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FEB 12 1971

Docket No. 50-231

Change No. 2
License No. DR-15

General Electric Company
ATTN: Dr. Karl Cohen, General Manager
Breeder Reactor Development Operation
310 DeGuigne Drive
Sunnyvale, California 94086

Gentlemen:

In letters dated July 8 and September 3, 1970, you applied for Change No. 3 and Change No. 4, respectively, to the Technical Specifications appended to Provisional Operating License No. DR-15 for the Southwest Experimental Fast Oxide Reactor (SEFOR). Proposed Change No. 3, as amended, would authorize the use of guinea pig rods in the inner positions of the core during steady-state operation up to 20 MWt and during excursion testing, including the maximum planned transient. Proposed Change No. 4 was submitted in response to the requirements of Technical Specifications 3.10.E and 6.6.B.3 which require, prior to operation above 17.5 MWt and excursion testing, proposal of additional specifications that: (a) provide specific criteria for judging the acceptability of continued operation with loss of fuel clad integrity, (b) address necessary steps for the detection of loss of fuel clad integrity, and (c) provide quantitative definitions of limits of unexplained behavior of pertinent reactor parameters. Amendments to the proposed changes were contained in your letters dated September 8, December 11, 22 and 31, 1970; January 28 and February 1, 1971. In addition, on December 2, 1970, you submitted a report on the determination of the SEFOR reactivity coefficients. We have combined the proposed changes and designated them as Change No. 2 to the Technical Specifications.

In our review, we have evaluated: the effect of slightly higher power densities in guinea pig rods on margins of safety to the damage threshold; the effect of initiating the maximum planned transient in the event the core contains a fuel rod with perforated cladding; the capability of detecting fuel clad perforations and anomalies in pertinent reactor parameters; and the increased surveillance that will be performed.

DV

A/60

We conclude that operation in the manner proposed does not present significant hazards considerations not described or implicit in the safety analysis report and that there is reasonable assurance that the health and safety of the public will not be endangered.

Accordingly, the attached replacement pages for the Technical Specifications, which incorporate the proposed changes, are hereby approved. The changes are indicated by brackets.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosures:
Revised pages for
Technical Specifications

Dispatched 2/16/71

OFFICE ▶	DRL	DRL	DRL	DRL	DRL	DRL
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DATE ▶	2/12/71	2/12/71	2/12/71	2/12/71	2/12/71	2/12

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

February 12, 1971

Docket No. 50-231

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Breeder Reactor Development Operation
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Peter A. Morris, Director
Division of Reactor Licensing

Enclosures:
Revised pages for
Technical Specifications

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 determined as follows, except that this limit does not apply during an excursion test initiated by the FRED:

1. If guinea pig rods are not present under the innermost refueling ports, the limit shall be 110% of rated flux.

2. If guinea pig rods are present under the innermost refueling ports, the limit shall be the lower of L_1 or L_2, where:

   L_1 = 1.05 (110N/600X) % of rated flux.

   L_2 = 110% of rated flux.

   N = total number of fuel rods in the core.

