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SUPPLEMENTAL

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR

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1.0 INTRODUCTION

By application dated July 25, 1967, the General Electric Company (GE), on its own behalf and acting for Southwest Atomic Energy Associates (SAEA), requested a Provisional Operating License to authorize operation of The Southwest Experimental Fast Oxide Reactor (SEFOR) at power levels up to 20 megawatts thermal (MWt).

By application dated September 27, 1968, GE requested that a license be issued to authorize operation of the reactor at power levels up to 1 MWt, pending completion of the Commission's review of the proposed 20 MWt operations. Following review of the application and amendments, through amendment 22, and receipt of a favorable report on 1 MWt operations from the Advisory Committee on Reactor Safeguards (ACRS), the Commission published a notice of proposed issuance of a provisional operating license authorizing such operations in the Federal Register on November 15, 1968. Following successful completion of preoperational testing of the facility, Provisional Operating License DR-15, authorizing GE to operate SEFOR at power levels up to 1 MWt, was issued on March 4, 1969.

We have reviewed the supplemental information provided in amendments 23 through 28 to the application, filed subsequent to our review of 1 MWt operations, and have concluded that operations of the facility at steady-state power levels up to 20 MWt and in the pulsed mode can be carried out as proposed without undue risk to the health and safety of the public. The ACRS considered this supplementary information at its March 1969 meeting and made a similar finding as stated in its letter dated March 13, 1969. A copy of the ACRS report to the Commission is included as Appendix A.

Section 2 of this Supplemental Safety Evaluation contains a summary of the principal site and design related aspects of the SEFOR facility which were covered in detail in our previous Safety Evaluation, dated November 16, 1968. Section 3 covers those aspects which were not fully resolved when 1 MWt operations were considered and supplements our previous Safety Evaluation.

The license amendment permitting such operations will not be issued until the Commission has inspected the facility and has determined that (1) it has been constructed in accordance with the application and the construction permit, (2) that the modifications discussed in Section 3 of this evaluation have been satisfactorily carried out and (3) plant emergency procedures for 20 MWt operations are complete.

2.0 BACKGROUND SUMMARY

This section presents a summary of site and design related matters which were discussed in detail in the safety evaluation published by the Division of Reactor Licensing on November 16, 1968 in connection with issuance of a 1 MWt provisional operating license for SEFOR. The conclusions reached on these matters have been confirmed by subsequent technical review.

2.1 Site

The SEFOR facility consists of the 20 MWt nuclear reactor, the reactor building, the operations and auxiliary buildings, and the waste storage vaults. It is located 19 miles south-southwest of Fayeteville in Washington County, Arkansas. The facility has a minimum exclusion distance of 0.4 miles, and the average population density within a 15-mile radius of the plant is about ten people per square mile. The site presents no unusual geological or hydrological problems. Special emphasis was given to the seismic and tornado design of the facility. We and our consultants reviewed the seismic design and analysis, based on a maximum horizontal ground acceleration of 0.30g, and concluded that the SEFOR facility would be able to withstand the combination of design loads and maximum earthquake forces. The facility was designed to withstand the effects of a direct strike by a tornado with a maximum wind velocity of 300 mph and associated missiles. An adequately protected 50 kw emergency diesel generator is installed at the facility to provide power to the primary auxiliary coolant pump and related auxiliaries in the event of a tornado. Provision of this diesel generator assures that if all other power is lost and the sodium-air heat exchangers are damaged, sufficient heat can be removed from the core (dissipated to the inner containment) to keep the sodium in the core from boiling and the fuel from being seriously damaged.

2.2 Containment

The reactor is doubly contained. The outer cylindrical steel containment vessel entirely encloses an inner, reinforced concrete, steel-lined containment. The inner containment (which is divided into cells housing the reactor, the entire primary coolant system, and associated equipment) contains an oxygen-depleted atmosphere during reactor operations and is adequately designed to contain the energy releases and missiles which might be associated with potential accidents. In particular, the series of concentric cylindrical steel annuli surrounding the reactor vessel in the core region and the restraints for the reactor vessel head and bottom shield plug are capable of absorbing a blast loading greater than that imposed by any postulated reassembly accident.

