SAN ONOFRE NUCLEAR GENERATING STATION

SEMI-ANNUAL OPERATING REPORT NO. 10

For the Period Including

January 1, 1972 to June 30, 1972

Submitted in Accordance With:

Operating License No. DPR-13
Paragraph 3.C (4)

Submitted By

Southern California Edison Company

San Diego Gas & Electric Company

I. HOURS OF USE AND ELECTRICAL OUTPUT

Gross generation of the San Onofre Nuclear Generating Station amounted to 1,223,400,000 KWH during the report period. The generator was on the line 2844.43 hours and the reactor was critical for 2921.53 hours.

II. SHUTDOWNS

72-1 (71-14)

At the beginning of 1972, the unit was off line for the second refueling. During the reporting period, the generator was out of service 1312.5 hours and the reactor subcritical for 1255.71 hours. The unit was returned to the line February 24.

72-2

On February 25, the unit was removed from service for balancing of the turbine shaft and mechanical overspeed trip testing. The unit was returned to service on February 26. The generator was off line 6.67 hours and the reactor subcritical for 2.92 hours.

72-3

On March 4, the unit was removed from service twice to conduct turbine overspeed tests and was returned to service the same day. The generator was off line 3.63 hours and the reactor subcritical 1.67 hours.

72-4

On March 24, the unit was removed from service to reset the turbine mechanical overspeed trip set point and it was returned to service the following day. The generator was off line 11.5 hours and the reactor subcritical 5.17 hours.

72-5

On April 29, the unit was removed from service to perform modifications to the main steam control valves. During the startup on April 30, the reactor and turbine were tripped on a high steam generator level signal. This incident is discussed in detail in letters to Dr. Peter A. Morris, dated May 9 and May 30. The unit was returned to service on May 1. The generator was off line 26.67 hours and the reactor subcritical 22.08 hours.

72-6

On May 18, the unit was removed from service to inspect the turbine control valves, complete reheater and condenser repairs and perform miscellaneous reactor plant maintenance. The generator was off line for 161.6 hours and the reactor was subcritical 157.92 hours.

III. RADIOACTIVITY LEVELS

A. Off-Site

The air and film badge samples as well as plant and animal samples from around the plant show no significant radiation levels above natural background.

B. On-Site

1. Radiation Surveys

The maximum radiation level measured during the report period was 125 R/hr on contact with the refueling cavity drain line.

Monitoring of the site perimeter air particulate samples showed no significant activity above natural background. The maximum level recorded during the reporting period was 7.76×10^{-12} µci/cc.

2. Contamination

Maximum surface contamination was 1.4×10^{7} dpm/ft^2 located in the reactor refueling cavity after refueling. The area was subsequently decontaminated.

3. Personnel Exposure

The maximum radiation exposure during the fourth quarter of 1971 was 320 mrem and during the first quarter of 1972, it was 2670 mrem.

IV. RADIOACTIVITY LEVELS IN THE REACTOR COOLANT SYSTEM

The maximum total gamma activity in the reactor coolant system was .244 µci/cc. The maximum tritium activity was 4.02 µci/cc.

V. ROUTINE RELEASES

The following radioactive waste disposals were made during the report period:

Type of Disposal	Amount	Activities	
		Curies, $\beta\gamma$	Curies, H^3
Liquid, gallons	1,086,142	12.668	1294.43
Gaseous, ft^3	613,932,264	11512.348	110.86
*Solid shipments, 55 gal drums	122	3.48	-
Resin, ft^3	200	72.0	-

*Includes nine 168 ft^3 wooden boxes.

An isotopic summary of the liquid and gaseous releases is listed below:

Isotope	Liquid	Gaseous
^{133}Xe	4.675	7181.4
^{133M}Xe	0	576.7
^{131M}Xe	3.861	3070.75
^{135}Xe	.0131	515.12
^{85}Kr	0	153.93
^{85M}Kr	0	14.35
^{51}Cr	.022	0
^{131}I	.376	3.82×10^{-5}
^{133}I	.019	0
$^{134-137}Cs$	3.122	4.27×10^{-4}
^{58}Co	.504	0
^{60}Co	.053	0
Unidentified	.018	.098
TOTAL	12.668	11512.348

VI. TESTING

At the completion of refueling in February 1972, hot zero power and full power physics tests were conducted. Measured rod worths and reactivity coefficients were in substantial agreement with predicted values. Hot full power tests were conducted in June at 2500 MWD/MTM. This data is still under evaluation.