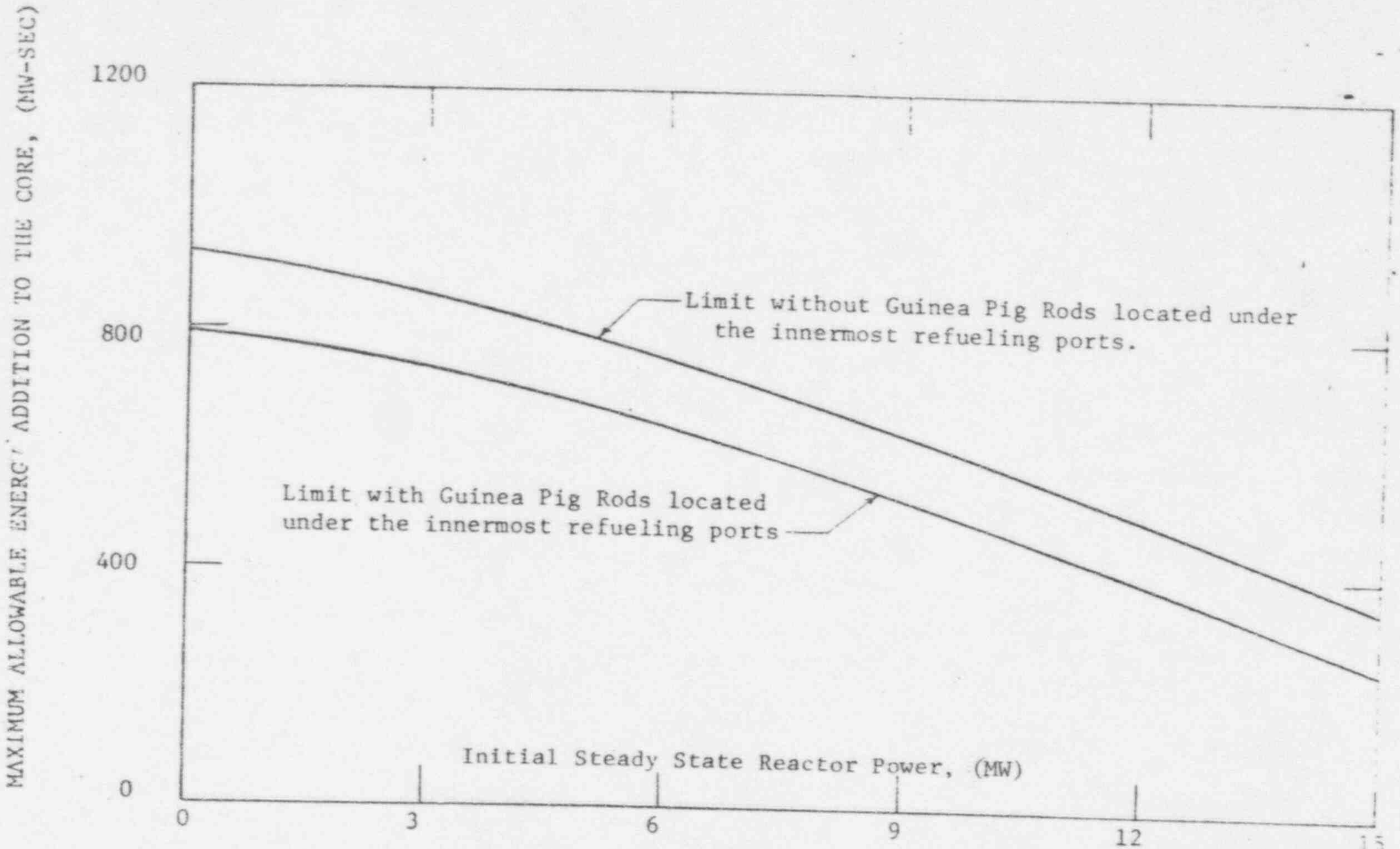
   X = ratio of guinea pig rod power to the power of a standard rod located nearest to the center of the core, as given in Reference 11.

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 $1050^{\circ}F$.

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 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 the safety limit given in paragraph 2.1.A.1 results in a maximum linear power density of 24 kW/ft in the standard hot fuel rod. Operation at the safety limit given by paragraph 2.1.A.2 results in a maximum linear power density of 25.2 kW/ft in a guinea pig rod located under the innermost refueling port. The values for X given in Reference 11 for calculating the limit of paragraph 2.1.A.2 will remain constant for the range of fuel rod loadings allowed by the Technical Specifications because the overall neutron flux profile will be constant. All 648 rod positions in the fuel channels (the central channel is a drywell for test devices) contain either fuel rods, B_4C rods, or stainless steel rods. The radial flux profile is constant with changes in core loading between the 600 and 648 fuel rods allowed by the Technical Specifications because loading changes are accomplished by interchanging fuel rods and B_4C or stainless steel rods rather than changing the size of the core. Local neutron flux perturbations due to the presence of a B_4C poison rod which could affect the value of X are small because the number of poison rods is limited by reactivity requirements and because the poison rods are distributed throughout the core, as required by 3.3.N, to minimize perturbations in the overall neutron flux profile. Burn-up effects on the flux profile are

