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

GENERAL ELECTRIC COMPANY

AND

SOUTHWEST ATOMIC ENERGY ASSOCIATES

DOCKET NO. 50-231

AMENDMENT TO PROVISIONAL OPERATING LICENSE

License No. DR-15
Amendment No. 1

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, 23, 24, 25, 26, 27, 28, 29 and 30 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, October 9, 1968, November 1, 1968, December 10, 1968, January 14, 1969, January 24, 1969, February 20, 1969, March 27, 1969 and May 16, 1969, 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 operation 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 steady-state power levels up to a maximum of 20 megawatts thermal, and in the pulsed mode in accordance with this license, as amended, 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;

Operating License No. DR-15 is hereby amended by restating subparagraphs 2.A.1, 2.A.3 and 3.A in their entirety to read as follows:

2.A.1 To possess, use and operate the reactor as a utilization facility at steady-state power levels up to twenty (20) megawatts thermal and in the pulsed mode;

2.A.3 To receive, possess and use up to 30 curies of Cobalt-60, up to 10 millicuries each of Krypton-85, Iodine-131, Xenon-133 and Cesium-137; and up to 2500 kilograms of natural and/or depleted uranium in connection with operation of the facility pursuant to 10 CFR, Part 30, "Rules of General Applicability to Licensing of Byproduct Material" and Part 40, "Licensing of Source Material".

3.A <u>Maximum Power Level</u>

General Electric is authorized to operate the facility at steady-state power levels up to a maximum of twenty (20) megawatts thermal and in the pulsed mode.

This amendment is effective as of the date of issuance.

FOR THE ATOMIC ENERGY COMMISSION

Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Attachment:
Appendix A, Technical Specification
Date of Issuance: JAN 2 1970

50-231 - SEFOR

APPENDIX A to LICENSE DR-15 - Technical Specifications for the SEFOR

(20 MW(t)

issued with Amendment No. 1 dated 1-2-70

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 for operation up to and including approximately 20 MWt.

Section 1

DEFINITIONS

1.1 Operable

The status of system or component when it is capable of performing its intended function in its required manner.

1.2 Operating

The status of system or component 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

Means 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

Means 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

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

Any one of the following:

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

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

(3) An unplanned reactor scram.

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

1.9 Degree of Redundancy

Defined as "R" in the formula, R=N-M, where N is the total number of operable 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.

1.10 Protection Instrument Channel

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

1.11 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.

1.12 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.

1.13 Channel Check

A visual inspection of the output analog signal indication during channel operation to determine that the channel is operating.

1.14 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.

1.15 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.

1.16 Gage Pressure, psig

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

1.17 FRED

The Fast Reactivity Excursion Device used to perform excursion tests.

1.18 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.

1.19 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.

1.20 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, thereby permitting certain safety functions to be bypassed(5) as specified in Section 3.1.

1.21 Master Mode Switch Positions

Establish the three basic conditions of reactor operation:

"NORMAL" - Used for all reactor operation except oscillator tests and excursion tests.

"OSCILLATOR TEST" - Provides safety system bypasses required for oscillator tests.

"EXCURSION TEST" - Provides safety system bypasses required for excursion tests.

1.22 Operate Mode Switch Positions

Provide safety system bypasses only when the Master Mode Switch is in the "NORMAL" position. The four positions of the Operate Mode Switch are:

"SECURED" - Used when the reactor is shut down with all reflector segments lowered.

"ZERO AND LOW POWER" - Used for criticality checks, reactor startup, and low power operation, at power levels which can be accommodated by the auxiliary coolant system. The low flow trips for the main coolant system are bypassed.

"HIGH POWER" - Used for reactor operation at all power levels. No safety system bypasses are in effect when the Master Mode Switch is in the "NORMAL" position and the Operate Mode Switch is in the "HIGH POWER" position.

"REFUELING" - Permits raising of no more than two reflector segments to check criticality.

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

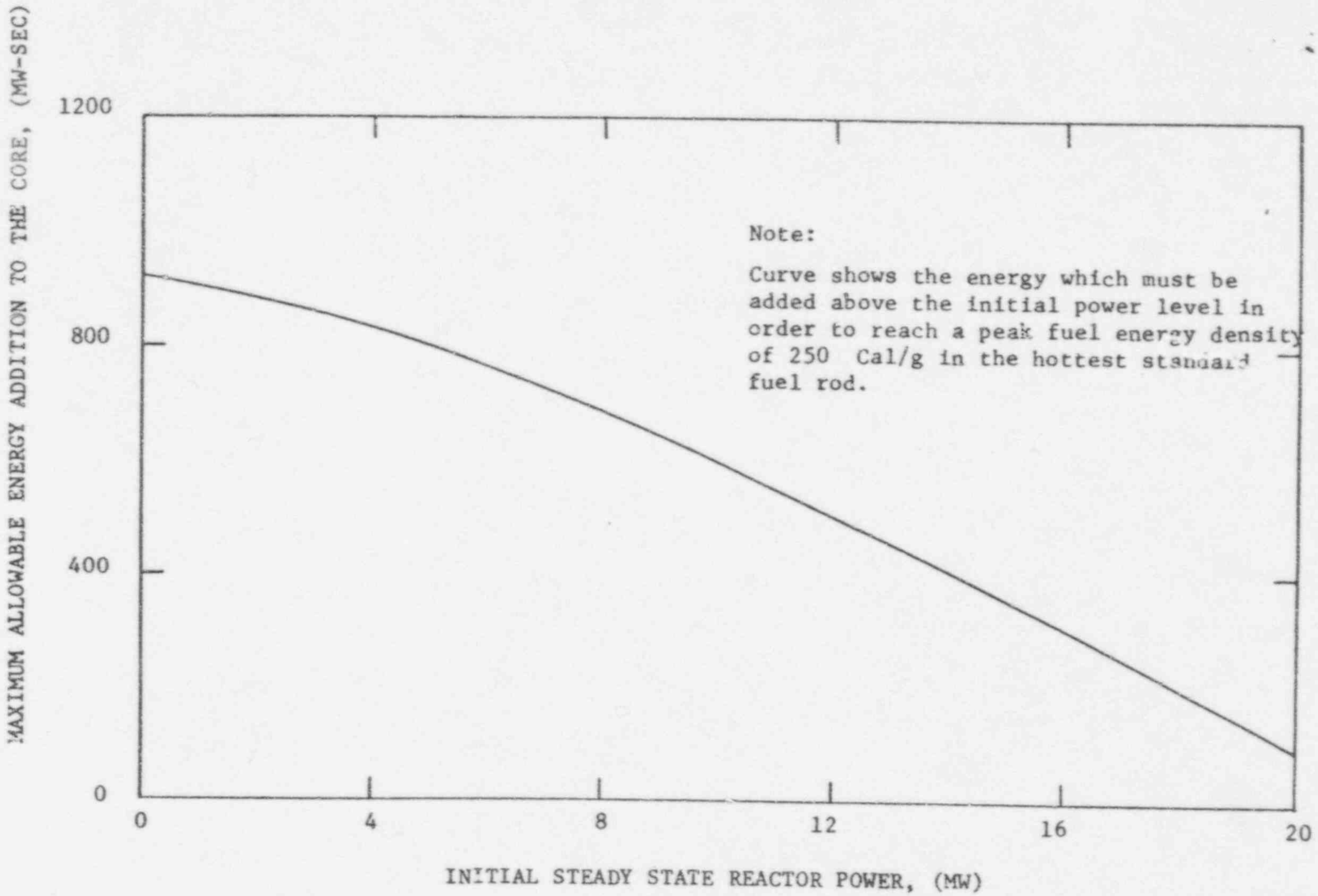
A. The maximum permissible reactor core flux shall be 110% of rated flux, except that the limit does not apply during an excursion test initiated by FRED.

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.



MAXIMUM ALLOWABLE ENERGY ADDITION FOR PLANNED TRANSIENT TESTS

Figure 2.1-1

Bases

The safety limit on reactor core flux of 110% of rated flux (defined in Section 1.6) corresponds to a reactor power level of 22 MWt. The steady state reactor power level is equal to the heat removed from the reactor vessel by the main primary and auxiliary primary coolant systems and is determined by measuring the coolant flow rate and coolant temperature rise from reactor vessel inlet to reactor vessel outlet for both of these coolant systems.

The reactor will normally operate at rated flux corresponding to a reactor power (defined in Section 1.5) of 20 MWt. The maximum linear power density in the standard hot fuel rod with the reactor operating at 20 MWt is 21.8 KW/ft using the minimum core loading of 600 fuel rods specified in Section 3.3, the core power peaking factors given in Reference 1, and assuming the full 20 MWt is being generated in the fuel meat. Operation at a reactor power of 22 MWt results in a maximum linear power density of 24 KW/ft in the standard hot fuel rod. The peak temperature in a fuel rod is calculated to reach the fuel solidus temperature(2) at a linear power density of 26 KW/ft. The accuracy of these calculations, determined by the present state of technology, does not completely preclude the possibility that a small amount of centerline fuel melting may occur at a linear power density of 26 KW/ft. Tests conducted under Task 3 of the SEFOR Preoperational Research and Development Program(3) and PA-10 of the AEC LMFBR Development Program(4) have demonstrated that oxide fuel can be operated for a large number of cycles at linear power densities grearer than 26 KW/ft without damage to the fuel. For example, results given in Reference 3 demonstrate that fuel can be cycled more than 100 times between power densities of 23-29 KW/ft (cycle frequency ~0.01 cps) with no damage. Fuel tests conducted under PA-10 have shown the capability of this type fuel to sustain burnups of tens of thousands of MWD/Ton at linear power densities in excess of 24 KW/ft.(4) In contrast, the SEFOR fuel will experience an estimated burnup of only 1500 MWD/Ton during the three-year experimental program. All of these tests were performed using sodium coolant at temperatures similar to those that will occur in SEFOR.

If the reactor outlet temperature were to increase from the nominal design value of 804°F to the safety limit of 1050°F at the same time the reactor power approached the safety limit of 22 MWt, the resulting increase in fuel temperature in the standard peak fuel rod would correspond to only a 0.7 KW/ft increase in the linear power density.

Additional margin at the safety limit of 22 MWt is provided by the fact that the 24 KW/ft maximum linear power density in the fuel at the safety limit of 22 MWt is based on the entire 22 MWt being generated in the fuel meat for a minimum core loading of 600 fuel rods. Physics calculations indicate that only 94% of the total energy generated inside the reactor vessel is generated directly in the fuel.[5] The remaining 6% is deposited in the coolant, structure and shielding inside the reactor vessel. This reduces the actual KW/ft generated in the fuel at 22 MWt from 24 KW/ft to 22.5 KW/ft. In addition, the nominal loading of SEFOR is predicted to be 630 fuel rods which will further reduce the peak KW/ft generated in the fuel at the safety limit of 22 MWt to 21.4 KW/ft. Based on these data, it is concluded that sufficient margin exists between operation at the 22 MWt safety limit and the point at which significant fuel rod damage would occur.

The rapid expansion of the fuel due to heating during planned prompt power transients exerts a dynamic load on the fuel rod, grid plate, and the core support shroud inside the reactor vessel. This dynamic load can be treated as an impulse in these structures since it is applied during a time interval that is small relative to the natural frequency response of the structures. The impulse is proportional to the peak power reached in the transient.[6] Analysis has shown that the 304 SS bolts (which attach the core support shroud to the reactor vessel) are the critical items[6,7]; i.e., they reach their working stress limits for this dynamic load before the other components do. The safety limit of a maximum core flux of 6.25×10^3 times rated flux (125,000 MWt) limits the impulse load so that the peak stress in the bolts does not exceed the yield strength. Significant margin exists above this safety limit because the stainless steel bolts are capable of carrying dynamic loads of several times the value corresponding to the yield strength without failing grossly.

The curve shown on Figure 2.1-1 defines the total energy which can be deposited in the core during a planned transient without exceeding a peak fuel rod centerline energy density of 250 cal/g. The 250 cal/g value is the calculated energy density from mixed oxide fuel just below its solidus temperature as obtained using the latest available data for mixed oxide fuel heat capacity.[8] Results from transient tests[3,4] on sodium-cooled oxide fuel have demonstrated that repeated transients up to fuel energy densities of 250 cal/g will not damage the fuel.

The safety limits of 1050°F vessel outlet temperature and 45.5 psig reactor vessel cover gas pressure are selected to limit the pressure and temperature for all components in the primary system to their design conditions. Margin between these safety limits and a damage limit is inherent in the codes used in their design (ASME Section III and the ASA Piping Code B 31.7).

The fuel will not approach a heat flux limit (burnout limit) because of the large amount of subcooling (approximately 600°F at the safety limit of 1050°F) and the good heat transfer characteristics of the sodium coolant. During the steady state operation at less than the limiting safety system settings specified in Section 2.2, the maximum combined stress in the fuel cladding in the standard hot rod is less than twice the at-temperature yield strength of 316 stainless steel[9] so that cyclic fatigue will not occur. During transient operation (either planned transients or accidental transients which exceed the normal reactor power and coolant temperatures) the maximum combined stress in the cladding of the standard hot rod may exceed the twice-yield design limit and the fatigue life of the cladding can become the limiting condition. The predicted fatigue life for the cladding is 750[10] cycles associated with the Maximum Planned Transients specified in Section 3.12.

This fatigue limit was obtained using the methods outlined in Section III of the ASME Code and the fatigue curves given in Code Case 1331.1. The cumulative fatigue damage for the planned transient test program is less than 0.02. Additional fatigue damage due to unplanned power or temperature transients approaching the safety limits of 22 MWt and 1050°F vessel outlet temperature will not contribute significantly to this cumulative total. For example, the predicted fatigue life for the cladding cycled from refueling conditions to an operating power of 22 MWt and a 1050°F vessel outlet temperature is 2000 cycles. Consequently, one cycle of this type (involving violation of two limiting safety system settings) would increase the cumulative usage factor less than 0.0005.

References

(1) SEFOR FDSAR, Volume I, Table IV-7, p 4-33.

(2) SEFOR FDSAR, Supplement 10, pp 1-110.

(3) SEFOR FDSAR, Supplement 3.

(4) GEAP 5198, 19th Quarterly Progress Report, pp 5-22 and 23.

(5) GEAP 5576, "Final Specification for the SEFOR Experimental Program," January 1968, p II-2.

(6) SEFOR FDSAR, Supplement 18, p 35.

(7) SEFOR FDSAR, Supplement 19, p 57.

(8) SEFOR FDSAR, Supplement 10, pp 1-126.

(9) SEFOR FDSAR, Supplement 10, pp 1-92.

(10) SEFOR FDSAR, Supplement 10, pp 1-109.

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
Low Level, Reactor Sodium	$\leq$	4 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.

*The operating flow set points shall be specified in written procedures.

Bases

The limiting safety system setting (LSSS) of 105% of rated flux provides a 5% margin below the safety limit of 110%. This will assure protection of the safety limit for normal reactor operation. The actual safety system setting will generally be less than 105% of rated flux since a large percentage of the plant operating time will be spent at power levels below 20 MWt where the trip setting would normally be reduced a corresponding amount. (Only a limited number of experiments will be conducted at rated flux.)[1]

The LSSS for reactor vessel sodium level provides assurance of reactor scram in the event that reactor cooling capability should be jeopardized because of a leak in the coolant system and consequent loss of sodium from the reactor vessel. Normal operation of the pump-around loop and overflow nozzle in the vessel will maintain the sodium at a constant level in the vessel. A loss of about 15 gallons of sodium from the reactor vessel will cause the level to fall below the level trip probe and scram the reactor. The level trip probes are 2 inches below the overflow nozzle, providing margin with respect to the LSSS of 4 inches.

The core outlet sodium high temperature trip at 900°F provides a 150°F margin to prevent the sodium temperature from reaching the safety limit. Analyses presented in the FDSAR[2] show that the coolant temperature will not approach the safety limit for accident conditions except for extreme assumptions involving failure to scram or failure of both main primary pump flywheels.

The low flow trip for the main primary coolant system provides assurance that the coolant temperature will not approach the safety limit due to loss of coolant flow. If the main primary coolant flow rate decreased to 80% of the set point value, the temperature rise across the vessel would increase less than 25% (to a vessel outlet temperature of 830°F) before the safety system would receive the scram signal and shut down the reactor. Thus, the low flow trip provides the earliest trip in the event of sudden reduction in coolant flow.

Adequate cooling of the reflector guide structure, segments, and neutron flux monitors, is required to assure operability of the reflectors and the neutron monitors. Thermocouples installed in the reflector guide structure are monitored by the safety system to provide this assurance. The temperatures at the

ten positions monitored are predicted to range in value from 200°F to 250°F with all reflector segments raised and a reactor power level of 20 MWt. The variations depend on whether the thermocouples are located in the inner or outer web of the guide structure. If one segment is lowered, the temperature levels in that region will increase by about 25°F. The actual trip level used for the safety system will be set a maximum of 135°F above the temperature readings observed at power for thermocouples on the inner diameter and radial web, and 80°F above the readings for thermocouples on the outer diameter of the guide structure. If the operating temperatures in the reflector region were to increase by these amounts (135°F or 80°F), a cooling flow reduction of less than 50% would be implied. Calculations have been made which show that the relative expansion of the guide structure for this condition would be less than 0.060 inch, and that the operation of the reflector segments would not be impaired.(3)

The limiting safety system settings of 350°F for thermocouples on the inner diameter and 275°F for thermocouples on the outer diameter may be more restrictive for operation at rated power than the values described above. The values are also safely below the temperature (400°F) at which the properties of the aluminum used in the guide structure begin to change significantly. These temperature limits also provide assurance that the temperature of the neutron monitors will remain below the manufacturer's certified operating temperature of 300°F.(4)

References

(1) GEAP 5576, "Final Specification for the SEFOR Experimental Program", January 1968.

