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OVERVIEW OF SANDIA NATIONAL LABORATORIES PULSE NUCLEAR REACTORS

MASTER

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ABSTRACT

Sandia National Laboratories has designed, constructed and operated bare metal Godiva-type and pool-type pulse reactors since 1961. The reactor facilities were designed to support a wide spectrum of research, development, and testing activities associated with weapon and reactor systems.

I. INTRODUCTION

Sandia National Laboratories has operated pulse nuclear reactor research facilities [1] for the Department of Energy since 1961. The reactors include both pool-type reactors and bare metal Godiva-type reactors. The Godiva-type reactors are the Sandia Pulse Reactor (SPR), Sandia Pulse Reactor II (SPR-II) and Sandia Pulse Reactor III (SPR-III). Sandia Pulse Reactor IIIM (SPR-IIIM) has been designed and the fuel is being fabricated. The pool-type reactors are the Annular Core Pulse Reactor (ACPR) and the Annular Core Research Reactor (ACRR). The reactor facilities have been used for research and development of nuclear and nonnuclear weapon systems, advanced nuclear reactors, reactor safety, simulation sources, and energy related programs. In addition to operation of the facilities, Sandia staff have contributed to the development of pulse reactors in the areas of thermomechanics, reactor physics and kinetics, reactor fuels, robotics, and instrumentation and diagnostics.

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II. SANDIA PULSE REACTOR

The SPR is sometimes referred to as Godiva III, since it is similar to Godiva II, which was designed and operated by the Los Alamos National Laboratory. The SPR was operated at the Sandia National Laboratories from 1961 to 1967. The fuel material was cast uranium enriched to 93.2% in ^{235}U with a total mass of uranium of 57.2 kg. The reactor geometry was a right circular cylinder with a domed cap. The diameter of the cylinder was 17.8 cm and the overall height was 14.3 cm. Three uranium bolts fastened the assembly together and to the steel support plate. There were four cylindrical cavities in the bottom to accommodate the safety block, two control rods and the burst rod. Figure 1 shows a cutaway diagram of the fuel assembly.

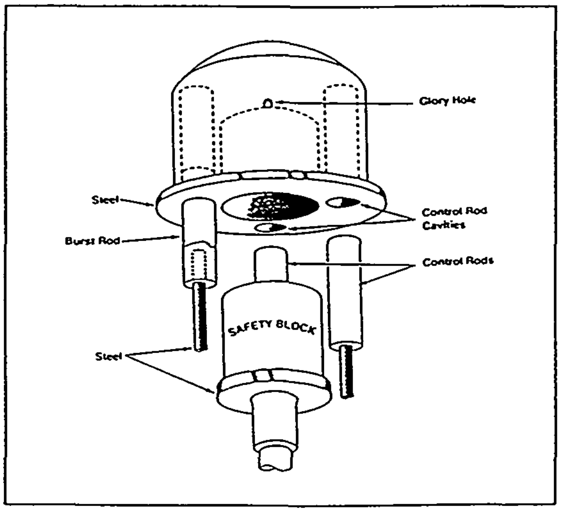

Figure 1. Diagram of SPR Fuel Assembly

DISCLAIMER

Compared to Godiva II, the SPR fuel components were machined to reduce the core reactivity and the control rod worth was increased to \$3 from \$2. The reactor stand was mounted on an hydraulically operated elevator that lowered the reactor into a shielded concrete pit immediately following an operation for rapid personnel entry. A through-hole, approximately 0.8 cm square, allowed for the irradiation of small objects. The fuel in SPR was cadmium plated rather than nickel plated. The cadmium provided oxidation reduction and decoupling from room return neutrons.

Figure 2 shows the SPR reactor stand in the operating position.

