident gamma absorption internal heat generation in the reactor vessel wall and thereby reduces the resulting thermal stresses. The thermal shield upper end is restrained
23. against inward and outward vibratory motion by restraints bolted to the core barrel cylinder. The lower end of the thermal shield is shrunk fit on the lower grid flange and secured by 96 high strength bolts.

f. Surveillance Specimen Holder Tubes

Surveillance specimen holder tubes are installed on the core support assembly outer wall to contain the surveillance specimen assemblies. The tubes extend from the top flange of the core support shield down toward the lower end of the thermal shield. The holder tube has a 3-1/2 inch offset to place the center line of the specimens approximately 3 inches from the vessel inside wall. B&W Topical Report BAW-10006, "Reactor Vessel Material Surveillance Program", describes the holder tubes and specimen capsules in detail.

g. Incore Instrument Guide Tube Assembly

The incore instrument guide tube assemblies guide the incore instrument assemblies from the instrument penetrations in the reactor vessel bottom head to the instrument tubes in the fuel assemblies. Horizontal clearances are provided between the reactor vessel instrument penetrations and the instrument guide tubes in the flow distri-
23. butor to accommodate misalignment. Fifty-two incore instrument guide tubes are provided and are designed so they will not be affected by the core drop described in 3.2.4.1.

h. Internals Vent Valves

Internals vent valves are installed in the core support shield to prevent a pressure unbalance which might interfere with core cooling following a postulated inlet pipe rupture. Under all normal operating conditions, the vent valve will be closed. In the event of the pipe rupture in the cold leg of the reactor loop, the valve will open to permit steam generated in the core to flow directly to the leak, and will permit the core to be rapidly re-covered and adequately cooled after emergency core coolant has been supplied to the reactor vessel. The design of the internals vent valve is shown in Figure 3-49.

Each valve assembly consists of a hinged disc, valve body with sealing surfaces, split-retaining ring, and fasteners. Each valve assembly is installed into a machined mounting ring integrally welded in the core support shield wall. The mounting ring contains the necessary features to retain and seal the perimeter of the valve assembly. Also, the mounting ring includes an alignment device to maintain the correct orientation of the valve assembly for hinged-disc operation. Each valve assembly will be remotely handled as a unit for removal or installation. Valve component parts, including the disc, are of captured-design to minimize the possibility of loss of parts to the coolant system, and all operating fasteners include a positive locking
4. device. The hinged-disc includes a device for remote inspection of disc function. Vent valve materials are listed in Table 3-16a.

The vent valve materials were selected on the basis of their corrosion resistance, surface hardness, antigalling characteristics, and compatibility with mating materials in the reactor coolant environment.

The arrangement consists of eight 14-in. inside diameter vent valve assemblies installed in the cylindrical wall of the internals core support shield (refer to Figure 3-46). The valve centers are coplanar and are 42 in. above the plane of the reactor vessel coolant nozzle centers. In cross section, the valves are spaced around the circumference of the core support shield wall.

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Table 3-16a

Internals Vent Valve Materials

Valve Part Name	Material and Form	Material Specification No.
Valve Body	304 S.S. Casting [(a)]	ASTM A351-CF8
Valve Disc	304 S.S. Casting [(a)]	ASTM A351-CF8
Disc Shaft	431 S.S. Bar [(b)]	ASTM A276 Type 431 Cond. T
Shaft Bushings	Stellite No. 6	
Retaining Rings (Top and Bottom)	15-5 pH (H 1100) S.S. forgings	AMS 5658
Ring Jack Screws	"A-286 Superalloy" S.S. [(c)]	AMS 5737 C
Jackscrew Bushings	431 S.S. Bar	ASTM A276 Type 431 Cond. A
Misc. Fasteners, covers, locking devices, etc.	304 S.S. plate bar, etc.	ASTM A240, ASTM A276

(a) Carbide solution annealed, C_{max} 0.08%, Co_{max} 0.2%

(b) Heat treated and tempered to Brinell Hardness Number (BHN) range of 290-320.

(c) Heat treated to produce a BHN of 248 min.

The hinge assembly consists of a shaft, two valve body journal receptacles, two valve disc journal receptacles, and four flanged shaft journals (bushings). Loose clearances are used between the shaft and journal inside diameters, and between the journal outside diameters and their receptacles. The hinge assembly is shown and the clearance gaps are identified in Figure 3-49a. The bushing clearances are listed in Table 3-16b.

The valve disc hinge journal contains integral exercise lugs for remote operation of the disc with the valve installed in the core support shield.

Table 3-16b

VENT VALVE SHAFT & BUSHING CLEARANCES

Clearance Gaps are Illustrated in Figure 3-49a

A. Cold Clearance Dimensions @ 70°F

Bushing I.D.	1.500	to	1.505	
Shaft O.D.	1.490	to	1.485	
	.010	to	.020	clearance (Gaps 1, 2, 7 & 8)

Body I.D.	2.000	to	2.003	
Bushing O.D.	1.997	to	1.995	
	.003	to	.008	clearance (Gaps 3, 4, 5 & 6)

Bushing End Clearance Gaps 9 + 10

Body Lugs	5.752	to	5.756	
Disc Hub	4.746	to	4.742	
	1.006	to	1.014	
	.996	to	.992	
	.010	to	.022	End Clearance (Gaps 9 + 10)
Bushing Flange	.249	x 4 =	.996	
	.248	x 4 =	.992	

B. Hot Clearance Differential Change from 70 to 580°F

Linear coefficient of thermal expansion of the materials for a temperature change of 70 to 600°F.

Shaft:	A286	9.8×10^{-6} in/in/°F
Bushing:	Stellite #6	8.1×10^{-6}
Bodies:	CF8 Stainless	9.82×10^{-6}

$$\Delta T = 580 - 70 = 510$$

Shaft $\Delta D = D\alpha\Delta T = 1.5\ (9.8 \times 10^{-6})\ 510 = .0075$

Bushing I.D. $= 1.5\ (8.1 \times 10^{-6})\ 510 = .0062$

$-.0013$ decrease

Bushing O.D. $= 2\ (8.1 \times 10^{-6})\ 510 = .0083$

Body I.D. $= 2\ (9.82 \times 10^{-6})\ 510 = .010$

$+.0017$ increase

Bushing Endplay Hot

CF8 Body $\Delta L = 1\ (9.82 \times 10^{-6})\ 510 = .0050$

Stellite #6 Bushing Flange $= 1\ (8.1 \times 10^{-6})\ 510 = .0041$

$.0009$ increase

The hinge assembly provides eight loose rotational clearances to minimize any possibility of impairment of disc-free motion in service. In the event that one rotational clearance should bind in service, seven loose rotational clearances would remain to allow unhampered disc free motion. In the worst case, at least four clearances must bind or seize solidly to adversely affect the valve disc free motion.

4. In addition, the valve disc hinge loose clearances permit disc self-alignment so that the external differential pressure adjusts the disc seal face to the valve body seal face. This feature minimizes the possibility of increased leakage and pressure-induced deflection loadings on the hinge parts in service.

The external side of the disc is contoured to absorb the impact load of the disc on the reactor vessel inside wall without transmitting excessive impact loads to the hinge parts as a result of a loss-of-coolant accident.

3.2.4.2 Core Components

The complete core has 177 fuel assemblies arranged in a square lattice to approximate the shape of a cylinder. All fuel assemblies are identical in mechanical construction, and mechanically interchangeable in any core location. Each fuel assembly will accept any control assembly. The reactivity of the core is controlled by 61 control rod assemblies (CRA) and 8 axial power shaping rod assemblies (APSRA). APSRA's are identical in physical configuration to the CRA's but have absorber material only in the lower portion of the rods. An orifice rod assembly (Figure 3-50) or a burnable poison rod assembly (Figure 3-51) is installed in fuel assemblies not containing an APSRA or a CRA. The orifice rod assemblies (ORA) limit guide tube bypass coolant flow through the fuel assembly guide tubes. The burnable poison rod assemblies (BPRA) assure a negative moderator temperature coefficient through core lifetime. The mechanical and geometric configuration of the CRA's, APSRA's, BPRA's and ORA's permit full interchangeability in any fuel assembly.

3.2.4.2.1 Fuel Assemblies

Description

a. General

The fuel is sintered low-enriched uranium dioxide cylindrical pellets. The pellets are clad in Zircaloy-4 tubing and sealed by Zircaloy-4 end caps, welded at each end. The clad, fuel pellets, end caps, and fuel support components form a "fuel rod". Two hundred and eight fuel rods, sixteen control rod guide tubes, one instrumentation tube assembly, seven segmented spacer sleeves, eight spacer grids, and two end fittings make up the basic "Fuel Assembly" (Figure 3-52). The guide tubes, spacer grids, and end fittings form a structural cage to arrange the rods and tubes in a 15 x 15 array. The center position in the assembly is reserved for instrumentation. Control rod guide tubes are located in 16 locations of the array. Fuel assembly components, materials, and dimensions are tabulated below:

Table 3-17
Fuel Assembly Components

Item	Material	Dimensions (In.)
Fuel Rod:		
Fuel	UO_2 Sintered Pellets	0.370 Dia.
Fuel Clad	Zircaloy-4	0.430 OD x 0.377 ID x 153-1/8 long
Fuel Rod Pitch		0.568
Active Fuel Length		144
24. Ceramic Spacer	ZrO_2	0.366 dia.
Metallic Spacer*	Zircaloy-4	0.359 dia.
Fuel Assembly:		
Fuel Assembly Pitch		8.587
Overall Length		165-5/8
Control Rod Guide Tube	Zircaloy-4	0.530 OD x 0.016 wall
Instrumentation Tube	Zircaloy-4	0.493 OD x 0.441 ID
End Fittings	Stainless Steel (Castings)	
Spacer Grid	Inconel-718 Strips	0.020 thick exteriors 0.016 thick interiors
Spacer Sleeve	Zircaloy-4	0.550 OD x 0.498 ID

27. *Oconee Unit III

b. Fuel Rod

The fuel is in the form of sintered and ground pellets of low enriched uranium dioxide. Pellet ends are dished to minimize differential thermal expansion between the fuel and cladding. The nominal density of the fuel is 93.5% of theoretical.

30. A maximum of 48 test fuel rods may be included in two Batch 2 fuel assemblies of Oconee 2; see Supplement 16.

Average design burnup of the fuel is 28,800 MWd/MTU with a peak design burnup of 55,000 MWd/MTU. At the peak design burnup, the fuel growth is calculated to be 9-1/2 volume per cent by the method given in Reference 39. Radial growth of the fuel during burnup is accommodated by pellet porosity, radial clearance between the pellets and the cladding, and by a small amount (less than 1% at the design burnup) of permanent strain in the cladding. Calculated peak burnup of the fuel is 42,000 MWd/MTU.

Below each fuel column is a thin wall stainless steel spacer which axially locates the bottom of the fuel column and separates the fuel from the lower fuel rod end cap. This pedestal is designed to collapse at a predetermined column load to prevent excessive axial strain in the cladding.

Above the fuel column is a thin-wall stainless steel spacer tube that separates the fuel from the fuel rod upper end cap. This spacer maintains the fuel column in place during shipping and handling. In operation, the spacer permits axial differential growth and thermal expansion between the fuel and the clad. This spacer also provides radial fuel rod cladding support. (In Oconee Unit No. 3, these spacers have
27 been replaced by springs which perform the same function.) Ceramic spacers are located between the fuel pellets and the corrugated tube spacers, to thermally insulate and separate fuel pellets from tube spacers. (In Oconee Unit 3, metallic and ceramic spacers are used.)

26 All fuel rods are internally pressurized with helium.

Fission gas release from the fuel is vented to voids within the fuel, the radial gap between the pellets and the cladding, and to the void spaces at top and bottom ends of the fuel rods.

C. Fuel Assembly

1. General

The fuel assemblies shown in Figures 3-52 and 3-52a are of the design to be used in Units 1, 2 and 3. Both are of the canless type where the eight spacer grids, end fittings, and the guide tubes form the
16 basic structure. Fuel rods are supported at each spacer grid by contact points integral with the wall of the cell boundary. The guide tubes are permanently attached to the upper and lower end fittings. Use of similar material in the guide tubes and fuel rods results in minimum differential thermal expansion.

2. Spacer Grids

Spacer grids are constructed from strips which are slotted and fitted together in "egg crate" fashion. Each grid has 32 strips, 16 perpendicular to 16, which forms the 15 x 15 lattice. The square walls formed by the interlaced strips provide support for the fuel rods in two perpendicular directions. Contact points on the walls of each square opening are integrally punched in the strips. On each of the two end spacer grids, the peripheral strip is extended and rigidly attached to the respective end fitting.

27

3. Lower End Fitting

The lower end fitting positions the assembly in the lower core grid plate. The lower ends of the fuel rods rest on the grid of the lower end fitting. Penetrations in the lower end fitting are provided for attaching the control rod guide tubes and for access to the instrumentation tube assembly.

4. Upper End Fitting

The upper end fitting positions the upper end of the fuel assembly in the upper core grid plate structure and provides means for coupling the handling equipment. An identifying number on each upper end fitting provides positive identification.

An internal hollow post in the center of the end fitting provides means for retention of either an orifice rod assembly or a burnable poison rod assembly.

Attached to the upper end fitting is a holddown spring and spider. This spring provides a positive holddown margin to oppose hydraulic forces resulting from the flow of the primary coolant.

Penetrations in the upper end fitting grid are provided for the guide tubes.

5. Guide Tubes

The Zircaloy guide tubes provide continuous guidance for the control rod assemblies when inserted in the fuel assembly and provide the structural continuity for the fuel assembly. Welded to each end of a guide tube are flanged and threaded sleeves, which secure the guide tubes to each end fitting by lock-welded nuts. Transverse location of the guide tubes is provided by the spacer grids.

6. Instrumentation Tube Assembly

This assembly serves as a channel to guide, position, and contain the in-core instrumentation within the fuel assembly. The instrumentation probe is guided up through the lower end fitting to the desired core elevation. It is retained axially at the lower end fitting by a retainer sleeve.

7. Spacer Sleeves

The spacer tube segments fit around the instrument tube between spacer grids and prevent axial movement of the spacer grids during primary coolant flow through the fuel assembly.

Evaluation

a. General

The basis for the design of the fuel rod is discussed in 3.1.2.4. Materials testing and actual operation in reactor service with Zircaloy cladding have demonstrated that Zircaloy-4 material has sufficient corrosion resistance and mechanical properties to maintain the integrity and serviceability required for design burnup.

b. Clad Stress and Strain

The cladding of fuel rods is subjected to external hydrostatic pressure, gradually increasing internal pressure, thermal stresses, vibration, and to the effects of differential expansion of the fuel and cladding caused by thermal expansions and by fuel growth due to irradiation effects. In addition, the properties of the cladding are influenced by thermal and irradiation effects. The analysis of these effects is discussed below.

Stress analysis for cladding is based on several conservative assumptions that make the actual margins of safety greater than those calculated. For example, it is assumed that the clad with the thinnest wall, the smallest fuel-clad gap, and the greatest ovality permitted by the specification is operating in the region of the core where performance requirements are most severe. Fission gas release rates, fuel growth, and changes in mechanical properties with irradiation are based on a conservative evaluation of currently available data.

c. Pressure Effects

Clad stresses due to internal and external pressure are considerably
below the material yield strength. Circumferential stresses due to
differential external pressure, calculated using those combinations
of clad dimensions, ovality, and eccentricity that produce the highest
stress, for pressurized and unpressurized fuel rods are shown in
Table 3-18. The maximum stress of 34,500 psi compression occurs in
the unpressurized fuel rod in the end void region. The stress is
composed of 21,500 psi compressive membrane stress plus 13,000 psi
compressive bending stress due to ovality at the clad O.D. For the
unpressurized fuel rods, the maximum stress in the heat-producing
9. zone is 34,200 psi at the design pressure and 28,700 psi at operating
pressure. The stresses for pressurized fuel rods are less than
stresses stated above for unpressurized rods.

In the heat producing zone, the stress and temperature is such that
the clad material may creep enough to allow an increase in clad
ovality until further creep is restrained by support from the fuel.
If fuel-clad contact occurs, the clad is subject to cyclic stresses/
strains which are a result of power and pressure transients. To
minimize clad fatigue damage, all fuel rods will be internally
pressurized with helium. Fatigue analyses, based on conservative
24. assumptions, show that the design limits previously specified
(3.1.2.4.2) are met for both pressurized and unpressurized fuel rods.

9. Even at the end of life, the fission gas pressure for unpressurized fuel rods does not exceed system operating pressure (3.2.3.2.3). The upper limit of internal pin pressure for pressurized fuel rods at the end of life has been conservatively calculated to be less than 2,600 psig. An internal pressure of 3,300 psi has been selected as the design basis. At this pressure the differential would result in a 9,000 psi tensile circumferential stress during operation at system pressure. This is about 1/4 of the yield strength and, therefore, is not a potential source of short-time burst. The possibility of stress-rupture burst has been investigated using finite-difference methods to estimate the long-time effects of the increasing design pressure on the clad. The predicted pressure-time relationship pro-
9. duces stresses that are approximately 1/3 of the stress levels that would produce stress rupture at the end of life. Outpile stress-rupture data were used, but the approximately 3:1 margin on stress is more than enough to account for decreased stress-rupture strength due to irradiation.

Table 3-18
Clad Circumferential Stresses

	Operating Condition	Calc. Stress, psi	Yield Stress, psi	Ultimate Tensile Stress, psi
1. & 3.	1. BOL[(a)] Expansion Void Clad			
	Conditions at Maximum Overpower			
	Total stress (membrane + bending) due to 2,500 psig system design pressure and internal pin pressure. Average clad temperature 650 F.			
	a. Unpressurized Fuel Rod Internal pin pressure of 100 psig.			
9. 1.	Stress	(-)34,500	45,000	
	b. Pressurized Fuel Rod Internal pin pressure of 1400 psig.			
	Stress	(-)13,200	45,000	
	2. BOL - Fueled Section Clad			
	Conditions at Maximum Overpower			
1.	Total stress (membrane + bending) due to differential pressures. Average clad temperature 729 F.			

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Table 3-18 (Cont'd)

Operating Condition	Calc. Stress, psi	Yield Stress, psi	Ultimate Tensile Stress, psi
b. Pressurized Fuel Rods			
9. Internal Pin Pressure of 1,500 psig			
Stress	(-) 6,100	44,000	47,000
5. EOL Shutdown			
1-1/3 Hours Later			
Conditions			
(50 F/h Pressurizer Cooldown Rate First Hour, up to 100 F/h Thereafter)			
Primary System Cooldown, 100 F/h			
1. System Pressure - 1,275 psig			
Average Clad Temperature - 400 F			
a. Unpressurized Fuel Rod			
Internal pin pressure 1,090 psig			
Stress	(-) 1,700	52,000	55,000
b. Pressurized Fuel Rod			
9. Internal pin pressure 1,200 psig			
Stress	(-) 700	52,000	55,000

(a) Cladding is specified with 45,000 psi minimum yield strength and 10 per cent minimum elongation, both at 650 F. Minimum room temperature strengths are approximately 75,000 psi yield strength (0.2 per cent offset) and
26 85,000 psi ultimate tensile strength.

(b) Cladding stresses due to fuel swelling are discussed further in a subsequent paragraph of this section, Effect of Zircalcy Creep.

(c) At 3300 psig internal fuel rod (design pressure.

The total production of fission gas in the hottest fuel rod assembly is based on the hot rod average burnup of 38,000 MWd/MTU. The corresponding maximum burnup at the hot fuel rod midpoint is 42,000 MWd/MTU. The design burnup is 55,000 MWd/MTU.

The fission gas release is based on temperature versus release fraction as shown in Figure 3-36. Fuel temperatures are calculated for small

27.

Table 3-18 (Cont'd)

Operating Condition	Calc. Stress, psi	Yield Stress, psi	Ultimate Tensile Stress, psi
b. Pressurized Fuel Rods			
9. Internal Pin Pressure of 1,500 psig			
Stress	(-) 6,100	44,000	47,000
5. EOL Shutdown 1-1/3 Hours Later Conditions			
(50 F/h Pressurizer Cooldown Rate First Hour, up to 100 F/h Thereafter)			
Primary System Cooldown, 100 F/h			
1. System Pressure - 1,275 psig			
Average Clad Temperature - 400 F			
a. Unpressurized Fuel Rod			
Internal pin pressure 1,090 psig			
Stress	(-) 1,700	52,000	55,000
b. Pressurized Fuel Rod			
9. Internal pin pressure 1,200 psig			
Stress	(-) 700	52,000	55,000

(a) Cladding is specified with 45,000 psi minimum yield strength and 10 per cent minimum elongation, both at 650 F. Minimum room temperature strengths are approximately 75,000 psi yield strength (0.2 per cent offset) and 85,000 psi ultimate tensile strength.

(b) Cladding stresses due to fuel swelling are discussed further in a subsequent paragraph of this section, Effect of Zircaloy Creep.

(c) At 3300 psig internal fuel rod (design pressure.

The total production of fission gas in the hottest fuel rod assembly is based on the hot rod average burnup of 38,000 MWd/MTU. The corresponding maximum burnup at the hot fuel rod midpoint is 42,000 MWd/MTU. The design burnup is 55,000 MWd/MTU.

The fission gas release is based on temperature versus release fraction as shown in Figure 3-36. Fuel temperatures are calculated for small radial and axial increments. The total fission gas release is calculated by integrating the incremental releases.

The maximum release and gas pressure buildups are determined by evaluating the following factors for the most conservative conditions.

(1) Gas conductivity at the end of life with fission gas present.

(2) Influence of the pellet-to-clad radial gap and contact heat transfer coefficient on fuel temperature and release rate.

(3) Unrestrained radial and axial thermal growth of the fuel pellets relative to the clad.

(4) Hot rod local peaking factors.

(5) Fuel temperature at reactor design overpower--The fuel temperatures used to determine fission gas release and internal gas pressure have been calculated at the reactor overpower condition (114 per cent). Fuel temperatures, total free gas volume, fission gas release, and internal gas pressure have been evaluated for a range of initial diametral clearances. This evaluation shows that the highest internal pressure results when the maximum diametral gap is assumed because of the resulting high average fuel temperature (Figure 3-39). The release rate increases rapidly with an increase in fuel temperature, and unrestrained axial growth reduces the relatively cold gas end plenum volumes. A conservative thermal expansion model is used to calculate fuel temperatures as a function of initial cold diametral clearance as outlined in 3.2.3.2.3.g(1).

9. (d) Design pressure exceeds calculated EOL pressure for both pressurized and unpressurized rods.

Collapse Margins

Short-time collapse tests have demonstrated a clad collapsing pressure in excess of 4,000 psi at expansion void maximum temperature. The collapse pressure margin for unpressurized fuel rods is approximately 1.7. Extrapolation to hot spot
9. average clad temperature (729 F) indicates a collapse pressure of 3,500 psi which gives a collapse margin of 1.4 for unpressurized fuel rods. The collapse margins for pressurized fuel rods exceed those for unpressurized rods. Outpile creep collapse tests have demonstrated that the clad meets the long-time (creep) collapse) requirement. Back-up radial support has been provided in the upper end void to assure clad dimensional stability in the vent that in-pile creep rates are sufficiently high to allow creep collapse of unsupported cladding. Test results summarized in Section 3.3.3.3.1 show the end void spacers are capable of providing backup support. The results of the tests show that creep collapse of the bottom end void will not occur since the clad temperature is about 90 F lower than that in the upper void region. The spacer in the bottom end void is therefore not required to provide radial support. Its geometry, however, is similar to the upper spacer, and it therefore provides added assurance of clad dimensional stability at the bottom void region.

Fuel Irradiation Growth and Fuel-Clad Differential Thermal Expansion

The results of tests and the operation of Zircaloy-clad UO_2 fuel rods indicate that the rods can be safely operated to the point where total permanent strain is 1-1/2 per cent, or higher, in the temperature range applicable to PWR cladding.(40) The design allowable strain is about 1 per cent (3.1.2.4.2.c).

Fuel rod operating conditions pertinent to fuel swelling considerations are listed below for end-of-life conditions.

Burnup (Design Value, MWd/MTU)	55,000
Minimum Fuel-to-Clad Gap (Beginning of Life), in.	0.0045
Pellet Nominal Diameter, in.	0.370
Pellet Density (Per Cent of Theoretical), %	93.5
Cladding (Zircaloy-4) Thickness, in.	0.0265

The capability of Zircaloy-clad UO_2 fuel in solid rod form to perform satisfactorily in service has been demonstrated through operation of the SA-1 assembly in the Dresden and Shippingport cores, and through results of their supplementary development programs, up to approximately 45,000 MWd/MTU.

As outlined below, existing experimental information supports the various individual design parameters and operating conditions up to and perhaps beyond the maximum design burnup of 55,000 MWd/MTU, but not in a single experiment. However, the B&W High Burnup Irradiation Program currently in progress does combine the primary items of concern in a single experiment, and the results will be available prior to operating.

Application of Experimental Data to Design Adequacy of the Clad-Fuel Initial Gap to Accommodate Clad-Fuel Differential Thermal Expansion

Experimental Work

Six rabbit capsules, each containing three Zr-2 clad rods of 5-in. fuel length, were irradiated in the Westinghouse Test Reactor(41) at power levels up to 24 kW/ft. The 94 per cent theoretical density (TD) UO_2 pellets (0.430 OD) had initial clad-fuel diametral gaps of 6, 12, and 25 mils. No dimensional changes were observed. Central melting occurred at 24 kW/ft only in the rods that had the 25 mil initial gap.

Two additional capsules were tested.(42) The specimens were similar to those described above except for length and initial gap. Initial gaps of 2, 6, and 12 mils were used in each capsule. In the A-2 capsule, three 38-in.-long rods were irradiated to 3,450 MWd/MTU at 19 kW/ft maximum. In the A-4 capsule, four 6-in.-long rods were irradiated to 6,250 MWd/MTU at 22.2 kW/ft maximum. No central melting occurred in any rod, but diameter increases up to 3 mils in the A-2 capsule and up to 1.5 mils in the A-4 capsule were found in the rods with the 2 mil initial gap.

Application

In addition to demonstrating the adequacy of Zircaloy-clad UO_2 pellet rods to operate successfully at the power levels of interest (and without central melting), these experiments demonstrate that the design initial clad-fuel gap of 4.5 to 9.5 mils is adequate to prevent unacceptable clad diameter increase due to differential thermal expansion between the clad and the fuel at beginning of life. A maximum local diametral increase of less than 0.001 in. is indicated for fuel rods having the minimum initial gap, operating at the maximum overpower condition.

Adequacy of the Available Voids to Accommodate Differential Expansion of Clad and Fuel, Including the Effects of Fuel Swelling

Experimental Work

Zircaloy-clad, UO_2 pellet-type rods have performed successfully in the Shippingport reactor up to approximately 40,000 MWd/MTU. Bettis Atomic Power Laboratory[39] has irradiated plate-type UO_2 fuel (96-98 per cent TD) up to 127,000 MWd/MTU and at fuel center temperatures between 1,300 and 3,800 F. This work indicates fuel swelling rates of 0.16 per cent $\Delta V/10^{20}$ f/cc until fuel internal voids are filled, then 0.7 per cent $\Delta V/10^{20}$ f/cc after internal voids are filled. This point of "breakaway" appears to be independent of temperature over the range studies and dependent on clad restraint and the void volume available for collection of fission products. The additional clad restraint and greater fuel plasticity (from higher fuel temperatures) of rod-type elements tend to reduce these swelling effects by providing greater resistance to radial swelling and lower resistance to longitudinal swelling than was present in the plate-type test specimens.

This is confirmed in part by the work of Frost, Bradbury, and Griffiths of Harwell[43] in which 1/4-in. diameter UO_2 pellets clad in 0.020 in. stainless steel with a 2 mil diametral gap were irradiated to 53,300 MWd/MTU at a fuel center temperature of 3,180 F without significant dimensional change.

In other testing[44] 0.150-in. OD, 82-96 per cent TD oxide pellets (20 per cent Pu, 80 per cent U) clad with 0.016-in. stainless steel with 6-8 mil diametral gaps have been irradiated to 77,000 MWd/MTU at fuel temperatures high enough to approach central melting without apparent detrimental results. Comparable results were obtained on rods swaged to 75 per cent TD and irradiated to 100,000 MWd/MTU.

Application

Based on the BAPL experimental data, swelling of the fuel rods is estimated as outlined below.

The fuel is assumed to swell uniformly in all directions, conservatively neglecting axial plastic flow into the end dishes. Thermal expansions are calculated as described in 3.2.3.2.3.g. If the fuel cracks, the crack voids are assumed to be available to absorb fuel growth.

The external effect of fuel swelling is assumed to occur at 0.16 per cent $\Delta V/10^{20}$ f/cc until the as-fabricated void in the 93.5 per cent pellets is filled. From that time on, swelling is assumed to take place at 0.7 per cent $\Delta V/10^{20}$ f/cc
1. until the maximum burnup of 13.4×10^{20} f/cc (55,000 MWd/MTU) is reached.

Studies of clad strain at various gaps indicate that the rod with the minimum gap experiences the greatest clad strain in spite of its improved gap conductivity. Clad permanent strain reaches a maximum at the end of life, and is 0.7 per cent for nominal density fuel. Clad strain for fuel rods with maximum density allowed by the specification will also meet the design maximum allowable permanent strain. The fuel in rods with nominal gaps, nominal density and average burnup will not grow sufficiently to cause a tensile hoop stress or strain in the cladding.

Fuel Swelling Studies at B&W

Experimental fuel swelling studies under inpile conditions simulating large reactor environments are under way. Parameters contributing to swelling are burnup, heating rate, fuel density and grain size, and clad restraint. These are being studied systematically by irradiating a series of capsules containing fuel rods. Test variables are shown in Table 3-19, and the program's schedule is given in Table 3-20. See also 3.3.3.3.3.

Test variables include heat rate, burnup, clad thickness, and fuel-to-clad gap. Postirradiation examination will include investigation of dimensional changes, metallographic examination of fuel and cladding, fission gas release correlations with test conditions, and other related observations.

Effect of Zircaloy Creep

The effect of zircaloy creep on the amount of fuel rod growth due to fuel swelling has been investigated. Clad creep has the effect of producing a nearly constant total pressure on the clad ID by permitting the clad diameter to increase as the fuel diameter increases. Based on out-of-pile data,[45] 1 per cent creep will result in 10,000 hours (corresponding approximately to the end-of-life diametral swelling rate) from a stress of about 22,000 psi at the 720 F average temperature through the clad at the hot spot. At the start of this higher swelling period (roughly the last 1/3 of the core life), the reactor coolant system pressure would more or less be balanced by the rod internal pressure, so the total pressure to produce the clad stress of 22,000 psi would have to come from the fuel. Contact pressure would be 2400 psi. At the end of life, the rod internal design pressure exceeds the system pressure by about 1,100 psi, so the clad fuel contact pressure would drop to 1,300 psi. Assuming that irradiation produces a 3:1 increase in creep rates, the clad stress for 1 per cent strain in 10,000 hours would drop to about 15,000 psi. Contact pressures would be 1,800 psi at the beginning of the high swelling period, 700 psi at the end of life. Since the contact pressure was assumed to be 825 psi in calculating the contact coefficient used to determine the fuel pellet thermal expansion, there is only a short period at the very end of life (assuming the 3:1 increase in creep rates due to irradiation) when the pellet is slightly hotter than calculated. The effect of this would be a slight increase in pellet thermal expansion and therefore in clad strain.

Table 3-19

B&W High Burnup Irradiation Program - Capsule Fuel Test

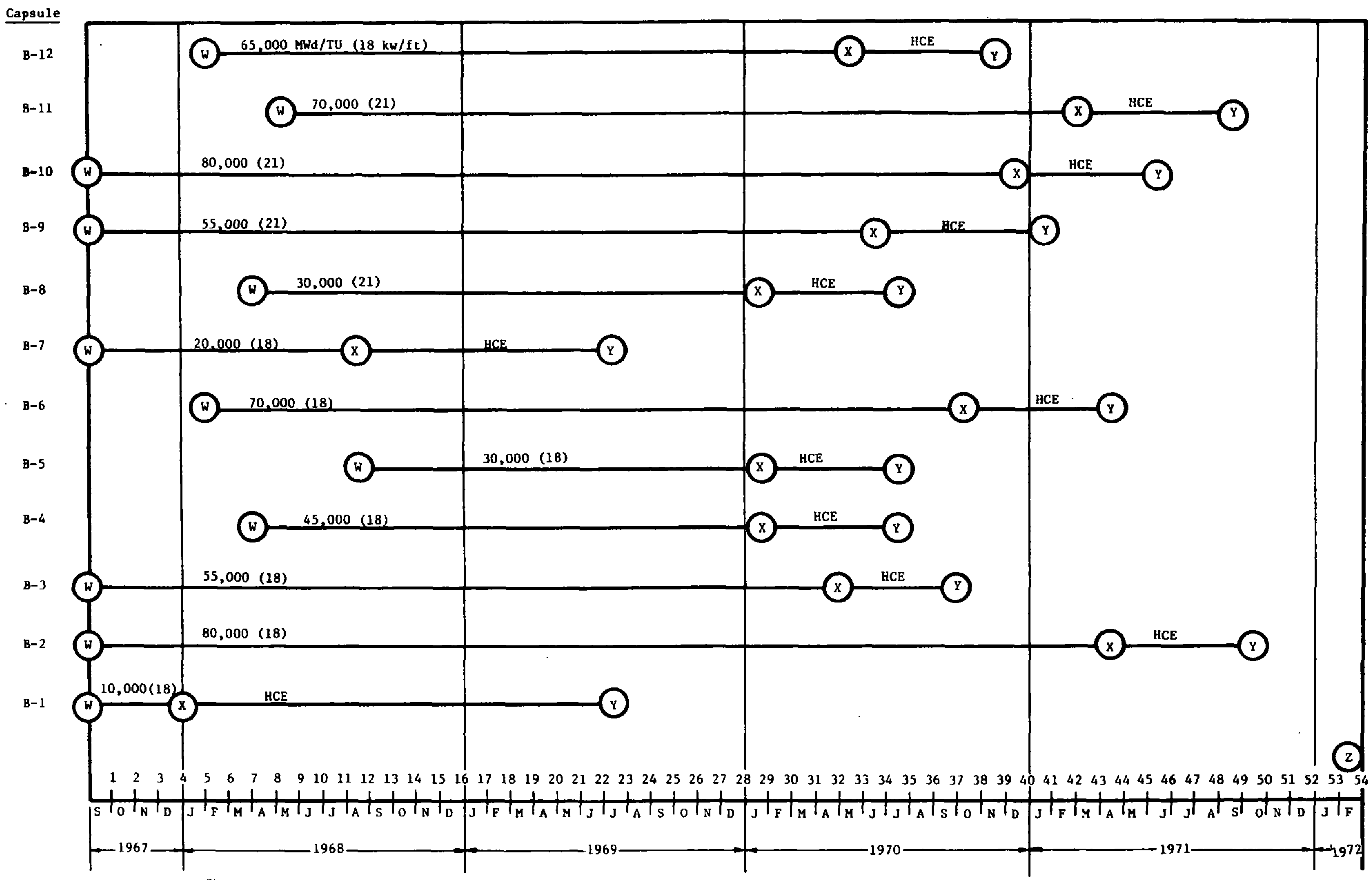
Identification			Burnup			Heat Rate			
Capsule	Fuel Rod	Irradiation Facility (3)	MWD/MTU $\times 10^{-3}$	Fissions/cc $\times 10^{-20}$ (1)	Irradiation Time, calendar months (2)	Initial, kw/ft	Final, kw/ft	Diametral Gap, mils	Clad Thickness, mils
B-1	B-1	RS-3	10	2.5	4	18	17.5	4-5	25
	B-2			2.5			17.5	7-8	25
	B-3			2.5			17.5	7-8	15
B-2	B-20	RS-4	17	3.8	10	18	16.9	Powder	25
	B-19		26	6.5			16.9	4-5	25
B-3	B-7	RS-6	30	7.5	11	18	16.1	4-5	25
	B-8			7.5			16.1	7-8	25
	B-9			7.5			16.1	7-8	15
B-4	B-10	RS-1	45	10.05	17	18	14.9	Powder	25
	B-11			11.25			14.9	4-5	25
B-5	B-13	RS-5	55	13.75	21	18	14.1	4-5	25
	B-14			13.75			14.1	7-8	25
	B-15			13.75			14.1	7-8	15
B-6	B-31	RS-2	70	15.63	26	18	13.3	Powder	25
	B-17			17.5			13.3	7-8	25
B-7	B-5	RS-5	80	17.87	30	18	12.5	Powder	25
	B-4			20.0			12.5	7-8	25
B-8	B-22	RL-1	30	7.5	11	21-1/2	20.5	4-5	25
	B-23			7.5			20.5	7-8	25
	B-33			7.5			20.5	7-8	15
B-9	B-25	RL-2	55	13.75	21	21-1/2	19.3	4-5	25
	B-26			13.75			19.3	7-8	25
	B-27			13.75			19.3	7-8	15
B-10	B-28	RL-3	80	20.0	30	21-1/2	17.8	7-8	25
	B-29			20.0			17.8	7-8	25
	B-30			20.0			17.8	7-8	15
B-11	B-24	RL-4	70	17.5	26	21-1/2	16.5	7-8	15
	B-34			17.5			16.5	7-8	25
	B-35			17.5			16.5	7-8	25
B-12	B-16	RS-3	65	14.50	24	18	13.3	Powder	25
	B-32			16.24			13.3	7-8	15

(1) Based on 200 Mev per fission.

(2) Based on 80 per cent reactor efficiency.

(3) Irradiation Location in BAWTR.

Table 3-20
B&W High Burnup Irradiation Program Schedule



LEGEND:
W - Start Irradiation
X - Complete Irradiation
Y - Interim Report Issued
Z - Final Report Issued
Capsules B-1 thru B-12
HCE - Hot Cell Examination

Overall Assembly

a. Assurance of Control Rod Assembly Free Motion

The 0.058 in. diametral clearance between the control rod guide tube and the control rod is provided to cool the control rod and to insure adequate freedom to insert the control rod. As indicated below, studies have shown that fuel rods will not bow sufficiently to touch the guide tube. Thus, the guide tube will not undergo deformation caused by fuel rod bowing effects. Initial lack of straightness of fuel rod and guide tube, plus other adverse tolerance conditions, conceivably could reduce the 0.088 in. nominal gap between fuel rod and guide tube to a minimum of about 0.038 in., including amplification of bowing due to axial friction loads from the spacer grid. The maximum expected flux gradient of 1.176 across a fuel rod will produce a temperature difference of 12 F, which will result in a thermal bow of less than 0.002 in. Under these conditions, for the fuel rod to touch the guide tube, the thermal gradient across the fuel rod diameter would have to be on the order of 300 F.

The effect of a DNB occurring on the side of a fuel rod adjacent to a guide tube would result in a large temperature difference. In this case, however, investigation has shown that the clad temperature would be so high that insufficient strength would be available to generate a force of sufficient magnitude to cause a significant deflection of the guide tube. In addition, the guide tube would experience an opposing gradient that would resist fuel rod bowing, and its internal cooling would maintain temperatures much lower than those in the fuel rod cladding, thus retaining the guide tube strength.

b. Vibration

The semiempirical expression developed by Burgreen[46] was used to calculate the flow-induced vibratory amplitudes for the fuel assembly and fuel rod. The calculated amplitude is less than 0.010 in. for the fuel assembly and less than 0.005 in. for the fuel rod. The fuel rod vibratory amplitude correlates with the measured amplitude obtained from a test on a 3 x 3 fuel rod assembly. In order to substantiate this conservatively calculated amplitude for the fuel assembly, a direct measurement has been obtained for a full size prototype fuel assembly during testing of the assembly in the Control Rod Drive Line Facility (CRDL) at the B&W Research Center, Alliance, Ohio. The maximum assembly amplitude determined by measurement was 0.005 inches.

c. Demonstration

In addition to the specific items discussed above, the overall mechanical performance of the fuel assembly and its individual components has been and is continuing to be demonstrated in an extensive experimental program in the CRDL (Paragraph 3.3.3.2).

3.2.4.2.2 Control Rod Assembly (CRA)

Each control rod assembly (Figure 3-54) has 16 control rods, a stainless steel spider, and a female coupling. The 16 control rods are attached to the spider by means of a nut threaded to the upper shank of each rod. After assembly, all nuts are lock welded. The control rod drive is coupled to the CRA by a bayonet type connection. Full length guidance for the CRA is provided by the control rod guide tube of the upper plenum assembly and by the fuel assembly guide tubes. The CRA's and guide tubes are designed with adequate flexibility and clearances to permit freedom of motion within the fuel assembly guide tubes throughout the stroke.

Each control rod has a section of neutron absorber material. The absorber material is an alloy of silver-indium-cadmium. It is clad in cold-worked type 304 stainless steel tubing. Stainless steel end pieces are welded to the tubing to form a water- and pressure-tight container for the absorber material. The stainless steel tubing provides the structural strength of the control rods and prevents corrosion of the absorber material. A tube spacer similar to that in the fuel assembly is used to prevent absorber motion within the cladding during shipping and handling, and to permit differential expansion in service.

Principal data pertaining to the CRA are shown in Table 3-21.

Table 3-21
Control Rod Assembly Data

Item	Data
Number of CRA	61
Number of Control Rods per Assembly	16
Outside Diameter of Control Rod, in.	0.440
3. Cladding Thickness, in.	0.021
Cladding Material	Type 304 SS, Cold-Worked
End Plug Material	Type 304 SS, Annealed
Spider Material	SS Grade CF3M
Poison Material	80% Ag, 15% In, 5% Cd
Female Coupling Material	Type 304 SS, Annealed
Length of Poison Section, in.	134
Stroke of Control Rod, in.	139

A CRA prototype of the B&W design has been extensively tested at reactor temperature, pressure, and flow conditions in the B&W test loop at their Alliance Research Laboratory. For test program description and results, refer to BAW-
21. 10029, "Control Rod Drive System Test."

These control rods are designed to withstand all operating loads including those resulting from hydraulic force, thermal gradients, and reactor trip deceleration. The ability of the control rod clad to resist collapse has been established in a test program on cold-worked stainless steel tubing, reported in the PSAR. Because the Ag-In-Cd alloy poison does not yield a gaseous product under irradiation, internal pressure and swelling of the absorber material will not cause excessive stressing or stretching of the clad.

Because of their length and the possible lack of straightness over the entire length of the rod, some interference between control rods and the fuel assembly guide tubes is expected. However, the parts involved, especially the control rods, are flexible and only small friction drag loads result. Similarly, thermal distortions of the control rods are small because of the low heat generation and adequate cooling. Consequently, control rod assemblies will not encounter significant frictional resistance to their motion in the guide tubes.

3.2.4.2.3 Axial Power Shaping Rod Assembly (APSRA)

Each axial power shaping rod assembly (Figure 3-53) has 16 axial power shaping rods, a stainless steel spider, and a female coupling. The 16 rods are attached to the spider by means of a nut threaded to the upper shank of each rod. After assembly all nuts are lock welded. The axial power shaping rod drive is coupled to the APSRA by a bayonet connection. The female couplings of the APSRA and CRA have slight dimensional differences to ensure that each type of rod can only be coupled to the correct type of drive mechanism.

When the APSRA is inserted into the fuel assembly it is guided by the guide tubes of the fuel assembly. Full length guidance of the APSRA is provided by the control rod guide tube of the upper plenum assembly. At the full out position of the control rod drive stroke, the lower end of the APSRA remains within the fuel assembly guide tube to maintain the continuity of guidance throughout the rod travel length. The APSRA's are designed to permit maximum conformity with the fuel assembly guide tube throughout travel.

Each axial power shaping rod has a section of neutron absorber material. This absorber material is an alloy of silver-indium-cadmium and is clad in cold-worked, Type 304 stainless steel tubing. The tubing provides the structural strength of the axial power shaping rods and prevents corrosion of the absorber material. Above the section containing the absorber material is a tubular follower made of cold-worked Zircaloy-4 tubing. This follower is attached to the absorber section by pinning to the end piece and is vented to permit the coolant-moderator to fill the void. Pertinent data on the APSRA is shown in Table 3-22.

Table 3-22

Axial Power Shaping Rod Assembly Data

Item	Data
Number of Axial Power Shaping Rod Assemblies	8
Number of Axial Power Shaping Rods per Assembly	16
Outside Diameter of Axial Power Shaping Rod, in.	0.440
1. Cladding Thickness, in.	0.021
Cladding Material	Type 304 Stainless Steel Cold Worked
16. Plug Material	Type 304 SS, Annealed
Poison Material	80% Ag, 15% In, 5% Cd
Spider Material	SS, Grade CF3M
Female Coupling Material	Type 304, SS, Annealed
Length of Poison Section, in.	36
Stroke of Control Rod, in.	139

These axial power shaping rods are designed to withstand all operating loads including those resulting from hydraulic forces and thermal gradients. The ability of the axial power shaping rod clad to resist collapse due to the system pressure has been established in a test program on cold worked stainless steel tubing. Because the Ag-In-Cd alloy does not yield gaseous products under irradiation, internal pressure is not generated within the clad. Swelling of the absorber material is negligible, and will not cause unacceptable clad strain.

Because of their great length and unavoidable lack of straightness, some slight mechanical interference between axial power shaping rods and the fuel assembly guide tubes must be expected. However, the parts involved are flexible and result in very small friction drag loads. Similarly, thermal distortions of the rods are small because of the low heat generation and adequate cooling. Consequently, the APSRA's will not encounter significant frictional resistance to their motion in the guide tubes.

3.2.4.2.4 Burnable Poison Rod Assembly (BPRA)

Each BPRA (Figure 3-51) has 16 burnable poison rods, a stainless steel spider, and a coupling mechanism. The coupling mechanism and the 16 rods are attached to the spider. The BPRA is inserted into the fuel assembly guide tubes through the upper end fitting. The coupling mechanism provides a means for positive coupling between the BPRA and the fuel assembly holddown latch.

The burnable poison rod is clad in cold-worked Zircaloy-4 tubing and Zircaloy-4 upper and lower end pieces. The end pieces are welded to the tubing to form a water and pressure-tight container for the absorber material. The Zircaloy-4 tubing provides the structural strength of the burnable poison rods.

In addition to their nuclear function, the BPRA also serve to minimize guide tube bypass coolant flow. Pertinent data on the BPRA is shown in Table 3-23.

Table 3-23
Burnable Poison Rod Assembly Data

	Item	Data
	Number of Burnable Poison Rods per Assembly	16
	Outside Diameter of Burnable Poison Rod, in.	0.430
16.	Cladding Thickness, in.	0.035
	Cladding Material	Zircaloy-4, Cold Worked
	End Cap Material	Zircaloy-4, Annealed
	Poison Material	B_4C in. Al_2O_3
16.	Length of Poison Section, in.	126
	Spider Material	SS, Grade CF3M
	Coupling Mechanism Material	Type 304 SS, Annealed

The burnable poison rods are designed to withstand all operating loads including those resulting from hydraulic forces and thermal gradients. The ability of the burnable poison rod clad to resist collapse due to the system pressure and internal pressure has been demonstrated by an extensive test program on cold-worked Zircaloy-4 tubing (3.3.3.3).

3.2.4.2.5 Orifice Rod Assembly (ORA)

Each orifice rod assembly (Figure 3-50) has 16 orifice rods, a stainless steel spider, and a coupling mechanism. The coupling mechanism provides a means for positive coupling between the ORA and the fuel assembly holddown latch when the orifice rods are inserted into the fuel assembly. The rods are spring loaded as shown to permit lateral movement of the rods in order to facilitate the instal-
1. lation of the orifice assembly into the guide tubes in the fuel assembly. The ORA serves to limit bypass flow through empty guide tubes. Pertinent data on the ORA
30. is shown in Table 3-24. A maximum of four test orifice rod assemblies may be included in Oconee 2; see Supplement 16.

Table 3-24
Orifice Rod Assembly Data

	Item	Data
	Number of Orifice Rods per Assembly	16
	Orifice Rod Material	Type 304 SS, Annealed
	Spider Material	SS, Grade CF3M
16.	Coupling Mechanism Material	Type 304 SS, and 17-4 pH, condition H 1100

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16. 3.2.4.3 Control Rod Drives (Reference Supplement 9 Revisions for Oconee 3)

3.2.4.3.1 Description

The control rod drive mechanism (CRDM) positions the control rod within the reactor core and indicates the location of the control rod with respect to the reactor core. The speed at which the control rod is inserted or withdrawn from the core is consistent with the reactivity change requirements during reactor operation. For conditions that require a rapid shutdown of the reactor, the shim safety drive mechanism releases the CRA and supporting CRDM components permitting the CRA to move by gravity into the core. The reactivity is reduced during such a rod insertion at a rate sufficient to control the core under any operating transient or accident condition. The control rod is decelerated at the end of the rod trip insertion by a buffer assembly in the CRDM upper housing. The buffer assembly supports the control rod in the fully inserted position. Criteria applicable to drive mechanisms for both control shim rod assemblies and axial power shaping rod assemblies are given below. Additional requirements for the mechanisms which actuate only control shim rod assemblies are also given below.

General Design Criteria

a. Single Failure

No single failure shall inhibit the protective action of the control rod drive system. The effect of a single failure shall be limited to one CRDM.

b. Uncontrolled Withdrawal

No single failure or sequence of dependent failures shall cause uncontrolled withdrawal of any control rod assembly (CRA).

c. Equipment Removal

The disconnection of plug-in connectors, modules, and subassemblies from the protective circuits shall be annunciated or shall cause a reactor trip.

d. Position Indication

Continuous position indication, as well as an upper and lower position limit indication, shall be provided for each CRDM. The accuracy of the position indicators shall be consistent with the tolerance set by reactor safety analysis.

e. Drive Speed

The control rod drive control system shall provide a single uniform mechanism speed. The drive controls, or mechanism and motor combination, shall have an inherent speed limiting feature. The speed of the mechanism shall be 30 in./min for both insertion and withdrawal. The withdrawal speed shall be limited to not exceed 25 per cent overspeed in the event of speed control fault.

f. Mechanical Stops

Each CRDM shall have positive mechanical stops at both ends of the stroke or travel. The stops shall be capable of receiving the full operating force of the mechanisms without failure.

g. Control Rod Positioning

The control rod drives shall provide for controlled withdrawal or insertion of the control rods out of, or into, the reactor core to establish and hold the power level required.

Additional Design Criteria

The following criterion is applicable only to the mechanisms which actuate control rod assemblies:

a. CRA Trip

The shim safety drives are capable of rapid insertion or trip for emergency reactor conditions.

Periodic venting of the center control rod drive mechanism during operation is planned for Oconee Unit 1 until adequate confidence levels show that venting is no longer useful or necessary. To accomplish this venting, a control rod drive insert assembly has been designed and will be mounted on the center drive in Unit 1. Seismically designed piping will then connect this closure assembly to a high pressure sample container outside the secondary shield wall. The liquid level in the high pressure container will be monitored to detect any gas.
18. The liquid sample will be returned to the reactor coolant system via another line. Double valving outside the secondary shield wall isolates the sample container when not in use.

To provide a reactor coolant system liquid sample for total dissolved gas concentration analysis, a sample line has been added to the letdown line (see Figure 9-2A). Technical Specification 3.1.10 describes the pressure versus temperature relationship for dissolved gases in the reactor coolant system.

All CRD mechanisms will be vented during initial heat up and after operation of the reactor coolant pumps.

3.2.4.3.2 Control Rod Drive Mechanisms

The control rod drive mechanisms provide for controlled withdrawal or insertion of the control rod assemblies out of or into the core and are capable of rapid insertion or trip. The drive mechanisms are sealed, reluctance motor-driven screw units. The CRDM data is listed in Table 3-25.

Table 3-25
Control Rod Drive Mechanism Design Data

Mechanism Function	Shim Safety	Axial Power Shaping
Type	Roller Nut Drive	Roller Nut Drive
Quantity	61	8
Location	Top-mounted	Top-mounted
Direction of Trip	Down	Does not trip
Velocity of Normal Withdrawal and Insertion, in./min.	30	30
Maximum Travel Time for Trip		
2/3 Insertion, s	1.40	Drive has no trip function
3/4 Insertion, s	1.50	Drive has no trip function
Length of Stroke, in.	139	139
Design Pressure, psig	2,500	2,500
Design Temperature, F	650	650
Weight of Mechanism (App.)	940 lb	940 lb

Shim Safety Drive Mechanism

The shim safety drive mechanism consists of a motor tube which houses a lead
screw and its rotor assembly, and a buffer. The top end of the motor tube is
16. closed by a closure and vent assembly. An external motor stator surrounds the
motor tube (a pressure housing) and position indication switches are arranged
outside the motor tube extension.

