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THE OMEGA WEST REACTOR: DESIGN BASIS AND PHYSICS MEASUREMENTS

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ABSTRACT

The Omega West Reactor (OWR) has been operated by the Los Alamos National Laboratory without accident or major operational incident since August, 1956. The OWR is, perhaps, one of the few remaining US reactors that was not built to a set of standards but was designed to the experience base of several reactors that had been operated during the late 1940's and early 1950's. In addition, physics parameters were measured during its lifetime in somewhat unusual but innovative ways. The design approach can be summarized as having the following attributes:

- Utilization of a well-tested and proven technology (e.g., MTR-type fuel);
- Use of extremely large safety factors (including those used for thermal hydraulic parameters, biological shield design, etc.);
- Design simplicity and component replaceability; and,
- Utilization of results of tests of similar cores to establish nominal physics parameters.

INTRODUCTION

The Omega West Reactor (OWR) has been operated by the Los Alamos National Laboratory without accident or major operational incident since August, 1956. Over the years, the OWR has provided a reliable source of neutrons for a variety of experimenters - ranging from weapons developers to environmental scientists. Although it was built primarily to serve the needs of Los Alamos staff, collaborations with universities and industry have flourished. Over the recent past, the primary mission for the reactor has evolved to one of isotope production and plans are now under development to devote the majority of operations in support of medical radioisotope production. Because of the OWR's original robust design, it is currently estimated that the reactor has some 10 - 15 years of productive life remaining.

The OWR is, perhaps, one of the few remaining US test and research reactors that was not built to a set of standards but was designed by Laboratory staff to a very conservative set of safety criteria. As is explained in sections below, the experience base of several reactors that had been operated during the late 1940's and early 1950's was utilized to develop the design. In addition, physics parameters have been measured in somewhat unusual but innovative ways that are also discussed below.

OWR DESCRIPTION

The OWR is a thermal, heterogeneous, closed tank-type test and research reactor that is light-water moderated and cooled (Figure 1). The core comprises a rectangular array of four rows (numbered 2-5) by nine columns (designated A-I) of fuel elements or in-core sample positions (Row 1 comprises a lead gamma-ray shield and Row 6 is the location of a beryllium reflector) Normal operations are at a steady state power of 8 MW utilizing either 31 or 33 fuel elements - allowing for up to five in-core sample positions. The reflector is made up of 21 beryllium blocks. In addition, two gamma ray shields, a