On March 4, turbine overspeed tests were conducted to verify overspeed reductions accruing from modifications accomplished during the refueling. Although the overspeed reduction was less than predicted, the unit was released for operation at 450 MW_e on March 25 after the turbine mechanical overspeed trip setting was lowered to 102.3%.

The sphere integrated leak rate test conducted in December, 1971, indicated a 24 hour leak rate of .132%.

VII. PRINCIPLE DESIGN CHANGES AND MAINTENANCE

The following design changes were completed during this report period:

No.	Description	Safety Analysis Summary
70-13	Install a condensate flush line to vacuum pump inlet lines.	Equipment installed is not nuclear safety related.
70-37	Install ammonia stripping columns on air ejector.	Equipment installed is not nuclear safety related.

No.	Description	Safety Analysis Summary
71-03	Install additional waste gas compressor.	Increases system reliability. Equipment is equivalent to original design.
71-08	Install steam generator high level turbine trip.	Does not change plant operational procedures. Electrical circuitry conforms to existing standards. Parallel devices are powered from separate vital buses.
71-09	Install auxiliary salt water cooling pump.	Installation conforms to existing standards and improves system reliability.
71-10	Install manually operated backup valves for POV 9 & 10.	Identical valves installed. Piping conforms to original piping specifications.
71-16	Modify reheater drain and steam supply valves.	Equipment installed is not nuclear safety related.
71-19	Provide diesel generator "Out of Automatic" alarm.	Failure of equipment installed will in no way effect diesel generator operation.
71-20	Modify hotwell conductivity alarm presentation.	Equipment installed is not nuclear safety related.
71-25	Install volume control tank vent valve bypass.	Installation conforms with original specifications for VCT vent. Manual valve, NC, to be used for maintenance of normal vent valve.
71-26	Replace strong motion accelerograph system.	Equipment installed is not nuclear safety related.
71-27	Install speed sensing device on turbine.	Equipment installed is not nuclear safety related.
72-1	Modify Number 2 Hot Box.	Equipment installed is not nuclear safety related.
72-2	Modifications to reduce turbine overspeed.	Equipment installed is not nuclear safety related.
72-3	Modify CRDM cooling fan start circuit.	Improves system reliability. New temperature switch is from same manufacturer and of equivalent construction.

72-4	Install temperature monitoring system in circulating water conduits.	Equipment installed is not nuclear safety related.
72-5	Install modified UV attachments on SCRAM breakers.	Improved UV attachment recommended by manufacturer. Breaker reliability improved.
72-6	Install piping to send steam generator blowdown to outfall.	Equipment installed is not nuclear safety related.
72-7	Modify rod position recorder causing spare channels to fail up scale.	Equipment installed is not nuclear safety related.
72-8	Install drain valve on excess letdown system.	Valves meet present system specifications.
72-9	Replace existing rod control slave cycler connectors.	Improved connectors from original manufacturer installed.
72-10	Replace existing backup overspeed device with a CFF relay.	Replacement relay utilized in separate applications has exhibited superior reliability to existing device.
72-11	Modify turbine governor valve test circuitry.	Equipment installed is not nuclear safety related.
72-12	Replace reactor coolant piping insulation.	Replacement insulation meets specifications for original insulation.
72-13	Modify fuel assemblies for region 5 fuel.	Small reduction in flow from modified assemblies is within design limits.
72-15	Replace steam generator manway cap screws with studs and bolts.	Studs and bolts are equivalent in design to cap screws.
72-17	Install Radwaste/Heating and Ventilation Alarm panel in the Control Room.	Equipment installed is not nuclear safety related.
72-22	Replace main feedwater regulating valve positioners with pneumatic positioners.	Replacement positioners utilized in other applications have exhibited superior reliability to existing positioners.

Other significant maintenance items accomplished during this period were:

1. The second refueling of the reactor was accomplished during January and February. Fifty-two fuel assemblies, region 5, were placed in the core. Four of the plutonium enriched rods from fuel assembly D 51 X were removed and replaced with natural uranium fuel rods. The following significant events occurred during the refueling:

   a. Three of the reactor thermal shield flexures were inspected by borescope. Based on the satisfactory condition of the flexures, the normal refueling sequence was initiated.

   b. The test specimen surveillance capsule was removed and shipped off-site.

   c. One of the RCC drive shafts was replaced with a spare due to a bent shaft coupling assembly.

   d. With the exception of a bent thermocouple guide tube, no abnormalties were found during the TV inspection of the instrument package, core baffle attachment or RCC drive shafts and hubs.

   e. Eddy current testing on selected tubes of the "B" steam generator was accomplished. No abnormal conditions were detected.

   f. In service inspection of the following components:

      (1) All steam generator inlet and outlet nozzles.

