UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

GENERAL ELECTRIC COMPANY

AND

SOUTHWEST ATOMIC ENERGY ASSOCIATES

DOCKET NO. 50-231

PROVISIONAL OPERATING LICENSE

License No. DR-15

The Atomic Energy Commission having found that:

a. The application for provisional operating license, as amended (Amendment Nos. 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18, 19, 20, 21, 22, and 23 to the license application, dated July 25, 1967, December 5, 1967, December 27, 1967, January 18, 1968, February 14, 1968, February 29, 1968, February 29, 1968, March 2, 1968, March 11, 1968, April 25, 1968, May 17, 1968, May 24, 1968, July 18, 1968, August 6, 1968, September 25, 1968, September 27, 1968, September 30, 1968, October 1, 1968, and October 9, 1968, respectively) complies with the requirements of the Atomic Energy Act of 1954 as amended (the "Act") and the Commission's regulations set forth in Title 10, Chapter 1, CFR;

b. The facility has been constructed in accordance with the application, as amended, and the provisions of Provisional Construction Permit No. CPPR-17;

c. There are involved features, characteristics, and components as to which it is desirable to obtain actual operating experience before the issuance of an operating license for the full term requested in the application;

d. There is reasonable assurance (i) that the facility can be operated at power levels up to a maximum of 1 megawatt thermal in accordance with this license without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

e. The applicants are technically and financially qualified to engage in the activities authorized by this license, in accordance with the rules and regulations of the Commission;

f. The applicants have furnished proof of financial protection to satisfy the requirements of 10 CFR, Part 140;

g. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

Provisional Operating License No. DR-15 is hereby issued to the General Electric Company (General Electric) and Southwest Atomic Energy Associates (SAEA) as follows:

1. This license applies to the plutonia-urania-fueled, fast-spectrum, sodium-cooled experimental reactor (the "facility"). The facility, known as the Southwest Experimental Fast Oxide Reactor (SEFOR), is located in Cove Creek Township, Washington County, Arkansas, approximately 19 miles southwest of Fayetteville, Arkansas, and is described in license application Amendment No. 5 "Facility Description and Safety Analysis Report," Volumes I and II, as supplemented by Amendment Nos. 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18, 19, 20, 21, 22, and 23. Said "Facility Description and Safety Analysis Report" in Amendment No. 5, as supplemented and amended, is hereinafter referred to as the Safety Analysis Report.

2. Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities", and subject to the conditions and requirements incorporated herein, the Atomic Energy Commission (the "Commission") hereby licenses:

   A. General Electric Company and Southwest Atomic Energy Associates to acquire and possess legal title to the facility as a utilization facility; and

   B. General Electric Company and Southwest Atomic Energy Associates, with General Electric acting for itself and for SAEA:

      1. To possess, use, and operate the reactor as a utilization facility at power levels up to one megawatt thermal;

      2. To receive, possess, and use at any one time up to 550 kilograms of plutonium as contained in SEFOR fuel elements, foils and sources, and 0.5 kilogram of uranium 235 as foils, check sources or in instruments in connection with operation of the facility pursuant to the Act and 10 CFR Part 70, "Special Nuclear Material".

3. To receive, possess, and use [illegible] curies of cobalt 60 as sealed calibration sources and 2500 kilograms of natural and/or depleted uranium as contained in SEFOR fuel elements, foils and instrument check sources in connection with operation of the facility pursuant to 10 CFR Part 30, "Rules of General Applicability to Licensing of By-Product Material" and Part 40, "Licensing of Source Material".

4. To possess, but not to separate, such by-product and special nuclear material as may be produced by operation of the facility, pursuant to the Act and 10 CFR Parts 30 and 70.

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Level

General Electric is authorized to operate the facility at steady state power levels up to a maximum of 1 megawatt thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A attached hereto are hereby incorporated in this license. General Electric shall operate the facility in accordance with the Technical Specifications and may make changes therein only when authorized by the Commission in accordance with the provisions of Section 50.59 of 10 CFR Part 50.

C. Reports

In addition to reports otherwise required under this license and applicable regulations:

(1) General Electric shall inform the Commission of any incident or condition relating to the operation of the facility which prevented or could have prevented a nuclear system from performing its safety functions

as described in the Technical Specifications. For each such occurrence, General Electric shall promptly notify by telephone or telegraph the Director of the appropriate Atomic Energy Commission Regional Compliance Office listed in Appendix D of 10 CFR Part 20 and shall submit within ten (10) days a report in writing to the Director, Division of Reactor Licensing ("Director, DRL") with a copy to the Division of Compliance.

(2) General Electric shall report to the Director, DRL, in writing within thirty (30) days of its observed occurrence any substantial variance disclosed by operation of the facility from performance specifications contained in the Safety Analysis Report or the Technical Specifications.

(3) General Electric shall report to the Director, DRL in writing within thirty (30) days of its occurrence any significant changes in transient or accident analysis as described in the Safety Analysis Report.

(4) As soon as possible after the completion of six months of operation of the facility (calculated from the date of initial criticality), General Electric shall begin submitting reports in writing in accordance with the requirements of the Technical Specifications.

D. Records

General Electric shall keep facility operating records in accordance with the requirements of the Technical Specifications.

4. This license is effective as of the date of issuance and shall expire eighteen (18) months from said date, unless extended for good cause shown, or upon the earlier issuance of a superseding operating license.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Attachment:
Appendix A, Technical Specifications

Date of Issuance: MAR 4 1989

Received w/Ltr Dated

APPENDIX A

TO

PROVISIONAL OPERATING LICENSE DR-15

TECHNICAL SPECIFICATIONS

FOR THE

SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR

GENERAL ELECTRIC COMPANY

AND

SOUTHWEST ATOMIC ENERGY ASSOCIATES

DOCKET NO. 50-231

TECHNICAL SPECIFICATIONS

FOR THE

SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR

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INTRODUCTION

These technical specifications, which have been prepared in accordance with the requirements of 10 CFR 50.36, incorporate the significant safety limits, functional performance requirements, operating limits, administrative requirements and surveillance schedules applicable to the Southwest Experimental Fast Oxide Reactor referred to as SEFOR.

The reactor is designed to operate at approximately 20 megawatts thermal (MWt) but initial operation will be limited to a power level of 1 MWt in order to permit initial fuel loading and preliminary testing.

These specifications will be modified and supplemented at such time as the operating license is amended to authorize operation at a power level of 20 MWt.

Section 1

DEFINITIONS

1.1 Operable

A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

1.2 Operating

A system or component shall be considered to be operating when it is performing its intended function in its required manner.

1.3 Reactor Conditions

The reactor shall be considered to be Secured or Shutdown as indicated below. The reactor shall be considered operating under all conditions not covered in 1.3(A) and 1.3 (B) below.

A. Secured

Secured shall mean that either or both of the following requirements are satisfied:

(1) The reactor contains less than the minimum amount of fuel required to achieve criticality with sodium in the reactor.[1]

(2) At least nine reflector segments are fully lowered, the reactor coolant temperature is 300°F or higher, and the reactor operate mode switch is locked in the "SECURED" position.

B. Shutdown

Shutdown shall mean that either or both of the following requirements are satisfied.

(1) At least nine reflector segments are fully lowered, the reactor coolant temperature is 300°F or higher, and a senior licensed operator is in charge of operations.

(2) At least eight reflector segments are fully lowered, the reactor coolant temperature is 300°F or higher, the reactor operate mode switch is in the "REFUELING" position, and a senior licensed operator is in charge of operations.

1.4 Reactor Shutdown Actions

A. Rundown

A hydraulically driven, automatic lowering of the reflector segments one at a time.(2)

B. Scram

An automatically or manually initiated lowering of all operating reflector segments at the maximum rate.(3)

1.5 Reactor Power

The rate at which heat is generated within the reactor vessel.

1.6 Rated Flux

The neutron flux that corresponds to a steady state reactor power level of 20 MWt.

1.7 Containment Integrity

The SEFOR containment system consists of an inner and outer barrier. Containment integrity means that all of the following conditions are satisfied:

A. The automatic containment isolation valves on the ventilation lines through the outer barrier are operable or are closed.

B. All doors in the inner containment barrier are closed and operating, the valves in the purge lines bypassing the batch tanks are closed and locked, and all batch tanks are either operable or have all their batching valves closed.

C. At least one door in each air lock through the outer containment barrier is closed and both are operable.

D. The 12'3" diameter equipment door in the outer containment barrier is closed and operating.

E. The atmosphere within the inner containment barrier contains less than 5 V/o oxygen.[4]

F. Leakage rates through the inner and outer containment barriers are equal to or less than the maximum allowable rates specified in Section 3.

