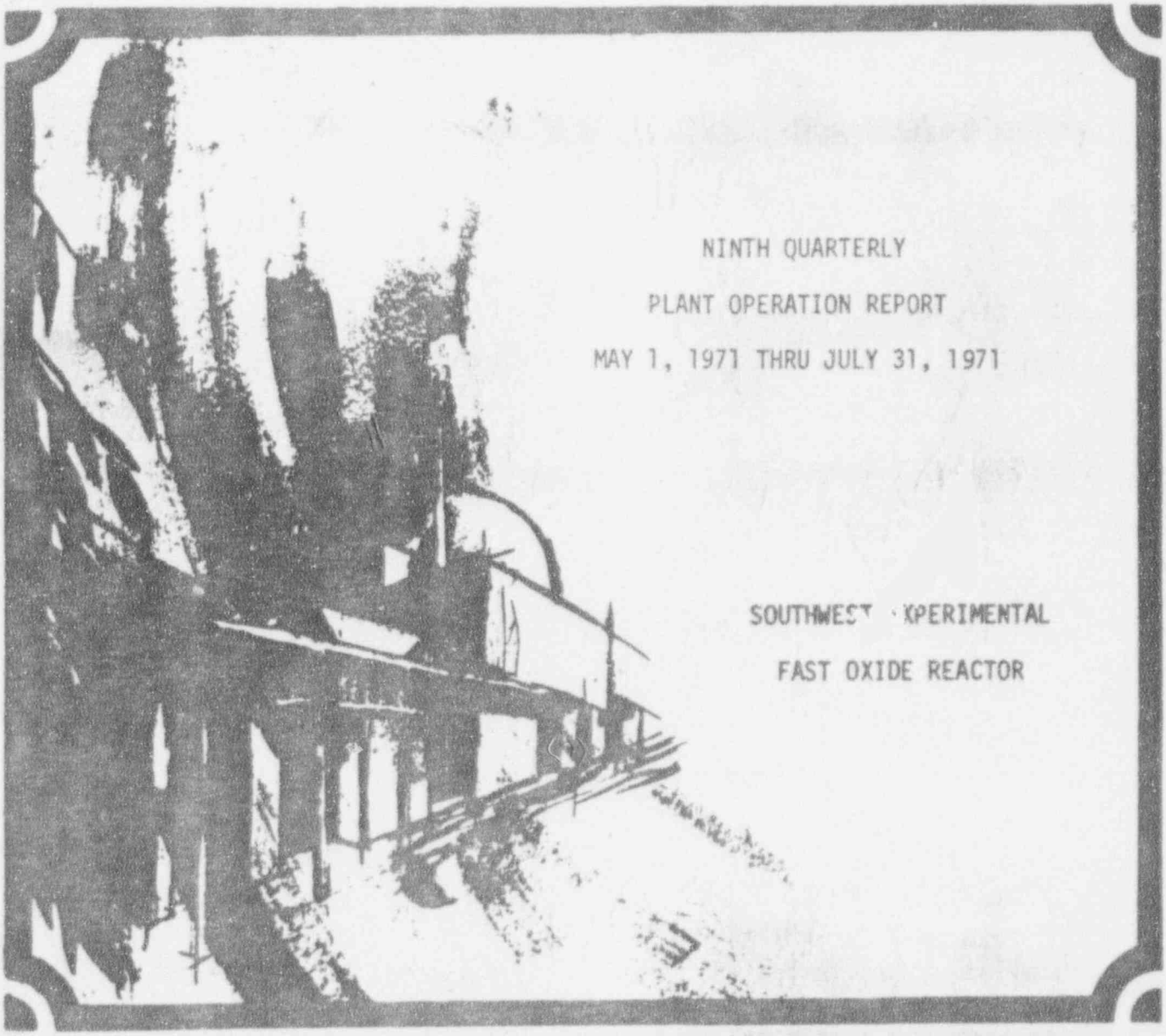

NINTH QUARTERLY
PLANT OPERATION REPORT
MAY 1, 1971 THRU JULY 31, 1971

SOUTHWEST EXPERIMENTAL
FAST OXIDE REACTOR

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SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR

NINTH QUARTERLY

PLANT OPERATION REPORT

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Prepared for transmittal to the Division of Reactor Licensing, United States Atomic Energy Commission, as required by License DR-15.

BREEDER REACTOR DEPARTMENT
GENERAL ELECTRIC COMPANY
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NINTH QUARTERLY
PLANT OPERATION REPORT

A. Introduction

This report is submitted in fulfillment of the requirements of License DR-15 for the report period of May 1, 1971 thru July 31, 1971.

B. Summary of Plant Operations

These data are the result of reactor operation for the period of May 1, 1971 thru July 31, 1971.

1. Operating Data

Reactor Critical	150.0 hours
Maximum Power Level	20.0 MW
Longest Continuous Run to Date (December 27, 1970, thru January 3, 1971)	153.5 hours

2. Plant Shutdowns

The reactor was shutdown on June 15, 1971 to perform the Annual Containment Leak Test and to accomplish modifications to the reactor outer head seal and the Refueling Cell crane system. The inert cells were purged to nitrogen and argon on July 29, 1971. At the end of the report period, the Refueling Cell argon purity was being achieved to terminate the outage on August 1, 1971.