      (2) All RCP outlet nozzles and the "A" pump inlet nozzle.

      (3) Pressurizer surge line welds and selected portions of the pressurizer vertical and circumferential welds.

      Minor surface cracks, porosity indications and slag marks were found and removed by grinding. Surface indications were found on the four "A" steam generator inlet nozzle alignment lug welds. Since these lugs now serve no useful purpose, they were removed from the inlet and outlet nozzles of all three steam generators.

      (4) The remote examination device (RED) was used to ultrasonically inspect the inside diameters of the reactor vessel outlet nozzles and the reactor vessel to flange weld. No abnormal indications were detected. In using the RED, two small Adiprene pads (1/8" thick by 1 1/2" diameter) were dropped from the device. One of the pads was located and recovered but the other was not found. It was determined that the material dissociates below operating temperature and pressure and is harmless in the reactor coolant system.

(5) Ultrasonic inspection was accomplished on the top and bottom of the reactor vessel head to flange weld and 14 of of the reactor head studs. No abnormal indications were found.

g. Magnetic particle and fluorescent penetrant checks were made on selected main steam relief header safety valve welds. No abnormal indications were found.

h. A failed fuel pin was discovered in each of two spent fuel assemblies. This incident is reported in detail in a letter to the Commission dated February 11, 1972.

2. During refueling, nuclear instrumentation source range channel detectors and thimble cables were replaced. In March, Channel 1201 detector and cables were replaced again.

3. The number one seal was replaced on reactor coolant pump C.

4. The nuclear instrumentation system intermediate and power range channel detector thimble cables were replaced.

5. Five reactor coolant system temperature detectors were replaced.

6. During the hydrostatic test of the reactor coolant system following refueling, an increase in activity of ORMS Channel 1216, Steam Generator Blowdown, was observed. Chemical analysis and visual investigation disclosed a leaking tube in Steam Generator "C". Eddy current testing verified leak location and the absence of leaks in adjacent tubes. The tube was explosively plugged.

7. The cryogenic waste gas treatment system installation was 95% installed.

8. The main generator hydrogen blower blading was repaired.

9. Twenty one blades in the first stationary row of blades on the generator end of the HP turbine were found missing. Due to the unavailability of replacement blades, the first row of stationary and rotating blades were machined off on the governor and generator ends of the HP turbine.

10. The lower bearing of the SE condensate pump was replaced.

11. All wearing members, the pump shaft and mechanical seals were replaced on the South Charging Pump. The pump motor was rewound and balanced.

12. New mechanical seals were installed on the North Charging Pump.

13. A new stem and cage assembly were installed on a pressurizer power relief valve.

14. Tube side leakage on the west reheaters was corrected.

15. Epoxy coating repairs to the inlet tube ends of the main condensers were completed. Approximately 750 inserts were installed in the inlet tube ends of the #4 condenser water box.

16. Fifteen spent fuel assemblies from region 1 were shipped off site.

HEM:dkm

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SAN ONOFRE NUCLEAR GENERATING STATION

SEMI-ANNUAL OPERATING REPORT NO. 11

FOR THE PERIOD INCLUDING

July 1, 1972, to December 31, 1972

Submitted in Accordance With:

Operating License No. DPR-13
Paragraph 3.C(4)

Submitted By

Southern California Edison Company

San Diego Gas & Electric Company

I. HOURS OF USE AND ELECTRICAL OUTPUT

The gross and net electrical generation of the San Onofre Nuclear Generating Station Unit 1 amounted to 1,741,800 MWH and 1,655,675 MWH respectively during the report period. Gross thermal power generated was 5,015,364 MW_t for the period. The generator was on the line 3,974.78 hours and the reactor was critical for 4,028.10 hours. Cycle III burnup at the end of the reporting period was 6320 MWD/MTM.