1.8 Abnormal Occurrence

Abnormal occurrence shall be defined as any one of the following:

(1) Any operating condition that exceeds a limiting safety system setting specified in Section 2.

(2) Any operating condition that violates a limiting condition for operation specified in Section 3.

(3) Any unplanned reactor scram.

(4) Any uncontrolled or unplanned release of radioactive material.

1.9 Written Order

A written instruction approved and signed by the SEFOR Facility Manager or his designated representative.

1.10 Degree of Redundancy

Degree of redundancy is defined as "R" in the formula, R=N-M, where N is the total number of channels which are able, singly or in coincidence, to perform a required safety action, and M is the minimum number of such channels which when tripped, will always perform the required safety action. For purpose of calculating R, a tripped channel shall be considered inoperable.

1.11 Protection Instrument Channel

The arrangement of components and sensors required to generate a single trip signal related to a plant condition requiring protective action.

1.12 PROTECTION LOGIC SUB-CHANNEL - The arrangement of relay contacts from the respective protection instrument channels, including the coils of the devices operated by such contacts.

1.13 PROTECTION LOGIC CHANNEL - The arrangement of contacts of the devices operated by the respective Protection Logic Sub-Channel, and including the feeders and test devices for the trip actuators.

1.14 MODE SWITCH - A control switch which provides electrical contacts to control the bypass relays. These contacts, together with permissive function contacts, allow the bypass relays to be energized and thereby provide a bypass of the safety function contacts.[5]

1.15 Channel Check - A visual inspection of the output analog signal indication during channel operation to determine that the channel is functioning properly.

If a surveillance function should not be performed because of an extended shutdown that function shall be performed before reactor operation is resumed, except that the channel check may be made after the applicable channel is operating.

1.16 Channel Test - Initiation of an input signal that will cause the instrument channel to respond through the preset trip points and thus determine that the trips respond correctly at the prescribed levels.

1.17 Channel Calibration - Adjustment of the channel instrumentation such that it responds in accordance with design requirements. Calibration shall be deemed to include the Channel Test as described above.

1.18 Gage Pressure, psig

Differential pressure measured with respect to the ambient pressure surrounding the named vessel.

1.19 FRED

The Fast Reactivity Excursion Device used to perform excursion tests.

1.20 Excursion Test

A planned test in which reactor power is rapidly increased as a result of use of a specially built device (FRED) installed in the center drywell of the reactor.

1.21 Oscillator Test

A planned test in which reactivity, main primary coolant flow rate, and main secondary coolant flow rate, are varied in a sinusoidal manner, either independently or in conjunction with each other.

References:

(1) SEFOR FDSAR, Volume II, paragraph 12.2.1, item 1, page 12-3; and Table XII-1, 0 Power, page 12-2.

(2) SEFOR FDSAR, Volume I, paragraph 10.2.2.2.3, page 10-5.

(3) SEFOR FDSAR, Volume I, paragraph 4.5.2.7, page 4-50.

(4) SEFOR FDSAR, Volume I, paragraph 7.2.2, page 7-1.

(5) SEFOR FDSAR, Volume I, paragraph 10.3.1.2.4, page 10-23; and Table X-5, page 10-20.

2.1 Safety Limits

Applicability

Applies to process variables which affect the integrity of the primary system.

Objective

To assure the protection of the primary process system barriers against uncontrolled release of radioactivity.

Specification

A. The maximum permissible reactor core flux shall be 110% of rated flux, except that the limit does not apply during an excursion test initiated by FRED.

B. The maximum permissible reactor core flux during an excursion test after the FRED is fired shall be 6.25×10^3 times rated flux.

C. The maximum permissible integrated energy deposited in the core during an excursion test shall not exceed the limit shown in Figure 2.1-1. The limit is exceeded when reactor conditions result in a point above the limit line.

D. The reactor vessel outlet coolant temperature shall not exceed 1050°F.

E. The maximum permissible reactor vessel cover gas pressure shall be 45.5 psig.

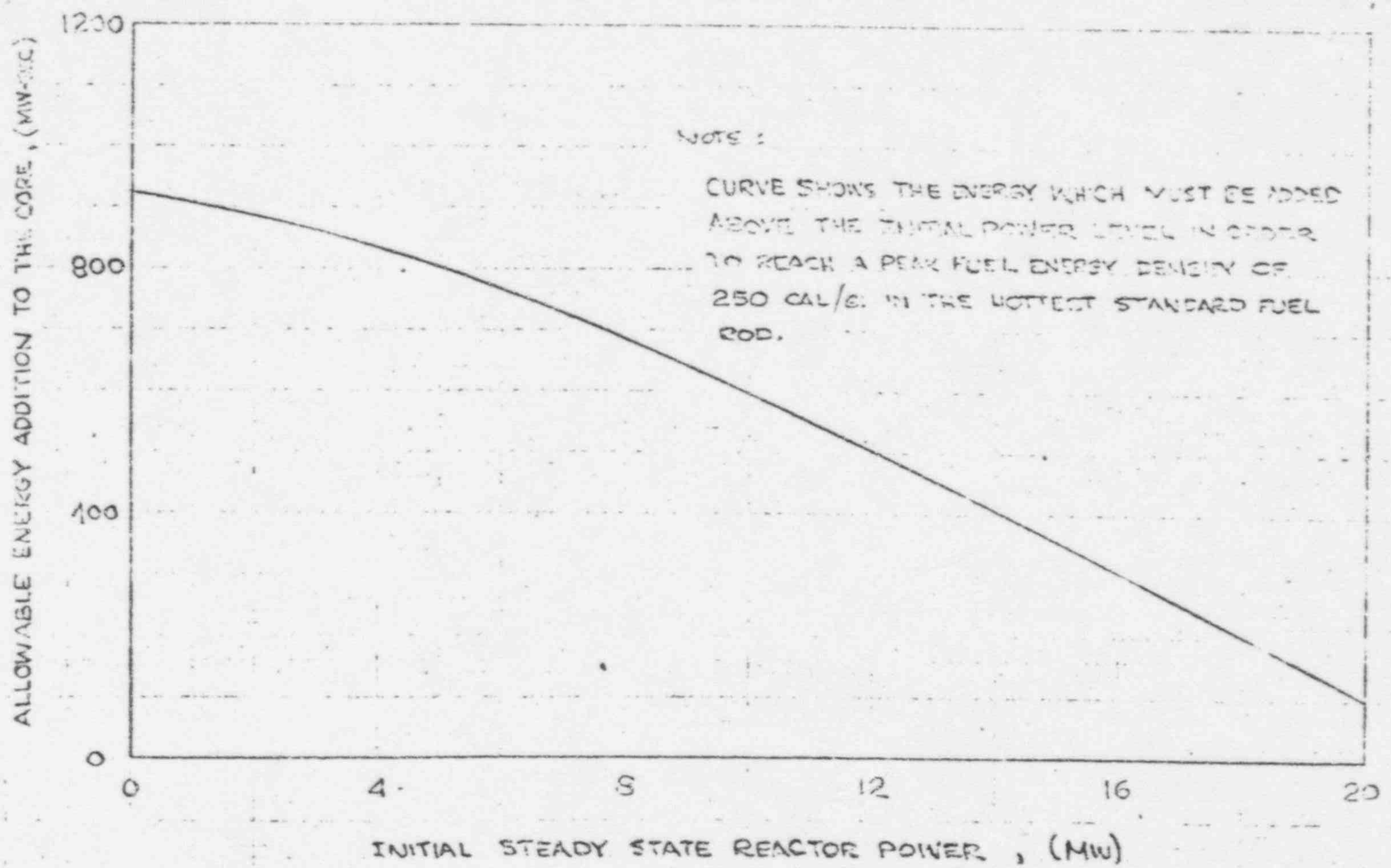


FIG. 2.1-1 : ALLOWABLE ENERGY ADDITION FOR PLANNED TRANSIENT TESTS

2.2 Limiting Safety System Settings

Applicability

These limits apply to trips which scram the reactor

Objective

To prevent reactor process parameters from reaching safety limits.

Specification

The Limiting Safety System Settings shall be as given in Table 2.2-1

TABLE 2.2-1

SCRAM FUNCTION

FUNCTION		SAFETY SYSTEM SETTINGS
High Flux, Wide Range Monitor	$\leq$	105% of Rated Flux
Lower Level, Reactor Sodium	$\leq$	3 inches below lip of operating level overflow pipe
High Temperature, Core Outlet-Upper Region	$\leq$	900°F
Low Flow, Main Primary	$\leq$	20% below the operating flow set point
High Temperature, Reflector Region	$\leq$	350°F for thermocouples on the reflector guide structure inner diameter and radial web.
	$\leq$	275°F for thermocouples on the reflector guide structure, outer diameter.