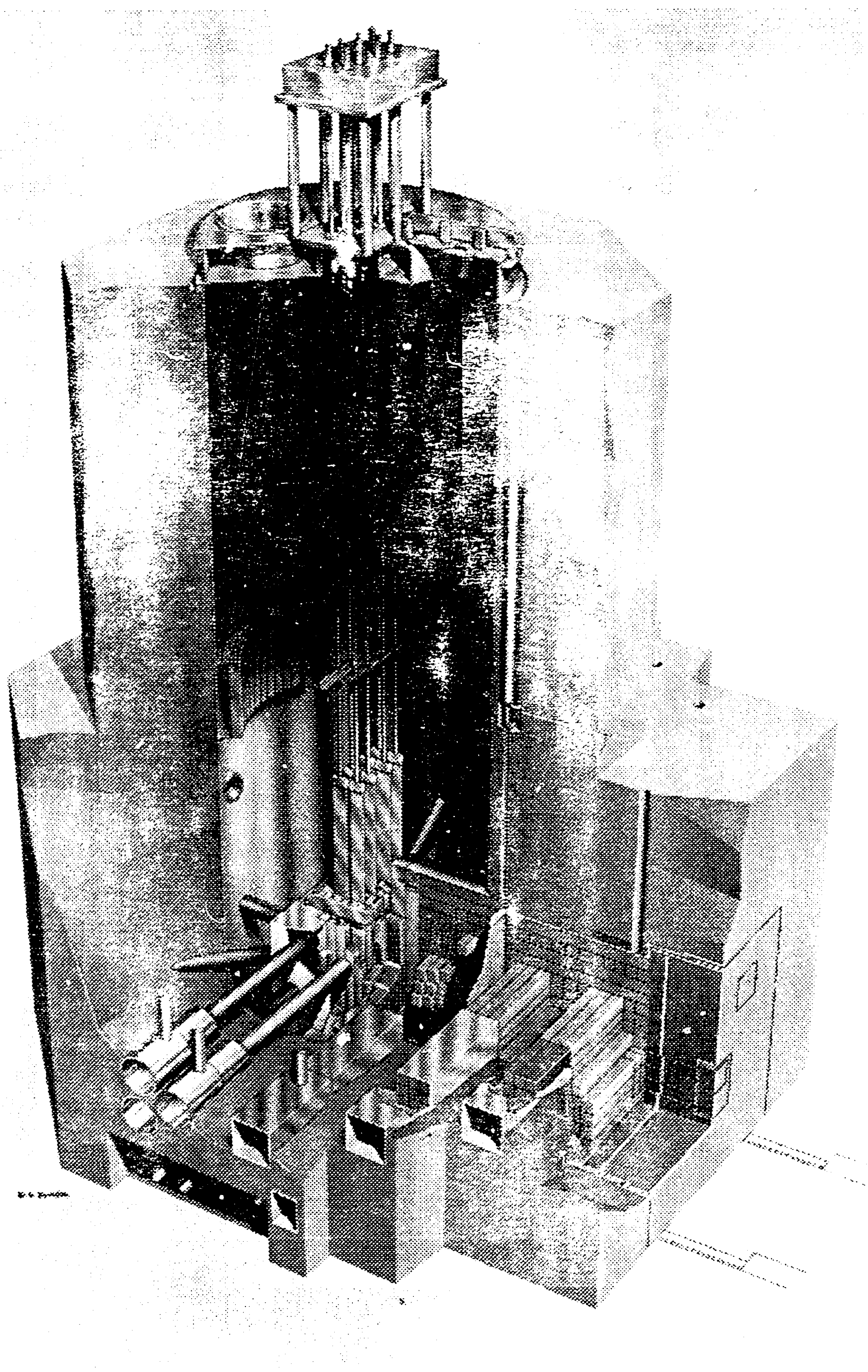

Figure 1 Omega West Reactor

5.7-m thick lead plate and a 12.7-cm thick bismuth shield (Figure 2), are located on the opposite side of the core and allow for experiments with a minimum of gamma-ray or fast-neutron interference to be conducted in a "thermal column" that is made up of stacked graphite within a shield.

Materials Testing Reactor (MTR) - type fuel elements are utilized in the OWR core with each element being made up of 18 or 19 aluminum-clad plates that contain highly-enriched U_3O_8 in an aluminum matrix within a layer of cladding on each side (Figure 3). The active portion of each element is about 0.625 m in length. However, the overall length of each element, including the aluminum end caps is 1.1 m. The core is supported by an aluminum grid structure that is located inside a 7.3 m-high, 2.4-m diameter, stainless steel reactor tank vessel. A biological shield of high-density concrete in an irregular octagonal shape surrounds the tank and thermal column. This shape was chosen to maximize the number of experiment ports available for research. Irradiation facilities at OWR comprise a total of nine six-inch beam ports, five in-core sample positions, sixteen pneumatic or hydraulic rabbit ports, and fifteen thermal column ports . Instrumentation ports extend underneath the core from a recess at the bottom of the south face.

Control is provided by eight blade-type poison rods. The rods are 3 meters in length and are made up of three sections: two aluminum end sections and a central 0.6-m long borated stainless steel section (1.28 wt% B_{nat}). The reactor is designed such that only one control blade may be moved at a time in the outward direction. In the automatic power control mode, however, the motion of the four middle control rods is "ganged" and their movement is controlled by small ac motors coupled to the main motors by a clutch and chain drive mechanism.

Cooling for the core is provided by light water that is circulated downward through and around the core at a rate of 13,250 liters per minute. More than half of this flow traverses directly through the core fuel elements while the remainder flows around the core. There is also a provision for operating the OWR, up to a power of 0.5 MW, in a natural convection mode. In this mode, the cooling water that is heated by the core, is forced upward by natural convection. It travels through a "flapper" valve that opens under its own weight whenever normal coolant flow is secured (Figure 4). The water then moves down through a U-tube pipe as it cools and then back into the core (note that this mode of cooling is also a defense against overheating of the core should forced flow be lost during normal operations).

