# SAN ONOFRE NUCLEAR GENERATING STATION

## SEMIANNUAL OPERATING REPORT NO. 6

For The Period Including
January 1, 1970 To June 30, 1970

Submitted In Accordance With:

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

Submitted By:

Southern California Edison Company
San Diego Gas & Electric Company

July 10, 1970

## I HOURS OF USE AND ELECTRICAL OUTPUT

Gross generation of the San Onofre Nuclear Generating Station amounted to 1,799,400,000 kwh during this period. The generator was on the line 4083.05 hours and the reactor was critical for 4100.61 hours.

## II SHUTDOWNS

### 70-1

On February 11, 1970 the unit was taken off the line for a scheduled outage to conduct routine turbine tests, modify the reheater drain system and perform miscellaneous maintenance. The unit was returned to service on February 16, 1970. The generator was off the line for 102.72 hours and the reactor was subcritical for 98.28 hours.

### 70-2

On May 22, 1970 the unit was taken off the line for a scheduled outage to repair reheater tube leakage, modify the packing on the letdown isolation valve and the pressurizer spray valves and perform miscellaneous maintenance. During the startup on May 29, 1970, the operator noted that the turbine acceleration rate was faster than desired and manually tripped the turbine, thus causing a reactor trip. The unit was returned to the line at 11:37 a.m. the same day. Shortly after, an interstage drain line on the high pressure turbine was found leaking. The unit was removed from the line from 3:31 p.m. to 4:44 p.m. to repair the drain line. By 8:42 p.m. on May 29 the unit was loaded to 450 MWe. During this outage the unit was off the line for 157.23 hours and the reactor was subcritical for 144.11 hours.

## III RADIOACTIVITY LEVELS

### A. Off-Site

The radiation monitoring program continues with film badges placed at 10 locations and the use of two off-site high volume air samplers.

The results of the monitoring program during this reporting period have shown no significant radiation levels above background for any of the air or film badge samples collected.

None of the environmental monitoring programs, plant or animal samples, have shown any significant activity level above background.

### B. On-Site

1. Radiation Surveys

The maximum level during this period was 200 R/HR on contact with the north reactor coolant demineralizer. A level of 30 R/HR was measured on contact with the reactor coolant filter.

Monitoring of the site perimeter air particulate samples showed no significant activity above background. The maximum level indicated during this period was $8.3 \times 10^{-13}$ µci/cc measured in June, 1970.

2. Contamination

No airborne contamination problems were encountered during this reporting period.

Maximum contamination was 899,000 dpm/ft$^2$ in the northeast corner of the charging pump room. The area was decontaminated to clean area limits.

3. Personnel Exposure

The maximum individual exposure during any one month was 640 mrem. Although the June film badge report has not been received, exposures as indicated from dosimeter readings are not expected to exceed the reported high.

IV <u>RADIOACTIVITY LEVELS IN PRINCIPLE SYSTEMS</u>

The maximum total gamma activity in the reactor coolant system was 1.91 µci/cc. The maximum activity due to tritium was 3.73 µci/cc.

A small tube leak in the "B" steam generator has resulted in measurable activity in the steam generator blowdown and feedwater samples. Tritium analyses of the feedwater samples indicated a maximum of $1.0 \times 10^{-3}$ µci/cc. Steam generator "B" blowdown analyses indicated a maximum of $1.2 \times 10^{-3}$ µci/cc Tritium and $5.23 \times 10^{-4}$ µci/cc Iodine.

The highest activity level detected in the component cooling water system was $1.05 \times 10^{-4}$ µci/cc.

The refueling water storage tank contained a maximum activity level of $2.05 \times 10^{-6}$ µci/cc.

V <u>ROUTINE RELEASES</u>

The following radioactive waste disposals were made during this period:

| Type of Disposal | | Activity (Beta-Gamma) Curies | Activity (Tritium) Curies |
|---|---|---|---|
| Liquid Releases | 3,174,395* gallons | 1.8123 | 2742.1091 |
| Gaseous Releases | 2,453,664 $ft^3$ | 848.122 | 6.21 |
| Solid Shipments | 47 55 gallon drums | .0594 | ---- |
| Demineralizer Resin Shipment | 75 $ft^3$ | 2.00 | ---- |

* Includes turbine plant leakage and steam generator blowdown after the "B" steam generator tube leak occurred. This source contributed 2,801,451 gallons, 0.0612 curies $\beta$-$\gamma$, and 3.8026 curies $H^3$ to the total liquid releases.