Section 3

LIMITING CONDITIONS FOR OPERATION

These requirements specify the minimum performance capability of each system required for safe reactor operation.

3.1 Reactor Safety System

Applicability

Applies to the reactor safety system.

Objective

To assure that process parameters will not exceed safety limits.

Specification

The limiting conditions for operation shall be as specified in Tables 3.1-1 through 3.1-6.

TABLE 3.1-1

INSTRUMENTATION THAT INITIATES SCRAM ACTION

Function	I Minimum Number of Operable Channels	II Minimum Degree of Redundancy	III Conditions Permitting Bypass
High Flux, Wide Range Monitor	3	2	None
Loss of Power, Bus 2A	2	1	1) MASTER mode switch in NORMAL, and 2) OPERATE mode switch in SECURED, and 3) all ten reflector segments full down.
Loss of Power, Main 2.4 KV Bus	2	1	None
Low Level, Reactor Sodium Level	3	2	1) MASTER mode switch in NORMAL, and 2) OPERATE mode switch in SECURED, and 3) all ten reflector segments fulldown, or 1) MASTER mode switch in NORMAL, and 2) OPERATE mode switch in REFUELING, and 3) eight reflector segment full down, and 4) reactor sodium below 400°F, and 5) primary loop pressure below 1 psi. or

Table 3.1-1 (Continued)

Function	I Minimum Number of Operable Channels	II Minimum Degree of Redundancy	III Conditions Permitting Bypass
			1) MASTER mode switch in NORMAL, and 2) OPERATE mode switch in ZERO AND LOW POWER, and 3) High flux scram trip set to 500 kw or less, and 4) The reactor vessel head is removed.
High Temperature, Core Outlet-Lower Region	3	2	None
High Temperature, Core Outlet-Upper Region	3	2	None
Low Flow, Main Primary Loop	2	1	1) MASTER mode switch in NORMAL, and 2) OPERATE mode switch in SECURED, and 3) all ten reflector segments full down, or 1) MASTER mode switch in NORMAL, and 2) OPERATE mode switch in ZERO AND LOW POWER or 1) MASTER mode switch in NORMAL, and 2) OPERATE mode switch in REFUELING, and 3) eight reflector segments full down, and 4) reactor sodium below $400^{o}F$, and 5) primary loop pressure below 1 psi.

Table 3.1-1 (Continued)

Function	I Minimum Number of Operable Channels	II Minimum Degree of Redundancy	III Conditions Permitting Bypass
Sodium Leak, Auxiliary Primary Loop and	6* 6	6* 5	None None
Low Level, Main Secondary Expansion Tank	2	1	None
Low Level, Auxiliary Secondary Expansion Tank	2	1	None
High Temperature, Main Secondary Cold Leg	2	1	None
Low Flow, Main Secondary Loop	2	1	Same as for "Low Flow, Main Primary Loop"
High Temperature, Reflector Region	9	8	None
Low or High Pressure, Reflector Drive Accumulators	9	8	None
Very High Radiation, Containment Ventilation Exhaust	2	1	None
Nitrogen Radiation Monitor**	1	1	None

*Applies only to circuits which act in coincidence with the nitrogen radiation monitor.

**Used only in coincidence with sodium leak detectors in the auxiliary primary loop.

Table 3.1-1 (Continued)

Function	I Minimum Number of Operable Channels	II Minimum Degree of Redundancy	III Conditions Permitting Bypass
Protection Logic Sub-Channel (Busses A, B, & C)	3	2	None
Protection Logic Sub-Channel (Busses D, E, & F)	2	1	None
Protection Logic Channel (Solenoid Busses Test Sections 1 & 2)	2	1	None

TABLE 3.1-2

TRIP SETTINGS FOR SCRAM ACTION

FUNCTION		TRIP SETTING
Loss of Power, Essential Bus 2A	≥	380 Volts
Loss of Power, Main 2.4 KV Bus	≥	1.8 Kilovolts
High Temperature, Core Outlet-Lower Region	≤	900°F
Leak, Auxiliary Primary Loop		Electrical Short
Low Level, Main Secondary Expansion Tank	≤	17 inches below sodium level at design power
Low Level, Auxiliary Secondary Expansion Tank	≤	7 inches below sodium level at design power
High Temperature, Main Secondary Cold Leg	≤	800°F
Low Flow, Main Secondary	≤	20% less than the operating flow set point
High or Low Pressure, Reflector Drive Accumlators	≤ ≥	225 psi (high) 150 psi (low)
Very High Radiation, Containment Ventilation Exhaust	≤	10 X Radiation Level at design power level

TABLE 3.1-3

INSTRUMENTATION THAT INITIATES A "BLOCK RAISING OF THE REFLECTOR SEGMENTS"

Function	I Minimum Number of Operable Channels	II Minimum Degree of Redundancy	III Conditions Permitting Bypass
Low Flux Source, Range Monitor	2	1	May be bypassed when either intermediate range monitor is upscale by at least one decade, or when the reactor vessel head is removed and special startup instrumentation is installed in the center drywell
Low Flux, Wide Range Monitor	3	2	May be bypassed when all wide range monitor range switches are on or below (more sensitive) a position that results in an indication between 30% and 100% of full scale at 0.2 watts.
"Operate" mode bypass switch in the "SECURED" position	2*	1*	None
Protection Logic Sub-Channel	2	1	None
Protection Logic Channel	2	1	None

*Redundant contacts on the switch.

TABLE 3.1-4

TRIP SETTINGS TO BLOCK REFLECTOR RAISE ACTION

FUNCTION	TRIP SETTINGS
Low Flux, Source Range Monitor	$\geq$ 10% of Full Scale Linear
Low Flux, Wide Range Monitor	$\geq$ 10% Full Scale

TABLE 3.1-5

INSTRUMENTATION THAT INITIATES CONTAINMENT ISOLATION ACTION

Function	I Minimum Number of Operable Channels	II Minimum Degree of Redundancy	III Conditions Permitting Bypass
High Radiation, Containment Vent Exhaust	2	1	None
Very High Radiation, Containment Vent Exhaust	2	1	None
Protection Logic Sub-Channels (Busses J, K, & L)	2	1	None
Protection Logic Channels (Solenoid Busses A & B)	2	1	None

TABLE 3.1-6

TRIP SETTINGS FOR CONTAINMENT ISOLATION ACTION

FUNCTION	TRIP SETTING
High Radiation, Containment Ventilation Exhaust	$\leq$ 2 X radiation level at normal design power level
Very High Radiation Level, Containment Ventilation Exhaust	$\leq$ 10 X radiation level at normal design power level

3.2 Reactor Control System

Applicability

Applies to the reactor control system.

Objective

To assure proper operation of the reflector segments and the availability of adequate shutdown margin.

Specification

A. At least nine reflector segments and their associated drives shall be operable.

B. The insertion rate of any reflector segment shall be less than 1.2 inches per second.

C. Upon initiation of a scram signal, the time required, including safety system response time, for each operable reflector segment to move through 90% of its stroke from the fully raised position shall not exceed one second when the Master Mode Switch is in either the Normal or Oscillator Test position and 1.4 seconds when it is in the Excursion Test position.

D. Whenever outer containment integrity is intentionally breached, the the electrical power circuit to the reflector control hydraulic power supply shall be de-energized.

H. Guinea pig fuel rods of 25.0% plutonium enrichment shall only be located below the six refueling ports. No guinea pig rods shall be located under the three innermost refueling ports during steady state reactor operations above 17.5 MWt.

I. No fuel rods shall be placed in the center drywell.

J. Fission chambers, experimental foils or oxide fuel samples having a total reactivity worth of less than 60¢ and containing a total of not more than 0.5 Kg fissile material may be placed in the center drywell for irradiation at power levels equal to or less than 100 KWt. Experimental foils containing less than 10 mg of fissile material may be irradiated at reactor power levels above 100 KWt.

K. Fuel rods which have defects as defined below shall not be reinserted in the core:

   1. Visual observation of cladding rupture, cladding perforation or other visual defects may cast reasonable doubt on the integrity of the rods.

   2. Local swelling of the cladding is in excess of 10 mils or bowing of the rod is sufficient to prevent re-insertion of the rod into the core.

   3. The column height of either fuel segment has increased by more than 1/2 inch.

3.3 Reactor Core

Applicability

Applies to reactor core loading configurations.

Objective

To assure that core physics parameters remain within the expected range and that fuel rod cladding integrity is maintained.