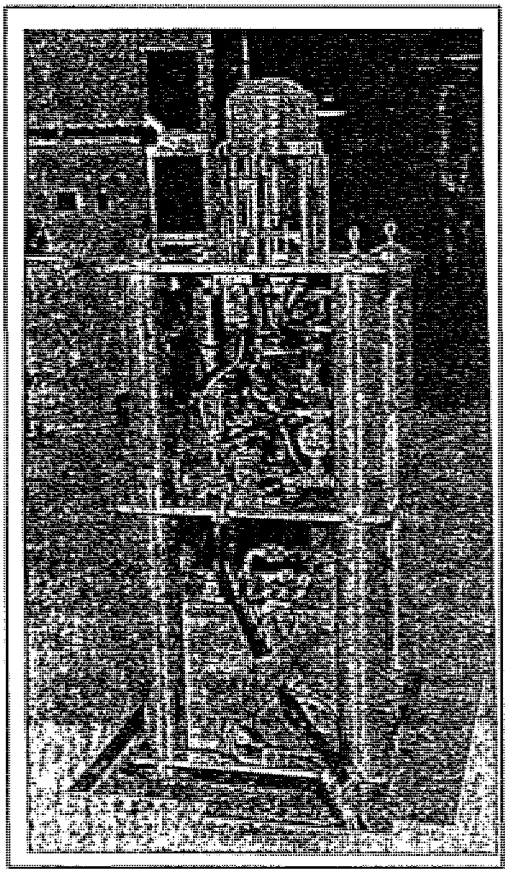

Figure 2. SPR reactor stand on elevator in reactor room

The reactor was designed for operation in a heavily shielded reactor building (Kiva) or at an outdoor site approximately 300 m away from the control room. The design burst for SPR was 2×10^{16} fissions which produced a 110°C fuel temperature rise.

III. SANDIA PULSE REACTOR II

SPR-II [2, 3] is a fast burst reactor, designed and constructed by Sandia National Laboratories, that became operational in 1967. The SPR-II core is solid-metal fuel enriched to 93 percent ^{235}U. The uranium is alloyed with 10 weight percent molybdenum to ensure the phase stabilization of the fuel. The core consists of six fuel plates divided into two assemblies of three plates each. The three lower plates are attached to an electromechanical drive mechanism. In the drive mechanism is an electromagnet such that for pulse operations with yields greater than 150°C, the shock forces break the magnetic coupling and rapidly drive the safety block down. Five vertical holes through the core assembly accommodate the irradiation cavity, three fuel control rods and one fuel burst rod. The burst rod is hydraulically driven to achieve the high rate of reactivity insertion required for pulse production. The primary shutdown mechanism is the inherent negative temperature coefficient of reactivity caused by the thermal expansion of the fuel. In operation, the reactor is surrounded by a shroud coated with a B-10 loaded silastic material. The shroud provides a flow channel for nitrogen cooling gas and the B-10 silastic decouples the core from backscattered neutrons. The reactor has a central cavity 3.8 cm in diameter and contains 105 kg of fuel alloy.

Figure 3 shows a cutaway diagram of the SPR-II fuel assembly with its square shroud.

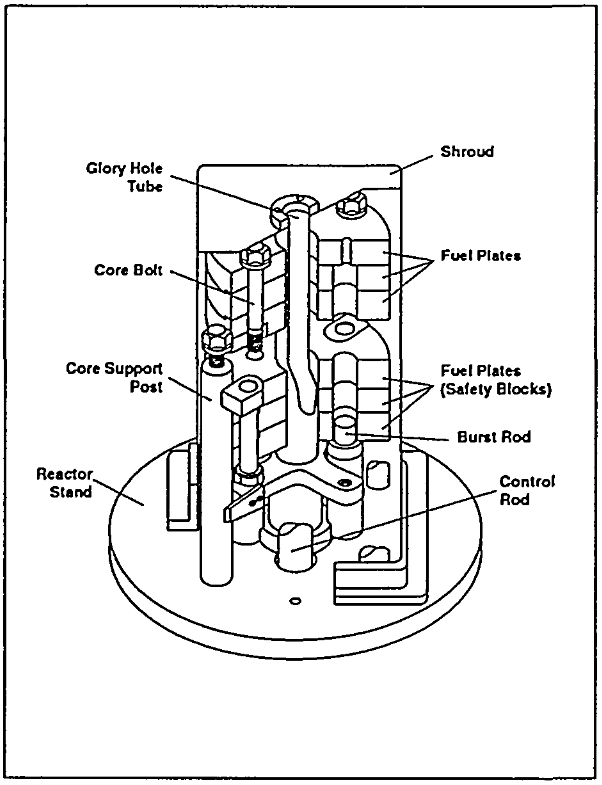

Figure 3. Cutaway diagram of the SPR-II Reactor with its decoupling shroud

The design goal for SPR-II was to produce a fast neutron fluence of 10^{15} n/cm^2 in a 3.8 cm diameter glory hole. The design fluence of 10^{15} n/cm^2 resulted in a temperature rise in the fuel of 560°C. Operation of the reactor at this yield or higher resulted in fuel plate and control rod cracking. The major improvements in SPR-II as compared to SPR were the central irradiation cavity and high yield pulsing. The fuel assembly was designed with 0.050 cm gaps between the fuel plates to reduce stress in the support bolts and to mitigate axial stress wave propagation. Subsequent modifications included slots in the fuel between the glory hole surface and the control rod holes to reduce the hoop stress on the inner surface where radial cracks had occurred during a large pulse.