3. Reactor Scrams (See Table I)

Equipment	7
Personnel	1
Manual	11*
Other (Loss of Site Power)	1
Total	20

4. Cover Gas Activity

The Cover Gas Monitor was in service during the quarter, and indicated no anomalous fission gas activity.

Ten cover gas samples were obtained between May 6 and May 8; six in June, and one in July** to quantitatively measure the isotopic constituents. These samples consisted of routine monthly cover gas analyses, special experiments to further refine sampling and identification techniques, and pre and post-FRED transient samples. No significant increase in the concentration of the fission products in the cover gas was observed.

Preliminary examination of these data indicate good correlation with other cover gas samples obtained since December, 1970.

* All eleven planned as part of Test Program.

** The reactor was shut down during the entire month of July for maintenance.

5. Major Items of Plant Maintenance, Instrumentation and Control Work

A total of 196 malfunctions were corrected, distributed as follows:

Mechanical	58
Electrical	60
Instrumentation	78

Significant malfunctions included:

Reactor Building Pressure Controller
Reflector #9 Upper Limit Switch
Positioner Motor Control Circuit & Reductor
MG Set 1B - Reverse Power Relay
Scram Solenoid 7A position switch
480 Volt Load Center 2B
SRM #2
Safety System Relays
Upper Limit Circuit #6 Reflector
Main Primary Pump Control Circuit
WRM's 1, 2, & 3 Electrometer
2.4 KV Bus Undervoltage Relay 227-C
Argon High Velocity Check Valve
Argon Compressor
Breathing Air Compressor
Freon Units 8118, 8119
Containment Leaks
MG Set 1A-DC Exciter Relay

6. Surveillance Testing

a. Compliance testing was conducted in accordance with the Technical Specifications using LTP's (License Test Procedures).

Weekly Tests	223
Bi-Weekly Tests	10
Monthly Tests	43
Quarterly Tests	44
Semi-Annual Tests	5
Annual Tests	4
Total	329

b. Maintenance Calibration Testing was conducted in accordance with Technical Specifications.

Monthly Calibrations	5
Semi-Annual Calibrations	16
Annual	2
Total	23

c. The annual containment leak test was performed during the weeks of June 14 and June 21, 1971. The refueling cell was subsequently modified, as discussed in item 9.i of this report, and inner containment leaks were located and repaired. The inner containment leak test was then repeated during the week of July 19, 1971, to demonstrate the integrity of the refueling cell modifications. The results for both leak tests are presented below:

	Allowable Leakage, %/Day	Measured Leakage Rates, %/Day	
		June, 1971	July, 1971
Inner Containment	16.5	9.00 ± 0.39	4.4 ± 0.39
Outer Containment	1.4	0.23 ± 0.15	(Not Measured)

d. Samples of the sodium for the SEFOR primary loop were obtained June 16, 1971. One set of samples was retained on site for radiochemical analysis. The other set of 3-sample cups were sent to Vallecitos Nuclear Laboratory, where they were analyzed for metallic impurities as well as radiochemically.

These results are summarized below. An increase in the aluminum concentration was noted, and an analysis of a second sample is in progress to corroborate these results.

Metallic Constituents of SEFOR Sodium in Sample Obtained June 16, 1971

Element	Concentration (ppm)
Al	100
Ca, Co, Mg	6
Cr, Fe	4
Ag, Si	2
Mn	0.6
P	<100
Sb, Nb	< 30
B, Li	< 10
Ba	< 5
Sn, Cu	< 3
Pb, Ni, Ti, Zn	< 1
Mo, V	< 0.3
Be, Bi	< 0.1

< Represents lower limit of detection for instrument used.

The carbon content was 22 ppm, while the U-235 was 0.2 ppb and the U-238 was 5 ppb.

Radiochemical Data

Nuclide	dpm/16 gm sample
Na-22	1.1×10^6
Ag-110	4.4×10^4
Sb-124	$<4 \times 10^3$

7. Radiation Monitoring Program

a. Environmental Sampling (May 1, 1971 through July 31, 1971)

Number of Vegetation Samples Analyzed	15
Number of Soil Samples Analyzed	6
Number of Water Samples Analyzed	8

1) Results of Vegetation Analyses

Average Radioactivity Content (pC_i/gm-ash)

Month	Alpha	Beta
May	<15	1108
June	<15	1151
July	<15	917
Recheck Level	50	1820
Pre-Operational Avg.	13	987

No evidence of Co-60, I-131, or Na-24 was observed in the vegetation samples before transmittal from the site.

2) Results of Soil Analyses

Average Radioactivity Content (pC_i/gm)

Month	Alpha	Beta
May	24.3	25
June	22	27
July	23	22
Recheck Level	32	45
Pre-Operational Avg.	25	34

No evidence of Co-60 or Cs-137 was observed above detection limits.