Specification

A. The reactor shutdown margin at 350°F shall be equal to or greater than 1 $ and extrapolation of data obtained at or above 350°F shall demonstrate that the reactor would be subcritical at 300°F with one operable reflector segment raised to its most reactive position.

B. The excess reactivity available at rated power (20 MWt) shall be equal to or less than 0.5 $ when the core inlet temperature is at 700°F.

C. The reactor power coefficient of reactivity at constant inlet temperature and constant coolant flow rate shall be negative.

D. The isothermal temperature coefficient of reactivity at "zero" power shall be negative.

E. Following initial operations at a power level of 10 MWt, the reactor shall not be operated unless operating data from SEFOR demonstrate that the net non-fuel coefficient is negative and that the Doppler coefficient is negative with a magnitude equal to or greater than 0.005.

F. The reactor shall have a phase margin of at least 30 degrees at the point where the Nyquist plot crosses the unit circle.

G. The reactor shall have at least 600 fuel rods in the core at power levels above 10 MW.

3.4 Sodium Coolant System

Applicability

Applies to the main and auxiliary sodium coolant systems and the irradiated fuel storage tank.

Objective

To assure reliable and adequate cooling of the core and to limit potential radiological effects of the primary sodium.

Specification

A. Each primary and secondary sodium coolant loop shall be operable and the sodium temperature shall be 300°F or greater.

B. The pump-around loop shall be operable.

C. The argon cover gas system and the argon vent vacuum pump shall be operable.

D. The fission product monitor shall be operating at power levels above 1 MWt.

E. The plugging temperature in each coolant loop and the irradiated fuel storage tank shall not exceed 425°F, and the plugging temperature be at least 25°F below the sodium coolant temperature.

F. The cover gas pressure in the main and auxiliary secondary sodium expansion tanks shall be equal to or greater than the cover gas pressure in the reactor vessel.

G. The sodium leakage rate in the main IHX at normal operating pressures shall be less than 3 gal/hr.

H. The sodium leakage rate in the auxiliary IHX at normal operating pressures shall be less than 3 gal/hr.

I. The cover gas pressure in the reactor vessel shall not exceed 25 psig.

3.5 Containment System

Applicability

Applies to the operating status of the inner and outer containment barriers.

Objective

To minimize and limit the inadvertent release of radioactive materials to the environs.

Specification

A. Containment integrity shall be maintained when either of the following conditions exist:

   1. The reactor is operating.

   2. When the reactor head or the irradiated fuel storage tank cover is not in place.

B. The following exceptions to specification A.2, above, shall be permitted.

   1. The equipment door in the outer containment may be left open while the fuel transfer cask is being used to transfer material to or from the refueling cell provided the reactor head is in place.

   2. The marine hatch in the man access panel in the refueling cell may be used to transfer material into or out of the refueling cell during the core loading process prior to initial reactor operation. The hatch shall be closed whenever reflector segments are inserted.

   3. If the reactor head or the irradiated fuel tank cover is not in place and equipment failure prevents replacement, inner containment may be temporarily breached to effect repairs.

C. The horizontal transfer port shall not be used.

D. The specified leakage rates for the outer and inner containments shall be as follows:

Outer Containment:

1. L_{t_o} = 1.2% of V_o in 24 hours at 10 psig.

2. L_o = 1.4% of V_o in 24 hours at 10 psig.

Inner Containment:

3. L_{t_i} = 14.8% of V_i in 24 hours at 10 psig.

4. L_i = 16.5% of V_i in 24 hours at 10 psig.

The values to be used for containment volumes are:

V_o = 70,000 cu. ft. for the outer containment

V_i = 73,000 cu. ft. for the inner containment

E. The reactor shall not be made critical if the leakage rate of the inner containment exceeds L_{t_i} or if the leakage rate of the outer containment exceeds L_{t_o}.

F. If the measured leakage rate of the outer containment exceeds L_{t_o}, but does not exceed L_o, individual penetrations shall be repaired as necessary, and the integrated leak test shall be repeated until the measured leakage is less than L_{t_o}.

G. If themeasured leakage rate of the outer containment exceeds L_o, the procedure described in F shall be followed. In addition, the surveillance period for those penetrations which require repairs in order to reduce the containment leakage below L_{t_o} shall be shortened to two months for a period of one year.

H. If the measured leakage rate of the inner containment exceeds L_{t_i}, but does not exceed L_i, individual penetrations shall be repaired as necessary, and the integrated leak test shall be repeated until the measured leakage is less than L_{t_i}.

I. If the measured leakage rate of the inner containment exceeds L_i, the procedure described in H shall be followed. In addition, the surveillance period for testable penetrations of the inner containment which require rep...r in order to reduce the leakage rate below L_i shall be shortened to two months for a period of one year.

J. The leakage rate for a single electrical or piping penetration through the outer containment shall not exceed 1/87 times L_{t_o}. The total leakage rate for these penetrations shall not exceed 0.4 times L_{t_o}.

K. The total leakage rate through the group of 15 pipe tunnel penetrations which penetrate both containment barriers shall not exceed $\frac{15}{8700}$ times L_{t_o}.

L. The leakage rate through each of the following five groups of components shall not exceed 0.1 times L_{t_o}.

1. The doors of the personnel lock.
2. The doors of the emergency escape lock.
3. The equipment door.
4. The vacuum breaker valves.
5. The reactor building ventilation valves.

M. The leakage rate through each pressure door in the inner containment shall not exceed 0.001 times L_{t_i}.

N. The leakage rate through each testable assembly in the man access panel shall not exceed 0.001 times L_{t_i}.

O. The oxygen content of the inner containment atmosphere shall be less than 5% by volume.

P. The freon content of the inner containment atmosphere shall be less than 1000 ppm.

Q. The water content of the inner containment Argon atmosphere shall be less than 125 ppm.

R. The dewpoint of the inner containment nitrogen atmosphere shall be less than 60^oF.

S. The rate of change in temperature of the inner containment or outer containment atmospheres shall be less than 5^oF per hour if the temperature change exceeds 50^oF.

3.6 Electrical Systems

Applicability

Applies to electrical power supply system including emergency power supplies.

Objective

To assure a reliable source of power for operation of vital equipment during reactor operation.

Specification

A. The 69 kV, 2.4 kV, and 480 V systems shall be energized.

B. The 125 V dc, + 26.5 V dc, and ± 26.5 V dc systems shall be operable with batteries for each system fully charged, and the battery chargers for each system shall be in service.

C. The main emergency diesel generator shall be capable of delivering rated power to the 480 V system within 1 minute after receipt of a start signal.

D. The total amount of diesel fuel in the underground diesel fuel storage tank and the main emergency diesel day tank shall be at least 910 gallons.

E. The auxiliary diesel shall be operable whenever the reactor is operated above 5 MWt.

3.7 Radioactive Waste Control System

Applicability

Applies to those components which control the collection, storage, and release of radioactive waste materials.

Objective

To assure the capability for safe control of radioactive waste materials and to define the limiting conditions for release of effluents from the reactor system.

Specification

A. At least one of the three waste gas compressors shall be operable.

B. For reactor startup, at least two waste gas compressors shall be operable.

C. The rate of discharge of radioactive effluents from the plant stack shall not exceed:

1. Annual average release rate, except halogens and particulates with a half-life greater than 8 days:
   $4.0 \times 10^{10} (\sum_x C_x)\ \mu Ci/sec.$ [(1)]
2. For periods less than 48 hours in any seven consecutive days, hourly average release rate, except halogens and particulates w th half-lives greater than 8 days:
   $1.7 \times 10^{11} (\sum_x C_x)\ \mu Ci/sec.$ [(1)]
3. Annual average release rate of radioactive halogens and particulates with half-lives greater than 8 days: $5.6 \times 10^{7} (\sum_x C_x)\ \mu Ci/sec.$ [(1)]
4. For periods less than 48 hours in any seven consecutive days, hourly average release rate of radioactive halogens and particulates with half-lives greater than 8 days: $5.6 \times 10^{8} (\sum_x C_x)\ \mu Ci/sec.$ [(1)]

[(1)] C_x is the concentration of radioisotope x in $\frac{\mu Ci}{ml}$ and must satisfy $\sum_x \frac{C_x}{(MPC)_x} \leq 1$; where the $(MPC)_x$ are equal to the values in 10 CFR 20.

D. The gross instantaneous activity of the liquid effluent at the point of release to the tile field shall be equal to or less than the MPC_w values specified in 10 CFR 20, Appendix B, Table II. If the isotopic content is unidentified, the gross activity shall be equal to or less than the unidentified mixture MPC_w of 1×10^{-7} μCi/ml given in 10 CFR 20, Appendix B.