(2) SEFOR FDSAR, Volume II, Section 16.3, pp 16-10, ff.

(3) SEFOR FDSAR, Supplement 11, Appendix A and B.

(4) SEFOR FDSAR, Supplement 17, p G-5.

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 Bux	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 full down, 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°F, and 5) primary loop pressure below 1 psi. or 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.

Table 3.1-1 (Continued)

Function	I Minimum Number of Operable Channels	II Minimum Degree of Redundancy	III Conditions Permitting Bypass
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°F, and 5) primary loop pressure below 1 psi.
Sodium Leak, Auxiliary Primary Loop	6* and 6	6* and 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

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

Table 3.1-1 (Continued)

Function	I Minimum Number of Operable Channels	II Minimum Degree of Redundancy	III Conditions Permitting Bypass
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
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

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

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 Accumulators	≤ ≥	225 psi (high) 150 psi (low)
Very High Radiation, Containment Ventilation Exhaust	≤	10 X Radiation Level at design power level
Nitrogen Radiation Monitor	≤	10 X Radiation Level at design power level

* The operating flow set points shall be specified in written procedures.

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 channel.
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

TABLE 3.1-7

INSTRUMENTATION THAT INITIATES AUXILIARY PRIMARY PUMP TRIP

Function	I Minimum Number of Operable Channels	II Minimum Degree of Redundancy	III Conditions Permitting Bypass
High Temp. Pump Duct Wall	1	0	May be bypassed subsequent to trip actions to permit pump start.

TABLE 3.1-8

TRIP SETTINGS FOR AUXILIARY PRIMARY PUMP TRIP

FUNCTION	TRIP SETTING
High Temp. Pump Duct Wall	$\leq$ 1000°F

TABLE 3.1-9

INSTRUMENTATION THAT INITIATES VENTING OF PRIMARY DRAIN TANK COVER GAS

Function	I Minimum Number of Operable Channels	II Minimum Degree of Redundancy	III Conditions Permitting Bypass
High Differential Pressure, Reactor Cover Gas - Primary Drain Tank Cover Gas	1	0	Reactor Shutdown or Secured

TABLE 3.1-10

TRIP SETTING FOR VENTING OF PRIMARY DRAIN TANK COVER GAS

FUNCTION	TRIP SETTING
High Differential Pressure, Reactor Cover Gas - Primary Drain Tank Cover Gas	$\leq$10 psid

Bases

Tables 3.1-1, 3.1-3, and 3.1-5 list the protection instrument channels which provide input to the reactor safety system and constitute the minimum functional condition for the safety system. Column I in each of these tables specifies the minimum number of operable instrument channels and protection logic sub-channels and channels required for reactor operation, and Column II specifies the minimum degree of redundancy required. These requirements assure that no single failure can prevent protection of the reactor by the safety system.[1]

The Master Mode Switch and Operate Mode Switch are used to bypass some of the trip functions as indicated by Column III in Tables 3.1-1 and 3.1-3. The contacts of the two mode switches are connected in series so that both are required to make up the necessary closed circuits to energize the bypass relays. These two mode switches are the only means of bypassing the trip functions listed in Table 3.1-1 and 3.1-3 with the exception that temporary bypasses will be installed on the reactor sodium low level trip and low level source range trip for the initial loading of fuel into the reactor. The bases for all of the bypasses are discussed below.

Loss of Power Bus 2A

This function is bypassed only in the "SECURED" mode when all reflector segments are down. The reactor is therefore shut down, and the safety trip functions are not required to be operational.

The bypass is provided to enable testing and adjustment of the on-site electrical generation facilities with the reactor shut down, while maintaining energization of the safety system to determine the effects of the transfer from one electrical system to another on other safety related equipment. The bypass is required because the under voltage safety trip devices on Bus 2A are fast acting and would trip on the few cycles interruption caused by the transfer.

Low Level, Reactor Sodium Level

The sodium level in the reactor is lowered during refueling operation so that the tops of the fuel rod extension pieces can be seen above the sodium level. This level is well below the low reactor sodium trip level for power operation so that this safety trip must be bypassed to energize the safety bus and close the scram valves. The scram valves must be closed so that two of the reflector

segments can be raised to verify the shutdown reactivity margin specified in Section 3.3.A. Several permissives are provided for the bypass condition to assure that certain conditions exist before the bypass can be made up. The permissive requiring that eight reflector segments be full down limits the maximum number of reflector segments that can be raised to two. This provides added assurance that a criticality accident will not occur with a low sodium level in the reactor. The other two permissives (sodium temperature less than 400°F and cover gas pressure less than 1 psig) are related to standard operating procedures, and provide assurance that the sodium level and low flow trips will not be bypassed by this mode switch position while the reactor is operating.

During initial fuel loading, it is necessary to bring the reactor critical with the head off and the sodium level lowered below the low reactor sodium level trip. During this phase of operation, the Operate Mode Switch is in the "ZERO AND LOW POWER" position, but there is an additional restriction that the wide range monitor range switches must have a stop at 500 KW to assure that the scram trip is set at a low power level. Under this condition of operation, adequate coolant flow is provided by the auxiliary coolant system to dissipate the 500 KW of heat. In addition, the high flux trip (500 KW or less) and sodium high temperature trip provide protection to limit the reactor power.

Low Flow Main Primary and Main Secondary Loop

During zero and low power operation and during refueling operation, it is necessary to lower the flow in both the primary and secondary loops to a point below which the trips can be maintained in the untripped condition (i.e., where the trip point level and the flow level converge within the band of noise, error, etc). Reduction in the flow rate to this level is necessary to limit temperature transients in the system after shutdown or during low power operation and to prevent gas entrainment in the main system with low reactor sodium level during refueling.

Adequate protection from the sodium high temperature trips (core outlet, lower region) remains in effect should the reactor power accidentally be raised to a level for which the shutdown cooling flow rate is not adequate.

Low Flux, Source Range Monitor

During initial fuel loading, it will be necessary to place the reactor Operate Mode Switch in the "ZERO AND LOW POWER" position in order to raise all reflectors and check criticality. During the first steps in fuel loading, the neutron flux level will be below the low level trip for the safety system. Therefore, this trip will have to be bypassed. The flux will be monitored during this time by in-core startup instrumentation as specified in Section 3.9.

Low Flux, Wide Range Monitor

Inherent noise in the wide range monitor system at low flux levels during reactor startup requires that the wide range monitors be operated in a range where the instrument is driven down-scale. This results in a down-scale trip that is automatically bypassed until the neutron flux reaches the level where the noise in the wide range monitor is negligible. This corresponds to the range where the power level is about 0.2 watt.

The intermediate range monitors provide flux level indication and short reactor period protection for this range of neutron flux level.

Table 3.1-2 provides the trip settings for those trip functions that initiate scram. Those items specified in Table 2.2 are not repeated here. The bases for these trip functions and their settings are given below:

Loss of Power Bus 2A

Provides protection against loss of the source of instrument battery charging power. The instruments could draw down the battery reserve energy required for emergency operation without immediate indication if this trip was not available. Normal voltage is 440, but the equipment will operate safely at less than 380 volts. The trip is set at 380 to avoid spurious scram due to normal fluctuations in the power supply system.

Loss of Power Main 2.4 KV Bus

This trip provides diversity to the loss of main coolant flow trip. Fluctuations of 600 volts may occur and the equipment is designed to operate with variations larger than this.

High Temperature Core Outlet Lower Region

Provides diversity for the upper region outlet temperature measurement and provides protection against events which might allow a portion of the primary

flow to bypass the upper region outlet thermocouples. (See Specification 2.2 for bases for temperature setting in outlet core region.)

Auxiliary Primary Loop Leak

Provides reactor scram in the event of a leak in the Auxiliary Primary Loop between the pump and reactor vessel, and assists in determination of the leak location, which is necessary to provide continuity of cooling in the event of large leaks in the auxiliary inlet line. Electrical contact type leak detectors are used, so that the presence of sodium causes an electrical short which is detected by the safety system.

Two of the seven leak detectors used for the auxiliary primary system are located (a) at the reactor vessel inlet nozzle and (b) about 10 inches upstream from this nozzle. If either one of these leak detectors fails to operate properly, that channel shall be placed in a tripped condition so that a signal from any one of the remaining six leak detectors or from the nitrogen radiation monitor will cause the required safety action. If both of these leak detectors fail, only one of these channels shall be placed in the tripped condition. Each of the other five leak detectors has a spare, which may be used if any of these five should fail to operate properly.

These leak detectors are required to prevent excessive loss of sodium in the event of a major pipe break in the auxiliary primary pump discharge line. After adequate natural circulation capability of the auxiliary primary loop has been demonstrated, the emergency procedures for sodium leaks will be revised to shut off the auxiliary primary pump in the event of any pipe break indicated by a low level in the reactor vessel. This revision will provide assurance of core cooling capability in the event of a pipe break, so that removal of failed leak detectors from the safety system can be justified if necessary.

Low Level Main Secondary Expansion Tank

Provides an indication of potential loss of secondary sodium inventory which could be a precursor to a number of adverse events, such as a sodium fire, introduction of argon gas to the secondary coolant loop, or excessive leakage within the IHX. The allowable 17 inch level drop provides the necessary margin for the 12 inch change due to thermal expansion in the loop between 350°F fill and design power operating conditions.

Low Level Auxiliary Secondary Expansion Tank

Same basis as that provided for Low Level Main Secondary Expansion Tank. The system is smaller, however, and thermal expansion from the 350°F fill condition to design power will cause a level change of only 3-1/2 inches.

High Temperature Main Secondary Cold Leg

Provides early response to loss of cooling in the forced air heat exchanger, and also provides diversity for the High Temperature Core Outlet Trip. The LSSS of 800°F is required to permit completion of planned tests at reactor inlet temperatures up to 760°F with reactor power less than 20 MWt. The secondary coolant system has been designed for temperatures above this LSSS.[2] The reactor outlet temperature would be 1054°F and the reactor inlet temperature would be 950°F at 20 MWt with design coolant flow rates if the secondary cold leg temperature were 800°F. However, these conditions could not be achieved because the available excess reactivity would limit reactor power to a value below 20 MWt at reactor inlet temperatures above 800°F.

Low Flow Main Secondary

Provides diversity for the High Temperature Core Outlet trip in the event that reactor cooling capability is reduced by loss of flow in the secondary loop.

High or Low Pressure Reflector Drive Accumulator

Assures proper operation of the hydraulic system accumulator. The set points have been established on the basis of normal fluctuations in accumulator pressure, which may fluctuate between 150 and 225 psi and still function properly with margin.[3]

Very High Radiation Containment Ventilation Exhaust

Provides indication of radioactivity within the air zone region. This could represent a loss of integrity of the inner containment barrier or inner containment atmosphere control. Background level during operation is expected to be less than 1 mR/hr. The tenfold increase would represent an approach to 10 mR/hr, which would allow about twelve minutes for personnel evacuation of the containment. The reactor is scrammed because it may be associated with the increase in activity.

Nitrogen Radiation Monitor

A scram will be initiated automatically in the event that the nitrogen radiation monitor reading is high, coincident with a leak indication in the auxiliary

primary loop. This monitor is designed to detect airborne Na-24 shortly after a leak occurs[4] and provides added assurance that small leaks will be detected.

Table 3.1-4 provides the trip setting for the trip functions in Table 3.1-3. The bases for these are as follows:

Low Flux Source Range Monitor

Provides protection against instrument failure as well as operator error. When the source range monitor is reading about 10% of full scale, assurance is provided that proper instrumentation is available for reactor startup.

Low Flux Wide Range Monitor

Same bases as that for Low Flux Source Range Monitor.

Bypass Protection

Redundant contacts on the Operate Mode Switch will block the raising of reflector segments if the Operate Mode Switch is in the "SECURED" position.

Table 3.1-6 provides the trip settings for the trip functions in Table 3.1-5. The bases for these are as follows:

High Radiation Containment Ventilation Exhaust

The trip occurs at two times the normal background which is an indication of a radioactivity change in the air zone. This change may be a precursor to more significant changes and, therefore, the containment isolation occurs to minimize potential radioactive release to the environment.

Very High Radiation Containment Ventilation Exhaust

Refer to previous discussion on reactor scram trip settings.

Tables 3.1-7 and 3.1-9 list additional protective devices for the reactor which provide assurance of adequate core cooling in the event of a major sodium system leak.[5,6] The trip settings[5,6] for these devices are given in Tables 3.1-8 and 3.1-10, respectively.

References:

(1) Proposed I.E.E.E. Criteria for Nuclear Power Plant Protection Systems, (I.E.E.E. No. 279, August 30, 1968; Section 4.2, 4.11 & 4.12)

(2) SEFOR FDSAR, Supplement 15, p. 163.

(3) SEFOR FDSAR, Supplement 11, Table 7.[illegible].13, p. 7-15.

(4) SEFOR FDSAR, Supplement 19, p. 52.

(5) SEFOR FDSAR, Supplement 23, p. 4.

(6) SEFOR FDSAR, Supplement 22, p. 4.

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 breached, the electrical power circuit to the reflector control hydraulic power supply shall be de-energized.

Bases

Reactivity is controlled by means of ten reflector segments located outside of the reactor vessel.[1] The reflector segments (also referred to as reflector rods and control rods in the FDSAR) are raised to increase reactivity. Two of the segments are designated as fine reflector segments because they can be accurately positioned throughout their range of travel. Either one or both of these segments will normally be set at an intermediate position in order to hold reactor power at a desired level. The other eight segments are designated coarse reflector segments and are normally set in the fully-raised or fully-lowered position for steady-state reactor operation, since some drift to lower reactivity may occur if they are left in a mid-range position. There will be at least 1$ shutdown margin even if one operable reflector segment remains in its most reactive position as specified in Section 3.3A.

The limit on the maximum rate of reactivity increase for a reflector segment is based on analysis of reflector segment insertion accidents.[2] For a reactivity insertion rate of 10¢/second, corresponding to 1.2 inches/second reflector segment raising rate, these analyses show that the increase in fuel temperature would be less than 25°F if the accident should occur at rated power, and about 50°F if the accident should occur at source-level power. In the latter case, the peak fuel temperature would be of the order of 350°F. The normal reflector segment raising rate is less than 0.75 inch/second and the control drives are interlocked so that only one drive can be actuated at any one time.

The requirement that the reflector segment reach 90% of its downward stroke in less than one second following initiation of scram is consistent with the scram behavior used for the SEFOR safety analysis (see Figure XVI-1[3] of the SEFOR FDSAR) which indicates that a scram at this rate would prevent damage to the fuel and the reactor systems even under abnormal accident conditions.

During the planned transient test programs (when the Master Mode Switch is in the "EXCURSION TEST" position) it will be necessary to delay scram by 0.4 second to obtain accurate test measurements.[4] This additional delay in the reactor scram was included in the transient analysis performed for the planned transient test program[5] and therefore was included in the results used to establish the safety limits (Section 2.1) and the limiting conditions for operation (Section 3.12) for the planned transient test program.

During periods when the outer containment is intentionally breached (for example, during movement of irradiated fuel from the refueling cell to the fuel transfer cask as covered by Specification 3.5.B.1) the control drive system will be de-energized to provide additional protection against the possibility of a criticality accident.

References

(1) SEFOR FDSAR, Volume I, Para. 4.5.2, p.4-45.

(2) SEFOR FDSAR, Volume II, Para. 16.2.2, p. 16.4.

(3) SEFOR FDSAR, Volume II, Figure XVI-1.

(4) SEFOR FDSAR, Supplement 17, p. 6.1.

(5) SEFOR FDSAR, Supplement 10, pp. 1-51 & 1-52.

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$ with one operable reflector segment raised to its most reactive position, 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. The core reactivity shall not be increased by adding fuel rods to compensate for an inoperable reflector segment.

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 ($T\frac{dk}{dT}$) 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 if the scram trip point is set at a power level greater than 1 MWt.

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 channel (or in a drywell in the center channel) 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 levels above 100 KWt.

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

   1. Cladding rupture, cladding perforation, or other observable defects which may cast reasonable doubt on the integrity of the rods.

   2. Local swelling of the cladding in excess of 10 mils or bowing of the rod sufficient to prevent reinsertion of the rod into the core.

   3. An increase of more than 1/2 inch in the column height of either fuel segment.

L. The gross gamma cover gas monitor shall be demonstrated to be capable of detecting a fission gas release equivalent to about 1% of the 20 MWt equilibrium inventory in a fuel rod before the reactor is operated above 10 MWt. If such sensitivity is not demonstrated, a more sensitive monitor shall be installed.