The inner and outer containment barriers are capable of withstanding all internally generated missiles and resisting the temperatures and pressures that might result from the remote event of a nuclear excursion that destroys the reactor core. The outer containment houses reactor support equipment and is designed to withstand the pressures and temperatures which could result from an accidental sodium fire during periods of reactor shutdown when the inner containment is permitted to contain air.

2.3 Reactor Design

The reactor is designed either to be operated at a steady-state power level of 20 MWt or to be subjected power excursions produced by rapid ejection of a neutron absorbing slug through use of specially designed Fast Reactivity Excursion Device (FRED). The principal objective of the experimental program is to demonstrate the effectiveness of Doppler reactivity feedback in limiting power excursions. The core was deliberately designed to minimize the effects of reactivity feedback from other sources, such as fuel expansion.

The core nuclear design analyses have been substantiated by mockup experiments in the ZPR-3 facility, operated by Argonne National Laboratory. The Doppler coefficient of reactivity ($T\frac{dk}{dT}$) of -0.0085 was determined from measurements involving small heated fuel samples in ZPR-3. Although there is some uncertainty in this value, it is negative and of such magnitude that there is reasonable assurance that the SEFOR coefficient will be more strongly negative than the specified limiting value for operations (-0.005). A Doppler coefficient of only -0.003 would still prevent fuel center melting during planned transients. Based on ZPR-3 data, a maximum of 30¢ in positive reactivity could be inserted through partial voiding of coolant in the core. This amount of reactivity would not cause core damage if it were added during steady-state operation. If this reactivity were to be added during a power excursion experiment, the initial transient would be essentially terminated before the effects of voiding could take place, and the additional energy produced would not be significant.

The reactor core is approximately 36 inches high and 34 inches in diamater and is fueled with stainless steel-clad fuel rods containing mixed PuO_2-UO_2 fuel pellets. The core is assembled into 109 close-packed hexagonal channels. Each channel, is typically loaded with six fuel rods, a central beryllium oxide (BeO) rod, and a tightening sleeve. The weight of the fuel rods plus frictional forces, derived from contact with the segments of the tightener sleeve are adequate to restrain the fuel rods from vertical motion due to hydraulic forces within the channels. Reactivity effects of fuel bowing were

examined and because of these restraints are expected to be of no importance in transients. Channels are restrained against radial movement by being clamped around the periphery at the top and bottom of the core by a system of torsion bar springs.

Based on our review of the materials and material properties, the fuel manufacturing process and GE's program on fuel rod quality control, and the low maximum fuel burnup of 1500 MWd/T, we conclude that the fuel element design is adequate to meet the conditions of the planned SEFOR program.

2.4 Reactor Control

Reactor control is accomplished by vertical movement of ten reflector segments which are external to and surround the core section of the reactor vessel. Eight of the ten segments are used as coarse control rods, being either fully inserted or fully withdrawn. The remaining two segments are used for fine control with one or both assuming intermediate positions during reactor operation. The control segments are driven hydraulically. A fast reactor shutdown (scram) is accomplished upon signal from the reactor safety system by venting the control rod drive pistons and allowing the reflector segments to drop away from the core by gravity, assisted by hydraulic pressure. Following modifications to the support of the guide rails and hydraulic circuit, and readjustment of the hydraulic and pneumatic system pressures, we concluded that the system can overcome the most adverse combinations of pressures and frictional forces and provide an adequate and reliable reflector control system. We also concluded that even in the remote event of a gross sodium leak in the vicinity of a reactor vessel nozzle, a sufficient number of control segments will remain capable of being withdrawn to shut the reactor down.

2.5 Reactor Coolant System

SEFOR has one main sodium coolant system which is designed to remove 20 MWt from the reactor core. The main loop has also been designed to provide natural circulation for decay heat removal. This capability will be demonstrated early in the facility test program with the use of nuclear heating. It is backed up by an auxiliary sodium coolant system which is designed for 1 MWt decay heat removal. All components in SEFOR coolant systems in contact with sodium vapor are fabricated of type 304 stainless steel. The sodium pumps, reactor shielding, and reflectors, are cooled by a nitrogen coolant system designed to remove approximately 1.5 MWt. All three systems ultimately reject the heat they absorb to the atmosphere via air-cooled heat exchangers.