E. The waste gas discharge radiation monitors shall be operating during periods of waste gas release.

F. The stack dilution fan shall be operating during periods of waste gas release.

G. The liquid waste radiation monitor shall be operating during periods of liquid waste release.

3.8 Irradiated Fuel Storage Tank

Applicability

Applies to parameters associated with the irradiated fuel storage tank whenever one or more fuel rods are stored in the tank.

Objective

To maintain safe conditions in the irradiated fuel storage tank.

Specification

A. The sodium temperature in the irradiated fuel storage tank shall be maintained between 300°F and 500°F.

B. The sodium level in the tank shall be maintained at or above the opening of the discharge duct attached to nozzle N-6.

C. Items A and B do not apply when the tank contains only new fuel rods and/or fuel rods that have been irradiated at reactor power levels of 100 KWt or less.

D. The criticality factor within the tank shall be less than 0.95.

E. The cover gas pressure in the tank shall be less than 5 psig.

F. The lower surface of the tank cover shield blocks shall not be elevated more than eight feet above the refueling cell floor.

References

1 SEFOR FDSAR, Volume 1, 9.4.4, page 9-8 and Table IX-3

2 SEFOR FDSAR, Supplement 17, Answer to Question M-6, page M-9.

3.9 Operations Conducted with Reactor Vessel Head Removed

Applicability

Applies to handling operations conducted while the reactor vessel head is removed.

Objective

To maintain safe conditions while the reactor vessel head is removed.

Specification

A. Either the main or the auxiliary coolant system shall be operable, unless the conditions specified in 1.3A(1) are satisfied and the core decay heat level is below 1 KWt.

B. The reactor Operate Mode Switch shall not be placed in the "HIGH POWER" position. It may be placed in the "ZERO AND LOW POWER" position when the Wide Range Monitor contains a physical stop which limits the high flux scram set point to 500 KWt or less. The "SECURED" and "REFUELING" positions may be used according to Standard Operating Procedures.

C. No handling operations shall be conducted in the refueling cell when the reactor Operate Mode Switch is in the "ZERO AND LOW POWER" position.

D. Before raising the reflectors or moving fuel in the core, the in-core start-up instrumentation shall register more than six neutron counts/sec with a fuel loading equal to or greater than the first 36 fuel elements loaded into the innermost fuel assemblies (six fuel elements per fuel assembly.)

E. Criticality checks are to be made between loading increments during the initial loading to critical by raising all reflectors and noting the change in count rate on the in-core neutron detector and the Source Range Monitor (when the SRM count rate is in the useable range). After the first two loading increments (108 fuel rods in each increment) no more than one rod in excess of one-half the number of additional rods predicted for criticality will be loaded in one increment. The number of fuel rods for criticality will be predicted by plotting curves of $\frac{\text{plutonium mass}}{\text{count rate}}$ vs. Pu mass for count rate values obtained from at least two detectors with all reflectors fully raised as well as fully lowered. The most conservative plot shall be used to establish loading increments.

F. The reactor vessel sodium temperature shall be less than 450°F.

3.10 Approach to Power

Applicability

Applies to reactor power limits during the initial approach to full power for Core I and Core II.

Objective

To provide a method of assuring a safe and orderly approach to full power.

Specification

Reactor power shall be limited to 1 MWt.

3.11 Oscillator Tests

Applicability

These limits apply to tests in which the rod oscillator mechanism is used to vary one or more reactor parameters on a periodic basis.

Objective

To specify additional limits which are applicable only during oscillator tests.

Specification

A. The amplitude of reactor power oscillation shall be equal to or less than ±20% of the indicated power level.

B. The amplitude of reactivity oscillator shall be equal to or less than ±10 cents.

C. The amplitude of oscillation in the main primary and main secondary coolant flow rate shall be equal to or less than ±1000 GPM.

D. The amplitude of the coolant temperature oscillation at the vessel inlet or at the vessel outlet shall be equal to or less than ±60°F.

3.12 Excursion Tests

Applicability

These limits apply when excursion tests are conducted with the Fast Reactivity Excursion Device (FRED).

Objective

To specify additional limits which are applicable only during excursion tests.

Specification

The FRED device may not be used during operations up to one MWt. Completion of the FRED is not required for such operation.

Section 4

SURVEILLANCE REQUIREMENTS

General

This section specifies the minimum surveillance needed to assure that the limiting conditions for operation are met. The time intervals specified in this section shall be valid only during periods of normal reactor operation and do not apply in the event of reactor shutdown for periods longer than the specified interval between tests. If a surveillance function should not have been performed because of an extended shutdown, that surveillance function shall be performed before reactor operation is resumed, except that channel checks may be made after the applicable system is operating. Specified intervals of three months and four months may be adjusted plus or minus two weeks, and specified intervals of six months or longer may be adjusted plus or minus one month to accommodate normal test schedules. Following repairs or maintenance which could alter or impair the performance of a system, tests shall be performed to verify that the system is operable.

4.1 Reactor Safety System

Applicability

These tests apply to instrumentation and sensors used to monitor parameters associated with reactor or radiological safety.

Objective

To maintain proper calibration of instruments and sensors used in the safety system so that plant parameters do not exceed safety limits, and to assure operability of instruments used to monitor radiological safety.

Specification

A. Channels shall be tested, calibrated, and checked as indicated in Table 4.1-1.

B. Radiation monitors shall be tested and calibrated as indicated in Table 4.1-2.

ABBREVIATIONS USED IN TABLES 4.1-1 AND 4.1-2

ea strt up	- Each start up of the reactor from shutdown.
1/d	- Once per day
1/wk	- Once per week
1/mo	- Once per month
1/3 mo	- Once every three (3) months
1/6 mo	- Once every six (6) months
N/A	- Not Applicable

TABLE 4.1-1

MINIMUM FREQUENCIES FOR TESTING

OF SAFETY INSTRUMENTATION

Channel	Channel Check	Channel Test	Channel Calibration	Remarks
Source Range Monitor	ea strt up	1/wk	1/6 mo	
Wide Range Monitor	1/d	1/wk	1/6 mo	
Undervoltage Relays	N/A	1/mo	1/6 mo	
a) 2.4 KV Main Bus				
b) 480 V Bus 2A				
Sodium Level Probes	1/d(1)	1/mo(2)	1/6 mo	(1) Check of DC current to the probe. (2) Change of process level to effect a level trip.
a) Reactor Level				
b) Aux. Expansion Tank				
c) Main Expansion Tank				
Temperature Monitors	1/d	1/wk	1/6 mo	
a) Reactor Core Outlet Upper Region				
b) Reactor Core Outlet Lower Region				
c) Reactor Cavity				
d) Main Secondary Cold Leg				
Flow Monitors	1/d	1/wk	1/6 mo	
a) Main Primary				
b) Main Secondary				
Pressure Switches	N/A	1/3 mo	1/6 mo	
a) Reflector Accumulator Switch (Hi)				
b) Reflector Accumulator Switch (Lo)				
Reflector Accumulator Leak Detector	N/A	1/3 mo	1/6 mo	
Ventilation Radiation Monitor	1/d	1/wk	1/6 mo	

TABLE 4.1-1 (Cont.)

Channel	Channel Check	Channel Test	Channel Calibration	Remarks
Sodium Leak Detector	N/A	1/mo$^{(1)}$	1/6mo$^{(2)}$	(1) Apply a test short outside containment. (2) Apply a test short at detector connector, and perform continuity test
Manual Scram a) Right Side Button b) Left Side Button	N/A	1/mo	N/A	
Manual Containment Isolation	N/A	1/mo	N/A	
Manual Initiation of Block Raise Action with Operate Mode Switch in "SECURED" position	N/A	1/mo	N/A	
Scram Protection Logic Sub-Channel	N/A	1/mo	N/A	
Scram Protection Logic Channel	N/A	1/mo	N/A	
Containment Isolation Sub-Channel	N/A	1/mo	N/A	
Containment Isolation Channel	N/A	1/mo	N/A	
Block Raise Action Sub-Channel	N/A	1/mo	N/A	
Block Raise Action Channel	N/A	1/mo	N/A	

TABLE 4.1.2

MINIMUM FREQUENCIES FOR TESTING

OF RADIATION MONITORING INSTRUMENTATION

Channel	Channel Check	Channel Test	Channel Calibration	Remarks
Nitrogen Radiation Monitor	1/d	1/wk	1/6 mo	
Liquid Waste Radiation Monitor	1/d	1/wk	1/6 mo	Test prior to each release of radioactive material.
Waste Gas Discharge Radiation Monitors	1/d	1/wk	1/6 mo	Test prior to each release of radioactive material.
Spent Fuel Storage Monitor	N/A	N/A	1/6 mo	Test prior to each movement of fuel to or from the irradiated fuel storage tank.
Area Radiation Monitor	1/d	1/2 wk	1/6 mo	Calibrate with Co-60 source
Refueling Cell Radiation Monitor	1/d	1/wk	1/6 mo	Calibrate with Co-60 source
Cutie Pie Survey Meter	Each time used	N/A	1/mo	Calibrate with radioactive source
Beta-Gamma Survey Meter	Each time used	N/A	1/mo	Calibrate with radioactive source
Alpha and Neutron	Each time	N/A	1/mo	Calibrate with radioactive source

4.2 Reactor Control System

Applicability

Applies to the core configuration and to the reflector control and drive system.