also negligible for the planned SEFOR program. They are in the conservative direction (reducing guinea pig rod power density) because the flux profile flattens as plutonium is preferentially burned out at the core center. Changes in the power profile at the guinea pig rod locations with variations in reflector position are also negligible because allowable reflector patterns are restricted by the excess reactivity limit given by 3.3.B. At 20 MWt, an unbalanced pattern of nine reflector segments fully raised and one segment partially raised would have no significant effect on the peak power density of the guinea pig rods.

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 greater 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 $\sim$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 limits given in paragraphs 2.1.A.1 or 2.1.A.2, the resulting increase in fuel temperature in the standard peak fuel rod and guinea pig rods would correspond to only a 0.7 kW/ft or 0.8 kW/ft increase in the linear power density of these rods, respectively.

Additional margin at the safety limit is provided by the fact that the maximum linear power density discussed above is based on the total power 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 of the hottest standard rod at 22 MWt from 24 kW/ft to 22.6 kW/ft. For a guinea pig rod, the actual kW/ft at the safety limit defined in paragraph 2.1.A.2 is reduced from 25.2 to 23.7 kW/ft. In addition, the nominal loading of SEFOR is predicted to be approximately 636 fuel rods which will further reduce the peak kW/ft generated in the fuel at the safety limit defined in paragraph 2.1.A.1 to 21.2 kW/ft. The loading will not affect the peak kW/ft for the guinea pig rods at the safety limit. The 23.7 kW/ft generated at the safety limit in a guinea pig rod located under the innermost refueling ports corresponds to a linear power density 5% greater than that of a standard rod at minimum core loading and 110% of rated flux. This increase is justified by increased surveillance of the guinea pig rods as specified in paragraph 4.3.E. Based on these data, and the limited number of guinea pig rods that can be loaded under the innermost refueling ports, it is concluded that sufficient margin exists between operation at the safety limits given in paragraphs 2.1.A.1 and 2.1.A.2 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

(T25,000 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 two curves shown on Figure 2.1-1 define the total energy which can be deposited in the core during a planned transient with and without guinea pig rods present under the innermost refueling ports 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 transients tests[3,-] on sodium-cooled oxide fuel have demonstrated that repeated transients up to fuel energy densities of 250 cal/g will not damage the fuel. Initial fuel temperatures are based on the data and methods given in Reference 10. The difference between the two curves shown on Figure 2.1-1 is the result of differences in power peaking factor for the standard hot rod and the guinea pig rods which influence the initial fuel temperature and the energy deposited in a particular rod.

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 maximum heat flux of a fuel rod is limited by boiling, not by a burnout limit, because gross sodium boiling and associated expulsion of sodium from the core will occur before a burnout heat flux is reached. However, gross sodium boiling will not occur because of the large amount of subcooling in a channel (approximately 600°F in the hot channel at the safety limit of 1050°F when guinea pig rods are not present under the innermost ports and approximately 400°F at the safety limit of 1050°F when guinea pig rods are present under the innermost ports) and the good heat transfer characteristics of the sodium coolant. During 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 and guinea pig rods 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 and guinea pig rods 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 cycles[10] and 300 cycles for the hot standard and guinea pig rods, respectively, for strain cycles associated with the Maximum Planned Transients specified in Section 3.12.