Based on the codes used in design, fabrication and testing of the reactor coolant system, as well as on the load combinations and stress limits considered, we and our consultants concluded that the mechanical design of the system is satisfactory.

To provide a means to alert operators to the need for corrective action to assure that functional capability of reactor cooling systems will not be severely degraded, a variety of leak detection methods are used. The leak detection monitors consist of 10 sodium level indicators, 6 smoke-type detectors, 2 sodium-24 activity monitors and 97 electric-contact-type detectors. We conclude that the functional capability and distribution of these leak detectors provide sufficient diversity and are adequate to minimize the consequences of sodium leaks by initiating scrams or alarms.

3.0 SUPPLEMENTARY EVALUATION

This section presents an evaluation of those aspects of 20 MWt plant operations and experimental uses of the facility which were still under review at the time the evaluation of 1 MWt operations was completed. The discussion herein supplements the Safety Evaluation dated November 16, 1968.

3.1 Experimental Program

The principal feature of the SEFOR experimental program is the conduct of a planned series of power excursion tests. The purpose of these tests is to demonstrate the effectiveness of Doppler reactivity feedback in limiting the energy release during power excursions in a plutonia-urania fueled fast reactor. All other tests will be of types that have been routinely carried out at other reactor facilities.

The power excursion tests will use a pneumatically operated Fast Reactivity Excursion Device (FRED) which has the capability of rapidly ejecting a slug of neutron absorbing material from the core. The following limitations have been established in the technical specifications. Reactivity insertion rates must not exceed 20$/sec, the maximum ejectable slug worth within the FRED must not exceed 1.30$, the Doppler reactivity coefficient ($T\frac{dk}{dT}$) must be negative with an absolute magnitude of 0.005 or greater, and failed fuel must not be present. In addition, the tests will be carried out in a stepwise manner, and measured results will be compared with analytical predictions at each step in the program to identify any unexpected deviations. Based on our review of detailed calculations by GE, we conclude that the transient program can be conducted within these limitations without expectation of reactor damage or consequent hazard to the public.

Controlled performance of the power excursion tests depends on the reliability of the FRED. Consequently the design and test results of the device have been reviewed in detail. The FRED is mounted on a Positioner Drive atop the reactor vessel head. To induce a transient, the neutron-absorbing slug at the bottom of a shaft is fired upward by sudden pressurization of a pneumatic cylinder at the top of the shaft. When the stroke is complete, the gas is vented and the slug drops back into the core. By adjusting the height of the Positioner Drive, the position of the neutron-absorbing slug with respect to the core can be varied from fully inserted to fully withdrawn prior to firing. Hence, the maximum rate of change of reactivity can be varied from essentially zero up to 20$ per second.

Accidental firing of the device is prevented by three independently actuated latches that must be withdrawn prior to firing, and by an interlock which prevents pressurization of the pneumatic cylinder unless the master switch on the control room console is in the Excursion Test position. Excessively fast withdrawal rates, owing to excessive gas pressure, have been protected against by pressure regulator and relief valves, blow-out diaphrams and the relative insensitivity of slug withdrawal time to variations in gas pressure. The inadvertent use of a neutron-absorber slug of improper worth has been adequately protected against by allowing only one neutron-absorber slug in the refueling cell at a time, electro-etching the worth of each slug on its surface, noting the power depression as the slug is lowered into the core and correlating the reflector segment motion required to return to the initial power, and by progressively approaching larger transients by lowering the positioner drive. The device is equipped with adequate means for monitoring position through position lights, rotary counters and indexing scales.

We conclude that the design of the FRED is adequate and that it should operate in a reliable and predictable way during the experimental program.