The control rod drive output element is a non-rotating translating lead screw
coupled to the control rod. The screw is driven by separating anti-friction
roller nut assemblies which are rotated magnetically by a motor stator located
outside the pressure boundary. Current impressed on the stator causes the
separating roller nut assembly halves to close and engage the lead screw.
Mechanical springs disengage the roller nut halves from the screw in the ab-
sence of a current. For rapid insertion, the nut halves separate to release
the screw and control rod, which move into the core by gravity. A hydraulic
buffer assembly within the upper housing decelerates the moving CRA to a low
speed a short distance above the CRA full-in position. The final CRA deceler-
ation energy is absorbed by the down-stop buffer spring. The CRDM is a totally
sealed unit with the roller nut assemblies magnetically driven by the stator
coil through the motor tube pressure housing wall. The lead screw assembly is
connected to the control rod by a bayonet type coupling. An anti-rotation de-
vice (torque taker) prevents rotation of the lead screw while the drive is in
16. service. A closure and vent assembly is provided at the top of the motor tube
housing to permit access to couple and release the lead screw assembly from
the control rod. The top end of the lead screw assembly is guided by the buf-
fer piston and its guide. Two of the six phase stator housing windings are
energized to maintain the control rod position when the drive is in the hold-
ing mode.

The CRDM is shown in Figures 3-55 and 3-56. Subassemblies of the CRDM are
described as follows:

a. <u>Motor Tube</u>

The motor tube is a three-piece welded assembly designed and manu-
factured in accordance with the requirements of the ASME Code, Sec-
tion III, for Class A nuclear pressure vessel. Materials conform to
ASTM or ASME, Section II, Material Specifications. All welding shall
be performed by personnel qualified under ASME Code, Section IX,
Welding Qualifications. The motor tube wall between the rotor as-
sembly and the stator is constructed of magnetic material to present
a small air gap to the motor. This region of the motor tube is of
low alloy steel clad on the inside diameter with stainless steel or
with Inconel. The upper end of the motor tube functions only as a
pressurized enclosure for the withdrawn lead screw and is made of
stainless steel transition-welded to the upper end of the low alloy
steel motor section. The lower end of the low alloy steel tube sec-
tion is welded to a stainless steel machined forging which is flanged
at the face which contacts the vessel control rod nozzle. Double
gaskets, which are separated by a ported test annulus, seal the
flanged connection between the motor tube and the reactor vessel.

b. Motor

The motor is a synchronous reluctance unit with a slip-on stator. The
rotor assembly is described in Paragraph (f). The stator is a 48-slot
four-pole arrangement with water cooling coils wound on the outside of
its casing. The stator is encapsulated after winding to establish a
16. sealed unit. It is six phase star-connected for operation in a pulse-
stepping mode and advances 15 mechanical degrees per step. The stator
assembly is mounted over the motor tube housing as shown in Figure 3-55.

c. Plug and Vent Valve

The upper end of the motor tube is closed by a closure insert assembly
containing a vapor bleed port and vent valve. The vent valve and
16. insert closure have double seals. The insert closure is retained by
a closure nut which is threaded to the inside of the motor tube. The
sealing for the closure is applied by jackscrews threaded through the
closure nut.

d. Actuator

The actuator consists of the translating lead screw, its rotating nut assembly, and the torque taker assembly on the screw. The actuator lead screw travel is 139 inches.

e. Lead Screw

The lead screw has a lead of 0.750 in. The thread is double lead
with a single pitch spacing of 0.375 in. Thread lead error is held
to close tolerances for uniform loading with the roller nut assemblies.
1. & 3. The thread form is a modified ACME with a blank angle that allows the
roller nut to disengage without lifting the screw.

f. Rotor Assembly

The rotor assembly consists of a ball bearing supported rotor tube carrying and limiting the travel of a pair of scissors arms. Each of the two arms carry a pair of ball bearing supported roller (nut) assemblies which are skewed at the lead screw helix angle for engagement with the lead screw. The current in the motor stator (two of a six winding stator) causes the arms that are pivoted in the rotor tube to move radially toward the motor tube wall to the limit provided thereby engaging the four roller nuts with the centrally located lead screw. Also, four separating springs mounted in the scissor arms keep the rollers disengaged when the power is removed from the stator coils. A second radial bearing mounted to the upper end of the rotor tube has its outer race pinned to both scissor arms thereby synchronizing their motion during engagement and disengagement. When a three

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phase rotating magnetic field is applied to the motor stator, the resulting force produces rotor assembly rotation.

g. Torque Extension Tube and Torque Taker

The torque extension tube is a separate tubular assembly containing a keyway that extends the full length of the lead screw travel. The tube assembly is supported against rotation and in elevation by the upper end of the motor tube extension. The lower end of the tube assembly supports the buffer and is the down stop. A set of indexing serrations mate to prevent rotation and orient the torque extension tube with the motor tube below the cap and vent valve assembly. An integral shoulder at the top of the tube rests against a step in the motor tube inside diameter to provide a vertical support.

The torque taker assembly consists of the position indicator permanent magnet, the buffer piston, and a positioning key. The torque taker key fixed at the top of the lead screw is mated with the torque extension tube keyway to provide both radial and tangential positioning of the lead screw.

h. Buffer

The buffer assembly is capable of decelerating the translating mass from the unpressurized terminal velocity to zero velocity without applying greater than ten times the gravitational force on the control rod. The water buffer consists of a piston fixed to the top end of the screw shaft and a cylinder which is fixed to the lower end of the torque extension tube. Twelve inches above the bottom stop, the piston at the top of the screw enters the cylinder. Guiding is accomplished because the piston and torque key are in a single part, and the cylinder and keyway are in a single mating part. As the piston travels into the cylinder, water is driven into the center of the lead screw through holes in the upper section which produce the damping pressure drop. The number of holes presented to the buffer chamber is reduced as the rod moves into the core, so that the damping coefficient increases as the velocity reduces, thereby providing an approximately uniform deceleration. A large helical buffer spring is used to take the kinetic energy of the drive line at the end of the water buffer stroke. The buffer spring accepts a five-foot per second impact velocity of the drive line and control rod with an instantaneous overtravel of one inch past the normal down stop. The inclusion of this buffer spring permits practical clearances in the water buffer.

i. Lead Screw Guide

The lead screw guide bushing acts as a primary thermal barrier and as a guide for the screw shaft. As a primary thermal barrier, the bushing allows only a small path for free convection of water between the mechanism and the closure head nozzle. Fluid temperature in the mechanism is largely governed by the flow of water up and down through this bushing. The diametral clearance between screw shaft and bushing is large enough to preclude jamming the screw shaft and small enough to hold the free convection to an acceptable value. In order to obtain trip travel times of acceptably small values, it is necessary to provide an auxiliary flow path around the guide bushing. The larger area path is necessary to reduce the pressure differential required to drive water into the mechanism to equal the screw displacement. The auxiliary flow paths are closed for small pressure differentials (several inches of water) by ball check valves which prevent the convection flow but open fully during trip.

j. Position Indications

Two methods of position indication are provided; one, an absolute position indicator and the other, a relative position indicator. The absolute position transducer consists of a series of magnetically operated reed switches mounted in a tube parallel to the motor tube extension. Each switch is hermetically sealed. Switch contacts close when a permanent magnet mounted on the upper end of the lead screw extension comes in close proximity. As the lead screw (and the control rod assembly) moves, switches operate sequentially producing an analogue voltage proportional to position. The accuracy of the analogue signal is ±1.44 per cent and produces a readout of approxi-
16. mately ±2.46 per cent accuracy. Additional reed switches are included in the same tube with the absolute position transducer to provide full withdrawal and insertion signals. The relative position indicator consists of a small pulse-stepping motor driving a potentiometer that generates a signal accuracy of ±0.7% producing a position readout of ±1.7% accuracy.

k. Motor Tube Design Criteria

The motor tube design complies with Section III of the ASME Boiler and Pressure Vessel Code for a Class A vessel. The operating transient cycles, which are considered for the stress analysis of the reactor pressure vessel, are also considered in the motor tube design.

Quality standards relative to material selection, fabrication, and inspection are specified to insure safety function of the housings essential to accident prevention. Materials conform to ASTM or ASME, Section II, Material Specifications. All welding shall be performed by personnel qualified under ASME Code, Section IX, Welding Qualifications. These design and fabrication procedures establish quality assurance of the assemblies to contain the reactor coolant safely at operating temperature and pressure.

In the highly unlikely event that a pressure barrier component or the control rod drive assembly does fail catastrophically, i.e., ruptured completely, the following results would ensue:

1. Control Rod Drive Nozzle

   The assembly would be ejected upward as a missile until it was stopped by the missile shield over the reactor. This upward motion would have no adverse effect on adjacent assemblies.

2. Motor Tube

   The failure of this component anywhere above the lower flange would result in a missile-like ejection into the missile shielding over the reactor. This upward motion would have no adverse effect on adjacent mechanisms.

Axial Power Shaping Rod Drive

For actuating the partial length control rods which maintain their set position during a reactor-trip of the shim safety drive, the CRDM is modified so that the roller nut assembly will not disengage from the lead screw on a loss of power to the stator. Except for this modification, the shim drives and the axial power shaping rod drives are identical.

3.3 TESTS AND INSPECTIONS

3.3.1 NUCLEAR TESTS AND INSPECTION

3.3.1.1 Critical Experiments

An experimental program (47-49) to verify the relative reactivity worth of the CRA has been completed. Detailed testing established the worth of the CRA under various conditions similar to those for the reference core. These parameters include control rod arrangement in a CRA, fuel enrichments, fuel element geometry, CRA materials, and soluble boron concentration in the moderator.

Gross and local power peaking were also studied, and three-dimensional power-peaking data were taken as a function of CRA insertion. Detailed peaking data were also taken between fuel assemblies and around the water holes left by withdrawn CRA. The experimental data have been analyzed and were used to bench mark the analytical models used in the design.

3.3.1.2 Zero Power, Approach to Power, and Power Testing

Boron worth and CRA worth (including stuck-CRA worth) will be determined by physics tests at the beginning of each core cycle. The boron worth and CRA worth at a given time in core life will be based on CRA position indication and calculated data as adjusted by operating data.

The reactor coolant will be analyzed in the laboratory periodically to determine the boron concentration, and the reactivity held in boron will then be calculated from the concentration and the reactivity worth of boron.

The method of maintaining the hot shutdown margin (hence stuck-CRA margin) is related to operational characteristics (load patterns) and to the power-peaking restrictions on CRA patterns at power. The CRA pattern restrictions will insure that sufficient reactivity is always fully withdrawn to provide adequate shutdown with the stuck-CRA margin. Power peaking as related to CRA patterns and shutdown margin will be predicted by calculations.

Operation under power conditions will normally be monitored by in-core instrumentation, and the resulting data will be analyzed and compared with multi-dimensional calculations.

3.3.2 THERMAL AND HYDRAULIC TESTS AND INSPECTION

3.3.2.1 Reactor Vessel Flow Distribution and Pressure Drop Test

A 1/6-scale model of the reactor vessel and internals has been tested to evaluate:

a. The flow distribution to each fuel assembly of the reactor core and to develop any necessary modifications to produce the desired flow distribution.

b. Fluid mixing between the vessel inlet nozzle and the core inlet, and between the inlet and outlet of the core.

c. The overall pressure drop between the vessel inlet and outlet nozzles, and the pressure drop between various points in the reactor vessel flow circuit.

d. The internals vent valves for closing behavior and for the effect on core flow with valves in the open position.

The reactor vessel, flow baffle, and core barrel were made of clear plastic to allow use of visual flow study techniques. All parts of the model except the core are geometrically similar to those in the production reactor. However, the simulated core was designed to maintain dynamic similarity between the model and production reactor.

Each of the 177 simulated fuel assemblies contains a calibrated flow nozzle. The test loop is capable of supplying cold water (80 F) to three inlet nozzles and hot water (180 F) to the fourth. Temperature was measured in the inlet and outlet nozzles of the reactor model and at the inlet and outlet of each of the fuel assemblies. Static pressure taps were located at suitable points along the flow path through the vessel. This instrumentation provided the data necessary to accomplish the objectives set forth for the tests.

3.3.2.2 Fuel Assembly Heat Transfer and Fluid Flow Tests

Although the design of the reactor is based on the W-3 heat transfer correlation, B&W is conducting a continuous research and development program for fuel assembly heat transfer and fluid flow applicable to the design of the reactor. Single-channel tubular and annular test sections and multiple rod assemblies have been tested at the B&W Research Center. This test work substantiates the thermal design of the reactor core. Some of this test work is described briefly below.

3.3.2.2.1 Single-Channel Heat Transfer Tests

A large quantity of uniform flux, single-channel, critical heat flux data has been obtained. References to uniform flux data are given in BAW-168.[53] The effect on the critical heat flux caused by nonuniform axial power generation in a tubular test section at 2,000 psi pressure was investigated as early as 1961.[50] This program was extended to include pressures of 1,000, 1,500, and 2,000 psi and mass velocities up to 2.5 x 10^6 lb/h-ft^2.[51] The effect on the critical heat flux caused by differences in axial power distribution in an annular test section was recently investigated at reactor design conditions.[52] Data were obtained at pressures of 1,000, 1,500, 2,000, and 2,200 psi and at mass velocities up to 2.5 x 10^6 lb/h-ft^2.

The tubular tests included the following axial heat flux shapes where $P/\overline{P}$ is local to average power:

a. Uniform Heat Flux $(P/\overline{P}) = 1.000$ constant

b. Sine Heat Flux $(P/\overline{P})_{max} = 1.396$ @ 50% L

c. Inlet Peak Heat Flux $(P/\overline{P})_{max} = 1.930$ @ 25% L

d. Outlet Peak Heat Flux $(P/\overline{P})_{max} = 1.930$ @ 75% L

Tests of two additional, nonuniform, 72-in. heated length, tubular specimens were undertaken to obtain data for peaking conditions more closely related to the reference design. The additional flux shapes being tested are:

a. Inlet Peak Heat Flux $(P/\overline{P})_{max} = 1.65$ @ 28% L

b. Outlet Peak Heat Flux $(P/\overline{P})_{max} = 1.65$ @ 72% L

These tests will cover approximately the same range of pressure, mass velocity, and ΔT as the multiple-rod fuel assembly tests.

3.3.2.2.2 Multiple-Rod Fuel Assembly Heat Transfer Tests

Critical heat flux data have been obtained from 6-ft long, 9-rod fuel assemblies in a 3 x 3 square array. A total of 513 data points was obtained covering the following conditions:

$$10 \leq \Delta T_S \leq 300$$

$$1,000 \leq P \leq 2,400$$

$$0.2 \times 10^6 \leq G \leq 3.5 \times 10^6$$

where

ΔT_S = inlet subcooling, F

P = pressure, psia

G = mass velocity, $lb/h\text{-}ft^2$

The geometry of this section consisted of nine rods of 0.420-in. diameter on a 0.558-in. square pitch. Analysis of the last data of this set is in progress.

3.3.2.2.3 Fuel Assembly Flow Distribution, Mixing, and Pressure Drop Tests

Flow visualization and pressure drop data have been obtained from a 10-times-full-scale (10X) model of a single rod in a square flow channel. These data have been used to refine the spacer grid designs with respect to mixing turbulence and pressure drop. Additional pressure drop testing has been conducted using 4-rod (5X), 4-rod (1X), 1-rod (1X), and 9-rod (1X) models.

Testing to determine the extent of interchannel mixing and flow distribution has also been conducted. Flow distribution in a square 4-rod test assembly has been measured. A salt solution injection technique was used to determine the average flow rates in the simulated reactor assembly corner cells, wall cells, and unit cells. Interchannel mixing data were obtained for the same assembly. These data have been used to confirm the flow distribution and mixing relationships employed in the core thermal and hydraulic design. Flow tests on a mockup of two adjacent fuel assemblies have been conducted. Additional mixing, flow distribution, and pressure drop data will be obtained to improve future core power capability. The following fuel assembly geometries have been tested to provide additional data:

a. A 9-rod (3 x 3 array) mixing test assembly, to determine flow pressure drop, flow distribution, and degree of mixing.

b. A 64-rod assembly simulating larger regions and various mechanical arrangements within a 15 x 15 fuel assembly and between adjacent fuel assemblies to determine flow distribution in the assembly and between adjacent assemblies.

3.3.3 FUEL ASSEMBLY, CONTROL ROD ASSEMBLY, AND CONTROL ROD DRIVE MECHANICAL TESTS AND INSPECTION

To demonstrate the mechanical adequacy and safety of the fuel assembly, control rod assembly (CRA), and control rod drive, a number of functional tests have been performed, are in progress, or are in the final stages of preparation.

16. #### 3.3.3.1 <u>Prototype Testing</u> (Reference Supplement 9 Revisions for Oconee 3)

A full-scale prototype fuel assembly, CRA, and control rod drive have been tested in the Control Rod Drive Line (CRDL) Facility located at the B&W Research Center, Alliance, Ohio. This full-sized loop is capable of simulating reactor environmental conditions of pressure, temperature, and coolant flow. To verify the mechanical design, operating compatibility, and characteristics of the entire control rod drive fuel assembly system, the drive was stroked and tripped approximately 200 per cent of the expected operating life requirements.

A portion of the testing was performed with maximum misalignment conditions. Equipment was available to record and verify data such as fuel assembly pressure drop, vibration characteristics, and hydraulic forces and to demonstrate control rod drive operation and verify scram times. All prototype components were examined periodically for signs of material fretting, wear, and vibration/fatigue to insure that the mechanical design of the equipment met reactor operating requirements.
21. Test results are given in B&W Topical Report BAW-10029, "Control Rod Drive System Test."

3.3.3.2 <u>Model Testing</u>

Many functional improvements have been incorporated in the design of the fuel assembly as a result of model tests. For example, the spacer grid to fuel rod contact area was fabricated to 10 times reactor size and tested in a loop simulating the coolant flow Reynolds number of interest. Thus, visually, the shape of the fuel rod support areas was optimized with respect to minimizing the severity of flow vortices and pressure drop. A 9-rod (3 x 3) assembly using stainless steel spacer grid material has been tested at reactor conditions (640 F, 2,200 psi, 13 fps coolant flow) for 210 days. Two full-sized canned fuel assemblies with stainless steel spacer grids have been tested at reactor conditions, one for 40 days and the other for 22 days. A prototype canless fuel assembly using Inconel 718 spacer grids has been tested for approximately 90 days, approximately half of that time at reactor conditions. The principal objectives of these tests were to evaluate fuel assembly and fuel rod vibration and/or fretting wear resulting from flow-induced vibration. Vibratory amplitudes have been found to be very small, and, with the exception of a few isolated instances which are attributed to pretest spacer grid damage, no unacceptable wear has been observed.

3.3.3.3 Component and/or Material Testing

3.3.3.3.1 Fuel Rod Cladding

Extensive short-time collapse testing was performed on Zircaloy-4 tube specimens as part of the B&W overall creep-collapse testing program. Initial test specimens were 0.436 in. OD with wall thicknesses of 0.020 in., 0.024 in., and 0.028 in. Ten 8-in.-long specimens of each thickness were individually tested at 680 F at slowly increasing pressure until collapse occurred. Collapse pressures for the 0.020-in. wall thickness specimens ranged from 1,800 to 2,200 psig, the 0.024-in. specimens ranged from 2,800 to 3,200 psig, and the 0.028-in. specimens ranged from 4,500 to 4,900 psig. The material yield strength of these specimens ranged from 65,000 to 72,000 psi at room temperature and was 35,900 psi at 680 F.

Additional Zircaloy-4 short-time collapse specimens were prepared from material with a yield strength of 78,000 psi at room temperature and 48,500 psi at 615 F. Fifteen specimens having an OD of 0.410 in. and an ID of 0.365 in. (0.0225-in. nominal wall thickness) were tested at 615 F at increasing pressure until collapse occurred. Collapse pressures ranged from 4,470 to 4,960 psig.

Creep-collapse testing was performed on the 0.436-in. OD specimens. Twelve specimens of 0.024-in. wall thickness and 30 specimens of 0.028-in. wall thickness were tested in a single autoclave at 680 F and 2,050 psig. During this test, two 0.024-in. wall specimens collapsed during the first 30 days and two collapsed between 30 and 60 days. None of the 0.028-in. wall specimens had collapsed after 60 days. Creep-collapse testing was then performed on thirty 0.410-in. OD by 0.365-in. ID (0.0225-in. nominal wall) specimens for 60 days at 615 F and 2,140 psig. None of these specimens collapsed, and there were no significant increases in ovality after 60 days.

The results of the 60-day creep-collapse testing on the 0.410-in. OD specimens showed no indication of incipient collapse. The 60-day period for creep-collapse testing was used since it exceeds the point of primary creep of the material, yet is sufficiently long to enter the stage when fuel rod pressure begins to build up during reactor operation, i.e., past the point of maximum differential pressure that the cladding would be subjected to in the reactor.

These tests were followed by additional creep-collapse tests in which 60 specimens of variable wall thickness were subjected to a pressure of 2,085 psi at 685 F until collapse occurred. The cladding wall thickness was 0.0285, 0.0263, 0.0251, and 0.0240 inch. The cladding thickness included the range of tolerances for production cladding, and the pressure represented the fuel rod maximum pressure differential at operating conditions. The temperature was selected to conservatively approximate in-pile creep rates. It was found that the 0.024-in. wall specimens collapsed in less than a month, and several 0.0263-in. wall specimens collapsed in less than 3 months. In view of the unknown increase of in-pile creep rates as compared with out-of-pile creep rates, it was decided to provide backup support for the cladding in the upper end void region where cladding temperatures of 650 F occur in hot channels.

A thin-walled stainless steel tube was selected as a backup spacer. This spacer has the desirable property of providing radial support without causing large restraint for the axial expansion of the fuel. Development tests have been performed to select a spacer that can withstand shipping acceleration of fuel pellets and provide the required backup support for the cladding. The tubes were encapsulated in 0.016-in. wall Zircaloy tubing and subjected to a pressure of 2,750 psi at 850 F. This represents a 10 per cent margin on system design pressure at the operating temperature for the spacer tube. On sectioning the specimens, there was no measurable deformation of the tube.

3.3.3.3.2 Fuel Assembly Structural Components

The structural characteristics of the fuel assemblies which are pertinent to loadings resulting from normal operation, handling, earthquake, and accident conditions were investigated experimentally in test facilities such as the CRDL Facility. Structural characteristics such as natural frequency and damping were determined at the relatively high (up to approximately 0.300 in.) amplitude of interest in the seismic and LOCA analyses. Natural frequencies and amplitudes resulting from flow-induced vibration were measured at various temperatures and flow velocities, up to reactor operating conditions.

3.3.3.3.3 High-Burnup Fuel Irradiations

The primary purpose of the B&W High-Burnup Irradiation Program is to determine the swelling rate of UO_2 as a function of burnup using fuel rods of the same design as the core. In addition to determining the swelling rate, the effect of several other variables, including density, heat rate, and cladding restraint, are being investigated.

The program consists of capsules, some of which will operate at a heat rate of 18 kW/ft and others at a heat rate of 21.5 kW/ft. The pellets, other than ^{235}U content, conform to the reactor fuel specifications. The target burnup ranges from 10,000 to 80,000 MWd/MTU with eight capsules exceeding 45,000 MWd/MTU. The specimens are not tested with an external pressure. However, two different cladding thicknesses, 0.015 and 0.025 in., are used to vary the restraint offered by the cladding. The fuel rods are irradiated with a cladding surface temperature of 650 F. The diametral gaps between the pellets and cladding vary from 4-5 to 7-8 mils, to give smeared densities of about 92.3 and 90.8 per cent, respectively. These gaps and smeared densities are consistent with the fuel rod specifications. The insertion date for the first capsule was September 5, 1967. See Tables 3-19 and 3-20 herein.

The tests are oriented toward the determination of the behavior of materials in an irradiation environment and to determine the optimum geometric and material properties for the specific application. The information is essential for design of advanced cores, but is not considered critical in the sense that the burnup program must be completed to insure safe operation. Adequate information is available in the literature, as discussed in 3.2.4.2.1.

3.3.3.4 Control Rod Drive Tests and Inspection

16. 3.3.3.4.1 Control Rod Drive Developmental Tests (Reference Supplement 9 Revisions for Oconee 3)

The testing and development program for the roller nut drive has been completed. The prototype drive was tested at the B&W Research Center at Alliance, Ohio. Wear characteristics of critical components have indicated that material compatibility and structural design of these components would be adequate for the design life of the mechanism. The trip time for the mechanism as determined under test conditions of reactor temperature, pressure, and flow was well within the specification requirements. B&W Topical report BAW-10029 summarizes the
21. 8. results of the test program.

3.3.3.4.2 Production Tests

Production tests discussed in this section will be performed either on the drives installed or on drives manufactured to the same specifications. The finished control rod drive will be proof-tested as a complete system, i.e., mechanisms, motor control, and system control working as a system. This proof-testing will be above and beyond any developmental testing performed in the product development stages.

Mechanism production tests will include the following:

a. Ambient Tests

Coupling tests.
Operating speeds.
Position indication.
Trip tests.

b. Operational Tests

Operating speeds.
Position indication.
Partial- and full-stroke cycles.
Partial- and full-stroke trip cycles.

3.3.4 INTERNALS TESTS AND INSPECTION

The internals upper and lower plenum hydraulic design is evaluated and guided by the results from the 1/6 scale model flow test which is described in 3.3.2.1. These test results have guided the design to obtain minimum flow maldistribution and test data allowed verification of vessel flow and pressure drops.

The effects of internals misalignment was evaluated on the basis of the test results from the CRDL tests described in 3.3.3.4. These test results, correlated with the internals guide tube design, insure that the CRA can be inserted at specified rates under conditions of maximum misalignment.

Internals shop fabrication quality control tests, inspection, procedures, and methods are similar to those for the pressure vessel described in 4.3.11.2. The internals surveillance specimen holder tubes and the material irradiation program are described in 4.4.6.

A listing is included herewith for all internals nondestructive examinations and inspections with applicable codes or standards applicable to all core structural support material of various forms. In addition, one or more of these examinations is performed on materials or processes which are used for functions other than structural support (i.e. alignment dowels, etc.) so that virtually 100 percent of the completed internals materials and parts are included in the listing. Internals raw materials are purchased to ASME Code Section II or ASTM material specifications. Certified material test reports are obtained and retained to substantiate the material chemical and physical properties. All internals materials are purchased and obtained to a low cobalt limitation. The ASME Code Section III, as applicable for Class A vessels, is generally specified as the requirement for reference level non-destructive
4. examination and acceptance. In isolated instances when ASME III cannot be applied, the appropriate ASTM Specifications for non-destructive testing are imposed. All welders performing weld operations on internals are qualified in accordance with ASME Code Section IX applicable Edition and Addenda. The primary purpose of the following list of non-destructive tests is to locate, define, and determine the size of material defects to allow an evaluation of defect, acceptance, rejection, or repair. Repaired defects are similarly inspected as required by applicable codes.

a. Ultrasonic Examination

   (1) Wrought or forged raw material forms are 100 percent inspected throughout the entire material volume to ASME III, Class A.

   (2) Personnel conducting these examinations are trained and qualified.

b. Radiographic Examination (includes X-ray or radioactive sources)

   (1) Cast raw material forms are 100 percent inspected to ASME III Class A or ASTM.

   (2) All circumferential full penetration structural weld joints which support the core are 100 percent inspected to ASME III Class A.

   (3) All radiographs are reviewed by qualified personnel who are trained in their interpretation.

c. <u>Liquid Penetrant Examination</u>

(1) Cast form raw material surfaces are 100 percent inspected to ASME III Class A or ASTM.

(2) Full penetration non-radiographic or partial penetration structural welds are inspected by examination of root, and cover passes to ASME III Class A.

(3) All circumferential full penetration structural weld joints which support the core have cover passes inspected to ASME III Class A.

(4) Personnel conducting these examinations are trained and qualified.

d. <u>Visual (5X Magnification) Examination</u>

This examination is performed in accordance with and results accepted
on the basis of a B&W Quality Control Specification which complies
with NAV-SHIPS 250-1500-1. Each entire weld pass and adjacent base
metal are inspected prior to the next pass from the root to and in-
4. cluding the cover passes.

(1) Partial penetration non-radiographically or non-ultrasonically feasible structural weld joints are 100 percent inspected to the above specification.

(2) Partial or full penetration attachment weld joints for non-structural materials or parts are 100 percent inspected to the above specification.

(3) Partial or full penetration weld joints for attachment of mechanical devices which lock and retain structural fasteners.

(4) Personnel conducting these examinations are trained and qualified.

After completion of shop fabrication, the internals components are shopfitted and assembled to final design requirements. The assembled internals components undergo a final shop fitting and alignment of the internals with the "as built" dimensions of the reactor vessel. Dummy fuel and CRA's are used to insure that ample clearances exist between the fuel and internals structures guide tubes to allow free movement of the CRA throughout its full stroke length in various core locations. Fuel assembly mating fit is checked at all core locations. The dummy fuel and CRA's are identical to the production components except that they are manufactured to the most adverse tolerance space envelope, and they contain no fissionable or absorber materials.

All internal components can be removed from the reactor vessel to allow inspection of all vessel interior surfaces. Internals components surfaces can be inspected when the internals are removed to the canal underwater storage location.

The internals vent valves were designed to relieve the pressure generated by steaming in the core following a LOCA so that the core will remain sufficiently cooled. The valves were designed to withstand the forces resulting from rupture of either a reactor coolant inlet or outlet pipe. To verify the structural adequacy of the valves to withstand the pressure forces and perform the venting function, the following tests were performed:

a. A full-size prototype valve assembly (valve disc retaining mechanism and valve body) was hydrostatically tested to the maximum pressure expected to result during the blowdown.

b. Sufficient tests were conducted at zero pressure to determine the frictional loads in the hinge assembly, the inertia of the valve disc, and the disc rebound resulting from impact of the disc on the seat so that the valve response to cyclic blowdown forces may be determined analytically.

c. A prototype valve assembly was pressurized to determine the pressure differential required to cause the valve disc to begin to open. A determination of the pressure differential required to open the valve disc to its maximum open position was simulated by mechanical means.

d. A prototype valve assembly was successfully installed and removed remotely in a test stand to confirm the adequacy of the vent valve handling tool.

e. A 1/6 scale model valve disc closing force (excluding gravity) test as described in 3.3.2.1

f. The full-size prototype valve's response to vibration was determined experimentally to verify prior analytical results which indicated that the valve disc would not move relative to the body seal face as a result of vibration caused by transmission of core support shield vibrations. The prototype valve was mounted in a test fixture which duplicated the method of valve mounting in the core support shield. The test fixture with valve installed was attached to a vibration test machine and excited sinusoidally through a range of frequencies which encompassed those which may reasonably be anticipated for the core support shield during reactor operation. The relative motion between the valve disc and seat was monitored and recorded during test. The test results indicated that there was no relative motion of the valve to its seat for conditions simulating operating conditions. After no relative motion was observed or recorded during test, the valve disc was manually forced open during test to observe its response. The disc closed with impact on its seat, rebounded open and reseated without any adverse affects to valve seal surfaces, characteristics or performance. From this oscillograph record, the natural frequency of the valve disc was conservatively calculated as approximately 1500 cps; whereas, the range of frequencies for the Oconee system (including internals components) has been established as 15 to 160 cps. These frequencies are separated by an ample margin to conclude that no relative motion between the valve disc and its seat will occur during
21. normal reactor operation.

Each production valve will be subjected to tests (b) and (c) above except that no additional analysis will be performed in conjunction with test (b).

The valve disc, hinge shaft, shaft journals (bushings), disc journal receptacles, and valve body journal receptacles have been designed to withstand without failure the internal and external differential pressure loadings resulting from a loss-of-coolant accident. These valve materials will be nondestructively tested and accepted in accordance with the ASME Code III requirements for Class A vessels as a reference quality level.

During scheduled refueling outages after the reactor vessel head and the internals plenum assembly have been removed, the vent valves will be accessible for visual and mechanical inspection. A hook tool will be provided to engage with the valve disc exercise lug described in 3.2.4.1.2h. With the aid of this tool, the valve disc will be manually exercised to evaluate the disc freedom. The hinge design will incorporate special features, as described in 3.2.4.1.2h, to minimize the possibility of valve disc motion impairment during its service life. With the aid of the hook tool, the valve disc can be raised and a remote visual inspection of the valve body and disc sealing faces can be performed for evaluation of observed surface irregularities.

Remote installation and removal of the vent valve assemblies if required is performed with the aid of the vent valve handling tool which includes unlocking and operating features for the retaining ring jackscrews.

An inspection of hinge parts is not planned until such time as a valve assembly is removed because its free-disc motion has been impaired. In the unlikely event that a hinge part should fail during normal operation, the most significant indication of such a failure would be a change in the free-disc motion as a result of altered rotational clearances.

The capability for remote inspection and removal of the vent valves will be demonstrated following a written procedure "Reactor Internals Vent Valve Inspection Test." This test will demonstrate the proper use of the handling tool, the ability to remove and replace the vent valve remotely, and the use of the exercise tool. Accessibility and visibility will be evaluated during this test. This test will be performed after internals are installed and prior to the installation of the plenum assembly and prior to hot functional testing.

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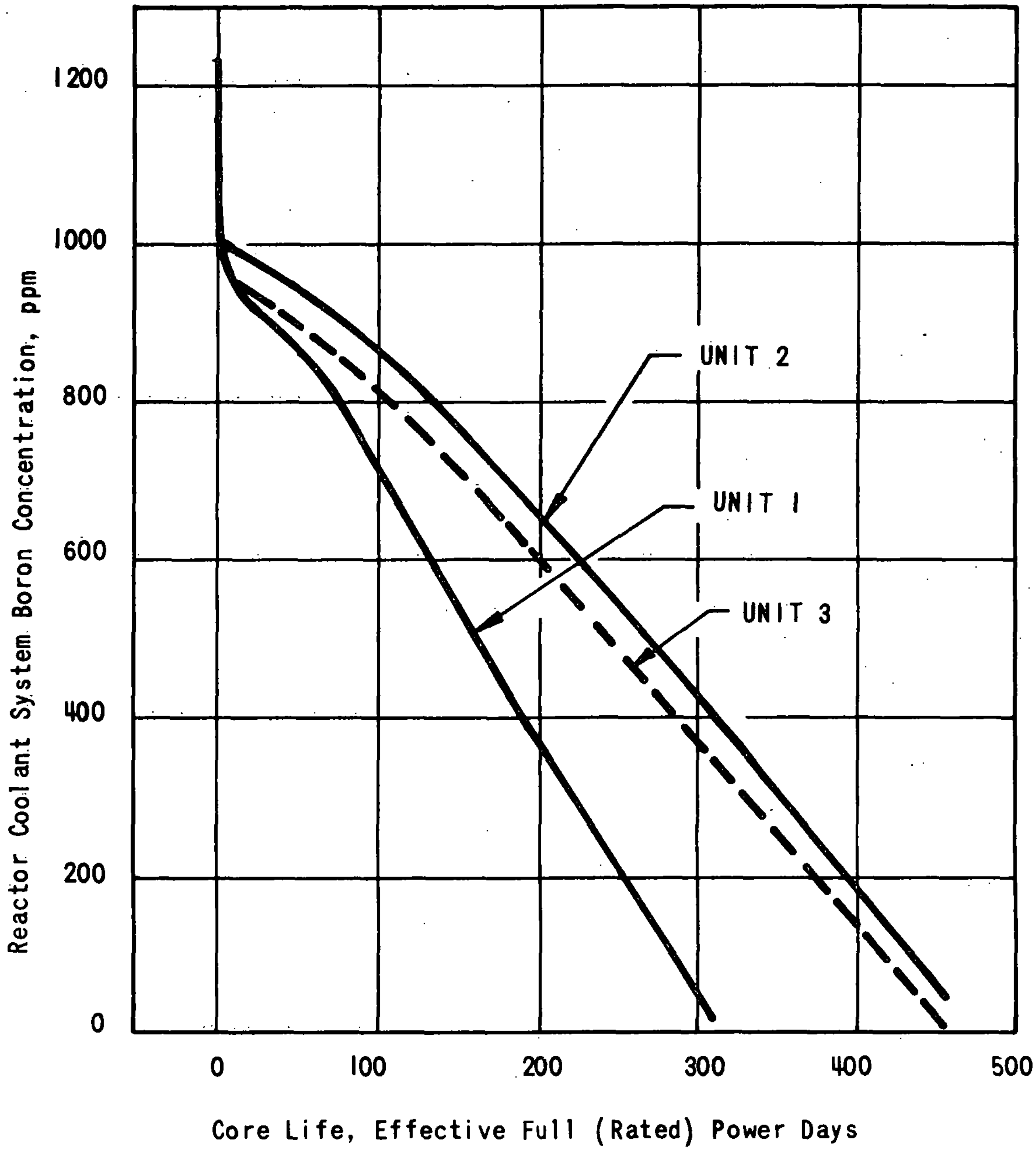
21.	BAW-10014	Analysis of Sustained Departure from Nucleate Boiling Operation
	BAW-10029	Control Rod Drive System Test (Nonproprietary version of BAW-10007, Rev. 1 and BAW-10007, Sup. 1)
	BAW-10035	Fuel Assembly Stress and Deflection Due to Loss-of-Coolant Accident and Seismic Excitation (Nonproprietary Version of BAW-10008, Part 2, Rev. 1)

23. BAW-10051 — Design of Reactor Internals and Incore Instrument Nozzles for Flow Induced Vibrations.

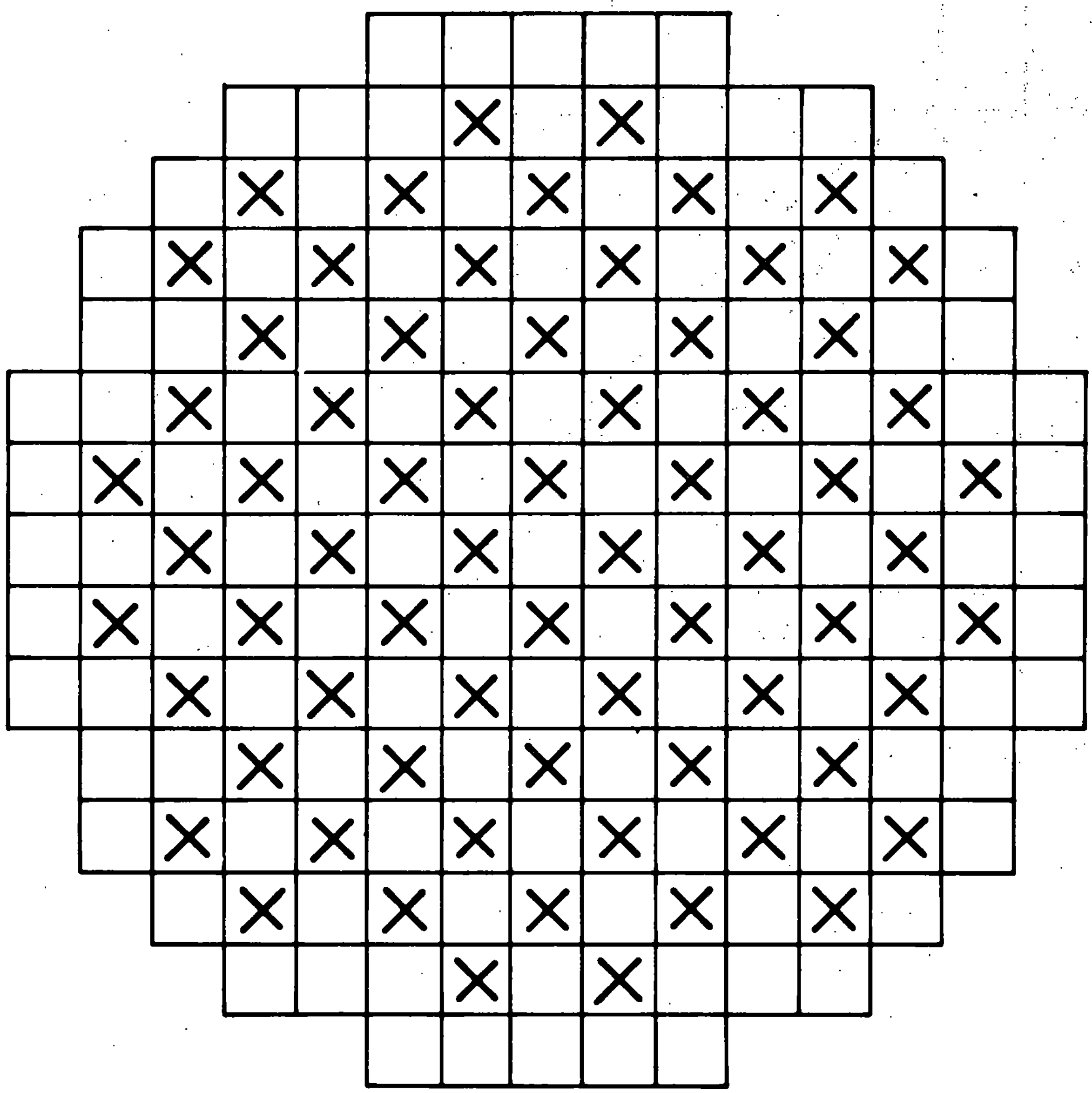


BORON CONCENTRATION VERSUS CORE LIFE

OCONEE NUCLEAR STATION

Figure 3 - 1

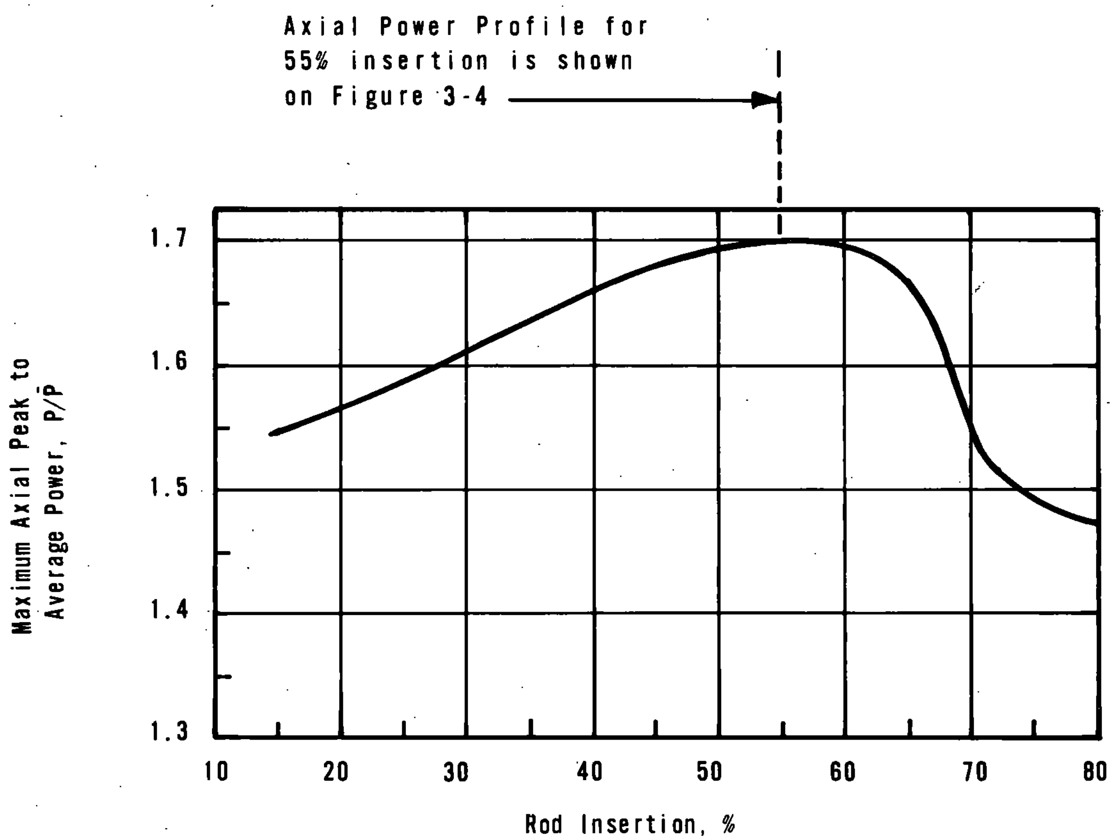
Rev. 22. 8/25/72



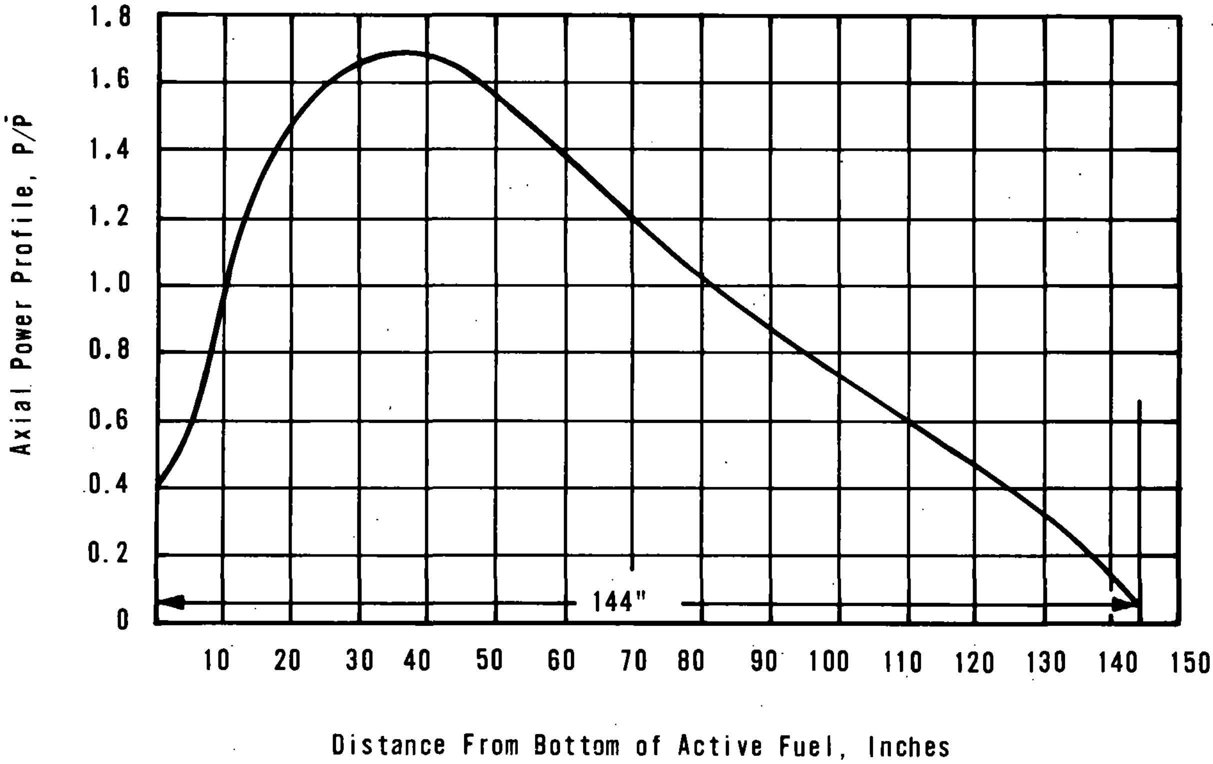
LOCATION OF FUEL ASSEMBLIES
CONTAINING BURNABLE POISON RODS

OCONEE NUCLEAR STATION

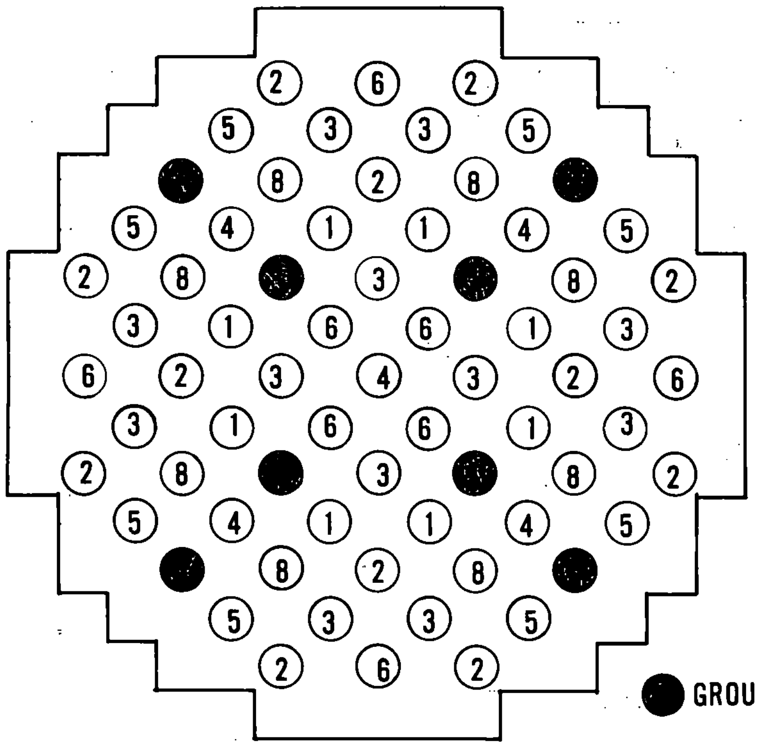
Figure 3 - 2



AXIAL PEAK TO AVERAGE POWER
VERSUS XENON OVERRIDE ROD INSERTION



AXIAL POWER PROFILE XENON OVERRIDE
RODS 55 PERCENT INSERTED



GROUP	NO. OF RODS	WORTH (%ΔP)	PURPOSE
1	8	3.3	SAFETY
2	12	2.3	"
3	12	3.3	"
4	5	.7	"
5	8	.7	DOPPLER
6	8	1.3	"
7	8	1.2	TRANSIENT
8	8	.6	APSR GROUP
TOTAL WORTH		13.4	

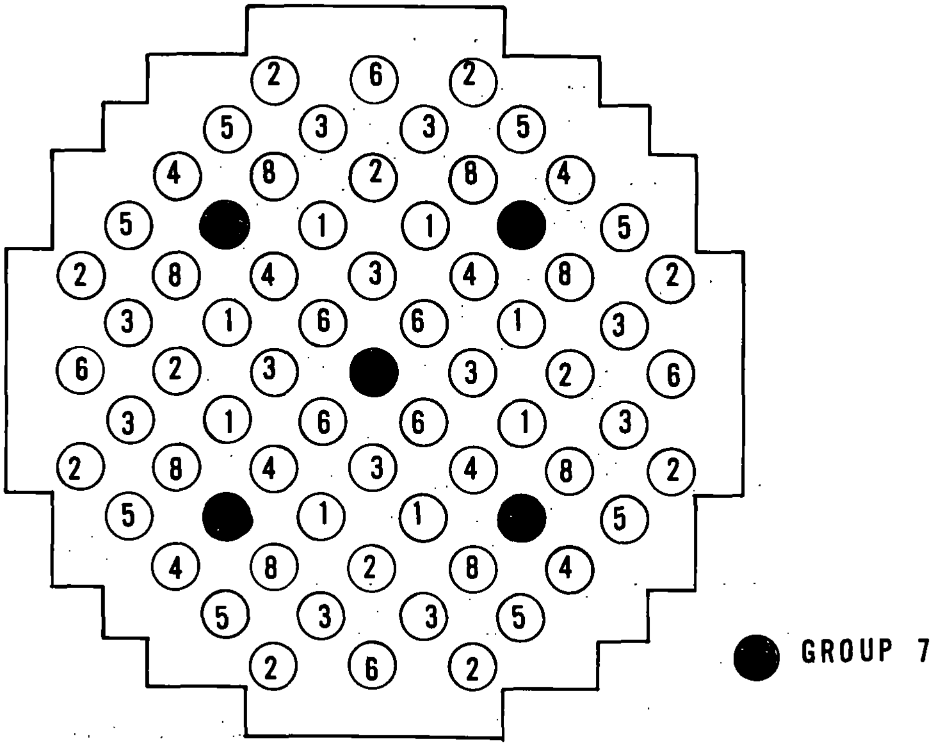
ROD GROUPS AND WORTH OF GROUPS WITHDRAWN IN ORDER (OCONEE UNIT 1, CYCLE 1, BOL)