A metallurgical [4] exam of the SPR-II cracked fuel revealed a brittle carbon dendritic phase at the grain boundaries which was a significant contributor to the fuel plate cracking. Figure 4 shows an SEM image of the U-10 Mo fuel plate preferentially etched to reveal the dendrites. The carbon impurity level in the plate was 650 ppm.

Figure 4. SEM image of a U-10 Mo SPR-II fuel plate showing carbon impurities at the grain boundaries

The SPR-II was designed to operate on the same elevator as SPR, and it has operated safely for extended periods of time with cracks in the fuel plates. The fuel in SPR-II was aluminum ion-plated for corrosion and oxidation control. The plating also acts as a fission product barrier to reduce contamination. Following the early core failures, the dry-air cooling system was replaced with a nitrogen cooling system using liquid nitrogen off-gassing. The timing of the pulse on SPR-II was controlled by use of an external pulsed deuterium-tritium neutron source in order to provide a fiducial for experiments requiring precise timing. Figure 5 shows the SPR-II reactor with the decoupling shroud removed.

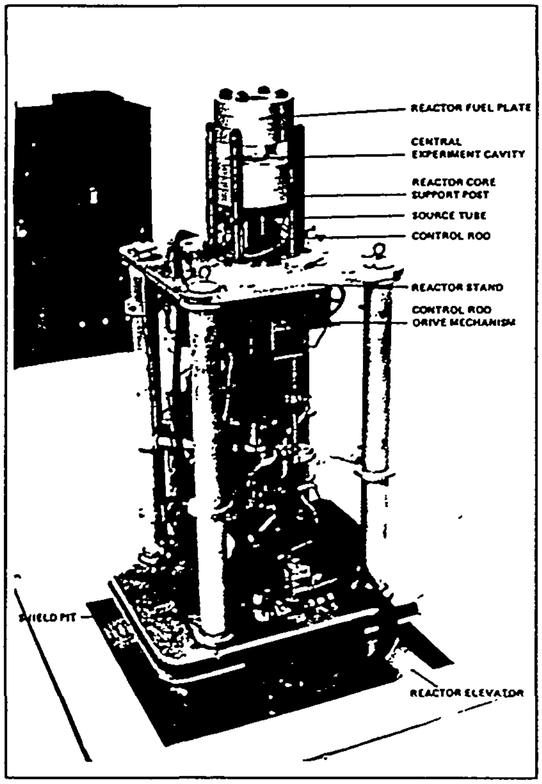

Figure 5. The SPR-II Reactor

IV. SANDIA PULSE REACTOR III

The SPR-III [5] is a fast-burst reactor, designed and constructed by the Sandia National Laboratories; it became operational in 1975. The SPR-III core consists of 18 fuel plates mechanically fastened into two halves of nine plates each. The fuel is 93 percent ^{235}U alloyed with 10 weight percent molybdenum to ensure phase stabilization. The nine upper plates are held stationary by the core support structure, while the nine lower plates are attached to an electromechanical drive mechanism. The plates are also aluminum coated similar to SPR-II. The total fuel alloy mass is 252 kg. Three reflector devices are used to control the reactor, while a fourth is the burst element. The burst element is electromagnetically driven to achieve the high rates of reactivity insertion needed for pulse production. The primary shutdown mechanism is the inherent negative temperature coefficient of reactivity caused by the thermal expansion of the fuel. In operation, SPR-III is covered by an aluminum shroud coated with ^{10}B that decouples the core from the backscattered neutrons. A 17-cm diameter central cavity extending through the core is the primary experiment cavity of SPR-III.

Figure 6 shows a cutaway view of the SPR-III reactor showing the top of the reactor table and the decoupling shroud.