M. If the gross gamma monitor becomes inoperable, the reactor shall be shut down, except under the following circumstances:
If a reactor test is in progress, (other than FRED transient test program) and the monitor should fail, reactor operation may continue for 24 hours, if no unexpected changes in cover gas activity indicative of changing fuel condition have been observed just preceding the failure, and if cover gas samples are taken for spectral analysis at intervals of approximately four hours.

Bases

The required shutdown margin stated in the FDSAR[1] is that the reactor be shut down with one operable reflector segment raised to its most reactive position at the refueling temperature. The nominal refueling temperature is 350°F. Specification A assures that this requirement will be met at both the nominal refueling temperature and the minimum allowable sodium temperature of 300°F as specified in Section 3.4. Note that when this specification is combined with Specification 3.2.A, the reactor in effect must be shut down by 1$ with two drives in their most reactive position.

The 0.5$ excess reactivity at 20 MWt was chosen because it is sufficient to carry out the planned experimental program, while at the same time limiting the maximum power achievable under accident conditions.[2]

The conclusions of the SEFOR safety analysis described in the FDSAR[3] are dependent primarily upon the presence of a negative Doppler coefficient ($T\frac{dk}{dT}$) with a magnitude equal to or greater than 0.004. This value was used in assessing the consequences of the MHA and setting design criteria for the containment structures. To a much lesser extent, and for much less severe potential accidents, SEFOR safety is dependent upon the calculated sodium and structures temperature coefficients.[4]

As in the case with the startup of any new reactor, one is faced with the dilemma of assessing its safety before there are operating data to verify that its characteristics are consistent with the assumption of the safety analysis.

In order to assure that the important safety characteristics of SEFOR are consistent with the safety analysis, critical experiments on a mockup SEFOR core were performed at the ZPR-III critical facility. The results of these experiments[5] indicate that the SEFOR Doppler coefficient is approximately $T\frac{dk}{dT} = -0.0085$ and that the sodium and quick acting structure (fuel) temperature coefficients are negative and of larger magnitude than those used in the safety analysis.

More accurate determination of the important safety characteristics of SEFOR cannot be made until the reactor operates at between five and ten megawatts so that the effect of significant changes in fuel temperature can be determined. At these power levels it will be possible to measure the Doppler and sodium temperature coefficients and to differentiate between them.

Although accurate measurements cannot be made at low power, gross measurements can be obtained to indicate that SEFOR characteristics are consistent with its predicted performance. A measurement of a non-negative zero power temperature coefficient or a non-negative power coefficient would indicate a discrepancy between predicted and actual performance. Under these circumstances, reactor power should not be increased unless additional measurements or analyses of the measurements indicated that the original measurements were in error. If this is not the case, the reactor shall be shut down and the safety implications of the observed behavior analyzed.

A similar situation exists with respect to Specification 3.3.E and the measurement of the Doppler coefficient at power levels up to 10 MWt.

The Nyquist Criterion for stability requires that the plot does not go through the point, -1, on the real axis. If the plot were to approach the critical point at some power level, the amplitude of the power oscillation resulting from a small reactivity oscillation would become undesirably large. The required phase margin of 30 degrees provides assurance that power oscillations of undesirably large magnitude will not occur and is in accordance with normal practice where problems of stability are involved.

The fully loaded core is expected to contain between 612 and 648 fuel rods. The power density in a fuel rod at a given reactor power will be inversely proportional to the number of fuel rods in the core. The minimum number of fuel rods allowed by specification G provides assurance that the linear power density at rated power will not exceed the values used to determine the safety limit on reactor power. (See Section 2.1).

There is no specific limit on the number of fuel rods below 1 MWt. However, criticality requirements will limit the minimum number of fuel rods at power levels below 1 MWt to a number close to 600 rods.(6)

The guinea pig rods can be removed through the small refueling ports in the reactor head without removing the head. They are intended to provide lead information on fuel performance.(7) Operation of the guinea pig rods at positions closer to the core center than the refueling port positions would subject the guinea pig rods to a higher than intended power density.

At a core power level of 17.5 MWt, the linear power density of the guinea pig fuel rods under the three innermost refueling ports is the same as that

of standard fuel rods at the center of the core during 20 MWt operation. Removal of guinea pig rods from these positions allows operation of the reactor above 17.5 MWt without exceeding this value of the linear power density.

Fission chambers (He-3) will be placed in the center channel to provide criticality measurements during initial loading of fuel into the core. After initial core loading, a drywell may be placed in the center channel. Experimental foils or fuel samples will be placed in the center channel (or in a drywell placed in the center channel) for physics experiments[8] at power levels less than 100 KWt during initial reactor operation. The amount of fuel in any sample will be well below 0.5 Kg. The drywell (which does not contain sodium) has limited cooling capacity and for this reason is not suitable for inclusion of fuel during power operations. At 100 KWt, the maximum power generated in 0.5 Kg of fissionable material is less than 300 watts, which can be dissipated to the surrounding structure (which is cooled by sodium flowing in adjacent channels) without a significant temperature rise. Small experimental foils (10 mg or less) may also be placed in the drywell for physics experiments at power levels above 100 KWt.

At 20 MWt, the maximum power generated in 10 mg of fissionable material is only about 1 watt. The reactivity worth of 10 mg of fissionable material in the center of the channel is negligible (about 0.001¢). The inadvertent addition of up to 60¢ in reactivity in the drywell with the reactor just critical and with no reactor scram is similar to the handling accidents discussed in the FDSAR.[9] The maximum fuel temperature for this case would be 1260°F.

Fuel rod defects are defined so as to prevent use of fuel rods which are distorted or which might distort to an extent that adequate cooling of the rod and/or surrounding rods might be compromised.

Pin hole leaks, which may occur in some fuel rods, are excluded from the definition of fuel rod defects.[10]

A rupture of the clad, if localized, might not, of itself, be sufficient to cause significant cooling loss. However, since such behavior is not expected, it would indicate the potential for future more serious damage to the rod, and removal from the core is prudent.

Localized expansion of the fuel rod of the order of a few mils may be expected in the SEFOR program, as indicated by the preoperational SEFOR fuel development program.(11) Local expansion of the order of 10 mils would not affect fuel rod cooling significantly. (The effect would be less than that of the local coolant perturbations due to fuel rod spacers.) Such an expansion is also not of concern as an immediate cause of fuel failure since the ductility of the SEFOR fuel cladding at maximum irradiation will be in excess of 15% uniform elongation, and fuel tests have shown that local distortions of as much as 20 mils will not cause a SEFOR fuel failure.(11) Nevertheless, under the SEFOR operating program, distortions of 10 mils would not be expected, and such distortions would be taken as an indication that the rod was not performing as expected. It would be prudent to remove the rod in order to preclude the possibility that additional operation would produce further distortions of more serious consequence.

The spacers in the core are such that bowing of a fuel rod to the extent permitted by the spacers would permit adequate cooling of the rod. No attempt will be made, however, to force rods into the core.

Changes in fuel element column length have been observed in fuel rods under normal operating conditions with no melting of the fuel meat occurring. Such changes do not by themselves indicate the imminence of fuel element failure. The limit of 1/2-inch change in fuel column height is set because larger changes are not expected in SEFOR; since such changes would indicate abnormal behavior, it is prudent to remove the fuel.

The conservative approach to the design of the reactor as discussed in Reference 12 (i.e., low fuel power density, relatively small core coolant temperature rise) provides the primary assurance of safe operation of the fuel and the reactor. Instrumentation to detect gross flow or power changes is provided which automatically scrams the reactor in the event of a significant change in either of these variables. The continuous fission gas monitor is included to provide the operator with some additional information to identify and follow trends (anomalies) that may develop during operation of the reactor. Operation of the reactor for up to one day with the continuous fission gas monitor inoperable is based on the need to:

(1) observe the continuous monitoring system under operating conditions to diagnose the cause of failure or mal-operation of the system;

(2) permit an orderly completion of a test series, so that tests completed prior to the failure do not have to be repeated;

(3) plan for an orderly shutdown of the reactor.

During periods of reactor operation when the continuous fission gas monitor is inoperable, batch samples will be taken at intervals of approximately four hours. This sampling frequency will assure that any trends that might develop will be identified.

References

(1) SEFOR FDSAR, Volume I, Para. 4.5.3.1, pp 4-50 and 4-51.

(2) SEFOR FDSAR, Volume II, Para. 12.3.6, pp 12-15 and 12-16.

(3) SEFOR FDSAR, Para. 16.4.2.5 and 16.4.2.6.1.1, pp 16-26 and 16-28.

(4) SEFOR FDSAR, Para. 16.2.1, p 16-4.

(5) SEFOR FDSAR, Appendix B, Para. B.5, p B-3.

(6) SEFOR FDSAR, Supplement 10, p 3-10.

(7) SEFOR FDSAR, Volume I. Para. 4.2.2.2.2, p 4-2.

(8) SEFOR FDSAR, Volume II, Para. 12.2.1, p 12-3.

(9) SEFOR FDSAR, Volume II, Para. 16.2.4.3.

(10) SEFOR FDSAR, Supplement 21, pp 2, 3.

(11) SEFOR FDSAR, Supplement 3.

(12) SEFOR FDSAR, Supplement 21, pp 1-17.

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 system shall be operable and the sodium temperature shall be 300°F or greater.

B. The reactor vessel pump-around loop shall be operable.

C. The argon cover gas systems for the primary and secondary sodium systems shall be operable. The systems for repriming the primary main and auxiliary coolant loops shall be operable.

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

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

F. The argon cover gas pressure in the primary drain tank shall not exceed the reactor cover gas pressure by more than 10 psi.

G. The systems (including the waste gas batch and decay tanks) for depressurizing the primary drain tank shall be operated within such limits that it is possible to reduce the primary drain tank pressure by 10 psi within 10 minutes.

H. A minimum reserve of 1300 gallons of sodium shall be maintained in the primary drain tank.

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

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

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

Sodium Coolant System

Bases

Each of the four sodium coolant loops, including pumps, heat exchangers, and associated controls and coolant equipment, must be operable during reactor operation to assure adequate core cooling capability for normal and emergency conditions. The 300°F minimum temperature in the sodium loops assures that the sodium temperature will be maintained above the plugging temperature to avoid potential oxide plugging problems. The 300°F value provides a reasonable margin above 275°F, which is expected to be the lowest plugging temperature that can be clearly determined. Plugging temperatures below 275°F are difficult to determine, because the characteristic drop in flow with decreasing temperature is not clearly distinguishable at lower temperatures. In addition, the 300°F minimum temperature in the primary loops assures adequate shutdown margin for the core as specified in 3.3.A.

The pump-around loop circulates sodium continuously between the reactor vessel and the primary drain tank. This loop must be operable during reactor operation to maintain the reactor sodium level within prescribed limits and to provide assurance that the loop is available for accident situations.(1)

The argon cover gas system is required to be operable to maintain the conditions described in Specification 3.4.D. The vent vacuum pump is required to reprime the auxiliary primary coolant system in the event sodium is lost from that system during some abnormal (accident) condition.

The cover gas pressure in the secondary system is set equal to the cover gas pressure in the reactor (which is at a lower elevation) so that the secondary sodium pressure in the IHX will be greater than the primary sodium pressure in the IHX. This will assure that leakage of radioactive sodium from the primary coolant system to the secondary coolant system will not occur.(2) Under normal operating conditions, the secondary sodium pressure will exceed the primary sodium pressure by about 30 psi in the main IHX and about 43 psi in the auxiliary IHX. Small leaks may occur in the IHX, but the differential pressure will prevent the radioactive primary sodium from entering the secondary system. If the primary and secondary loops were drained, the gas leakage through the allowable IHX leak, which corresponds to a hole size of about 22 mils for the main IHX, would be about 95 ft^3 in a 24 hour period at a pressure differential of 10 psi.(3) This is about .15% of the primary

containment volume, and would not contribute significantly to the inner containment allowable leakage rate of 20% of the contained volume over a 24 hour period at 10 psig. A leakage rate of 3 gal/hr in the main IHX would cause a change in sodium level of about 0.5 in/hr in the main secondary expansion tank. A leakage of 3 gal/hr in the auxiliary IHX will cause a change in sodium level of about 1.5 in./hr in the auxiliary secondary expansion tank. This leak rate at the higher pressure differential for the auxiliary IHX would indicate a hole smaller than 22 mils, and is therefore more conservative than the specification for the main IHX. The radiological effects resulting from a failure of the secondary coolant system boundary have been calculated for this leak size and are well within the 10 CFR 100 Guidelines.(4)

The sodium coolant system is designed to operate at reactor cover gas pressure down to 0 psig. However, in order to assure maximum protection for the main coolant pumps, i.e., prevent possible pump duct damage due to vibration, it is planned to operate the systems so that the pressure at the pump inlet is positive with respect to the 2 psig gas coolant surrounding the pump duct. At maximum sodium flows, the reactor cover gas pressure required to achieve this condition is 20 psig.(5) The upper limit of 25 psig corresponds to the pressure relief value for the rupture disk in the reactor cover gas system. This value is sufficiently high to permit operation at 20 psig with the normal control variations (20-22 psig) which occur during supply and venting operations.

The differential between the cover gas pressures in the primary drain tank and the reactor vessel is limited to 10 psi by a differential pressure alarm unit which automatically vents the drain tank whenever this pressure differential is exceeded. A check valve in the overflow line normally prevents reverse flow of sodium from the drain tank to the reactor vessel through this line. The 10 psi limit and the capability of reducing drain tank pressure as specified provide added assurance that an uncontrolled loss of sodium from the primary drain tank will not occur due to rapid depressurization of the reactor vessel in the event of a major pipe break.(6)

The 1300 gallons of reserve sodium in the primary drain tank will be adequate to refill the auxiliary loop in the event of a major pipe break.(6)

Setting the plugging temperatures at least 25°F lower than the minimum loop operating temperatures provides a means of assuring that oxide precipitation and potential plugging will not occur in the coolant system. The upper plugging temperature value of 425°F represents a limit which will not cause significant corrosion or mass transfer for the SEFOR operating temperatures. This conclusion is based upon results of tests run at GE under the AEC Sodium Mass Transfer Program.[7] Corrosion measurements were made with sodium containing 50 ppm oxygen (corresponding to a plugging temperature of 450°F) which indicate that corrosion rates are small for the SEFOR conditions.

References

(1) SEFOR FDSAR, Supplement 19, p. 52.

(2) SEFOR FDSAR, Volume I, Para. 5.2.2.2.1, p. 5-5.

(3) SEFOR FDSAR, Supplement 17, Answer J-2, p. J-1.

(4) SEFOR FDSAR, Supplement 21, p. 31 & 32.

(5) SEFOR FDSAR, Volume I, Table V-I, p. 5-2.

(6) SEFOR FDSAR, Supplement 23, pp. 2-5.

(7) GEAP-4831, p. 11, Sodium Mass Transfer XV, March, 1965, "Behavior of Selected Steel Exposed in Flowing Sodium Test Loops," by M. C. Rowland, D. E. Plumlee, R. S. Young.

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. This specification is applicable only when the subject vessel contains one or more fuel rods.

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, including through-head ports, is in place.
   2. Access to the refueling cell may be permitted provided that the following conditions are met:
      a. The reactor shall be secured.
      b. The purity of the inner containment atmosphere shall be maintained according to standard operating procedures.
      c. All handling operations in the refueling cell, except for material being moved through the marine hatch, shall be suspended while the marine hatch is open.
      d. The electrical power circuit to the reflector control hydraulic power supply shall be de-energized.
      e. Outer containment integrity shall be maintained.
   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, and the integrated leak test shall be repeated. This sequences shall be repeated until the measured leakage is less than L_{t_o}.

G. If the measured 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, and the integrated leak test shall be repeated. This sequence 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 repair 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 13 pipe tunnel penetrations which penetrate both containment barriers shall not exceed $\frac{13}{8700}$ 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 500 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 50°F.

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

T. The inner containment barrier shall be capable of withstanding 10 psig internal pressure.

U. The outer containment barrier shall be capable of withstanding 30 psig internal pressure.

Bases

Containment integrity will be maintained whenever the reactor is in operation. Containment isolation can be initiated by appropriate pressure, temperature, or radiation signals, or by manual trip.[1]

Containment integrity will also be maintained during fuel handling operations when the reactor vessel head or the spent fuel tank cover is removed. The potential for cell pressurization due to accidents during this operation will be very low, since the reactor will be shut down and the sodium temperature must be less than 450°F, as specified in Section 3.9. If the head is in place, but not bolted down, the containment may be breached to permit necessary handling or repair operations.

Use of the vertical fuel transfer port to transfer fuel from the refueling cell to the fuel transfer cask is described in Reference 2. The design of the cask and of the connection between the cask and the vertical fuel transfer port maintains the inner containment integrity and pressure containing capability throughout the transfer operation. The additional requirements that the reactor head be in place and that the control drive system be de-energized (Specification 3.2.D) make it very improbable that an accident resulting in pressurization of the inner containment could occur. Therefore, any accidental release of radioactivity (for example, by dropping and failing a fuel rod) could be controlled by operating the refueling cell as an alpha cell so as to prevent the release of the radioactive material directly to the environs.