OCONEE NUCLEAR STATION

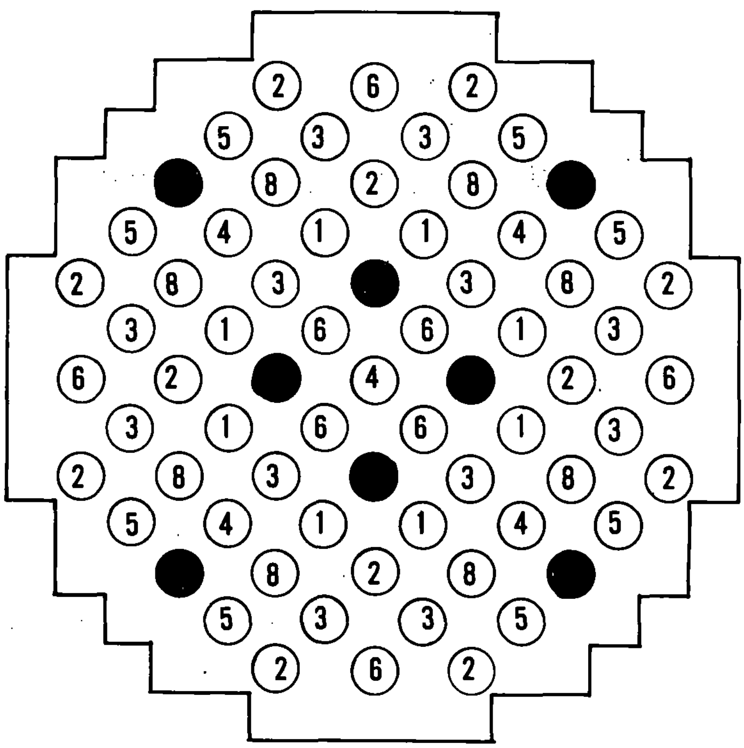
Figure 3 - 4A

(New) Rev. 4 4/20/70

CONTROL ROD GROUPING, 100 TO 200 DAYS



CONTROL ROD GROUPING, 200 TO 285 DAYS

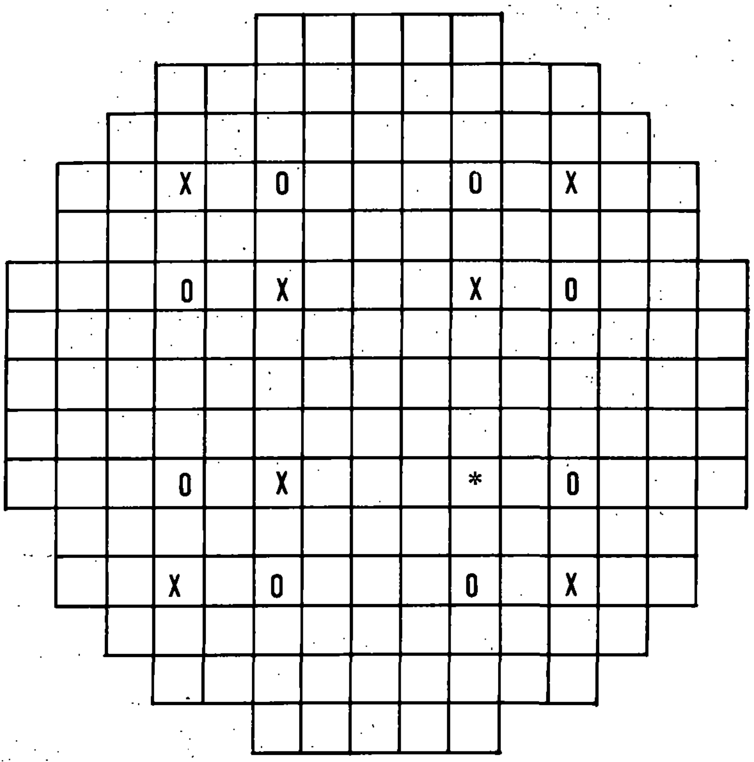


CONTROL ROD GROUPINGS (OCONEE UNIT 1, CYCLE 1)

OCONEE NUCLEAR STATION

Figure 3 - 4B

(New) Rev. 4 4/20/70



EJECTED ROD WORTH

RODS IN	WORTH OF EJECTED ROD (%ΔP)
TRANSIENT BANK	.3115
TRANSIENT BANK & APSR'S	.3508

X - ROD POSITION

0 - APSR POSITION

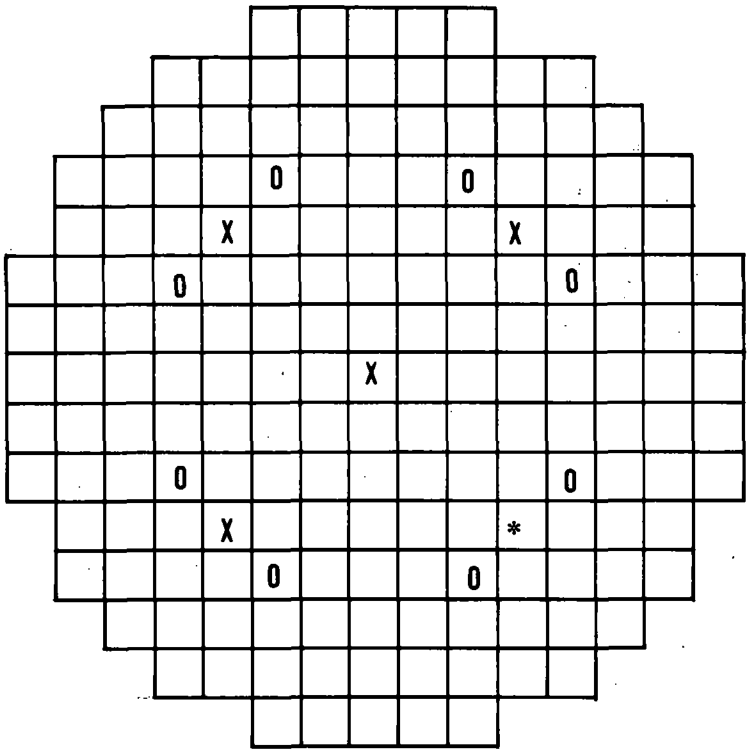
* - EJECTED ROD

0-100 DAY TRANSIENT BANK AT BOL HOT FULL POWER (OCONEE UNIT 1, CYCLE 1)

OCONEE NUCLEAR STATION

Figure 3 - 4C

(New) Rev. 4 4/20/70



EJECTED ROD WORTH

RODS IN	WORTH OF EJECTED ROD (%ΔP)
TRANSIENT BANK	.1993
TRANSIENT BANK & APSR'S	.3255

X - ROD POSITION
0 - APSR POSITION
* - EJECTED ROD

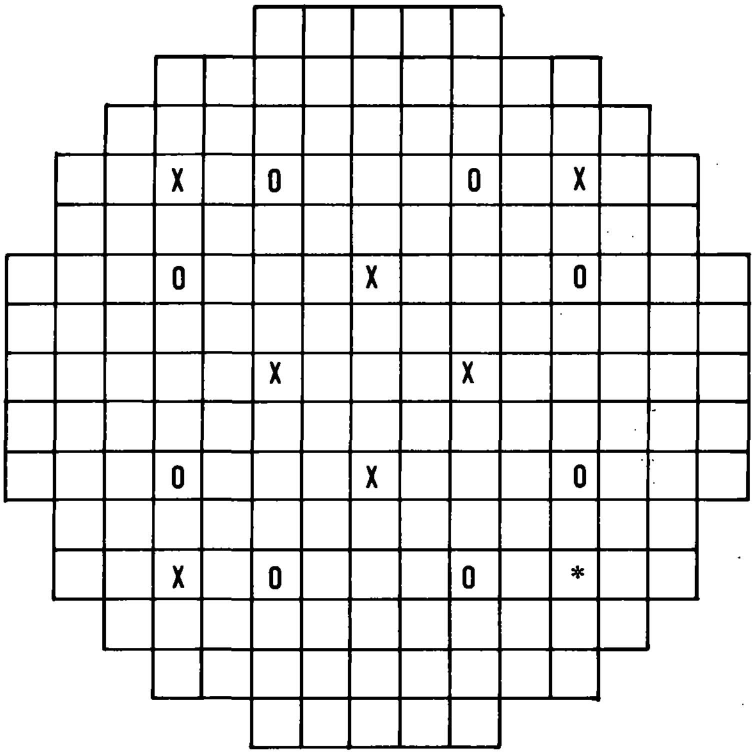
100-200 DAY TRANSIENT BANK
AT 100 DAYS

100 - 200 DAY TRANSIENT BANK AT
100 DAYS (OCONEE UNIT 2, CYCLE 1)

OCONEE NUCLEAR STATION

Figure 3 - 4D

(New) Rev. 4 4/20/70



EJECTED ROD WORTH

RODS IN	WORTH OF EJECTED ROD (%ΔP)
TRANSIENT BANK	.2966
TRANSIENT BANK & APSR'S	.3701

X - FULL LENGTH RODS
0 - APSR'S
* - EJECTED ROD

200-285 DAY TRANSIENT BANK AT 200 DAYS

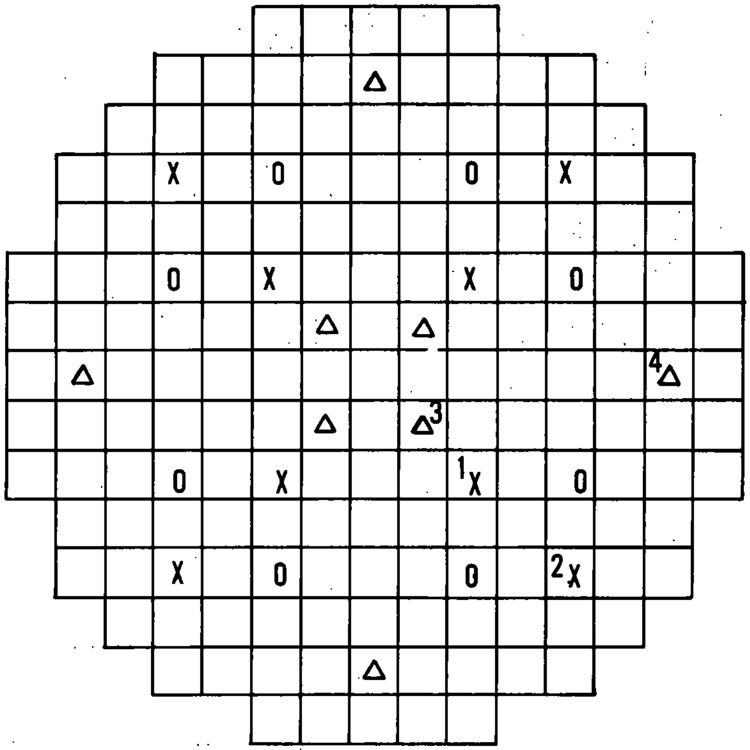
200-285 DAY TRANSIENT BANK AT 200 DAYS
(OCONEE UNIT 1, CYCLE 1)

CITIZENSHIP DUKE POWER SERVICE

OCONEE NUCLEAR STATION

Figure 3 - 4E

(New) Rev. 4 4/20/70



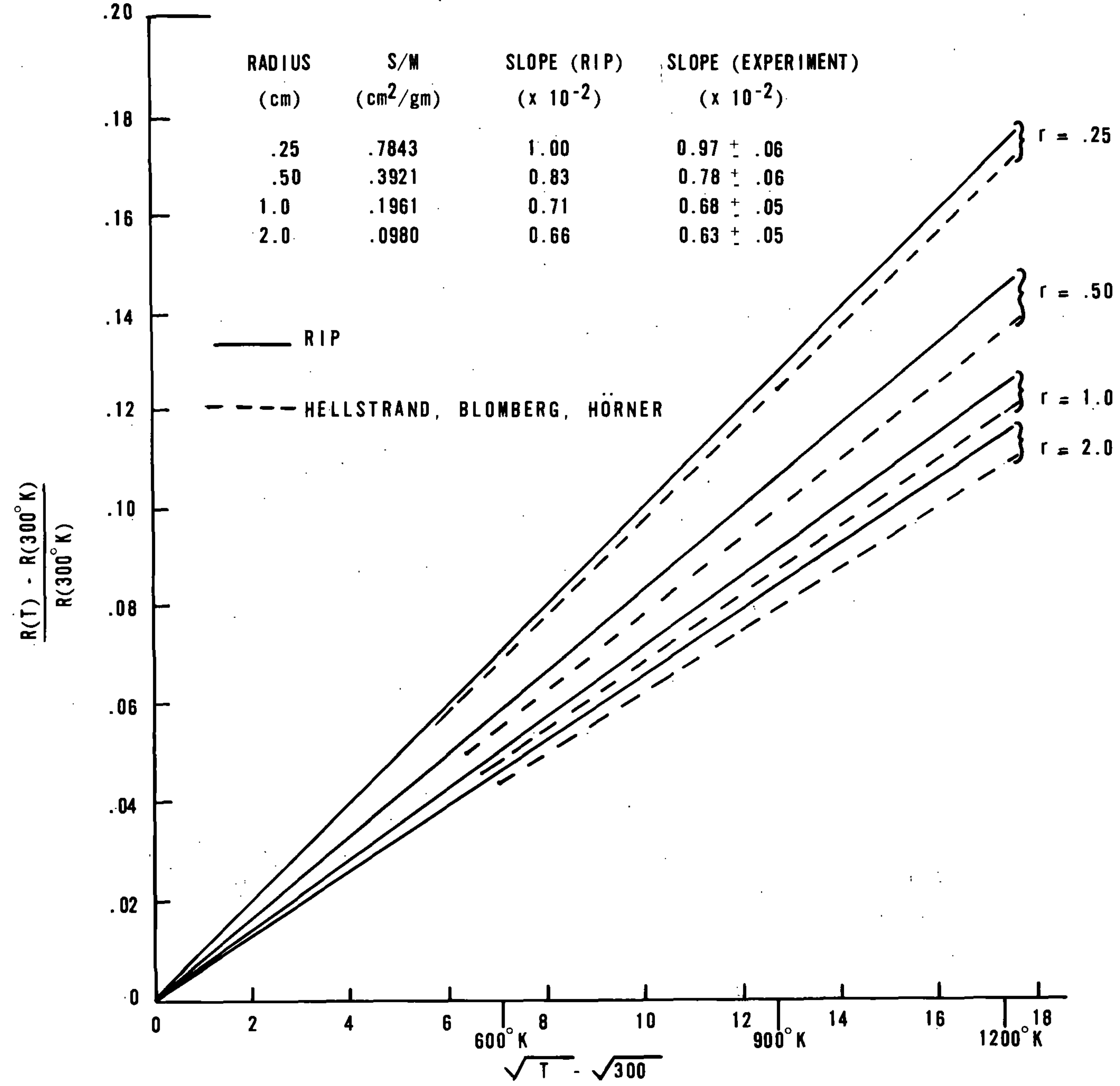
ROD EJECTED	EJECTED ROD WORTH	RODS IN
TRANSIENT ROD (1)	.306%	X TRANSIENT CRA
TRANSIENT ROD (2)	.265%	△ DOPPLER CRA
DOPPLER ROD (3)	.149%	0 APSR
DOPPLER ROD (4)	.324%	

EJECTED ROD WORTH BOL, HOT, ZERO POWER
(OCONEE UNIT 1, CYCLE 1)

OCONEE NUCLEAR STATION

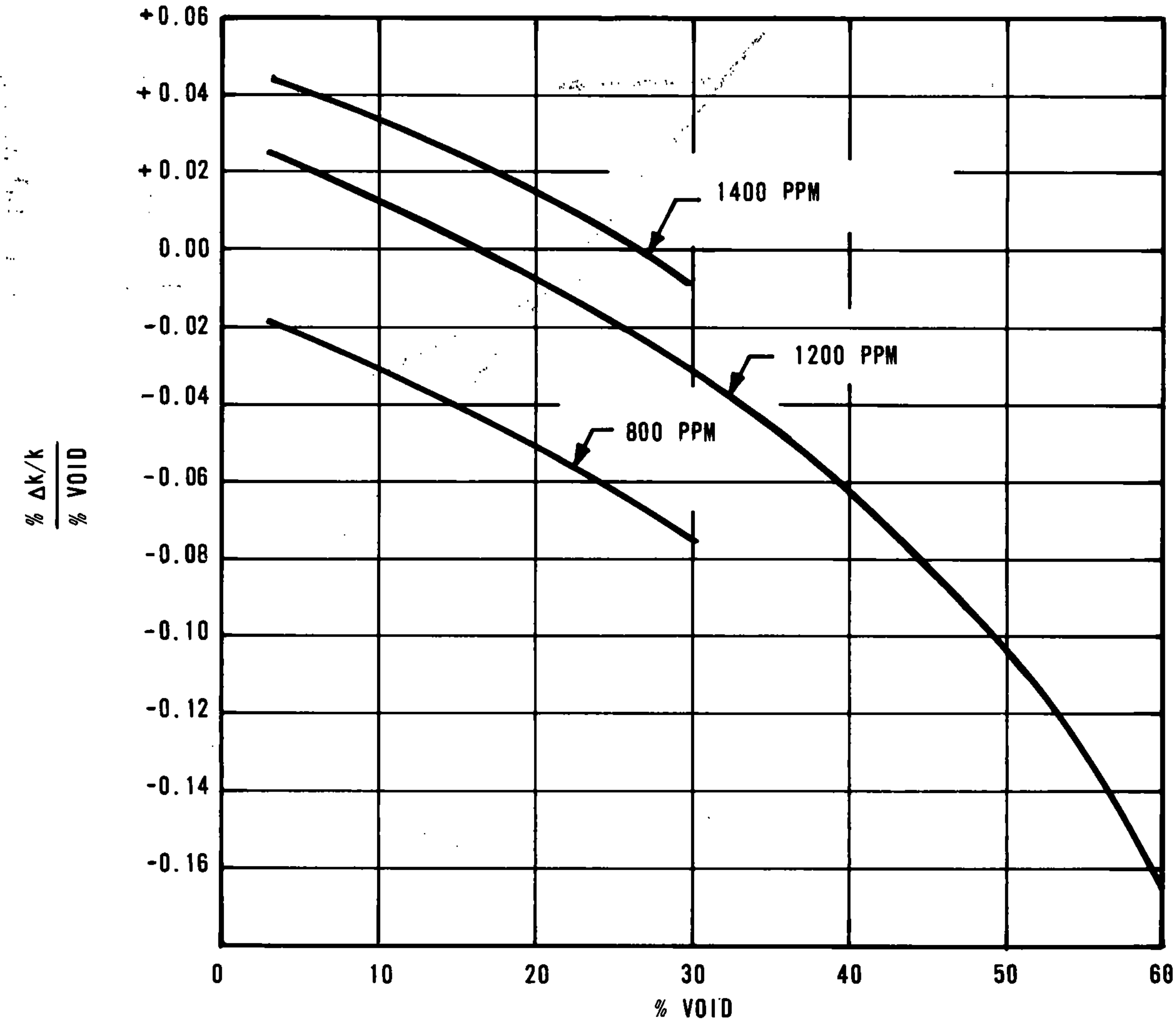
Figure 3 - 4F

(New) Rev. 4 4/20/70



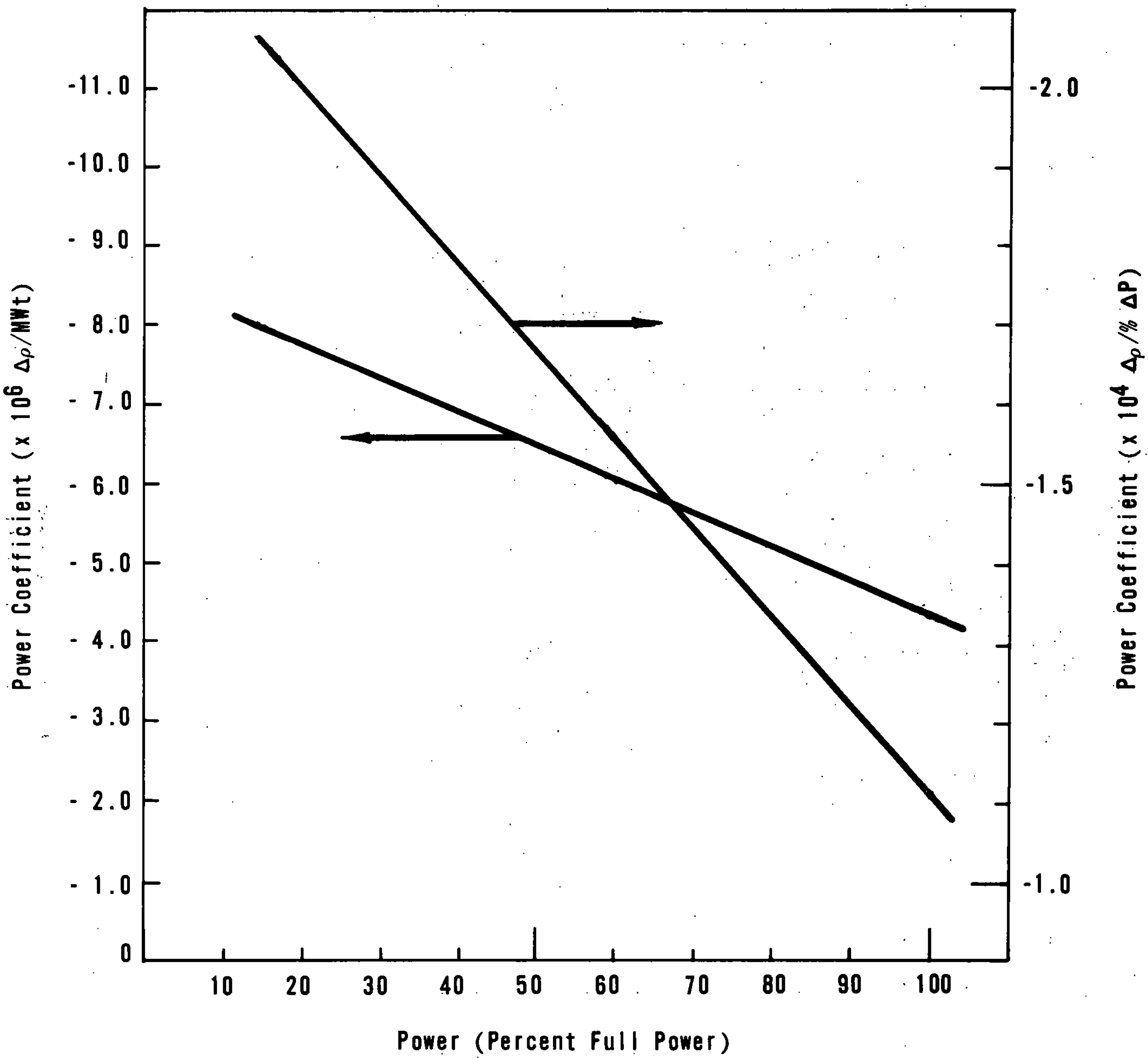
RADIUS (cm)	S/M (cm^2/gm)	SLOPE (RIP) ($\times 10^{-2}$)	SLOPE (EXPERIMENT) ($\times 10^{-2}$)
.25	.7843	1.00	0.97 ± .06
.50	.3921	0.83	0.78 ± .06
1.0	.1961	0.71	0.68 ± .05
2.0	.0980	0.66	0.63 ± .05

FRACTIONAL CHANGE IN THE RESONANCE INTEGRAL AS A FUNCTION OF $\sqrt{T} - \sqrt{300}$ FOR UO_2 RODS (T IN DEGREES K)



UNIFORM VOID COEFFICIENT FOR 177
ASSEMBLY CORE

OCONEE NUCLEAR STATION
Figure 3 - 5

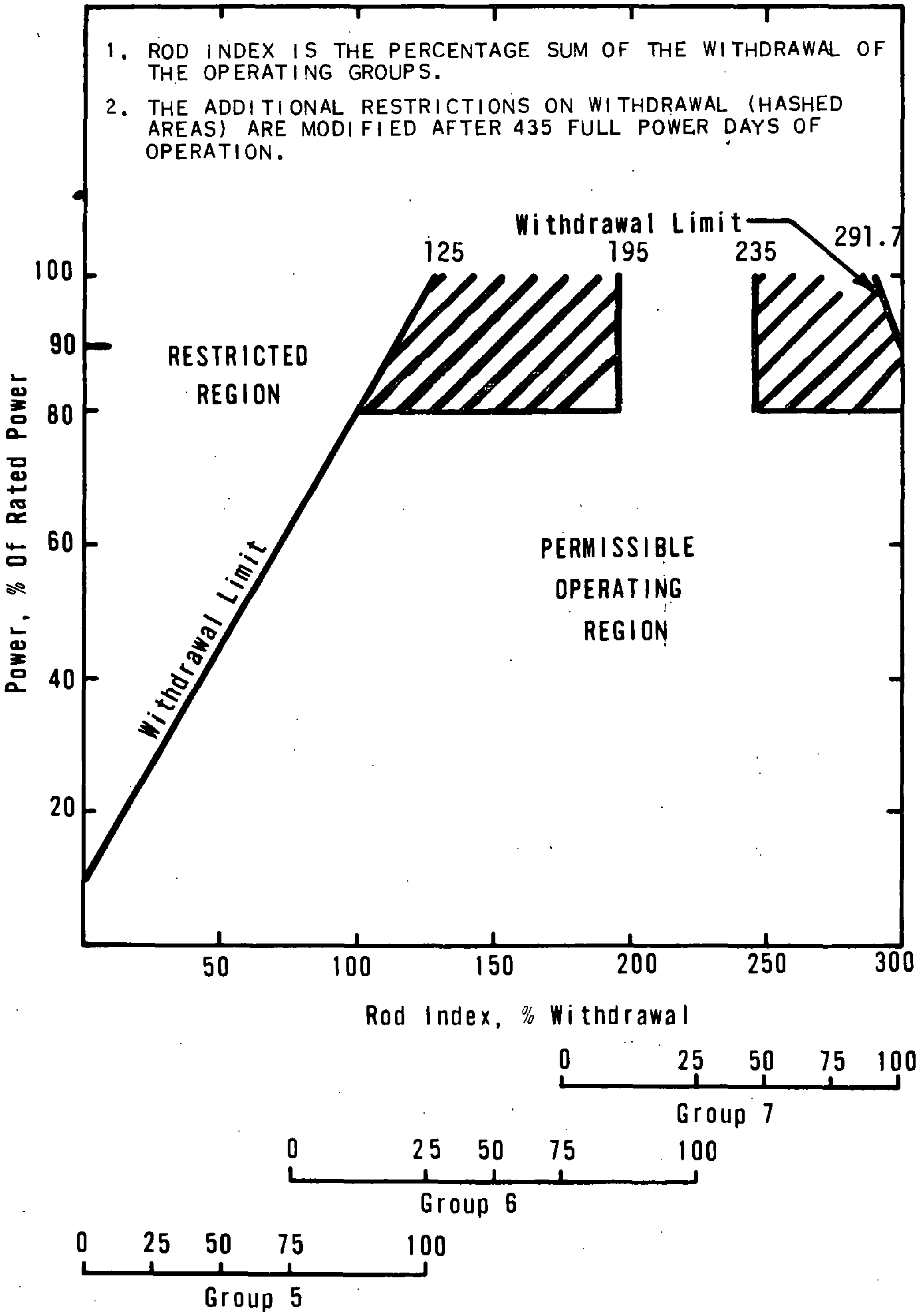


POWER COEFFICIENTS VS. POWER LEVEL

OCONEE NUCLEAR STATION

Figure 3 - 5A

(New) Rev. 4 4/20/70

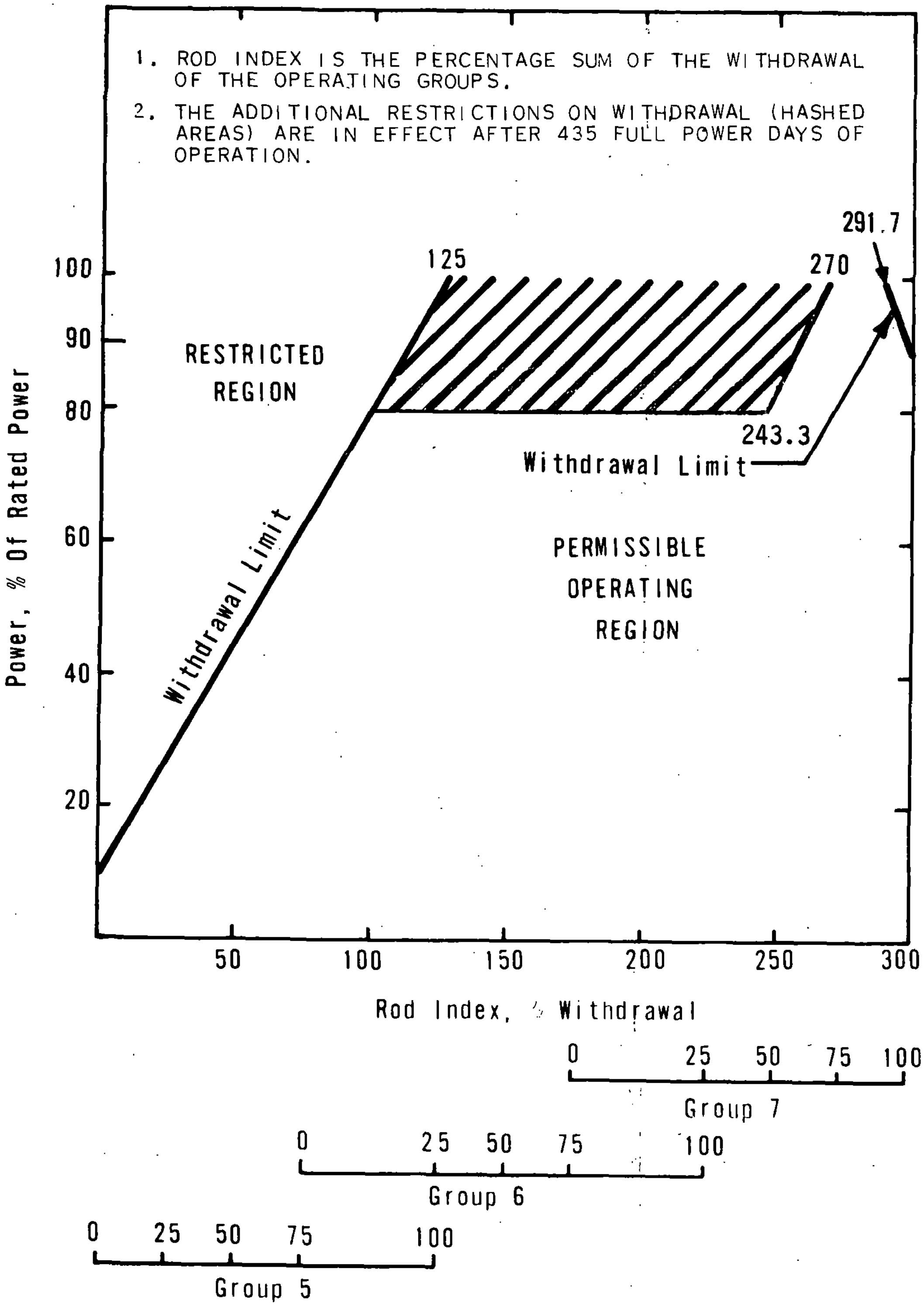


CONTROL ROD GROUP WITHDRAWAL LIMITS FOR 4 PUMP OPERATION

OCONEE NUCLEAR STATION

Figure 3.5.2-1-1

Rev. 30. 9/4/73

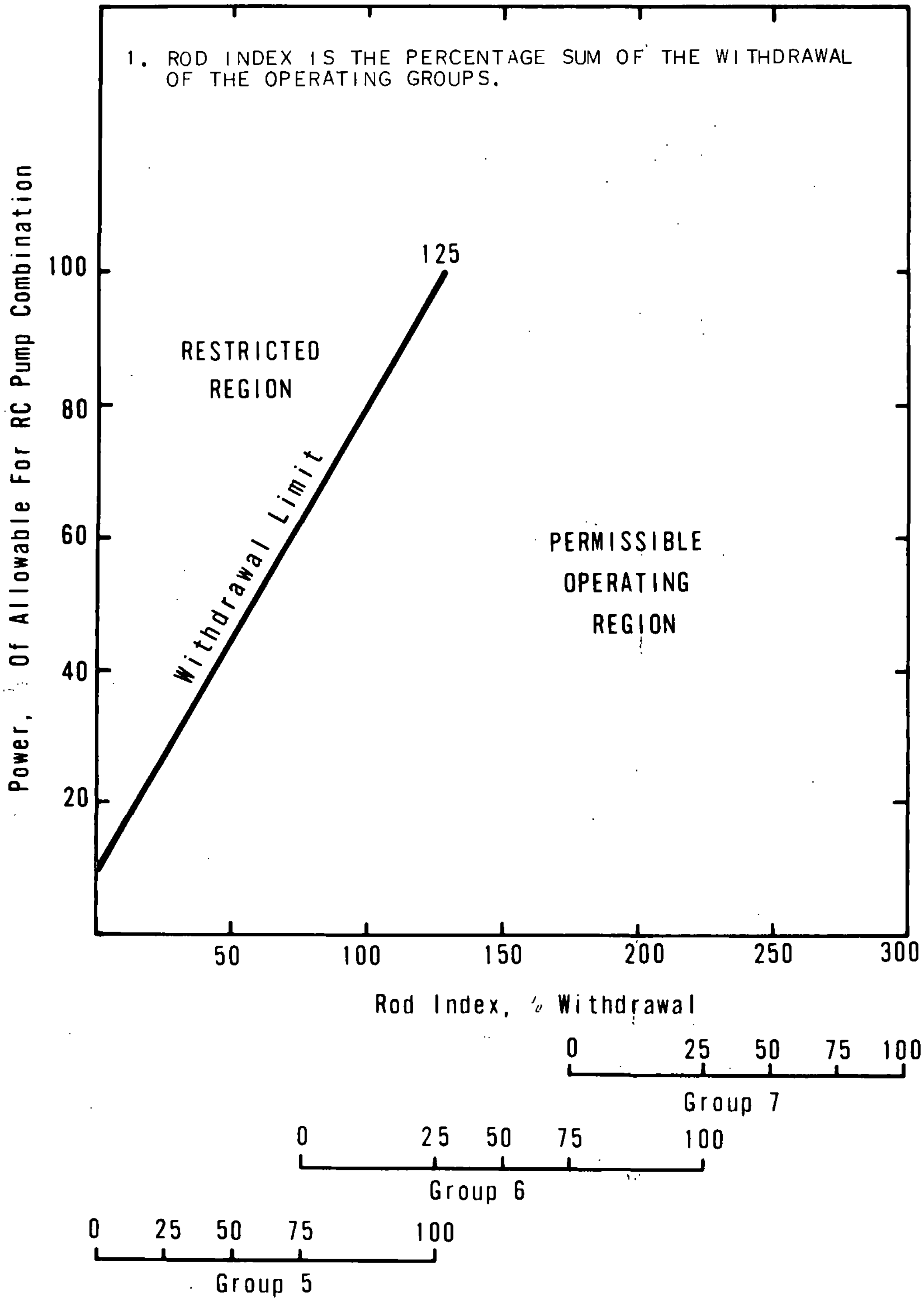


CONTROL ROD GROUP WITHDRAWAL
LIMITS FOR 4 PUMP OPERATION

OCONEE NUCLEAR STATION

Figure 3.5.2-1-2

Rev. 30. 9/4/73

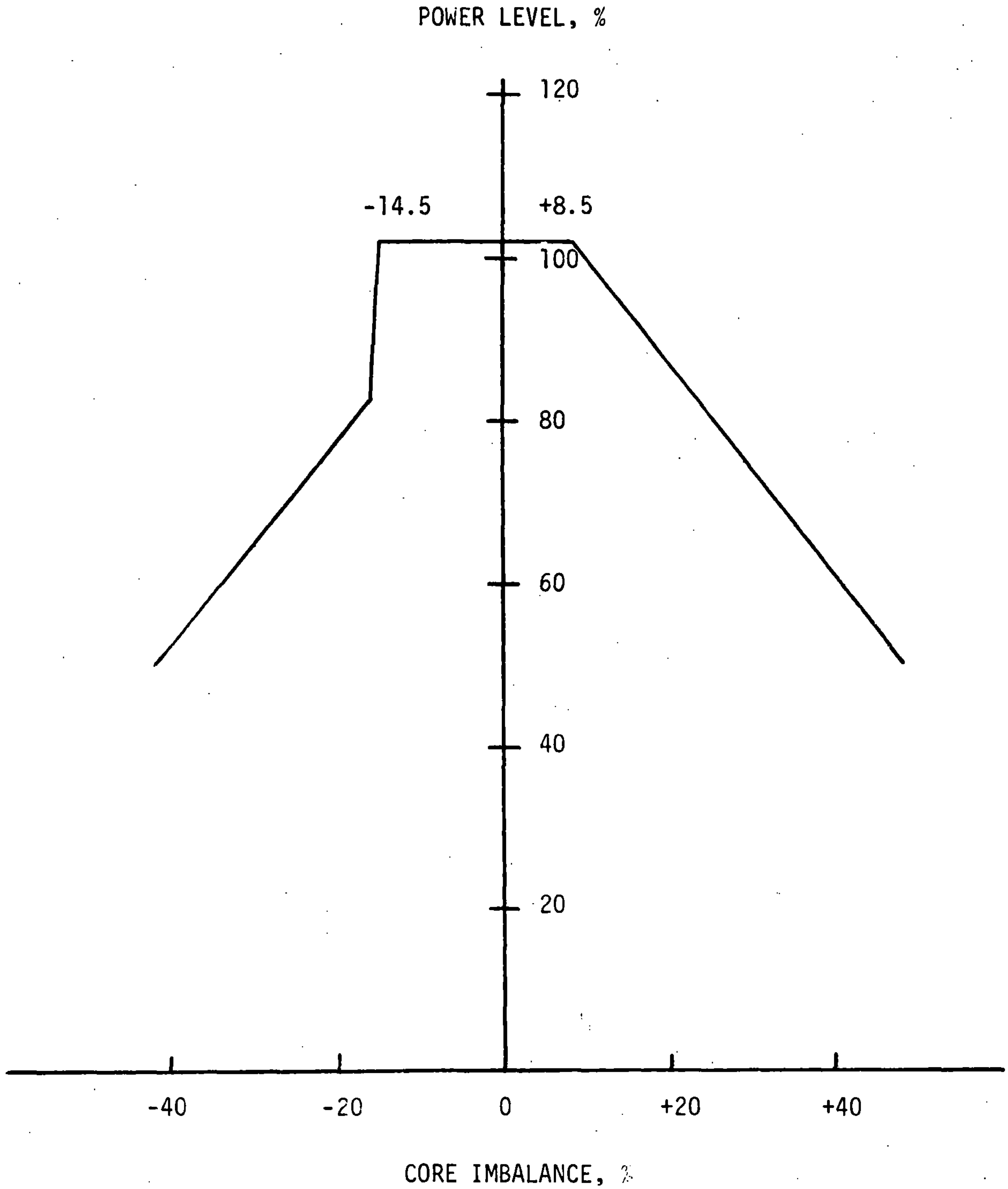


CONTROL ROD GROUP WITHDRAWAL
LIMITS FOR 3 AND 2 PUMP OPERATION

OCONEE NUCLEAR STATION

Figure 3.5.2 - 2

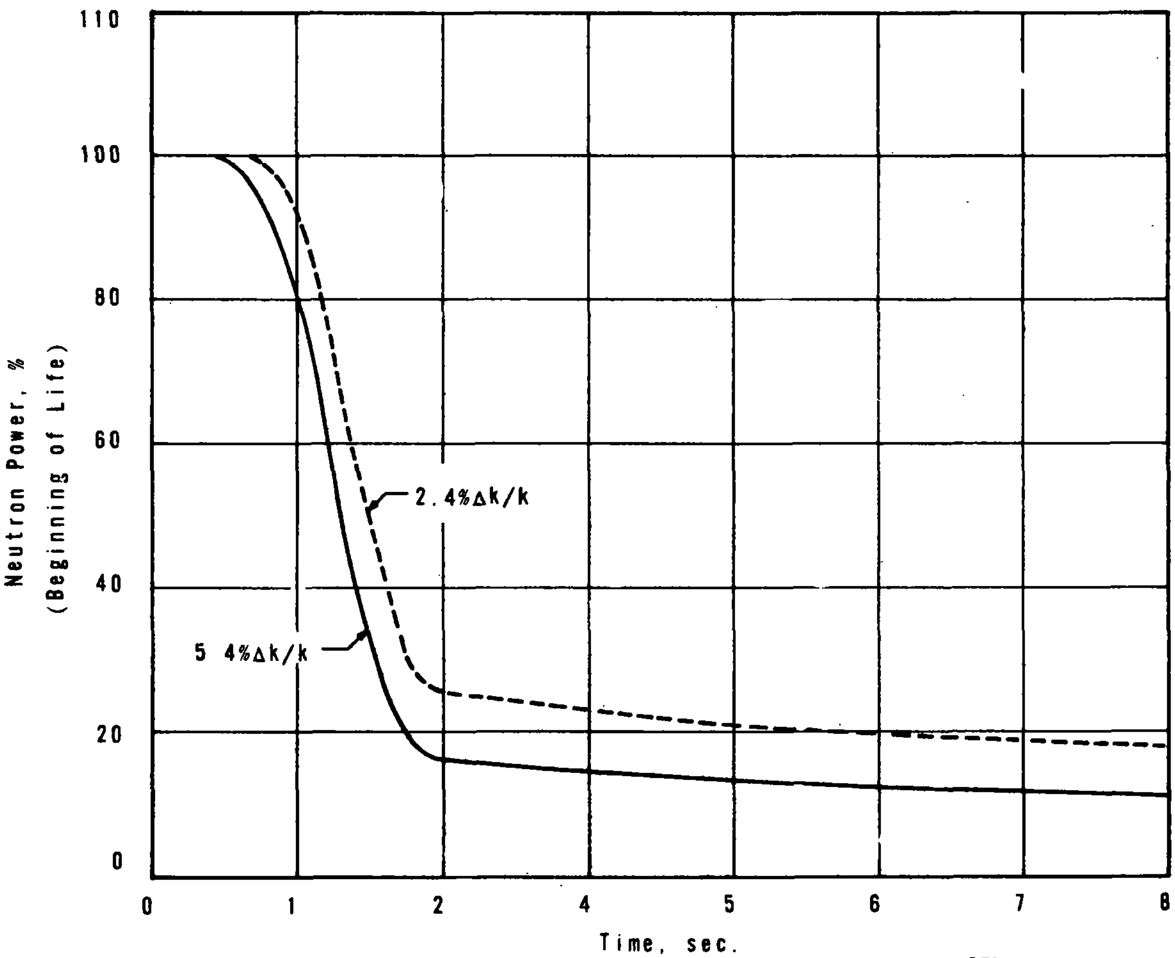
Rev. 30. 9/4/73



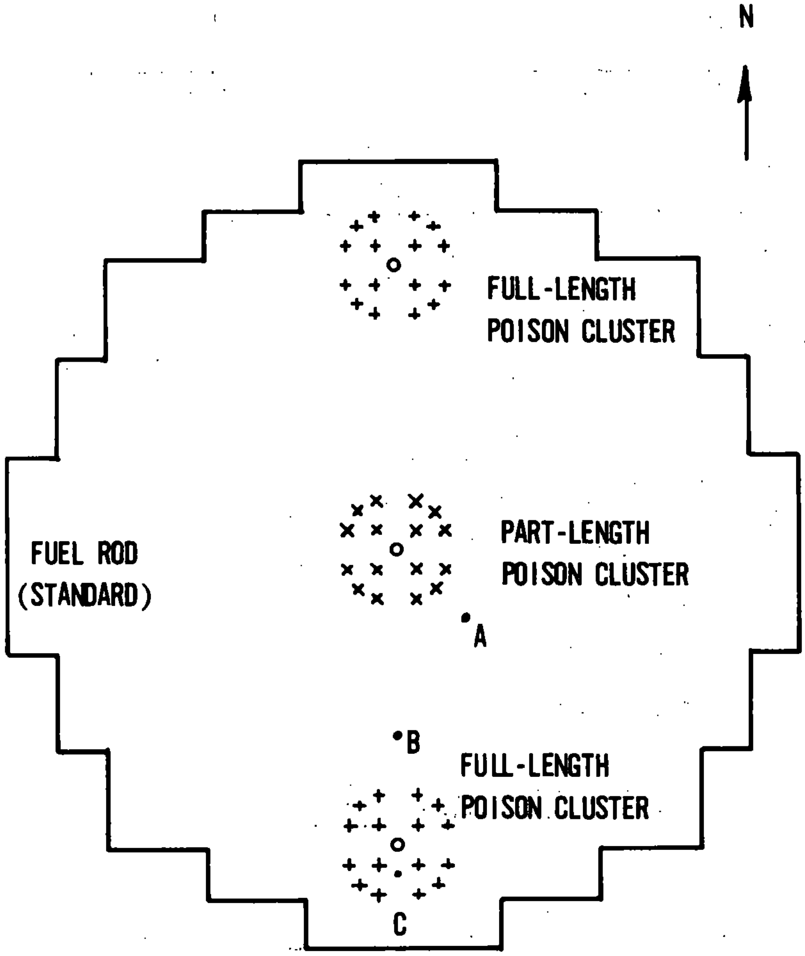
POWER-IMBALANCE ENVELOPE

OCONEE NUCLEAR STATION
Figure 3.5.2 - 3

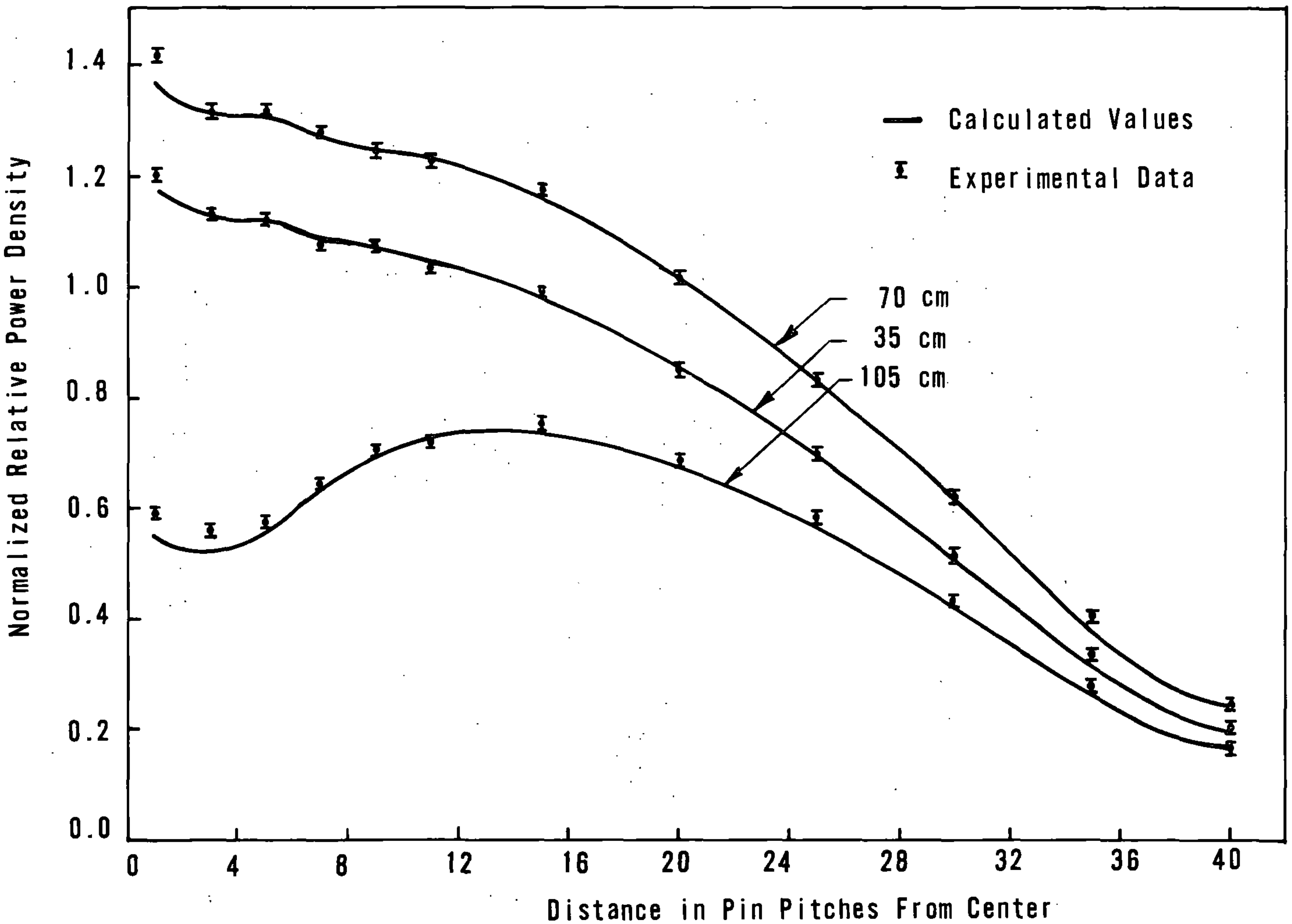
Rev. 30. 9/4/73



PER CENT NEUTRON POWER
VERSUS TIME FOLLOWING TRIP

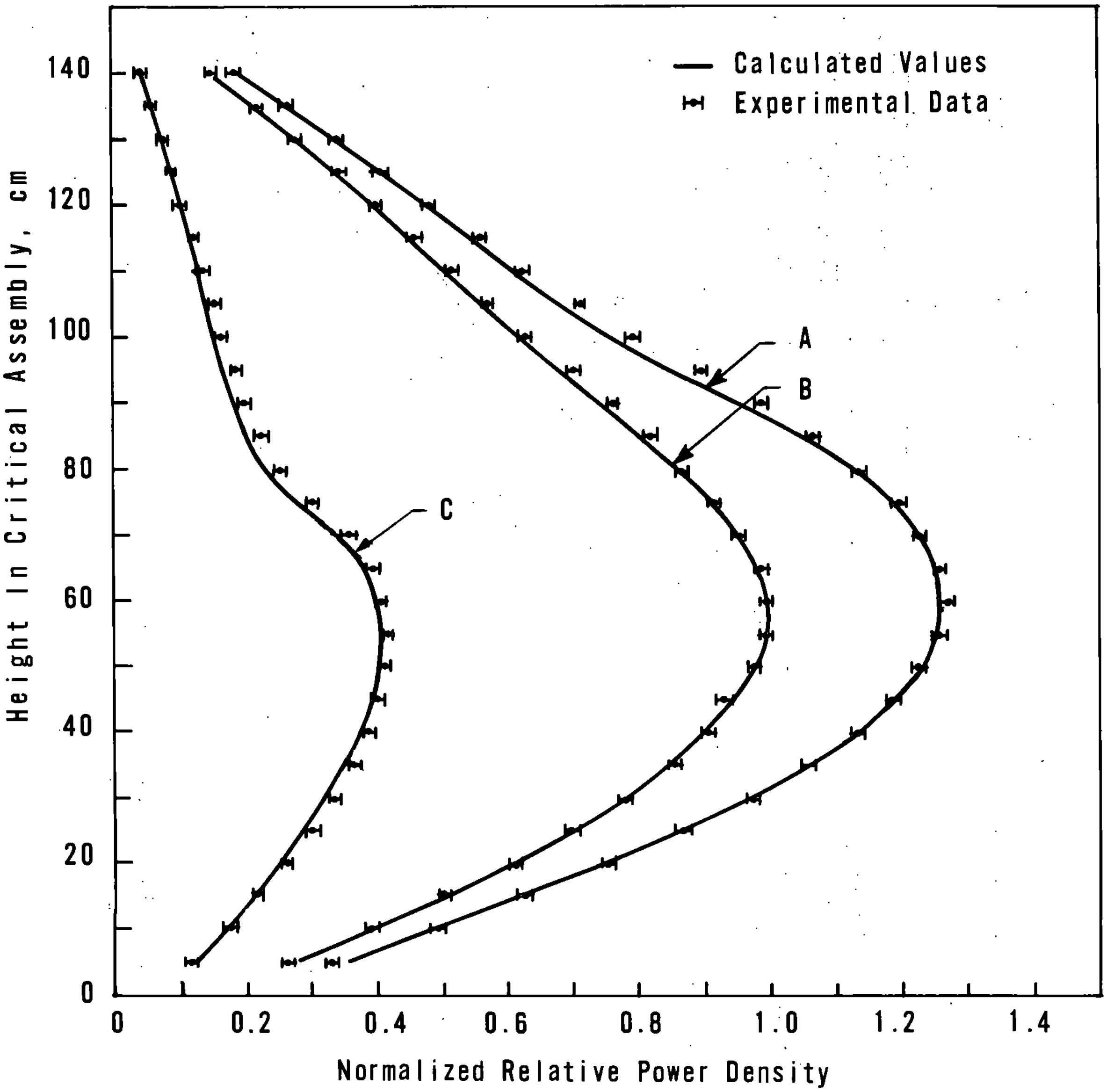


LOADING DIAGRAM FOR CORE IV-H



RADIAL POWER DISTRIBUTION AT THREE LEVELS FROM BOTTOM OF CORE

OCONEE NUCLEAR STATION
Figure 3-6B
(New) Rev. 4 4/20/70



AXIAL POWER DISTRIBUTION AT THREE SELECTED POSITIONS IN THE CORE

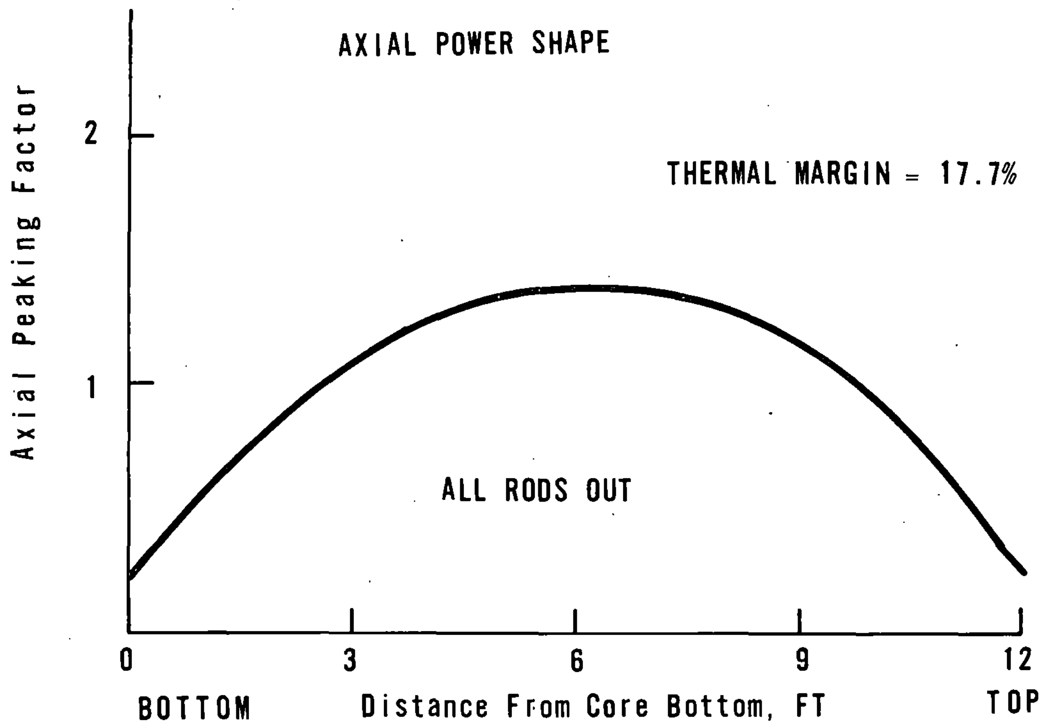
OCONEE NUCLEAR STATION

Figure 3 - 6C

(New) Rev. 4 4/20/70

1.44	1.44	1.42	1.39	1.30	1.17	.88	.46
1.44	1.43	1.42	1.39	1.34	1.15	.85	.43
1.42	1.42	1.41	1.42	1.30	1.07	.73	.33
1.39	1.39	1.42	1.35	1.19	.93	.55	
1.30	1.34	1.30	1.19	1.00	.69	.34	
1.17	1.15	1.07	.93	.69	.40		
.88	.85	.73	.55	.34			
.46	.43	.33					