3) Results of Water Analyses

Average Radioactivity Content (μC/mℓ)

Month	Alpha	Beta
May	$<1 \times 10^{-8}$	3×10^{-8}
June	$<1 \times 10^{-8}$	3×10^{-8}
July	$<1 \times 10^{-8}$	4.6×10^{-8}
Recheck Level	3×10^{-8}	1.5×10^{-7}
Pre-Operational Avg.	$<2 \times 10^{-9}$	6.1×10^{-8}

No Co-60 or Cs-137 were observed above detection limits.

b. Environmental Film Monitoring (May 1, 1971 thru July 31, 1971)

Number of Stations	17
Total Films Analyzed	51
Maximum Radiation Level Reported	16 millirad/quarter*
Maximum Radiation Level Reported during Pre-Operational Survey	8 millirad/month

c. Personnel Monitoring

1) Number of film badges issued:

May	48
June	69
July	79

2) Personnel Maximum Whole Body Radiation Received (Quarter ending June 30, 1970)** — 430 mrem

3) Personnel Maximum Whole Body Radiation Received during July, 1971 — 230 mrem

4) Number of Exposures to Radioactivity Concentrations in Air in Excess of that specified in 10 CFR 20 — None

5) Number of Radiological Spills or Contamination Incidents — One***

* The report for July, 1971, showed 16 millirad at 1 station. However, the control badge showed 4 millirad, indicating possible irradiation of the film in transit or storage. (The reactor was shut down during the entire month of July.)

** Reported on a Calendar Quarterly basis for April, May, and June, as per 10 CFR 20, Section 20.39 (ii).

*** Result of irradiating the temperature sensitive paints for the FRED poison rod. Necessitated minor refueling cell cleanup.

8. Off-Site Radioactivity Release and Shipments (May 1, 1971, through July 31, 1971)

a. Liquids

1) Fission and Activation Products (except tritium)

a)	Total curie activity released (Ci)	$<3.3 \times 10^{-6}$
b)	Total volume of liquid waste discharged (gallons)	1.15×10^{4}
c)	Total volume of dilution water (gallons	5.0×10^{4}
d)	Volume average concentration at discharge point $\frac{(1.a)}{(1.c)} \times 264$ (μCi/mℓ)	$<1.9 \times 10^{-8}$
e)	MPC used*(μCi/mℓ)	3.0×10^{-5}
f)	Percent of limit (%)	<0.06
g)	Maximum concentration released, averaged over not more than 24 hours (μCi/mℓ)	3.0×10^{-7}

2) Tritium

a)	Total curie activity released (Ci)	4.4×10^{-2}
b)	Volume average concentration at discharge point: $\frac{(2.a)}{(1.c)} \times 264$ (μCi/mℓ)	2.3×10^{-4}
c)	Percent of limit (%)	7.7

3) Estimated Carbon 14 release (Ci) 4×10^{-3}

Note: All liquids are released to a tile field. Measured concentrations refer to values at the point of discharge into the tile field.

b. Gaseous

1) Noble and Activation

a)	Total curie activity released (Ci)	6.38×10^{-3}
b)	Total volume of gas released (ft^3)	1.27×10^{3}
c)	Time average release rate $\frac{(b.1.a)}{7.9}$ (μCi/sec)	0.81×10^{-3}
d)	MPC used (μCi/mℓ)**	2×10^{-8}
e)	Licensed limit for annual average (μCi/sec)	800
f)	Percent of annual limit (%)	1×10^{-4}
g)	Maximum hourly average release rate (μCi/sec)	5.1×10^{-2}
h)	Licensed limit for hourly average (μCi/sec)	3400
i)	Percent of hourly limit (%)	1.5×10^{-3}

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* Na-22 identified as gamma emitter.

** Based on Kr-87 observed in cover gas.

2) Halogens with half-lives >8-days and particulates with half-lives >8-days.

a)	Total curie activity released (Ci)	$<3.6 \times 10^{-8}$
b)	Time average release rate $\frac{(b.2.a)}{7.9}$ (μCi/sec)	$<4.6 \times 10^{-9}$
c)	MPC used* (μCi/mℓ)	1×10^{-10}
d)	Licensed limit for annual average (μCi/sec)	5.6×10^{-3}
e)	Percent of limit (%)	$<1 \times 10^{-4}$
f)	Maximum hourly average release rate (μCi/sec)	$<9.4 \times 10^{-7}$
g)	Licensed limit for hourly average (μCi/sec)	5.6×10^{-2}
h)	Percent of hourly limit (%)	$<1.7 \times 10^{-3}$

c. Number of Samples analyzed during quarter ending July 31, 1971

Liquid	18
Gaseous	9

d. Number of Radwaste Discharges

Liquid	16
Gaseous	9

e. Radioactive Shipments

Quarterly Summary of Radioactive Shipments
May 1, 1971 thru July 31, 1971

Date	Description	To	Radioactive Content
7/7/71	Approximately 60 gm Na (Primarily Na-22)	GE (Vallecitos)	<1 mCi

* Based on the possible presence of I-131.

9. Significant Modifications Approved by Facility Manager and Completed During Report Period.

a. Sodium Level Switch Fuse Relocation

The fuse in the low sodium level switch bridge circuit was relocated, so that a blown fuse will cause a loss of power to both legs of the bridge circuit rather than to just one leg. This provides added assurance of detecting any malfunction of the bridge circuit, by causing a trip signal to occur in the event of a fuse failure.

b. Half-Ton Grapple Modification

The half-ton grapple was modified by the addition of two telescoping springs between the grapple body and the cable attachment. As the grapple is lowered and a rod becomes fully inserted, the springs are compressed by a 100 lb weight above them which permits time for coastdown to occur without creating a slack cable. When handling fuel with the half-ton grapple, the 100 lb weight is attached to the grapple to permit seating the tightener rods. Prior to modification, insertion of a tightener rod caused the load to drop off and the low load limit to shut off the hoist, but coastdown created a slack cable. The slack cable permitted the grapple and weight to fall to one side creating the potential for bending the rod.