Initial loading of fuel into the reactor will require transfer of a large number of fuel rods (several hundred) through the marine hatch in groups of six at a time. Bagging techniques will be used to maintain purity of the refueling cell atmosphere. The marine hatch will be closed before reflector segments are raised to check for criticality since this operation reduces the core shutdown margin. These operations will be conducted according to standard written operating procedures.

Man access to the refueling cell with the reactor vessel head removed will be required to perform necessary modifications and surveillance functions. Written procedures will be followed for all of these procedures, which may include the following operations:

(1) Final assembly operations for the Instrumented Fuel Assembly.

(2) Removal of reactor vessel surveillance samples.

(3) Checking of core clamps.

(4) Replacement of reactor head seals.

During these operations, the oxygen content of the inner containment will remain well below the allowable limits and will be maintained as low as possible by means of the standard operating procedures. Bagging techniques will be used for the marine hatch when necessary. Clean room inventory procedures will be used to account for all tools and other items taken into and out of the cell, and lanyards or other keepers will be attached to all tools to prevent their falling into the reactor. A temporary cover will be installed over the core whenever possible. The sodium level will be lowered to the refueling level (below the tops of the extension rods), thus significantly reducing the sodium surface area exposed to the cell atmosphere. The sodium temperature will be reduced to as low a value as is practical, and will be kept within the allowable limits of 300°F to 450°F.

Temporary breaching of the inner containment barrier to repair equipment necessary to cover an open vessel of sodium would be less hazardous than leaving the vessel open for an extended period of time. Written procedures will be used to carry out such an operation with special attention given to maintenance of the argon atmosphere in the refueling cell.

Use of the horizontal transfer port is not required, because handling operations that could be performed using the port are not planned. This port was included in the design for potential future use if a need for it should arise. No such need has been identified as yet.

The maximum allowable leakage rates are established on the basis of potential radiological effects as calculated for the CBRE.(3) These calculations show that the radiological effects calculated on the basis of the allowable leakage rates fall well within the guidelines given in 10 CFR 100.(4) The allowable leakage rates for each containment were calculated by the method recommended in Reference (5). Values for the outer containment and inner containment are identified by the subscripts, o and i, respectively, when appropriate. The allowable leakage rates(6) are based on the conditions used in calculating the radiological effects as shown in the FDSAR. The allowable leakage rate

for the inner containment is L_a = 20% of the contained volume at a pressure of 10 psig and 329°F. The pressure corresponds to that used for the radiological effects calculation, and the temperature corresponds to case 1e of Table XVI-13 in the FDSAR.[7] The leakage rate was not adjusted for temperature changes in the accident calculations, so the use of 329°F in the calculation of applicable limits is conservative.

If leakage of the inner containment exceeds the value of L_{t_i} or L_i, the penetrations must be repaired and the surveillance frequency must be increased, as indicated in specifications 3.5.H and 3.5.I, respectively. It would be desirable to avoid reaching these limits, both from a safety viewpoint and an operational viewpoint. Therefore, if the annual containment leak test shows that unidentified leakage from the inner containment exceeds 10% per day at 10 psig, a special effort will be made to identify and repair the source of the increased leakage.

For the outer containment, the allowable leakage rate is L_a = 2.5% of the contained volume at a pressure of 30 psig and a temperature of 140°F. The initial pressure used in the accident calculations was 0 psig, and the temperature was 100°F. The peak calculated temperature for this analysis was 140°F, and this value is used herein.

The allowable leakage rates for the outer containment under test conditions were then determined from the following expressions, as recommended by Reference (5) and the DRL staff:

$$L_o = L_a(R_p T_p / R_a T_a)^{1/2} \times (P_t / P_p)^{1/2}$$

$$L_{t_o} = 0.9 \times L_o$$

where L_o corresponds to the symbol L_t in Reference (5).

Values of L_i and L_{t_i} were determined for the inner containment by using the same expressions with appropriate values.

The contained volumes used in the radiological effects calculations[6] were smaller than the actual volume of the inner and outer containment. This produces conservative results, since the calculated pressure is higher and the calculated leakage of contamination is greater than if the actual volumes

were used. The volumes specified in this section are based on more recent calculations, and should be used for the allowable limits of leakage through individual penetrations. Measurement of over-all leakage does not require a knowledge of the contained volume, since it is based on the change in pressure, which in turn is proportional to the percent of contained volume, as required.

The electrical and process piping penetrations through the outer containment shell are capable of being pressurized for individual leak tests. There are 54 electrical penetrations and 20 process piping penetrations between the air zone and the atmosphere, plus 13 pipe penetrations between the nitrogen zone and the atmosphere (through both containment shells).[8] These add up to a total of 87 penetrations.

Each penetration is designed to be leaktight, but some deterioration may occur over a period of time. Thus, it is necessary to establish leakage rate criteria, against which each penetration can be tested in order to assure detection of possible failures. The leakage rate through a penetration will depend primarily on whether or not a failure has occurred, rather than on the size of the penetration. The allowable leakage rate through a single penetration is set at 1/87 of L_{t_o}. Thus, the over-all leakage rate would not approach the limiting value except in the unlikely event that a large number of individual penetrations had failed. To provide further protection against reaching the allowable leakage rate for the outer containment, the maximum allowable total leakage through piping and electrical penetrations is set at 0.4 times L_{t_o}, and the allowable leakage through the doors and ventilation valves in the outer containment is set at 0.5 times L_{t_o}. Thus, the total leakage through known penetrations of the containment is limited to 0.9 times L_{t_o}, the allowable value for the outer containment.

The 13 pipe tunnel penetrations which connect the nitrogen atmosphere to the secondary sodium room (outside the outer containment) require an additional limitation. Their leakage rate limits will be further reduced by a factor of 100 to account for the absence of a dilution effect in leaking from the inner containment directly to the environs. Calculations for the design basis accident showed that the radioactive inventory leaked to the environs was reduced by a factor of 100 due to dilution in the air zone. Thus, the radioactive inventory leaked to the environs through these penetrations would not

exceed the amount leaked during the design basis accident if a proportionate amount of the leakage were assumed to pass through these penetrations.

The leakage rate through each penetration will be determined by measurement of the rate of pressure decay after pressurization of the penetration to 30 psig. For a given leakage rate, the rate of pressure decay depends on the volume being tested. Since the volume contained in each penetration is small, the allowable pressure decay rates may be quite high. Therefore, a one hour test period should be sufficient for each penetration (rather than the 24 hour period required for the integrated leak test).

Test frequencies are specified to be consistent with current practices and proposed guides.[5] Variations in testing periods (see general statement at the beginning of Section 4) are allowed to permit scheduling the tests at convenient intervals in the experimental program. For example, in some instances, if the scheduled test comes up during a planned sequence of experiments, it may be convenient to defer the test until the series is completed.

Repairs or modifications to containment barriers, its appendages, or penetrations must leave the barriers at least equal to the original barrier design requirements. Tests to verify this must be performed before containment integrity is assured.

The oxygen and freon content of the inner containment atmosphere has an influence on the potential consequences of chemical reactions resulting from accidental sodium spills. The oxygen content of the inner containment atmosphere is limited to less than 5 vol %. An alarm unit provides warning if this limit is approached.[9] The freon content of the inner atmosphere is expected to be negligible. Analysis of accidents involving chemical reactions between sodium and oxygen, using normal air in the inner containment, and between sodium and 1500 lbs of freon in the inner containment result in temperatures and pressures which are within the design capabilities of the inner containment.[10] The limit of 5 vol % further reduces the consequences of such an accident and provides margin by reducing the rate of combustion. The limit of 500 ppm of freon corresponds to a total freon content of less than 2-1/2 lbs within the inner containment. This limit is far below the amount used for accident analyses, and is based on the capability of the halogen leak detector, which can measure freon concentrations from 10 ppm to 1000 ppm. Initially, the freon content of the inner containment atmosphere will be less

than 10 ppm. If a leak should occur, it will be detected readily by the halogen leak detector, and the reactor will be shut down while the leak is repaired. Following such an occurrence, however, extensive purging will probably be required to reduce the freon content to a value below the specified limit of 500 ppm. Further reduction to less than 10 ppm would be impractical. Nevertheless, with an initial freon concentration at any value less than 500 ppm, a new freon leak would be detected if it were to occur.

The water content of the inner containment argon atmosphere will be kept as low as possible for practical reasons. At the limit of 125 ppm, there would be no safety hazard, even with the reactor vessel head removed. However, this would result in some reaction between the water and sodium and the plugging temperature of the sodium would be increased. Thus it would not be desirable from the standpoint of operational convenience to expose the sodium surface in the reactor vessel or the irradiated fuel storage tank for an extended period of time unless the water content of the argon atmosphere is well below the limit of 125 ppm. It will be kept as low as possible even while the heads are in place so that a minimum water content can be assured whenever the sodium surface is exposed. The cell argon purifier is designed to maintain the water content below 10 ppm.

The dewpoint of the inner containment nitrogen atmosphere will normally be kept at about 40°F by means of a nitrogen cooler and moisture removal unit. The moisture content of the nitrogen atmosphere at this dewpoint is 36 grains of water per pound of dry nitrogen (based on humidity charts for normal air). This corresponds to a relative humidity of about 17% at 90°F (blower discharge temperature) or about 7% at 120°F (normal cell temperature). At the specified dewpoint limit of 50°, the corresponding values are 54 gr/lb, 26% R.H. at 90°F, and 11% R.H. at 120°F. The nitrogen cooling coils are designed to operate with a minimum evaporating temperature of 47°F which results in a minimum nitrogen temperature of 63°F. Therefore, condensation of water vapor will not be a problem since the specified maximum dewpoint of the nitrogen atmosphere (50°F) is 13° below the minimum nitrogen temperature in the system.

The temperature of the atmosphere within each containment will normally be held at a constant value. However, if temperature changes greater than 50°F are made, it is desirable to limit the rate at which the changes are made to minimize thermal stresses in the containment liners and concrete walls.

Stresses caused by a step change of 100°F in the steel liner result in stresses below yield in the steel liner.[11] The allowable change of 50°F in temperature of the cell atmosphere would result in steel liner stresses of less than half of the referenced values. For larger changes in temperature, the rate of 5°F/hour would result in still lower stresses. For a step change of 100°F in the steel liner, the surface temperature of the concrete would rise about 65°F in 24 hours.[12] A step change of 100°F in concrete temperature will not overload the concrete or reinforcing rods.[12] Thus, the specified limits assure adequate protection of the steel liner and the concrete.

The design pressures for the inner and outer containments are 10 psig and 30 psig, respectively.[13] The outer containment was initially tested at 30 psig, but subsequent annual tests shall be performed at pressures not to exceed 10 psig.[13]

References

(1) SEFOR FDSAR, Volume I, Table X-5, p. 10-20.

(2) SEFOR FDSAR, Supplement 17, Answer to Question K-2, pp. K-3ff.

(3) SEFOR FDSAR, Volume II, Para. 16.4.3, pp. 16-32ff.

(4) SEFOR FDSAR, Volume II, Table XVI-10, p. 16-43.

(5) "Proposed Technical Safety Guide III, Reactor Containment Leakage Testing & Surveillance Requirements," USAEC, December 15, 1966.

(6) SEFOR FDSAR, Volume II, Para. 16.4.3.3.2, p. 16-39.

(7) SEFOR FDSAR, Volume II, Table XVI-13, p. 16-50.

(8) SEFOR FDSAR, Supplement 17, Answer D-15.

(9) SEFOR FDSAR, Volume II, Figure XI-20.

(10) SEFOR FDSAR, Volume II, Table XVI-11, p. 16-49.

(11) SEFOR FDSAR, Supplement 17, Answer B-15.

(12) SEFOR FDSAR, Supplement 17, Answer B-1.

(13) SEFOR FDSAR, Volume II, Para. 7.2.3.2 and 7.3.3.4, pp. 7-7 and 7-18, respectively.

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 a battery charger 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 one 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. One hour availability shall be demonstrated.

Electrical Systems

Bases

The SEFOR facility is provided with a single source of externally generated power supplied by an overhead 69 kV transmission line.[1,2] High power operation of the reactor requires the 2.4 kV and 480 V electrical systems in order to prevent damage to equipment.

The reactor will be scrammed upon loss of power to any of the 2.4 kV or 480 V essential buses. The main emergency diesel generator will start automatically and is capable of supplying power to essential components required for cooling and process control.[3,4] The underground diesel fuel storage tank has a capacity of 1500 gallons of fuel.[5] The minimum requirement of 910 gallons will provide sufficient fuel for 24 hour operation.[6]

Three separate dc power systems are provided for various essential system loads throughout the Reactor and Operations Buildings.[7] These dc power systems provide a non-interruptible source of power for essential loads and for loads where power interruptions are undesirable. Power for the systems is normally supplied through battery chargers, so that the batteries float on the line and thus maintain a full charge. Each battery system is capable of supplying power for a minimum of 8 hours without recharging.[8] The one minute starting time requirement for the main diesel generator provides assurance of adequate reactor cooling capability. The flywheel system will provide power to the main primary coolant pump for one minute or longer. If power is not available to operate the auxiliary system pumps after one minute, core cooling will be provided by natural circulation. Calculations show that it would require one hour or more (depending on the decay heat generation) to reach a core outlet temperature of 1050°F under natural circulation conditions, even if the forced air heat exchangers in the main and auxiliary systems were not functioning (loss of all external heat dumps).[9] Thus, a delay of up to one hour in supplying emergency electrical power would not result in plant temperatures above that for which the design permits continuous operation for an indefinite period. However, it is desirable to have electrical power at all times, and the main diesel generator has been designed to provide power within less than one minute after a plant power failure.

The auxiliary diesel generator is provided to assure core cooling capability in case some unforeseen occurrence should prevent use of the main diesel generator. The auxiliary unit is not started automatically. As indicated above, the time limit of one hour for starting and supplying auxiliary emergency power provides ample margin before a temperature rise of safety significance occurs in the reactor system.

References

(1) SEFOR FDSAR, Volume II, Para. 15.2.2, p. 15-1.

(2) SEFOR FDSAR, Supplement 12, Section 1.

(3) SEFOR FDSAR, Volume II, Para. 15.4.2.1, p. 15-3.

(4) SEFOR FDSAR, Supplement 12, Para. 1.3.1, pp. 1-32 ff.

(5) SEFOR FDSAR, Volume II, Para. 15.4.2.2, p. 15-3.

(6) SEFOR FDSAR, Supplement 12, p. 3-5.

(7) SEFOR FDSAR, Supplement 12, Para. 1.4, pp. 1-44 ff.

(8) SEFOR FDSAR, Supplement 12, Para. 3.3.2, p. 3-8.

(9) SEFOR FDSAR, Supplement 19, p. 6.

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 with 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;

$$5.6 \times 10^{8} (\sum_x C_x) \mu Ci/sec.^{(1)}$$

D. The gross instantaneous activity of 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 minotors 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.

H. The inventory of noble gas in a single decay tank must be no greater than 17,000 Ci.

I. The inventory of halogens and particulate with a half-life greater than 8 days in a single decay tank must be no greater than 4.0 Ci.

J. The atmosphere in the refueling cell or the nitrogen zone will not be purged if the gas activity is greater than 2.6×10^{-4} μCi/cc.

Radioactive Waste Control System

Bases

Gaseous waste products from the reactor and auxiliary systems are collected through a manifold. The pressure in this manifold is maintained automatically between 10 and 17.7 psia by operation of one or more of the three radwaste compressors, depending on the flow rate required.[2] The ability to maintain constant pressure in the nitrogen cells and in the argon cell depends on the proper operation of this system. Normally only one compressor will be required to handle the intermittent flow rates in this system. The additional compressors are provided to accommodate certain purging operations[2] which will be performed on an infrequent basis.

Diffusion analysis employing meteorological data from the SEFOR site[3] and methods in References (4) and (5) shows that the release of radioactive gases resulting from failed fuel in the reactor after 15 minutes holdup in the gaseous waste system, at a continuous rate of 29,400 μCi/sec (Specification C-1) would not result in a whole body dose exceeding the 10 CFR 20 value of 0.5 REM in one year.

The maximum dose rate during unfavorable meteorological conditions is calculated to be 16 mREM/hr for a release of 1 Ci/sec. Thus, for a release rate of 125,000 μCi/sec, the maximum dose rate would be 2 mREM/hr. (Specification C-2). Based on this maximum dose rate, a release rate of 125,000 μCi/sec in any two days of a seven consecutive day period would not cause the whole body dose to exceed 100 mREM in a week (10 CFR 20). Release rates between 29,400 and 125,000 μCi/sec are tolerable for longer than 2 days as long as the doses do not exceed 100 mREM value in any 7 consecutive day period. Additionally, the short-term release rate is based on the maximum dose rate resulting from the worst meteorological conditions. These worst meteorological conditions are not experienced most of the time, so that the calculated short-term release rate of 125,000 μCi/sec is conservative. Thus, in any two day period, the dose from a release rate of 125,000 μCi/sec would normally be much less than 100 mREM.

The annual average ground level air concentration was calculated using meteorological data taken at the SEFOR site and methods reported in Reference (4) and (5). The maximum calculated off-site annual average concentration at ground level for a continuous release of 1 Ci/sec is 2.5×10^{-5} μCi/cc.