RADIAL POWER SHAPE

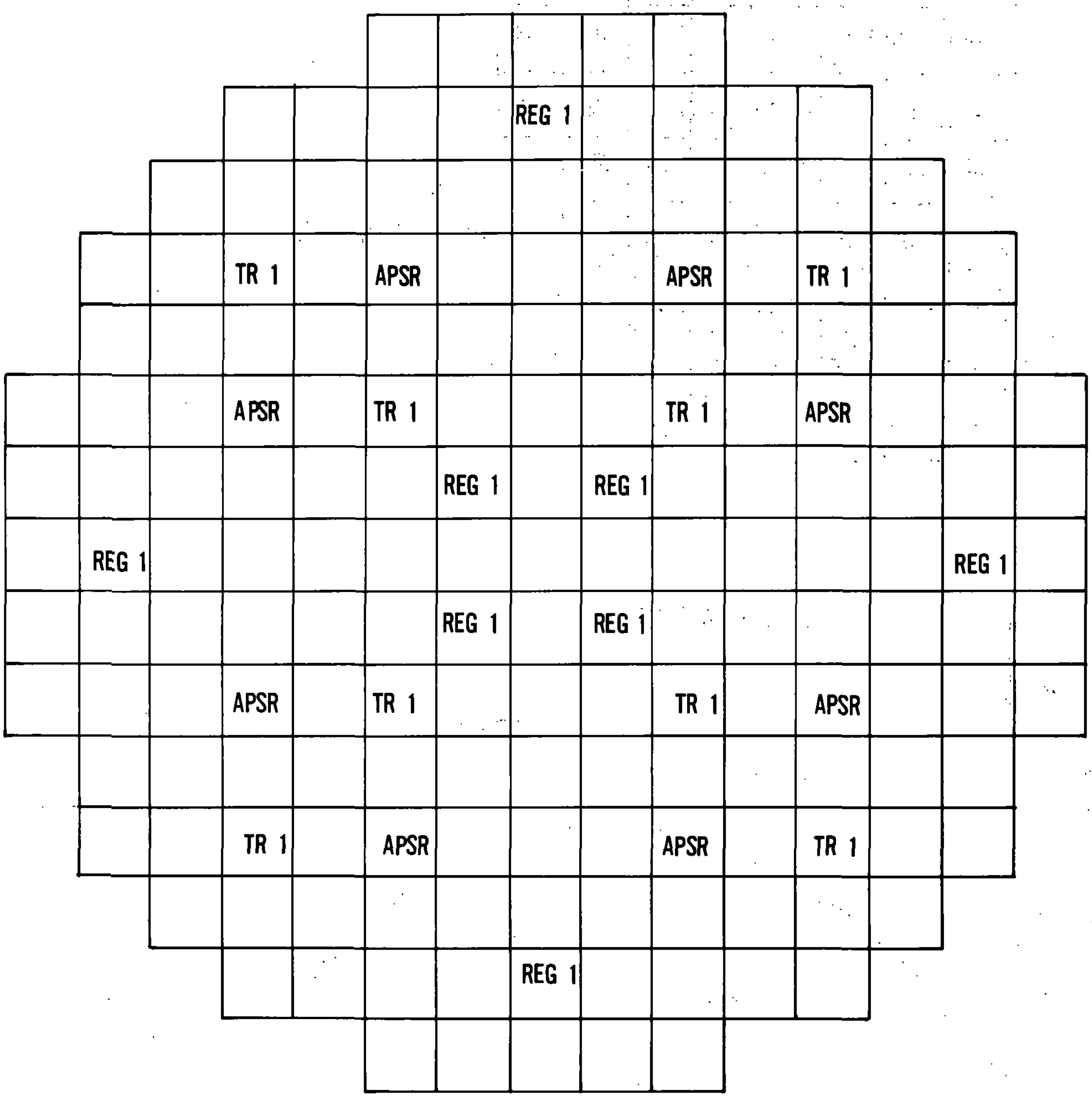


RADIAL AND AXIAL CORE POWER DISTRIBUTION FOR THE UNRODDED CORE AT BOL (OCONEE UNIT 1, CYCLE 1)

OCONEE NUCLEAR STATION

Figure 3 - 6D

(New) Rev. 4 4/20/70



OCONEE I ROD PATTERN 0-100 DAYS (CYCLE 1)

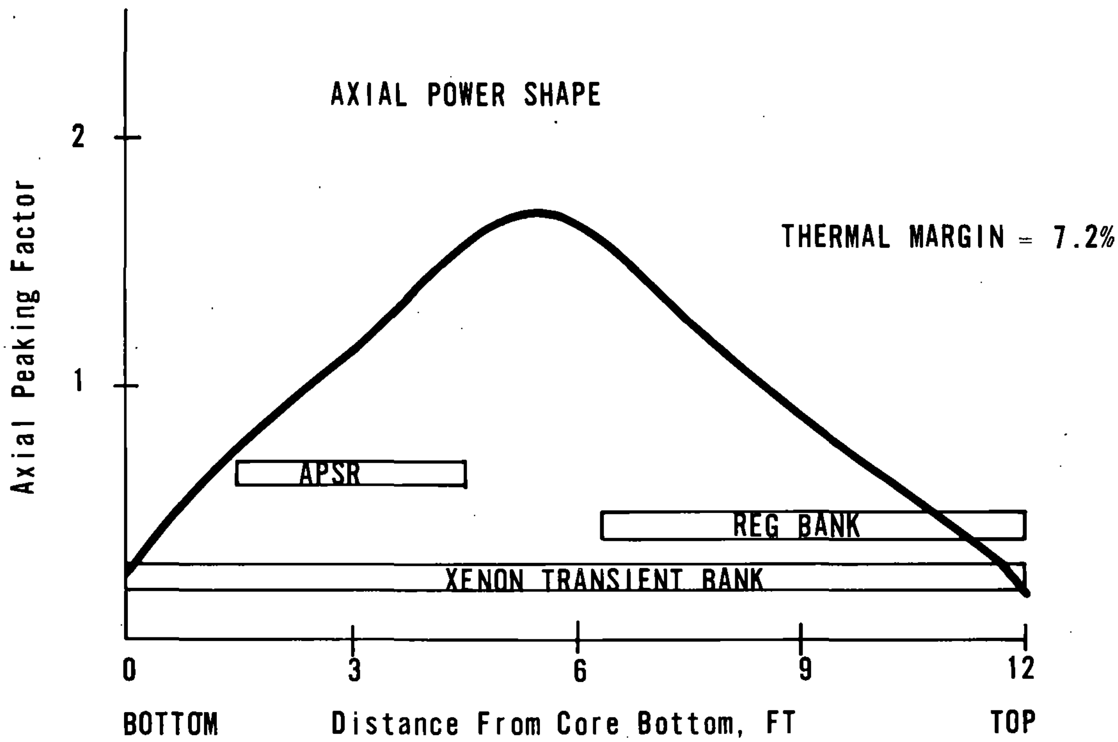
OCONEE NUCLEAR STATION

Figure 3 - 6E

(New) Rev. 4 4/20/70

1.31	1.27	1.34	1.47	1.51	1.40	1.01	.60
1.27	1.16	1.16	1.35	1.47	1.37	1.06	.59
1.34	1.16	.84	1.17	1.23	1.20	.93	.47
1.47	1.35	1.17	1.14	1.01	.95	.67	
1.51	1.47	1.23	1.01	.62	.60	.39	
1.40	1.37	1.20	.95	.60	.37		
1.01	1.06	.93	.67	.39			
.60	.59	.47					

RADIAL POWER SHAPE



RADIAL AND AXIAL CORE POWER DISTRIBUTION DURING XENON BURNOUT FOR FOUR DAY POWER TRANSIENT (OCONEE UNIT 1, CYCLE 1)

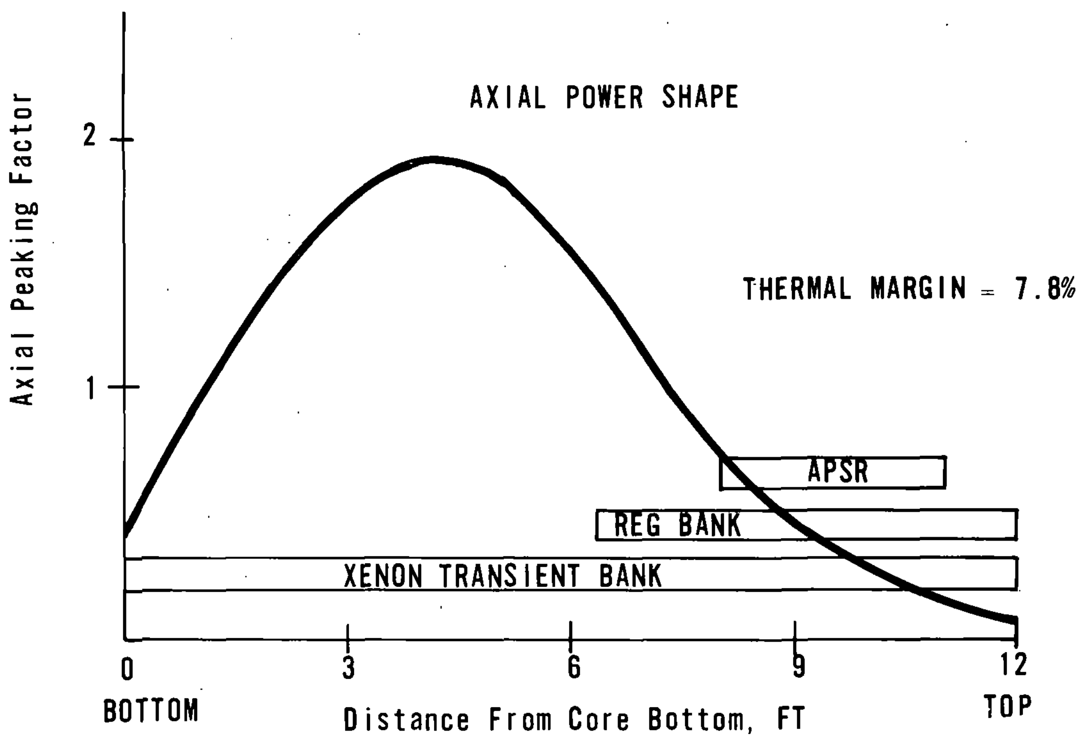
OCONEE NUCLEAR STATION

Figure 3 - 6F

(New) Rev. 4 4/20/70

1.31	1.27	1.32	1.43	1.48	1.40	1.04	.60
1.27	1.18	1.15	1.34	1.47	1.36	1.05	.58
1.32	1.15	.84	1.19	1.30	1.21	.91	.46
1.43	1.34	1.19	1.16	1.03	.95	.65	
1.48	1.47	1.30	1.03	.62	.59	.37	
1.40	1.36	1.21	.95	.59	.36		
1.04	1.05	.91	.65	.37			
.60	.58	.46					

RADIAL POWER SHAPE

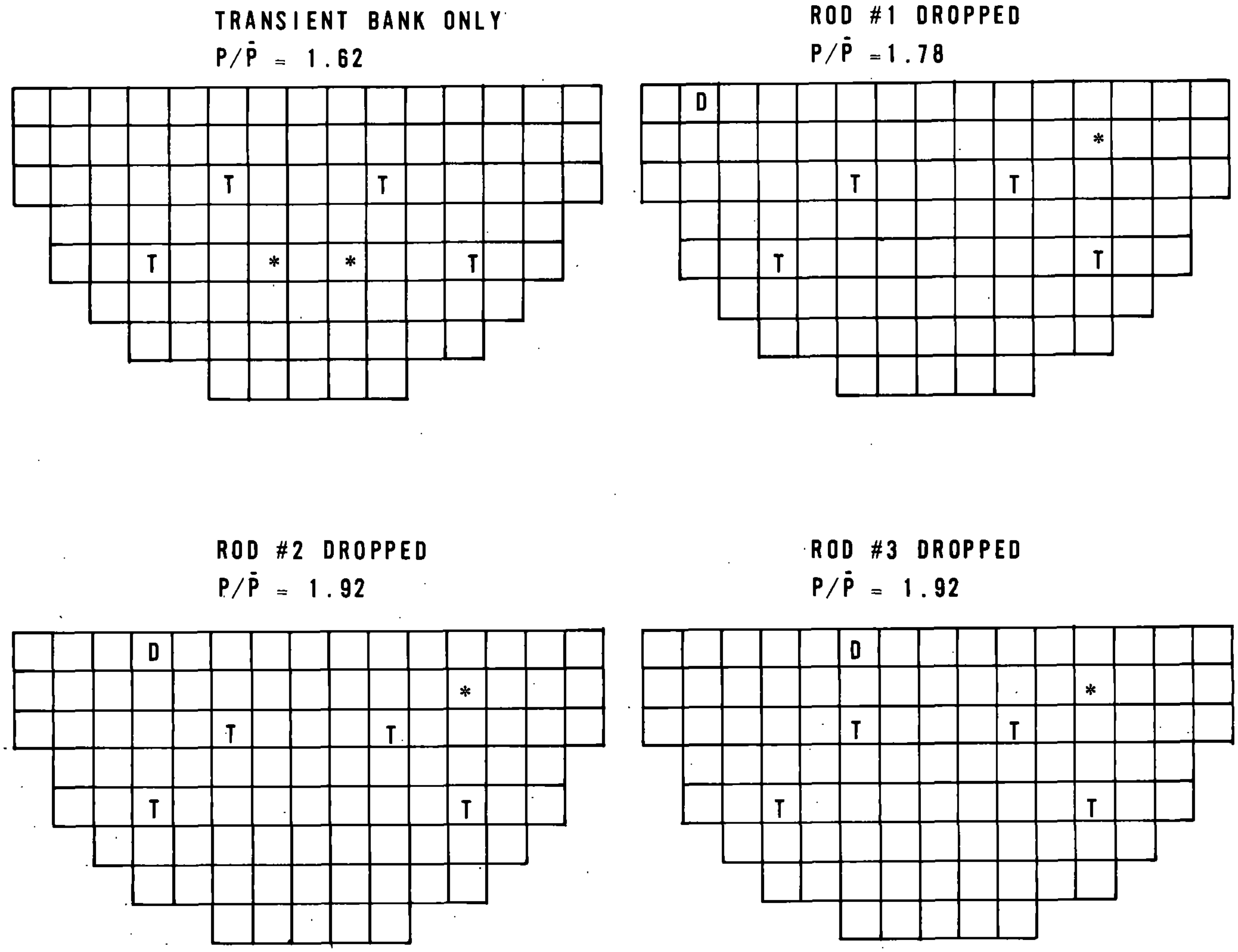


RADIAL AND AXIAL CORE POWER DISTRIBUTION DURING XENON BURNOUT FOR FOUR DAY POWER TRANSIENT (OCONEE UNIT 1, CYCLE 1)

OCONEE NUCLEAR STATION

Figure 3 - 6G

(New) Rev. 4 4/20/70



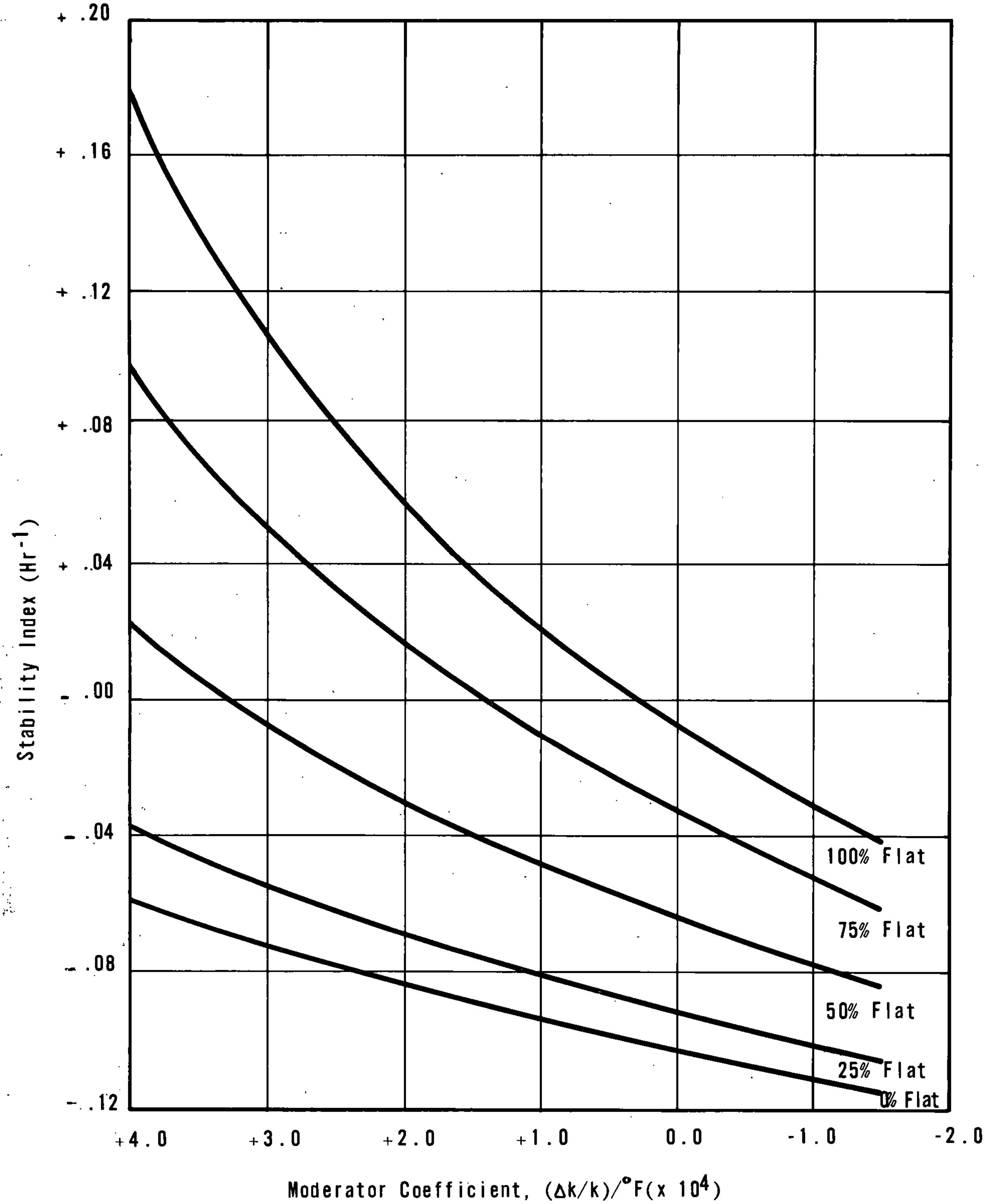
T - TRANSIENT ROD
D - DROPPED ROD
* - LOCATION OF POWER PEAK

POWER PEAKING CAUSED BY DROPPED RODS
(OCONEE UNIT 1, CYCLE 1, BOL)

OCONEE NUCLEAR STATION

Figure 3 - 6H

(New) Rev. 4 4/20/70

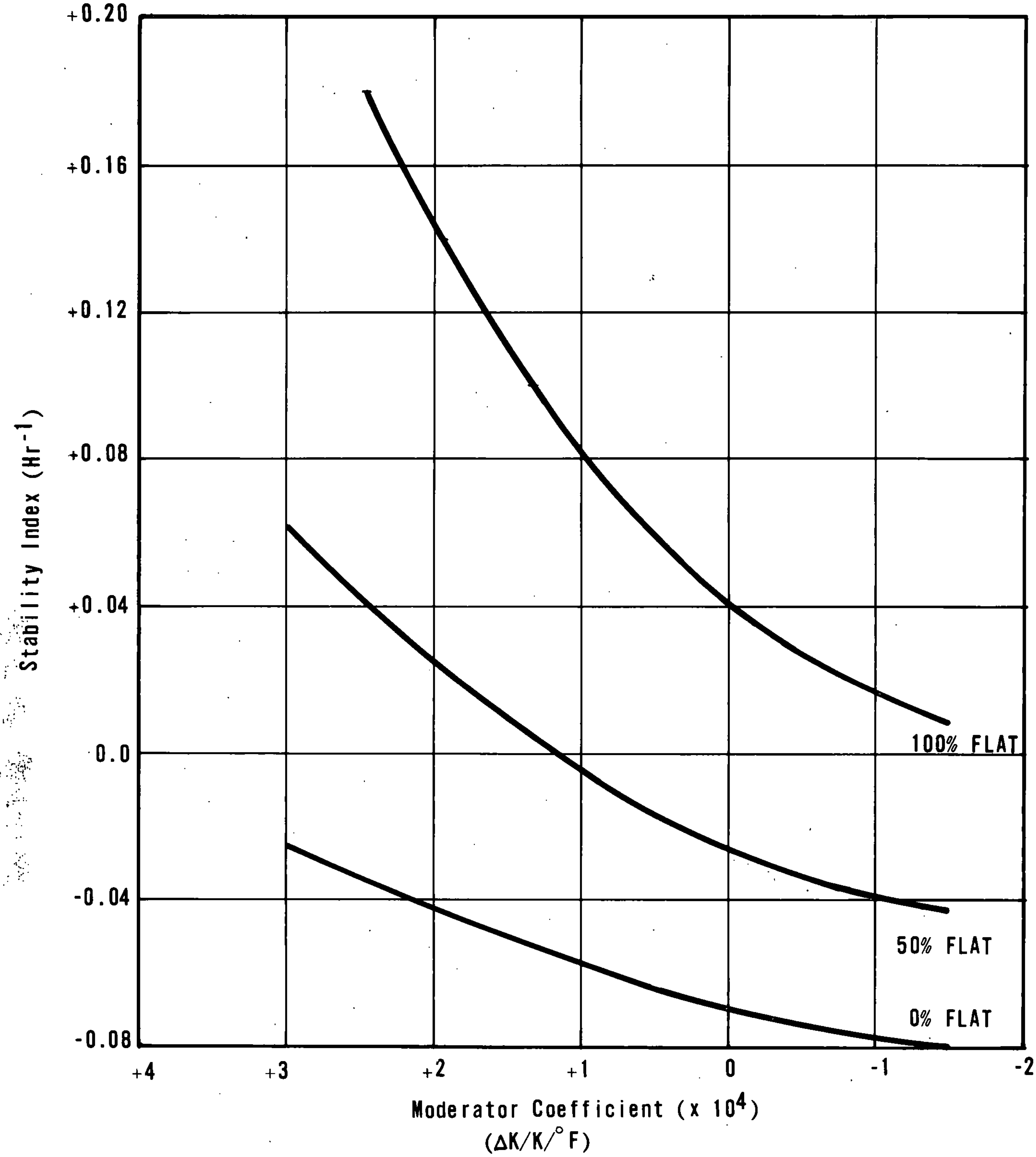


AZIMUTHAL STABILITY INDEX VS. MODERATOR
COEFFICIENT FROM 3 DIMENSIONAL CASE
(OCONEE UNIT 1, CYCLE 1)

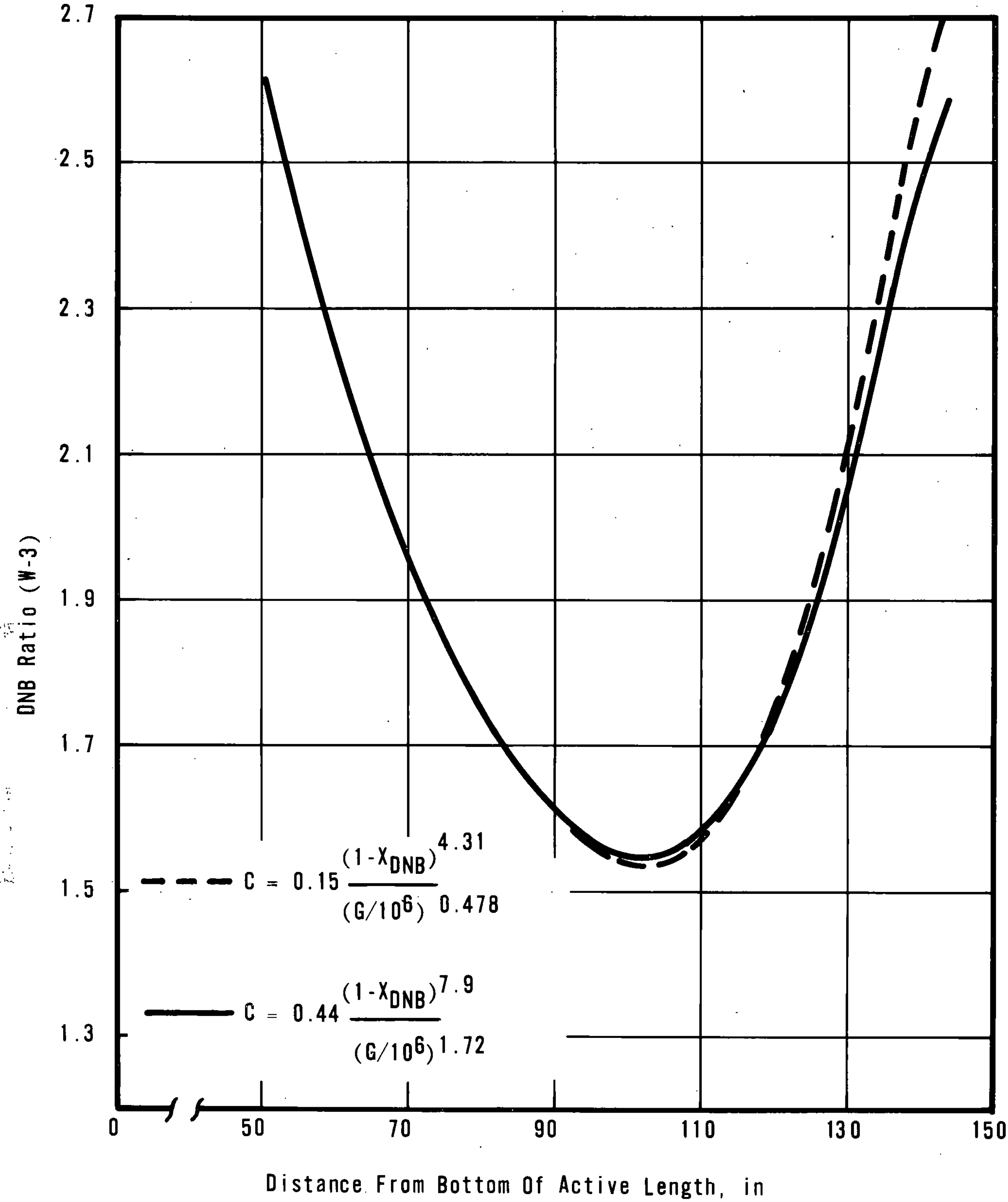
OCONEE NUCLEAR STATION

Figure 3 - 6i

(New) Rev. 4 4/20/70



AZIMUTHAL STABILITY INDEX WITH COMPOUNDED ERROR VS. MODERATOR COEFFICIENT CALCULATED FROM 3 DIMENSIONAL CASE (OCONEE UNIT 1, CYCLE 1)

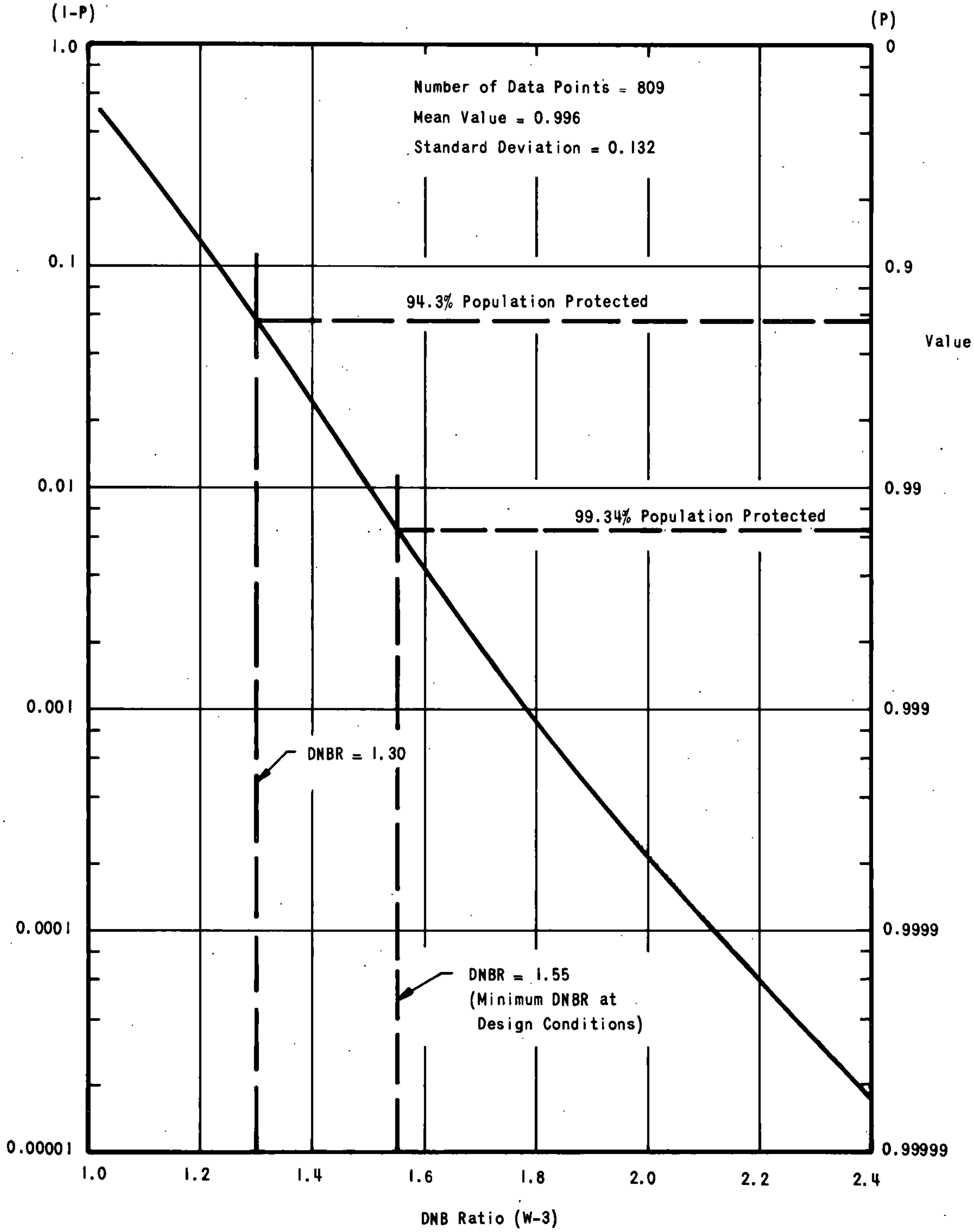


DNB RATIOS (W-3) IN THE HOT UNIT CELL
FOR THE OLD AND NEW C-FACTOR

OCONEE NUCLEAR STATION

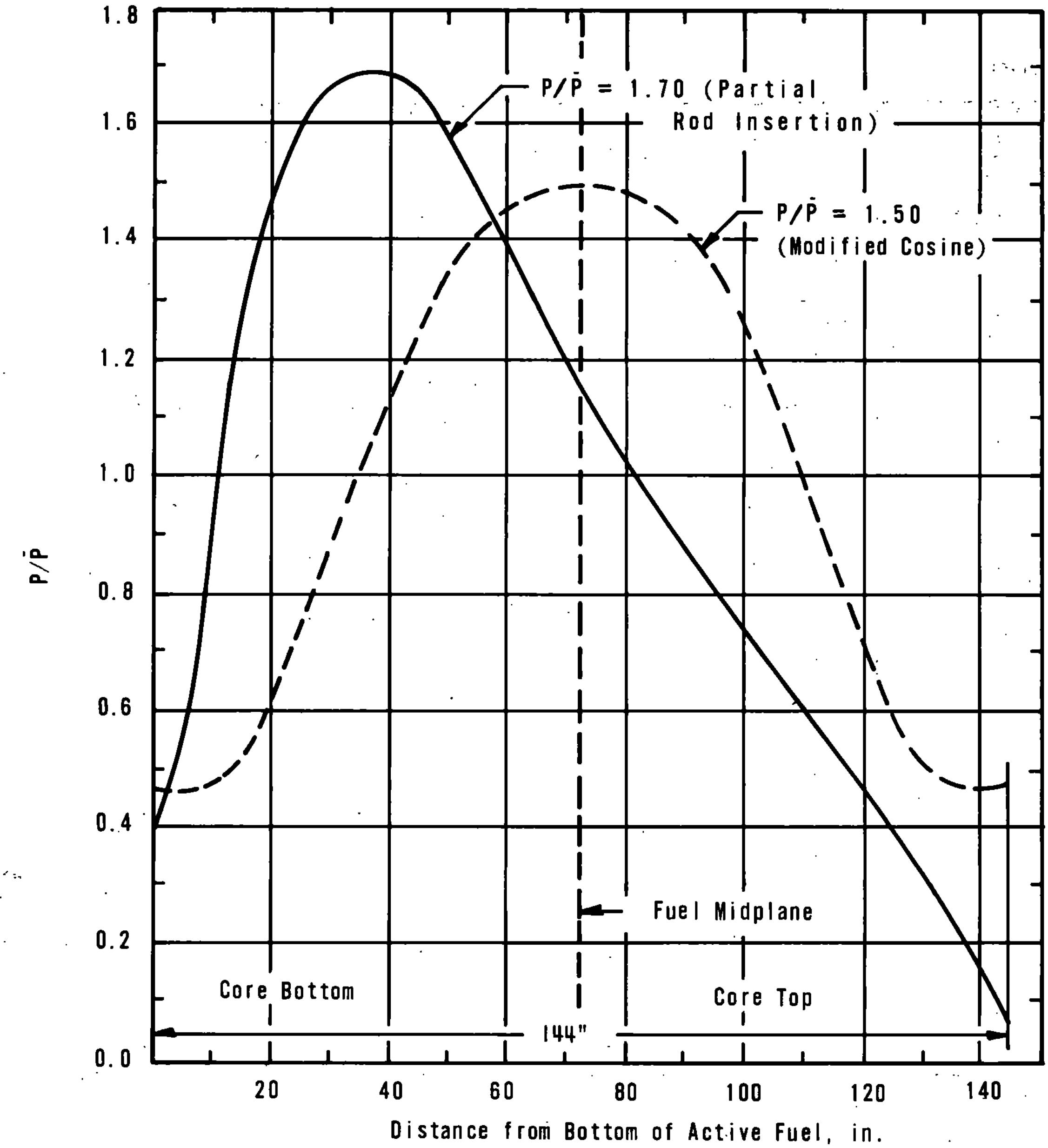
Figure 3 - 6K

(New) Rev. 4 4/20/70

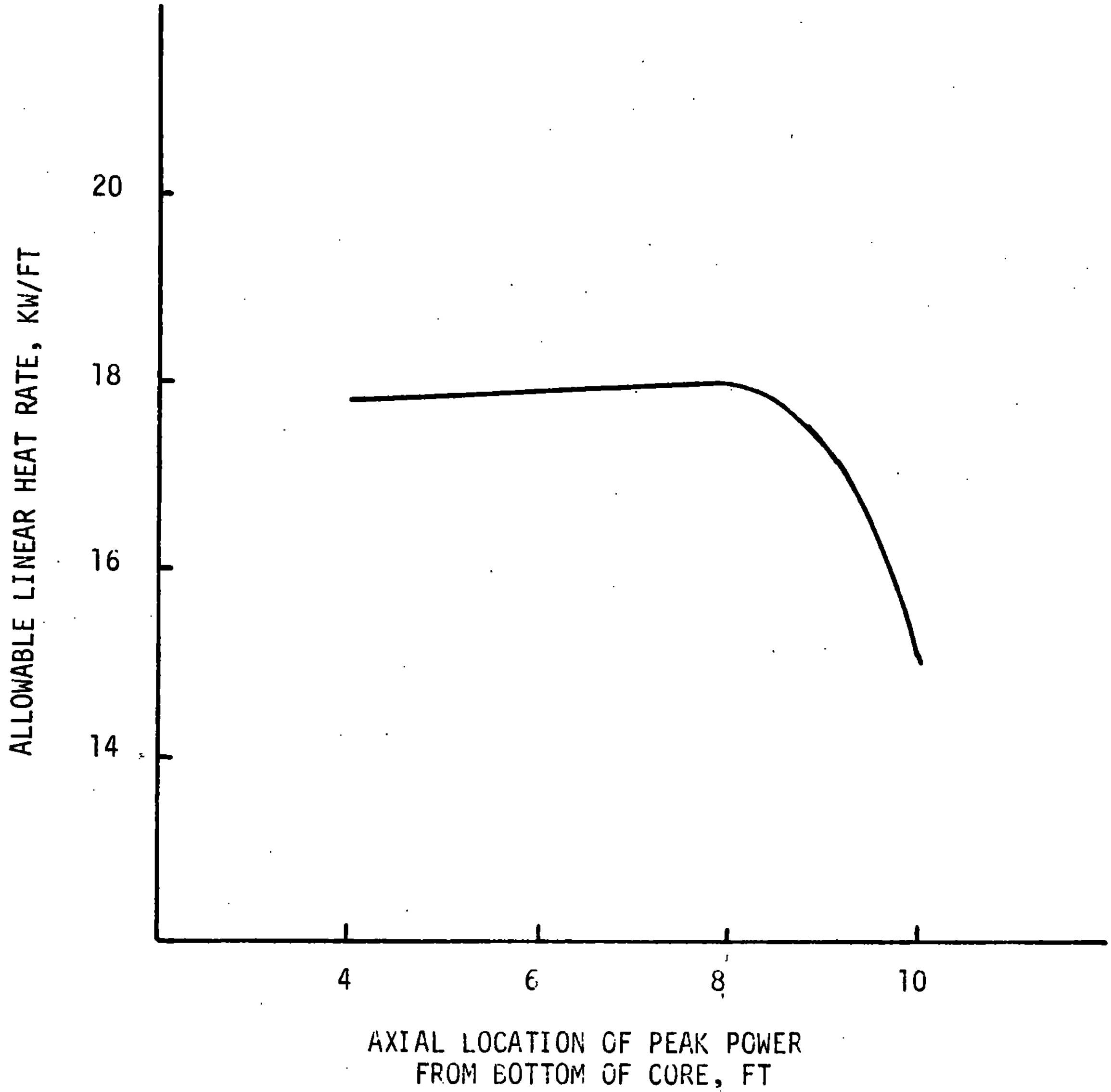


POPULATION PROTECTED, P, AND 1-P
VERSUS DNB RATIO (W-3)