These fatigue limits were 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 for standard rods and less than 0.05 for guinea pig rods. Additional fatigue damage due to unplanned power or temperature transients approaching

the safety limits of paragraphs 2.1.A and 2.1.D will not contribute significantly to this cumulative total. For example, the minimum predicted fatigue life for the cladding cycled from refueling conditions to the safety limits of 2.1.A and 2.1.D is 1000 cycles. Consequently, one cycle of this type (involving violation of two limiting safety system settings) would increase the cumulative usage factor less than 0.001.

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 11-2.

(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.

(11) SEFOR FDSAR, Supplement 10, p 1-113.

TABLE 2.2-1

SCRAM FUNCTION

FUNCTION		SAFETY SYSTEM SETTINGS
High Flux, Wide Range Monitor	≤	105% of Rated Flux, if guinea pig rods are not present under the innermost refueling ports If guinea pig rods are present under the innermost ports the setting shall be equal to or less than the lower of T_1 or T_2, where: $T_1 = \frac{110N}{600X}$ % of Rated Flux $T_2 = 105$ % of Rated Flux N = total number of fuel rods in the core X = ratio of guinea pig rod power of standard rod located nearest the center of the core. (Reference 6)
Low Level, Reactor Sodium	≤	4 inches below lip of operating level overflow pipe
High Temperature, Core Outlet	≤	900°F
Low Flow, Main Primary	≥	60% of the operating flow set point*
High Temperature Reflector Region**	≤	350°F for thermocouples on the reflector guide structure inner diameter and radial web.
	≤	275°F for thermocouples on the reflector guide structure, outer diameter.
	≤	450°F for thermocouples in the lower end of the reflector segments.

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

**At least ten thermocouples shall be connected in the safety system, including at least one in each reflector bay.

Bases

The limiting safety system setting (LSSS) of 105% of rated flux provides a 5% margin below the safety limit of 110% defined in paragraph 2.1.A.1. The limiting safety system setting of 110N/(600X) % or 105% of rated flux, whichever is less, provides a 5% margin below the safety limit of paragraph 2.1.A.2. This will assure protection of the safety limit for normal reactor operation. The LSSS for operation with guinea pig rods under the innermost refueling ports will limit the reactor power so that the maximum linear power density of a guinea pig fuel rod does not exceed the maximum power density for standard fuel rods at the safety limit of paragraph 2.1.A.1 with the minimum core loading. (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 two inches below the overflow nozzle, providing margin with respect to the LSSS of four 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 and segments are monitored by the safety system to provide this assurance. The guide structure temperatures at the 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

The ten reflector region thermocouples used by the safety system can be chosen from any of the applicable thermocouples listed in Table 2.2-1, since a different trip level can be set for each thermocouple. The choice of the ten thermocouples to be used for the safety system will be made so as to monitor temperatures in each of the ten reflector bays.

A safety system trip at 50% to 60% of the normal coolant flow rate also provides assurance that the temperature of the neutron monitors will remain below the manufacturer's certified operating temperature of 300°F.

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 11, p 7-20.

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

(6) SEFOR FDSAR, Supplement 10, p 1-113.

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 $350^{\circ}F$ 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 $350^{\circ}F$ shall demonstrate that the reactor would be subcritical at $300^{\circ}F$ 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 $700^{\circ}F$. 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° 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.

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.

N. When guinea pig rods are located under the innermost refueling ports, the B_4C poison rods in the core shall be distributed such that the number of poison rods in any quadrant of the core (determined by N-S, E-W centerlines) does not exceed the number of poison rods in any other quadrant by more than two. This specification shall not be applicable when the high flux trip level is set more than 10% below the LSSS.

of standard fuel rods at the center of the core during 20 MWt operation.

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 0.001c). The inadvertent addition of up to 60c 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.

(1) observe the continuous monitoring system and operating conditions to diagnose the cause of failure or maloperation 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) pla[illegible] 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.