Adjustment of the release rate to limit ground level concentration to the MPC_a value of I-131 divided by 700, in consideration of the milk production and consumption mode of exposure, results in a stack release rate of .0056 μCi/sec. (Specification C-3).

The maximum concentration off-site during favorable meteorological conditions is calculated to be 4.5×10^{-4} μCi/cc for a release of 1 Ci/sec. A release rate of .056 μCi/sec results in a concentration of I-131 of 2.5×10^{-11} μCi/cc, which is 1/4 of the MPC_a value in 10 CFR 20, Appendix B, Table II.

The gaseous radioactivity will be measured in the radioactive waste system prior to release from the radioactive waste storage tanks and also will be monitored in the stack. The activity of halogen and particulates with half-lives greater than 8 days will be ascertained prior to release by analyzing samples extracted from the waste gas storage tank. The liquid effluents from SEFOR will be controlled on a batch basis. Before release, each batch will be measured for gross radioactivity. During release the activity release rate will be measured.

The specified inventories represent an estimate of the noble gas and halogen inventory which would result from fission products leaking from an acceptable fraction of the fuel rods (equivalent to 100% release from 5 fuel rods at the core central location[6]). Accidental release of the specified inventories results in a whole body and thyroid dose considerably less than guideline values in 10 CFR 100.

The controlling isotope is A-41, which is a product of neutron activation in both the refueling cell (argon atmosphere) and the nitrogen zones (1% argon concentration assumed). Atmospheric dilution during adverse meteorological conditions will reduce concentration by a factor 1.6×10^{-3} at the site boundary. The maximum volume purge rate for the cells is 200 scfm. The specified concentration and the maximum purge rate will produce maximum concentrations at the site boundary less than the MPC_a value for A-41 in Table II, Appendix B, 10 CFR 20.

Other radioactive materials such as Na-24 or noble gas fission products have MPC_a values equal to or greater than A-41, therefore, if they contributed to the measured activity, the specification would have an increased margin with respect to 10 CFR 20.

References

(1) C_x is the concentration of radioisotope x in $\frac{\mu Ci}{mil}$ and must satisfy $\sum_x \frac{C_x}{(MPC)_x} \leq 1$; where the $(MPC)_x$ are equal to the value in 10 CFR 20, Appendix B, Table II.

(2) SEFOR FDSAR, Volume I, Para. 9.3.1.2, pp. 9-2 and 9-3.

(3) SEFOR FDSAR, Supplement 19, Appendix A. Summary of SEFOR Meteorological Data Final Report, May 15, 1967 to May 15, 1968.

(4) A Brief Survey of the Meteorological Aspects of Atmospheric Pollution, H.E. Cramer, Bulletin of the American Meteorological Society 40 (4): 165-171.

(5) Watson, E.C. and Gamertsfelder, C.C., "Environmental Radioactive Contamination as a Factor in Nuclear Plant Siting Criteria," HW-SA-2809, February, 1963.

(6) SEFOR FDSAR, Supplement 21, p. 32.

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. The detachable lower end of the dip-tube drain line shall not be installed when more than 30 irradiated fuel rods are stored in the irradiated fuel storage tank.

C. Items A and B do not apply when the tank contains only new fuel rods and/or no more than one irradiated fuel rod or one irradiated instrumented fuel assembly in each channel, provided the total decay heat load from the fuel rods in the tank is less than the decay heat load of 30 fuel rods, two days after shutdown following infinite irradiation in the center of the core at 20 MWt.

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

E. The 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.

Bases

The sodium temperature in the irradiated fuel storage tank will normally be maintained between 350°F and 400°F.(1) A value of 700°F was used as a basis for its design.(2) The specified upper limit of 500°F provides margin to assure that the design temperature will not be reached. The minimum value of 300°F provides assurance that sodium plugging or freezing will not occur.

The sodium level in the tank will remain well above the top of the fuel region as required by 3.8.B. It is not possible to lower the sodium below the specified level without installing a special dip-tube in the tank.(3)

The irradiated fuel storage tank is designed to store and cool up to one and one-half cores of irradiated fuel using liquid sodium. This capability will not be required before the end of the planned experimental program unless some unforeseen problem occurs. Since refueling is not necessary or planned during the planned experimental program, the only irradiated fuel that will be stored in the tank until the end of the program is a small number of fuel rods (30 maximum) that are part of the fuel surveillance program. Results from analyses have shown(4) that this number of fuel rods, irradiated for an infinite time at 20 MWt, can easily be cooled with only argon gas in the tank. If the decay heat load for a given fuel rod were less than the value assumed in the analyses, the cladding temperature would be reduced, even though the argon temperature remained at the assumed value (210°F). If the total decay heat load from rods stored in the tank were less than the assumed value, the argon temperature would be correspondingly reduced. Consequently, it will be safe to store more than 30 irradiated fuel rods in the tank provided the total decay heat load is less than the specified value. This heat load will be estimated based on the irradiation history of each rod. In addition, the gas temperature in the tank will be indicated by means of the thermocouples normally used to measure sodium temperature in the tank. Therefore, the specification is written to allow storage of more than the expected maximum of 30 irradiated rods without sodium in the tank to provide for the possibility of temporary storage of additional irradiated rods during investigation of unexpected reactor problems. The limit of one irradiated rod per channel (163 channels in the tank) limits the maximum cladding temperature to less than the normal operating temperature for fuel in the reactor. An exception is also made to permit storage of an irradiated instrumented fuel assembly, which contains

two fuel rods, in a channel without adding sodium to the tank. The number of irradiated IFA's is limited as there are only six positions in the reactor in which these assemblies can be installed. Calculations discussed in Reference 4, show that up to three irradiated fuel rods could be placed in a channel without exceeding normal operating clad temperatures.

The storage array and materials used in the tank are designed to permit storage of one and one-half reactor core loadings of fuel in such a manner that non-criticality is assured. A radiation monitor is provided to assure warning of any unforeseen increase in the criticality factor.

The gas pressure will normally be set at 1/2 psig. A value of 5 psig was used for tank design requirements, and the tank has been hydrostatically tested to a pressure equal to 1.5 times the sum of 5 psig cover gas pressure plus 127 inches of sodium at 400°F.[5] A rupture disk-type safety relief valve designed for 5 psig is installed in the tank cover gas system.

The tank cover shield blocks will normally be removed one at a time, which requires a lift of only one foot to permit stacking on an adjacent shield block. If all three shield blocks are removed and stacked on the head storage stand (as provided for in the equipment design), the third shield block would have to be raised about 6.5 feet above the cell floor. The tank and shield blocks have been designed so that if a shield block were dropped inadvertently from a height of up to 10 feet 9 inches above the cell floor (12 feet above the tank flange), it would not be able to enter the tank and damage the fuel rods.[6,7]

References

(1) SEFOR FDSAR, Supplement 13, Para. 2.2.4, p. 5.

(2) SEFOR FDSAR, Supplement 13, Para. 2.3.2, p. 6.

(3) SEFOR FDSAR, Supplement 13, Para. 2.4.8.1, p. 20.

(4) SEFOR FDSAR, Supplement 21, p. 34.

(5) SEFOR FDSAR, Supplement 13, Para. 2.5.5, p. 24.

(6) SEFOR FDSAR, Supplement 8, Answer to Question 6, p. 18 ff.

(7) SEFOR FDSAR, Supplement 19, Answer to Question 3, p. 12 ff.

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. Reflector segments may be raised while the reactor head is removed if the Wide Range Monitor contains a stop which limits the high flux scram set point to 500 KWt or less.

C. Handling operations shall not be conducted in the refueling cell unless the reactor is secured.

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

E. If the core is unloaded to less than the minimum critical mass, but not less than 432 fuel rods, fuel rods shall not be reloaded into the core and reflectors shall not be raised unless two neutron detectors are operating and registering more than two neutron counts per second.

F. When the core contains less than the minimum critical mass, criticality checks shall be made between loading increments by raising all reflectors and noting the change in count rate on at least two neutron detectors. If the core contains less than 432 fuel rods, loading increments shall not exceed 108 fuel rods. If the core contains more than 432 fuel rods, no more than six rods in excess of one-half the number of additional rods predicted for criticality shall be loaded in one increment.

Bases

During fuel handling operations, the sodium level will be lowered from the operating level to the refueling level. The auxiliary coolant system is designed to be operable over the complete design flow range with the vessel sodium level at the refueling level, and will be used to remove reactor decay heat, if necessary, when the vessel head is removed.(1) The main coolant system could also be used for this purpose but its operation would be restricted to low flow rates to avoid gas entrainment with the sodium level in the reactor vessel at the refueling level. Operating considerations make it more practical to rely on the auxiliary system for cooling for this condition.

The reactor vessel sodium level will be indicated by means of "J" probes, and will be lowered to the refueling level by draining sodium through the refueling nozzle. Since this nozzle is located about 3 inches above the top edge of the main outlet nozzle, the reactor vessel sodium level will always remain high enough to permit operation of either the main or auxiliary coolant system.

If the core contains less than a critical mass (Specification 1.3A(1)) and the decay heat is less than 1 KWt, the decay heat can be dissipated by heat losses from the reactor vessel and operability of a coolant loop is not required.

During criticality measurements after each incremental loading of fuel is completed, and during wet critical testing, it will be necessary to raise all reflectors and check criticality. The flux will be monitored during the loading process by at least two neutron detectors. The low-level reactor sodium trip will be bypassed to avoid the necessity for repeated changes in the reactor sodium level between the refueling level and the normal operating level. Adequate core cooling will be provided by operation of the auxiliary coolant system. The low level source range trip must also be bypassed, since the flux level will be very low until a large percentage of the core has been loaded with fuel.

When the full core loading is accomplished, physics data will be obtained on material worths, flux profiles, and reflector calibration by operating the reactor at power levels up to 100 KW. The vessel head will remain off during this operation. The neutron flux above the reactor vessel at 100 KW with the head off is comparable to the flux at 20 MWt with the head on. However, the gamma radiation level may be higher, depending on the duration of operation

and the consequent sodium activation. The maximum dose rate was calculated to be 3.5 mREM/hour outside the refueling cell with the reactor at 100 KW with the head off. Personnel access to this area will be controlled according to standard procedures (Section 6.2).

The stop limiting the high flux scram set point to 500 KW provides assurance that the auxiliary coolant system will be able to remove the heat generated in the core. This system is designed for a capacity of 2.5 MWt.

Material movement is not allowed in the refueling cell when the reactor is operating with the head removed to avoid the possibility of inadvertently changing the reactivity of the core by dropping objects into the core.

The reactor sodium temperature will normally be held between 350°F and 400°F during refueling operations. The upper limit of 450°F is set to avoid problems with sodium vapor at higher temperatures. Tests have indicated that above about 550°F significant sodium vapor is generated which could obscure vision in the refueling cell.

The minimum critical mass for Core I has been determined experimentally and corresponds to 522 standard fuel rods. The minimum critical mass for Core II will be greater than that for Core I by the equivalent of 24 to 40 rods, depending on the changes made. This number will be estimated, but will not be determined experimentally. The changes will depend on the test results obtained from Core I. It is expected that the Core II configuration will be achieved by removing BeO and B_4C from the core and adding fuel and steel to replace this material. Thus, unloading and re-loading of the core is not part of the planned program. If it became necessary to unload and then reload the core, relatively large loading increments could be safely used up to some number of rods significantly less than the minimum critical value, since the loading would be based on known nuclear parameters. Four increments of 108 fuel rods each provide a logical loading plan to obtain criticality checks with symmetrical fuel arrays. After 432 fuel rods are in the core, the source range monitors should be registering more than two counts per second, and will provide a clearly detectable signal for use in monitoring changes in count rate during additional fuel loading. Since the minimum critical core size, worth of fuel rods and other materials, and reflector worths, have all been determined experimentally, a minimum loading increment

of six fuel rods (one channel) is permissible during subsequent loading to criticality. The reactivity worth of six fuel rods added to the outer edge of the minimum critical core is about 0.8$, which is less than the worth of one reflector segment (1.2$).

<u>Reference</u>

(1) SEFOR FDSAR, Volume I, Para. 5.3.2, pp. 5-11 ff.

3.10 Approach to Power

Applicability

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

Objective

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

Specification

A. Reactor power shall be limited to 2 MWt initially. This limit shall be successively increased to values of 5, 10, 15, 17.5, and 20 MWt provided the conditions listed in Section 3.3C, D & E are satisfied at each of these limits in the approach to rated flux (power). Satisfactory results obtained at a given limit shall permit reactor operation up to and including the next scheduled step in the approach to power.

B. If at any power level, the analysis of the conventional oscillator tests indicates that the stability criterion of specification 3.3 F will not be met at some higher level of power, the reactor power may be raised only as high as the halfway point between the level at which the test is made and that at which the failure to meet the specification is indicated.

C. The reactor power limit shall not be increased above 15 MWt or above 17.5 MWt unless analysis of results from guinea pig fuel rod examinations shows that no damage to standard fuel rods is to be expected by operation at the next scheduled power level.

D. A reactor heat balance shall be made as soon as practicable after achieving steady state power levels of 5, 10, 15, 17.5, and 20 MWt, to determine the correlation between rated flux and reactor power.

E. Initial operation at 20 MWt and excursion tests shall not be performed unless the specifications identified in this section, and their bases, have been approved and incorporated into the SEFOR Technical Specifications. The General Electric Co. shall submit proposed specifications for Items 1 and 2 below to the Director, Division of Reactor Licensing, for approval, based on experience gained in operating SEFOR up to and including the 10 MWt power level and the then current state-of-the-art.

1. Quantitative definitions of limits of unexplained behavior of reactivity, reactor cover gas activity, and other parameters as discussed in Specification 4.9.B.1.
2. Necessary detection steps and specific criteria for judging the acceptability of continued steady state operation in the presence of a known loss of fuel clad integrity.

Bases

The reactor power limit will be increased in a step-wise manner with static and oscillator measurements made at the indicated power levels. The results from tests at each power level will be evaluated and compared to predicted results before proceeding to the next higher power level. Results from static and oscillator tests will be analyzed to verify that the minimum conditions for operation specified in Section 3.3 C, D & E are being met.

Reactor stability will be determined by means of conventional oscillator tests at each step in the approach to power. These tests will consist of measuring reactor flux and input reactivity as a function of time while the reactivity is oscillated and coolant flow rate is held constant. Data from these tests will be used to make Nyquist plots for each power level.

Guinea pig fuel rods[1] containing fuel pellets of 25% fissile plutonium will be placed in the core at positions located under through-head refueling ports, and will be removed for examination at scheduled intervals in the test program. Up to three of the guinea pig rods will operate at power densities up to 15% higher than a standard rod nearest the center of the core.

The specified guinea pig rod examinations after operation at power levels of 15 and 17.5 MWt were chosen such that satisfactory operating experience with the guinea pig rods at each of these power levels will provide assurance of satisfactory operation of standard fuel rods at the next higher power level.

Guinea pig rods nearest the center of the core will be removed prior to reactor operation above 17.5 MWt, so that no fuel rods will be operated at power densities in excess of that experienced by the hottest standard fuel rod at 20 MWt. (See Specification 3.3.H).

The initial calibration of the Wide Range Flux Monitor will be based on physics calculations. This calibration will be verified by experimental data as soon as practicable, and will be checked at the specified steps in the approach to power.

Reactor operating data and experience will be used to establish allowable limits for unexplained changes in reactivity and reactor cover gas activity. These data will be obtained during initial reactor operation up to and including 10 MWt. Observation of unexplained changes beyond these limits will require the actions specified in Section 4.9.

Reference

(1) SEFOR FDSAR, Volume I, Para. 4.2.2.4, p. 4-9.

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 less than $\pm$20% of the indicated power level.

B. The amplitude of reactivity oscillator shall be less than $\pm$10 cents.

C. The amplitude of oscillation in the main primary and main secondary coolant flow rate shall be less than $\pm$1000 GPM.

D. The amplitude of the coolant temperature oscillation at the vessel inlet or at the vessel outlet shall be less than $\pm$60°F.

Bases

Oscillator tests will be run by oscillating the reactivity, main primary coolant flow rate, and main secondary coolant flow rate, individually or concurrently, at controlled amplitudes, frequencies, and phase relationships. The reactivity oscillation will be accomplished by oscillating a poison rod, worth about 50¢, located in the center channel. The maximum reactivity amplitude of the oscillator will be $\pm$10¢. Most of the oscillator tests are expected to be run with reactivity amplitudes of less than $\pm$6¢. The reactivity amplitude and power amplitude limits are based on test program requirements for linear response.

The coolant pumps, heat exchangers, reactor vessel, and piping in the main coolant systems have been designed so that mechanical stresses associated with flow oscillations of $\pm$1000 GPM at 2 cycles per minute will be well below the fatigue limit for each component. The maximum frequency of flow oscillation will be 2 CPM. The maximum amplitude will be $<\pm$20% of the set point value, since the safety system will scram the reactor if the coolant flow rate falls to 80% of the set point value.