The OWR was operated 24-hours per day, five-days per week from 1960 until 1972 when the present 8-hour per day, five-day per week scheduled was assumed. However, plans to produce the medical radioisotope Molybdenum-99 (in the fission product mode) will require that the reactor resume a 24-hour per day, seven-day per week schedule. Shutdowns for the removal of irradiated Mo-99 producing targets will occur on a periodic basis throughout the 24-hour day, thereby, providing opportunities for the production of other medical and industrial radioisotopes within the many ports available for the irradiation of materials. Because the reactor is in an unclassified area, it is readily accessible to non-Laboratory experimenters who have established formal user agreements.

DESIGN BASIS

Unlike modern reactors, the OWR was designed to an experience base comprising operations and measurements at other reactors such as the 3-MW Low Intensity Testing Reactor (LITR), the 40-MW MTR, the Boiling Reactor Experiment (BORAX), and the Special Reactor for the analysis of Transients (SPERT) series of experimental reactors. The MTR fuel design was extensively tested at LITR. This fuel design was then used in the 40-MW MTR in a core arrangement that was essentially duplicated in the OWR design except for the annular beryllium reflector that completely surrounded the MTR core. The original OWR fuel was identical to the MTR fuel both in blade coolant channel spacing (0.297 cm) and in uranium loading (168 gm/element). Extensive heat transfer, thermal hydraulics, and reactor dynamics data were gathered at MTR and analyzed by OWR designers to define OWR design values.

Both BORAX and SPERT reactor operations, which utilized the same MTR-type fuel elements in configurations that were very similar to both the MTR and the OWR cores, yielded considerable data on reactor transients and associated accidents. The results of one series of SPERT core tests in particular, the SPERT-1 A-17/28 core (Figure 5), were used to "establish" the nominal OWR physics parameters such as neutron lifetime, α_T, and α_V. Only later, were measurements performed to validate these values.