II. SHUTDOWNS AND INCIDENTS

A. Shutdowns

1. 72-7 On July 8, 1972, a slight increase in radioactive concentration on ORMS Channel 1216, Steam Generator Blowdown was noted. Results of chemical analysis indicated that a leak existed in the Steam Generator system and that it was located in "C" Steam Generator.

   Sampling was then conducted on a frequent basis and leak rates were calculated and plotted through July 18 at which time the leakage was approximately 100 gal/day.

   At 10:29 PM, July 19, 1972, the unit was removed from service and the reactor manually shutdown by control rod motion and placed in a cold shutdown condition to locate and repair the tube leak in "C" Steam Generator.

   The steam generator secondary side was tested at 800 psig and one leaking tube was identified. Eddy current testing was conducted on 819 tubes. The leaking tube and seven others were plugged and the steam generator was again hydrotested. Another tube was found to be leaking. It and eight other tubes were plugged. Other maintenance included repairing tube leaks on the west reheaters, repairing the south water screen and bar rake, installing inserts and epoxy coating the condensers.

   On July 27, 1972, at 7:50 AM, during the plant startup following the steam generator tube repairs, the main exciter motor failed mechanically when being started. The problem was determined to be rubbing between the rotor and stator. The unit was returned to service at 4:43 PM, July 28, 1972, utilizing a spare portable exciter. The generator was off line for 210.23 hours and the reactor subcritical for 170.58 hrs.

2. 72-8 On July 29, 1972, at 2:23 PM, the unit tripped at 450 MW_e gross from loss of main generator field.

   The unit had recently been returned to service using a spare portable exciter in place of the main motor driven unit which had failed during starting. The main exciter had been repaired and was to be test run. When the 4 KV

motor ACB was racked into its cubicle, the spare exciter field breaker tripped open causing the unit to trip on loss of field. A "B" pallet switch in the main exciter 4 KV ACB opens the main exciter field breaker in the event the 4 KV ACB is opened. All of the main exciter field breaker trips had been connected to the spare exciter field breaker during the operation of the spare exciter. The tripping of the field breaker was a normal trip function. Evaluation of the trip indicated that turbine overspeed had reached 133%.

An analysis of this overspeed value was concluded and a determination made that the 133% was a reasonable figure for overspeed with the conditions that existed. (This item was discussed in a letter to the Commission dated August 28, 1972.) The unit was returned to service at 4:45 PM the same day. The generator was off line for 2.34 hours and the reactor at hot shutdown for 0.88 hours.

3. 72-9 On July 29, 1972, at 11:07 PM, the unit was removed from service to connect the main exciter after repairs had been completed on the motor. The unit was returned to service at 4:48 AM, July 30, 1972. The generator was off line for 5.68 hours and the reactor remained at hot standby.

4. 72-10 During a routine weekly "Control Rod Exercise Test" on September 20, 1972, an automatic load limit runback was initiated from "Nuclear Dropped Rod" circuitry while withdrawing Shutdown Bank 2 Control Rods to the normal full out position. After load had been stablized at 300 MWe gross, an investigation was made to determine the cause of the runback. A print out of the rod position recorder indicated that four rods in Shutdown Bank 2 had slipped approximately 150 steps into the core. An incore thermocouple map was made which verified that the rods had slipped. The reactor was then manually tripped at 8:20 AM. The rod control circuitry was inspected and contactors and relays cleaned but no abnormal conditions were found. The control rods were then exercised in an effort to repeat the malfunction without success. The reactor was returned to criticality at 11:32 AM and the unit synchronized to the line at 12:30 PM, September 20, 1972.

The generator was off line for 4.17 hours and the reactor at hot shutdown for 3.20 hours.

5. 72-11 The reactor had been brought critical and the unit was placed on the line at 12:30 PM, September 20, 1972. At 12:32 PM, a high temperature alarm was received on the east main feed pump inboard motor bearing. The pump was stopped immediately. Investigation revealed that the pump and motor would have to be cleared for work to correct the high bearing temperature problem.

The unit was removed from service and the reactor tripped at 4:23 PM. While examining the motor bearing, excessive thrust bearing clearance was found in the pump. Subsequent repair required complete pump disassembly, inspection and replacement of the rotating element with a spare.

The unit was returned to service at 9:16 AM, September 23, 1972. The generator was off line for 65.23 hours and the reactor at hot shutdown for 62.60 hours. This item was discussed in a letter to the Commission dated September 29, 1972.