Specification 3.3.N requires a reasonably uniform arrangement of B_4C poison rods in the core to provide assurance that the power density in guinea pig rods under the innermost refueling ports is not significantly greater than the value used to determine the LSSS. A uniform distribution of poison rods is desirable for most of the planned tests. However, some non-uniform arrangements at low or intermediate power may be required for special tests such as determination of material worths at zero power or determination of available excess reactivity during the approach to power. Such arrangements are permissible when the high flux trip level is reduced more than 10% below the LSSS, since the maximum effect of a single poison rod on guinea pig rod power density is only 1/4%.(13) The intent of limiting the applicability of this specification is not to permit grossly non-uniform arrangements of poison rods, but rather to permit the flexibility of arrangement which may be required during portions of the test program when the reactor is operated below the power levels at which special protection for guinea pig rods is required.

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.
(13) Addendum No. 2 to Proposed Change No. 3 to the Technical Specifications,

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

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

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 requirement that the sodium coolant systems shall be operating means that the sodium pumps shall be energized to provide forced circulation of the sodium. The fans and blowers for the sodium-to-air heat exchangers may be secured if the heat load is such that their operation is not required to hold the sodium temperature at the desired value. 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 than 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 operating 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]

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. The specifications previously identified in this section have been approved and are incorporated into the SEFOR Technical Specifications as Section 3.13.

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.

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 up to and including 10 MWt were used to establish allowable limits for unexplained reactor behavior. Section 3.13 was added to these specifications to specify these limits.

Reference

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

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 limits given in this specification also assure that the maximum energy addition to the core during a planned transient does not exceed that calculated for the Maximum Planned Transient.[10] This in turn assures that the maximum consequence of inadvertently running a transient with a defective (sodium-logged) fuel rod in the core would be limited to deformations corresponding to about 0.6% strain of the cladding of the defective fuel rod.[11] This amount of strain is only 4% of the minimum ductility of the SEFOR cladding at the end of the three-year experimental program.[12]

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 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 second.[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]

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) SEROR 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.

(10) SEFOR FDSAR, Supplement 10, page 1-48

(11) Additional Information Regarding Sodium Logging of SEFOR Fuel Rods, February 1, 1971

(12) SEFOR FDSAR, Supplement 21, page 4

3.13 Operating Limits

Applicability

Applies to parameters observed or measured during steady state reactor operation.

Objective

To establish limits on important parameters which will detect anomalous reactor behavior and to establish limits for steady state operation with known loss of clad integrity.

Specification

A. The limits for unexplained behavior shall be as given below. If these limits are exceeded, the actions specified in Section 4.9.B shall be taken.

   1. An unexplained increase of more than 500 mr/hr in the reading of the cover gas monitor shall be considered anomalous.

   2. A change in steady state reactivity of more than ± 10 cents from the predicted value at the reactor operating conditions shall be considered anomalous.

   3. Changes to the equation used to predict reactivity shall be reported to the DRL if such changes, excluding those due to core loading adjustments and burnup, result in a change of more than ± 20 cents from the equation previously reported to the DRL, when the equation is evaluated at a power level of 20 MWt with an average coolant temperature of 760°F.

4. The main primary coolant flow rate shall be considered anomalous if it differs by more than ±300 gpm from the calculated flow rate for flow rates above 1000 gpm, or if it deviates from an established flow rate by more than ±300 gpm at flow rates below 1000 gpm.

5. The auxiliary primary flow rate shall be considered anomalous if it differs by more than ±20 gpm from the calculated flow rate.

6. A difference of more than 60°F between the upper reactor vessel outlet temperature and the Resistance Temperature Detectors (RTD's) in the reactor vessel main primary outlet pipe shall be considered anomalous.