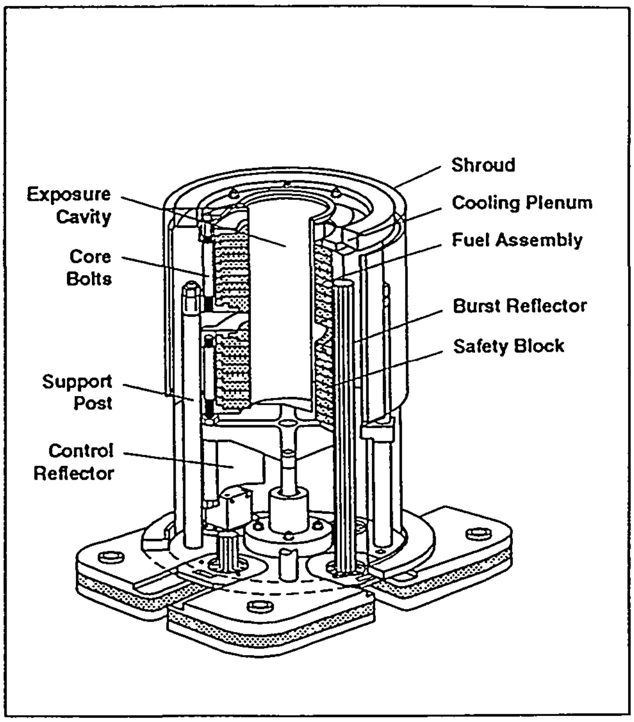

Figure 6. Cutaway view of the SPR-III fuel assembly and shroud.

The design goal for SPR-III was to produce a fast neutron fluence of 6×10^{14} n/cm^2 in the 17 cm diameter glory hole, which resulted in a temperature rise of 450°C. The fuel plates were designed with no holes in them since the core used external bolts and support plates. This eliminated a source of stress risers in the fuel. The plates were supported on a 1.27 cm rim on the outer radius with a 0.05 cm gap between the fuel plates near the center. The gaps eliminated axial stress wave propagation and reduced bolt stress.

There were no cracks in the fuel plates until 1993 when it was discovered that three fuel plates had cracked. The cause of the cracking was thought to be the result of fatigue and stress corrosion cracking. The reactor has continued operations with the cracked fuel.

The SPR-III reactor is operated in the same reactor room as SPR-II on an interchangeable basis. Figure 7 shows the SPR-III stand without its decoupling shroud.

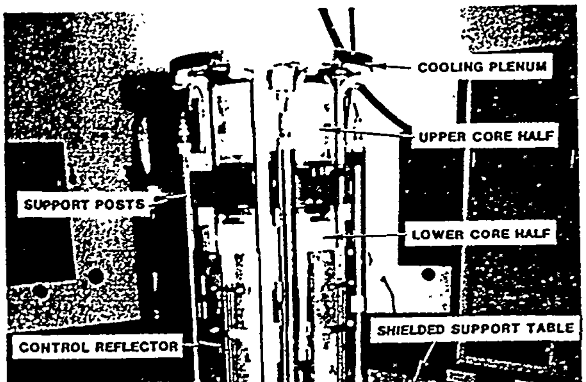

Figure 7. SPR-III reactor stand in operating position.

Prior to design of SPR-III, reflector control concepts were explored using tangential and normal reflector motion effected by linear or rotary drive systems [6]. A series of neutron transport calculations were performed to determine the effectiveness of various insertion techniques and materials as a function of thickness as shown in Figure 8.