6. 72-12 On September 12, 1972, it was determined by chemical analysis that "A" Steam Generator was experiencing primary to secondary side leakage at the rate of 2.37 gallons per day. Sampling was then conducted daily until October 13, 1972, at which time the leakage was 121.7 gallons per day.

The unit was removed from service at 10:43 PM, October 13, 1972, for locating and repairing the tube leakage. The reactor was manually shutdown by control rod motion and placed in cold shutdown condition. One leaking tube was identified during the initial hydrostatic test. Eddy current testing was then performed on 127 adjacent tubes. No thinning beyond 50% was noted. The previously identified leaking tube was plugged and the steam generator successfully hydrostatic tested.

Major maintenance items during the shutdown included plugging reheater tubes, repairing the north bar rake, replacing #2 control valve servo motor relay plunger and installing 820 nylon inserts in the inlet side of the south B condenser.

The unit was returned to service at 9:14 AM, October 20, 1972. The generator was off line for 154.52 hours and the reactor subcritical for 151.63 hours.

B. Incidents

The following incidents occurred during the report period.

1. On Tuesday, July 18, 1972, at 1:25 PM, the No. 1 and No. 2 Diesel-Generators were started, synchronized to the No. 2 480 v bus and loaded to 200 KW each for the routine weekly test as per Operating Instruction S-2-11.

At 2:19 PM, the Control Operator observed the No. 1 and 2 Diesel-Generators apparently out of step with each other. The generators A.C. amps were taking large swings 180° apart. The operator shut down the No. 1, then the No. 2 diesel using the remote stop button which also separates the generators from the diesel bus by opening the 480 v ACB's B-10-1 and B-10-2. The operator then opened the tie ACB, B-10-4, de-energizing the diesel bus.

The No. 2 diesel-generator was started for test and would not develop voltage. The No. 1 diesel was started and tested satisfactorily as per Operating Instruction S-2-11. The No. 2 diesel-generator was found to have a broken wire (F2) in the exciter field (stator). The wire was repaired, the generator tested satisfactorily for operability and returned to standby

2. On Saturday, December 9, 1972, at 6:57 PM, an automatic load runback was initiated from nuclear dropped rod indications on power range channel 1207 with a load reduction to 280 MW_e gross.

   Number 3 vital bus transferred from the Number 3 inverter to its emergency power source. The transfer caused a momentary power loss to NIS Channel 1207, initiating a "Nuclear Dropped Rod" indication. Analysis of the thermocouple map revealed no significant deviations from normal and the unit load of 450 MW_e gross was restored at 8:10 PM that evening.

   Investigation revealed that the inverter low voltage power supply had failed. The power supply was replaced with a spare and the inverter returned to service at 11:31 PM the same evening.

III. RADIOACTIVITY LEVELS

A. Off-Site

The air and film badge samples as well as plant and animal samples from around the plant show no significant radiation levels above natural background.

B. On-Site

1. Radiation Surveys

   The maximum radiation level measured during the report period was 70 R/hr on contact with the reactor coolant filter. The filter was subsequently changed.

   Monitoring of the site perimeter air particulate samples showed no significant activity above natural background. The maximum level recorded during the reporting period was 2.26×10^{-12} µci/cc.

2. Contamination

   Maximum surface contamination was 4.03×10^6 dpm/ft^2 located on the spent fuel pit decontamination pad. The area was subsequently decontaminated.

3. Personnel Exposure

   The maximum radiation exposure during the second quarter of 1972 was 920 Mrem; during the third quarter of 1972, it was 500 Mrem.

IV. RADIOACTIVITY LEVELS IN THE REACTOR COOLANT SYSTEM

The maximum total gamma activity in the reactor coolant system was 1.92 μCi/cc. The maximum tritium activity was 4.26 μCi/cc.