B. The reactor shall not be operated, except for diagnostic tests, if the cover gas monitor reading exceeds 9000 mr/hr.

Bases

The reactor and auxiliary systems (including the radwaste system) have been designed so that steady state operation with five failed fuel rods can be accommodated [1,2]. The cover gas monitor is capable of detecting loss of clad integrity if it occurs [3]. The cover gas monitor will also respond to fission products escaping from pin hole leaks, which may occur in some fuel rods but which are not classed as cladding failures or defective fuel. (4)

Analyses of the reactor cover gas have shown that the normal background activity is due to 38 sec Ne^{23} and 1.8 hr A^{41}, with Ne^{23} being the major contributor. Due to the short half-life of Ne^{23}, the cover gas monitor may show a temporary increase on the order of 100 mr/hr due to cover gas system transients, such as cover gas pressure control valve operation. Such increases would not be considered anomalous, since the Ne^{23} comes from the sodium, rather than the fuel.

The normal background activity due to A^{41} and Ne^{23} may result in a cover gas monitor reading of as much as 100 mr/hr at 20 MWt, depending on operating conditions, such as sodium temperature and sampling flow rate.

The activity in the cover gas may also increase due to pin hole leaks in some fuel rods or as the result of cladding failures. The initial increase in activity from any cause will be treated as an anomaly if it exceeds the specified limit, and appropriate actions will be taken. When reactor operation is subsequently resumed, as provided for in Section 4.9 and other relevant sections of the Technical Specifications, the new level of activity will be considered to be the normal background for purposes of identifying subsequent anomalous conditions.

Changes in cover gas activity are reported as required by paragraph 6.6.A.4 of the Technical Specifications.

If the fission gases were released from an average fuel rod due to cladding failure following operation at 20 MWt for three days, the cover gas monitor reading would increase by 1610 mr/hr due to the fission gases [5]. The specified anomaly limit is approximately one-third of this value, which provides additional safety margin. The fission gases actually present in any fuel rod will depend on its operating history, but the value of 500 mr/hr represents a reasonable lower limit for detection of a significant release of fission gases. This limit also assures the capability of detecting a change in cover gas activity due to loss of cladding integrity which exposes fuel to the sodium.

The SEFOR reactor operates over a wide range of temperature and power conditions in the course of the defined experimental program. To follow normal experimental reactor conditions requires a predictive capability for a broad range of conditions. To establish definitive criteria for anomalous reactivity, careful reactivity balances and comparisons have been maintained during zero power testing, fuel arrangement and the power ascension to 10 MWt. Inconsequential random and systematic errors normally are less than the ±10 cents maximum disparity between predicted and measured reactivity values.[6] The limit is restrictive enough to alert the operator and staff to items of consequence, e.g., significant changes in the reactor coefficients, erroneous fuel arrangements or other problems.

The equation used to predict the steady state reactivity value at given reactor operating conditions was given in Reference 6. Changes may be made to this equation from time to time, based on changes in core loading, fuel restructuring, or other changes which are properly identified. All changes will be documented and will be reviewed by the Site Safety Committee. The specified reporting requirement allows the necessary flexibility for making appropriate changes and also provides assurance that the DRL will be informed of significant or unforeseen changes in core reactivity.

Comparison of the coolant flow rate to the pump characteristic provides a cross comparison between the pump performance and the magnetic flowmeter. The plant is protected from abrupt and large losses of flow by the low flow trips at 80% of the set point flow rate for the main coolant system. The requirement for comparison of measured flow to predicted flow will alert the operator to any deterioration in performance of either the pump or the magnetic flowmeter. The specified limits are large enough to exclude variations due to random errors and repeatability considerations, but small enough to detect incipient problems before they have a detrimental effect on reactor cooling.[6]

Because the outlet temperatures vary so widely over the course of the experiments, it is necessary to have criteria which will be applicable to all conditions. The comparison between the vessel exit RTD's and the upper reactor vessel outlet temperatures provides a cross-check on both instruments. The allowable variation was obtained by examining the difference between these two temperature devices over the testing completed up to 10 MWt, including the natural circulation tests. The value of ±60°F is slightly larger than any variation obtained to date. Calculations indicate that for full flow, full power, there should be 12°F difference for the as-designed core, although larger differences may occur due to local temperature variations.[6] A difference in these temperatures of 60°F would correspond to approximately 35% of the total vessel flow bypassing the core compared to the design condition of 10% bypass leakage. It could also be caused by changes in flow distribution due to orificing effects at low flow rates. These conditions are not detrimental to the core.[7] The resulting upper region temperatures would be below the trip limit of 900°F for these sensors. The 60°F limit would assure detection of such a condition before the situation became serious.