Our review of the transient program included consideration of the ability of the reactor vessel and its internals to withstand the dynamic loading that may be imposed as a result of the FRED transients. We conclude that a power excursion caused by reactivity insertion at a rate of 30$/sec can be accommodated without damage of any kind. For the design basis reactivity insertion rate of 50$/sec, we have determined that the limiting loadings would be imposed on the bulkheads in the fuel rods, which separate the upper and lower fuel columns and on the bolts that attach the core shroud to the reactor vessel. The loadings on the bulkheads would be considerably below the elastic limit of the welds holding them in place, and the shroud bolts would elongate a maximum of 4.5%, compared with the 50% elongation need to cause bolt failure. Since reactivity inserting rates associated with use of the FRED will be limited to a maximum of 20$/sec, we conclude that the transient program can be conducted without causing excessive impulse loads on the core and its supporting structure.

Based on our review of the SEFOR application, we have concluded that GE has established a test program for SEFOR that can be carried out safely, and that adequate criteria have been established for the conduct of the transient test program involving the FRED.

3.2 Reactor Surveillance

3.2.1 Anomalous Reactor Behavior

Unanticipated changes in reactivity, fission product activity in the reactor cover gas and core outlet temperatures are the most likely indications of anomalous reactor behavior. Quantitative limits on such changes cannot be established until the reactor has been operated at significant power levels long enough to establish a base line of measured values which are representative of normal operations. Until such quantitative limits are established, the Technical Specifications require that any deviation from expected behavior be reviewed in accordance with the following procedure.

The Technical Specifications direct attention to the need for deliberate evaluation of any anomalous condition and a conscious safety oriented decision as to whether or not operations should continue. The shift supervisor must make the initial evaluation. His decision is reviewed by the SEFOR Site Safety Committee. Additional review and investigation, of anomalous indications, will be conducted by the technical staff of the General Manager, Breeder Reactor Development Operation and the SEFOR Safety Review Committee (comprised of senior GE technical personnel). The Technical Specifications also require that the results of these reviews be reported to the AEC on a timely basis. In its letter on SEFOR the ACRS recommended that "... prior to operation at full power (20 MWt), the applicant and the Regulatory Staff should agree on quantitative definitions of suitable limits on unexplained behavior of reactivity, coolant temperatures and other parameters of significance". Implementation of this recommendation will be accomplished by a requirement that GE submit additional technical specifications, and bases, for Commission approval prior to full power operation. These new specifications are to be based on operating experience of SEFOR up to and including the 10 MWt level.

We conclude that adequate attention will be given to any indications of anomalous reactor behavior and that there is little likelihood that reactor operation will be permitted by GE unless it is clearly safe to proceed.

3.2.2 Fuel Surveillance

Although SEFOR should initially contain high quality fuel we have considered the possibility of a fuel rod failure during reactor operation. There is no concern over fission product leakage from very small cladding defects, such as porous welds, since adequate provisions have been made to store and dispose of fission products from such sources. There would be concern, however, if cladding should become distorted enough to interfere with coolant flow or should develop leaks large enough to permit sodium logging of fuel rods or to permit fuel to be exposed to the bulk coolant. It is clear that some degree of cladding failure can be tolerated during steady state operation of the reactor without creating an unsafe situation. However, if steady-state operations are conducted with failed fuel present, there is need to carefully monitor the reactor to assure that the degree of failure does not progress to the extent that core integrity would be jeopardized.

Failed fuel rods should be absent from the core when pulsed operations with the FRED are conducted. Thus far there is no definitive experimental evidence that the damage to an already failed rod cannot propagate during the pulse and therby affect other nearby fuel rods. Consequently, a failed fuel rod would introduce an element of uncertainty that should be avoided. GE has agreed that pulsed operations will not be conducted if the core contains failed fuel.

With respect to steady-state operations, the ACRS has recommended that before operations at full power, necessary detection steps and specific criteria be developed to judge the acceptability of continued operations in the presence of known loss of fuel clad integrity. To a large extent these measures must be implemented in terms of accumulated experience with methods and equipment to be employed by GE in ascertaining fuel condition, as discussed below. Until specific limitations are established in the technical specifications, the specifications related to anomalous reactor behavior as discussed in the preceding section, will govern operations under conditions where symptoms of fuel failure may occur. The technical specifications prohibit operations at full power until GE has submitted a report on operations through 10 MWt, and the technical specifications are changed to include specific provisions with respect to steady-state operations with failed fuel.