Oscillation of flow rates or reactivity will cause coolant temperature oscillations. The amplitude of the temperature oscillation will depend on the amplitudes and phase relationships between reactivity and coolant flow oscillations. In some tests, the temperature amplitude will be zero. The maximum amplitude is expected to be less than 30°F. Each of the components mentioned above has been designed such that temperature oscillations of $\pm$60°F at the reactor vessel inlet or outlet will not result in thermal stresses above the fatigue limit.[1]

All values of Limiting Safety System Settings are applicable during the oscillator tests. Therefore, all parameters will remain within their normal operating limits during these tests. The oscillator tests have been simulated on the analog model of SEFOR.[2] These simulations showed that no combination of oscillating parameters within the limits specified in this section would result in unsafe operation of the reactor.

References

(1) SEFOR FDSAR, Supplement 16, Section IX.C, Table 2.

(2) SEFOR FDSAR, Volume II, Appendix A.

3.12 Excursion Tests

Applicability

These limits apply when excursion tests are conducted with the Fast Reactivity Excursion Device (FRED). The prompt critical test program for Core I and Core II shall not be initiated until DRL has completed its review of the special reports described in Specification 6.6.B.5 and determined whether or not additional specifications are required.

Objective

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

Specifications

A. Experimental Program with FRED

The experiments with the FRED shall be carried out in three phases as indicated below. Progress to the next phase shall be contingent upon adequate agreement with predicted results.

1. Familiarization Tests
   a. The worth of the poison slug used in these tests shall be 0.5$ or less.
   b. The initial reactor power level shall be equal to or less than 15 MWt.
2. Sub-prompt Critical Tests
   a. The worth of the poison slug used in the sub-prompt critical tests shall be 0.98$ or less.
   b. The initial reactor power level for sub-prompt critical tests with the FRED shall be equal to or less than 15 MWt.
3. Prompt Critical Tests
   a. The worth of the poison slug used in the prompt critical tests shall be equal to or less than 1.3$ if the magnitude of the sodium-in negative Doppler coefficient ($T\frac{dk}{dT}$) is equal to or greater than 0.008. The worth of the poison slug shall be equal to or less than 1.2$ if the magnitude of the sodium-in negative Doppler coefficient ($T\frac{dk}{dT}$) is less than 0.008.
   b. The initial reactor power level for prompt critical tests shall not exceed 11 MWt.

B. General

1. An excursion test shall be performed with the FRED only if the analysis of previous operating data indicates that the sodium-in Doppler coefficient ($T\frac{dk}{dT}$) is negative and has a magnitude equal to or greater than 0.005.
2. Th time required for the poison slug to travel the first 20 inches after lift-off shall be equal to or greater than 0.085 second.
3. The main primary coolant flow rate shall be at least 4000 GPM when the poison slug is ejected from the core.
4. The scram system may be modified in accordance with the description in Supplement 17 to the FDSAR, page G-1, to provide an additional 400 millisecond delay between a scram signal and the point in time when the reflector segments start to move.
5. Whenever a poison slug worth more than 1$ is lowered into the core by means of the FRED, containment integrity shall be maintained and the isolation valves on the outer containment ventilation lines shall be closed.
6. If fuel rod inspections called for in Section 4.3 indicate that the limits of fuel defects (as defined in Section 3.3.K.2 and 3) are being approached, succeeding excursion tests shall be limited to power excursions below that at which such behavior was observed.
7. The initial reactor power level for excursion tests shall be equal to or greater than 0.1 MW.
8. System checkout of the FRED components may be performed at power levels less than 0.1 MW, provided the initial position of the poison slug is more than 20 inches above the core midplane.
9. The core coolant inlet temperature shall not be below 700°F at the start of each excursion test.
10. Excursion tests shall not be conducted if there is evidence that the core contains defective fuel rods.

Bases

The experimental program with FRED is graded so that small transients precede larger transients. The information from the small transients will be used - (1) to evaluate the performance of the reactor, (2) to compare the performance with predicted behavior, and (3) to predict performance of the reactor for the larger prompt critical transients.

The characterization of the tests into the categories of (1), (2) and (3) above is self-explanatory. The maximum power levels indicated in each case are to assure that the safety limits as given in Section 2.1 will not be violated and are in accordance with Figure 2.1-1 and the explanation in Section 2.1.

The value of the Doppler coefficient for SEFOR Core I with sodium in the core is estimated to be $T\frac{dk}{dT} = -0.0085$. This value was verified experimentally by means of Doppler measurements on the SEFOR mockup in the ZPR-III Critical Facility.[1] From further measurements on this mockup, it was established that the Doppler coefficient with sodium out is 17.5% lower than the value with sodium in, or $T\frac{dk}{dT} = -0.0070$ for SEFOR Core I with sodium out.

The safety analysis of the MHA for SEFOR was based on a sodium-out Doppler coefficient of $T\frac{dk}{dT} = -0.004$, which corresponds to a sodium-in Doppler coefficient ($T\frac{dk}{dT}$) of -0.005.[2] The demonstration of a negative Doppler coefficient ($T\frac{dk}{dT}$) with a magnitude equal to or greater than 0.005 during the approach to maximum power will verify predictions of this coefficient based on the ZPR-III measurements and will provide the basis for safe performance of prompt critical tests in SEFOR.

The total reactivity worth of each poison slug used in the FRED will be known and the value will be checked before each transient test. The maximum reactivity insertion rate will be limited to less than 20$ per second by limiting the reactivity worth of the slug to 1.3$ and by limiting the minimum allowable time for the slug to travel the first 20 inches to .085 seconds.[3] This time will be measured by means of the lift-off switch and a proximity switch which marks 20 inches of travel by the poison slug. The safety of the plant has been assessed for a maximum rate of 50$ per second with a sodium-out Doppler coefficient ($T\frac{dk}{dT}$) of -0.004.[4]

The minimum allowable coolant flow rate is set at 4000 GPM to assure adequate core cooling capability during the excursion tests.[5]

Analyses of the requirements for excursion tests show that reactor scram must be delayed for an interval of up to 0.4 second following firing of the FRED in order to obtain needed experimental data.[6] The safety analysis of the transient operation of SEFOR was performed using a 400 millisecond delay in addition to the normal scram system delay. In addition, the conclusion of all safety analyses of SEFOR would not be affected by the 400 millisecond delay since at rated power this delay would correspond to an addition of only 8 MW seconds to the core energy content. The MHA assumes no prompt scram, but rather a scram occurring several seconds after a power excursion so as to significantly increase the severity of the accident. The time delay circuit will not be installed until the start of the excursion tests program and will be interlocked so that it will be effective only when the Master Mode Switch is in the "EXCURSION TEST" position.

Containment integrity will always be maintained during reactor operation. Specification B-5 provides additional assurance of containment integrity in the event of an accidental release of radioactive material to the outer containment region during the prompt critical testing.

Fuel rods will be examined non-destructively prior to start of the excursion tests and at selected intervals during the tests to verify that a fuel rod defect has not occurred. Results[7] from the transient fuel irradiations program carried out as part of the SEFOR Pre-operational Research and Development Program have shown that the profilometer measurements that will be made on the fuel after each excursion test will detect fuel rod expansion well before fuel defects with significant safety implications occur.

The minimum limit on initial power level limits the rate of energy release during the excursion. The limit provides assurance that the impulse load on the vessel internals will not exceed allowable limits.[8] The magnitude of the impulse load is not strongly dependent on the initial power. As an example,reducing the initial power of the Maximum Planned Transient from 9 MWt to 0.09 MWt increases the impulse load 24%. However, the impulse load corresponding to the safety limit given in Section 2.1 is approximately 800% larger than the Maximum Planned Transient, so the impulse load for transients

starting from initial power levels as low as 0.1 MWt is still well below the safety limit.

Initial checkout tests of the FRED after it is installed on the reactor head will be performed with the reactor either sub-critical or at low power level (less than 0.1 MWt). The FRED will have a negligible effect on reactivity when it is in a position more than 20 inches above the core midplane.

The minimum limit of 700°F on the core coolant inlet temperature is to assure that the total reactivity of the core is maintained at the equivalent of 50¢ excess at 20 MW conditions.[9] At lower temperatures, the excess core reactivity would be higher. The 50¢ excess limit assures that the Maximum Planned Transient will not be initiated from a power level in excess of 11 MWt, and also limits the final reactor power if the reactor does not scram and the FRED slug remains out of the core following any excursion test.

References

(1) SEFOR FDSAR, Appendix B, Section B.5, p. B-3.

(2) SEFOR FDSAR, Section 16.4.2.6.1.1, p. 16-28.

(3) SEFOR FDSAR, Volume II, Section 13.4.3.

(4) SEFOR FDSAR, Volume II, Section 16.2.7.

(5) SEFOR FDSAR, Volume II, Section 16.2.7, p. 16-10.

(6) SEFOR FDSAR, Supplement 17, p. G-1.

(7) SEFOR FDSAR, Supplement 3, Section 5.1.3.

(8) SEFOR FDSAR, Supplement 19, p. 57.

(9) SEFOR FDSAR, Volume II, Section 12.3.6, pp. 12-15, 16.

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 facility operation and do not apply in the event of reactor shutdown for periods longer than the specified interval between tests. If a surveillance function has not 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. Determination of excess reactivity and reactor heat balances must necessarily be made after reactor startup. 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.

For extended periods of open head operation of the reactor, surveillance time intervals may be extended, for systems or components whose safety function is rendered inoperable, until the reactor head is replaced.

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. Channels shall be tested individually to assure continued independence as well as operability.

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 (3)	(1) Check of DC current to the probe. (2) Change of process level to effect a level trip. (3) Initial calibration shall be based on verification of probe position. Subsequent calibration shall use J-probes as a reference.
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				
e) Aux. Primary Pump Duct Wall				
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	
High Differential Pressure Monitor (Reactor cover gas - drain tank cover gas)	N/A	1/mo	1/6 mo	

TABLE 4.1-1 (Cont.)

Channel	Channel Check	Channel Test	Channel Calibration	Remarks
Sodium Leak Detector	N/A	1/mo (1)	1/6 mo (2)	(1) Apply a test short outside containment. (2) Apply a test short at detector connector, and also 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 radio-active material.
Waste Gas Discharge Radiation Monitors	1/d	1/wk	1/6 mo	Test prior to each release of radio-active material
Spent Fuel Storage Monitor	N/A	N/A	1/6 mo	Test prior to each movement of fuel to or from the irrad-iated fuel storage tank.
Area Radiation Minitor	1/d	1/2 wk	1/6 mo	Calibrate with radioactive source.
Refueling Cell Radiation Monitor	1/d	1/wk	1/6 mo	Calibrate with radioactive 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.
Reactor Fission Gas Monitor	1/d	1/2 wk	1/6 mo	Calibrate with radioactive source.

Reactor Safety System

Bases

Instrument check is performed at the shortest time interval deemed practical considering this mode of operation. The reactor is normally operated in a series of short test runs, usually of less than one day duration so that the reactor is started up and shut down each day.

The elapsed time between Tests is selected to make the expected percentage of unprotected time for a particular function to be less than 10^{-2}%. The elapsed time between Tests for the various protection function trip logic combinations was calculated using the following formula,(1) where f is the single channel unsafe failure rate:

Logic	One-out-of-Two	One-out-of Three or More	Two-out-of-Three or More
Elapsed time between Tests	$\leq \sqrt{\frac{3x10^{-4}}{f}}$	$\leq \sqrt{\frac{4x10^{-4}}{f}}$	$\leq \sqrt{\frac{10^{-4}}{f}}$

Unsafe failure rates for the IN/MAC (GE-NEBS manufactured nuclear instrumentation) solid-state instrumentation are based on both experience and analysis. Unsafe failure rates were calculated by use of component failure rate data and analysis of circuit operation. Experience has been accumulated at the KRB nuclear power facility in Germany from August, 1966 to April, 1968. From this experience, the Total Failure Rate $\leq$ $3x10^{-4}$ failures/operating hours. No unsafe failures were reported; however, one can show by circuit analysis that the unsafe failure rate is at least a factor of ten (10) less than the total failure rate. We have conservatively used $3x10^{-5}$ as the unsafe failure rate.

Unsafe failure rates for the GE/MAC (GE-West Lynn manufactured industrial instrumentation) solid-state instrumentation are based on experience from several industrial installations that have reported their failures for a considerable period of operation.

For the MV/I the following data are available;

4 failures from 30 inst each operated 5780 hrs

no failures from 11 instr each operated 18,785 hrs

no failures from 12 instr each operated 12,000 hrs.

These data support a mean time to failure of 53,000 hrs or larger at the 90% confidence level.

For the Trip Unit the following data are available:

no failures from 100 instr each operated 5780 hrs

no failures from 8 instr each operated 18,785 hrs

no failures from 8 instr each operated 12,000 hrs

These data support a mean time to failure of 340,000 hrs or larger at the 90% level.

From these results the unsafe failure rate for these instruments is estimated to be $2x10^{-6}$ failures/hr for the MV/I and $3x10^{-7}$ for the Trip Unit.

Test intervals for the balance of system or sub-system components are based on failure rate data reported in Reference 2.

Calibrate intervals are based on both experience and long-term drift specifications reported by manufacturers for the instruments. The drift is considered to be a systematic error and the calculated deviation in the specified interval will not result in any unsafe failures. Due to the nature of operation of most of the protection function channels, instrument operational drift has practically no effect on safe operation, since drift is accounted for in the trip set point, but calibration is deemed important in the sense of preventive maintenance to assure that the previous good experience is continued.

References

(1) Rasmussen, Jens and Timmermann, P., Safety and Reliability of Reactor Instrumentation with Redundant Instrument Channels, Danish Atomic Energy Commission, Risö Report No. 34, Jan. 1962.

(2) Green, A.E., Bourne, A.J., Safety Assessment with Reference to Automatic Protection System for Nuclear Reactors, British Report AHSB (SR.117), Parts I, II, & III, 1966.

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, except that the determination of available excess reactivity (3.3B) shall not be required until a steady state reactor power of 10 MWt has been achieved. 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 occurred which would result in exceeding the limits specified in Paragraphs 3.3A, 3.3B, and 3.3C. These daily checks shall also be used to verify the correlation between reactor power and neutron flux.

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

C. An operability test for the reflector segments and scram system shall be performed before each scheduled reactor startup, but this test shall not be required more often than once per day; 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 such that all ten are so tested before the cycle is repeated.

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.

Bases

The shutdown margin and excess reactivity of the reactor are not expected to change significantly over a long period of time unless core modifications are made. Therefore, a demonstration of the limits specified in 3.3A and 3.3B once every four months is adequate. In addition, the reflector position required to achieve criticality will be checked during each startup and daily checks of reactor heat balance data will be made. Since reactor operation for extended periods of time is not planned, any change in shutdown margin or reactivity will be noted without delay. In the case of core rearrangement, measurements will be made using the calibrated control segments. Results from these measurements will be used to extrapolate to the conditions specified in 3.3A and 3.3B using the best available data for the extrapolations.

The excess reactivity available at 20 MWt must be determined from test data at a power level high enough to provide reasonable accuracy. It is also desirable to determine this value before a large amount of test data have been obtained, in the event that an adjustment in core loading is required. A power level of 10 MWt satisfies these requirements. Preliminary determination of core excess reactivity will be made at as low a power as is practicable.

The four month interval for shutdown scram and drive speed test is adequate to minitor wear and other factors that might affect reflector speeds. Equipment wear in four months is expected to be negligible as indicated during R&D testing of the drive systems.(1) During the prototype test drive and scram motion equivalent to more than 10 times that expected in the three year experimental program was checked.(2) In all of the tests the drives never failed to scram when the initial action was taken, and no detectable change in drive performance was observed.

The operational check of the scram system for each reflector during each reactor startup will verify the operability of the most important components of the safety system. The frequent checking of these components added to the complete system test at four month intervals assures proper operation of the reactor scram system.

Extended periods of reactor operation are not planned at this time. However, if this should occur, the required movement of each reflector rod will assure its continued operability.

References

(1) SEFOR FDSAR, Supplement 11, Section 7.

(2) SEFOR FDSAR, Supplement 11, p. 7-16.

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. After reaching a power level of 15 MWt and before reaching 17.5 MWt, the interval between fuel rod examinations shall not exceed 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 fuel rod should indicate a defect as described in Section 3.3K, additional fuel rods shall be examined to determine the extent of additional defects if any.

Bases

The bases for fuel rod examinations specified in this section are given in Section 3.10, "Approach to Power", and Section 3.2, "Excursion Tests". The same fuel rod will be chosen for examination following each test, insofar as practicable, to provide comparative data on the effects of each test.

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 limiting conditions for operation are met.

Specification

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 if it contains sodium.

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

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

E. The Marmon clamp connections on the auxiliary primary reactor vessel inlet and outlet dip-tubes shall be leak tested at least quarterly.

F. The vacuum breaker system for the reactor cover gas shall be functionally tested at least quarterly with at least five tests performed prior to reaching 10 MWt operation.

G. The capability of isolating the primary drain tank shall be functionally tested at least quarterly.

H. The argon vent valve for rapid venting of the primary drain tank shall be functionally tested at least quarterly.

I. The system capability required to meet specification 3.4.G shall be demonstrated at least annually.

J. 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.

K. The plugging temperature of the primary sodium system shall be measured daily when the plugging temperature exceeds 400°F and at intervals not to exceed one week when the plugging temperature is below 400°F.