## VI TESTING

An improved boron depletion curve for predicting boron concentration versus Cycle 1 burnup was adopted in March, 1970. Reactivity follow data exhibited a gradual departure above the original prediction after approximately 50% of Cycle 1 lifetime. Sources of the discrepancy were considered, including both experimental error associated with the measurements and correction factors and analytical error associated with the prediction. The study indicated that although some error is associated with measurements and correction factors, sufficient data existed to rule this out as a major source of the discrepancy. Subsequently, the prediction, which was made prior to startup, was examined with respect to current technology for reactivity depletion calculations and consistency with data from other operating PWRs. As a result of this investigation, the original prediction was revised utilizing more recent technology in the areas of plutonium cross sections, the representation of fission products and neutron spectrum variation with burnup. Reactivity follow comparison to the revised curve since its adoption has been good.

Core physics parameters were measured at full power on March 21 and 22 and April 6 and 8, 1970 and at hot zero power on May 29, 1970. The following parameters were measured:

Hot zero power

1. Differential rod worth, control bank 2, from 300 steps to 60 steps
2. Boron coefficient
3. Temperature coefficient

Full power

1. Differential rod worth, control bank 2, from 325 steps to 295 steps
2. Boron coefficient

3. Temperature coefficient
4. Power coefficient

The results of both tests indicated no significant deviations from predicted values. Utilization of the full power results for reactivity follow corrections has resulted in more consistent data, especially during transient conditions.

A turbine plant performance test was conducted on March 21 and 22, 1970 at five load points to determine an operating heat rate curve for fuel contract calculations. The results verified the routine monthly performance test results and were appreciably better than the design curve. This is attributed to better than design heat transfer across the steam generators which results in higher secondary plant efficiency. Measurement of the control rod insertion limit was accomplished on May 22, 1970. The insertion limit curve was updated and the corresponding changes were made in the shutdown margin computer.

## VII PRINCIPAL DESIGN CHANGES AND MAINTENANCE

The following design changes were accomplished or were in the process of being accomplished during this reporting period:

| No. | Item | Purpose |
|---|---|---|
| 70-1 | Relocation of the normal suction of the refueling water filter pump. | Prevent recirculation of radioactive water into refueling water storage tank. |
| 70-2 | Install a hose station on the primary water makeup line to the pressurizer relief tank. | Provide primary water for washdown of the reactor cavity liner and refueling equipment. |
| 70-3 | Install a charging pump auto start alarm | Prevent possible charging pump damage by alerting operator immediately. |
| 70-4 | Modify drain line on moisture separator reheaters. | To assure excess water does not accumulate in reheater shell. |
| 70-5 | Install valve in air ejector vent line. | To provide a backpressure to stabilize the air ejector flow meter. |
| 70-6 | Install shutoff valves in turbine first point extraction drains. | To permit removal of piping and orifices for maintenance. |

| No. | Item | Purpose |
|---|---|---|
| 70-7 | Install drain traps on condenser air removal lines. | To drain accumulated condensate and prevent restrictions in the air removal lines. |
| 70-8 | Install three Pyr-A-Larms in the HP turbine housing. | Assure prompt detection of turbine control oil fires. |
| 70-9 | Install a fire hose connection in the containment sphere. | To permit use of a fire hose within the sphere. |
| 70-10 | Modify the backup over-speed protection relay. | Provide greater reliability, set point stability and eliminate circuit interference caused by testing the speed recorder. |
| 70-11 | Install LP drain on the steam seal spillover dump line to the fifth point extraction line. | To drain accumulated condensate and prevent water hammer in the northwest fifth point extraction line. |
| 70-12 | Install a check valve in the distilled water make-up line to the condensate storage tank. | To prevent lower quality condensate water from contaminating the primary water storage tank. |
| 70-13 | Install a condensate flush line to the vacuum pump inlet lines. | To prevent seawater from entering the hotwells. |
| 70-15 | Rewire 15 Darling valves in the safety injection system. | To avoid excessive stressing of yoke leg studs during closing operation of the valves. |
| 70-16 | Install an oil demister in the existing turbine lube oil reservoir vapor extractor discharge system | Prevent lube oil discharge from the vapor extractor. |
| 70-17 | Relocate and replace a portion of the radwaste discharge piping. | To reduce radiation levels to uncontrolled area limits and reduce crud buildup in the piping. |

| No. | Item | Purpose |
|---|---|---|
| 70-18 | Install ring tracks around the OD of the six reactor vessel nozzles. | To permit accurate ultrasonic inspection of the safe end welds. |
| 70-19 | Install improved valve bonnets on PCV-430C, 430H and LCV-1112. | To allow more space for packing to reduce stem leakage. |

Other significant maintenance items accomplished during this period are listed below:

1. The north charging pump shaft failed on January 7, 1970. A new shaft and wearing components of an improved material were installed.