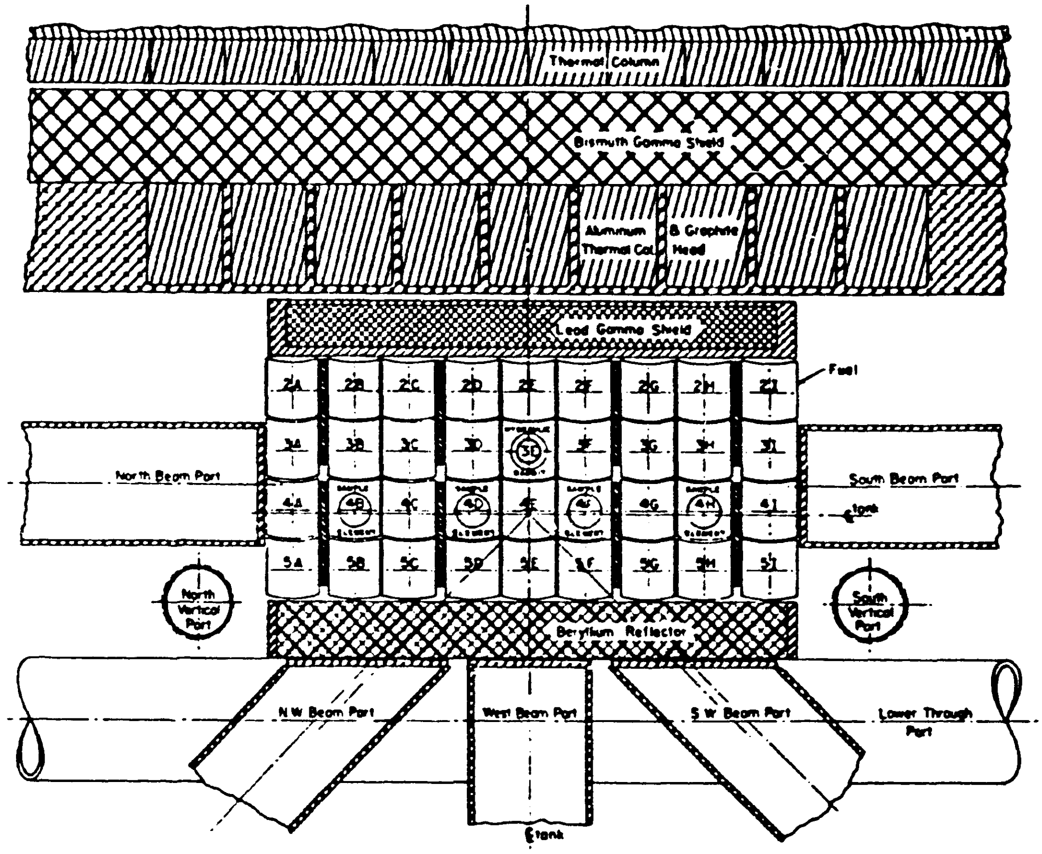

Figure 2 OWR Core Layout

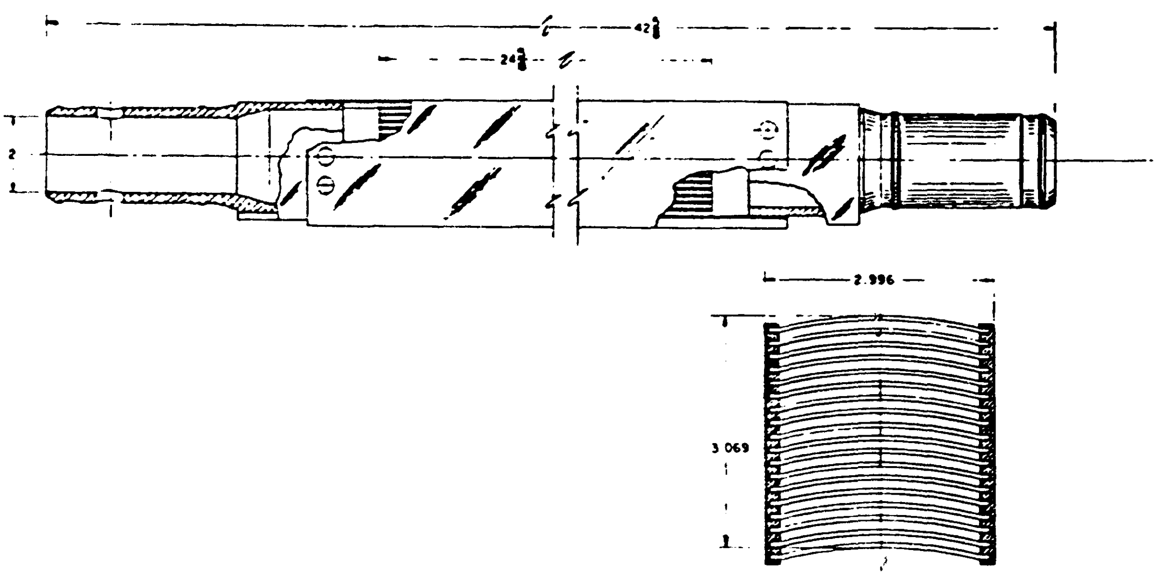

Figure 3 OWR Fuel Element

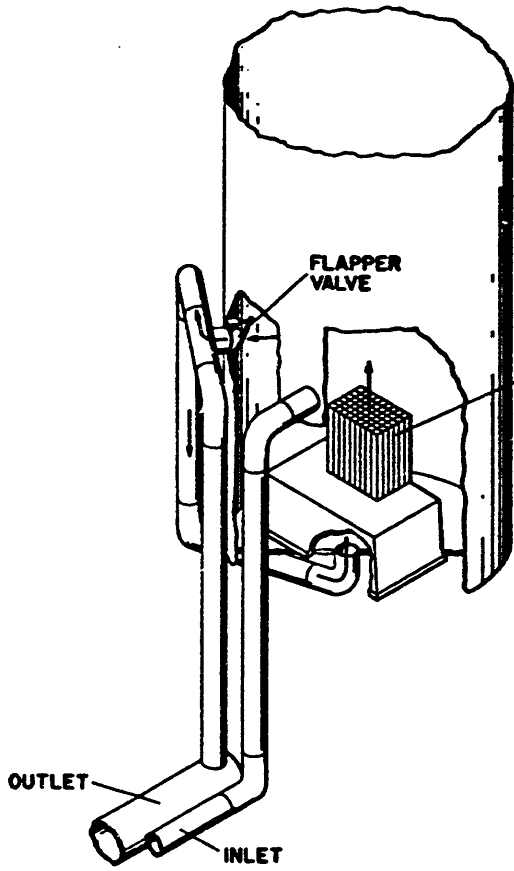

Figure 4 OWR convective Flow Loop

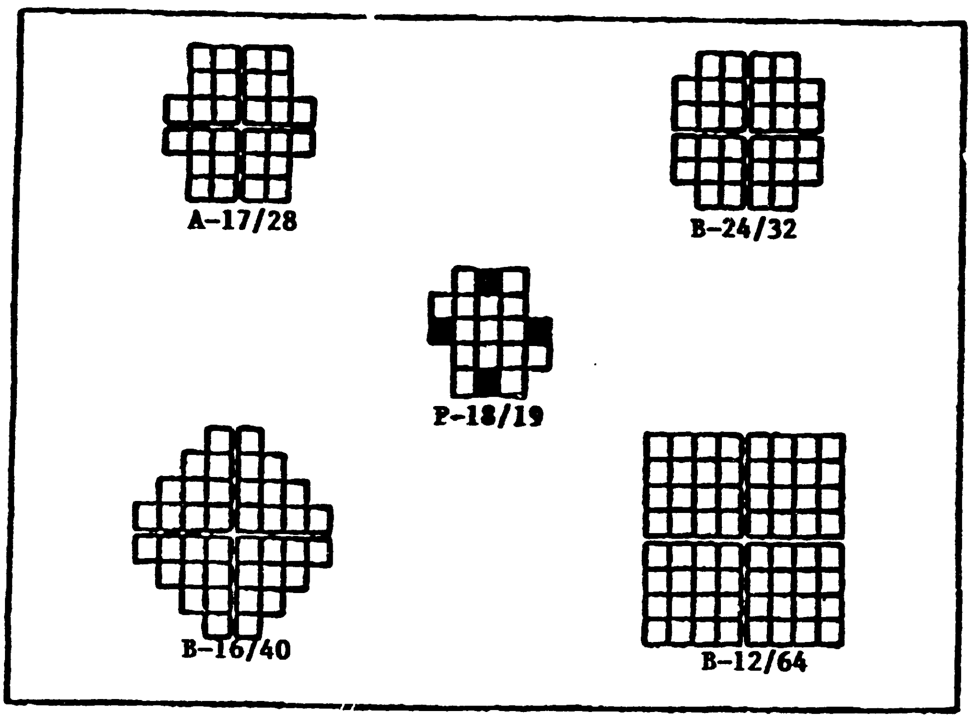

Figure 5 SPERT Core Configurations

Hence, the OWR design basis and basis for operation (Reference 1) were essentially defined by insuring, through technical specifications (TSs), that physical operating conditions never exceeded accident onset values for these type cores by extremely large safety factors. As an example of this design approach, reactivity insertion rates for the onset of core damage resulting in fission product release with MTR fuel were established to be about $100/sec. or for periods < 4 msec in BORAX and SPERT tests. To be absolutely certain that such values would never be approached in OWR operations, OWR TSs were written to specify that the period limiting safety system setting (scram setpoint) would be a factor of 1000 times greater than this threshold, i.e., 4 sec. In addition, operating procedures require that operators attempt to keep periods above 30 sec. Hence, the overall design approach can be summarized as having the following attributes:

- Utilization of a well-tested and proven technology (e.g., MTR-type fuel);
- Use of extremely large safety factors (including those used for thermal hydraulic parameters, biological shield design, etc.),
- Design simplicity and component replaceability; and,
- Utilization of results of tests of similar cores to establish nominal physics parameters.

As another example of the conservatism used to define operational limits for heat transfer parameters, TSs specify that the minimum critical heat flux ratio (CHFR) shall be maintained at a value ≥ 2.0. Typically, the CHFR is maintained at about 6 to 7 during normal, 8-MW operation. Heat transfer calculations show that if the OWR were to be operated at the power safety limit of 14 MW and the primary flow rate was allowed to drop to its limiting safety system setting of 11,730 l/min. the CHFR would only decrease to about 2.3.