Objective

To assure safe control of the reactor under all operating conditions.

Specification

A. The core loading limits specified in paragraphs 3.3A and 3.3B shall be demonstrated at least once every four months. After each core rearrangement, compliance with these limits shall be checked by use of calibrated control segments and extrapolation to the specified temperature conditions using the best available data. In addition, the daily checks indicated in Section 4.9A shall be reviewed to assure that no reactivity changes have occured which would result in exceeding the limits specified in paragraphs 3.3A, 3.3B and 3.3C.

B. The requirements of Section 3.2 for the reflector segments shall be demonstrated at least quarterly.

C. Before each scheduled reactor startup, all operable reflector segments shall be raised one at a time to a minimum height of ten inches, be driven down and be scrammed from a minimum height of six inches. One reflector segment shall be scrammed from the fully raised position. A different reflector segment shall be chosen each time for the scram from full height.

D. During continuous reactor operation for periods longer than one week, each operable reflector segment shall be moved through at least 25% of its stroke at time intervals not to exceed one week.

4.3 Reactor Fuel Rods

Applicability

Applies to fuel rod examination made in the refueling cell.

Objective

To assure maintenance of fuel rod cladding integrity during reactor operation.

Specification

A. Two or more guinea pig fuel rods which have operated at power densities higher than the power density of standard fuel rods nearest the center of the core shall be removed from the reactor after operation at reactor power levels of 15 and 17.5 MWt, and shall be examined in the refueling cell by visual observation, dimensional checks, and gamma scans. The maximum interval between these examinations, after reaching a power level of 15 MWt, shall be six months.

B. Before the start of the sub-prompt critical excursion tests and before the start of the prompt critical excursion tests, a minimum of one guinea pig fuel rod and one standard fuel rod shall be examined by the methods described in "A" above.

C. After each prompt critical excursion test, at least one guinea pig rod and one standard rod shall be examined by the methods described in "A" above.

D. If the examination of a guinea pig rod should indicate damage as described in Section 3.3K, additional guinea pig rods shall be examined to determine the extent of additional damage, if any.

4.4 Reactor Coolant System

Applicability

This series of tests applies to the sodium coolant systems.

Objective

To provide for surveillance of sodium system components to assure that the limiting conditions for operation are met.

Specifica[illegible]

A. The reactor safety vessel shall be leak tested at intervals not to exceed six months.

B. The irradiated fuel storage tank safety vessel shall be leak tested at intervals not to exceed six months while in service.

C. The auxiliary inlet check valve shall be operationally tested at least quarterly.

D. The check valve in the reactor overflow line shall be operationally tested at least once quarterly.

E. The Marmon clamp on the auxiliary primary reactor vessel outlet dip tube shall be leak tested at least quarterly.

F. The vacuum breaker valve for the reactor cover gas shall be functionally tested at least quarterly.

G. Bar-type tensile specimens shall be removed from the reactor vessel following the 3rd, 4th, 5th, 8th, and 10th year of reactor operation and subjected to specified tests. (1) (2)

H. The plugging temperature of the primary sodium system shall be measured daily when the plugging temperature exceeds $400^{o}F$ and at intervals not to exceed one week when the plugging temperature is below $400^{o}F$.

I. The plugging temperatures of the main secondary and auxiliary secondary sodium systems shall be measured daily when the plugging temperature exceeds 400°F, weekly when the plugging temperature is between 300°F and 400°F, and monthly when the plugging temperature is below 300°F.

J. The plugging temperature of the sodium in the irradiated fuel storage tank shall be measured weekly when the plugging temperature exceeds 400°F, and monthly when the plugging temperature is below 400°F.

References

(1) SEFOR FDSAR, Supplement 16, Section VII, pp 7-1 ff.

(2) SEFOR FDSAR, Supplement 19, Answer to Question 9, pp 67 ff.

4.5 Containment System

Applicability

Applies to inner and outer containment barriers, including penetrations, isolation valves and high velocity check valves.

Objective

To determine that the containment system continues to meet specifications with regard to allowable leakage and valve operation.

Specification

A. Inner and Outer Containment Leak Tests

A containment leak test shall be performed annually for each barrier at a pressure differential of 10 psig across each containment barrier.

B. Outer Containment Penetration Leak Tests

1. The following penetrations shall be tested quarterly for leakage at the indicated pressures:
   a. Inner and outer doors of the personnel lock and emergency escape lock: 10 psig and 1 psig.
   b. Equipment door: 30 psig and 1 psig, quarterly and following each closure.
   c. Vacuum breaker valves and reactor building ventilation valves: 30 psig and 1 psig.
   d. Piping and electrical penetrations through the outer containment: 30 psig.
2. Piping and electrical penetrations may be tested individually or may be manifolded together in groups and tested simultaneously.

3. Penetration leakage rates shall be calculated from the pressure decay rate, based on the free volume contained in the penetration(s) and piping used for the test. Test periods of one hour or longer shall be used to determine leakage rates.

4. If the leakage rate for a group of penetrations shows that leakage through one or more penetrations may exceed the limits specified in section 3.5, the penetrations in that group shall be tested individually.

5. Individual penetrations shall be leak tested as specified above, whenever they are modified or repaired.

C. Inner Containment Penetration Leak Tests

1. Pressure doors in the inner containment shall be leak tested quarterly and prior to reactor startup following final door closing by pressurizing the cavity between double seals.

2. Man-access panel glove port covers, the marine hatch, and the neck ring assemblies shall be leak tested quarterly and prior to reactor startup following use of this equipment.

3. The man-access panel mounting gasket, windows, helmet, base rings, and electrical penetrations in this panel shall be tested using soap-bubble technique or its equivalent with normal refueling cell to air zone differential pressure each time they are replaced. Leaks shall be repaired without undue delay.

4. All other penetrations of the inner containment shall be checked as part of the annual inner containment leak test.

D. Isolation Valves

All isolation valves which are actuated by the reactor safety circuitry shall be checked quarterly for closing in response to manual scram or simulation of two out of three signals from any one of the following sources:

High pressure nitrogen header

High radiation containment vent exhaust

Very high radiation containment vent exhaust

Malfunction vent radiation monitor

High pressure containment building.

E. High-Velocity Check Valves

High velocity check valves in the argon and nitrogen systems shall be checked annually for closing in response to a differential pressure at the orifice taps corresponding to a flow rate of 2000 scfm or on manual initiation from the control panel.

F. Nitrogen Cooling Refrigerant Isolation Valves

The supply line solenoid valves and return line back pressure valves shall be tested semi-annually for closing in response to loss of pressure in the liquid supply lines.

G. Waste Gas Discharge Filter

Pressure drop across the waste gas discharge filter shall be monitored each time gas is released from the decay tanks, and the trend shall be plotted. A decrease in pressure drop may be indicative of filter damage and shall be cause for filter inspection (and if damaged, replacement) prior to further usage.

H. <u>Oxygen, Water, and Freon Content of Inner Containment</u>

1. The oxygen and Freon content of the inner containment atmosphere shall be monitored daily.
2. The water content of the argon region of the inner containment shall be monitored daily.
3. The water content of the nitrogen region of the inner containment shall be monitored at least weekly.

4.6 Emergency Electrical Power System

Applicability

These tests apply to the emergency power systems.

Objective

To assure availability of the emergency power system at all times.

Specification

A. Main Diesel Generator 480 V AC Supply

1. The diesel generator shall be started and loaded to 540 kw at monthly intervals.

2. Automatic emergency starting of the main diesel generator shall be demonstrated once each month.

3. Operability of the emergency system tie breakers shall be demonstrated semi-annually.

B. Auxiliary Diesel Generator

The auxiliary diesel generator shall be started and loaded to 30 kw at monthly intervals whenever its operability is required by Section 3.6. One hour availability shall be demonstrated.