OCONEE NUCLEAR STATION
Figure 3 - 7

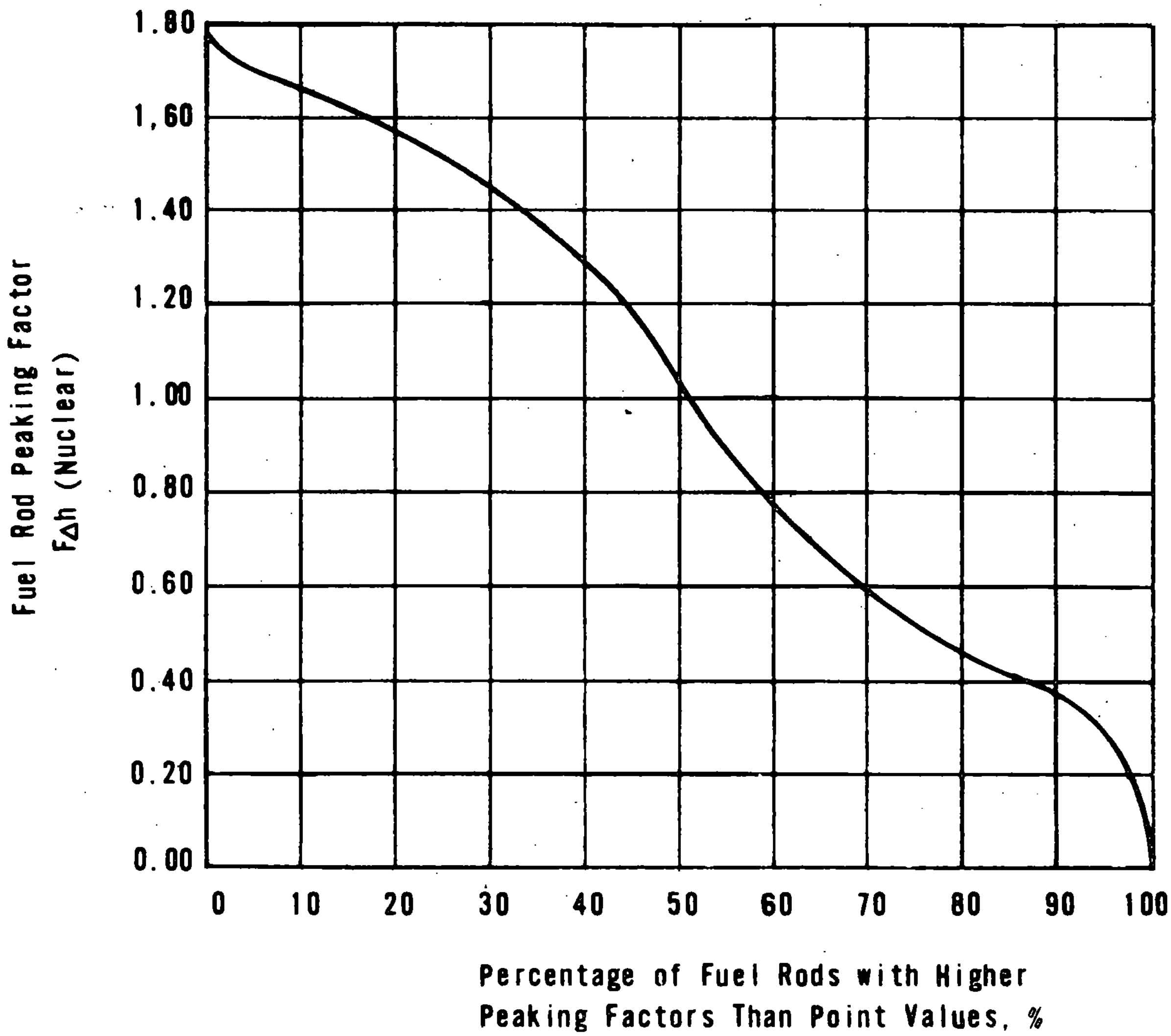


POWER SHAPE REFLECTING INCREASED AXIAL POWER PEAK FOR 144-INCH CORE

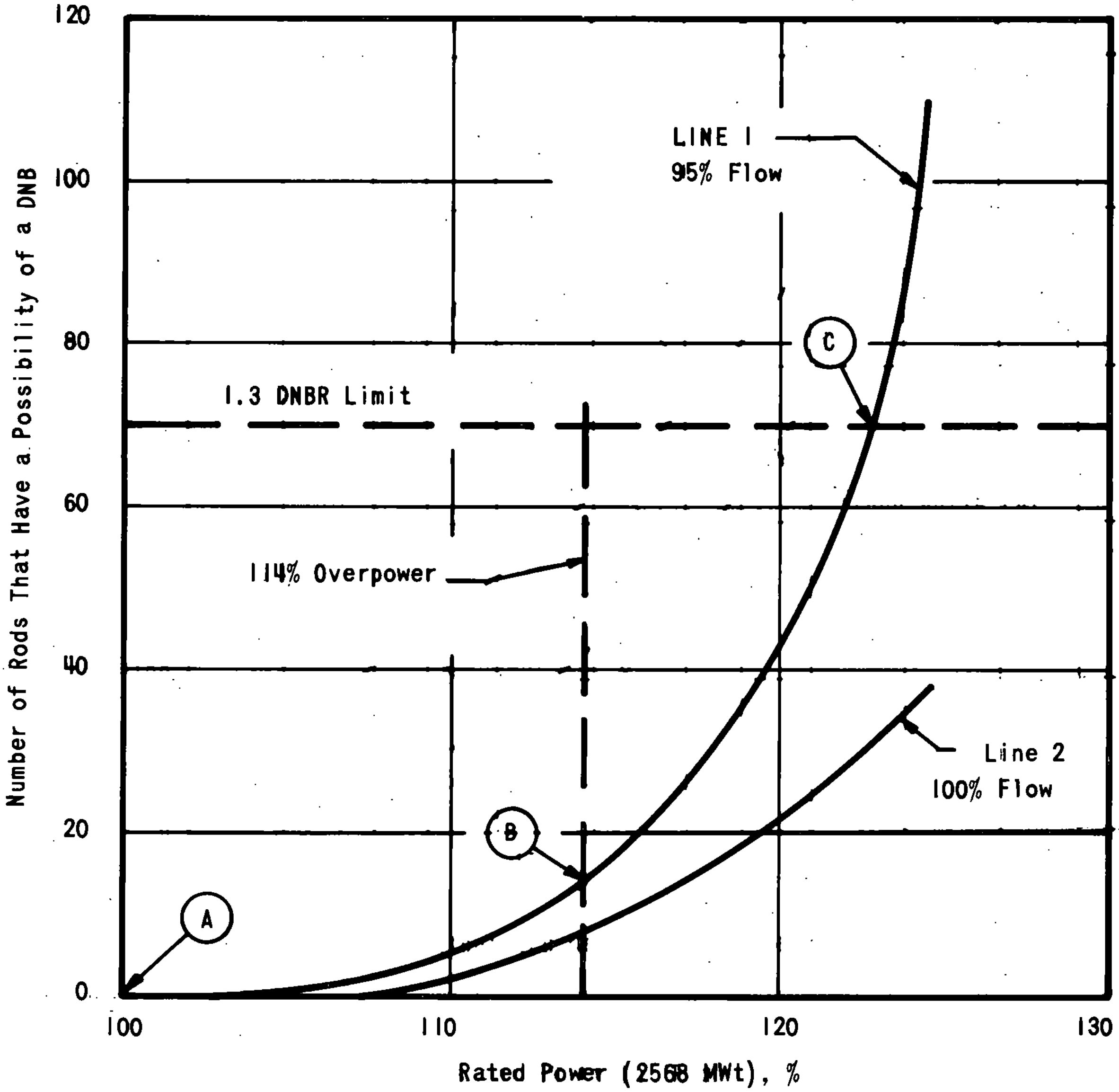


MAXIMUM ALLOWABLE LINEAR HEAT RATE
PER INTERIM ACCEPTANCE CRITERIA

OCONEE NUCLEAR STATION
Figure 3.5.2-4



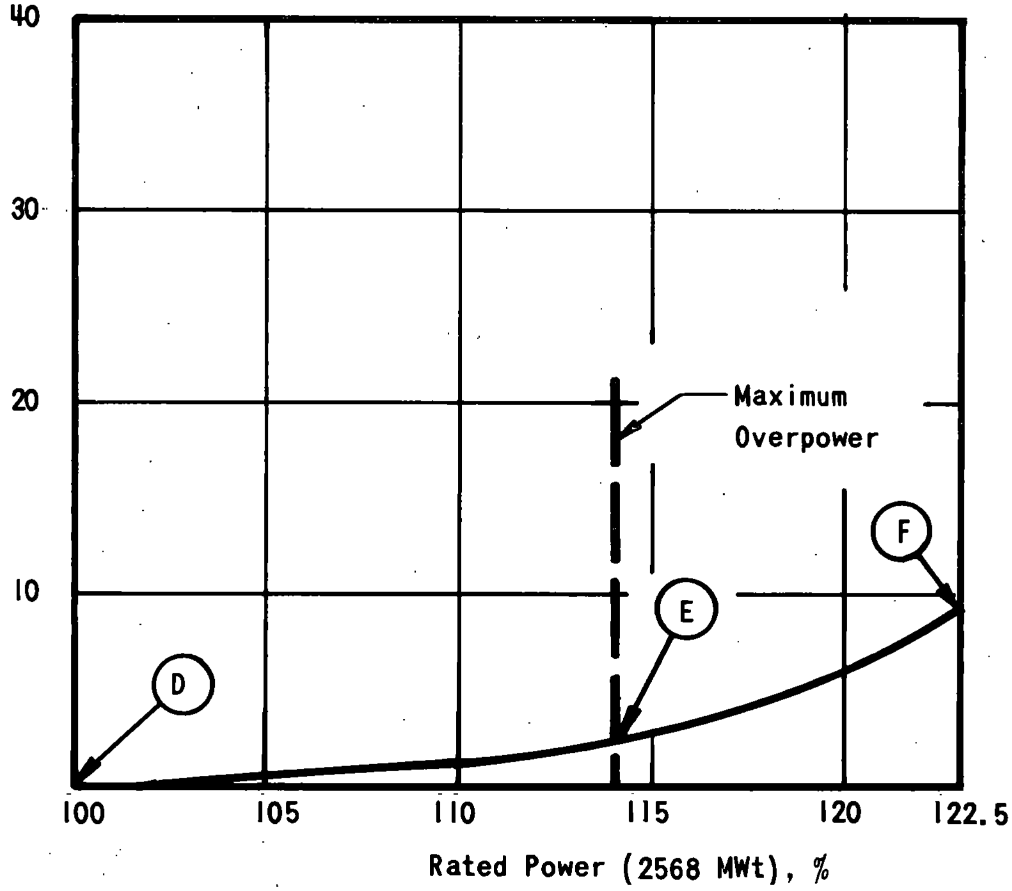
DISTRIBUTION OF FUEL ROD PEAKING



POSSIBLE FUEL ROD DNB'S FOR MAXIMUM DESIGN CONDITIONS - 36,816 ROD CORE

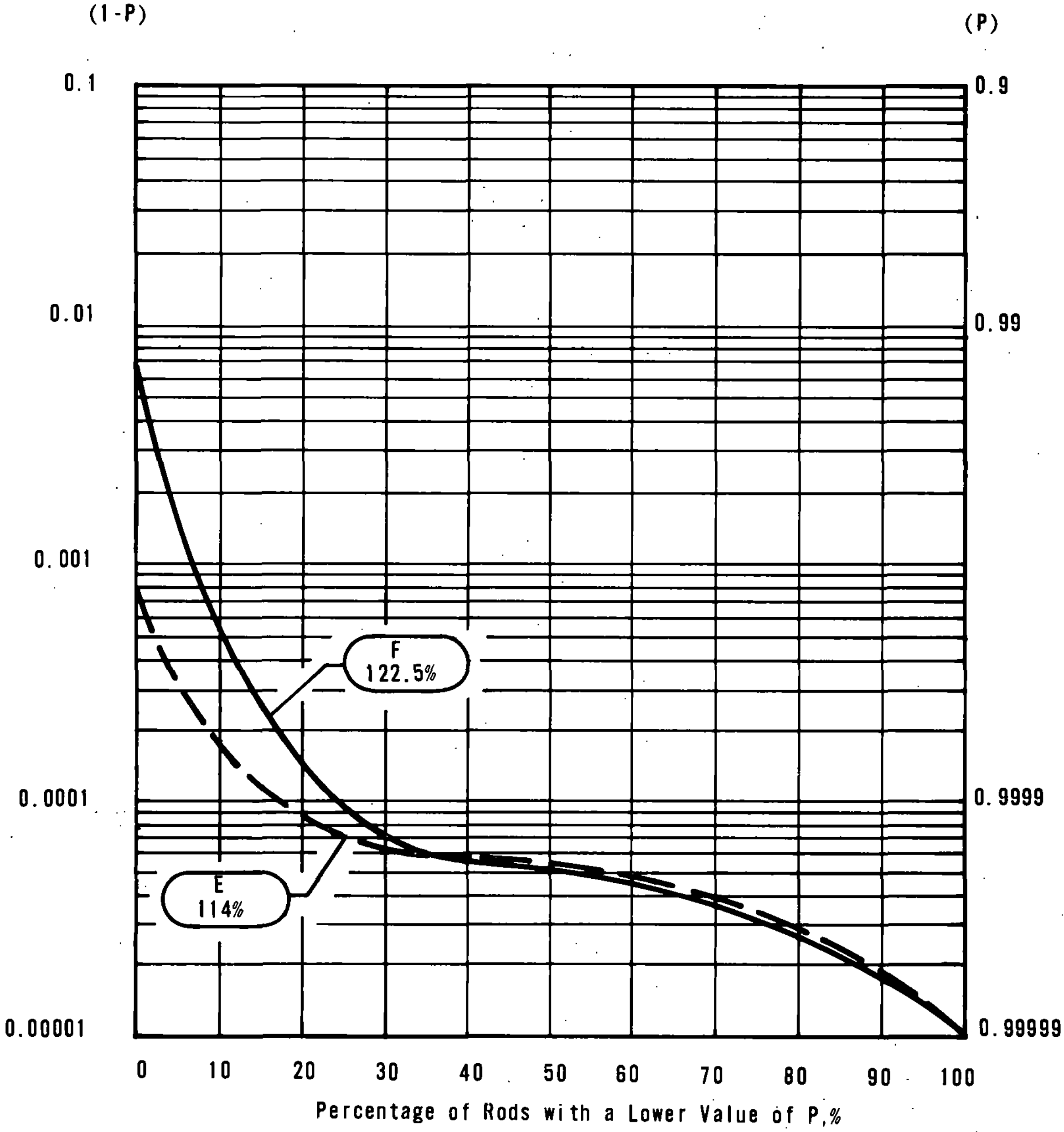
OCONEE NUCLEAR STATION

Figure 3 - 10

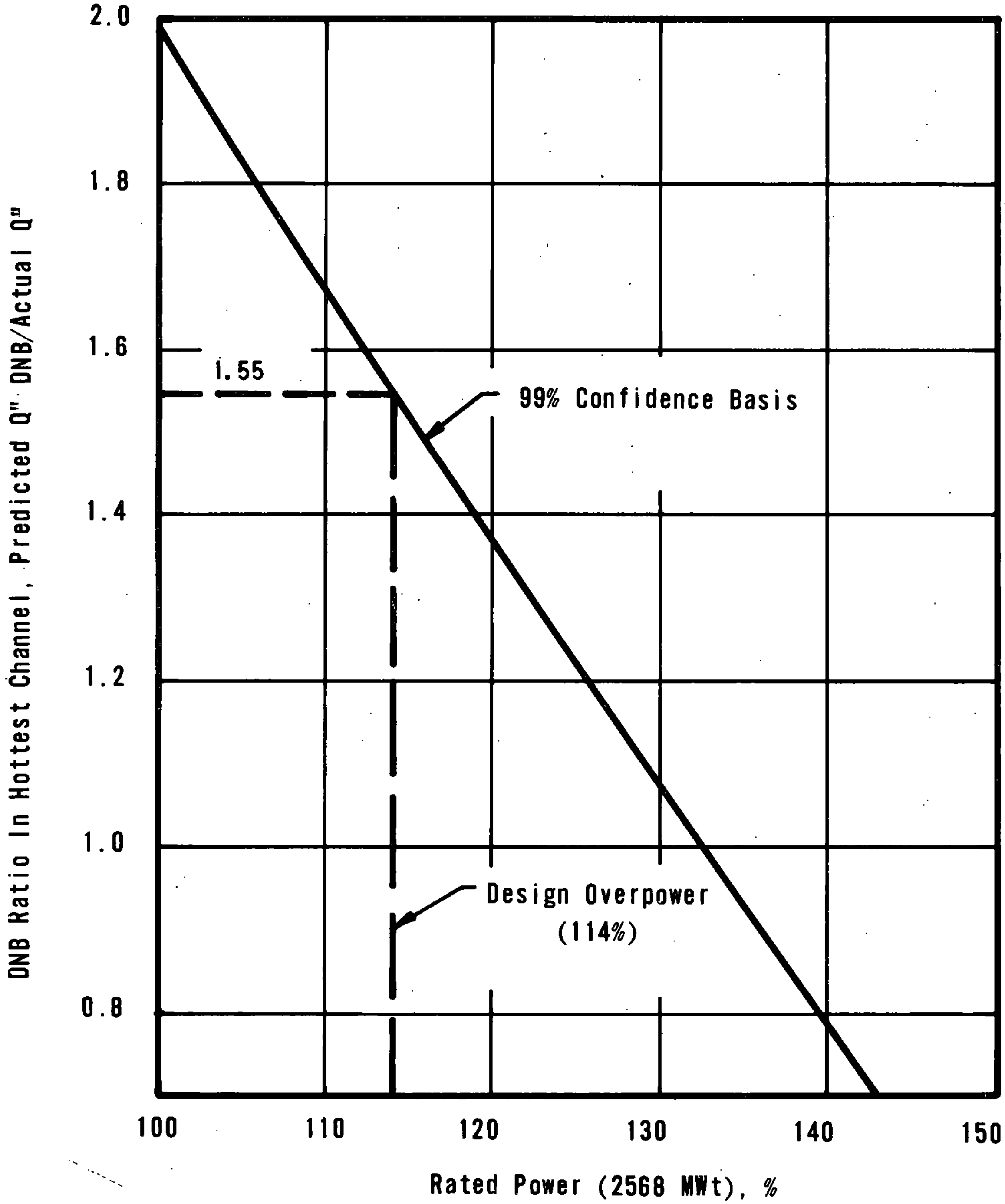


POSSIBLE FUEL ROD DNB'S FOR
MOST PROBABLE CONDITIONS -
36,816-ROD CORE

OCONEE NUCLEAR STATION
Figure 3 - 11



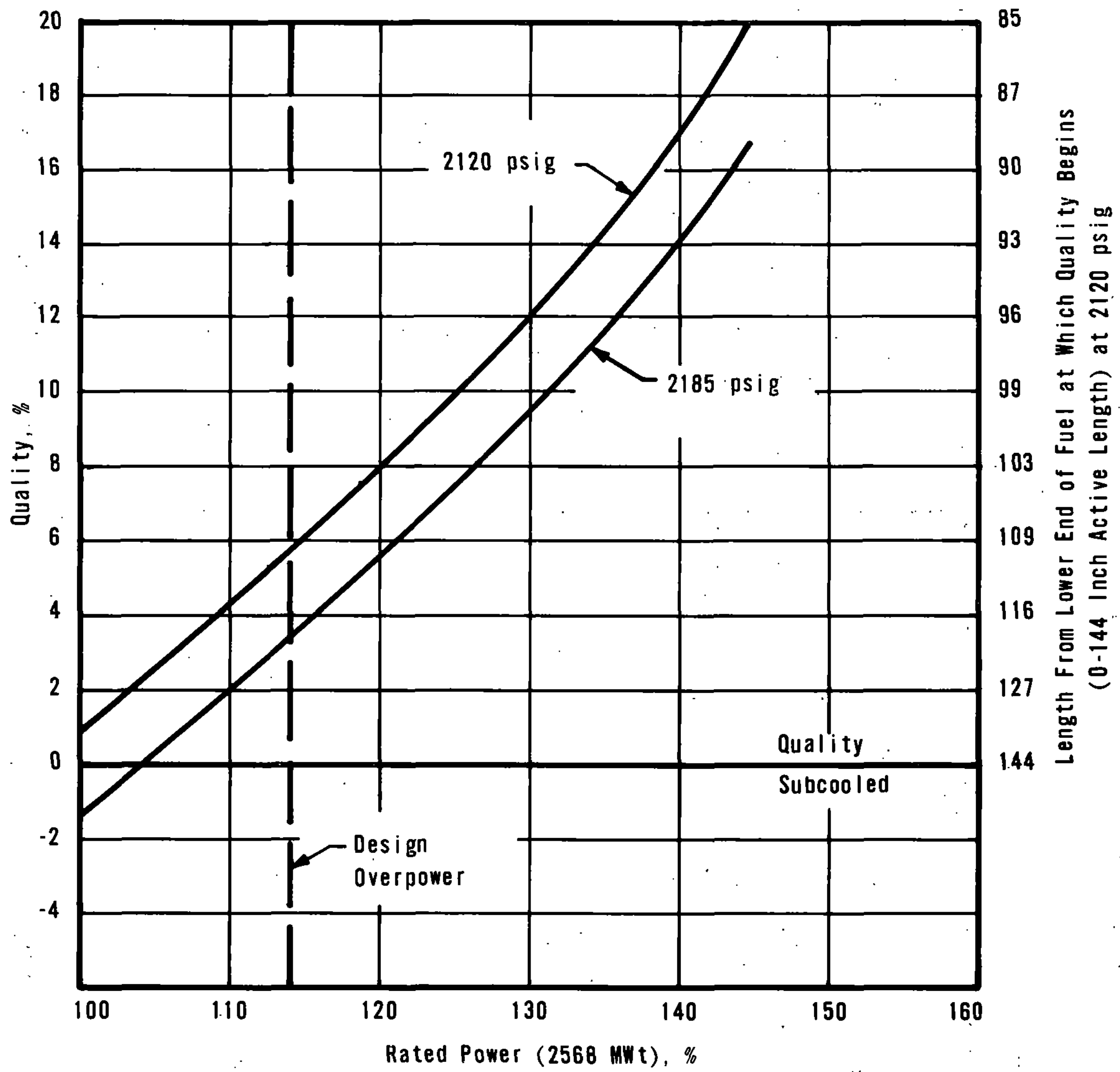
DISTRIBUTION OF POPULATION PROTECTED, P, AND 1-P VERSUS NUMBER OF RODS FOR MOST PROBABLE CONDITIONS



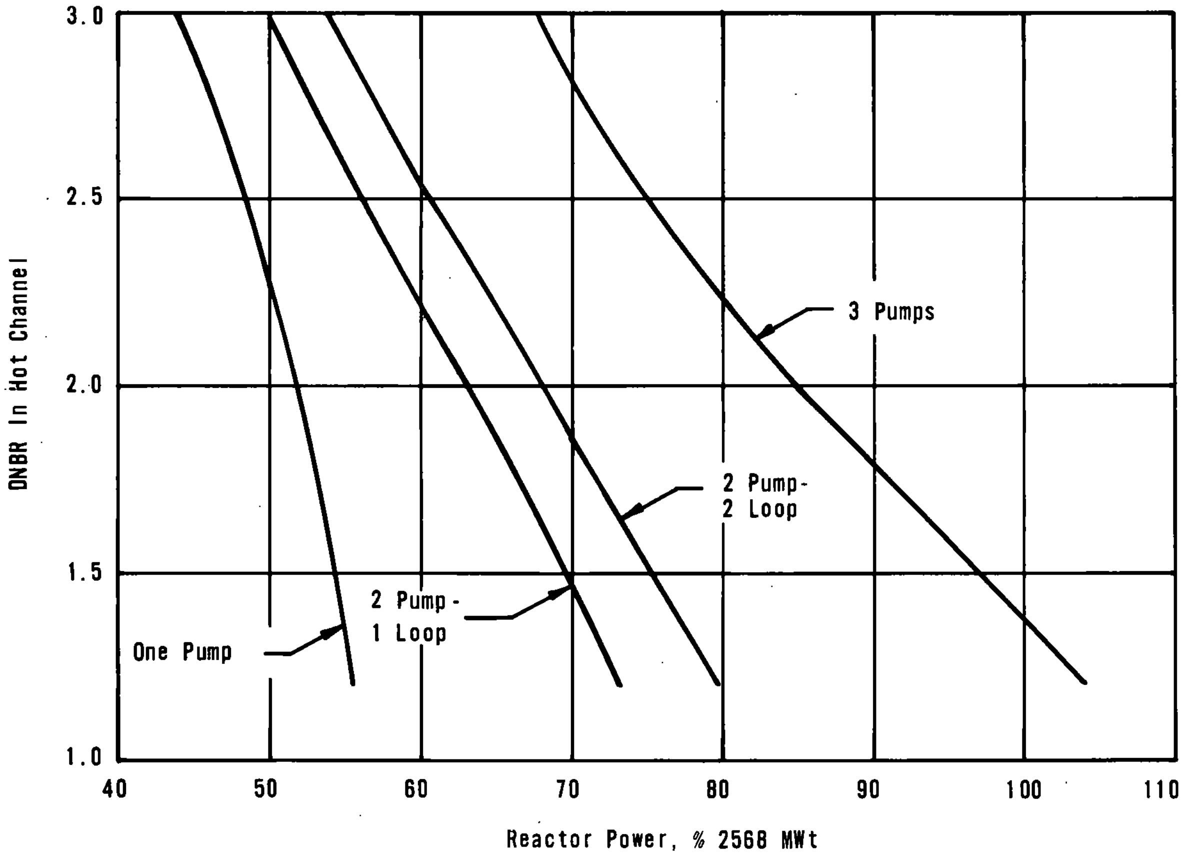
DNB RATIOS (W-3) IN HOT UNIT
CELL VERSUS REACTOR POWER

OCONEE NUCLEAR STATION

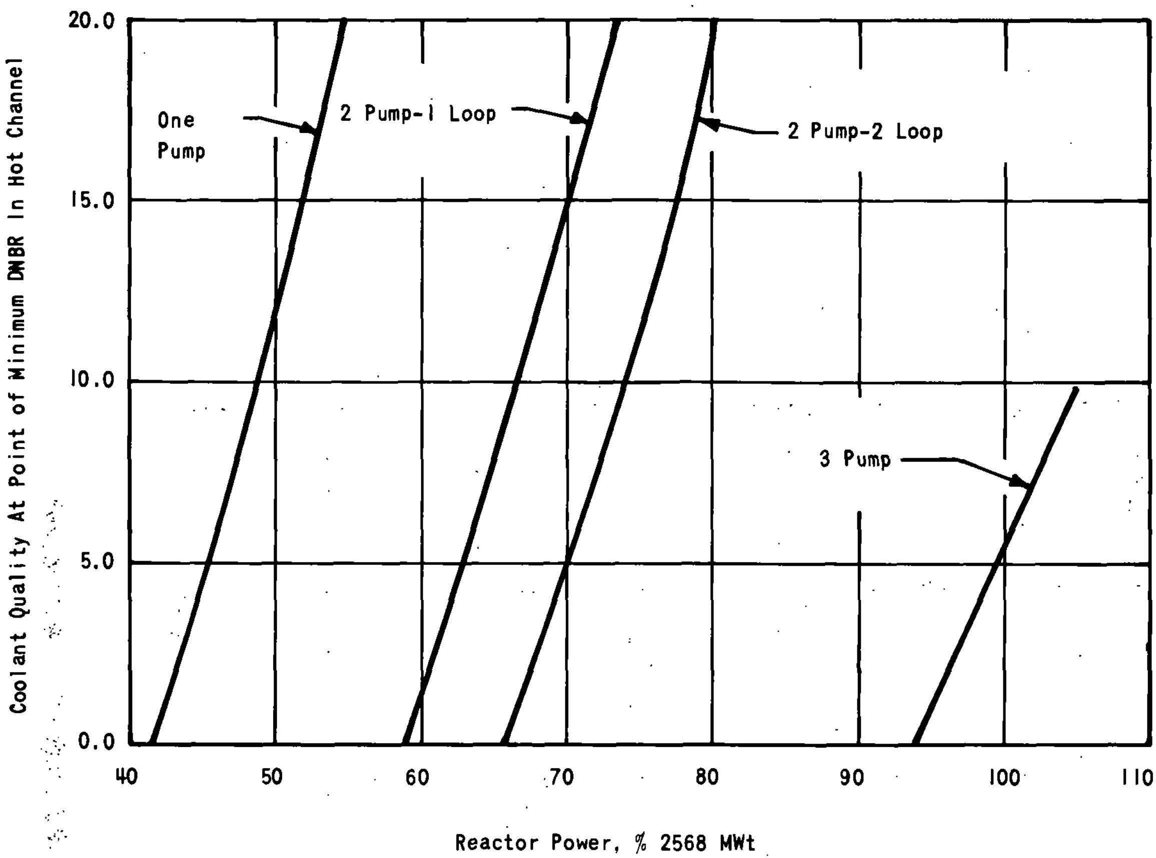
Figure 3 - 13



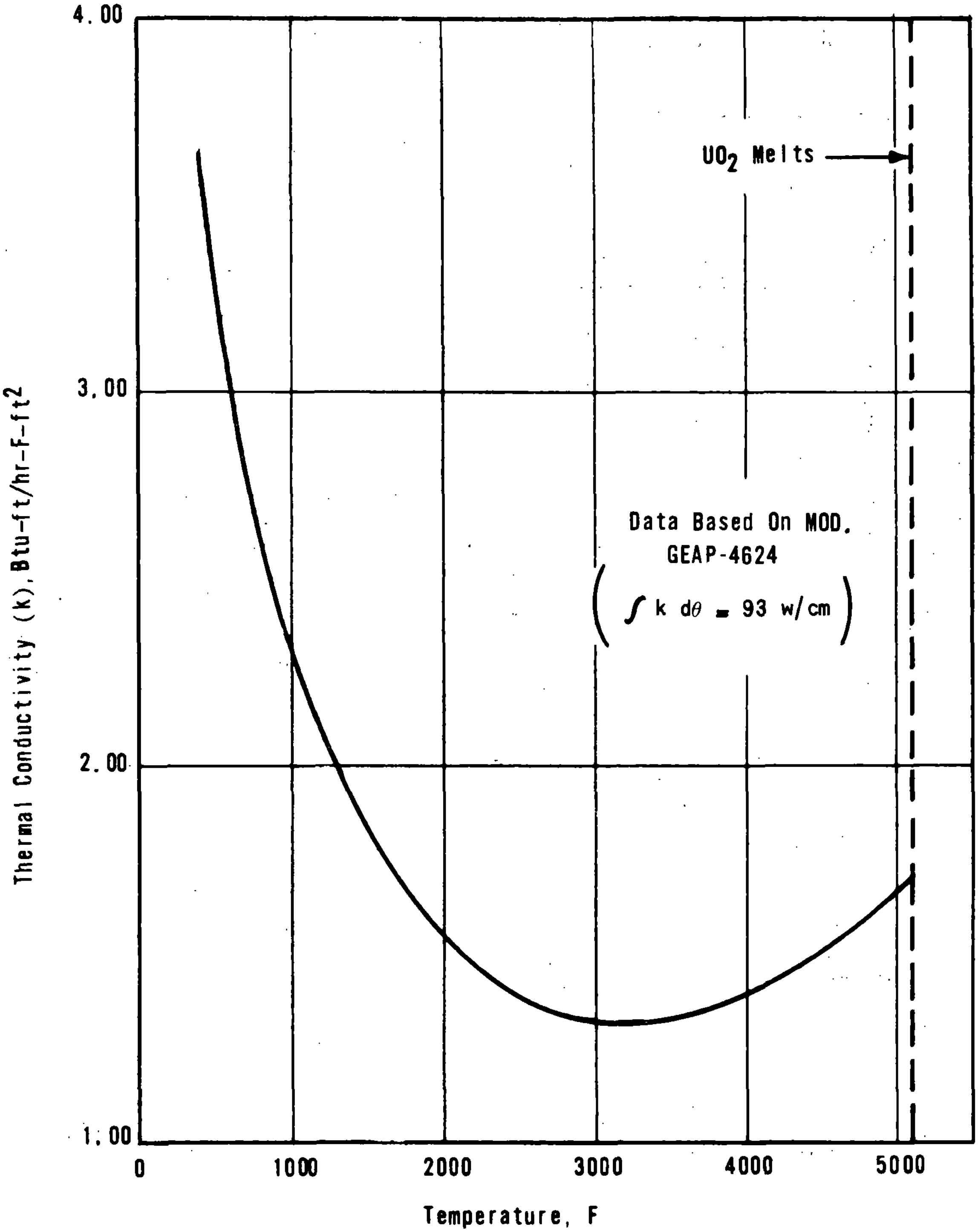
MAXIMUM HOT CHANNEL EXIT QUALITY VERSUS REACTOR POWER



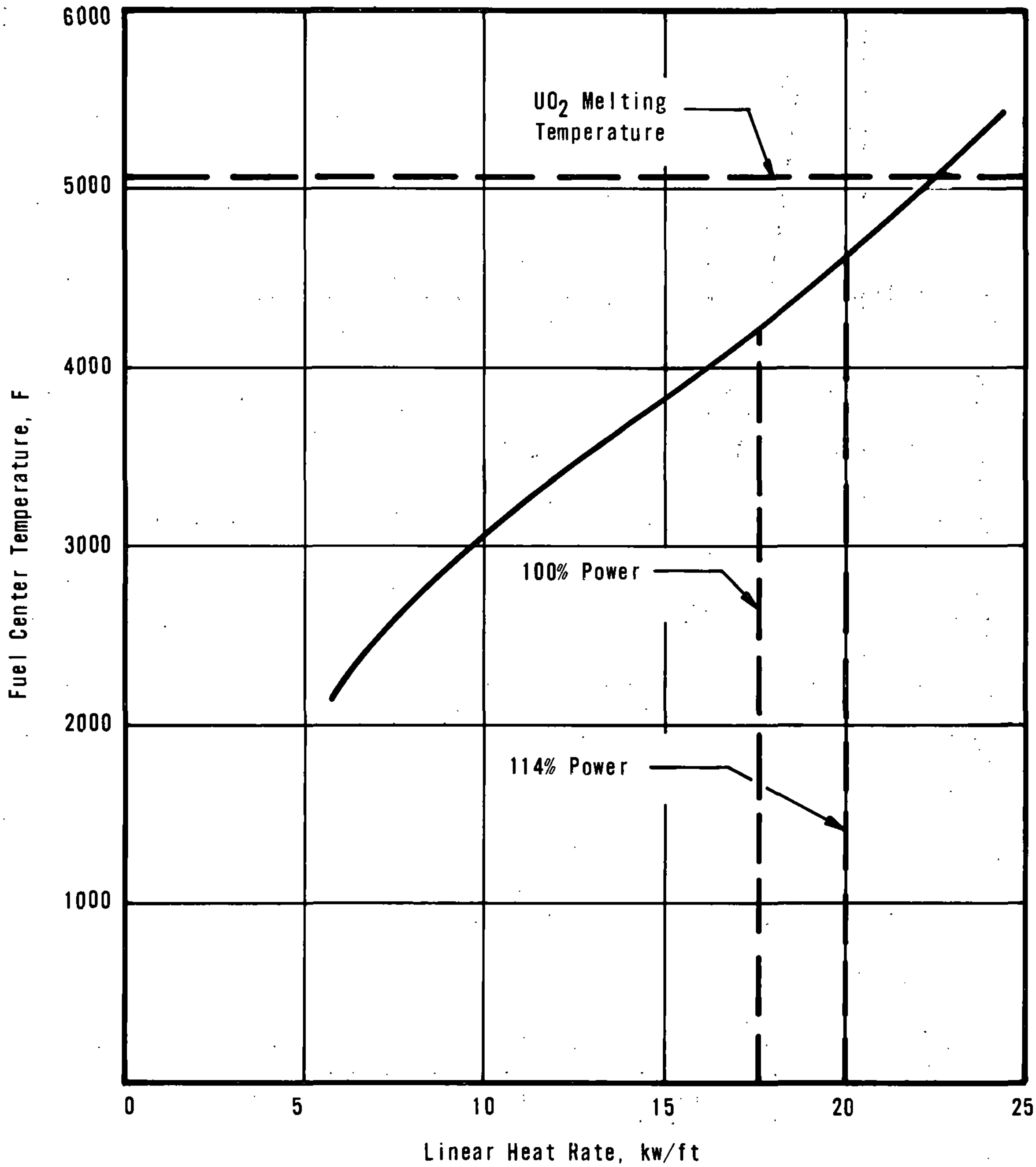
HOT CHANNEL DNB RATIO (W-3) VERSUS POWER
FOR PARTIAL PUMP OPERATION



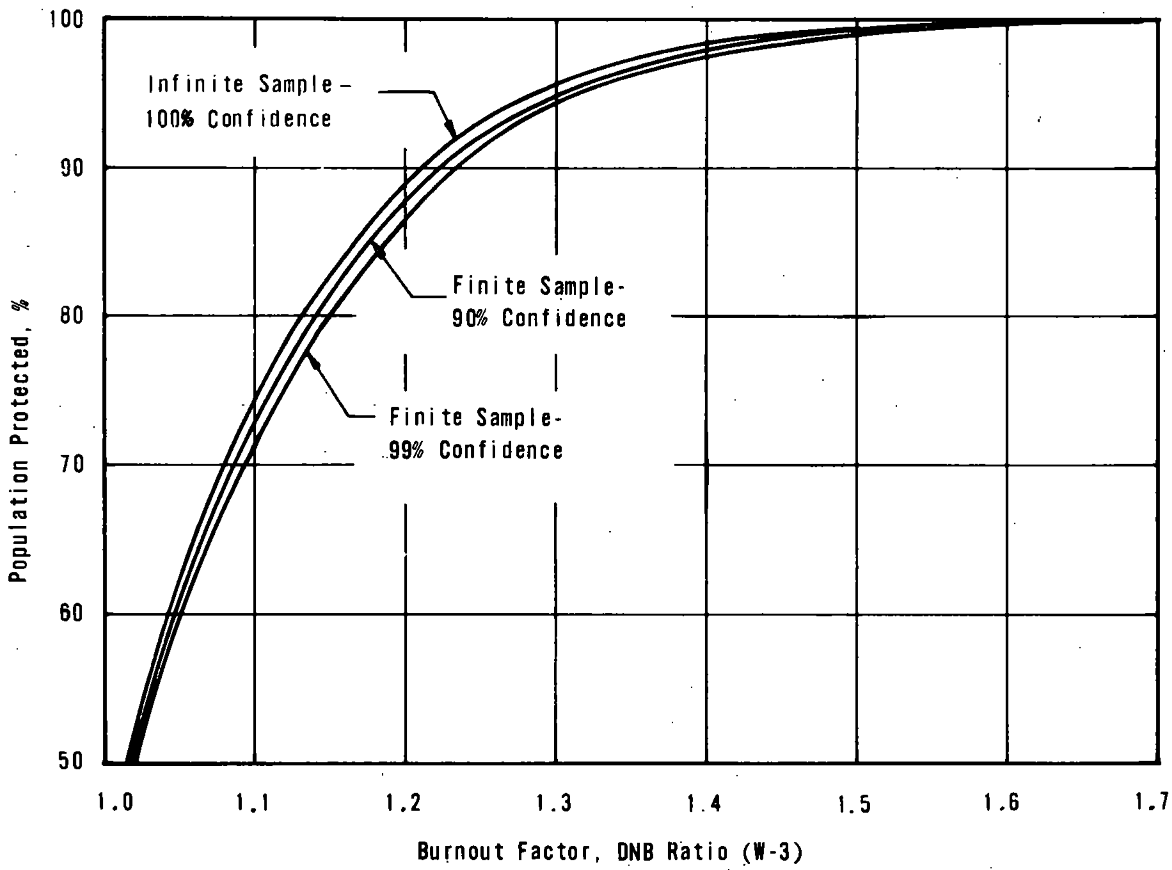
HOT CHANNEL QUALITY AT X_{DNB} VERSUS POWER FOR PARTIAL PUMP OPERATION



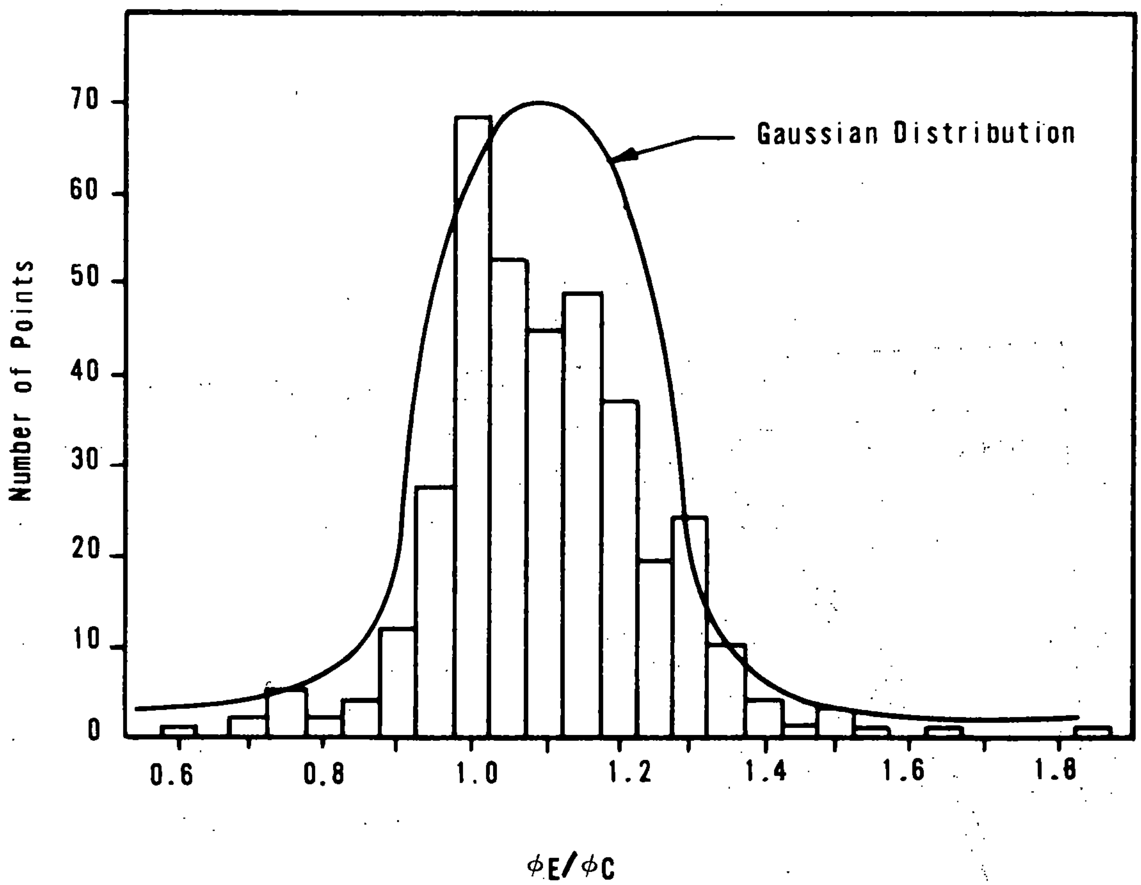
THERMAL CONDUCTIVITY OF UO_2



FUEL CENTER TEMPERATURE AT THE HOT SPOT VERSUS LINEAR POWER AT BEGINNING OF LIFE

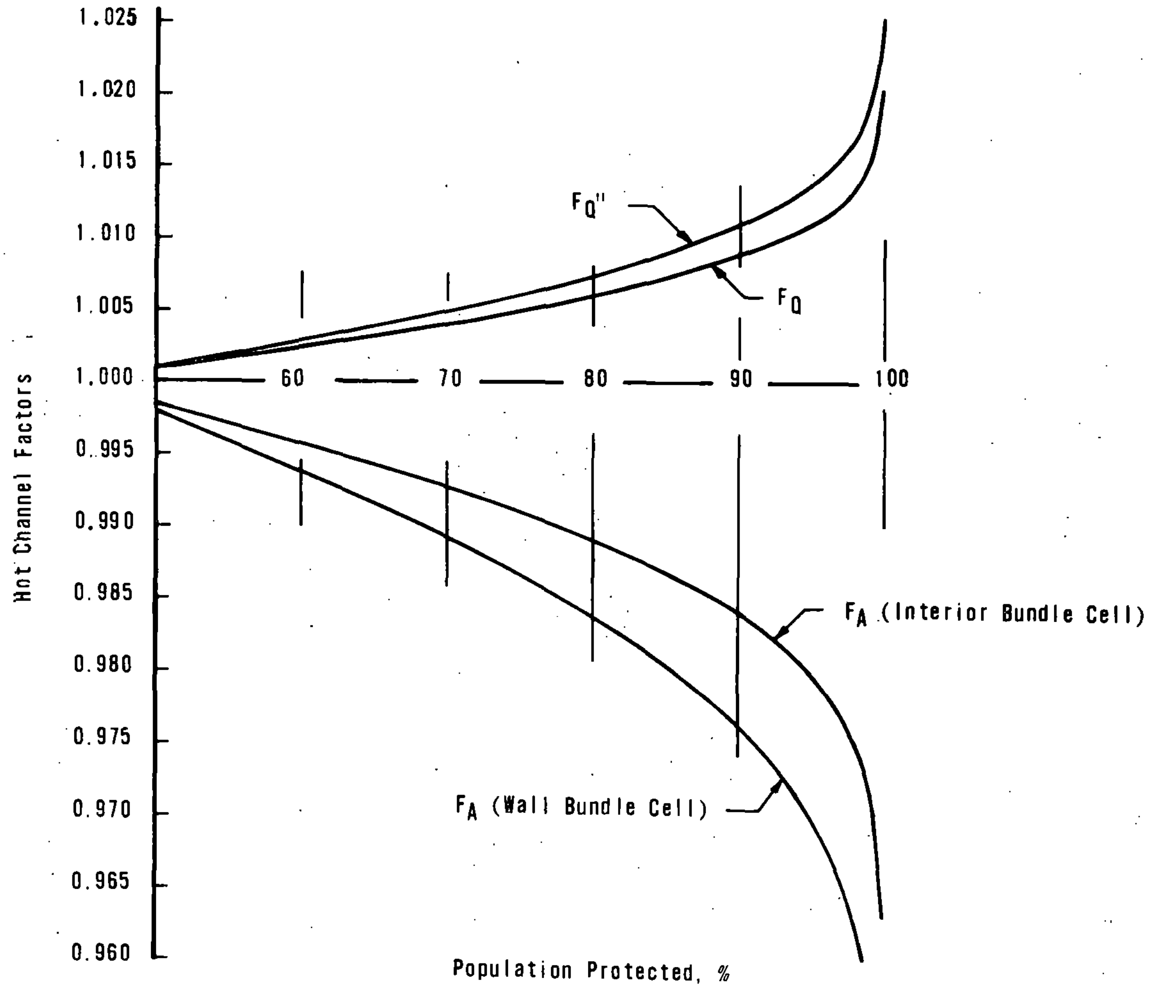


BURNOUT FACTOR (W-3) VERSUS POPULATION
FOR VARIOUS CONFIDENCE LEVELS



NUMBER OF DATA POINTS VERSUS ϕ_E/ϕ_C

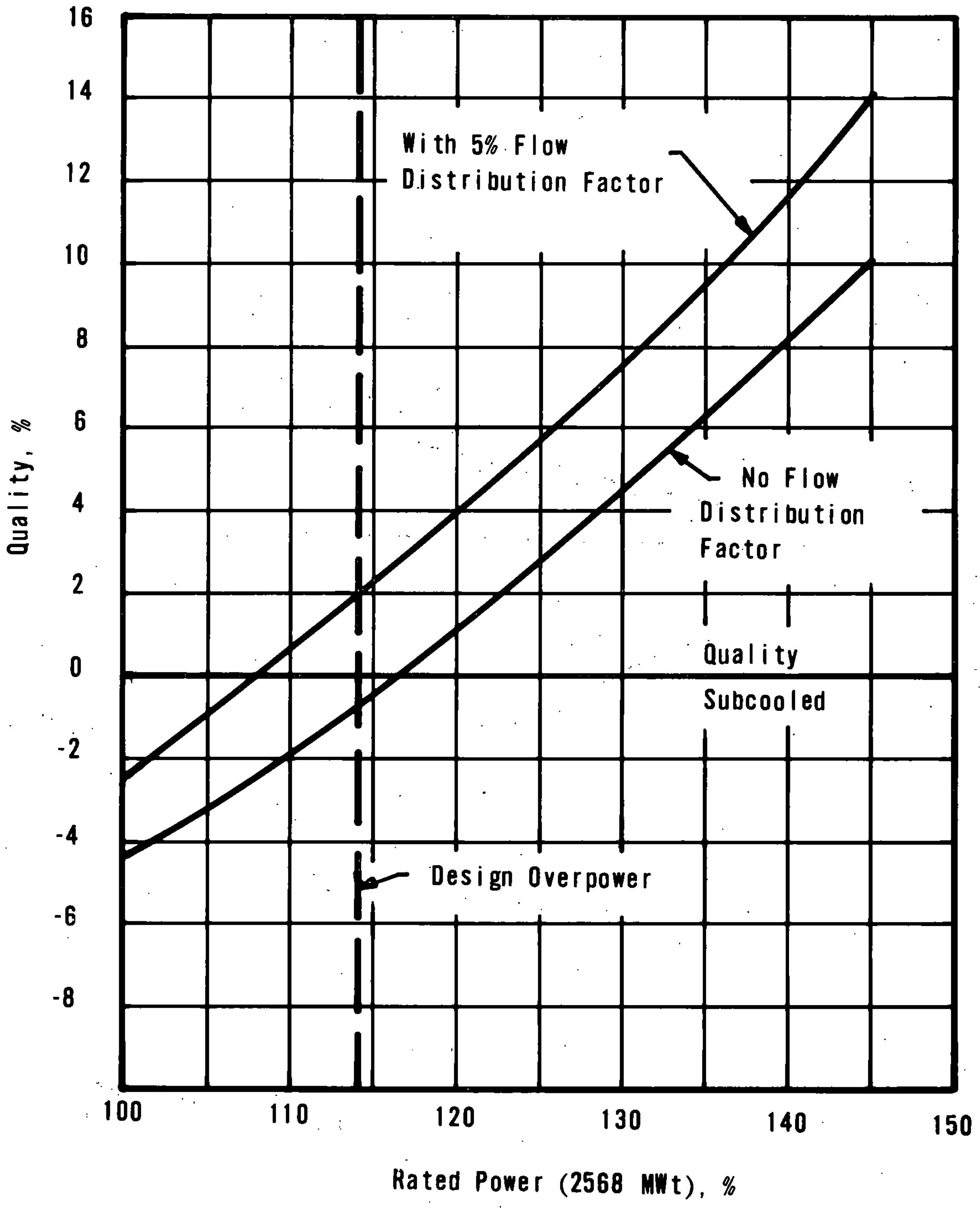
OCONEE NUCLEAR STATION
Figure 3 - 20



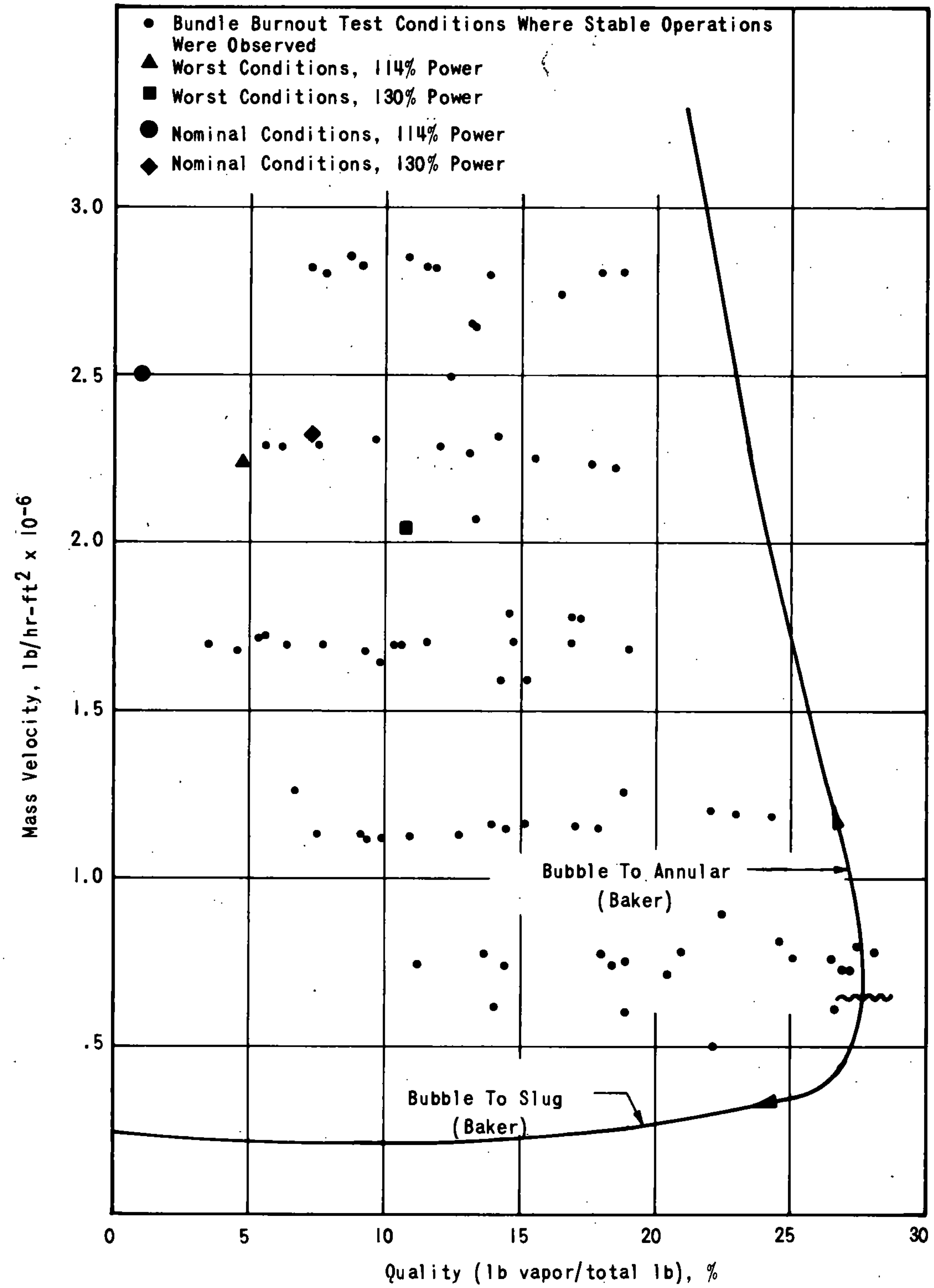
HOT CHANNEL FACTORS VERSUS
PER CENT POPULATION PROTECTED

OCONEE NUCLEAR STATION

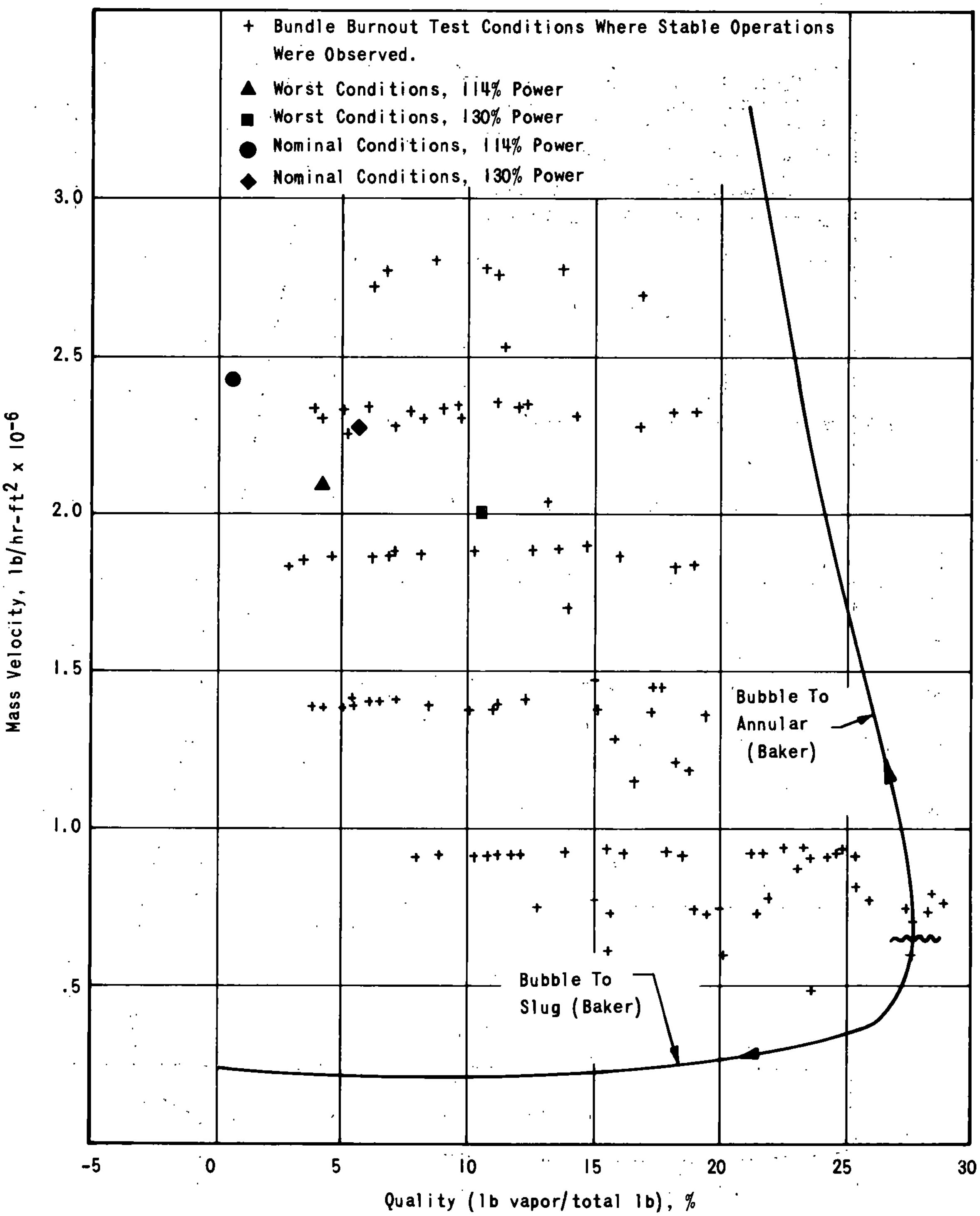
Figure 3 - 21



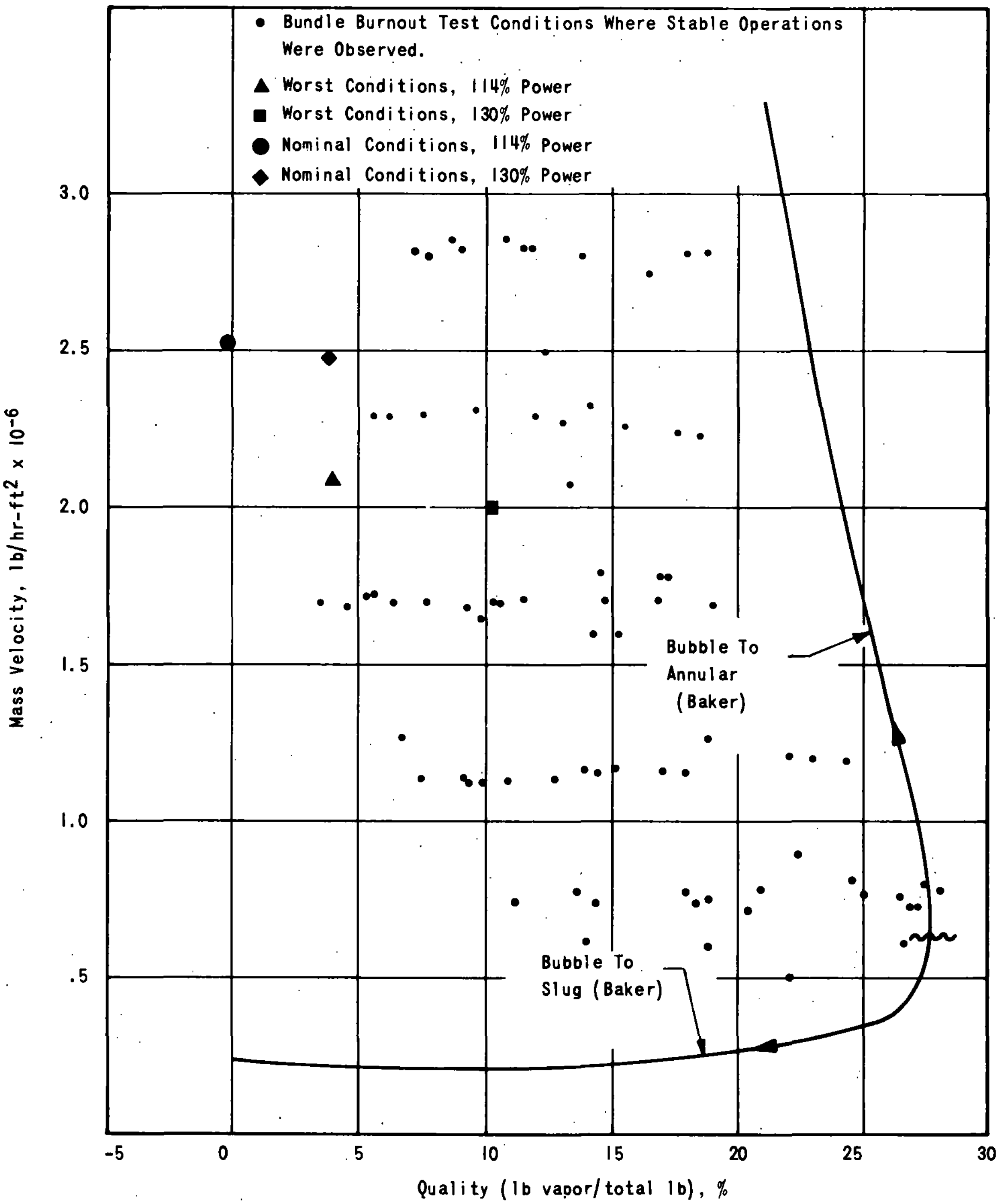
DESIGN HOT CHANNEL AND NOMINAL CHANNEL EXIT QUALITIES VERSUS REACTOR POWER (WITHOUT ENGINEERING HOT CHANNEL FACTORS)