c. Safety System Jumper Panel

A safety system jumper panel was installed in the Control Room to eliminate the use of alligator clip wire lead jumpers and the attendant potential for shorting between terminals or improper terminal selection. The new panel uses a jack plug to create a circuit to jumper a selected function. The new jumper devices are controlled in the same manner as the wire jumpers, by storage in a locked cabinet, with each use entered in a jumper log.

d. Man Access Suit Flowmeter

A flowmeter equipped with a high and low flow switch was installed in the Man Access air supply line. The low flow trip closes the shut-off valve for the vacuum exhaust system by means of a three-way solenoid which vents the pressure from the diaphragm of the shut-off valve. Electrical power to the solenoid valve is supplied thru a pair of normally

open contacts which are actuated by the low flow trip. Loss of electrical power does not cause the vacuum exhaust shut-off valve to close. An audible alarm is actuated by either the low flow or a high flow switch. A manual globe valve was installed on the outlet of the flowmeter to regulate flow thru the meter.

e. Pump-Around-Pump, Volt-pac Control

The gear package for the Pump-Around-Pump volt-pac was modified to require 60 seconds for changing the voltage from zero to full voltage. Previously the time required was 30 seconds. For reactor operation, the flow is limited to 1 gpm to provide greater sensitivity for detection should a small leak occur in the primary coolant system. Due to the low pump voltage and pump head required to maintain this flow rate, system changes which affect bus voltage or cover gas pressure cause the pump around pump flow rate to drift. The flow rate must then be returned to an acceptable value by making small corrections to the pump voltage. This modification slows the response of the volt-pac control system to improve the ability to make small corrections in the applied pump voltage.

f. Core Flux Detector System Calibrator Circuit Shielding

The source of 60 cycle noise in the signals from the transient fission chambers was created by unshielded cables in the calibrator circuits. Shielded cables were installed to eliminate the noise.

g. Measurement Circuit for Excursion Mode Time Delay

A circuit has been provided to more accurately measure the Excursion Mode Time Delay. A set of spare contacts on the FRED Fire Switch and spare contacts on the relays in the scram chasses were used to provide a signal to a scaler timer to give more accurate measurement capability of the Excursion Mode Time Delay. The circuit does not affect the normal function of the Safety System.

h. Reactor Vessel Head Seal Backup

A seal was installed between the reactor vessel outer head and the vessel support skirt to provide a cover gas backup seal. When the primary coolant temperature is increased, the resulting rotation of the reactor vessel flange temporarily reduces the tension in the outer head bolts. (See item 7, page 19, of the Eighth Quarterly Plant Operation Report.) In some instances, this may cause cover gas leakage to occur. The backup seal was installed to reduce this leakage to a negligible amount.

Following the installation, tests were performed at rated pressure to verify the effectiveness of the seal. As a part of the seal installation, a seal clamp was added under the nut and washer on each outer head stud. In order to provide adequate thread engagement of the stud tensioner puller bar socket on the outer head studs, the tensioner support barrel was shortened by one inch.

i. Refueling Cell Crane Emergency Retrieval System

The Refueling Cell and the Refueling Cell Crane have been modified to permit recovery should a malfunction of one or more of the components occur. The modification involved:

1) drilling one hole in the north wall of the Refueling Cell for trolley retrieval,

2) drilling four holes in the west wall of the Refueling Cell for bridge retrieval,

3) installation of the bridge retrieval system,

4) installation of parts to permit trolley retrieval,

5) changes in the ½ ton and 10 ton load blocks to permit attachment of auxiliary hoist cables,

6) changes in the ½ ton grapple weight,

7) changes in the reactor vessel head lifting fixture to permit manual lowering or raising of the head.

Following the modification, the operability of the crane system was demonstrated. A test was performed to demonstrate the ability to move the bridge and trolley using the emergency retrieval system. The inner containment leak test was repeated to demonstrate the integrity of the welds and the new penetrations added as a part of the crane emergency retrieval modification. Some volt-pacs and conduit runs were relocated to provide room for the refueling cell penetrations added as part of the modifications.

j. FRED Positioner Lower Limit Block

The reactivity insertion inferred from the measured transient flux data during the familiarization transients was about 4% greater than that determined from measurements of the poison slug reactivity worth during static tests. A lower limit block (shim) was installed on the FRED positioner to limit its travel when the poison slug was lowered into the core. With the shim installed, the maximum static worth of the sub-prompt poison slug was limited to 0.94$, which would correspond to a dynamic worth of 0.98$.

k. Fission Product Monitor Hold-up Volume

A 4 inch schedule 40 stainless steel pipe, 11 feet in length was inserted into the Cover Gas Monitor loop inside the Refueling Cell near the point at which the loop penetrates the wall. The additional loop volume is designed to reduce the Cover Gas Monitor background signal created by the presence of Ne^{23}, an activation product (half life 38 seconds) produced by an n, p reaction on Na^{23}. This additional volume of 1 ft^3 provides a delay of approximately 2½ minutes (or nearly four Ne^{23} half lives) at a flow rate of 0.4 ft^3 per minute.

l. WRM Picoammeter Protection

The Wide Range Monitor circuitry was modified to limit the input current to each WRM to approximately 1.2 milliamperes, which is less than the saturation value of the input electrometer tube, but about three times the value at 20 MWt.