C. Batteries

Periodic checks of battery performance shall be made on all three battery stations (+125 V DC, $\pm$ 26.5 V DC and +26. 5 V DC) and the diesel starting batteries as follows:

1. Measure and record daily the battery floating bus voltage, the pilot cell specific gravity reading and adjacent cell temperature. The designated pilot cell shall be changed each month.

2. Measure and record monthly the floating charge, amount of water added, and specific gravity of each cell. Measure and record the temperature of every sixth cell.

3. Inspect all electrical connections for tightness every six months. At the same time, subject each battery station to a heavy discharge condition. Monitor bus voltage and current as a function of time to establish that each battery station performs as expected, and check amperehour rating against draw-down. Except for the diesel starting batteries, calibrate panel voltmeters against a known standard.

4.7 Piping System Snubbers

Applicability

Applies to all pipe snubbers in the reactor building.

Objective

To assure proper operation of the snubbers.

Specification

A. All pipe snubbers shall be checked for oil level and leakage at six-month intervals.

B. Samples of oil shall be placed near snubbers operating in representative radiation fields. The viscosity of oil samples shall be measured every six months to assure that it is in the range specified by the snubber manufacturer.

C. Representative pipe snubbers in accessible regions will be exercised at a yearly interval.

D. Pipe snubbers shall be replaced or repaired if such action is required to assure proper operation of the snubbers.

4.8 Environment

Applicability

Applies to the environmental surveillance program in the vicinity of the site.

Objective

To assure that release of radioactive material from the plant is not significantly affecting the level of radiation of the off-site environment.

Specification

The environmental surveillance program specified in References 1 and 2 shall be implemented.

References

1. SEFOR FDSAR, Volume 1, 9.4.4, page 9-8 and Table IX-3.
2. SEFOR FDSAR, Supplement 17, Answer to Question M-6, page M-9.

4.9 Unexplained Reactor Behavior

Applicability

Applies to unanticipated changes in reactor process and nuclear variables during operation.

Objective

To assure that reactor characteristics are properly interpreted and sufficiently understood for safe operation and safe conduct of the experimental program.

Specification

A. Long-term Unexplained Trends

In addition to the more frequently taken records, records shall be kept of observations made at least once daily of sodium flow for given pump conditions and the concurrent reactor sodium inlet and outlet temperatures, reflector segment positions and the reactor operating power. These records shall be examined daily during periods of reactor operation for short-term changes and analyzed in detail at least monthly to determine if there are any unexpected trends of significance which might indicate a change in reactor or component performance. Any unexpected trends which are observed shall be reviewed by the Site Safety Committee. Reactor operation at a higher power level shall not take place unless identified long-term trends are satisfactorily explained, or the Site Safety Committee concludes that the long-term trends observed do not indicate a deterioration of performance which could affect plant safety during the next planned period of operation. The conclusions of the Site Safety Committee shall be documented and transmitted to the SEFOR Safety Review Committee and the General Manager, APO, immediately after their conclusions are reached. In the event that there is no satisfactory explanation of long-term trends, independent evaluation by members of the Safety Review Committee shall be obtained within one month after identification of the trend. The General Manager, APC, upon advice of his technical staff, the SEFOR Site Safety Committee, and the Safety Review Committee shall make a determination of the future mode of operation of the reactor. These determinations and the supporting documentation shall be transmitted within one week to the AEC.

B. Short-term Unexplained Reactor Behavior

Reactor operation shall be continuously monitored for short-term changes. For purposes of this specification, short-term changes shall include but not be limited to, unexplained changes in reactivity, primary coolant flow rate, or upper reactor vessel outlet temperature. Upon observation of a short-term change that is not readily explainable and which in the operator's judgement has possible safety significance, reactor power shall be reduced at least to a level of 50% of that a which the change was observed. The SEFOR site facility manager shall be notified immediately. Reactor power shall not be increased unless the cause of the change has been determined. If the cause is not immediately apparent, the SEFOR site manager shall determine whether operation at reduced power may continue or whether the reactor should be shut down. As soon as practical, he shall call a meeting of the Site Safety Committee which shall investigate the unexplained occurrence and recommend further action. If the cause of the occurrence is not identified or if it is determined that there is a potential safety problem, resumption of operation at the initial power level where the change was observed shall not be permitted until a report has been made to the General Manager of APO and an investigation has been conducted by members of the APO technical staff. The General Manager, APO, may authorize higher power operation of SEFOR upon evaluation of the reports of his technical staff and the Site Safety Committee. Upon such authorization, a report of his decision and supporting documentation shall be forwarded to the AEC. If operation is resumed, the conclusions of the Site Safety Committee and the APO technical staff shall be documented and circulated to the SEFOR Safety Review Committee and their independent evaluation shall be obtained within one week of the resumption of operation.

Section 5

DESIGN FEATURES

Applicability

Applies to those features of the plant that are not covered elsewhere in these specifications and are applicable to physical barriers.

Objective

To control changes and maintain safety margins in the design and location of equipment.

Specifications

The reactor facility shall be located within a restricted area. The exclusion distance shall be at least 0.4 mile.

Section 6

ADMINISTRATIVE CONTROLS

6.1 Organization, Review, and Audit

This specification applies to the organization for management and for review and audit of facility operations. Its objective is to delineate responsibility for management of the facility, to assure maintenance of a high level of staff competence, and to specify an independent program for review and audit of facility operation.

A. Organization

1. The organization for management, operation, and audit of SEFOR facility operations shall be as given in Figure 6-1. The overall full-time responsibility for operation of the SEFOR facility and compliance with these Technical Specifications shall reside in the SEFOR Facility Manager, who in turn is responsible directly to the APO General Manager. The APO General Manager reports to the Nuclear Energy Division General Manager. As indicated in Figure 6-1, technical engineering assistance will be provided as necessary through the APO Manager of Plant Engineering and Projects.
2. The minimum functional organization required for operation of the facility shall be as follows:
    a. An operating shift shall consist of a shift supervisor and at least three additional operators. Refueling operations may be carried out under the supervision of a senior operator with a minimum crew of two persons trained in fuel handling procedures.
    b. When the reactor is secured, a reactor operator or a senior supervisor and two additional persons trained in carrying out emergency procedures shall be at the site.

c. A licensed senior operator shall be in charge of startup, approach to power, normal operation, recovery from unplanned or unscheduled reductions in power, shutdown, and refueling operations.

d. Personnel requiring Part 55 licenses shall be as indicated in Figure 6-1.