2. Thirty-five LVDTs of an improved design were installed during the May, 1970 outage.

3. Shielding was installed around the reactor coolant filter. This reduced the contact radiation level of 30 REM/HR to 175 MREM/HR on contact with the shielding.

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# SAN ONOFRE NUCLEAR GENERATING STATION

## SEMIANNUAL OPERATING REPORT NO. 7

For The Period Including
July 1, 1970 To December 31, 1970

Submitted In Accordance With:

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

Submitted By:

Southern California Edison Company
San Diego Gas & Electric Company

January 15, 1971

## I. HOURS OF USE AND ELECTRICAL OUTPUT

Gross generation of the San Onofre Nuclear Generating Station amounted to 1,406,400,000 KWH during this period. The generator was on line 3187.37 hours and the reactor was critical for 3228.12 hours.

## II. SHUTDOWNS

### 70-3

On October 2, 1970 the unit was removed from service for refueling, in-service inspection, maintenance, and relocation of the existing 220 KV & 138 KV switchyards. The unit was returned to service on November 20, 1970. The generator was off the line for 1156.98 hours and the reactor remained subcritical for 1119.71 hours.

### 70-4

On November 23, 1970 the unit was removed from service to complete relocation of the 220 KV switchyard. The unit was returned to service on November 26, 1970. During this outage, the generator was off the line for 72.65 hours and the reactor remained subcritical for 69.17 hours.

## III. RADIOACTIVITY LEVELS

### A. Off-Site

The radiation monitoring program continues with film badges placed at 11 locations and the use of two off-site high volume air samplers.

The results of the monitoring program during this reporting period have shown no significant radiation levels above background for any of the air or film badge samples collected.

None of the environmental monitoring programs, plant or animal samples, have shown any significant activity level above background.

### B. On-Site

1. Radiation Surveys

The maximum radiation level measured during this period was 600 R/HR on contact with the south purification demineralizer. A level of 30 R/HR was measured on contact with the reactor coolant filter.

Monitoring of the site perimeter air particulate samples showed no significant activity above background. The maximum level indicated during this period was $6.7 \times 10^{-13}$ μci/cc measured in September, 1970.

2. Contamination

No airborne contamination problems were encountered during this reporting period.

Maximum contamination was $1.713 \times 10^7$ dpm/ft$^2$ located in the reactor cavity after the borated water was pumped back to the refueling water storage tank at the conclusion of refueling. The cavity was decontaminated to approximately 100,000 dpm/ft$^2$ in most areas.

3. Personnel Exposure

The maximum individual exposure during any one month was 2030 mrem and during either quarter was 2150 mrem.

Twenty-seven personnel received more than 1250 mrem during the final quarter of 1970 due to refueling outage activities; however, none exceeded the 5(N-18) accumulated lifetime dose limit.

## IV. RADIOACTIVITY LEVELS IN PRINCIPLE SYSTEMS

The maximum total gamma activity in the reactor coolant system was 2.58 μci/cc. The maximum activity due to Tritium was 3.20 μci/cc.

The steam generator "B" tube leak reported for the previous period continued until repairs were accomplished during the refueling. Prior to the repair, Tritium analyses of the feedwater samples indicated a maximum of $2.55 \times 10^{-3}$ μci/cc and steam generator "B" blowdown analyses indicated a maximum of $2.62 \times 10^{-3}$ μci/cc Tritium and $6.29 \times 10^{-4}$ μci/cc Iodine.

Maximum activity levels in other systems are listed below:

| System | Activity μci/cc |
|---|---|
| Component Cooling Water | $4.20 \times 10^{-4}$ |
| Refueling Water Storage Tank | $4.57 \times 10^{-3}$ |
| Spent Fuel Pit | $1.58 \times 10^{-2}$ |

## V. ROUTINE RELEASES

The following radioactive waste disposals were made during this period:

| Type of Disposal | Amount | Activity (Beta-Gamma) Curies | Activity (Tritium) Curies |
|---|---|---|---|
| Liquid Releases, gallons* | 704,689 | 1.9491 | 2027.27 |
| Gaseous Releases, cu. ft. | 132,055,602 | 758.23 | 4.5649 |
| Solid Shipments, 55 gal. drums** | 138 | 8.41 | - |

* In addition to the above, 1,705,310 gallons containing 0.0504 Ci $\beta,\gamma$ and 10.134 Ci Tritium activity were released in steam generator blowdowns and turbine plant leakage.