During the sub-prompt transient ($0.93 from 5 MW) on May 26, 1971, the peak power attained was about 110 MW. WRM No. 1 and No. 2 output signals were connected to the Data Acquisition System. The signal from WRM No. 1 reached saturation and then turned abruptly downscale. Subsequent tests indicated a zero shift had occurred and the WRM had become less stable. Measurements were performed to determine the input currents at which each WRM saturated. Since the input currents to WRM No. 2 and No. 3 reached saturation values during the measurements, zero drift and some instability were observed on these two instruments. Attempts to operate the reactor in steady state conditions subsequently were interrupted by spurious high flux scrams which occurred during changing of the range selector switch. The scrams were caused by instable electrometer pairs in the picoammeter module of amplifier No. 1 in the WRM circuitry. The electrometer pairs were replaced and the WRM stability was restored. The current limiting circuit installed was designed to prevent damage to the new electrometer pairs.

The current limiting circuit involved the addition of two diodes (1N3575) in parallel, but with opposite directions of current flow, in series with a set of contacts on the K1 relay and connected between the WRM input and the common (ground). The current limiting circuit was installed only in the previously unused uppermost range of the WRM which is above the 0 to 125% of power range. The resistor-capacitor feedback network which existed in this range (R1 and C15) was removed. When the WRM range selector switch was placed in this position (identified as "0-125% Excursion"),

the resistor-capacitor feedback network of the 0-125% range was used, the connection being made between the two range positions by a diode (1N914). Only in this position was relay K1 energized to connect the current limiting circuit into the input circuit of the WRM. Thus, for all steady-state operation, the current limiting circuit was not connected.

Bench tests of the current limiting circuit with input currents up to 10 milli-amps indicated proper functioning of the circuit. The maximum current that can be produced by the neutron detector for the WRM is approximately 8 milli-amps (determined by the high voltage power supply). Current leakage thru the diode is of the order of 10^{-7} ma (about 10^{-4}% power), which will have a negligible effect on the input signal during operation in the power ranges or during the transients.

m. Cleanout "Y" on the NaK Bubbler

A "Y" was installed on the NaK Bubbler inlet line to allow cleaning the inlet pipe. The pressure drop through the NaK Bubbler had increased over the past few months because of oxide deposits in the inlet pipe. The cleaning which was performed following the "Y" installation resulted in a reduction in the pressure drop across the bubbler.

n. Transient Flux Detector System

Lead shot had been placed around the fission chambers to reduce the gamma dose rate on the chamber. The use of lead shielding created a softer neutron spectrum at the detector and a large number of lower energy gammas. Both effects contributed to a higher background current. The softer neutron spectrum caused a higher capture to fission ratio, and hence more background from the decay of U-239. The low energy gamma radiation contributed more heavily to the photoelectric effect than the high energy gammas. Replacement of the lead by B_4C reduced the capture to fission ratio by hardening the neutron spectrum.

o. FRED Position Drive Unit Modification

The positioner drive motor and gear reductor were modified so that a man in a man access suit can install the unit and make proper alignment with the drive shaft on the positioner. The alterations provide vertical, lateral and rotational adjustment on the mounting base. The grease that was formerly used in the reductor gear box was replaced with standard oil (NRRO-85 SSU at 210°) to eliminate gear tooth wear which had required premature replacement of the reductor gear. An "O" ring seal was added on the lower part of the gear box at one of the shaft bearings.

C. Other Reportable Items

1. Safety System Relay Malfunctions

During the performance of the Surveillance Test LTP M-Q-9 "Monthly Channel Test of the Reactor Safety System Automatic Trip Functions" on May 23, 1971, the high flux trip for WRM No. 3 did not cause a scram when the test toggle switch was actuated. The switch was actuated several times, but no trip occurred. The source of the malfunction was identified as stuck contacts on mercury-wetted relay K1 (Automatic Electric Co., Series V-4, Part No. PM-4400-150A) in Chassis C of the safety system. These contacts are in series with the scram contactor coil across the 125 VDC supply. Contact protection circuits had been added in parallel with the scram contactor coil. Chassis C was removed from the panel and inspected. No indication of an overcurrent condition (melted insulation, burned components, or discoloration) was observed. The relay was removed for inspection and when it was inverted, the contacts opened. A new relay was installed and the surveillance test was completed with satisfactory results. The high flux trips for WRM No. 1 and No. 2 functioned properly in all tests. Since the trip logic is one-out-of-three, high flux scram protection was maintained at all times.

During the performance of the above surveillance test, it was also found that the power supply breaker for the main secondary coolant system pump (MSP) winding did not open when the test switch for the high pump winding temperature trip was actuated. The source of this malfunction was identified as stuck contacts (mercury-wetted) on relay K15 in Safety Chassis B1. This pair of contacts transmits a 26.5 VDC signal through Auxiliary Trip Chassis Z to trip the MSP supply breaker. The safety system scram trips in Chassis B1 functioned properly. The contacts on the relay which malfunctioned did not open when the relay was removed for inspection. A new relay was installed and surveillance tests were performed with satisfactory results. Examination of the components and wiring in Chassis B1 revealed no indication of an overcurrent condition. Both relays were replaced and satisfactory system operation was demonstrated. The relays which malfunctioned were returned to the manufacturer with a request for an analysis of the cause of the malfunction, and an estimate of the expected lifetime for mercury-wetted relay contacts. The manufacturer attributes the relay failure to "currents or voltages in excess of specified limits, thereby destroying the protective mercury coating." The request for an estimate of an expected lifetime was not answered by the vendor, however a test was recommended which would test the relays for bridging action which may be related to an individual relay life expectancy. The manufacturer reiterated the need for some form of contact protection, with a preferred contact protection being a resistor-capacitor network. The manufacturer's recommendations are under review by the SEFOR Staff and BRD Engineering.