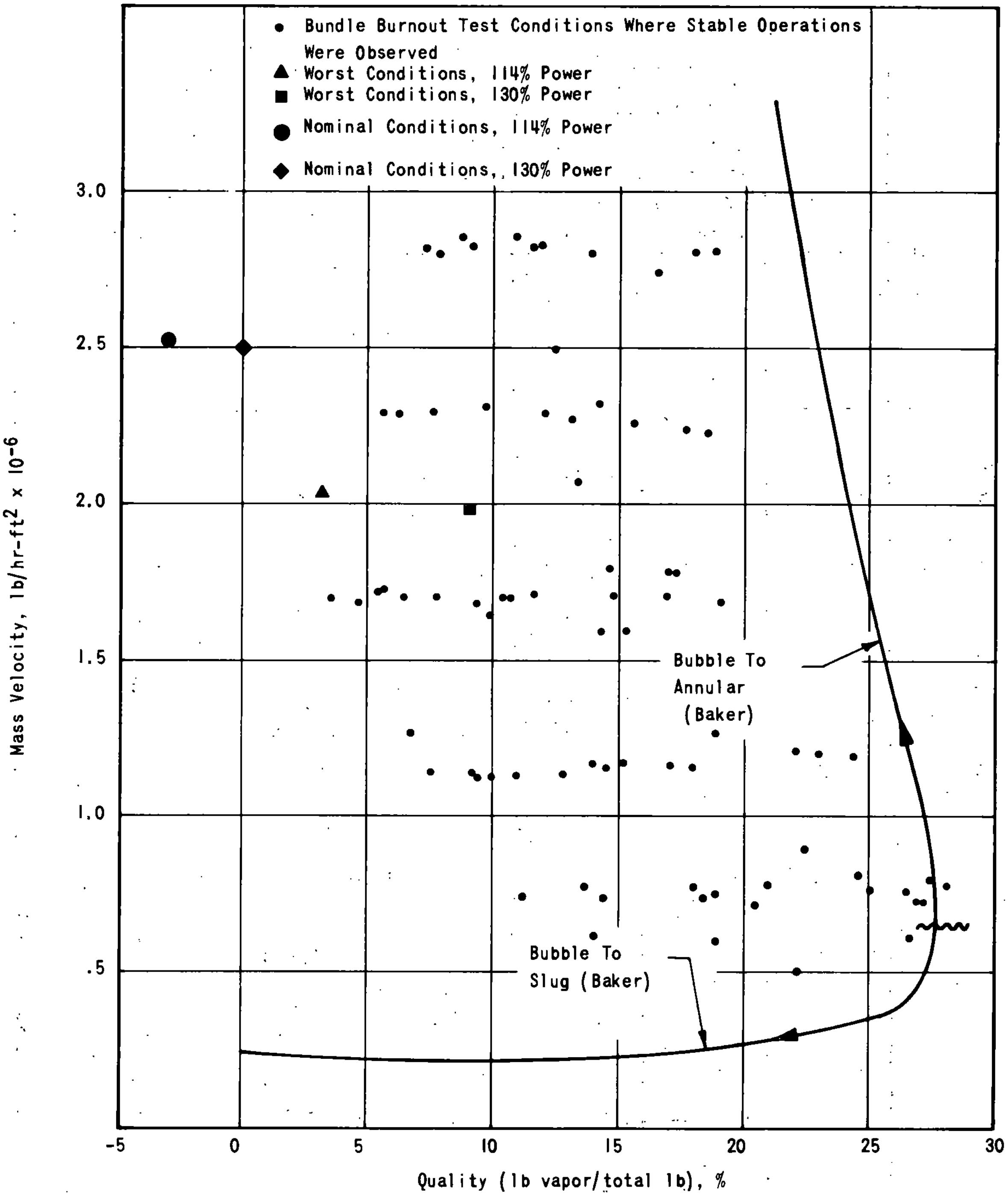
FLOW REGIME MAP FOR THE HOT UNIT CELL



FLOW REGIME MAP FOR THE HOT CONTROL ROD CELL



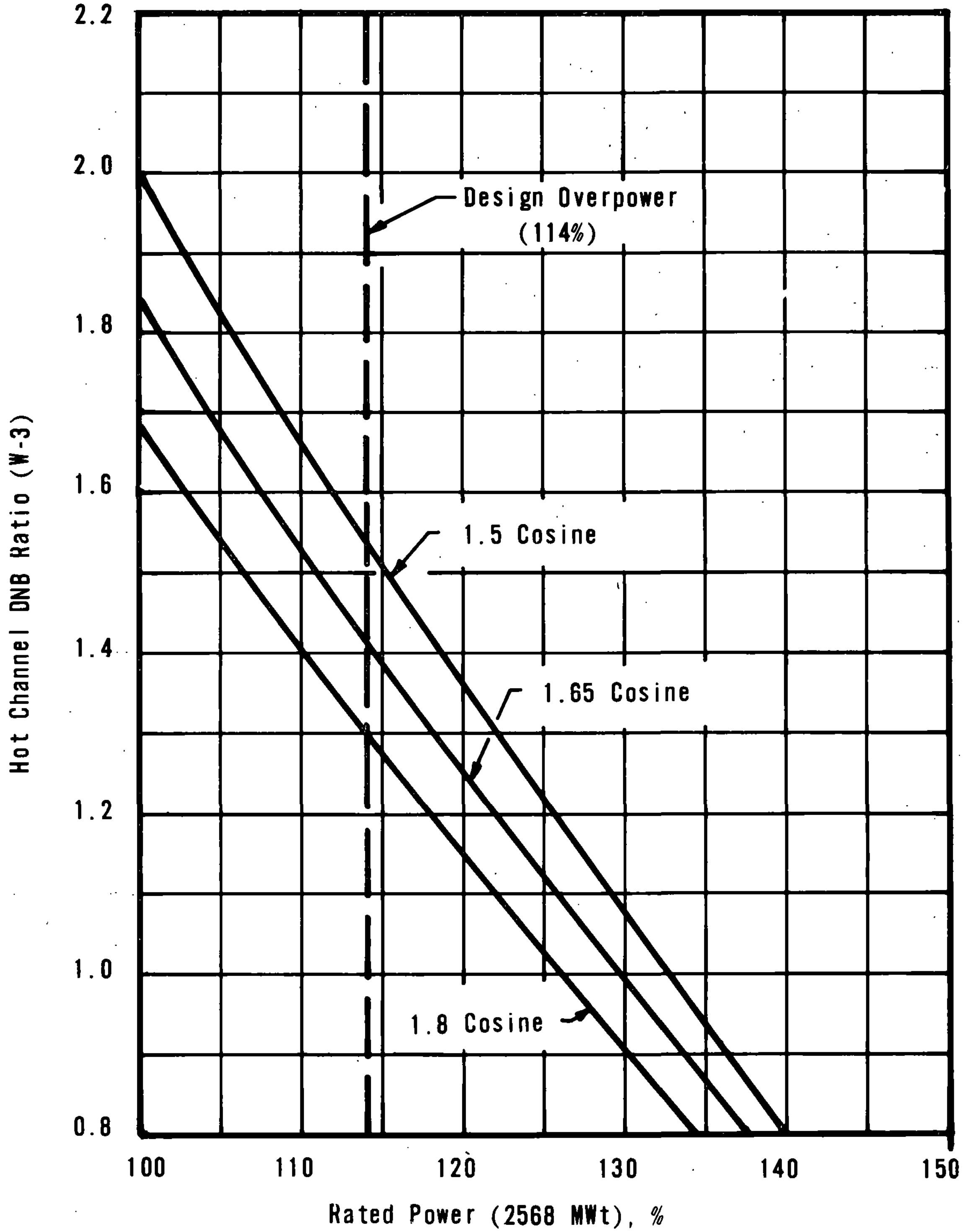
FLOW REGIME MAP FOR THE HOT WALL CELL



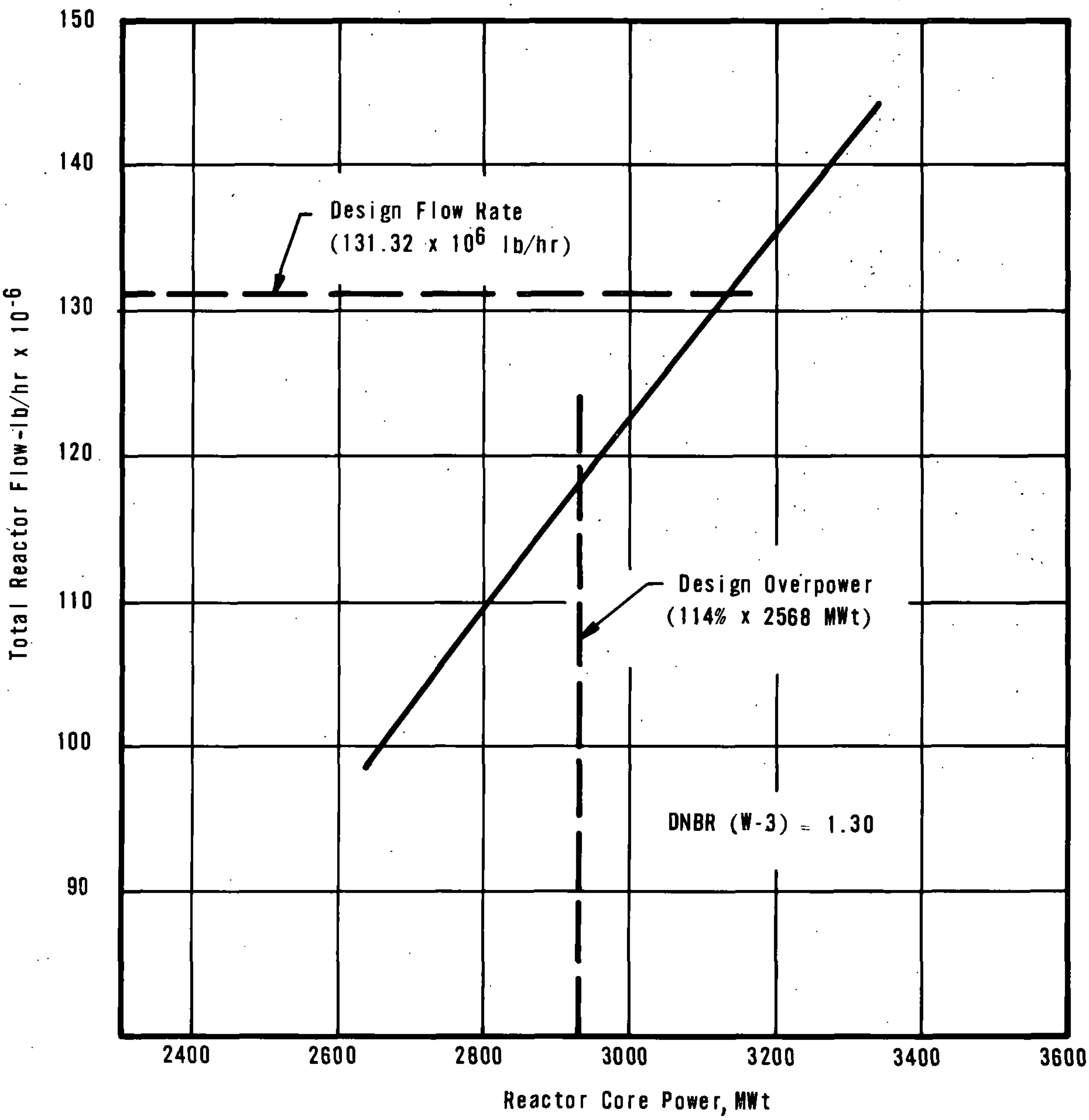
FLOW REGIME MAP FOR HOT CORNER CELL

OCONEE NUCLEAR STATION

Figure 3 - 26



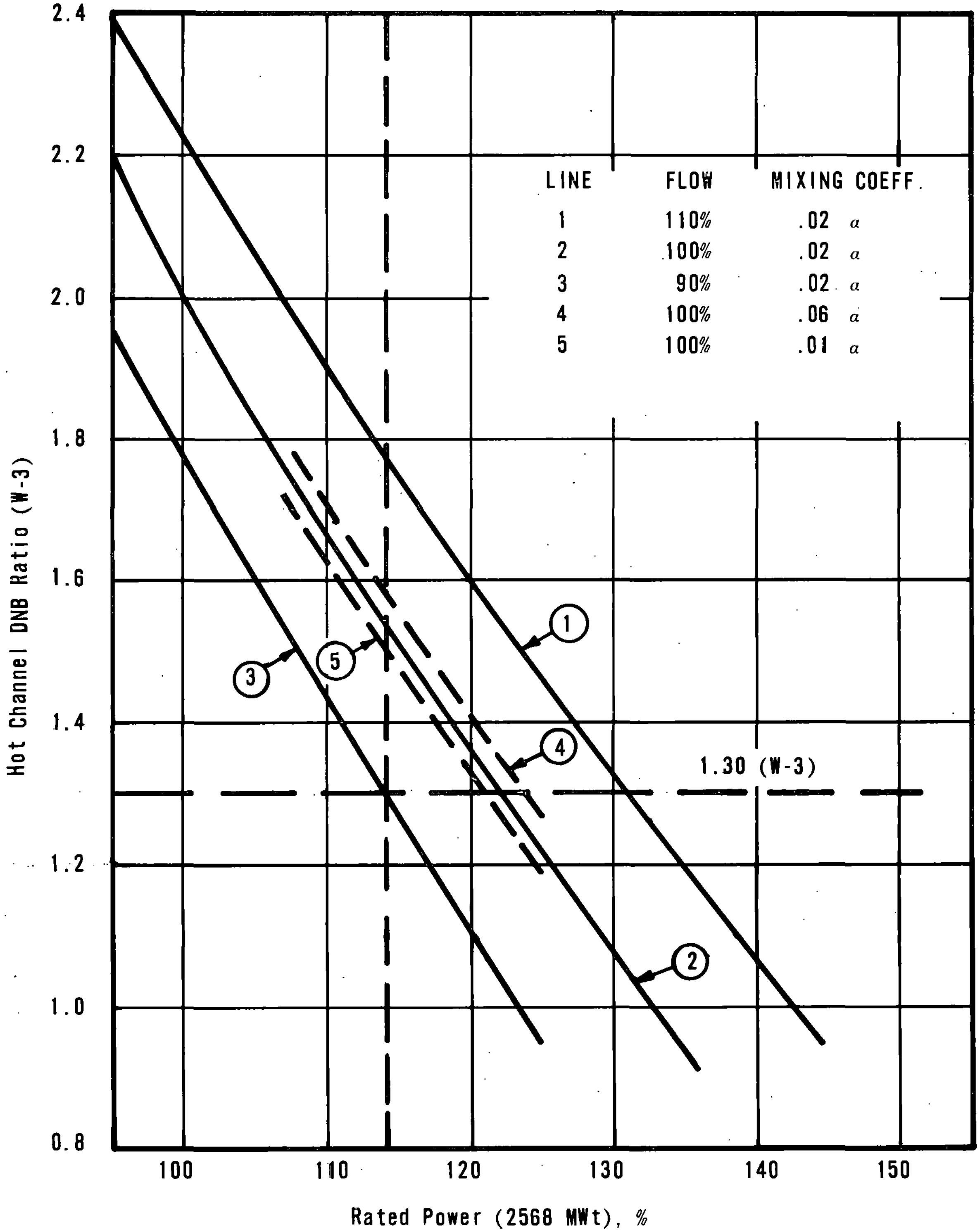
HOT CHANNEL DNB RATIO (W-3) VERSUS
POWER FOR VARIOUS AXIAL FLUX SHAPES



REACTOR COOLANT SYSTEM
FLOW VERSUS POWER

OCONEE NUCLEAR STATION

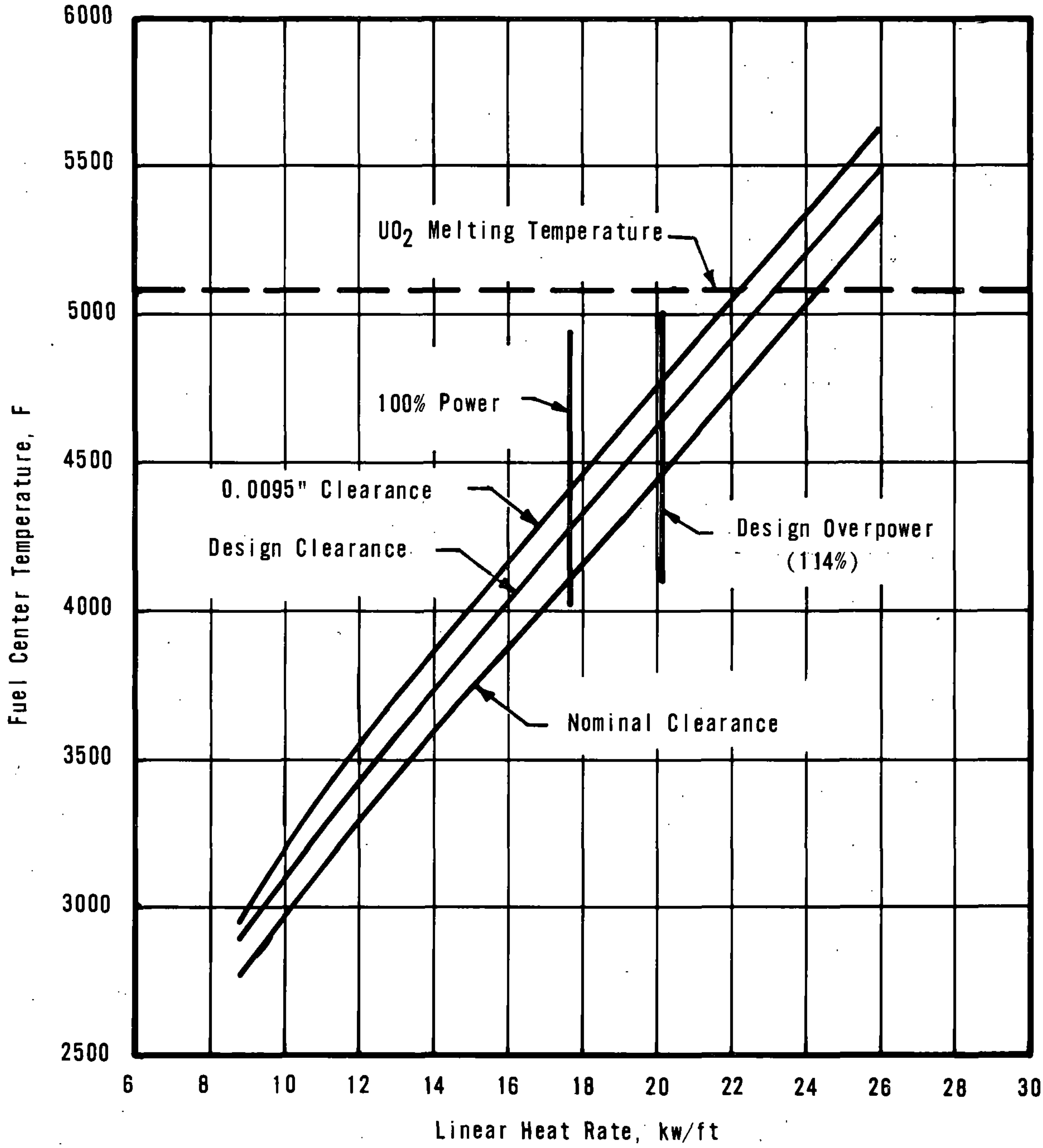
Figure 3 - 28



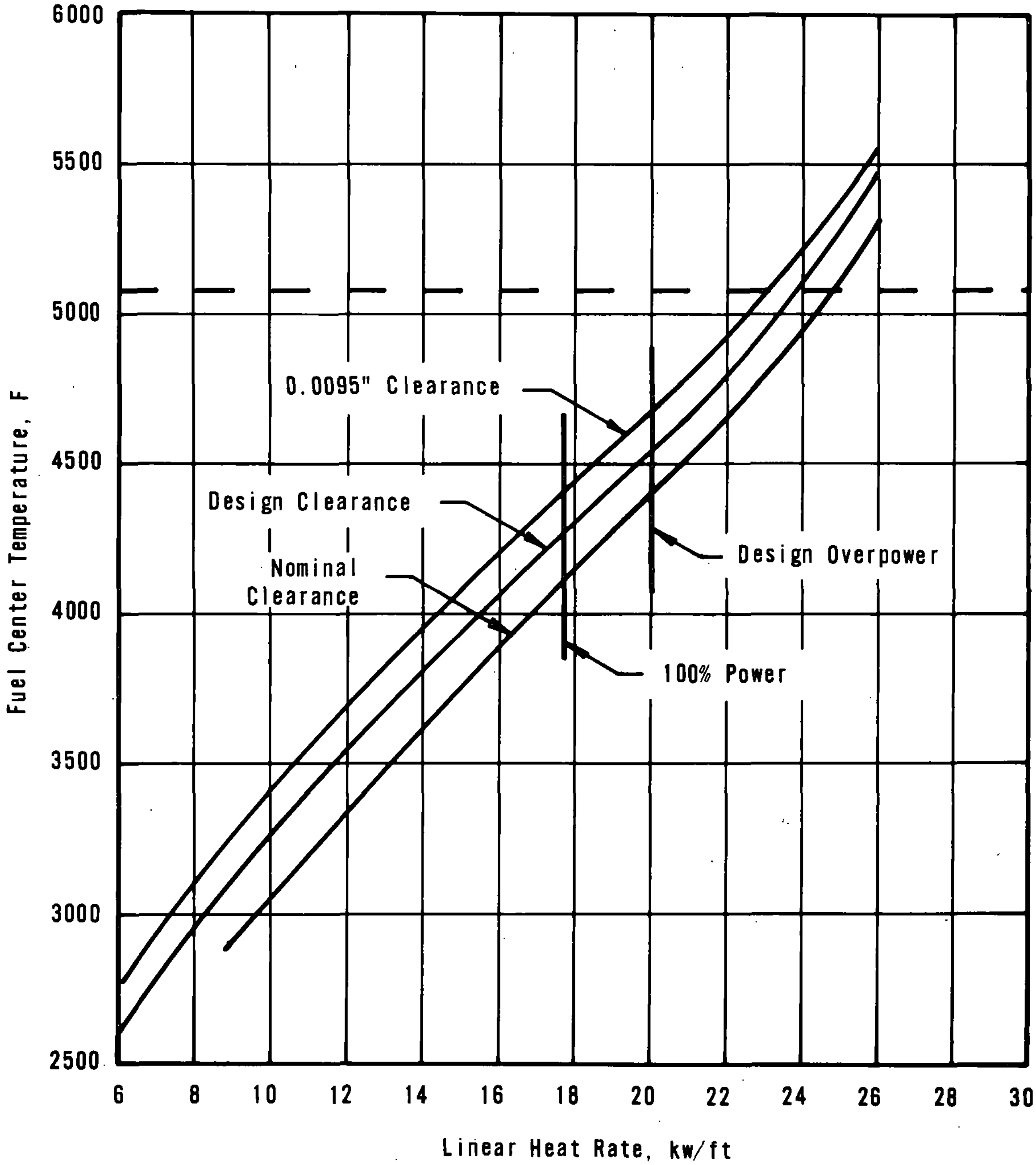
HOT CHANNEL DNB RATIO (W-3) VERSUS POWER WITH REACTOR SYSTEM FLOW AND ENERGY MIXING AS PARAMETERS

OCONEE NUCLEAR STATION

Figure 3 - 29

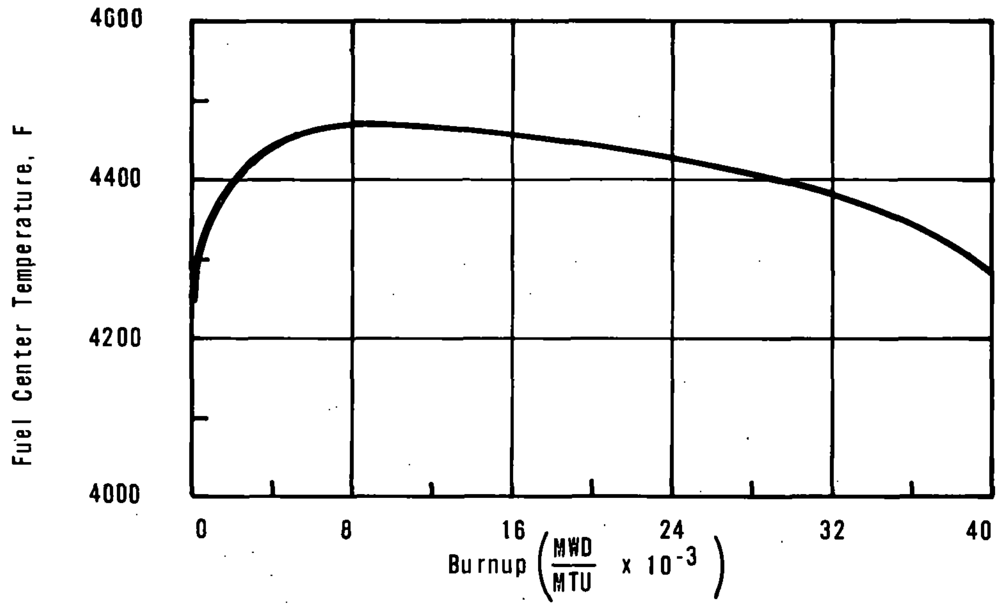


FUEL CENTER TEMPERATURE FOR
BEGINNING OF LIFE CONDITIONS

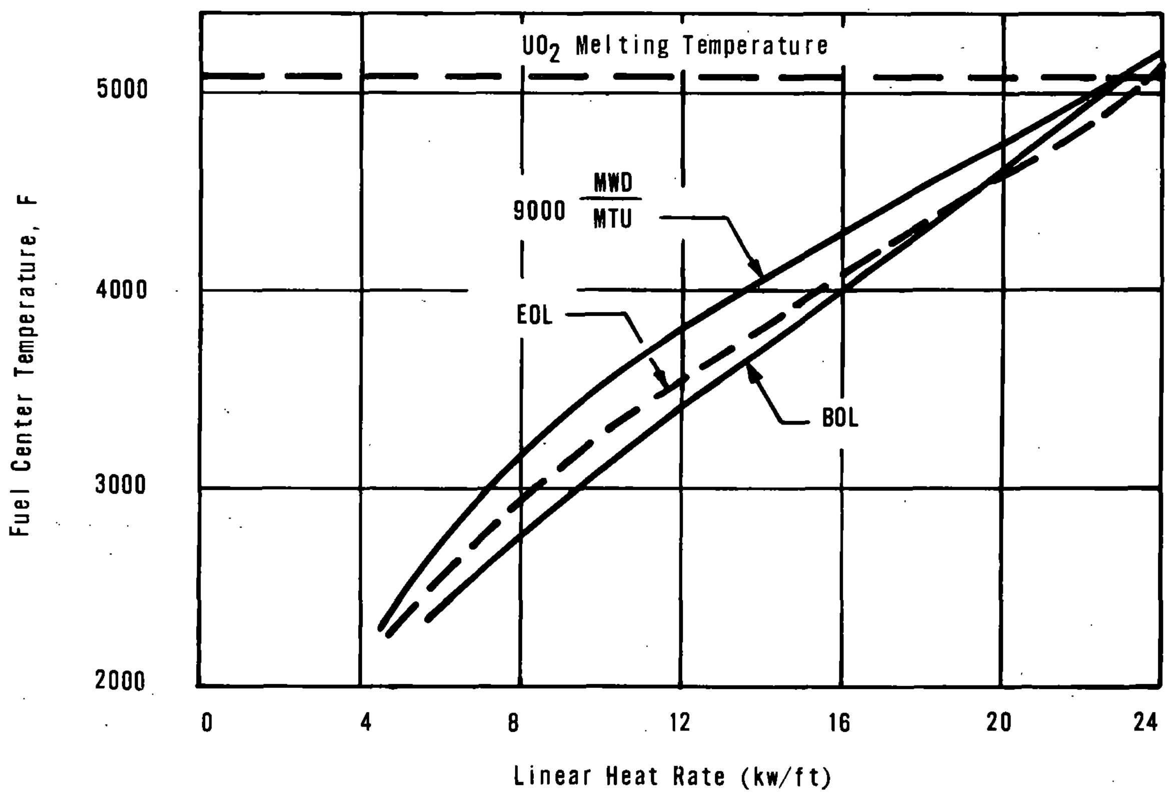


FUEL CENTER TEMPERATURE FOR
END OF LIFE CONDITIONS

OCONEE NUCLEAR STATION
Figure 3 - 31

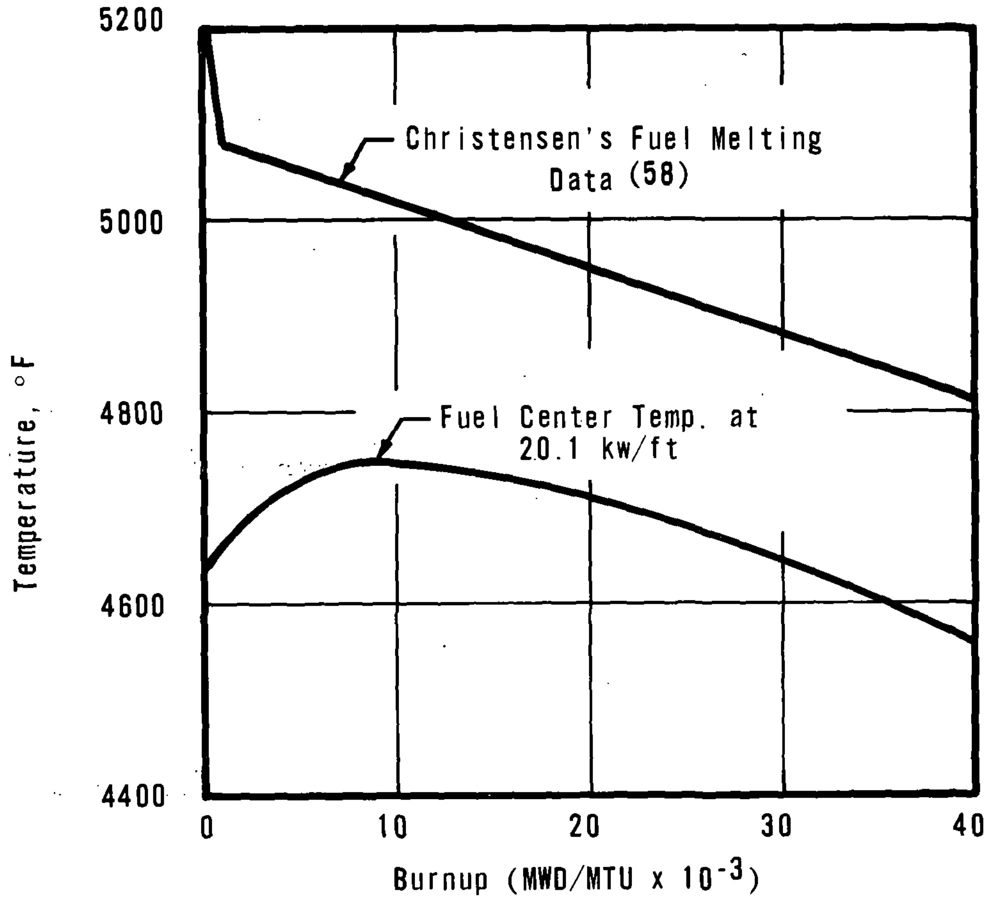


Center Fuel Temperature Versus
Burnup at 17.63 KW/FT

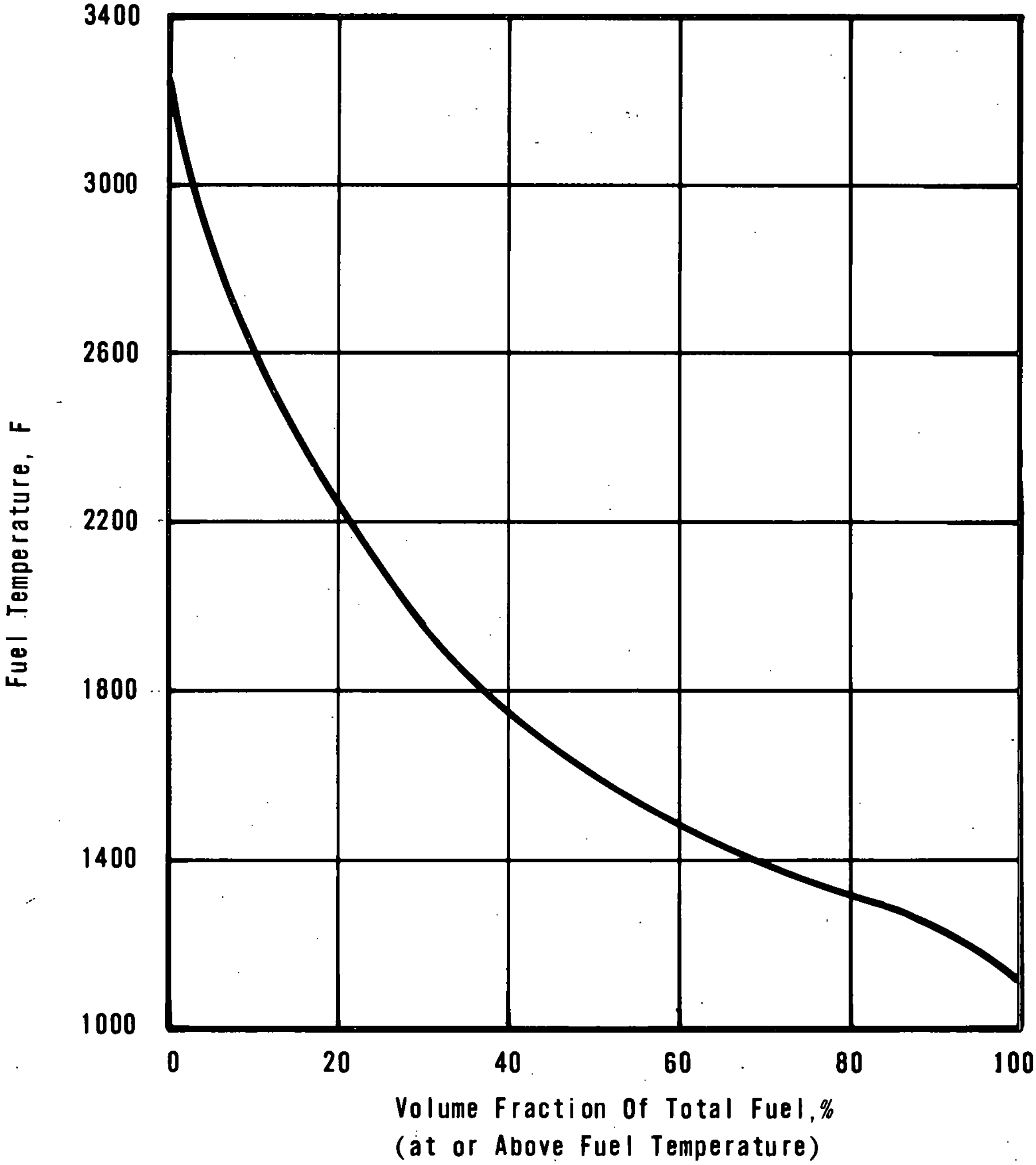


Center Fuel Temperature Versus Linear Heat
For Various Burnup Conditions

BURNUP EFFECTS ON FUEL CENTER
TEMPERATURE



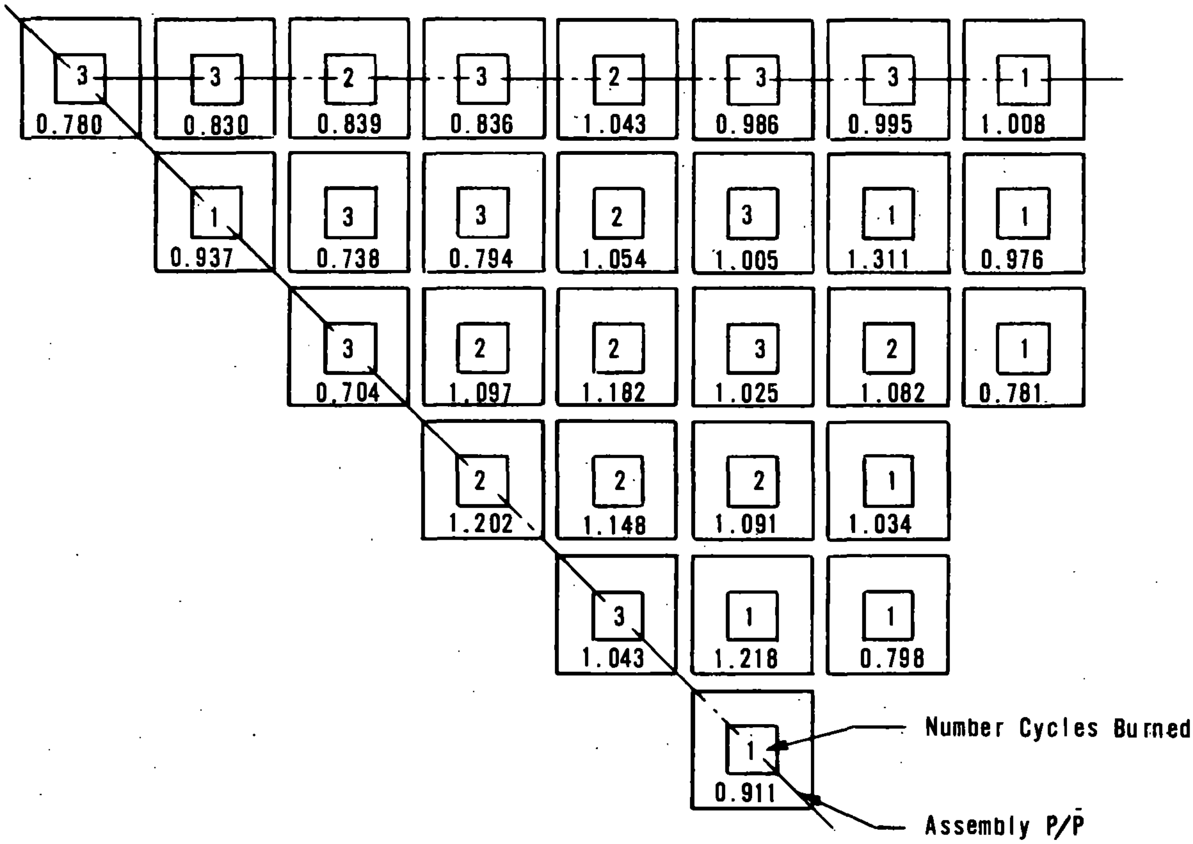
COMPARISON OF FUEL MELTING TEMPERATURE
TO MAXIMUM FUEL TEMPERATURE WITH BURNUP



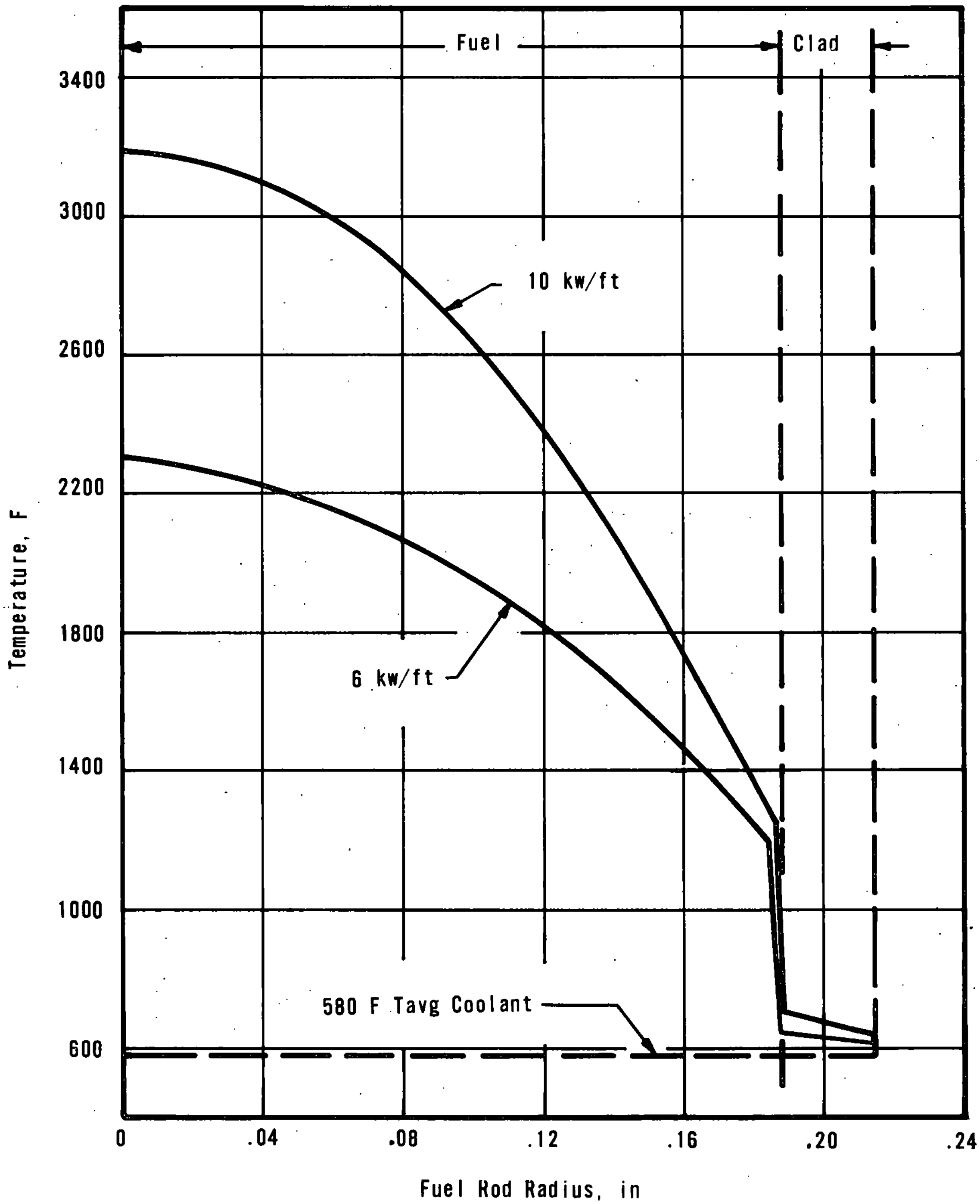
FUEL TEMPERATURE VERSUS TOTAL
FUEL VOLUME FRACTION FOR
EQUILIBRIUM CYCLE AT END OF LIFE

OCONEE NUCLEAR STATION

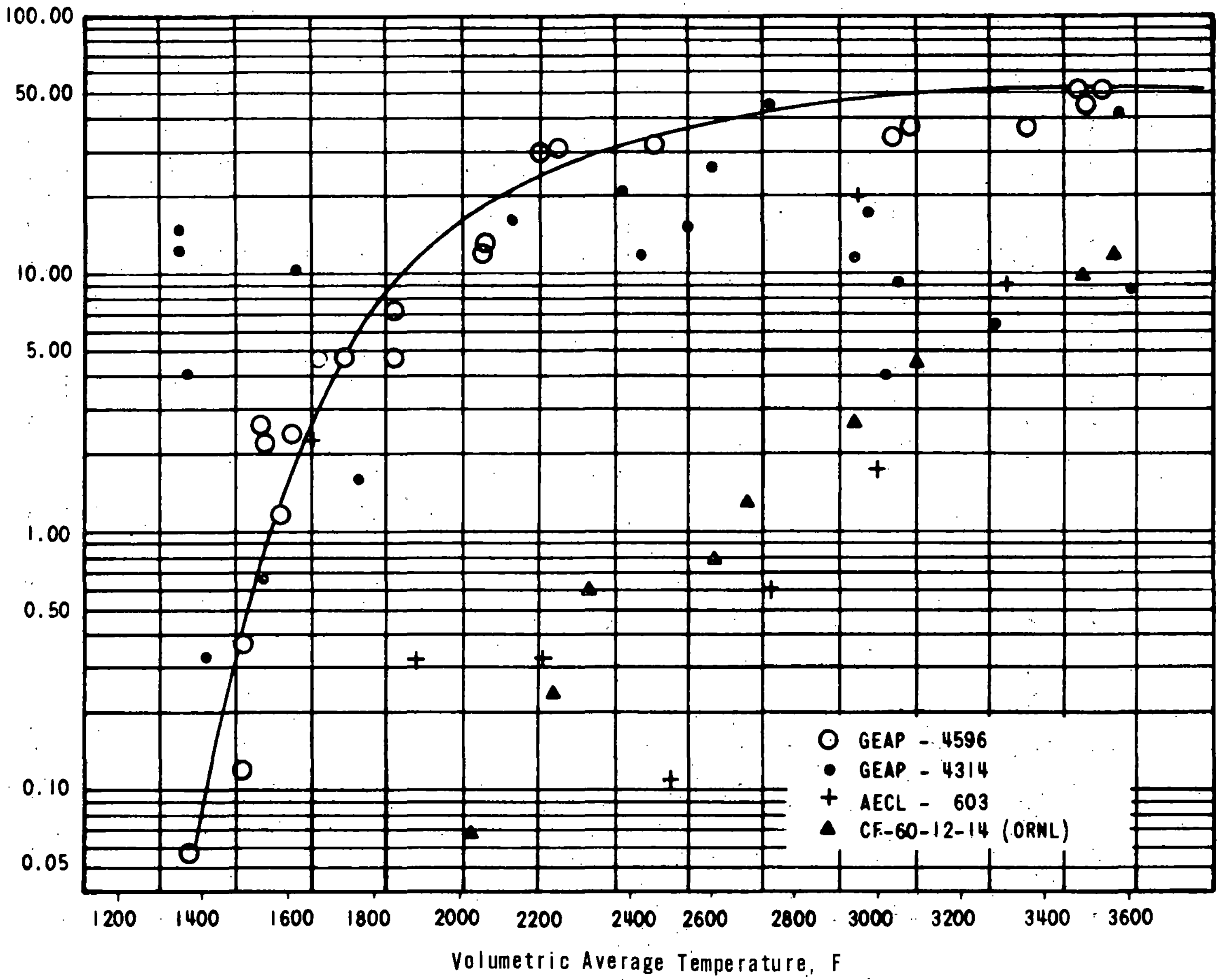
Figure 3 - 33



TYPICAL REACTOR FUEL ASSEMBLY POWER
DISTRIBUTION AT END-OF-LIFE EQUILIBRIUM
CYCLE CONDITIONS FOR 1/8 CORE



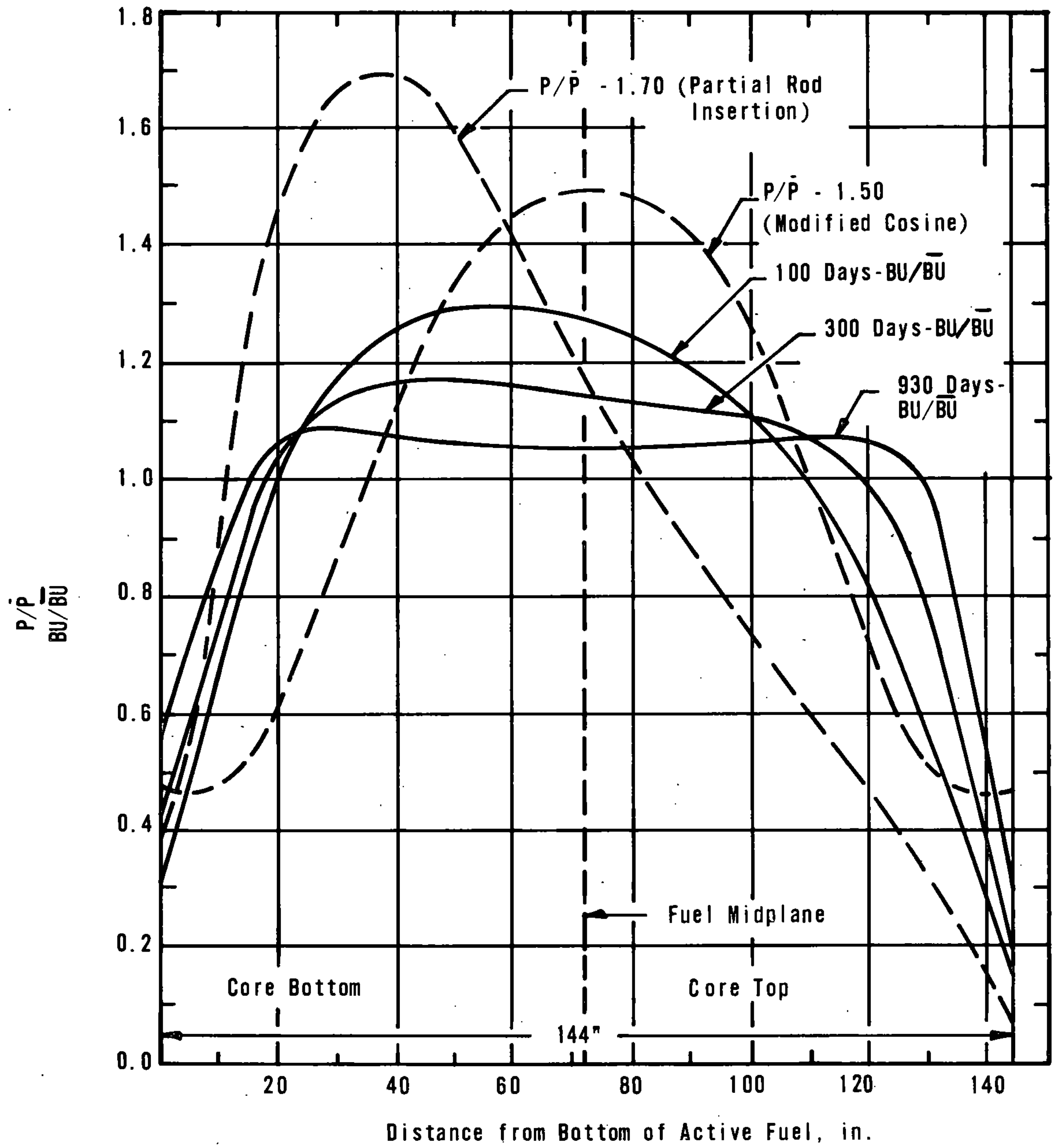
FUEL ROD TEMPERATURE PROFILES AT 6 AND 10 KW/FT