EARLY EXPERIMENTS

The first two years of OWR operations were devoted to low-power testing and experimentation. It was necessary to methodically check all equipment and instrumentation and to determine the response characteristics of the reactor. In conversations with original OWR designers and operators, and as chronicled in operating logs, it is clear that the freedom that existed at that time to experiment with the reactor made for a delightful work environment and much experimentation was accomplished in the process. For example, in the first six months of operation, 57 different core configurations were taken critical! The following is a partial list of other early experiments performed at OWR:

- Subcritical multiplication measurements;
- Optimal nuclear instrumentation positioning;
- Control rod calibration by the period method;
- Testing of cadmium vs. boron-stainless steel control rods;
- Use of multiple core configurations to determine relative worths of core positions;
- Indium foil flux mapping;
- Reactivity worth measurements for sample holders;
- Thermal column head leakage measurements;
- Corrosion experiments;
- End-port gamma heating measurements; and,
- Reactor tank gas evolution measurements.

PHYSICS PARAMETER MEASUREMENTS

Several of the OWR core physics parameters mentioned in literature are actually based on empirical data taken during the operation of the reactors mentioned above. In fact, although no criticals experiments were ever performed for the OWR core before it was built, because it was fashioned closely after the SPERT-1 A-17/28 core, the A-17/28 core, for all practical purposes, comprised our "criticals experiment." Some of the physics parameters for that core are listed in Table 1.

Table 1
SPERT-1 A-17/28 Static Core Characteristics (Ref. 3)

Parameter	Value
Temp. Coefficient (¢/°C) @ 20 °C	-0.67
Moderator Void Coefficient	
Core Average (¢) Core Average (¢/cm^3) Central (¢/cm^3)	-25 -0.046 -0.093
l/β_{eff} (msec)	7

Interesting experiments have been conducted to demonstrate that the OWR core parameters are indeed within the range of those values noted above. Measurements of a gross temperature coefficient (α_T) were performed in 1962. In this measurement, temperature was measured as it was allowed to rise while the reactor was maintained critical at a constant power level of 10 kW via control blade positioning. The blade positions necessary to keep power constant at 10 kW were noted. The difference in rod positions as the temperature increased 11 °C was a positive 0.2032 cm, indicating a negative temperature coefficient. From a previous calibration run, the rod worth was known to be 4.96 x 10^{-3} $\Delta k/k$-cm Hence the core-averaged $\alpha_T = \partial r/\partial T$ was found to be -9.16 x 10^{-5} $\Delta k/k$°C, corresponding to about -1.4 ¢/°C (with a β of 0.0065), i.e., well within the value for the SPERT core.

An "average" α_T (over power) was also determined by noting the difference in rod positions and temperatures while adjusting power to just critical at both 10 kW and 2 MW. The difference in rod positions as the temperature increased about 12 °C was a positive 0.381 cm. Hence, with the same rod worth and β used above, the average temperature coefficient was found to be -1.57 x 10^{-4} $\Delta k/k$°C or about -2.4 ¢/°C.

To determine one value of α_V, a coefficient that is defined for specific locations in the core,. an innovative method was utilized to experimentally find α_V for a position just next to the core. For the SPERT reactor, nearly continuous curves of reactivity vs. moderator density were developed by uniformly displacing water within the fuel channels of the fuel elements with magnesium strips. By removing these strips, one by one, reactivity vs. moderator density curves were developed. This particular experiment would probably never be performed with the OWR core because of the potential for either fuel element water passage blockage or hot-spot formation on fuel plates.

The OWR is equipped with two "through ports" that extend from one side of the reactor to an opposite side for ease of usage with some experimental configurations. Both through ports are located on the same cross section of the reactor (North-South) and installed horizontally and inset approximately one inch into the outer surface of the Be reflector (Fig. 1). Because of their position, they are closely coupled to the core in terms of reactor neutronics. To determine the magnitude of this coupling, graphite plugs that are normally installed within the entire length of the ports, were removed from the lower through port to effect some repairs during December, 1956. The work presented an opportunity to perform an indirect measure of a void coefficient in that, once the graphite plugs were removed, the port could be flooded with water and then be drained to measure the effect on reactivity. The port comprises a relatively large volume just next to the core (about 12.97 liters). The experiment was performed with all but one of the control blades at identical heights. During drainage of the port, control blade Number 1 had to be pulled a total of 7.366 cm to keep the reactor at the same power level - indicating a negative void coefficient for this area. From a previous calibration run, it was known that the worth of the blade was 3.465 x 10^{-2} $\Delta k/k$-cm. Hence, using a β of 0.0065, the void coefficient was found to be -2.56 x 10^{-3} $\Delta k/k$ or -39.4 ¢.