3. Qualifications with regard to education and operating experience for key supervisory personnel shall be as follows:

a. <u>SEFOR Facility Manager</u>

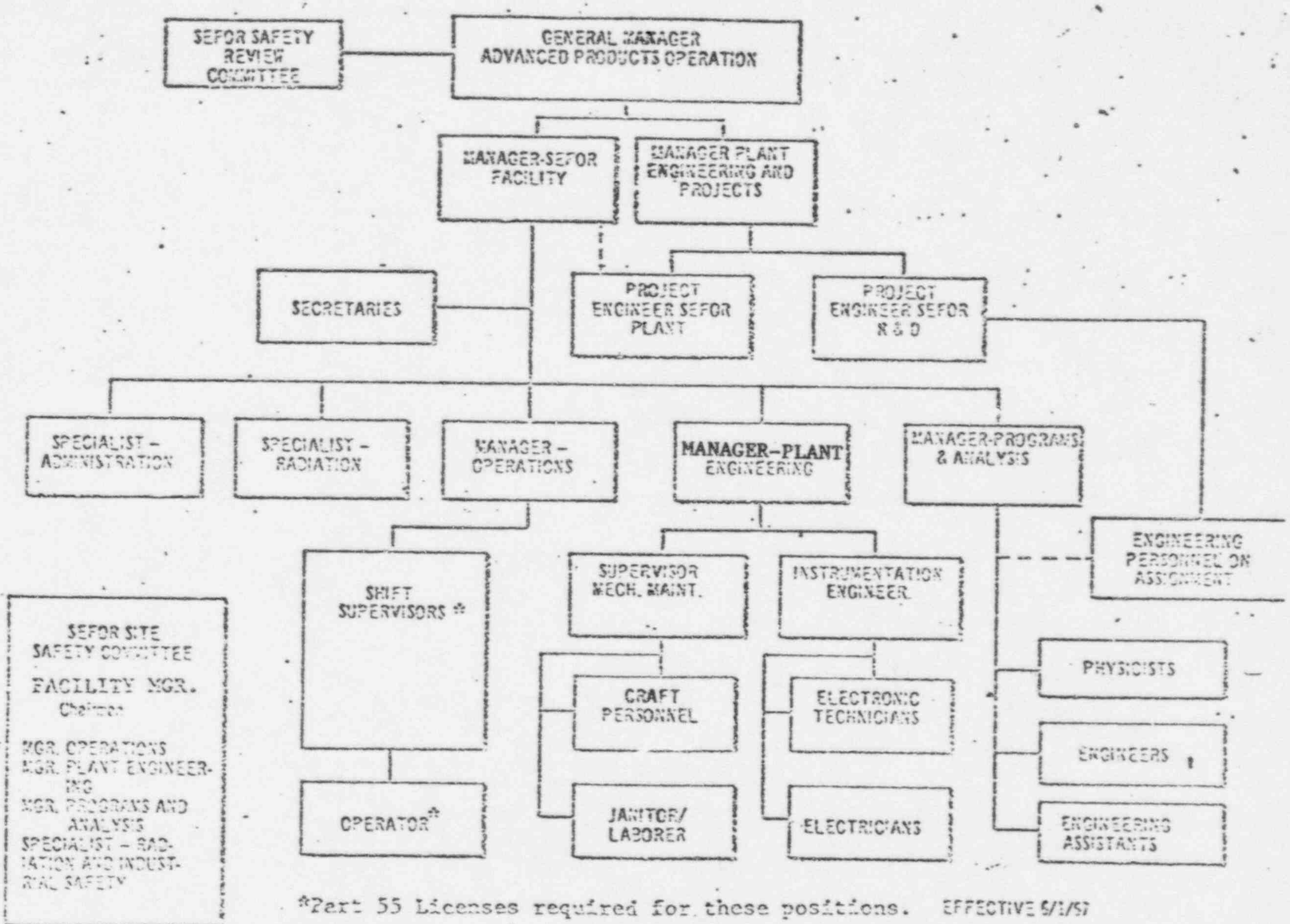
B.S. in Engineering or Science or equivalent in experience. Ten years experience in the design, construction, installation, operation, development, and maintenance of nuclear facilities.

Demonstrated detailed and comprehensive knowledge in related technical fields, including reactor physics, radiological hazards control, nuclear engineering and instrument engineering.

Five years experience in the supervision and management of the construction and operation of reactor facilities. Demonstrated ability to plan, organize, and direct reactor plant operations for several reactor types.

b. <u>Manager, Plant Engineering</u>

B.S. in Engineering or Science or equivalent in experience. Ten years experience or equivalent in the operation and maintenance of power-generation facilities, including a minimum of three years in responsible supervisory positions in the operation or maintenance of such facilities.



*Part 55 Licenses required for these positions. EFFECTIVE 6/1/67

6.1.3

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Ability to plan, program, and direct activities of engineering and craft personnel.

Demonstrated ability in the design and application of equipment and devices, and have a thorough understanding of process equipment such as pumps, fans, heat exchangers and generators, heaters, etc., as applicable to nuclear facilities.

c. Manager, Operations

B.S. in Engineering or Science or equivalent in experience. Five years experience in the operation and maintenance of reactor or nuclear power facilities, including minimum of one year in supervisory positions in the operation and maintenance of such facilities.

Demonstrated ability to organize and coordinate plant operations. Comprehensive knowledge of problems associated with startup and initial operation of reactor facilities, including knowledge of radiological hazards, technical aspects of reactor operation of control systems, radiation shielding, contamination control, etc.. Demonstrated good judgment necessary to make correct decisions under rapidly changing conditions.

d. Manager, Programs and Analysis

B.S. in Engineering or Science or equivalent in experience. Five years experience in the design, operation, analysis and programming of a variety of reactor types or nuclear power facilities, including at least one year in responsible supervisory position in such organizations.

Comprehensive knowledge of reactor physics, reactor design, reactor operation, radiation shielding, fluid flow, thermodynamics, instrumentation, and related technologies.

Demonstrated capability for directing the efforts of physicists and engineers.

Ability to develop techniques and test procedures to carry out a reactor experimental program.

Demonstrated knowledge of the practical aspects of the operation of reactors, including the characteristics, limitations, and safe operating requirements.

e. Specialist, Radiation and Industrial Safety

B.S. in Chemistry or Chemical Engineering or equivalent in experience.

Three years experience in analytical chemistry and health physics, and one year in radio-chemistry.

Deomonstrated ability in evaluation of radiation hazards, design and development of radiation monitoring equipment, and in conducting health-physics studies.

Thorough understanding of radiation dosimetry and a working knowledge of design of radiation facilities, shielding calculations and design of ventilation control, radioactive waste processing, calibration of radiation measuring instrumentation, maximum permissible radiation exposure levels, and good radiological safety and health protection practices.

Must be cognizant of local and state industrial safety requirements. Demonstrated ability in teaching, lecturing, and implementing safe practices and procedures.

f. Supervisor, Mechanical Maintenance

High school education and apprenticeship training in metal-working fields, coupled with at least ten years practical experience in mechanical shop or industrial processes, including maintenance and fabrication.

Five years of supervisory responsibility in a mechanical shop or industrial establishment.

Comprehensive knowledge of all phases of mechanical maintenance, including such mechanical crafts functions as machining, pipe fitting, welding, carpentry, and rigging on reactor process equipment, including those which have been exposed to radiation. Cognizance of radiation and safety procedures and regulations, as applicable to nuclear facilities.

g. Instrumentation Engineer

B.S. in Electrical Engineering or equivalent in experience.

Three years experience in design, installation, calibration and maintenance of process and nuclear instrumentation.

Cognizance of significance of control and instrumentation systems with respect to reactor operation and safety. Demonstrated ability to analyze systems for adequacy to meet systems requirements and to conceive, assemble, and install necessary modifications to meet systems requirements.

h. Shift Supervisor

B.S. in Engineering or Physics or equivalent in experience.

Three years experience in the operation of reactor or nuclear facilities.

Knowledge of reactor startup methods and procedures, including radiological hazards and their control, plant maintenance, modern physics and other technical aspects of reactor facilities.

Ability to plan, coordinate, and direct the efforts of operations personnel as indicated by previous supervisory experience or satisfactory progression in job positions.

Licensed as a Senior Reactor Operator.

B. Review and Audit

Organizational units for the review and audit of plant operations shall be constituted and have the responsibilities and authorities outlined below:

1. SEFOR Site Safety Committee

a. Membership

Chairman: Facility Manager or designated alternate.

Manager, Operations or designated alternate.

Manager, Plant Engineering or designated alternate.

Manager, Program and Analysis or designated alternate.

Specialist, Radiation and Industrial Safety or designated alternate.

b. Meeting Frequency

At least every two weeks and more often as deemed necessary by the Chairman.

c. Quorum

Chairman or his designated alternate, plus two other members.

d. Responsibilities

(1) Review all proposed normal, abnormal, emergency procedures, and procedures for maintenance which are significant to reactor safety and proposed changes to these procedures.

(2) Review all proposed tests for the planned experimental program, and plant tests which may have significance to reactor safety.

(3) Review proposed changes to Technical Specifications.

(4) Review proposed changes and modifications to plant systems or equipment which would require a change in, or would be covered by procedures in d(1) above.

(5) Review plant operation to detect potential safety hazards.

(6) Review reported violations of Technical Specifications.

(7) Perform special reviews and investigations and make recommendations thereon as requested by the SEFOR Facility Manager.

(8) Report to the APO General Manager and to the Chairman of the SEFOR Safety Review Committee on all reviews and investigation conducted under items d(6) and d(7) above.

(9) Make tentative determinations regarding whether proposals considered by the committee involve unreviewed safety questions in accordance with 10 CFR 50.59 and recommend the necessary actions or safety analyses to the Facility Manager.

e. Authority

(1) The SEFOR Site Safety Committee shall be advisory to the Facility Manager.

(2) The SEFOR Site Safety Committee shall review and recommend to the Facility Manager approval or disapproval of proposals under items d(1) through d(5) above.

a. In the event of disagreement between the recommendations of the SEFOR Site Safety Committee and actions contemplated by the Facility Manager on safety matters, the decision and action to be taken shall be the responsibility of the Facility Manager.

f. Records

Minutes shall be kept for all meetings of the SEFOR Site Safety Committee. Copies of the minutes shall be forwarded to the APO General Manager, the Chairman of the SEFOR Safety Review Committee, and to the Manager, Safeguards and Analysis.

g. Procedures

Committee rules and regulations shall be prepared and maintained describing the function of the committee, its meeting schedule, methods for review and approval of evaluations and recommendations, or designation of meetings and such other matters as may be appropriate.