The criterion for continued reactor operation with possible or known loss of cladding integrity is based on the ability of the cover gas monitor to detect additional changes in cover gas activity. The cover gas monitor has a full scale capability of 10,000 mr/hr. If the reading were to exceed 9000 mr/hr, detection of changes on the order of 500 mr/hr could not be assured.

A full scale reading of 10,000 mr/hr could be caused by various levels of cover gas activity, depending on isotopic composition. For example, if cladding defects in five average rods permitted the release of fission gases with no additional hold up time, the cover gas monitor would reach its full scale reading after three days' operation at 20 MWt [5].

Reactor operation for the purpose of performing diagnostic tests is permitted when the cover gas monitor reading exceeds the specified limit, so that cover gas samples and other data can be obtained to aid in the diagnosis of the problem.

Data obtained from the isotopic analysis of cover gas samples and the results of fuel rod examinations, required by other sections of the Technical Specifications, will be used to determine whether or not there is evidence that the core contains defective fuel rods. Excursion tests are prohibited if there is such evidence, [8] but steady state operation is permissible. [9]

References

1. SEFOR Technical Specifications, p 3.7-4, bases for 3.7.H

2. SEFOR FDSAR, Supplement 21, Section I.

3. SEFOR FDSAR, Supplement 21, Section II.

4. SEFOR Technical Specifications, pp 3.3-5, 6, bases for 3.3.K

5. Table II: Rev. 2 of Proposed Change No. 4 to the SEFOR Technical Specifications, December 22, 1970.

6. Rev. 1 of Proposed Change No. 4 to the SEFOR Technical Specifications, December 11, 1970.

7. SEFOR FDSAR, Supplement 18, pp 9-11.

8. SEFOR Technical Specifications, paragraph 3.12.B.10.

9. SEFOR Technical Specifications, Section 4.9.B

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, 17.5, and 20 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 on one guinea pig fuel rod and one standard fuel rod shall be examined by the methods described in "A" above. If guinea pig rods are located under inner refueling ports, one of them should be the one examined.

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. If guinea pig rods are located under inner refueling ports, one of them should be the one examined.

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.

E. Following the examination after operation of 20 MWt as specified in 4.3.A, two or more (if available) 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 examined by the methods described in "A" above at intervals such that the rod exposure during the first interval does not exceed a core integrated power

of 300 MWt-days with the reactor operating at power levels greater than 17.5 MWt, and during successive intervals, does not exceed a core integrated power of 600 MWt-days with the reactor operating at power levels greater than 17.5 MWt. The maximum time interval between examinations of guinea pig fuel rods located under the innermost ports shall not exceed six months. Extended outages of more than one month shall not be included in the determination of this surveillance interval.

Bases

The bases for fuel rod examinations specified in this section are given in Section 2.1, "Safety Limits", Section 3.10, "Approach to Power", and Section 3.12, "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.

B. Short-Term Unexplained Trends

1. Verification that reactor parameters meet the requirements of paragraph 3.13.A shall be obtained at each planned stable condition of reactor operation and at least once per shift when stable operating conditions are maintained. If such verification is not obtained, reactor power shall be reduced to a level of no more than 50% of that at which the failure to meet limits 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

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.

6. A report which identifies changes made to the reactivity equation as required by specification 3.13.A.3 This report shall describe the reasons for the changes and the bases upon which continued reactor operation is deemed to be safe.