2. Cladding Temperatures for the FRED Poison Rod

The maximum cladding temperature for the FRED poison slugs was originally estimated to be from 1370°F to 1550°F. (Reference Supplement 19, page 37, 38.) Material compatability tests were run at 1600°F to demonstrate the capability of the poison rod to operate at the expected temperatures. Subsequent thermal analysis of the final design resulted in estimated cladding temperatures of up to 1760°F for some of the planned test points. The initial power levels for the sub-prompt critical and super-prompt critical transient tests were therefore reduced accordingly, so that the maximum estimated cladding temperature was 1540°F.

Prior to the performance of the sub-prompt transient tests, steady state tests were performed, using temperature sensitive paints applied to the poison rod, to investigate the cladding temperatures in the reactor. However, the data obtained from these tests were inconclusive. As a result, the maximum allowable initial power level for transient tests (with the poison rod inserted into the core) was limited as discussed above, based on temperatures calculated from heat transfer analyses.

3. Reactor Head Bolt Surveillance

Corrective actions taken to reduce the head bolt stresses to acceptable values included a reduction in the bolt pre-load to a value of 3000 psi on April 18, 1971. The pre-load reduction required use of the stud tensioner, and during this process the strain gages and leads were damaged. (These strain gages had been used to investigate bolt stresses, as reported on page 19, item 7, of the Eighth Quarterly Plant Operation Report.) Repairs were attempted but subsequent data obtained from the gages proved to be unreliable.

After the bolt pre-load was reduced, and the reactor coolant system had experienced several thermal cycles at 10°/hr, nine of the outer head bolts were checked for possible elongation by applying a pre-load of 3000 psi with the stud tensioner and measuring the nut rotation which could be accomplished at this pre-load. This method is not very accurate, since a nut rotation of 3/16 inch (measured on the nut O.D.) is equivalent to a change of about 3600 psi in bolt stress and nut rotation depends on the torque applied by hand through the stud tensioner apparatus. Variations of up to 5/16 of an inch had been previously noted for measurements on adjacent bolts. Measurements obtained during this inspection show less variation with five measurements indicating zero rotation. (See table below.) Therefore, considering the accuracy of this measurement, it was concluded that there was no loss in bolt pre-load during thermal transients with the initial

pre-load reduced to 3000 psi. The results of the m[illegible]ments (made on June 17, 1971), are given below:

Bolt	Nut Rotation When Tensioned to 3000 psi
7-1	0
11-1	3/16
14-1	0
4-2	0
14-2	1/4
3-3	1/16
7-3	0
12-3	0
15-3	1/4

The submittal containing information regarding Proposed Change No. 5 to the Technical Specifications, dated April 28, 1971, stated that the periodic surveillance program would be initiated prior to August 31, 1972, and that one of the outer head bolts examined in March, 1971, would be re-examined between the sub-prompt and super-prompt transient test programs. These requirements have been met or exceeded as discussed below.

(1) All of the outer head bolts and all of the inner head bolts were examined with ultrasonic test equipment on July 16, 1971. The results of these tests showed no observable defects in any of the bolts.

(2) Four outer head bolts were removed for visual inspection, dye penetrant checks, and length measurements in July, 1971. Three of these bolts had been examined previously in March, 1971. The visual examination and dye checks showed no evidence of cracks, wear, corrosion, or galling. The thread on bolt number B35 was damaged for about 1 1/2 turns on the nut end of the bolt, but this does not affect its serviceability. Bolt number D3209-13-47 was installed to replace a bolt which had sustained damage to several threads due to a malfunction of the stud tensioner.

The length of the bolts had not changed, as indicated by the data in the following table:

Outer Head Bolt Examination Results

Bolt No.	Old Position	New Position*	Bolt Length at 79°F March, 1971	July, 1971
B35	14-2	5-2	24.215	24.215
B9	15-3	12-3	24.363	24.362
B14	14-1	12-1	24.315	24.313
D3209-13-47	--	12-2	--	24.420

*The bolts were moved to the new position in July, 1971, so that they would be more accessible for future surveillance.

4. Primary Drain Tank Venting System Malfunction

Preparations for scheduled fuel surveillance operations required reduction of the cover gas pressure in the reactor vessel and primary drain tank on May 8, 1971. Because a gradual increase in restriction to flow through the normal primary drain tank venting line had been observed during previous use of this system, a decision to use the emergency venting system at a controlled flow rate was made. The drain tank pressure was satisfactorily reduced by this method, but the results indicated that the emergency system was also partially restricted.

The emergency drain tank venting system provides a backup to the shut-off valve in the pump around loop line, by automatically de-pressurizing the drain tank if a primary coolant system failure were to cause a reduction in reactor vessel cover gas pressure. This backup function provides added assurance that a sufficient amount of reserve sodium will be available to cool the core following such a failure.

Since the available information indicated that a Limiting Condition for Operation (LCO) could not be met, reactor operation was not resumed until the cause of the malfunction could be investigated and corrective action could be taken.