L. The plugging temperatures of the main secondary and auxiliary secondary sodium systems shall be measured daily when 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.

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

N. The argon cover gas system and the argon vent vacuum pump shall be operationally tested at least monthly.

O. It shall be demonstrated at least once prior to 10 MWt operation and yearly thereafter, that the remote refilling of the primary auxiliary coolant loop can be accomplished within 30 minutes by the minimum operating staff using the emergency procedures.

P. During initial ascent to full power operation, cover gas samples for spectral analysis shall be taken to the extent necessary to establish baseline data on cover gas activity levels as a function of reactor power. Dependence on temperature and flow will also be determined. Thereafter, spectral analysis shall be performed at least monthly, before and after each series of FRED transients below 90¢, before and after each transient above 90¢, and at any time the gross-gamma monitor indicates an unexpected increase in cover gas activity.

Q. Samples of the primary coolant shall be taken for analysis, at intervals not to exceed three months and following each prompt-critical FRED transient.

Bases

Leak checking of the reactor safety vessel and the fuel storage tank safety vessel will be accomplished by establishing a pressure differential between the safety vessels and the surrounding atmosphere, then monitoring for pressure change. No deterioration of these vessels is expected in a six month period because they are located in an inert, non-corrosive gas environment and are not subject to wear or significant loading conditions. Leak testing of the irradiated fuel storage tank is not required until it is filled with sodium and put into service.

The three month interval for testing the check valves is believed sufficient to assure satisfactory operation. A check valve of identical design and fabrication was tested at SEFOR operating conditions in a sodium loop for a period of 1500 hours. This would correspond to 62 days of continuous operation at full-flow conditions. The tests included periodic flow reversals to check operability of the valve. No deterioration in performance of the valve was observed, and a visual inspection following the test showed no change in appearance of the valve components.

The quarterly intervals for testing the functional performance of the Marmon clamps, the reactor vessel vacuum breaker system, and drain tank isolation valves, are based upon satisfactory operation of these components in sodium test loops and other sodium systems. A three month interval would be adequate to detect a gradual deterioration in performance. A sudden change in performance could not be detected without continuous monitoring, which would be impractical for these components. The five tests of the vacuum breaker system are required before reaching 10 MW to demonstrate the functional reliability of this system. It is noted that the probability that these components would be called upon to function in their intended manner is extremely low. The probability that this situation would arise coincidentally with an undetected failure of these devices is even more remote.

The quarterly interval for testing the functional performance of the argon vent valve used for rapid venting of the primary drain tank is based on satisfactory operation of this type of valve in other systems.

The capability of reducing drain tank pressure at a faster rate than normal is provided to minimize the possible loss of sodium in the event of a major pipe break, combined with failure of an engineered safeguard.[1] This

capability is tested only once a year to avoid unnecessarily severe testing of a system which is required for protection against a low probability event.

The maximum rate at which the drain tank pressure can be reduced is determined by the initial value of the drain tank pressure and the gas flow rate provided by the radwaste compressor. The pressure of the gaseous radwaste decay tank which is on-line will have a secondary effect on the system capability, since an increase in this pressure will tend to reduce the volumetric efficiency of the compressor.

The initial demonstration of system capability will show that the drain tank pressure can be reduced by 10 psi in 10 minutes or less, and will determine whether or not the maximum decay tank pressure must be limited in order to meet this requirement. Pressure and temperature measurements from this test will be used to calculate the gas flow rate. Subsequent demonstrations of the venting capability will consist of shorter tests to avoid possible carry-over of sodium vapor from the drain tank to the radwaste system. For these tests, the drain tank pressure will be reduced approximately 2 psi. Pressure and temperature data from these tests will be used to show that the system flow rate is sufficient to meet specification 3.4.G by comparison to the original test data. The decay tank pressure will be maintained below the limit determined by the initial test described above.

Tensile specimens shall be removed from the reactor vessel following the 3rd, 4th, 5th, 8th and 10th years of reactor operation. These specimens will be tested and compared to control specimens.[2,3] The program follows the recommended practice of the American Society for Testing and Materials and is based on the most recently published data concerning irradiation effects on Type 304 stainless steel as well as the BRDO cladding development program.

During initial operation of the primary sodium system, or after the system is exposed to the refueling cell atmosphere, plugging temperatures may exceed the maximum allowable value of 425°F, and cold trapping will be required. Plugging temperatures will be determined daily or more frequently to assure that the cold trap is operating effectively. When the plugging temperature has been reduced to 400°F, it should not increase unless the system is re-exposed to a source of contamination. After stable conditions have been achieved, as indicated by daily measurements of plugging temperatures below 400°F over a period of one week, the intervals between plugging temperature determinations will be increased to a maximum of one week. More frequent

determinations will be made at any time the system is exposed to possible contamination, as indicated by an increase in the plugging temperature, until it is re-established that the plugging temperature is stable below 400°F. More frequent determinations will also be made when the reactor head is removed to assure that the plugging temperature remains at least 25°F below the primary system temperature.

The main and auxiliary secondary systems are closed, separate systems. Once a low plugging temperature has been established in each system, no reason has been identified for it to increase. During initial operation of each system, and whenever the system is exposed to possible contamination, plugging temperatures will be measured more frequently than specified herein, until stable conditions have been achieved. When the plugging temperature has been reduced below 300°F, monthly determinations will provide adequate surveillance.

The irradiated fuel storage tank is essentially separate from the main coolant system and will remain a closed system for long periods of time. Surveillance requirements are similar to those specified for the secondary coolant systems.

The operation of the argon cover gas system is checked each time there is a change in the primary coolant temperature requiring a feed or bleed of argon gas to maintain cover gas pressure at 20 psig. This occurs during reactor startup and shutdown or during significant power changes. The tests at monthly intervals will provide a systematic check of the entire system. The operation of the argon vent vacuum pump will be checked monthly by demonstrating its capability to pull a vacuum on the argon vent tank.

Refilling of the auxiliary primary loop within a given time is required only in the event of a major pipe break accident, and demonstration of this capability once a year is considered adequate. The time required to refill the loop was estimated to be about 30 minutes, based on the time required to exhaust the vent tank and cycle the valves, and on the number of times this had to be done to refill the loop.[4] Demonstration of this capability must be done with the main loop filled, since it would be unwise to drain both loops for a surveillance test.

The bases for the cover gas monitor are given in Section 3.3.

Sodium samples will be taken periodically and analyzed and the results compared with data obtained from the analysis of reactor cover gas samples. The interval of three months between samples is adequate to establish background information and to identify long term trends in fission product activity in the primary coolant. The frequency of sodium sampling will be increased during the planned prompt transient testing when samples of the primary coolant can be readily obtained without significantly affecting the reactor operating schedules.

Each sample obtained for analysis will be examined for evidence of fission products by means of a multi-channel analyzer (gamma scan). Samples will be analyzed for evidence of carbon and metallic impurities quarterly and whenever such analyses are expected to help explain anomalous reactor behavior.

References

(1) SEFOR FDSAR, Supplement 23, pp. 3,4,5.

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

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

(4) SEFOR FDSAR, Supplement 18, p.7.

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, quarterly and following each closure.
   c. Vacuum breaker valves and reactor building ventilation valves: 30 psig and 1 psig. Operability of the vacuum breaker valves shall also be demonstrated.
   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 trip 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. Gaseous Radwaste Release Flow Control Valve

The flow control valve shall be checked quarterly for closing in response to a signal from the waste gas discharge radiation monitor channel.

I. Liquid Radwaste Discharge Pump

The pump shall be checked quarterly for shutdown in response to a signal from the liquid waste discharge radiation monitor channel.

J. Oxygen, Water, and Freon Content of Inner Containment Atmosphere

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.

K. A sample of condensate water removed from the nitrogen cooling system dryer shall be obtained for pH analysis quarterly after the reactor has operated at 1 MWt. If the pH of the condensate deviates outside the range 5.0 -9.5, specimens located within the inner containment shall be examined for evidence of corrosion within 90 days.

Bases

Test frequencies are specified to be consistent with current practices and proposed guides.[1] Variations in testing periods are allowed to permit scheduling the tests at convenient intervals in the experimental program. For example, in some instances, if the scheduled test comes up during a planned sequence of experiments, it may be convenient to defer the test until the series is completed.

Repairs or modifications to the containment barriers, its appendages, or penetrations must leave the barriers at least equal to the original barrier design requirements. Tests to verify this must be performed before containment integrity is assured.

The leakage rate through each penetration through the outer barrier will be determined by measurement of the rate of pressure decay after pressurization of the penetration to 30 psig. For a given leakage rate, the rate of pressure decay depends on the volume being tested. Since the volume contained in each penetration is small, a one hour test period will be sufficient (rather than the 24 hour period required for the integrated leak test).

The oxygen content of the argon and nitrogen zones and the freon content of the nitrogen zone will be continuously recorded during normal operation. The recorder charts will be checked daily to verify compliance with specified limits.

The water content of the argon cell atmosphere will also be continuously recorded and checked daily. Water content of the nitrogen cell atmosphere will be checked weekly or whenever the oxygen content shows evidence of inleakage of air. A dehumidification unit for the nitrogen zone will maintain the dewpoint below 40°F. The argon purifier will maintain the water content in the refueling cell atmosphere below 125 ppm.

If fluorine is dissolved in the condensate from the nitrogen zone dehumidifier, the pH of the condensate would go below 5.0 for a fluorine concentration of 0.2 ppm or greater. This concentration would occur if the concentration of freon in the nitrogen zone were reduced by 1 ppm, assuming that the freon had decomposed to form fluorine and chlorine gas, and that these gases were absorbed in 10 gallons of condensate. The presence of these gases would not

necessarily indicate corrosion problems, however, since the dewpoint of the nitrogen is kept well below the cell temperatures. If the pH of the condensate increased above 9.5, it would indicate the possibility that very fine particles of NaOH were circulating in the system and collecting in the condensate. Therefore, if the pH falls outside the range of 5.0 to 9.5, test coupons in the cell will be removed and examined for evidence of corrosion.

Reference

(1) "Proposed Technical Safety Guide III, Reactor Containment Leakage Testing and Surveillance Requirements," USAEC, December 15, 1966.

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.

C. Batteries

Periodic checks of battery performance shall be made on all three battery stations (+125 V DC, ± 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 drawdown. Except for the diesel starting batteries, calibrate panel voltmeters against a known standard.

Bases

The monthly test of the main diesel generator is primarily to check for failures and deterioration in the system since last use. The testing will be conducted up to equilibrium operating conditions not only to demonstrate starting ability, but also to demonstrate ability to operate satisfactorily under loaded conditions. Load test will be made with generator subjected to a load of at least 540 kW. The semi-annual tests are more comprehensive in that this functionally tests the emergency power distribution system and the logic under which it operates.

The auxiliary diesel generator will also be tested monthly up to equilibrium operating conditions to demonstrate its operability as required by Section 3.6. Load tests will be made with the generator subjected to a load of at least 30 kW. The ability to start the diesel and pick up loads on the bus within one hour should also be demonstrated.

Station batteries will deteriorate with time, but precipitous failure is very uncommon. The type of surveillance called for in this specification is that recommended by the manufacturer, and which in the past has proven adequate in predicting failure of a cell before it becomes inoperable.

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 these 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.

Bases

Operability of the pipe snubbers is dependent primarily on the maintenance of an adequate supply and quality of oil in the snubbers. A radiation-resistant oil has been supplied with the pipe snubbers. Therefore, a six month surveillance interval is considered adequate. The manufacturer's recommended maintenance is an annual inspection of the clean exposed portion of the piston rod for scratches or other damage and a level check of the hydraulic fluid with addition of fluid only if required.

No mechanism for deterioration of snubber performance has been identified except for loss of oil supply or deterioration of the oil. All snubbers have been located so that the oil supply is visible. However, some snubbers are located where accessibility is very limited and would entail extensive time and difficulty due to background radiation. Deterioration of the oil is not expected since a radiation resistant oil is used. The viscosity of oil samples located in representative areas will be measured at six month intervals to verify this conclusion.

Specification A and B will assure that each snubber has an adequate supply of oil and that deterioration of oil has not taken place. Exercising representative pipe snubbers at yearly intervals is intended to assure that there is no problem of general deterioration of snubber performance. Snubbers will be replaced if required.

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.

Bases

To provide assurance that release rates established in Specification 3.8 are not significantly affecting the level of radioactivity in the environment, it is necessary to conduct the specified program. Environmental radioactivity changes are specified in 10 CFR 20 which may constitute concern and require a review of the radioactive effluent releases.

References

(1) SEFOR FDSAR, Volume 1, 9.4.4, p. 9-8 and Table IX-3.

(2) SEFOR FDSAR, Supplement 17, Answer to Question M-6, p. 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 main and auxiliary sodium flow for given pump conditions and the concurrent reactor sodium inlet and outlet temperatures, reflector segment positions, gross gamma activity in the reactor cover gas, 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, BRDO, 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, BRDO, 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 DRL.

B. Short-Term Unexplained Trends

1. Reactor operators shall make frequent observations of reactor operations with the objective of detecting unexplained deviations from expected conditions. For purposes of this specification, such deviations shall include unexplained changes in reactivity, primary coolant flow rate, gamma activity in the reactor cover gas, and upper reactor vessel outlet temperature. If reactor operating experience indicates that other parameters indicative of the status of the core and primary coolant system detect unexplained deviations from expected conditions which are not detected by the parameters mentioned above, those additional parameters shall be included in the group of parameters which are frequently observed. Upon observation of a short-term change that is not readily explainable and which in the operator's judgment has possible safety significance, reactor power shall be reduced to a level of no more than 50% of that at which the change was observed. The SEFOR 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 Facility 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 BRDO and an investigation has been conducted by members of the BRDO technical staff. The General Manager, BRDO, 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 DRL within one week. If operation is resumed, the conclusions of the Site Safety Committee

and the BRDO 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.

2. If cover gas activity increases are apparently associated with unexplained reactivity changes of measurable magnitude, the reactor shall be immediately shut down to investigate the cause. If it is confirmed that the reactor cover gas activity increase and reactivity change are related, reactor operations shall not be resumed until authorized by the General Manager of BRDO as discussed in 1. above.
3. If there is positive indication that cladding failure has occurred, the reactor shall be shut down for examination of all fuel rods which are accessible by means of through-head ports. The reactor may be started up if defective fuel rods, as defined in Specification 3.3.K, are located and removed. If, after examination of all fuel rods located under through-head ports, defective fuel rods are not located, the reactor may be started up if, in the judgment of the Facility Manager, it is safe to proceed.
4. In conjunction with fuel rod examinations for the purpose of locating failed fuel, at least two sodium samples shall be obtained from the reactor vessel. One sodium sample shall be kept for an archive sample, and another shall be examined for evidence of defective fuel.

Bases

The direct responsibility for safe operation of SEFOR on a day-to-day basis is vested in the Facility Manager and his staff. Although general review and evaluation of SEFOR operations and the performance of the SEFOR operating staff will be made by the Safety Review Committee, the BRDO General Manager and his staff, and the DRL, this observation is not continuous on a day-to-day basis, and, unless gross and serious errors and conditions exist, is largely in the nature of an advisory function. To have it otherwise leads to the danger of very inefficient operation and, more seriously, might tend to reduce safety by diluting the responsibility of the site manager and his staff. The present specification is intended to preserve this responsibility while at the same time, under special circumstances which might have serious safety implications, to assure that independent reviews are made by competent individuals who are not directly involved in the day-to-day operation of the reactor. In the case of anomalous behavior of the reactor, it is believed that evaluation by technically qualified individuals who are not involved with the day-to-day operation may provide additional perspective to the situation and may point up potential problems, which are not evident to those involved in the reactor operation.

In the case of anomalous behavior occurring over a long period of time, basic responsibility for determining whether continued operation is safe is vested in the SEFOR site manager and the Site Safety Committee with the independent reviews to be made by the Safety Review Committee and the BRDO General Manager and his staff. The potentially more serious situation is anomalous behavior that occurs over a short period of time. In this case, basic responsibility for return to power level at which the behavior was observed is vested in the BRDO General Manager, who has at his disposal the services of a staff of over 200 nuclear energy specialists, including the designers of the SEFOR reactor. He is able to marshall this technical capability on short notice and thus can render decisions, based on detailed technical evaluation by the most qualified individuals who will perform this work in a competent and expeditious manner. He will have the benefit of the judgment of the SEFOR site manager and his technical staff. Additional evaluation for the benefit of the General Manager will be forthcoming at a later date from the SEFOR Safety Review Committee, also composed of highly competent technical individuals who are distributed throughout various components of the General Electric Company other than BRDO.

Short-term unexplained variations in process variables called out in the specification are not in themselves an indication that an unsafe reactor condition exists; however, they may be symptomatic of potential safety problems of a serious nature. The site manager is charged with the responsibility of determining whether other conditions that occur should be defined as short-term (or long-term) unexplained behavior.

The actions of the SEFOR Site Safety Committee and the Safety Review Committee in the case of unexplained reactor behavior are consistent with their responsibilities as given in Section 6 of these technical specifications. Section 6 describes the composition of these committees.