V. ROUTINE RELEASES

The following radioactive waste disposals were made during the report period:

		Activities	
Type of Disposal	Amount	Curies, βγ	Curies, H^3
Liquid gallons	5,402,472	17.622	2183.11
Gaseous, $ft.^3$	660,371,601	7637.18	170.020
Solid shipments	Sixty-nine 55-gal drums and six 168 ft^3 wooden boxes	4.19	-
Resin, ft^3	None	-	-

An isotopic summary of the liquid and gaseous releases is listed below:

Isotope	Liquid, Ci	Gaseous, Ci
^{133}Xe	0.752	6079.46
^{133M}Xe	0	566.48
^{131M}Xe	7.726	752.08
^{135}Xe	0.044	33.09
^{85}Kr	0	185.01
^{85M}Kr	0	20.74
^{51}Cr	0.058	0
^{131}I	1.574	0
^{133}I	0.768	0
$^{134-137}Cs$	5.533	0
^{58}Co	1.020	0
^{60}Co	0.142	0
Unidentified	0.005	0.32
Total	17.622	7637.18

VI. TESTING

Power defect and xenon follow tests were conducted on July 19 and 20, 1972, from near full power to zero power. The measured results indicated a good correlation with predictions based on WCAP-7799.

VII. PRINCIPLE DESIGN CHANGES AND MAINTENANCE

The following design changes were completed during this report period:

No.	Description	Safety Analysis Summary
71-23	Install cryogenic gaseous radwaste system	Failure of this system will result in releases that do not exceed limits previously evaluated in the FERSA.
72-14	Install orifice-manometer type air meters on the air ejectors.	Failure of the meters does not affect air ejector operations. Flow continues to the same atmosphere vent system and is monitored in the same manner as the original installation.
72-21	Modify Switch S-501-C of ORMS Channel 1211.	The function of the equipment is not changed. Modification permits calibration at a higher scale.
72-23	Modify 220 KV Generator PCB closing circuits	The proposed change is a safety interlock to prevent equipment damage. It does not provide automatic synchronization but monitors normal manual synchronization procedures and can be bypassed in the event of failure.
72-26	Modify Switch S-501-C on ORMS 1212, 1215, 1216 & 1217.	The function of the equipment is not changed. Modification permits calibration at a higher scale.
72-27	Modify Oscillograph inputs to include rotor speed.	The proposed change utilizes an existing but unused tachometer generator. The output of this tachometer will be connected to a spare oscillograph point to provide a continuous rotor speed record.

Other significant maintenance items accomplished during this period were:

1. Repaired tubeside leakage on the east and west reheaters.
2. Lapped, reseated and tested pressurizer relief valve 532.
3. Repaired north and south bar rakes.
4. Installed inserts and epoxy repaired the main condenser.
5. Repaired a mechanical rub on the main exciter motor.
6. Adjusted the turbine control valves.
7. Replaced the mechanical seals on the north turbine plant cooling water pump.
8. Installed a seal water line to the auxiliary salt water pump.

9. Replaced the south waste gas compressor rupture disk.

10. Repaired the turbine generator voltage regulator.

11. Repaired west feedwater pump oil drive unit .

12. The following breakers were overhauled:

    a. Santiago East & West 220 KV
    b. Mission East 220 KV
    c. Chino East 220 KV
    d. Unit 1 East & West 220 KV
    e. 61 West 220 KV

13. Repaired the pressurizer spray valve controller.

14. Loop "C" cold leg RTD failed. Utilized a spare for operation.

15. Repaired steam generator "B" main feedwater regulating valve stem by welding connecting block threads.

16. Repaired NIS Channel 1202 by installing a new BF_3 detector.

17. Reseated auxiliary steam feed pump valves.

18. Replaced #2 control valve servo motor relay plunger.

19. Replaced plug and cage on pressurizer valve CV 545.

20. Replaced cam followers on sphere personnel hatch door.

21. Overhauled the center and north instrument air compressors.

22. Realigned the north primary makeup pump.

23. Replaced pump vanes on ORMS Channel 1215.

24. Replaced pump motor on ORMS Channel 1211.

25. Replaced pump on ORMS Channel 1212.

26. Repaired linkage to steam generator feedwater pressure recorder transmitter.

27. Repaired choke on #3 inverter.

28. Replaced pilot valve on east evaporator shell side level control.

29. Replaced diode and capacitor on CVI temperature controller.

30. Thirty seven spent fuel assemblies were shipped off site this period.

Corrections and additions to the Semi-Annual Operating Report No. 10, January 1, 1972, to June 30, 1972.

Liquid effluent gallons were reported as 1,086,142 gallons and should be corrected to 1,086,192 gallons.

The number of curies of gaseous ^{131M}Xe released was reported as 3070.75 and should be corrected to 3073.25.

The number of curies of gaseous ^{133M}Xe released was reported as 576.7 and should be corrected to 574.2.

KSF:dkm

REACTOR THERMAL POWER - 1972