Initial inspection of the control valves, located within the nitrogen atmosphere of the inner containment, did not reveal the cause of the malfunction, although a subsequent test did cause a flow restriction in the normal vent system valve due to condensation of sodium vapor. The nitrogen atmosphere was then changed to air and inspections of the system indicated that sodium or sodium oxide, deposited on the check valve between the vent line and the gaseous radwaste header, had apparently caused the check valve to restrict the flow. The

valve was cleaned, the system was returned to service, and normal operation was demonstrated by standard surveillance test procedures. The rate of flow in the normal vent line had been controlled by partially throttling a valve in the line. Restriction to flow in the normal vent system had occurred earlier due to sodium condensation in the throttled valve. To eliminate this problem a short length of ¼ - inch tubing was installed in the normal vent section of the system to provide the required flow control with the manual shut-off valve in the full open position. A performance criterion has been defined for normal operation of the system, to assure early detection of any future malfunctions of this nature.

The check valve was re-inspected on June 26, 1971, following the annual containment leak check. This inspection revealed a small amount of sodium had been deposited on the valve, but the amount was on the order of 10% of that which had been previously observed. The valve was cleaned and the system was returned to service. It was decided that the surveillance of this system should be increased by adding an annual inspection and cleaning of the check valve.

5. Nitrogen Blower Control Switch Malfunction

During the performance of the surveillance test, LTP SA-0-5, Semi-Annual Test of Electrical Distribution System Logic on July 30, 1971, a malfunction was detected which prevented a nitrogen blower from starting. The sequence of events leading up to the detecting of the malfunction was as follows:

> No nitrogen blowers were operating. The "blower preferred" start switch was in "position 3" (No. 3 blower). No. 3 Blower control switch was in "pull to lock" position (blower could not start). The No. 3 blower control switch was shifted to the "after trip" position, but the blower failed to start.

Diagnosis of the malfunction revealed that the resistance of the start-initiating contacts on the control switch was so high that sufficient current flow could not occur. Visual inspection of the contacts verified that the contacts were dirty or corroded and that the normal wiping action of the contacts on closing did not occur. The contacts were cleaned and the surveillance test completed with satisfactory results. The control switch has subsequently been replaced with a new unit whose contact wiping action was verified to be correct. If a loss of site power had occurred, and the diesel generator had started, and if the "blower preferred" switch had been in "position 3," the blower may not have started automatically. However, either of the other two blowers could have been started automatically by shifting the "blower preferred" switch or another blower could have been started by manual action of the control switch for that blower. Since the temperature increase in the nitrogen zone is slow even with no blower operating, no immediate unsafe or adverse plant condition would have occurred.

The "blower preferred" switch is routinely kept in "position 1" to limit the load on the diesel generator during automatic loading. Only in the unusual event that No. 1 blower was out of service might the No. 3 blower have been selected for preferred starting.

6. Reactor Sodium Temperature Change Rates

A pre-load stress of 3000 psi was established in the outer head bolts on April 18, 1971, after stable temperature conditions had been attained with a sodium temperature of less than 500°F. The inner head bolts were retensioned to a pre-load stress of 7700 psi when the head was reinstalled following the IFA installation.

The sodium coolant temperature changes were limited to 10°F/hr for temperature changes greater than 125°F. This rate prevents the combined pre-load and temperature induced bolt stresses from exceeding the ASME pressure vessel code allowable values.

7. Sodium Temperatures for Fuel Surveillance

The 10°F/hr temperature change rate further lengthened the time required to decrease the temperature to less than 400°F for fuel surveillance. The possibility of fuel removal and insertion at temperatures greater than 400°F was investigated. Data presented in the FDSAR showed that the stresses resulting from inserting a fuel rod at room temperature (90°F) into sodium at 400°F do not have a significant effect on the predicted fatigue life of the fuel rod. Insertion of a fuel rod into sodium at temperatures above 400°F will not reduce the predicted fatigue life if the initial temperature difference between the fuel rod and sodium does not exceed 310°F. This is true for clad temperatures up to 800°F. (Ref. ASME Code Case 1331-1 which gives a single fatigue curve for high alloy steels at temperatures of 800°F or less).

Tests were performed to determine the cool-down rate for a fuel rod after heating the rod in the vacuum distillation station to 700°F. The coolest point on the rod was determined to be 285°F after 20 minutes cooling in the refueling cell. The time to transfer a rod from the distillation station to the reactor (fully inserted) is less than 20 minutes. A sodium temperature of 575°F allows insertion of a heated rod without exceeding the 310 °F temperature difference. Reinsertion of fuel rods into the core after heating in the distillation station to 700° is now accomplished with a maximum sodium temperature of 575°.

8. Main Primary Pump Power Supply Malfunction

During reactor operation at low power on June 10, 1971, a malfunction of a component in the emergency power supply for the main primary pump was detected. The malfunction did not affect the coolant flow rate under normal operating conditions. However, if a plant power

loss had occurred, the primary coolant flow rate after reactor scram would have been about 40 percent of the normal value. This would result in only a slight increase in reactor coolant temperature during the first minute after a plant power loss.

When the malfunction was discovered, the reactor was shut down until repairs were completed.

A flywheel is provided for each motor generator set to provide a temporary source of energy for the main primary pump power supply in the event of loss of power to the M-G sets. When such loss of power occurs, the generator excitation is automatically switched from the normal rectified a-c supply to the 125 V d-c storage batteries. A follower circuit maintains the battery supplied excitation at approximately the same voltage as the normal excitation to minimize the change in coolant flow rate on loss of normal power. The excitation is maintained at a constant voltage for a short time (about 20 seconds to 1 minute) after loss of power. A time delay relay is then actuated to reduce the excitation voltage to zero and prevent overheating of the field windings after the motor-generator set rotational speed falls below the value required for adequate cooling of the windings.