PER CENT FISSION GAS RELEASED AS A FUNCTION OF THE AVERAGE TEMPERATURE OF THE UO_2 FUEL

OCONEE NUCLEAR STATION

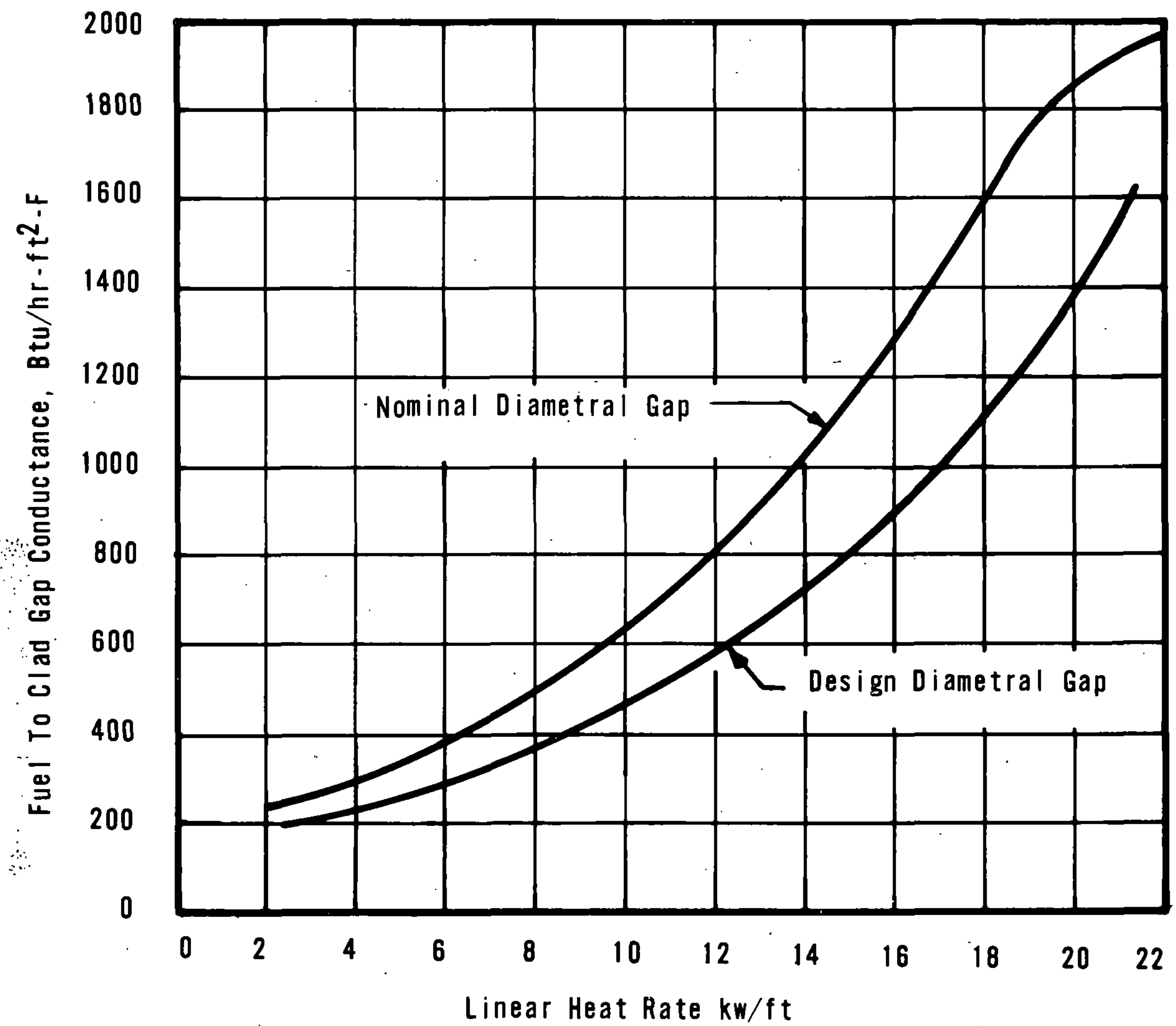
Figure 3 - 36



AXIAL LOCAL-TO-AVERAGE BURNUP AND INSTANTANEOUS POWER COMPARISONS

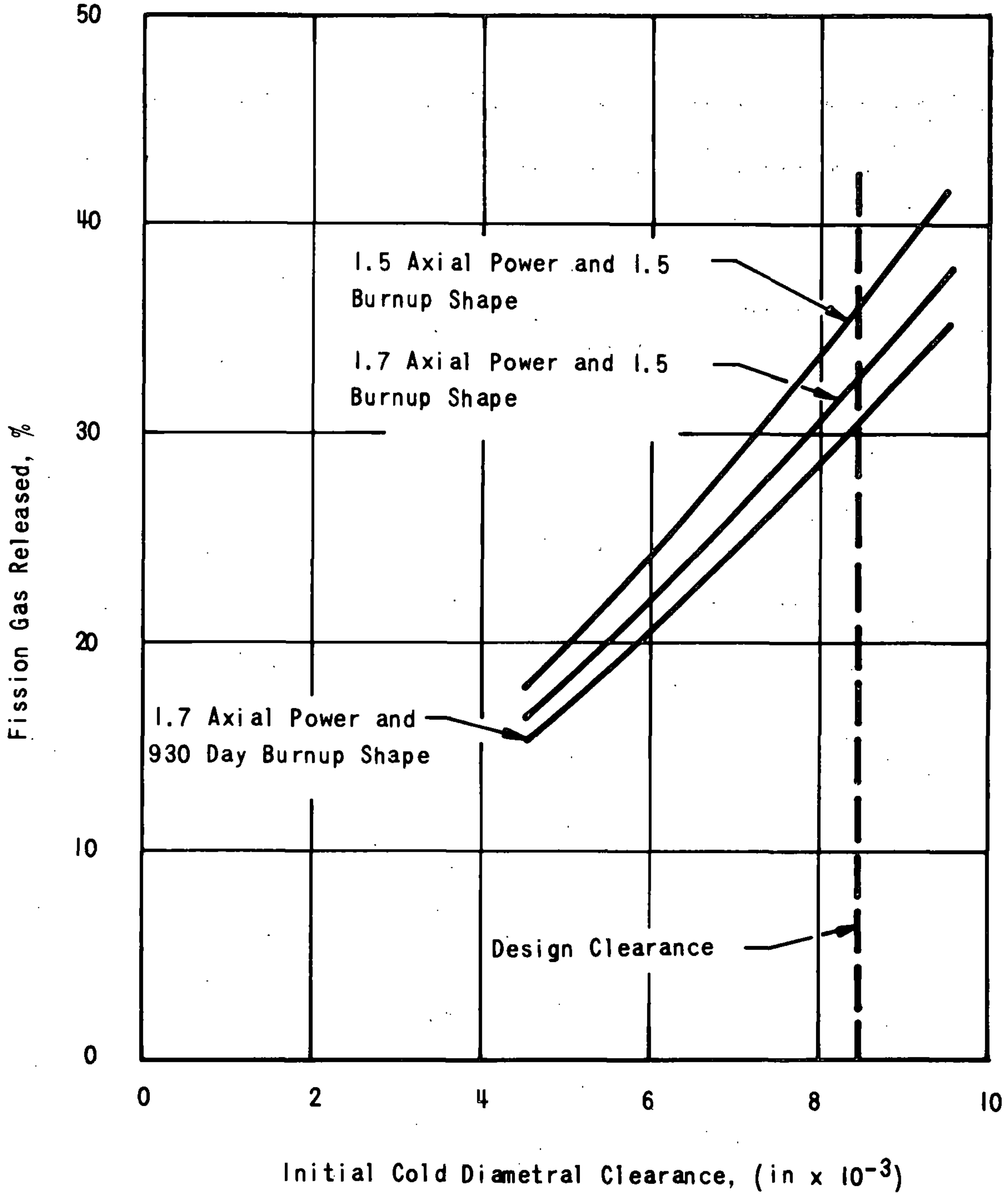
OCONEE NUCLEAR STATION

Figure 3 - 37



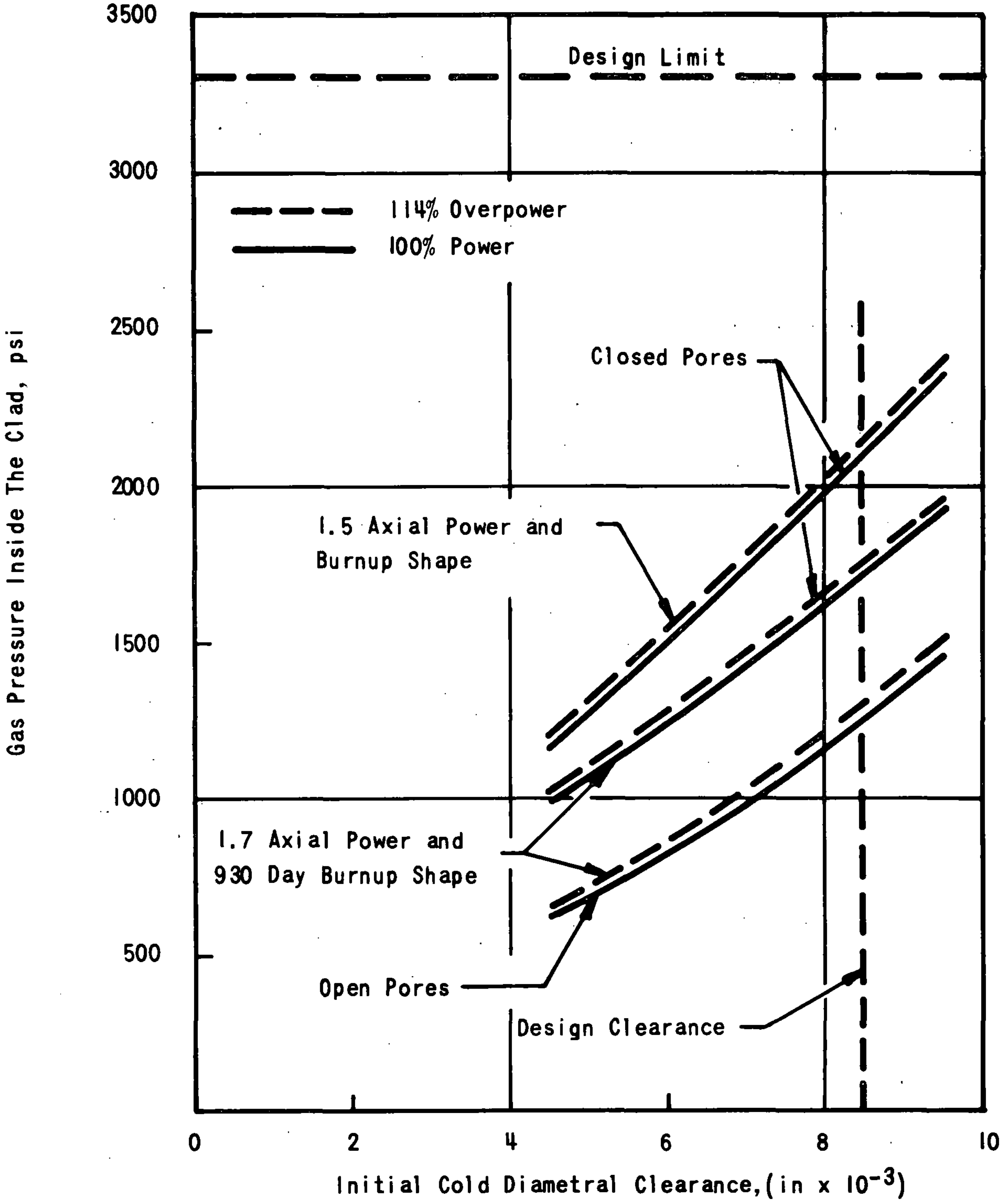
FUEL-TO-CLAD GAP CONDUCTANCE FOR
END OF LIFE CONDITIONS

Figure 3-38



FISSION GAS RELEASE FOR 1.5 AND
1.7 MAX/AVG AXIAL POWER SHAPES

OCONEE NUCLEAR STATION
Figure 3 - 39
Rev. 1 9/15/69

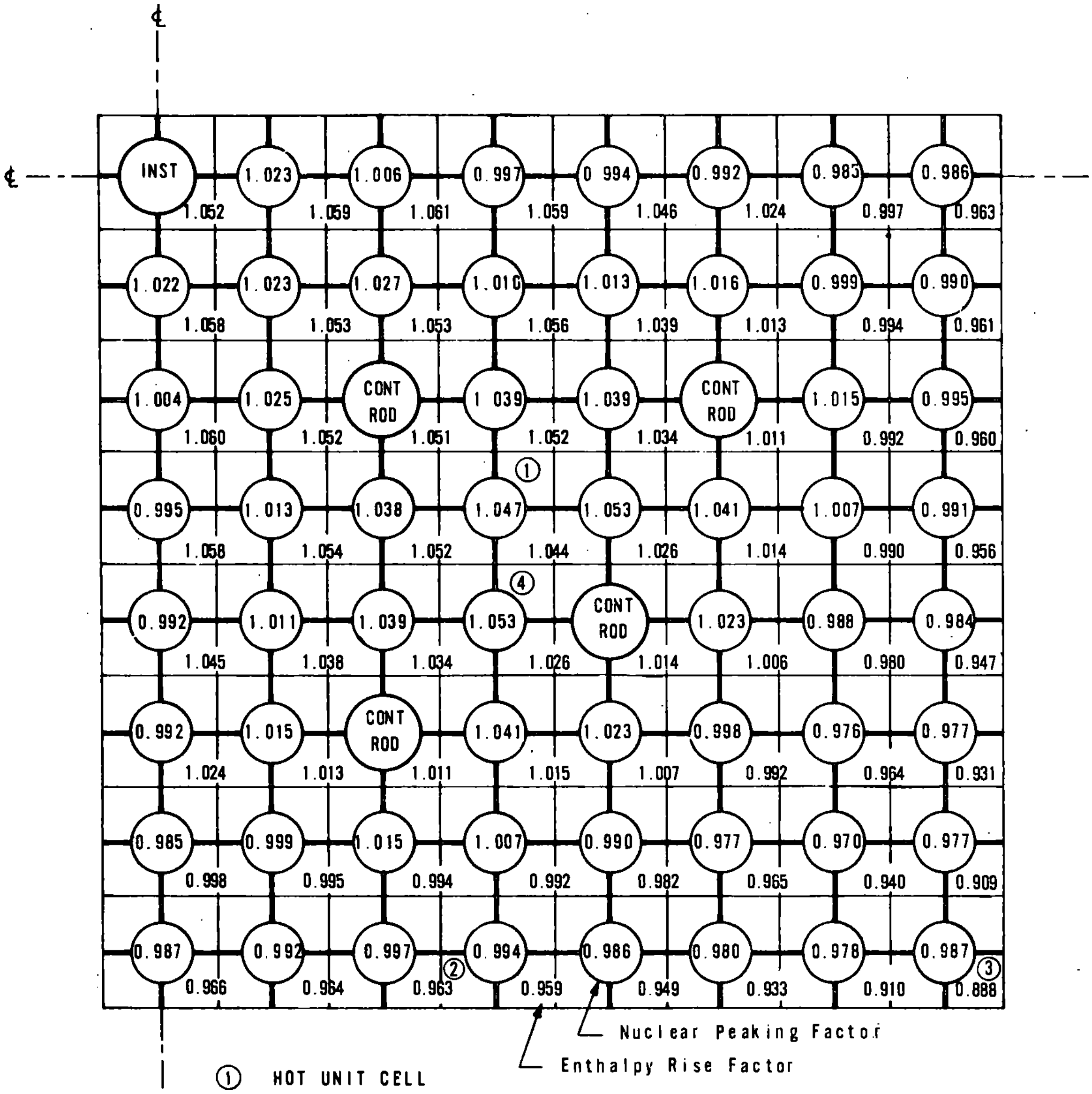


GAS PRESSURE INSIDE THE CLAD FOR
VARIOUS AXIAL POWER AND BURNUP SHAPES

OCONEE NUCLEAR STATION

Figure 3 - 40

Rev.1 9/15/69



① HOT UNIT CELL

② HOT WALL CELL

③ HOT CORNER CELL

④ HOT CONTROL ROD CELL

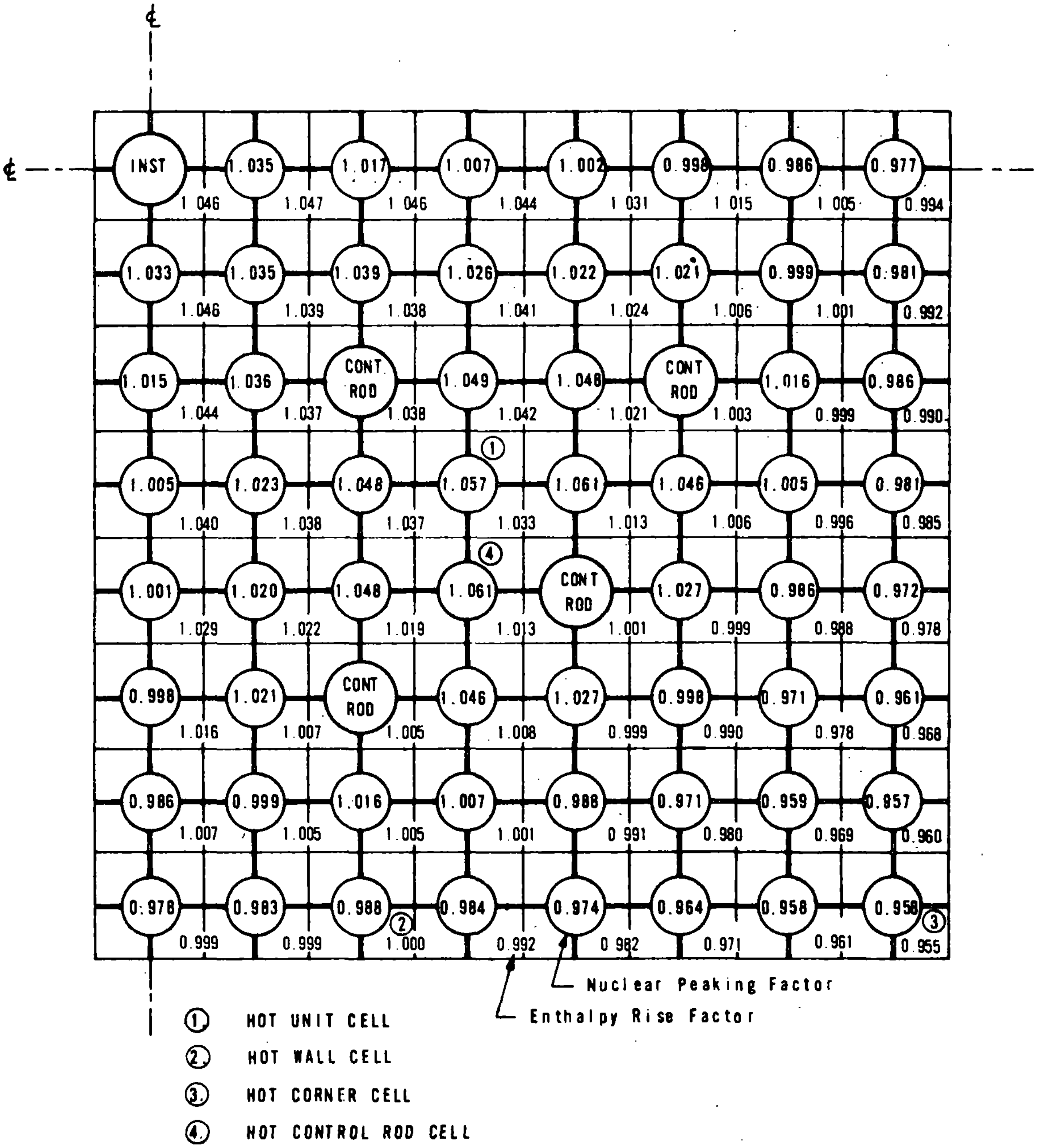
NOMINAL FUEL ROD POWER PEAKS AND CELL EXIT ENTHALPY RISE RATIOS

DUKE POWER CITIZENSHIP SERVICE

OCONEE NUCLEAR STATION

Figure 3 - 41

Rev. 3 3/16/70

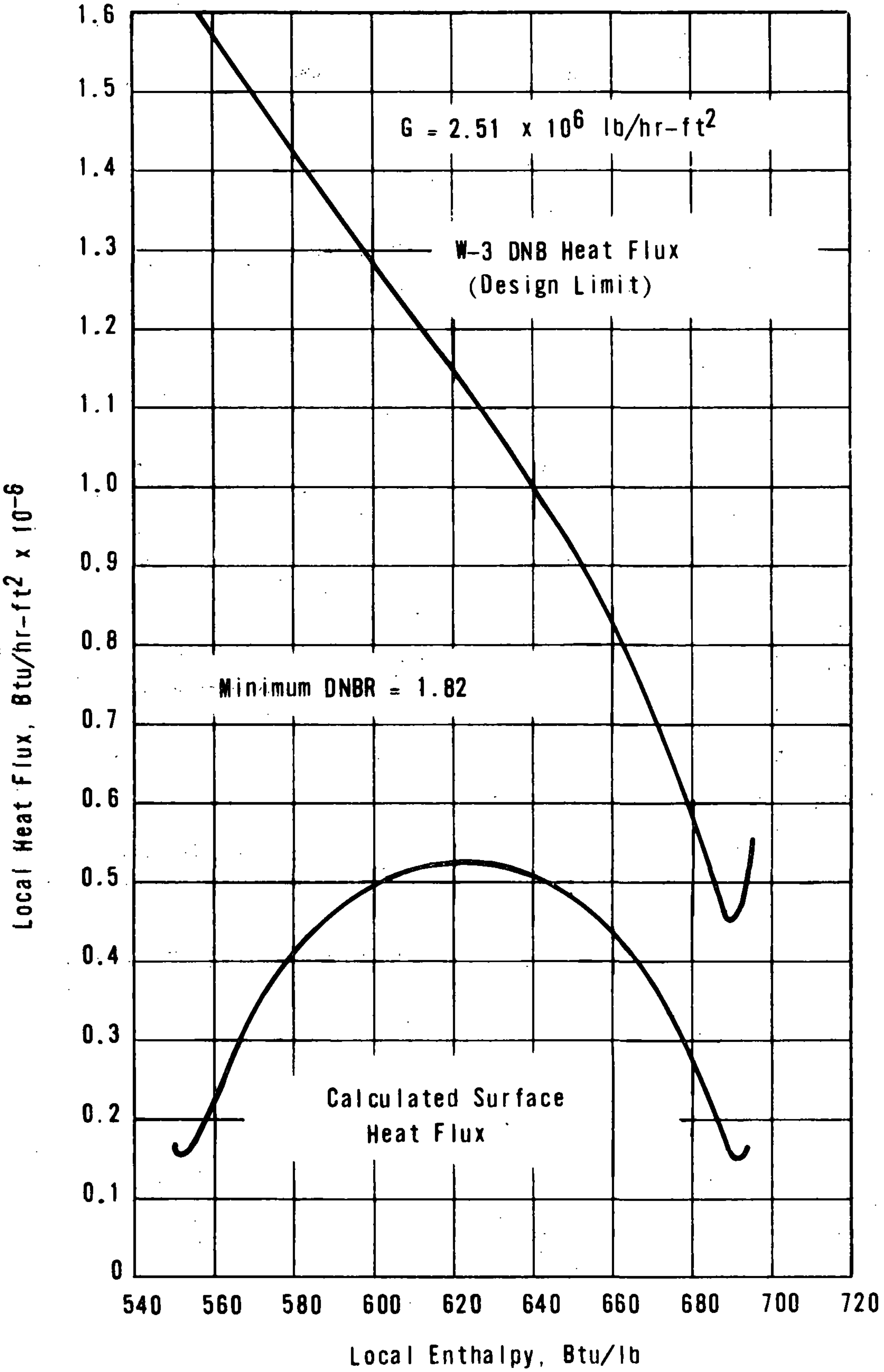


MAXIMUM FUEL ROD POWER PEAKS AND CELL EXIT ENTHALPY RISE RATIOS

OCONEE NUCLEAR STATION

Figure 3 - 42

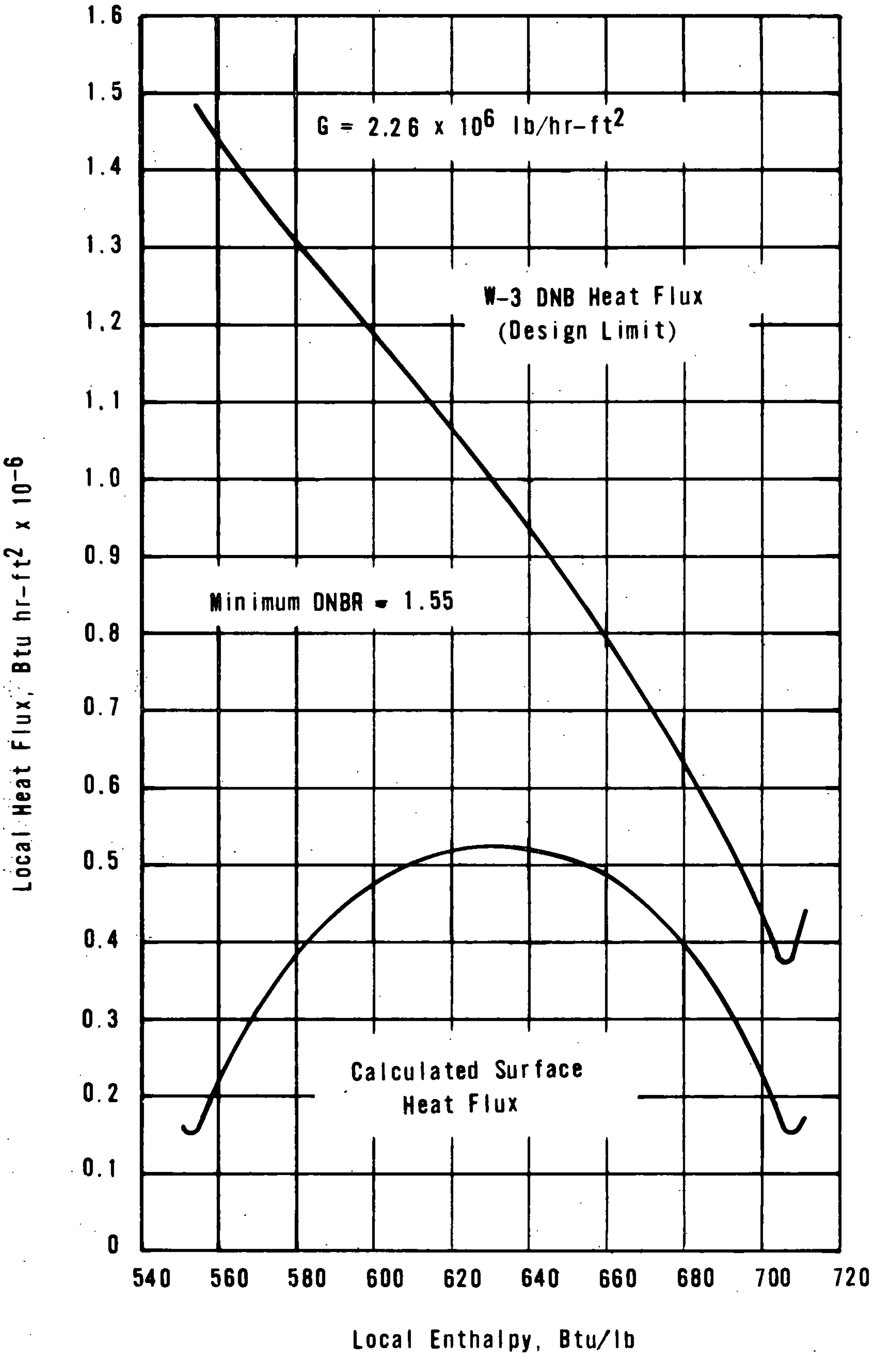
Rev. 3 3/16/70



CALCULATED AND DESIGN LIMIT LOCAL HEAT FLUX VERSUS ENTHALPY IN THE HOT UNIT CELL AT THE NOMINAL CONDITION

OCONEE NUCLEAR STATION

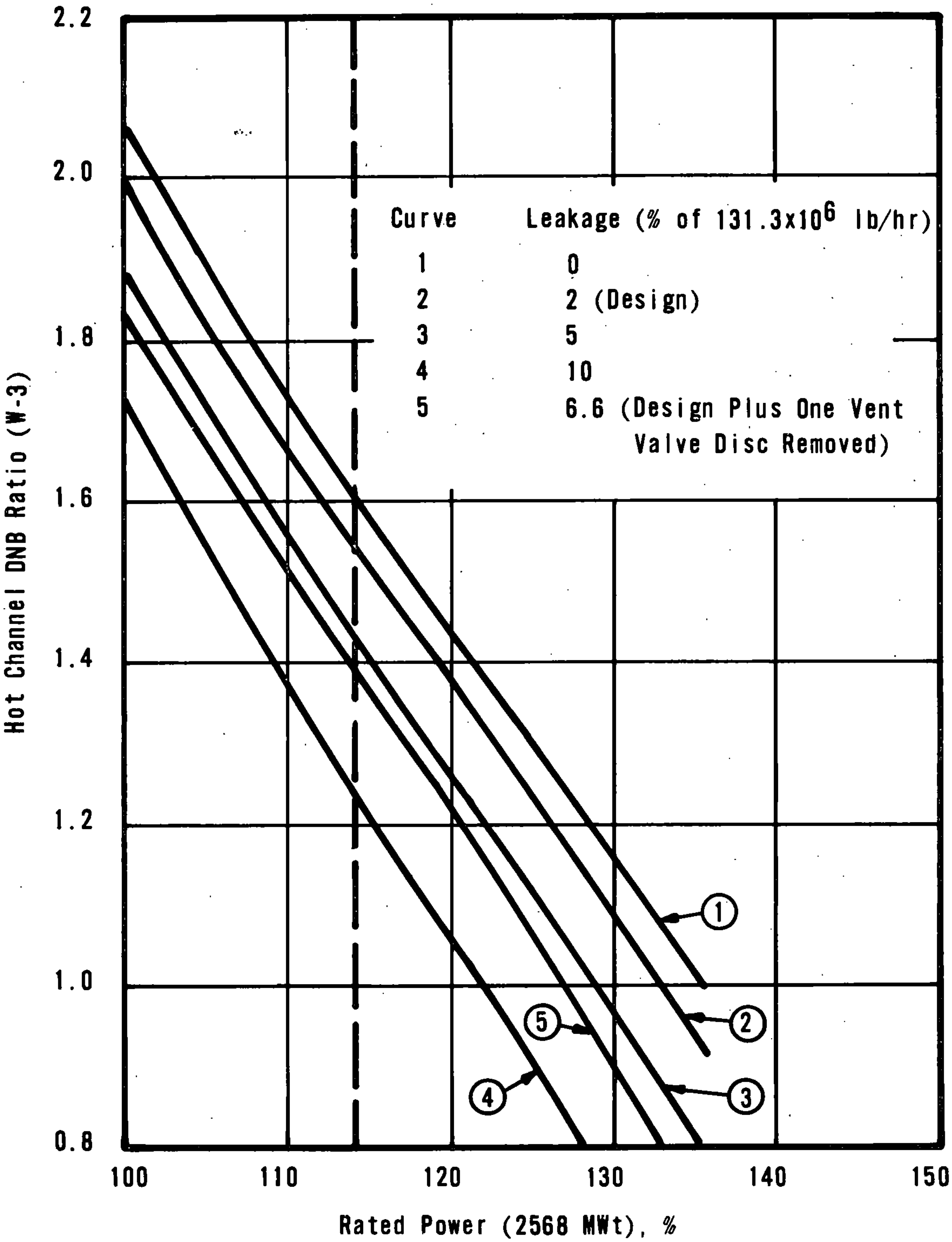
Figure 3 - 43



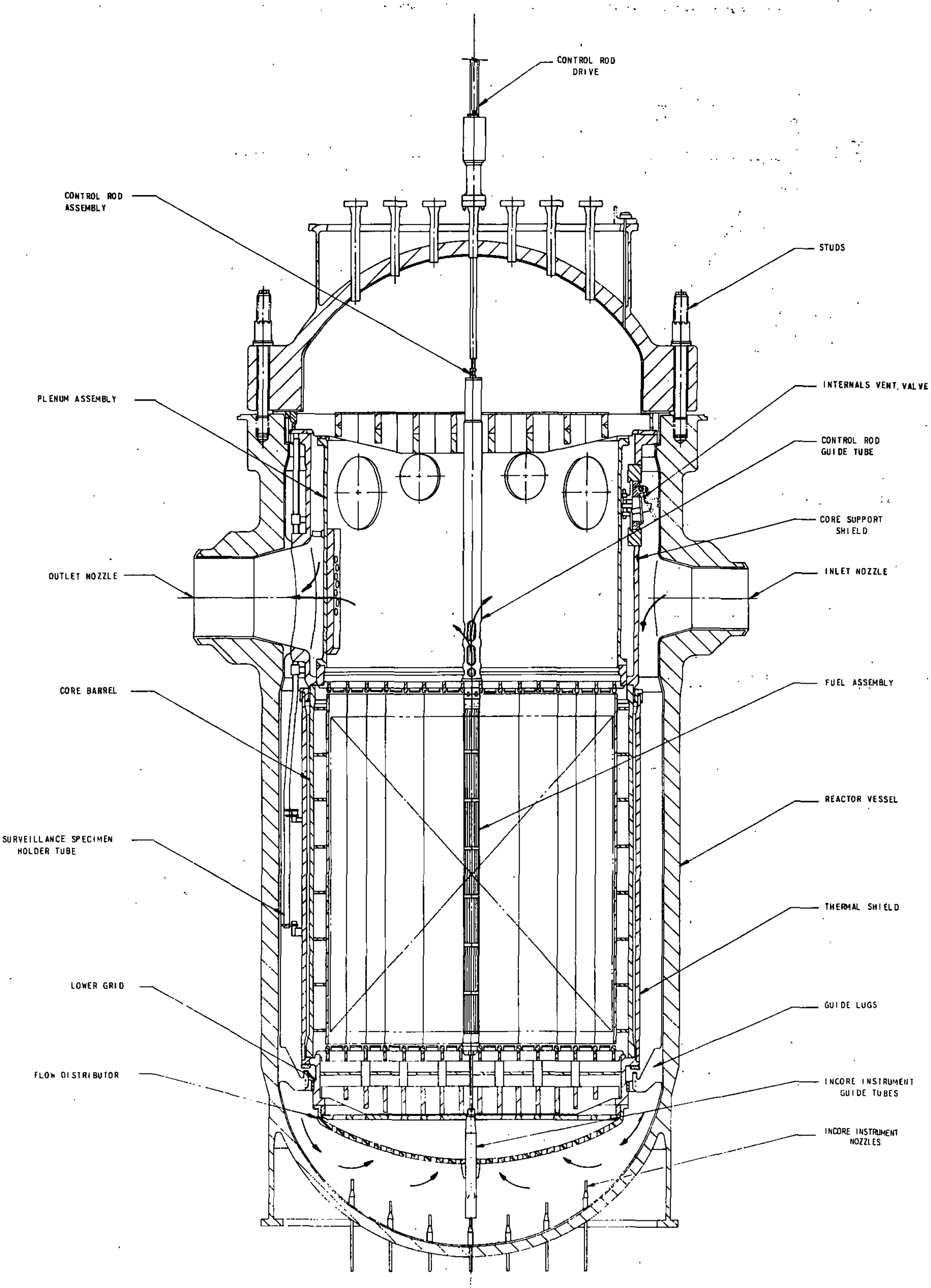
CALCULATED AND DESIGN LIMIT LOCAL HEAT FLUX VS ENTHALPY IN THE HOT UNIT CELL AT THE DESIGN CONDITION

OCONEE NUCLEAR STATION

Figure 3 - 44



DNB RATIO (W-3) VERSUS POWER FOR VARIOUS INLET-TO-OUTLET CORE BYPASS LEAKAGE

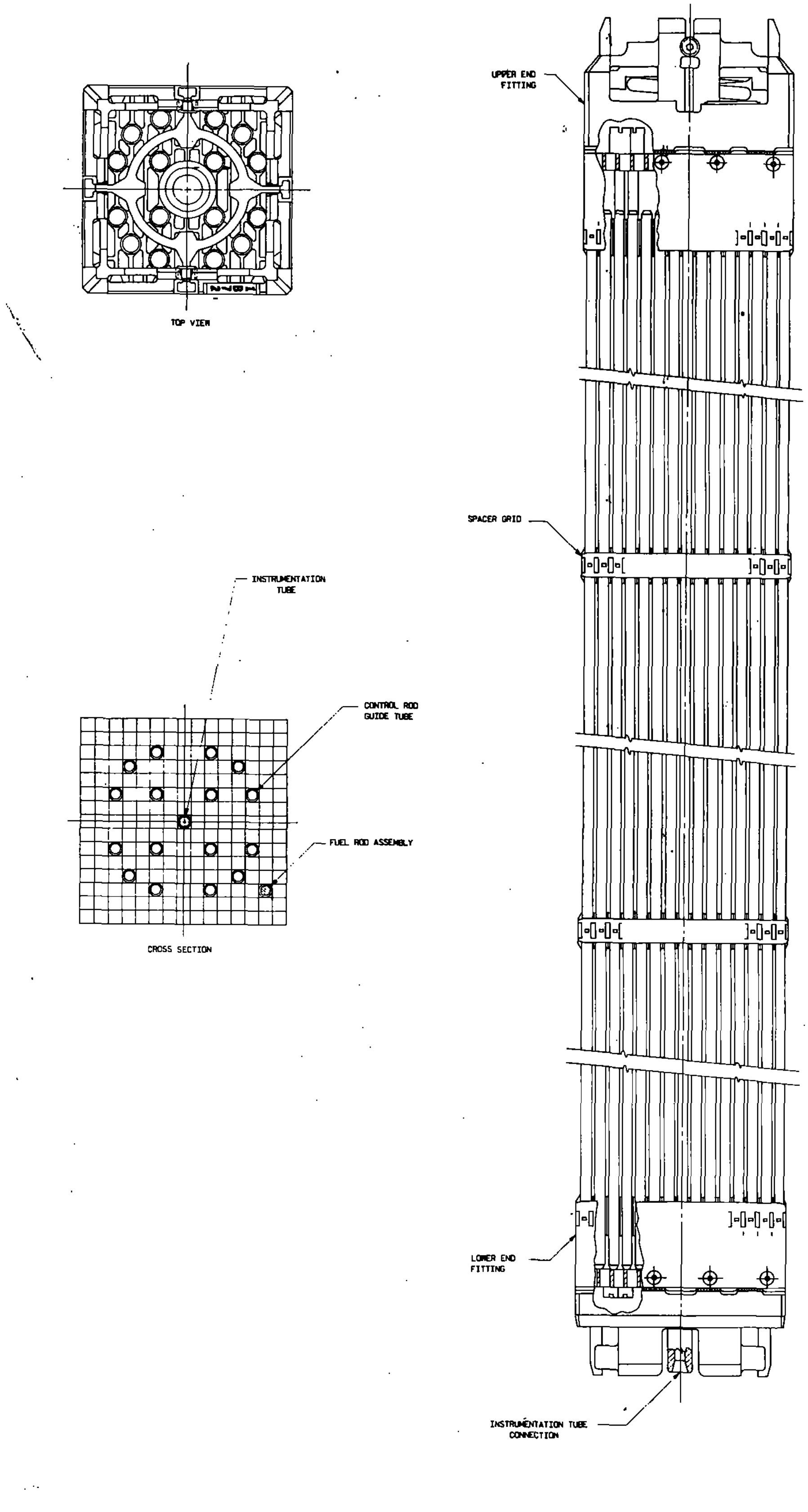


REACTOR VESSEL AND INTERNALS - GENERAL ARRANGEMENT

OCONEE NUCLEAR STATION

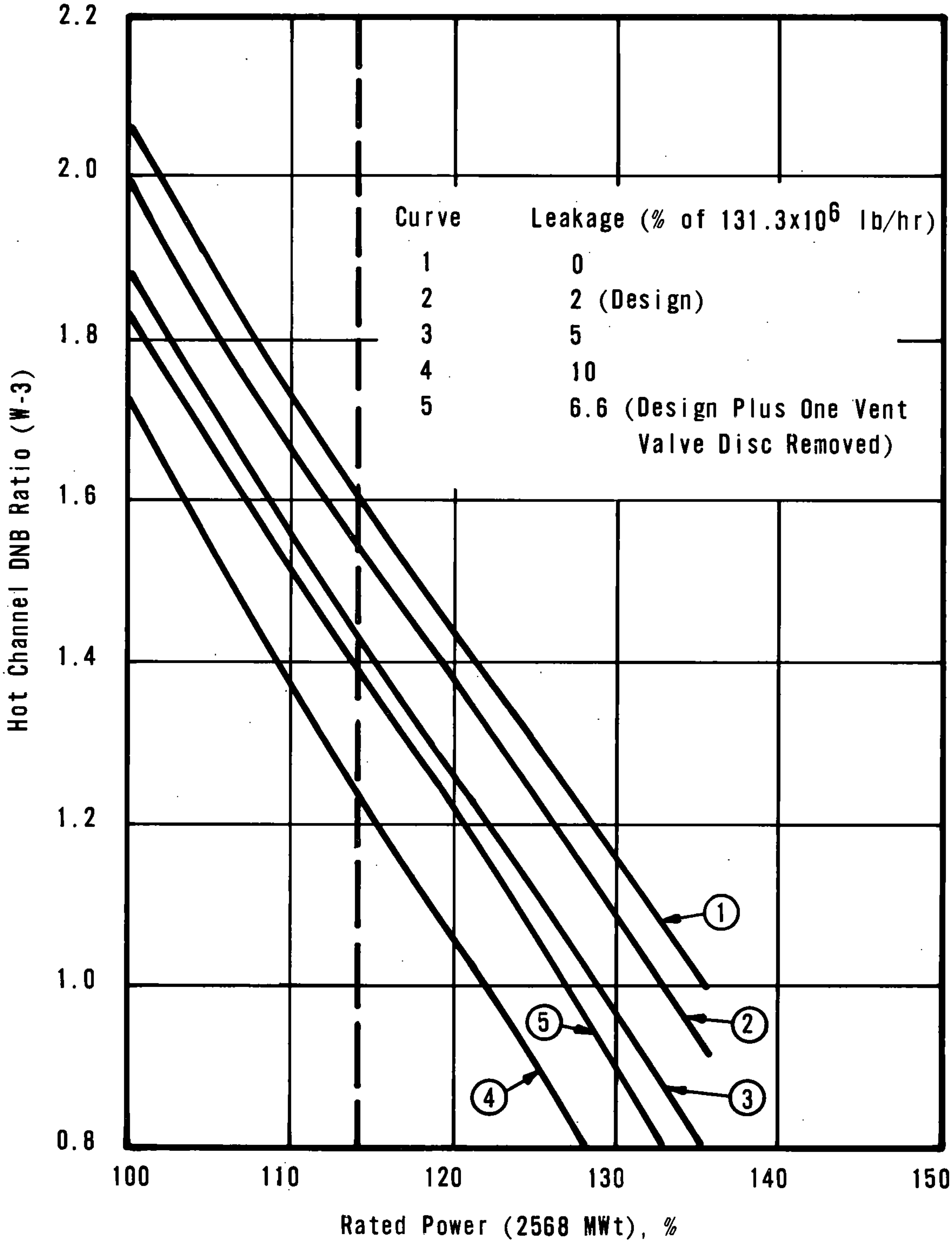
Figure 3 - 46

Rev. 1 9/15/69
Rev. 23 9/15/72

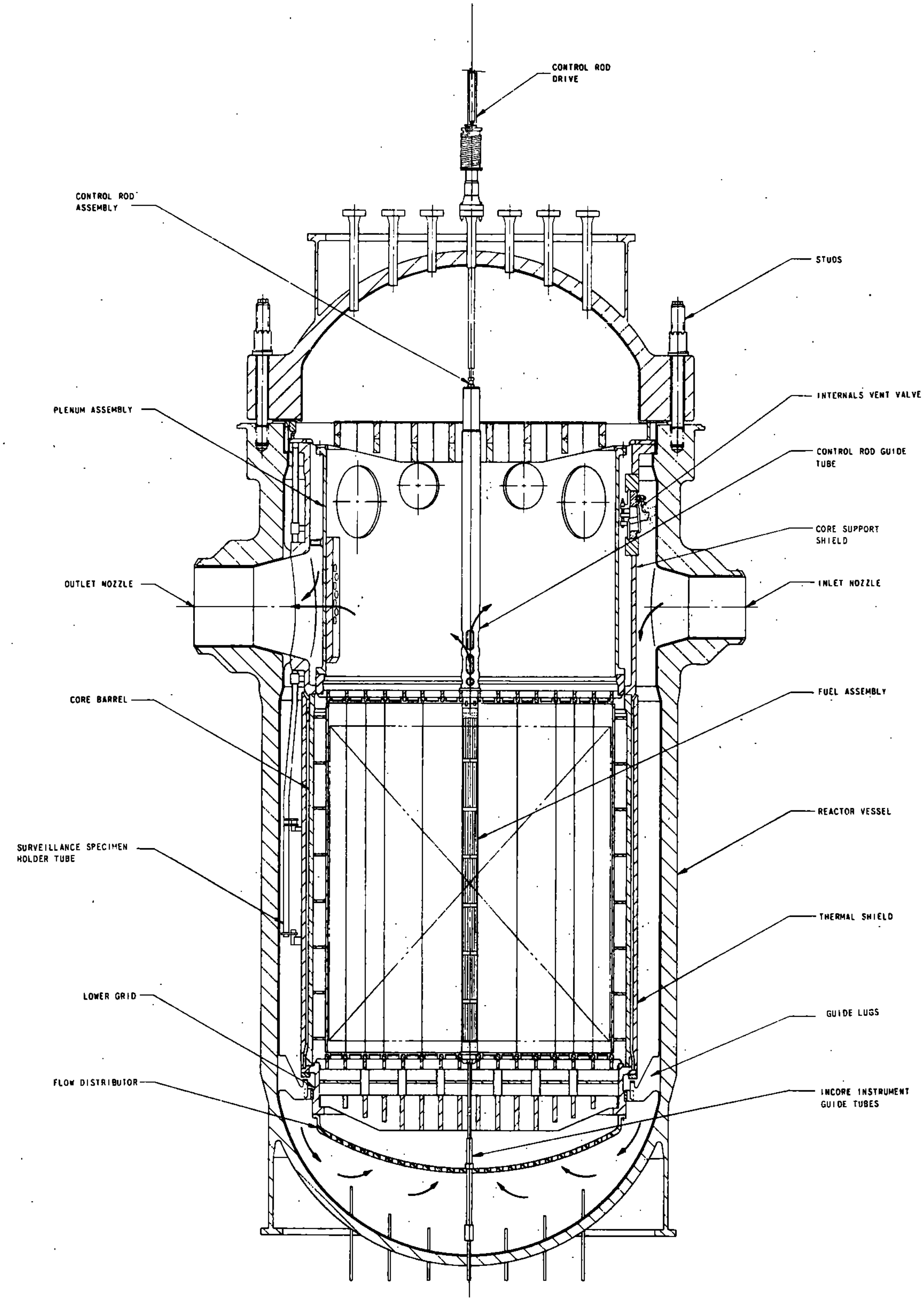


FUEL ASSEMBLY (Shortened end fittings)

OCONEE NUCLEAR STATION
Figure 3 - 52A
(New) Rev. 16 7/30/71



DNB RATIO (W-3) VERSUS POWER FOR VARIOUS INLET-TO-OUTLET CORE BYPASS LEAKAGE

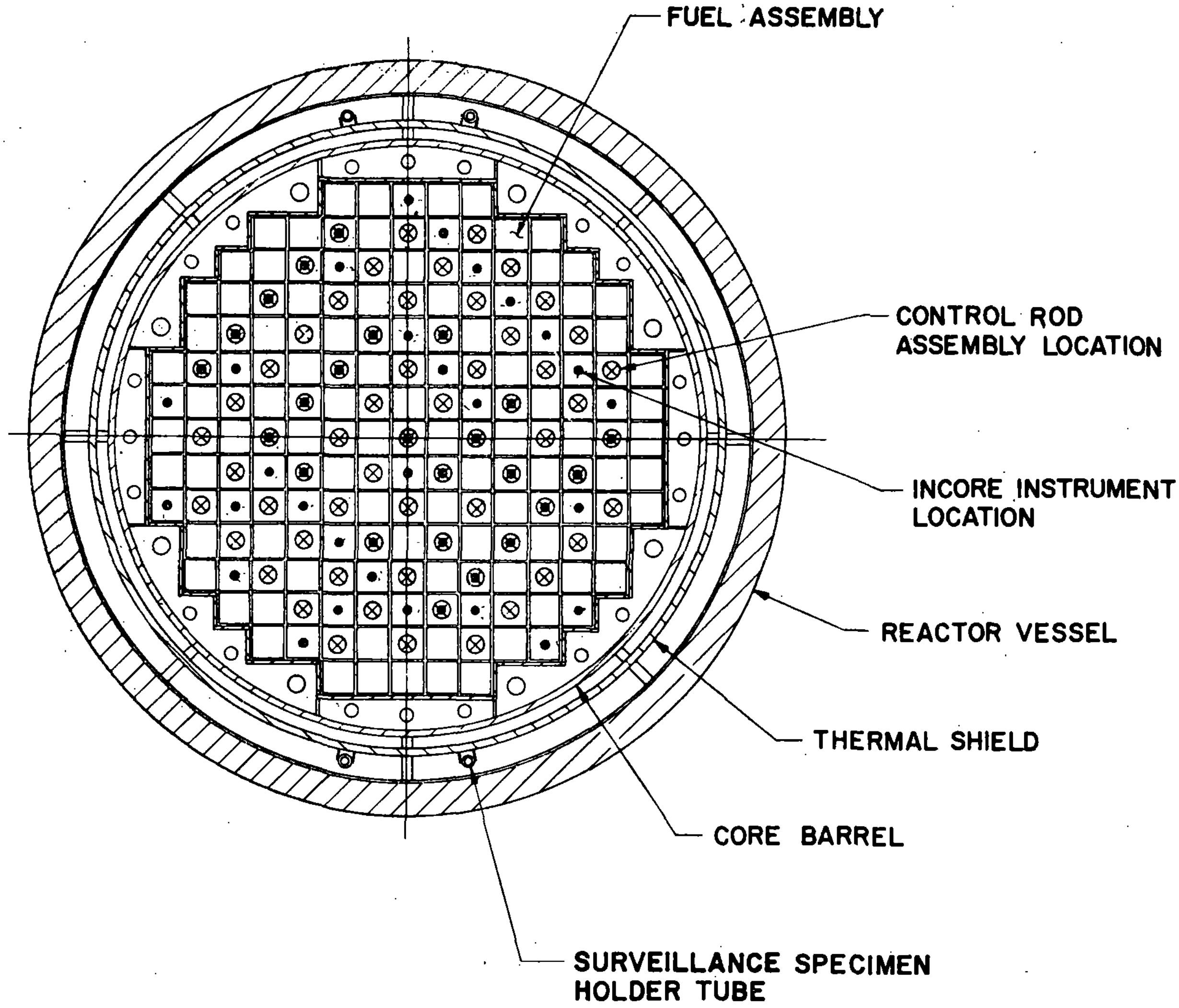


REACTOR VESSEL AND INTERNALS - GENERAL ARRANGEMENT

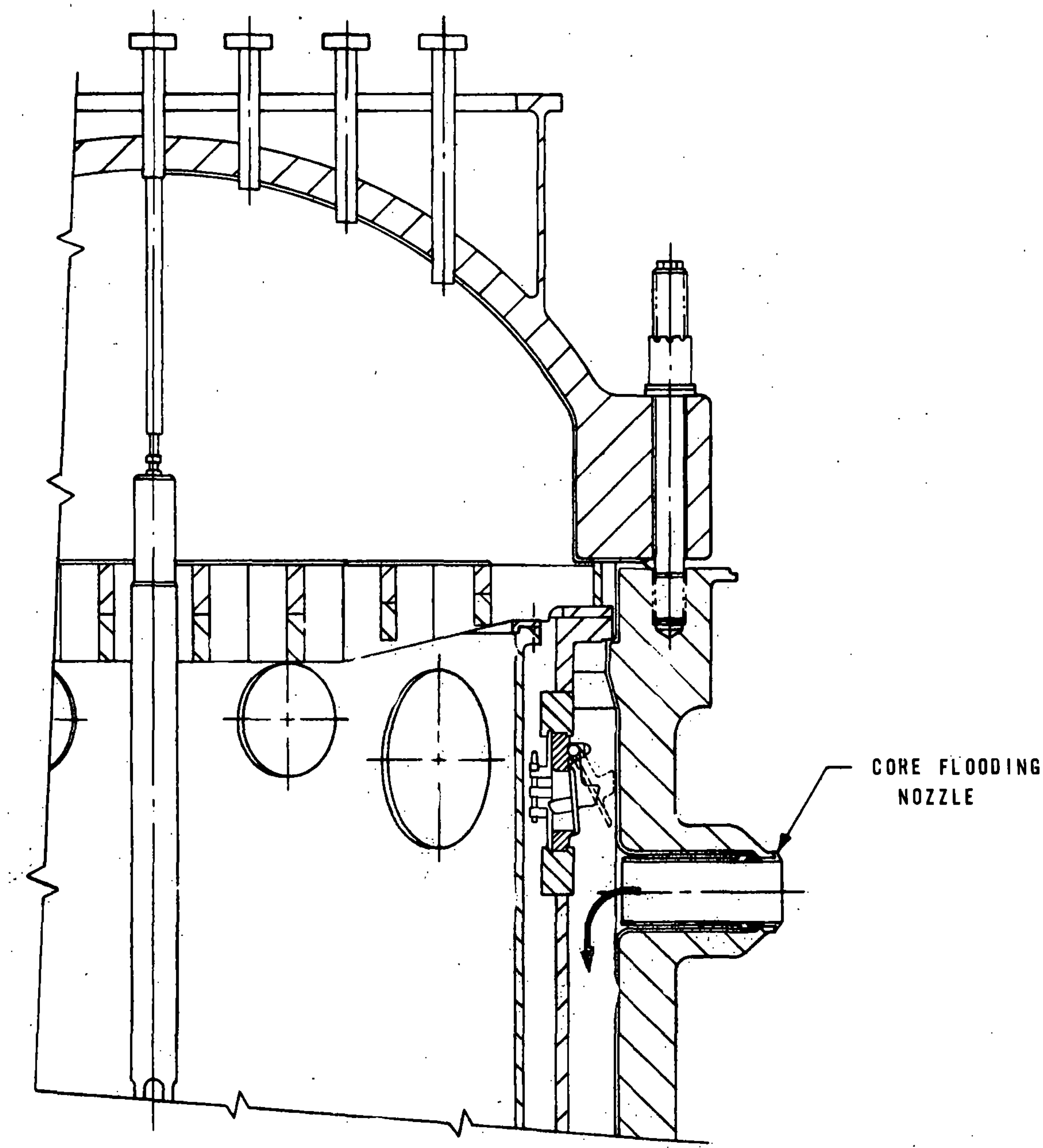
OCONEE NUCLEAR STATION

Figure 3 - 46

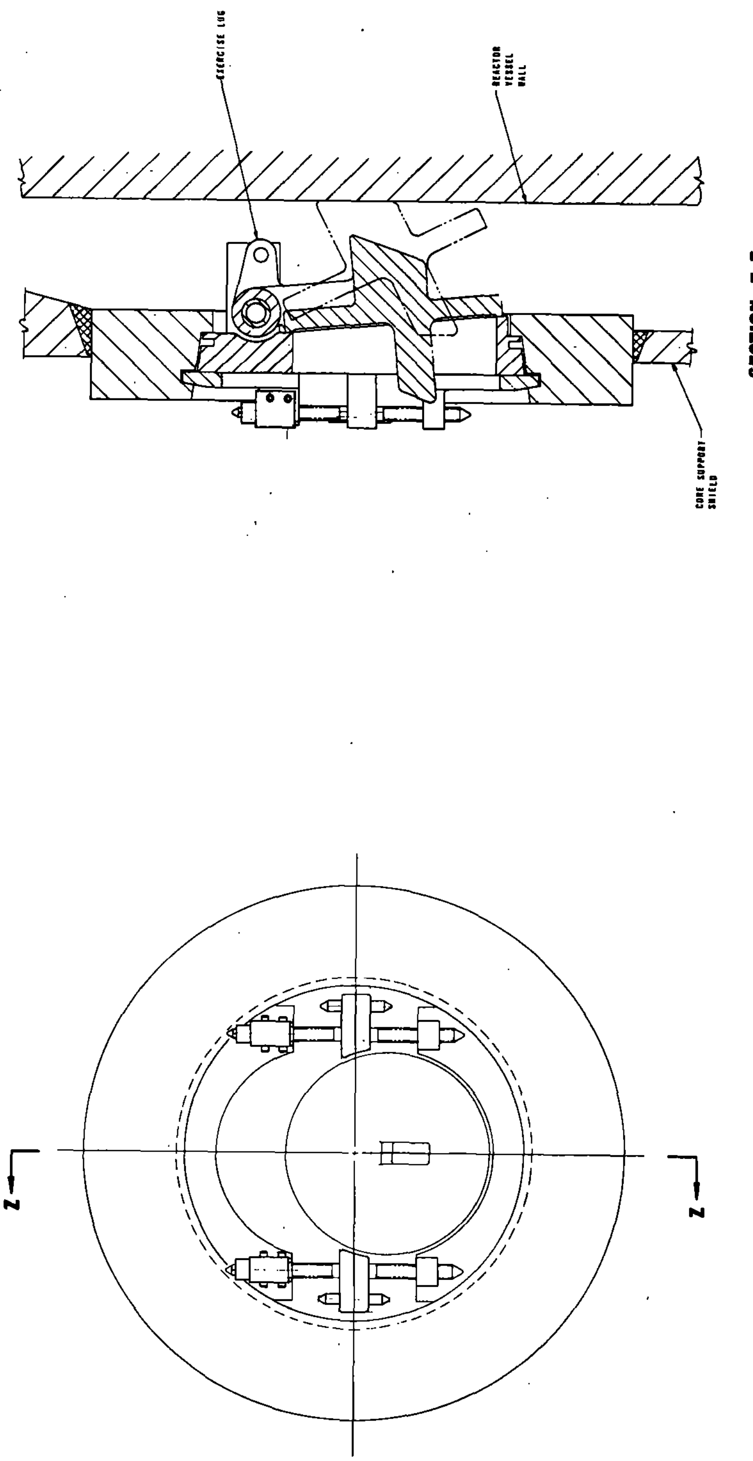
Rev. 1 9/15/69



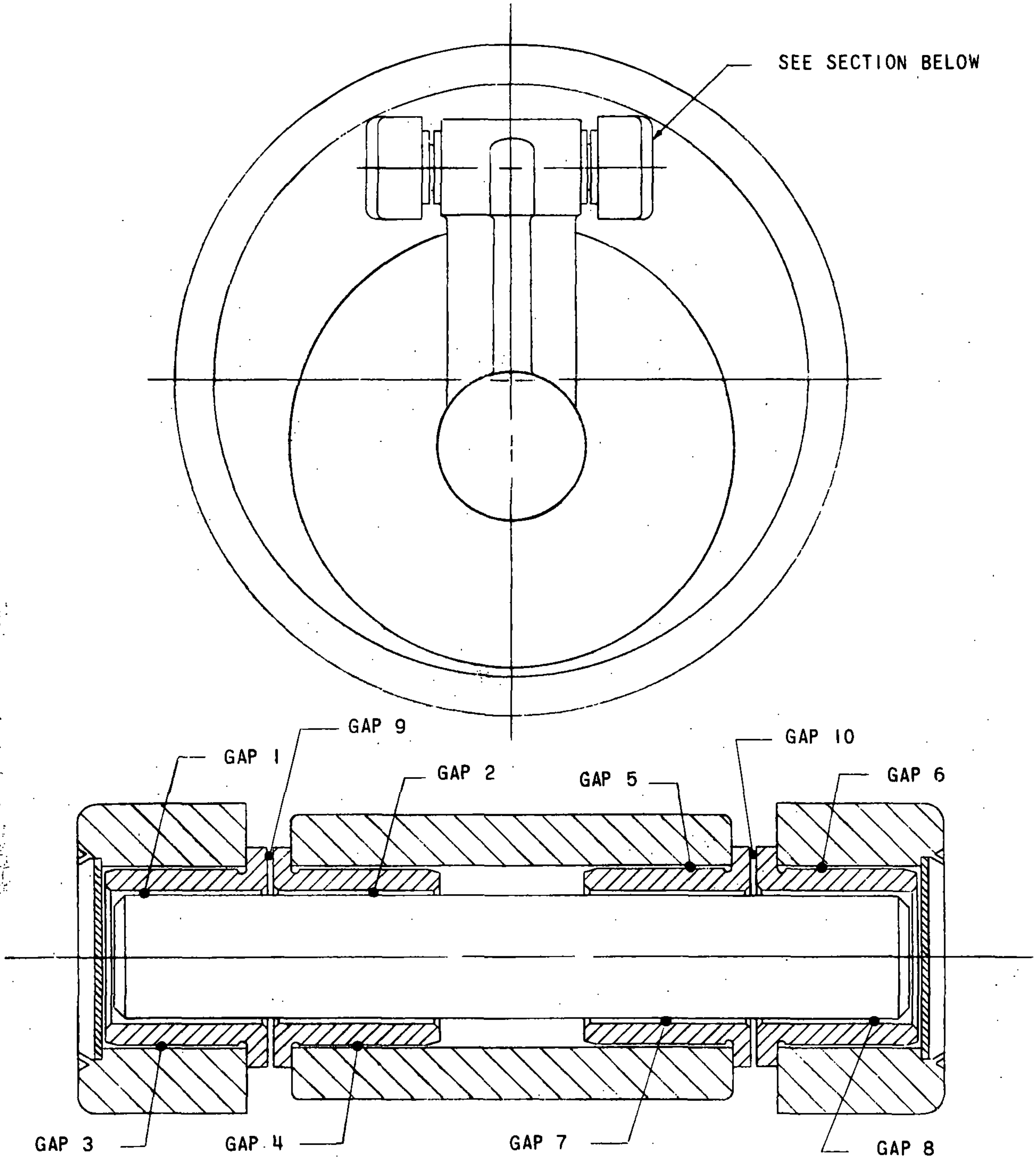
REACTOR VESSEL AND INTERNALS-
CROSS SECTION