** Three packages of wood (22 cu. ft.) were also shipped. The activity in these packages is included in the solid waste total.

## VI. TESTING

Cycle I core physics parameters were measured at full power on September 23 and 29, 1970 and at hot zero power on October 3, 1970. The following parameters were measured:

Hot zero power

1. Differential rod worth, control bank 2, from 300 to 75 steps.
2. Boron coefficient
3. Integral worth of all 45 rod cluster control assemblies.

Hot full power

1. Differential rod worth, operating band of control bank 2.
2. Temperature coefficient
3. Power coefficient
4. Boron coefficient

The results of both tests indicated no significant deviations from predicted values.

Cycle II core physics parameters were measured at hot zero power on November 18 and 19, 1970 and at full power on December 1 and 3, 1970. The following parameters were measured:

Hot zero power

1. All rods out critical boron concentration
2. Temperature coefficient
3. Differential rod worth of control banks 1 and 2.
4. Integral rod worth of shutdown banks 1 and 2.
5. Boron coefficient

Full power

1. Differential rod worth, operating band of control bank 2.
2. Temperature coefficient
3. Power coefficient
4. Boron coefficient

The results of both tests agree closely with the design predictions.

Results from the forced vibration testing accomplished on the principle components of the nuclear steam supply system in August, 1969 confirm that the system is conservatively designed and could withstand the effects of severe earthquakes without endangering the public. Natural frequencies of vibration were experimentally determined for the reactor coolant pumps, the pressurizer, primary coolant piping and the steam generators.

Accelerometers were not placed on the reactor vessel. However, it was inferred from measurements and data on other components that the natural frequency of the vessel was less than the calculated design value. Additional testing was accomplished in October, 1970 to obtain additional information on reactor vessel response. Data reduction for this testing is in progress.

In November, 1970 instrumentation was installed at selected locations on the Reactor Coolant System to measure the level and spectral nature of system noise during operation. Data will be taken periodically during full power operation by Battelle-Northwest personnel as part of the Edison Electric Institute program to evaluate acoustic emission techniques for continuous integrity surveillance of reactor pressure boundaries.

## VII. PRINCIPLE DESIGN CHANGES AND MAINTENANCE

The following design changes were completed during this reporting period:

| No. | Item | Purpose |
|---|---|---|
| 69-36 | Install emergency auxiliary power system (3 KVA inverter for telemetering to Dispatcher) in SDG&E 138 KV switchyard. | To assure continuous AC power to the telemetering equipment if the normal source is lost. |
| 69-38 | Reroute sphere equalizing line from below ground to above ground. | To eliminate possibility of moisture collection resulting in a "loop seal" in the equalizing line. |
| 70-20 | Increase the set point of the test pump relief valve (RV-259) from 2750 psig to 3000 psig. | To prevent lifting of the relief valve at normal operating pressure. |
| 70-22 | Install an annunciator window. Window should state auto rod withdrawal not reset. | To provide visual display of rod withdrawal permissive circuit in addition to the backlit push button presently provided. |

| No. | Item | Purpose |
| --- | --- | --- |
| 70-41 | Replace existing meteorological tower with a new 120 foot tower. | To meet current AEC criteria. |
| 70-42 | Replace primary water flush valve No. 336 and add one additional in-series shutoff valve. | To provide a positive primary water supply shutoff. |
| 70-43 | Add high/low limiters to control banks 1 and 2 shutdown margin computer circuitry. | To represent actual shutdown margin requirements for Cycle II. |

Work on the following design changes was in progress during this reporting period:

| No. | Item | Purpose |
| --- | --- | --- |
| 70-30 | Install a back flow preventer between the service water and domestic water systems. | Prevent service water from entering the domestic water system. |
| 70-31 | Install steel plugs in roof of reactor auxiliary building and piping penetration enclosure. | Provide access to the reactor coolant and seal water supply filters. |
| 70-36 | Install motor operators on the feedwater block valves. | Provide remote manual operation of the block valves. |