FLUX MAPPING

A method used to predict the amount of U-235 burned out of each fuel element in the core required that the relative neutron flux be mapped each time the core configuration was changed appreciably, as in going from a 31-element to a 33-element array.

To perform this measurement, the relative neutron flux in a section of an element was determined by inserting a brass wire between two fuel plates in the center of the element. The reactor was then operated at a low power, typically 10 kW, for a time long enough to activate the copper in the wire sufficiently to obtain good counting statistics in a reasonable time period. The count rate of Cu-64 positron annihilation was then observed with a sodium iodide detector, from which the activation was inferred. The neutron flux at a place in the core is directly proportional to the activation of the segment of the wire which occupied that place during irradiation.

Because placing the wires in the core was rather time consuming, it was not unusual to map every element in the core. When the core configuration was symmetric about a line through column E, it was preferred to map only half of the core completely on the assumption that the shape of the flux in the opposing half would be similar. Four or five wires irradiated simultaneously in the opposing half served to supply points through which the flux curves must pass in that half.

Because the burnup in the entire element in each core position was of interest, and the flux wires only represented the flux in the center of the elements, some correction factors were needed for elements in core positions adjacent to water boundaries and the beryllium reflector. Experiments to ascertain these correction factors were performed separately.

The accuracy of the flux map was continuously checked against the actual U-235 burnup in spent fuel elements, as reported by the Idaho Chemical Processing Plant (ICPP), which reprocessed OWR fuel. Typical discrepancies between calculated burnup and that reported by ICPP were on the order of 2% or less.

Flux mapping equipment included the brass wires, aluminum wire guides, and a counting carriage. To insert a wire into a fuel element, the wire was first fit into the slotted end of a wire guide and then held in the center of the guide by a pin inserted next to the wire. The wire and guide assembly was lowered onto the top of a fuel element by holding the triangular bail of the wire with a latching hook on a long pole. The guide was seated on the fuel element end box and the wire allowed to slide under its own weight through the guide and down between the central fuel plates. The bottom of the wire was bent back on itself to prevent presenting a sharp edge which could scratch a fuel plate. Because OWR fuel element plates are curved, the wire was curved to conform to the fuel blade shape for proper alignment.

After an irradiation period of 10 to 15 minutes, the reactor was shut down and the wires removed from the core and hung on hooks in the reactor tank for a few hours to allow for decay of short-term activity prior to handling. The wires were then taken out of the tank and removed from the guides. The wires were counted using standard techniques, which involved placing a wire on a grooved rack on a motorized carriage that ran the wire past a sodium iodide detector. The counts were collected on a 400-channel analyzer, and the data were punched out on a paper tape. Data reduction was accomplished with the aid of a CDC-6600 computer program, which also incorporated the boundary correction factors noted above. The reduced data were graphed to show the axial flux shape for each fuel element, and the relative flux values were used in subsequent burnup calculations.

CONCLUSIONS

The design of the OWR has resulted in safe and reliable operations for nearly thirty eight years. Although the reactor is currently down for repair of some degraded primary system components, it is expected that the reactor may have some 10 - 15 years of useful life as a result of its robust design. Because little had been documented during the early 1950s with regard to standard reactor physics measuring techniques, innovative methods were utilized at OWR to validate core physics parameters that had only been estimated during the design process.

REFERENCES

1. H. T. WILLIAMS, O. W. STOPINSKI, J. L. YARNELL, et. al., "1969 Status Report on the Omega West Reactor, with Revised Safety Analysis," Los Alamos Safety Analysis Report LA-4192 (1969 - with six (6) addendums dating to October, 1987 and references therein).
2. A. H. SPANO and R. W. MILLER, "SPERT-1 Destructive Test Program Safety Analysis Report," IDO 16790, Phillips Petroleum Company, Atomic Energy Division (1962).
3. S. G. FORBES, F. L. BENTZEN, P. FRENCH, et. al., "Analysis of Self-Shutdown Behavior in the SPERT I Reactor, IDO-16528, Phillips Petroleum Company, Atomic Energy Division (1959).

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