CORE FLOODING ARRANGEMENT



INTERNALS VENT VALVE

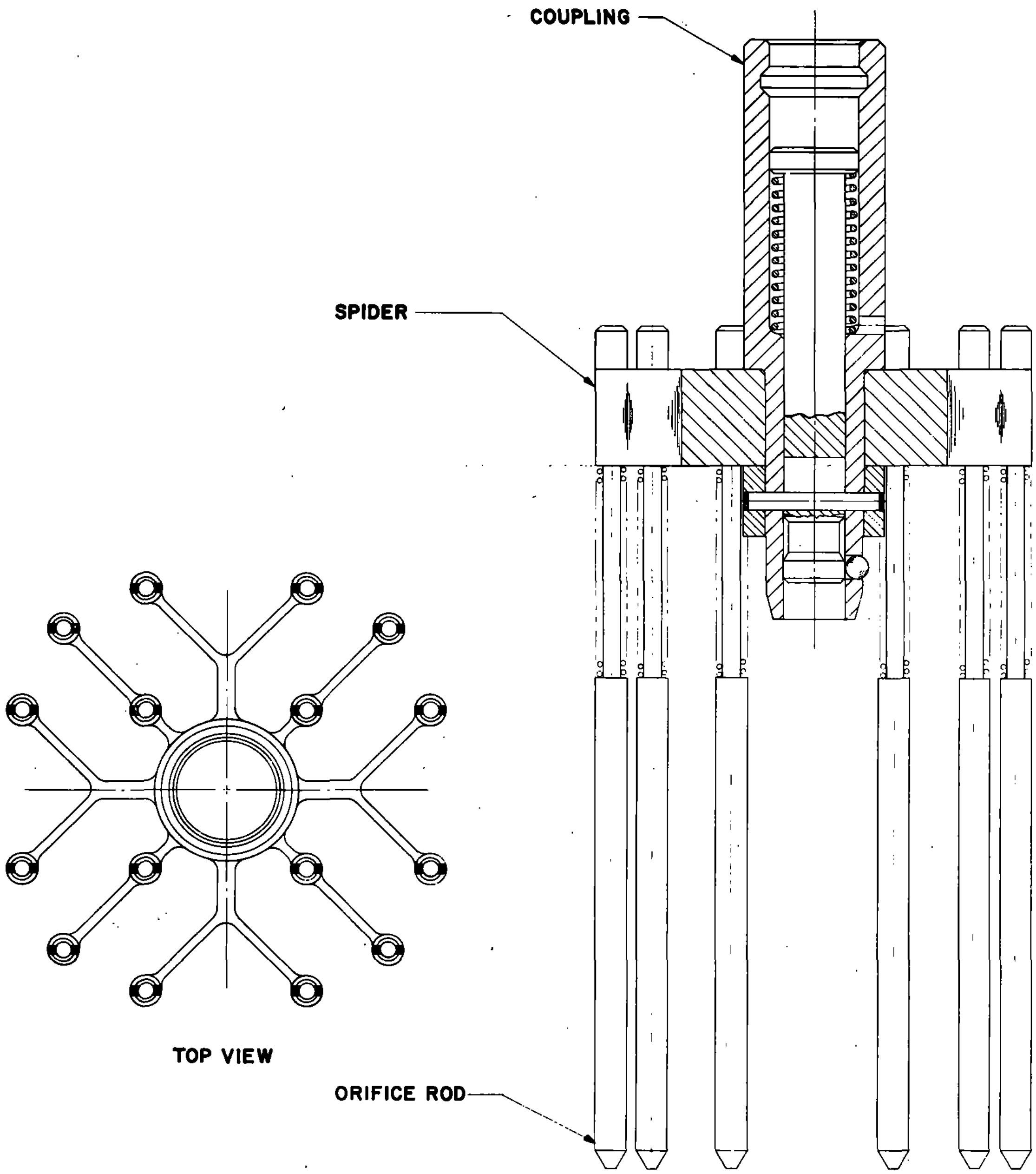


INTERNALS VENT VALVE
CLEARANCE GAPS

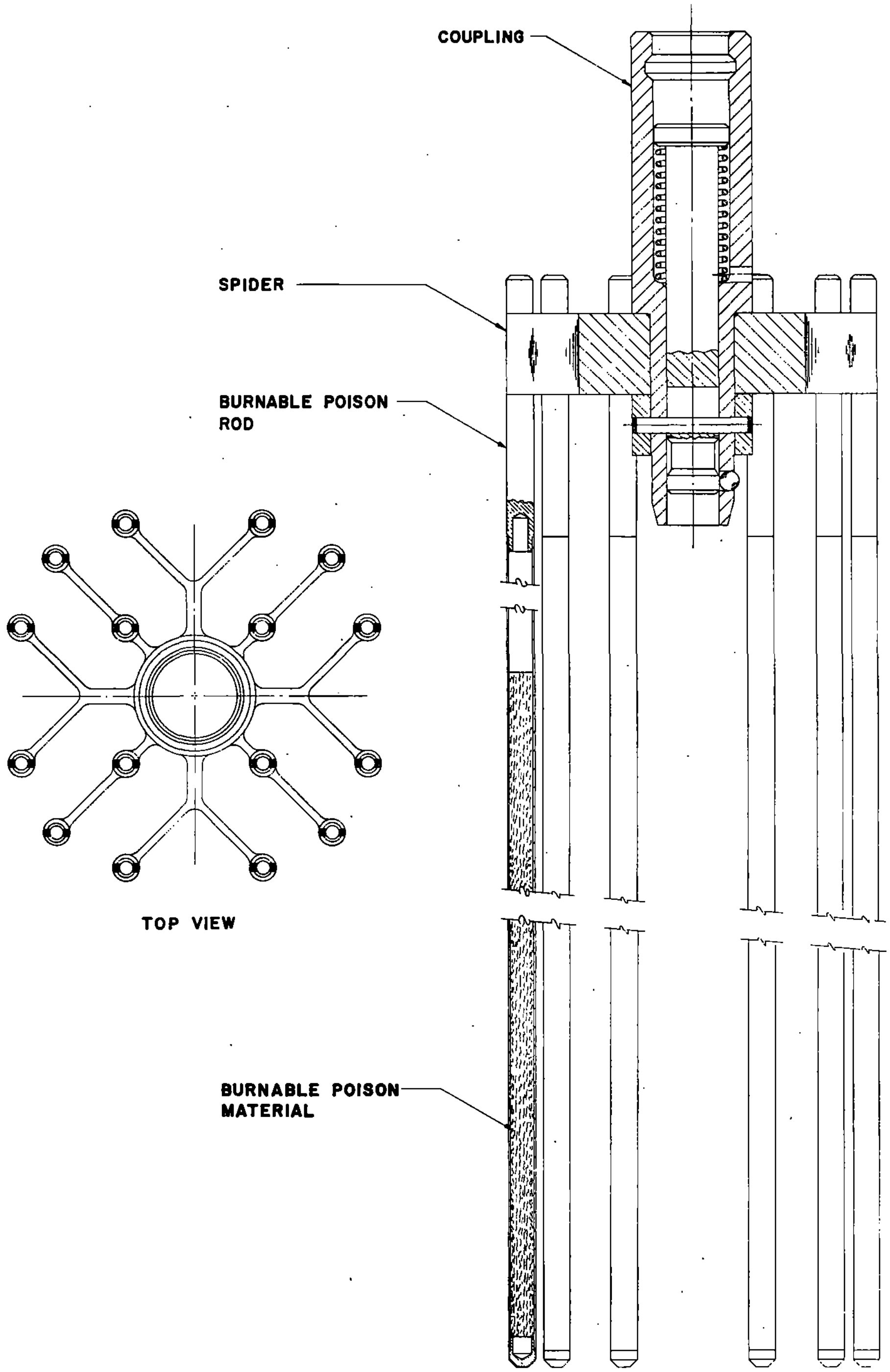
OCONEE NUCLEAR STATION

Figure 3 - 49A

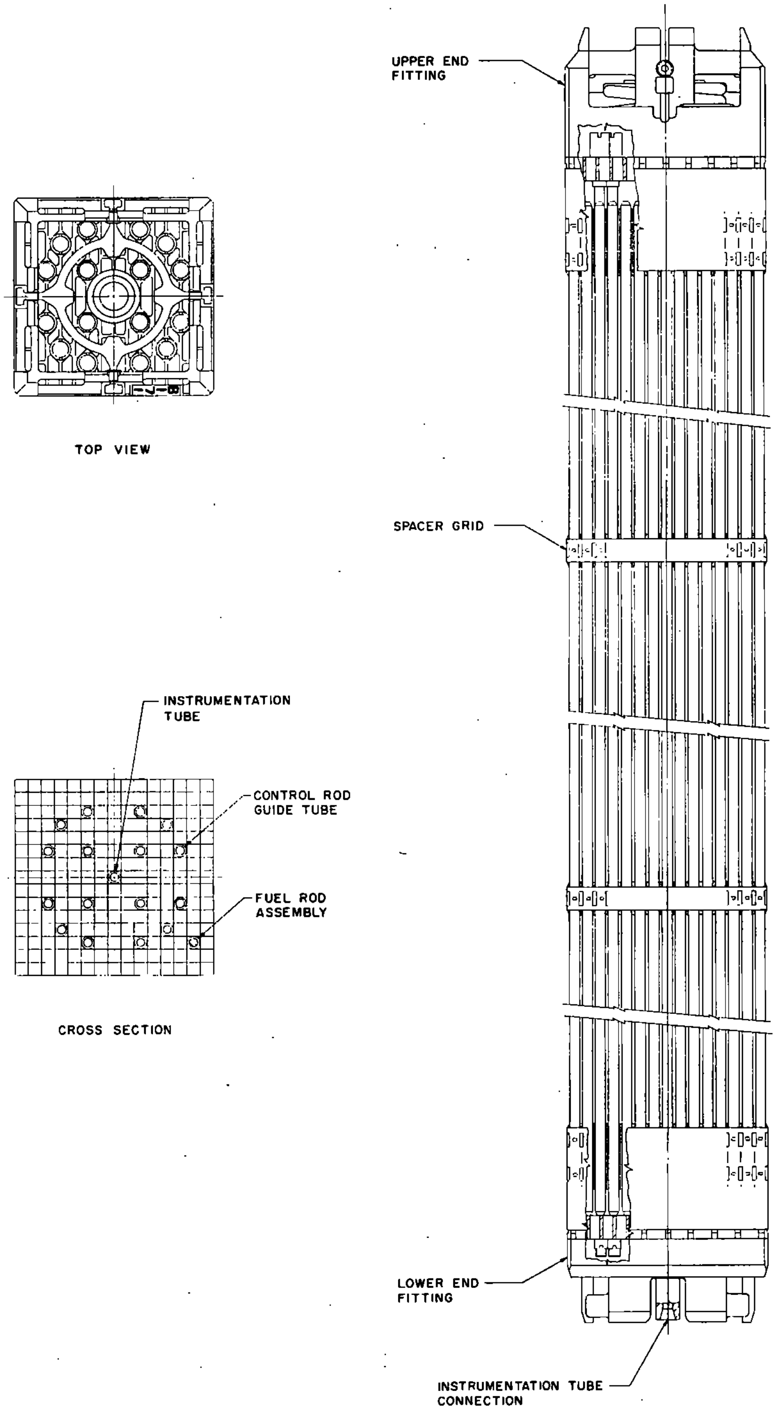
(New) Rev. 4 4/20/70



ORIFICE ROD ASSEMBLY



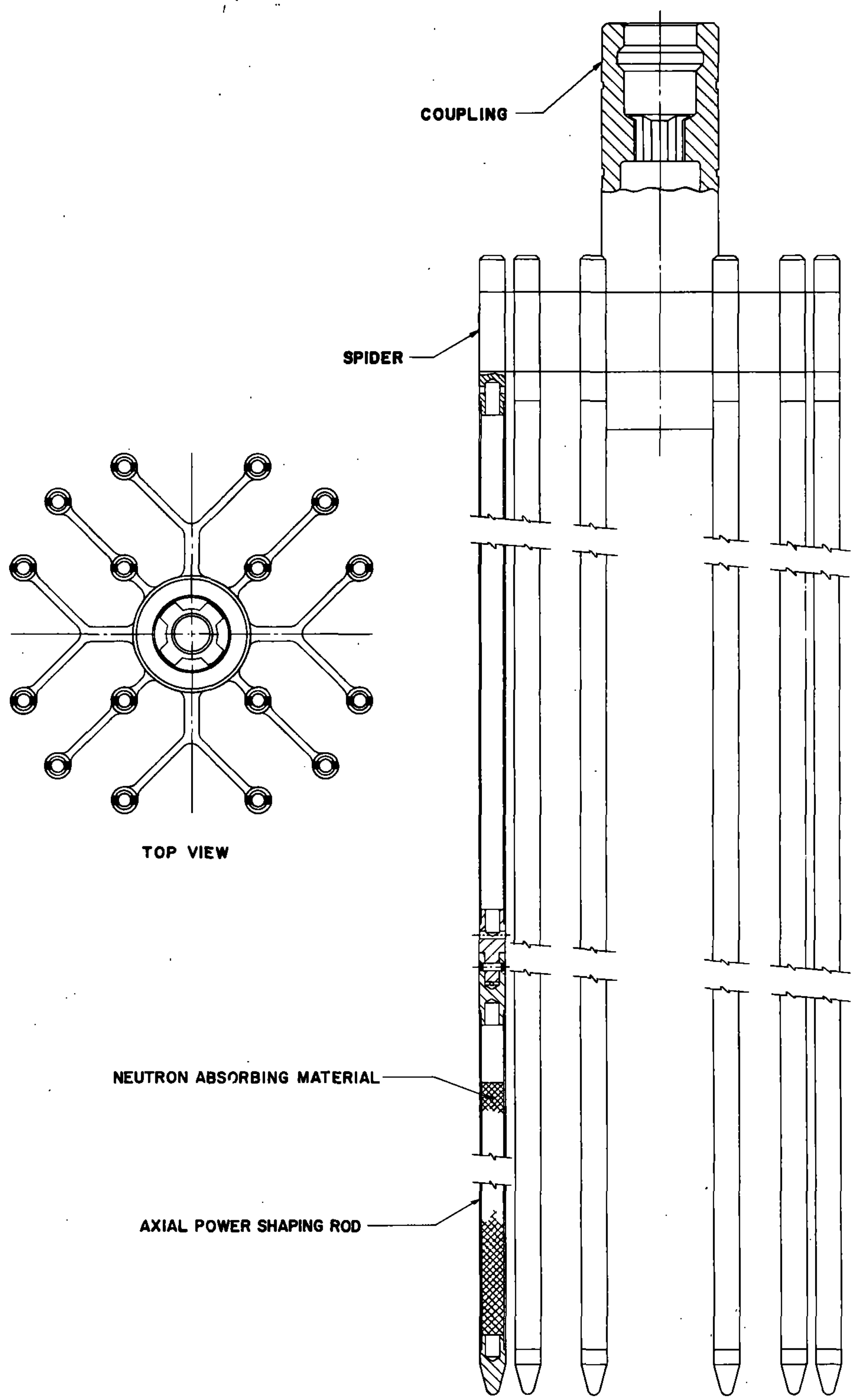
BURNABLE POISON ROD ASSEMBLY



FUEL ASSEMBLY

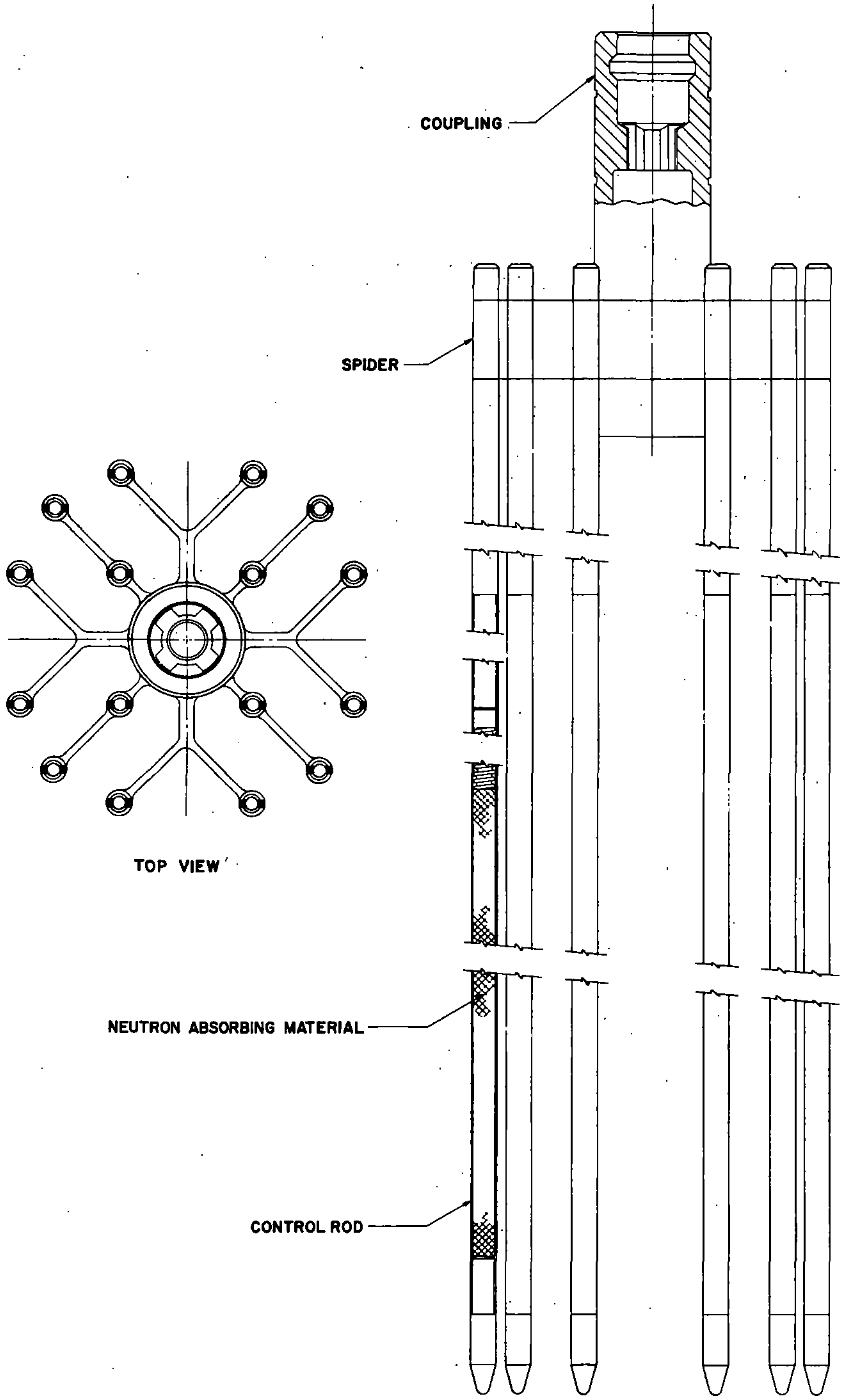
OCONEE NUCLEAR STATION

Figure 3 - 52



AXIAL POWER SHAPING ROD ASSEMBLY

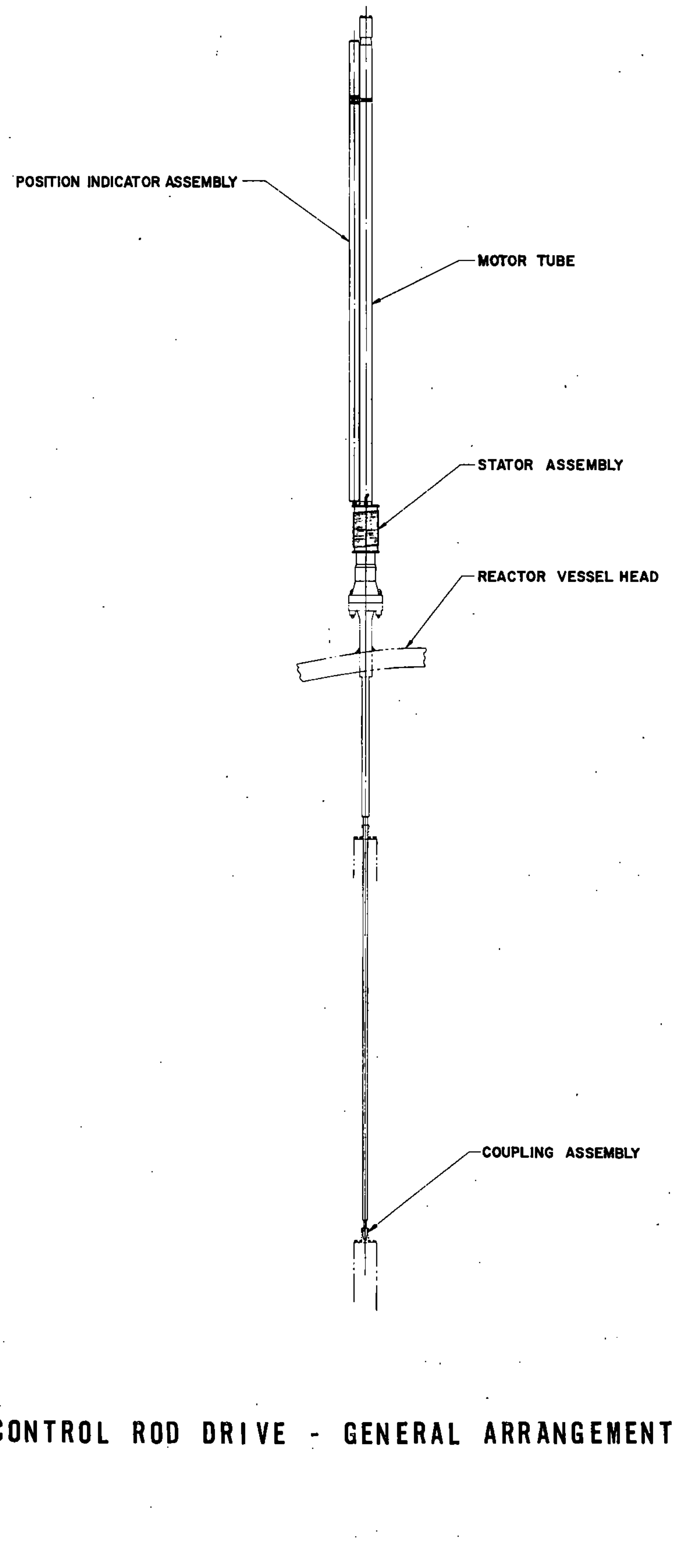
OCONEE NUCLEAR STATION
Figure 3 - 53
Rev. 1 9/15/69



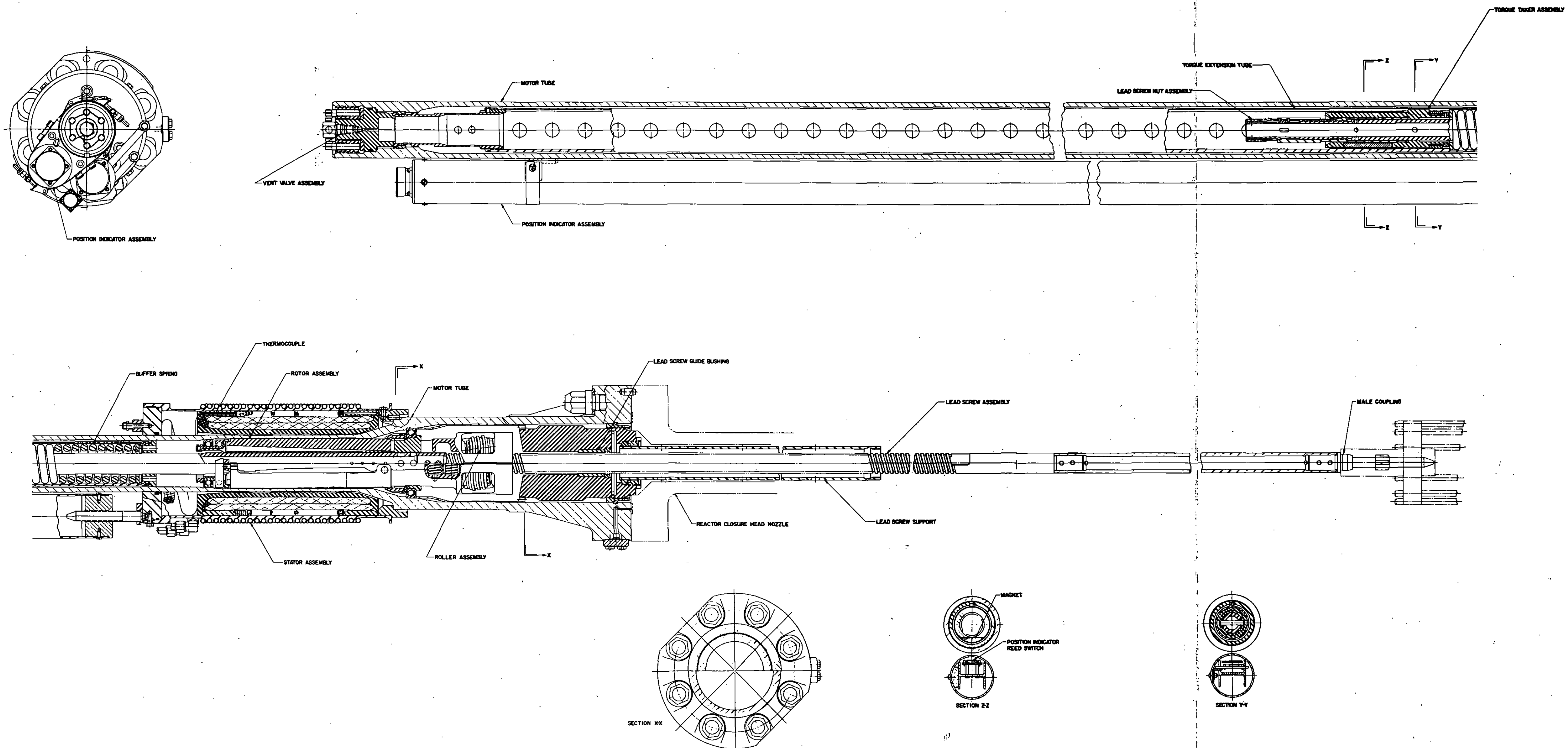
CONTROL ROD ASSEMBLY

OCONEE NUCLEAR STATION

Figure 3 - 54



CONTROL ROD DRIVE - GENERAL ARRANGEMENT



CONTROL ROD DRIVE - VERTICAL SECTION

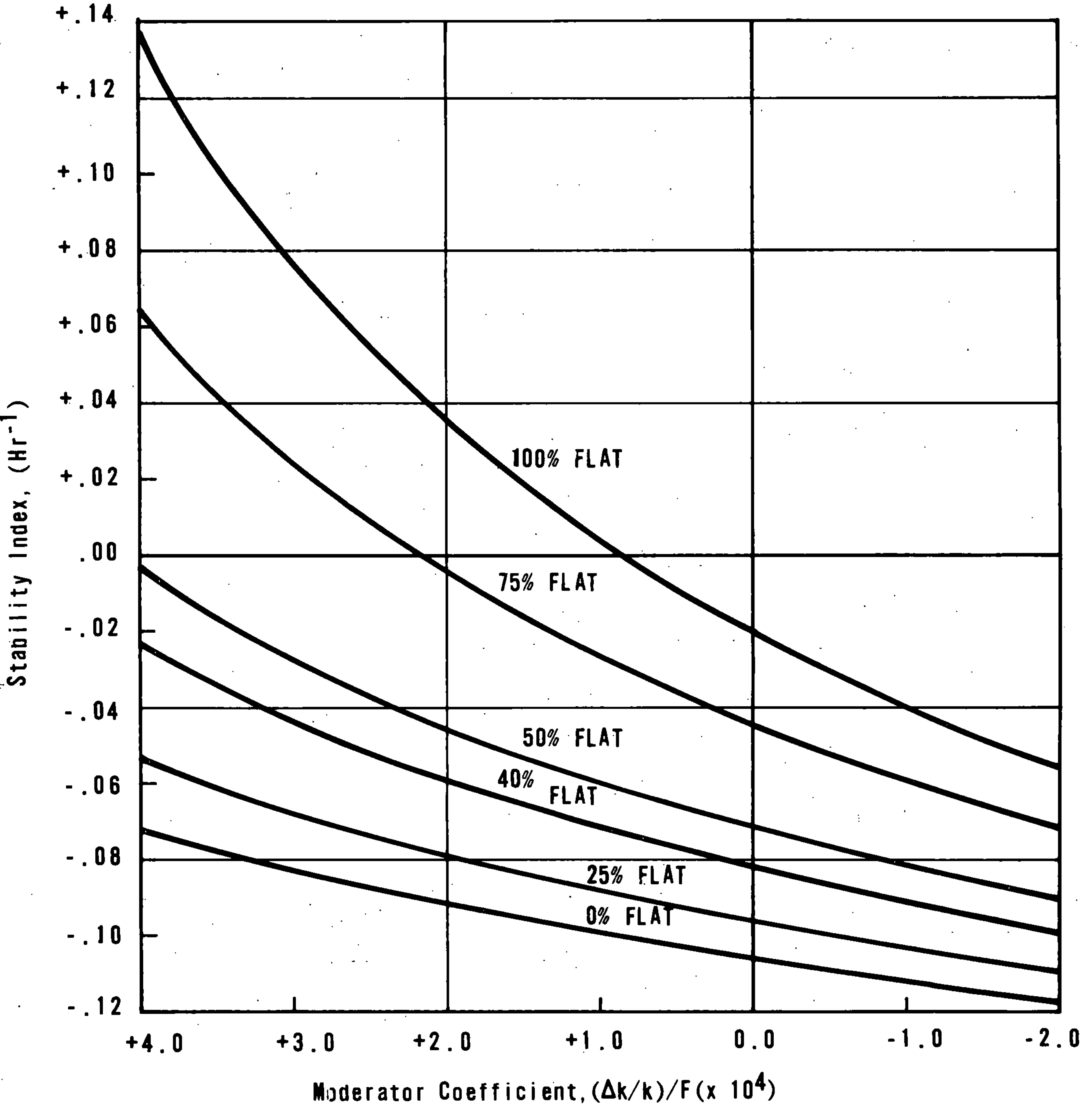
(Reference Supplement 9 Revisions for Oconee 3)

OCONEE NUCLEAR STATION

Figure 3 - 56

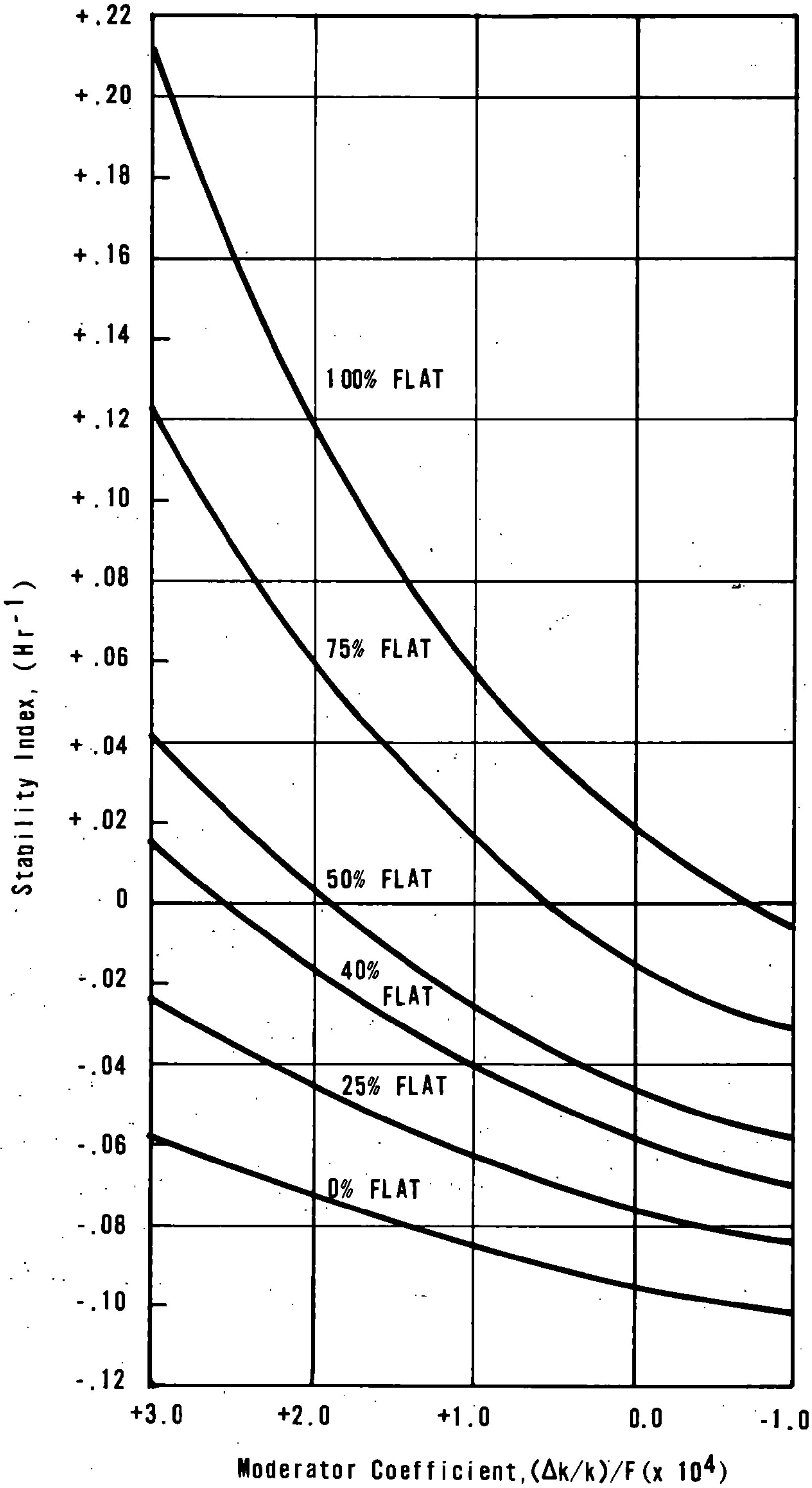
Rev. 5 5/25/70

Rev. 16. 7/30/71



AZIMUTHAL STABILITY INDEX VS. MODERATOR
COEFFICIENT FROM 3 DIMENSIONAL CASE
(Oconee Unit 2, Cycle 1)

OCONEE NUCLEAR STATION
Figure 3 - 6L
(New) Rev. 16 1971



AZIMUTHAL STABILITY INDEX WITH
COMPOUNDED ERROR VS. MODERATOR
COEFFICIENT FROM 3 DIMENSIONAL CASE
(Oconee 2, Cycle 1)

OCONEE NUCLEAR STATION
Figure 3 - 6M
(New) Rev. 16 7/30/71

LIST OF EFFECTIVE PAGES
FSAR APPENDIX 3-A

PRESSURIZED FUEL

Page	Revision
List of Effective Pages....	Rev. 26
Cover Sheet................	Rev. 10
3A-i	Original
3A-ii......................	Original
3A-iii	Rev. 24
3A-1	Rev. 24
3A-2	Rev. 26
3A-3	Rev. 24
Fig. 3A-2	Original

APPENDIX 3A

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LIST OF TABLES

LIST OF FIGURES

24.

APPENDIX 3-A

PRESSURIZED FUEL

3A.1 INTRODUCTION

24. The reactors for the Oconee Nuclear Station will utilize helium pressurized fuel rods. The entire core of all units will be pressurized. The use of pressurized fuel has been proposed and accepted for other recently licensed PWR's.

The initial pressurization affords a mechanical enhancement of the fuel rod structural margins over the fuel lifetime. Clad stresses and strains from routine reactor power and pressure transients will be minimized. Prepressurization of the fuel rods partially offsets the reactor coolant system external pressure, reducing the net compressive stresses in the clad. It lengthens the period of time before fuel-to-clad contact occurs from the combined action of fuel pellet swelling and clad creep, with a corresponding reduction in clad fatigue strain at the end of life, and an increase in the margin against fatigue failure.

The effects of initial pressurization on thermal, nuclear, and safety aspects of core operation have been investigated. The results of this study are given below.

3A.2 PRESSURIZATION EFFECTS ON FUEL TEMPERATURE AND INTERNAL PRESSURE

Fuel pin parameters affected by initial prepressurization are fuel temperature and fuel pin internal pressure. In order to determine the effective change in fuel pin parameters resulting from prepressurization, identical conditions for both the prepressurized and unpressurized fuel pins are considered. The methods used to evaluate the fuel pin parameters are explained in Sections 3.2.3.1.2 and 3.2.3.2.3(g) & (h). The hot pin normal power conditions have been used to determine the effect of prepressurization on fuel pin parameters. The maximum power in the hot pin varies over the life of the hot assembly as explained in Section 3.2.3.2.3(h) 1(c). This design line has been selected since it conservatively encompasses the peak conditions encountered during life. The average burnup in the hot fuel pin at the end of life is 38,150 MWD/MTU as determined in Section 3.2.3.2.3(h) 1(b). A 1.50 max/avg modified cosine power and burnup shape as shown in Figure 3-37 has been evaluated to insure adequate local fuel cladding strength for possible increases in local burnup values to 55,000 MWD/MTU.

An initially prepressurized fuel pin has lower fuel temperatures and larger internal pin pressures than an unpressurized pin for the same power history conditions. The expected internal pressure from BOL through a maximum average pin burnup of 38,150 MWD/MTU (corresponding to a local peak burnup of 57,000 MWD/MTU) is shown in Figure 3A-2. The design maximum burnup is 55,000 MWD/MTU. The EOL point on the curve representing the unpressurized case is consistent with the maximum pressure at nominal diametral clearance of 7 mils shown in Figure 3-40.

internal pressure buildup is based on nominal cold gap clearance and normal power, using the hot pin power envelope as discussed in Section 3.2.3.2.3(h). A study of the power histories of all fuel assemblies through five cycles to equilibrium conditions has shown that the power envelope used in the analysis is conservative. At the design peak burnup, the prepressurized fuel pin has an internal pressure less than nominal system pressure at hot normal power conditions as shown in Figure 3A-2.

Changes in fuel pin design, fuel pellet density, or fuel pellet enrichment due to prepressurization are not anticipated. Fuel temperatures representative of the EOL fuel handling and storage periods are lower for prepressurized pins than for unpressurized pins.

End of life fuel pin internal pressure is below nominal system pressure at hot normal power conditions. Therefore, fuel pin conditions arising from the clad transients specified in Section 3.2.3.1.3 and sustained DNB will not exceed fuel pin conditions previously analyzed for unpressurized fuel pins. Average clad temperatures for prepressurized fuel pins and unpressurized fuel pins are identical.

3A.3 NUCLEAR EFFECTS

The pressurization of fuel pins results in a decrease in fuel temperature. The effects of this fuel temperature decrease have been examined for the Oconee 1 core and are reported in the following table.

Table 3A-1
Effect of Pressurization on Fuel Pin Nuclear Parameters

	Effect	Unpressurized Fuel Assemblies	177 Pressurized Fuel Assemblies
26.	1. Critical boron concentration (hot full power, clean, all rods out), ppm	1284	1310
24.	2. Moderator Coefficient (not full power, no xenon), $\Delta\rho/^{o}F$	$+ 0.27 \times 10^{-4}$	$+ 0.32 \times 10^{-4}$
	3. Power Coefficient (hot full power, no xenon), $\Delta\rho$/MWt	-4.33×10^{-6}	$- 3.9 \times 10^{-6}$
	4. Total Rod Worth (61 full length rods), %$\Delta\rho$	12.10	12.10
	5. Stuck Rod Worth, %$\Delta\rho$	2.10	2.10
24.	6. Transient Rod Worth (hot full power, no xenon), %$\Delta\rho$	1.24	1.24
	7. Potential Ejected Rod Worth, %$\Delta\rho$	0.3115	0.3115

As demonstrated above, the effects of pressurizing 177 fuel assemblies are very small and do not affect plant safety.

3A.4 EFFECTS ON CORE SAFETY

As seen in the discussion on nuclear effects, the moderator temperature coefficient at BOL is slightly more positive and the power coefficient is slightly less negative. These effects will cause all reactivity transients to be essentially unchanged from the nominal cases presented in the safety analysis of Chapter 14. These reactivity coefficients, the boron concentration, and all of the control rod worths with pressurized fuel are well within the limits of the sensitivity analyses presented.

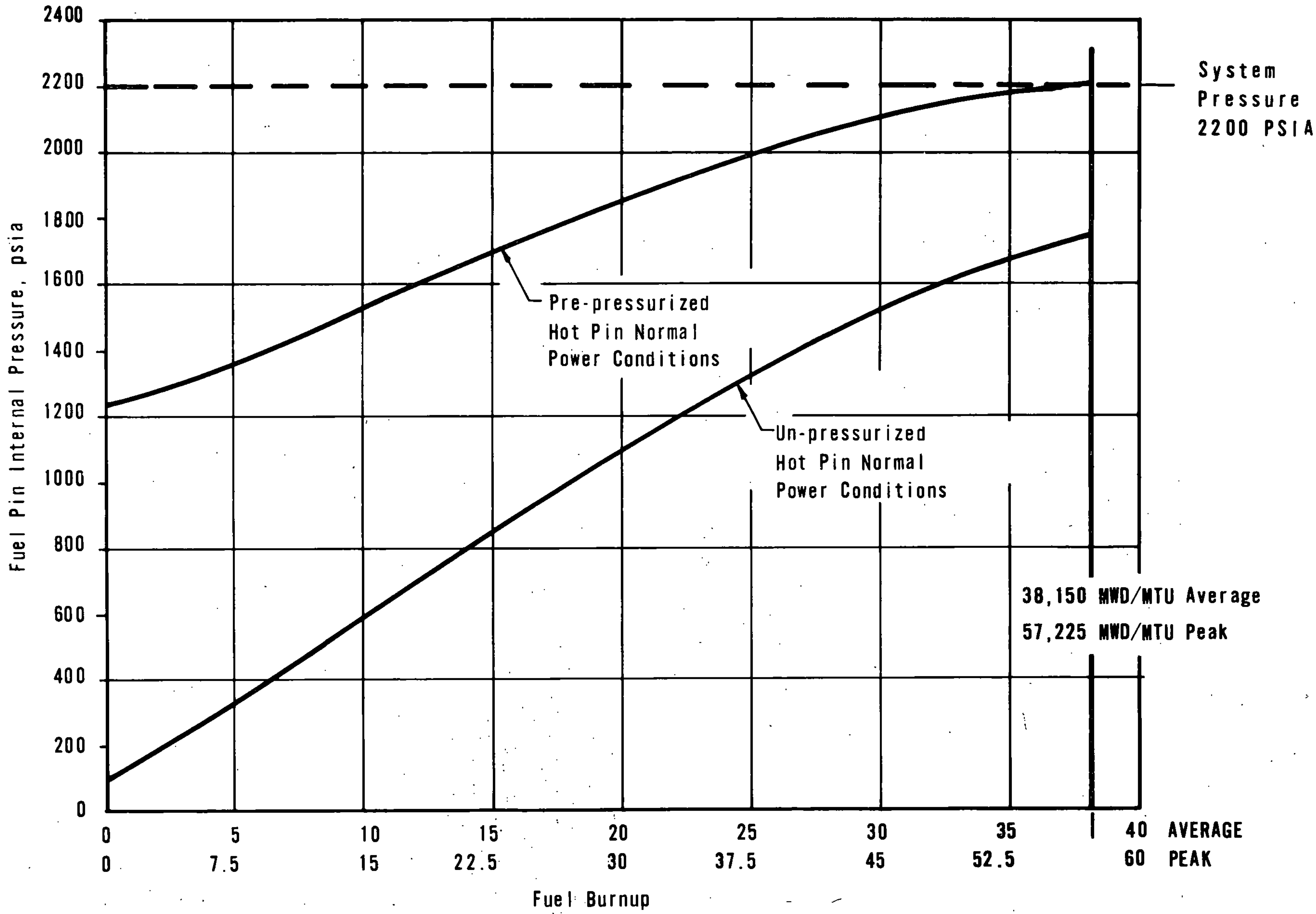
The fuel pressurization is particularly beneficial when considering transients where peak fuel or clad temperatures are important. In particular the peak clad temperature following a LOCA is greatly reduced because the initial fuel temperature prior to the accident is lower for the average pellet and at the hot spot when rods are pressurized. This reduced initial temperature reduces the amount of stored energy in the fuel which must be dissipated during the
24. degraded heat transfer conditions of the blowdown. The effect discussed will apply to the entire core in Units 1, 2, and 3.

3A.5 EXPERIMENTAL VERIFICATION PROGRAM

21. The experimental verification program is described in Section 4 of the Technical Specifications.

3A.5.1 FISSION GAS RELEASE

The fission gas release model is discussed in detail in Section 3.2.3.2.3(h). The design release rate curve (Figure 3-36) used to evaluate internal pressure buildup for both pressurized and unpressurized fuel rods is based upon the upper limit of the available test data, and contributes conservatism to the rod internal pressure calculations. Supplemental gas release data is being obtained from the B&W High Burnup Irradiation Program described in Sections 3.2.4.2.1 and 3.3.3.3.3.



HOT FUEL PIN INTERNAL PRESSURE
NOMINAL COLD DIAMETRICAL CLEARANCE

OCONEE NUCLEAR STATION
Figure 3A - 2