Final approval has been obtained for the following design changes; however, no work was accomplished during this reporting period:

| No. | Item | Purpose |
| --- | --- | --- |
| 70-23 | Add a control switch in each of the circulating water pump motor ACB cooling water POV circuits. | To open the POV's when the circ pumps are stopped |
| 70-25 | Revise radwaste effluent monitor R-1218 inlet and outlet piping and add inlet cooling system. | To reduce stray radiation pickup by monitor from large diameter inlet and outlet piping and to provide a cooling system for the gas stripper effluent. |
| 70-37 | Install ammonia stripping column on the air ejectors. | Avoid excessive ammonia in the condensate system. |

All design changes reported as accomplished or in progress in previous semi-annual reports have been completed except as noted below:

| No. | Item | Purpose | Status |
|---|---|---|---|
| 67-240 | Add monorail for aeroball system. | To remove concrete plug required for access to instrumentation. | Deleted |
| 69-51 | Install two permutit automatic gravity filters upstream of the service water pumps. Install a 4 inch gate valve in reservoir supply line and removal of the sand filters downstream of the domestic water tank. | To secure potable water for the domestic water system. | Work in progress. |
| 70-1 | Relocation of the normal suction of the refueling water filter pump. | Prevent recirculation of radioactive water into refueling water storage tank. | Work in progress. |
| 70-11 | Install L.P. drain on the steam seal spillover dump line to the fifth point extraction line. | To drain accumulated condensate and prevent water hammer in the northwest fifth point extraction line. | Approved- No work accomplished. |
| 70-13 | Install a condensate flush line to the vacuum pump inlet lines. | To prevent sea water from entering the hotwells. | Approved - No work accomplished. |

Other significant maintenance items accomplished during this period are listed below:

1. Lead shielding was installed around the spent fuel pit and refueling water tank recirculation filters in anticipation of high radiation levels after refueling.

2. The north charging pump was shutdown on September 10, 1970 due to an increase in vibration. An Ultrasonic test indicated that the pump shaft was cracked. A new shaft was installed and the pump was returned to service.

3. Four tubes were plugged in steam generator "B". Post refueling operation indicates that this repair has eliminated primary to secondary plant leakage.

4. The first refueling was accomplished during October and November, 1970. Fifty-two fuel assemblies were replaced with forty-eight uranium enriched stainless steel clad and four plutonium enriched zircalloy clad fuel assemblies.

5. A partial in-service and reactor internals inspection was accomplished as part of the refueling outage. Ultrasonic and dye penetrant testing was accomplished on the reactor vessel, steam generator and pressurizer nozzle outside diameters. Ultrasonic testing was accomplished on the reactor vessel to flange weld, reactor head to flange weld, steam generator support structure and reactor vessel studs. Borescope or periscope inspections were made on reactor thermal shield flexures, rod cluster control assemblies, fuel assemblies, source assemblies, flow mixers, control rod drive shafts, the core baffle attachment and the instrument package. No significant defects or abnormalities were found during any of the above tests or inspections.

6. One flow mixer and one control rod drive shaft were replaced due to damage incurred during refueling operations.

7. The valve stem was replaced and the stem nut was repaired on Safety Injection MOV 850B. The above damage was caused by improper torque settings on the MOV. Tests were performed to insure proper operation before the valve was returned to service.

8. An end bell and the stator were replaced on Safety Injection MOV 853A after damage caused by a broken rotor bearing support in the end bell.

9. The expansion joints in the ducts leading to the Reactor Cavity Cooling fans A-8, A-8S and A-8SS were replaced. This reduced leakage and thereby increased cooling air flow across the LVDTs.

10. Modifications were made to the boric acid heat tracing system (Design Change 70-35) to assure uniform temperatures in the piping system. These modifications included:

    a. Improvement of the separation of the temperature sensing element from the controller on the heat tracing circuits.
    b. Addition of heat tracing to four feet of the demineralized water lines.
    c. Separation of heat tracing for the south and north boric acid transfer pumps.

    This work was accomplished in lieu of installation of a small continuous recirculation pump as described in a letter from A. Arenal (Steam Generation Division of SCE) to L. D. Low (Division of Compliance of A.E.C.) dated 10-27-69. An extensive study indicated that the recirculation pump should not be installed for the following reasons:

a. The present system of recirculation using the boric acid transfer pumps for 10 minutes once each shift (three times per day) has proven to be satisfactory.
b. It would complicate the system unnecessarily and thus reduce its reliability.