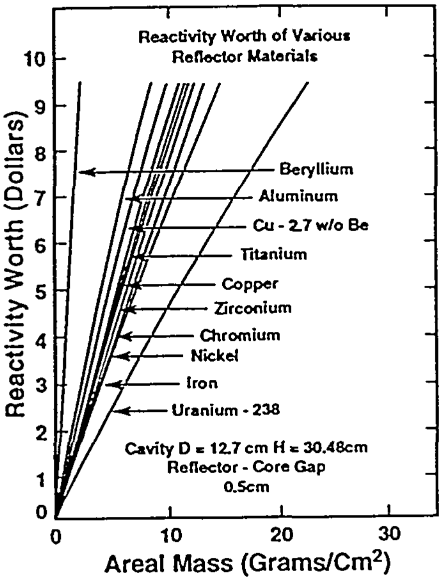

Figure 8. Reactivity worths of various reflectors for SPR-III-like reactor

It was determined that vertical linear motion adjacent to the core was preferred as it provided a typical " S" -shaped reactivity worth curve with $\Delta\rho = 0$ at the end of the stroke and it could be designed to incorporate a gravity-driven scram. Static experiments were then performed on SPR-II to benchmark the reflector reactivity calculations using reflectors of ^{238}U, copper, iron, lead, aluminum, and beryllium. Subsequent experiments [7] were performed to pulse SPR-II with an aluminum reflector assembly. There was no change in reactor pulse kinetics using the reflector for pulsing rather than the fuel pulse rod.

Extensive parametric neutron transport calculations were performed to determine pulse reactor critical dimensions, neutron spectra, fission and flux distributions, and prompt neutron lifetime using Monte Carlo [8] and discrete ordinates [9,10] techniques. Figure 9 gives the critical dimensions of an annular core with various cavity diameters as a function of core height.

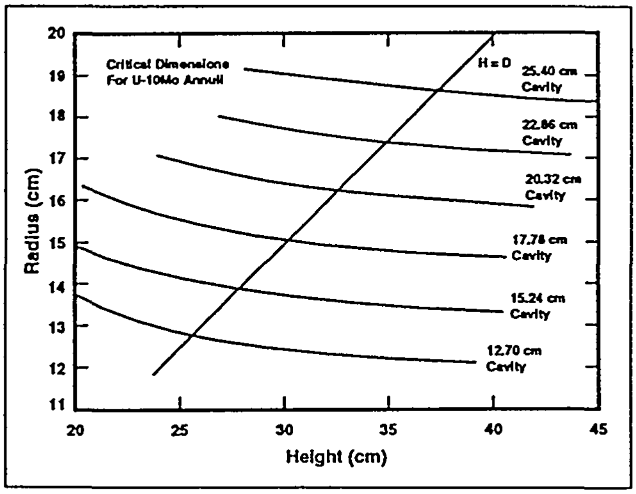

Figure 9. Critical dimensions for U-10 Mo annular cores

Numerical techniques for the analysis of fuel behavior under rapid fission heating conditions were developed. One-and two-dimensional methods were used for the design of SPR-III. A fuels development program was conducted using SPR-II to pulse fuel samples under conditions similar to those that would exist in SPR-III. These analyses are described by Reuscher [11] and Reuscher/Schmidt [12]; several different U-Mo alloys were tested for comparison [13]. Basically, a rod of alloy was placed in the central cavity of SPR-11. The rod was supported at the axial midpoint and displacement transducers measured the expansion of each free end. The yield stress was inferred from the oscillatory behavior of the rod and supporting calculations. The rod oscillated elastically at a frequency determined by the speed of sound in the alloy. The speed of sound is directly related to the modulus of elasticity of the alloy. The rod temperature rise was measured and a series of pulses produced information of expansion versus temperature. Calculations using measured material properties (modulus and expansion coefficient) were performed with a bilinear model of the modulus of elasticity for various yield stresses. These results are shown in reference [11], Figures 28 and 29. The calculated information was compared with experimental results to obtain the yield stress under pulse conditions.

A one-dimensional reactor (slab, cylinder, or sphere) coupled-kinetics stress analysis code was written by Reuscher [14] to provide a numerical method for evaluation of pulse reactor performance. The neutron, point kinetics equations are solved with displacement and stress component relations for a step insertion of reactivity. An experimental shutdown coefficient based on fuel expansion was used to provide negative feedback. The method of fuel support (free or fixed ends) was chosen. Pulse performance was calculated in terms of peak power, pulse width, total energy yield, temperature rise, fuel expansion, and stress.

V. SANDIA PULSE REACTOR IIIM

The SPR-IIIM is an improved version of SPR-III, which will go into operation in 1995. SPR-IIIM features a 19 cm diameter central cavity and many improvements in the fuel and mechanical design. The reactor is fully described in a paper by J. S. Philbin in this conference [15].