The applicant has elected to use a cover-gas gross gamma activity monitor and spectral analysis of cover-gas grab samples, as the basic means of detecting failed fuel elements. GE has agreed to demonstrate the adequacy of the sensitivity of the gross gamma monitor. If the monitor is not sufficiently sensitive to detect the cover-gas activity increase that would be expected from cladding failure on a single fuel rod, GE will install a more sensitive device.

Should the fuel failure detection system indicate the possible presence of a failed fuel rod, the readily accessible fuel rods (located under the through-head ports of the reactor vessel head) will be withdrawn from the reactor and inspected. Any failed elements located will be removed.

The ACRS has noted the potential usefulness of sodium sampling and analysis for characterizing impurities in the sodium and diagnosing and understanding the probable status of fuel element defects, and has suggested that relatively frequent sodium samples be taken. In accord with this suggestion, GE has agreed to sample and analyze primary sodium for the presence of fission products and other impurities of potential significance on a quarterly basis. In addition, sodium will be sampled in connection with fuel rod examinations for the purpose of locating failed fuel and following each prompt critical FRED transient.

We conclude that an adequate fuel surveillance plan has been established.

3.3 Accident Analysis

In our previous analysis for operation of SEFOR up to 1 MWt, we reviewed the accident potential of SEFOR at 20 MWt and during the proposed transient test program utilizing the FRED. We evaluated core flow blockage accidents, flow failure, loss of coolant, as well as the hypothetical Design Basis Acci- (DBA) for SEFOR. We have extended our review of the loss-of-coolant class of accidents and discuss below the plant modifications and conclusions for those analyses. Our overall conclusions related to the consequences of the accident analyses, based on the DBA, have not changed from those presented in our previous Safety Evaluation. However, our calculated dose values have been slightly reduced, and a summary of the revised results is presented herein.

3.3.1 Loss-of-Coolant Accidents

The loss-of-coolant class of accidents was discussed in our November 16, 1968 Safety Evaluation. Since that time we have examined in considerable detail the spectra of break locations and break sizes for the purpose of determining whether any improvements might be needed to assure effective core cooling in the event of a sodium leak.

Our analysis considered potential feedback effects from auxiliary systems to the primary coolant system during the course of the loss-of-coolant accident. As a result of our analysis, GE has agreed to make provision for automatic isolation of the four-inch reactor vessel drain line and depressurization of the primary drain tank should a 10 psig pressure differential develop between the tank and the reactor vessel. This measure was taken to prevent potential loss of reserve sodium through certain system breaks owing to excessive pressure in the drain tank. GE will modify the reactor vessel vacuum breaker system to assure seal material compatibility with the thermal environment, and will erect a baffle to protect the racks of solenoids (which actuate primary system valves) against splashing sodium. With proper operation of the primary system following a leak we concluded that coolant circulation can be restored to the core. In the event this cannot be accomplished, sodium can be added to the reactor vessel to make up boiling losses from the reactor vessel. Sufficient sodium is stored at the site for three days of such cooling, within which time additional supplies can be obtained.

3.3.2 Design Basis Accident

As discussed in our November 16, 1968 report, the Design Basis Accident was postulated to result from an extreme overpower condition that results in core collapse. A maximum reactivity insertion rate of 50$/second was calculated to result. The total energy for the compaction excursion was conservatively calculated to be 830 MW-sec of which 230 would appear as energy in vaporized fuel. A theoretical upper limit of the energy available as kinetic energy is 100 MW-sec. This is based on isentropic expansion of the vaporized fuel from the applicant's calculated peak pressure of 2000 psi to atmospheric pressure. Based on extrapolations of data from the KIWI experiment, GE believes that the actual available kinetic energy would be less than 20 MW-sec. The upper limit kinetic energy release of 100 MW-sec predicted by the analysis is significantly less than the containment design basis energy release of 400 MW-sec.

Since our previous review, we have considered the possibility that sodium might be involved in the excursion in such a way that its vaporization could increase the energy release. We have estimated that such involvement could increase the kinetic energy by a factor of two. However, the time scale of the release, about 3 milliseconds, is much longer than the several microsecond time scale assumed for the energy release used in containment design. Therefore, the margin between the calculated and design basis energy releases is considerably greater than two to one. Consequently, no loss of containment integrity should occur as a result of the energy release. The pressure buildup would be less than the design pressure of 10 psig, and the event would be adequately contained.