Section 5

DESIGN FEATURES

5.1 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.

Bases

The minimum exclusion distance is of significance from the standpoint of potential radiological doses. It is the intention of this specification to record the minimum distance so that future changes to plant facilities will maintain the minimum exclusion distance.

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

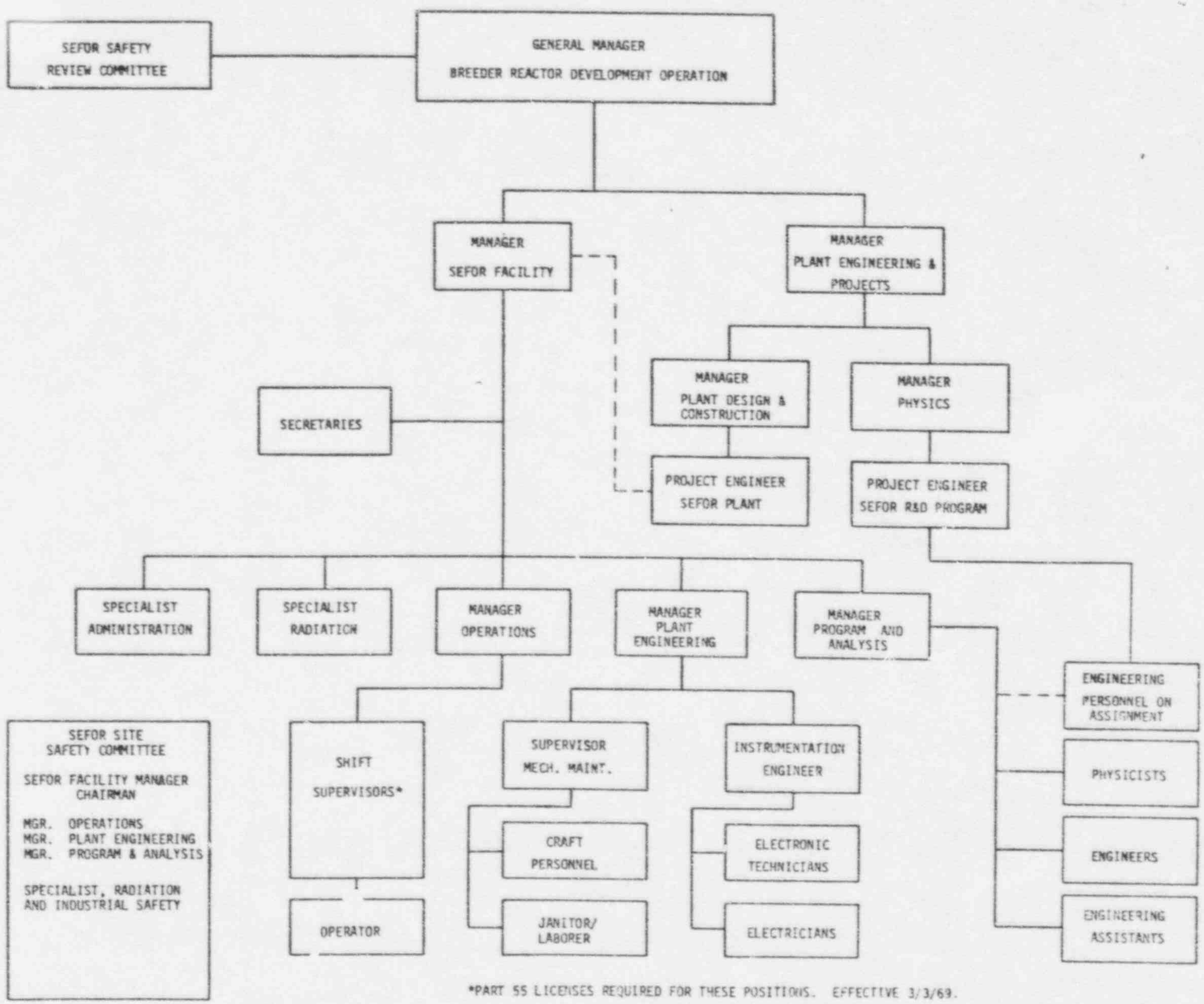
1. The organization for management, operation, and audit of SEFOR facility operations shall be as given in Figure 6-1. The overall full-time responsibility for operation of the SEFOR facility and compliance with these Technical Specifications shall reside in the SEFOR Facility Manager, who in turn is responsible directly to the BRDO General Manager. The BRDO 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 BRDO 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. At least one of the operators shall be licensed.

   b. When the reactor is secured, a licensed reactor operator and two additional persons trained in carrying out emergency procedures shall be at the site.

   c. A shift supervisor shall be in charge of startup, approach to power, normal operation, recovery from unplanned or unscheduled reductions in power, shutdown, handling or transferring of materials in the reactor vessel, and during any handling operations within the refueling cell when the reactor vessel head or ports have been removed.



*PART 55 LICENSES REQUIRED FOR THESE POSITIONS. EFFECTIVE 3/3/69.

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.

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. <u>Manager, Operations</u>

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.

Demonstrated 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. <u>Supervisor, Mechanical Maintenance</u>

High school education and apprenticeship training in metalworking 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. <u>Instrumentation Engineer</u>

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. <u>Shift Supervisor</u>

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 BRDO 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 be 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 BRDO 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.

2. SEFOR Safety Review Committee

a. Membership

(1) Chairman plus five members appointed by the BRDO General Manager.

(2) Manager, Nuclear Safety Appraisal Operation, Nuclear Energy Division, ex-officio member in addition to the appointees in (1) above.

(3) Technical consultants as deemed necessary by the BRDO General Manager.

(4) Members of the SEFOR Facility Operation (Figure 6-1) shall not serve as members of the Safety Review Committee.

(5) Qualifications of the Safety Review Committee with regard to the combined experience and technical specialties of the individual members shall be maintained at a high level. The Committee as a whole shall have competence in nuclear reactor technology encompassing the basic disciplines of chemistry, physics, engineering, and also nuclear safety and health physics. The minimum qualifications for membership of this Committee shall be determined by the BRDO General Manager. The membership of this Committee shall be selected by the BRDO General Manager with the advice and concurrence of the Manager, Nuclear Safety Appraisal Operation, Nuclear Energy Division. The performance of this committee shall be reviewed periodically by both the BRDO General Manager and the Manager, Nuclear Safety Appraisal Operation. Deterioration in its performance shall be avoided by replacement of individuals or a reconstitution of the committee, if necessary, at the discretion of the BRDO General Manager. Replacement shall be made from the manpower resources of the General Electric Company as a whole.

b. Meeting Frequency

Semi-annually and as required on call of the Chairman or the BRDO General Manager. During the first year of operation, the committee shall meet at least three times.

c. Quorum

Chairman or his delegated alternate plus three members.

d. Responsibilities

(1) Review proposed changes to the Operating License, including Technical Specifications.

(2) Review matters including proposed changes or modifications to plant systems or equipment as referred to it by the SEFOR Site Safety Committee or by the Facility Manager and review decisions with regard to unreviewed safety questions.

(3) Audit proposed changes to procedures, tests, experiments, and review those changes which may have safety significance.

(4) Review reports, minutes, and results of audit inspections and select for review specific items which may have safety implications.

(5) Conduct, or have conducted by a qualified group, audits of plant operation at least semi-annually to assure that emergency procedures are up to date and to assure compliance with Operating Procedures and Technical Specifications.

(6) Review reported instances of violations of Technical Specifications and recommend to the BRDO General Manager appropriate action to prevent recurrence of the violations.

(7) Review instances of significant equipment malfunctions which are of an unusual or unexpected type, and perform other special reviews and investigations as requested by the BRDO General Manager.

e. Authority

The function of the SEFOR Safety Review Committee shall be advisory, and it shall report directly to the BRDO General Manager.

f. Records

A report on each meeting of the SEFOR Safety Review Committee shall be prepared and forwarded to the BRDO General Manager and to such others as the Chairman or the BRDO General Manager may designate. This report shall contain the findings and recommendations resulting from each meeting. The BRDO General Manager shall have documented, within a reasonable period following the receipt of the report, the actions taken on the recommendations. Copies of this response shall be forwarded to the Chairman and members of the SEFOR Safety Review Committee.

g. Procedures

Procedures for committee operation shall be established by the Chairman of the committee and the BRDO General Manager. The BRDO General Manager shall review periodically the performance of the committee to assure an independent and comprehensive review of facility operation.

6.2 Plant Operating and Emergency Procedures

A. Detailed written procedures and instructions with check-off lists, approved by the SEFOR Facility Manager, shall be provided and used for the following conditions:

1. Refueling and refueling cell operations, normal startup, operation, and shutdown of the complete facility and of all systems and components involving nuclear and radiological safety of the facility.
2. All program tests and experiments.
3. Emergency conditions involving possible or actual releases of radioactive materials.
4. Preventive or corrective operations that could have an effect on reactor safety or affect nuclear or radiological safety.

B. Temporary operating procedures which do not decrease plant safety margins, may be used after review by the Manager, Program and Analysis, and approval by the Manager, Operations. Use of these procedures shall be documented in the operating records.

C. Modifications to experimental test procedures, which do not change the original intent or add to the scope of the test, and which do not reduce safety margins may be made with the approval of the Manager, Program and Analysis. Such changes shall be documented in the operating records.

D. Provisional test procedures which are complementary to experimental test procedures or that are of a diagnostic nature, and do not reduce plant safety margins, may be used after review by the Manager, Operations and approval by the Manager, Program and Analysis. Such procedures shall be documented in the operating records.

E. Radiation control procedures shall be established and all station personnel shall be instructed in these procedures. These procedures shall show the permissible radiation exposure levels and methods for control of radiation exposure.

The radiation protection program shall be organized to meet the requirements of 10 CFR 20 with the following exception:(1)

Paragraph 20.203 - caution signs, labels and signals. In lieu of the "control device" or the "conspicuous visible or audible alarm signal" specified in paragraph 20.203 (c) (2), each High Radiation Area in which the intensity of radiation is 100 mREM/hr or more shall be barricaded and physically posted as a High Radiation Area. Entrance thereto shall be controlled by requiring a special work permit and any individual or gruop of individuals permitted to enter such an area shall be provided with a radiation monitoring device which continuously indicates the radiation dose in the area.

In addition to the above requirement, any high radiation area in which the intensity of radiation is greater than 1000 mREM/hr, shall be provided with locked doors to prevent entry into such areas by unauthorized personnel. The keys to these doors shall be kept under the administrative control of the shift supervisor and the Manager, Operations.

F. Practice drills on Emergency Procedures shall be conducted prior to initial criticality and at least annually thereafter. Such drills shall include as a minimum, partial or complete evacuation of the site and a test of the adequacy of communications with off-site support groups.

Reference

(1) Letter to Dr. Peter A. Morris, U.S.A.E.C., Division of Reactor Licensing from Karl Cohen, BRDO, General Electric Company dated February 28, 1969.

6.3 Action to Be Taken in the Event of an Abnormal Occurrence

A. Limiting Safety System Setting

1. If a limiting safety system setting is violated without causing a reactor scram or other safety system action, the reactor shall be shut down immediately, and the occurrence shall be reported promptly to the Manager, SEFOR Facility.
2. A thorough investigation of the conditions related to the occurrence shall be made by the Site Safety Committee. The Committee shall prepare a report for each such occurrence, including an evaluation of the cause of the occurrence and recommendations for appropriate actions to prevent or reduce the probability of recurrence. Copies of this report shall be forwarded to the BRDO General Manager.
3. The DRL shall be promptly notified of the occurrence.
4. Reactor operations may be resumed after corrective action has been taken to prevent recurrence of the situation, provided such operation is recommended by the Site Safety Committee.

B. Limiting Conditions for Operation

1. If during operation, the limiting conditions for operation specified in Section 3 of these Technical Specifications are not met, the reactor shall be shut down.
2. The Site Safety Committee shall review the occurrence and ascertain that corrective measures are taken to prevent or minimize the probability of recurrence of the event. The committee shall prepare a report documenting its evaluation of the event and its recommendations and forward the report to the BRDO General Manager.
3. If a violation of the limiting conditions for operation causes a reactor shutdown, the DRL shall be notified by means of the quarterly operations report. This report shall include the circumstances of the abnormal occurrence and the remedial actions taken.
4. Reactor operations may be resumed when remedial actions have been taken and the limiting conditions for operation are met.

C. Component Failures and Deterioration of Fission Product Barriers

1. If component failures should occur which threaten the operability of an engineered safety system or if abnormal deterioration should be discovered which could threaten the integrity of one of the barriers to the release of fission products, the reactor shall be shut down.
2. The Site Safety Committee shall review the occurrence and ascertain that corrective measures are taken before reactor operations are resumed. The Committee shall prepare a report documenting the occurrence and the basis for any action it recommends and forward the report to the BRDO General Manager.
3. The DRL shall be notified of the occurrence as required by the license and by means of the Quarterly Operations Report. This report shall identify the circumstances and the remedial action taken.

D. Administrative and Procedural Controls

1. Should inadequacies in administrative or procedural controls be disclosed during the course of facility operations which create or could lead to an unsafe condition, immediate action shall be taken to put the affected portion of the plant in the safest possible condition.
2. The Site Safety Committee shall review the deficiencies and assure that corrective measures are taken before normal operations are resumed. The Committee shall prepare a report of the circumstances and action taken and forward the report to the BRDO General Manager.
3. The DRL shall be notified by means of the Quarterly Operations Report.

6.4 Action to Be Taken if a Safety Limit is Exceeded

If a safety limit is exceeded, the reactor shall be shut down immediately. Reactor operations shall not be resumed until authorized by the DRL. An immediate report shall be made to the General Manager of Breeder Reactor Development Operation and to the DRL. A complete analysis of the circumstances leading up to and resulting from the situation, together with a recommendation to prevent a recurrence shall be prepared by the SEFOR Site Safety Committee. This report shall be submitted to the Manager of the Breeder Reactor Development Operation and the Chairman, SEFOR Safety Review Committee for review and action. Appropriate analyses and reports, verbal and written, shall be submitted to the DRL.

6.5 Plant Operating Records

A. Records or logs relative to the following items shall be retained for the life of the plant:

1. Records of normal plant operation, including power level, periods of operation at each level and cumulative damage to primary and secondary coolant system components.
2. Records of principal maintenance activities, including inspection, repair, substitution or replacement of principal items of equipment pertaining to nuclear safety.
3. Records of abnormalities, including failures, emergency shutdowns and trips of equipment pertaining to nuclear safety.
4. Records of periodic checks, inspection and/or calibrations performed to verify that surveillance requirements (see Section 4) are being met.
5. Records of changes made to the plant as described in the Facility Description and Safety Analysis Report and supplements thereto.
6. Records of fuel rod data and fuel inventory histories.
7. Records of radioactivity in liquid and gaseous waste released to the environment, including time and duration of release.
8. Records of plant radiation and contamination surveys.
9. Records of off-site environmental monitoring surveys.
10. Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant who enter radioactive material areas.
11. Records of the experimental program and all tests performed with the facility. These shall be in the form of data recorded during the experimental procedure, applicable logs, and data gathered from the data acquisition system in the form of raw data on magnetic tapes and/or analyzed data resulting from the analysis of the raw data. Data from the experimental program may also be retained in the form of punched paper tape for certain specific cases.

6.6 Plant Operation Reports

Applicability

Applies to the preparation of a Quarterly Plant Operation Report and reports of experimental results.

Objective

To provide a timely summary of all aspects of facility operation and projected reactor experiments.

Specifications

A. A Quarterly Plant Operation Report shall be prepared and submitted to the Director, Division of Reactor Licensing. The report shall generally contain the following sections in the order listed:

1. Summary of plant operations.
2. Listing and classification of all plant shutdowns.
3. List of all "abnormal occurrences".
4. A detailed summary of changes in cover gas activity and any investigations involving suspected failed fuel.
5. Tabulation of all major items of plant maintenance.
6. Tabulation of major items of instrumentation and control work.
7. Summary of required surveillance tests and results.
8. Summary of health and safety operations including a summary of all radioactive releases and shipments.
9. Discussion and justification of significant changes in plant design.
10. Tabulation of significant changes in plant operating procedures related to transient tests.
11. Current schedule for planned transient experiments.

B. The following special reports shall be submitted in a timely manner.

1. A report summarizing the results of testing which demonstrates the natural circulation capability of the reactor coolant system.
2. A report summarizing the results from testing up to and including 10 MWt which demonstrates that the conditions specified in Section 3.3.E are being met.

3. A report(s) based on test data and operating experience up to and including 10 MWt which:
   a. Proposes quantitative definitions of suitable limits, and bases thereof, on unexplained behavior of reactivity, cover gas activity, and other parameters as discussed in Specification 4.9.B.1.
   b. Proposes the necessary detection steps and specific criteria, and the bases thereof, for judging the acceptability of continued steady state operation in the presence of a known loss of fuel clad integrity.
4. A report summarizing the results obtained from out-of-reactor and in-reactor testing of the FRED device.
5. A report summarizing all the data obtained from the experimental program up to and including results from the sub-prompt critical tests, as specified in Section 3.12.