VI. ANNULAR CORE PULSE REACTOR

The ACPR [16] was a TRIGA-type reactor, which was designed and constructed by General Atomics for Sandia National Laboratories and was operated from 1967 to 1977. The cylindrical uranium-zirconium hydride ($U-ZrH_{1.625}$) fuel-moderator elements were arranged in a close-packed lattice. The uranium enrichment was 20 percent. The main features which distinguish this reactor from other TRIGA reactors were the large (23 cm-inside-diameter) dry irradiation cavity in the center of the core and the reactivity insertion of $4.50 in the pulse mode. The annular-shaped core was formed by 156 fuel-moderator elements arranged in a hexagonal grid around the central cavity. The core was located in an open pool 3.1 m in diameter and 8.5 m deep. The top of the core was approximately 7 m below the surface of the pool water, which afforded more than adequate radiation shielding. Access to the central irradiation cavity was by a dry, air-filled 25 cm diameter loading tube which extended vertically upward from the center of the core. The fuel-moderator material, 3.5 cm in diameter by 38 cm in length, was sealed in 0.5 mm thick stainless-steel cladding with a specially designed dimpled surface to maintain a gas-filled insulating gap between the fuel-moderator material and the cladding. The fuel-moderator material was designed to operate at temperatures up to 1000°C. The core was cooled by natural convection and controlled by six control rods and three transient rods.

VII. ANNULAR CORE RESEARCH REACTOR

The ACRR [17] is an upgrade based on the ACPR to provide larger pulse fluences and greater steady-state powers. The core geometry is unchanged. It became operational in 1978. The annular core is formed by 236 cylindrical fuel elements arranged in a hexagonal grid around a 23 cm, dry irradiation cavity. The core fuel elements are made of a uniquely designed BeO-UO_2, fuel material held by niobium liners in a stainless steel cladding. The fuel is uranium enriched to 35 percent ^{235}U, with 21.5 weight percent UO_2, and 78.5 weight percent BeO. The fuel elements are designed to allow steady-state and pulsed operation at fuel temperatures up to 1400°C. The core is cooled by natural convection. The reactor is controlled by two fuel-followed safety rods, three poison transient rods, and six fuel-followed control rods. The fuel-followed rods make up part of the 236 fuel elements in the normal core configuration. In addition to the internal irradiation cavity, two interchangeable, fuel-ringed, external cavities (FRECs), one 38 cm and the other 51 cm in diameter, are available. These cavities are surrounded by U-ZrH fuel elements. The ACRR can operate in the steady-state, normal-pulse, reduced-tail pulse, dual-pulse, or tailored-transient-rod-withdrawal (TRW) mode. In steady-state mode, the reactor is capable of any power level up to 2 MW for an extended period. The ACRR has operated at steady-state power levels up to 4.5 MW for reactor behavior studies. In the pulse mode, the ACRR produces a peak power of 30,000 MW.

Figure 10 shows a diagram of the core grid plate with control, safety, and transient rod locations.

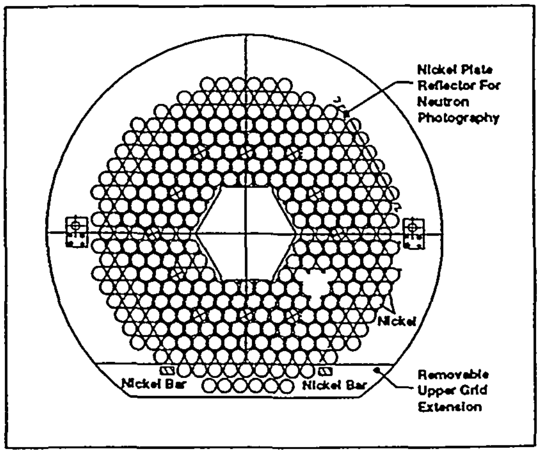

Figure 10. The grid plate diagram for the ACRR.

Figure 11 shows a cutaway of the ACRR facility with the 38 cm FREC installed and also showing the neutron radiography assembly.

During the design of ACRR, a fuels development program was conducted which examined U-ZrH, UO_2BeO, and UC-ZrC fuel types. The UO_2BeO fuel was selected because of its superior characteristics such as high melting temperature, phase change temperature, thermal conductivity, and heat capacity. An instrumented fuel element was developed to measure temperature reliably up to 1800°C [18].