3.3.3 Environmental Consequences

As a conservative evaluation basis we assumed that, as a result of a design basis accident, the entire core is volatilized with 100% of available fission products and 100% of the plutonium released into the inner containment space. As a conservative upper limit, we considered that 100% of the noble gases, 20% of the halogens, 20% of the solid fission products remained suspended in the containment atmosphere and available for leakage for the first two hours. On the basis of particle physics considerations, we assumed that 2.5% of the core inventory of plutonium (30 g/m^3) would remain airborne. We further conservatively assumed this fraction remains suspended for the duration of the accident.

The applicant's calculations of containment pressure versus time for the design basis accident have been used to predict leakage. We have conservatively assumed that the leak rate change is proportional to the square root of pressure. Based on predicted containment pressures, we have used the 30 hour period over which net leakage takes place as a calculational basis. Our meteorological model (Pasquill Type F, 1 m/sec) was used in making dose calculations. Tabulated below are the doses we have calculated for the controlled and low population distances, 0.4 mile and 12 miles, respectively.

	Thyroid Dose (Rem)	Whole Body Dose (Rads)	Bone Dose (Rem)
0.4 mile (2 hrs)	1.0	1×10^{-2}	35
12 miles (30 hrs)	0.1	2×10^{-3}	5

These calculated doses are well below the guideline values set forth in 10 CFR 100, and would not present an undue risk to the health and safety of the public.

4.0 TECHNICAL SPECIFICATIONS

GE submitted proposed Technical Specifications dated March 19, 1969, covering the steady-state operation of SEFOR up to full power (20 MWt) and the pulsed operation of the reactors. The Technical Specifications are in accordance with the guidelines published in the Federal Register on August 11, 1966. As a result of numerous discussions and meetings with GE, the proposed specifications were modified to incorporate comments from the ACRS, and the AEC's Divisions of Compliance and Licensing.

As recommended in the ACRS comments, the Technical Specifications contain the requirement that the experience gained in the operation of the reactor will be used to develop additional specifications which will define (1) the limits on anomalous behavior of reactor operating parameters beyond which corrective action must be taken, and (2) the necessary detection steps, and the specific criteria and bases for judging the acceptability of continued steady-state operation in the presence of a known loss of fuel clad integrity.

As a result of these comments, GE submitted a revised set of Technical Specifications, dated June 20, 1969, which with some minor modifications has been accepted and will be issued with the SEFOR full power license. These Technical Specifications incorporate all necessary requirements to assure safe operation of the reactor for full power, in the pulsed mode, and to assure that necessary safety features will be available in the event of malfunctions in the plant.

5.0 REPORT OF ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards reviewed the SEFOR for full power 20 MWt operation at its March 1969 meeting and issued its report on SEFOR on March 13, 1969. The ACRS report to the Commission is included as Appendix A. The Committee recommended that after obtaining some initial operating experience, due attention be given to development of specific criteria regarding anomalous changes in operating parameters and steady-state operation with a limited number of failed fuel rods. These items were discussed on pages 8 and 9, respectively. The Committee also suggested that the equipment which would permit obtaining sodium samples from the reactor more frequently, be completed. Sodium sampling is discussed on page 10 of this report. Implementation of these recommendations will be completed prior to 20 MWt operation. The Committee concluded that "... if due attention is given to the foregoing comments and if experience in the stepwise experimental program is favorable, the SEFOR can be operated at steady-state powers up to 20 MWt and in the pulsed mode, as proposed, without undue risk to the health and safety of the public".

6.0 CONCLUSION

Based on our review of the SEFOR application, as amended, and the report of the Advisory Committee on Reactor Safeguards, we conclude that GE has planned an adequate test program for SEFOR, that adequate criteria have been established for the conduct of power ascension to full power and the transient test program involving the FRED, and that the SEFOR facility can be operated as proposed without endangering the health and safety of the public.

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