During routine data collection, an operator noted that a resistor in the follower circuit for M-G set 1A was overheated and that the follower rheostat was driven to a low voltage position. The follower rheostat for M-G set 1B was in a position which would provide full flow after power loss occurred. The reactor, which was operating at criticality (zero power), was then shut down to permit correction of the malfunction.

The overheated resistor was in series with a time delay relay set of contacts and the rheostat motor winding across the 125 V d-c power supply. The relay contacts are enclosed in a phenolic case and are actuated by a plastic plunger through a hole in the side of the case. Investigation indicated that the plastic plunger was stuck in the hole and was holding the contacts in a partially closed position. The relay had been actuated last on June 2, 1971, when a plant power voltage dip caused transfer to the emergency power supply (flywheel) system. After the rheostat had run down to the zero voltage position following loss of normal power, a limit switch opened to remove power from the rheostat motor winding. When normal power was restored, the rheostat moved to a higher voltage position, closing the limit switch. The control circuit could then energize both the raise and lower windings on the rheostat motor. The motor was not damaged, but the resistor became overheated.

The malfunction was corrected by enlarging the hole through which the plunger operates, lubricating the plunger, and replacing damaged circuit board components.

9. Instrument Nitrogen Supply Lines to Valves in Nitrogen Zone

Item C.5 in the Eighth Quarterly Plant Operation Report described the breaking of the instrument nitrogen supply line to the Reactor Overflow Valve. An inspection of all such lines in the Nitrogen Zone was performed during the summer outage. The inspection was intended to detect damaged lines, lines with inadequate length for flexing, or inadequate protection against vibration, and the possibility of work-hardening. One potentially damaged line was removed which had a bulge near the ferrule. All other lines were found to be in satisfactory condition.

D. Safety Review and Audit Activities

1. The seventh meeting of the Safety Review Committee was he) at the site on June 8 and 9, 1971.

2. Twenty-one meetings of the Site Safety Committee were held during this quarter.

3. One trip was made to the site by G.E. personnel from the Sunnyvale office to review plant safety.

TABLE I

REACTOR SCRAMS

DATE	CAUSE	NUMBER
5/6/71	Site Power Loss During Thunderstorm	124
5/8/71	FRED Test (TP V-1) 80¢ @ 1 MW	125
5/15/71	FRED Test (TP V-1) 76¢ @ 1 MW	126
5/17/71	FRED Test (TP V-1) 93¢ @ 2 MW	127
5/25/71	FRED Test (TP V-1) 93¢ @ 2 MW	128
5/25/71	Operator Error (Improper Range Change WRM #2)	129
5/26/71	Spurious Trip WRM #1 While Switching Ranges	130
5/26/71	FRED Test (TP V-1) 93¢ @ 5 MW	131
5/28/71	Spurious Trip WRM #2 While Switching Ranges (125×10^{-1} to 40×10^{0} %P)	132*
5/29/71	Spurious Trip WRM #2 While Switching Ranges (125×10^{-1} to 40×10^{0} %P)	133*
5/29/71	Spurious Trip WRM #2 While Switching Ranges (125×10^{-1} to 40×10^{0} %P)	134*
6/3/71	Spurious Trip WRM #2 While Switching Ranges (125×10^{-1} to 40×1 %P)	135*
6/3/71	Spurious Trip WRM #1 While Switching Ranges (125×10^{-1} to 40×1 %P)	136*
6/6/71	Spurious Trip WRM #2 While Downscaling on Reflector Rundown (40×10^{-4} to 125×10^{-5} %P)	137*
6/7/71	FRED Test (TP V-1) 93¢ @ 5 MW	138
6/7/71	FRED Test (TP V-1) 93¢ @ 5 MW	139
6/8/71	FRED Test (TP V-1) 93¢ @ 10 MW	140
6/10/71	FRED Test (TP V-1) 93¢ @ 10 MW	141
6/12/71	FRED Test (TP V-1) 93¢ @ 10 MW	142
6/13/71	FRED Test (TP V-1) 93¢ @ 10 MW	143

* These scrams were related to the effects of the transient test on 5/26/71. The cause was corrected as described in item 1 on p 11 of this report

DEFINITIONS

ABC	Air Blast Cooler
APS	Auxiliary Primary System
ARM	Area Radiation Monitor
ASS	Auxiliary Secondary System
Aux.	Auxiliary
BRD	Breeder Reactor Department
CP	Corrective Procedure
EM	Electro-Magnetic
EP	Emergency Procedure
FCV	Flow Control Valve
FRED	Fast Reactivity Excursion Device
IFA	Instrumented Fuel Assembly
IFST	Irradiated Fuel Storage Tank
IHX	Intermediate Heat Exchanger
IRM	Intermediate Range Monitor
LTP	License Test Procedure
MPS	Main Primary System
MSS	Main Secondary System
NFSV	New Fuel Storage Vault
PAP	Pump-Around-Pump
PCV	Pressure Control Valve
PM	Preventive Maintenance
PTP	Provisional Test Procedure
PVT	Primary Vent Tank
Rx	Reactor
SRM	Source Range Monitor
TOP	Temporary Operating Procedure
TP	Test Procedure
WRM	Wide Range Monitor