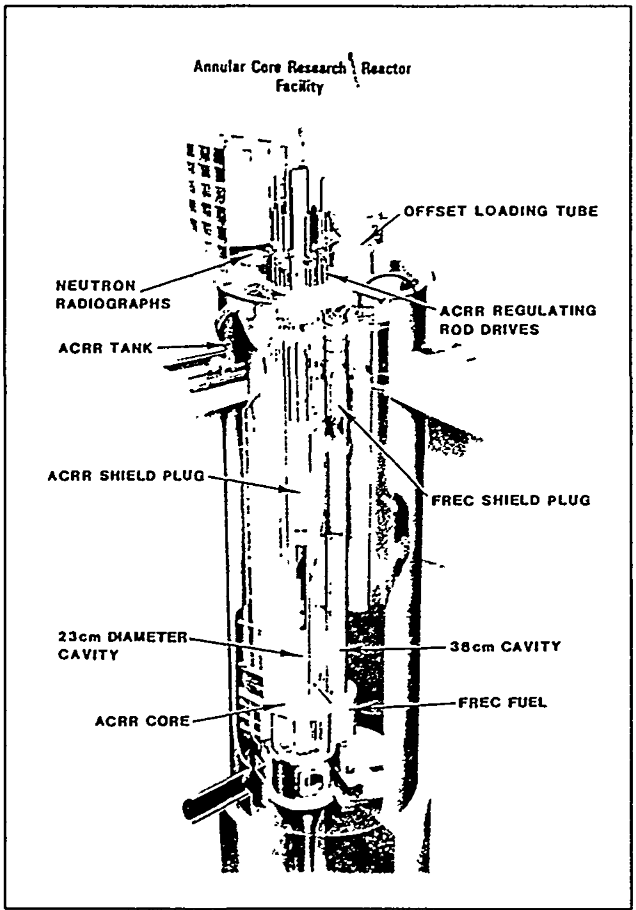

Figure 11. Cutaway diagram of the ACRR facility.

VIII. SUMMARY

Table 1 provides a compilation of the characteristics for the pulse reactors. Included are material descriptions, physical dimensions, and operational parameters.

In summary, Sandia National Laboratories has designed and operated six pulse research reactors for over three decades to provide research and test support to a number of weapon and energy related projects.

Table 1 Summary of Reactor Characteristics

	Godiva Type				Pool Type	
Descriptio	SPR	SPR II	SPR III	SPR IIIM	ACPR	ACRR
Fuel Material	U	U-10 Mo	U-10 Mo	U-10 Mo	U-Z_rH	UO_2BeO
Uranium Content %	100.0	90.0	90.0	90.0	12.0	21.0
Uranium Enrichment	93.2	93.2	93.2	93.2	20.0	35.0
Melting Point	1132.0	1135.0	1135.0	1135.0	1800.0	2450.0
Fuel Mass (kg)	57.9	105.0	252.0	~290.0	350.0	405.0
Fuel Clad/Coatin	Cadmium	Aluminum	Aluminum	Aluminum	Stainless Steel	Stainless Steel
Core Diameter (cm)	17.8	20.3	29.7	33.0	70.0	80.0
Core Height (cm)	14.3	20.8	36.8	36.8	38.1	50.8
Cavity Diameter (cm)	0.8	3.8	17.0	21.3	22.9	22.9/50.8
Core Cooling	Air	Nitrogen	Nitrogen	Nitrogen	Water	Water
Control System	Fuel Rods	Fuel Rods	Cu Reflectors	Cu Reflectors	Fuel Followed B_4C	Fuel Followed B_4C
Pulse System	Fuel Rods	Fuel Rods	Al Reflector	Al Reflector	B_4C Rod	B_4C Rod
Cavity Fluence (^{n}/cm^2)	2×10^{14}	8×10^{14}	6×10^{14}	6×10^{14}	3.6×10^{15}	6×10^{15}
Reactor Yield (MJ)	0.55	4.1	10.0	12.0	110.0	300.0
Temperature Rise (°C)	110.0	450.0	450.0	450.0	730.0	950.0
Pulse Width	50 µs	40µs	76µs	90.0µs	4.7ms	7.0 ms
Reactivity ($)	1.03	1.12	1.12	1.12	4.50	3.50
Peak Power (MW)	8,200	76,000	98,000	104,000	15,000	30,000
Operational Period	1961-1967	1967-Present	1975-Present	1995 -	1967-1977	1978-Present
Number of Operations	5,600	5,300	9,600	-0-	10,000	5,900

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