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THE ADVANCED NEUTRON SOURCE (ANS) PROJECT

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ABSTRACT

The Advanced Neutron Source (ANS) is a new user experimental facility for neutron research planned at Oak Ridge. The centerpiece of the facility will be a steady-state source of neutrons from a reactor of unprecedented flux. In addition, extensive and comprehensive equipment and facilities for neutron research will be included. The scientific fields to be served include neutron scattering with cold, thermal, and hot neutrons (the most important scientific justification for the project); engineering materials irradiation; isotope production (including transuranium isotopes); materials analysis; and nuclear science.

I. MISSION, PERFORMANCE GOALS, AND OVERALL REQUIREMENTS

The Seitz-Eastman Committee, in a 1984 National Academy Study,[1] was most influential in defining the scientific justification for an advanced steady neutron source and in specifying the broad performance capabilities required. The committee debated the performance capabilities of present research reactor technology and selected an achievable, but challenging, performance goal to meet the scientific needs of the user community. The Seitz-Eastman report was studied, and its findings endorsed, by the U.S. Department of Energy's (DOE's) own Energy Research Advisory Board (ERAB).[2]

Subsequently, workshops[3,4] and the National Steering Committee for an Advanced Neutron Source (NSCANS) defined the performance requirements in greater detail and also in quantitative terms. NSCANS continues to guide the ANS Project and to review the project team's work to ensure that user requirements are being met. The DOE

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orders, and other appropriate codes and regulations, further define design requirements (see Tables 1 and 2).

II. TECHNOLOGY AND BASIC DESIGN FEATURES FOR THE ANS REACTOR

The user requirements, particularly the need for a large, accessible volume of very high thermal neutron flux, determine the main features (high power and small size) of the reactor core design. To minimize technical risk, the project has adopted the involute, aluminum-clad cermet fuel plates and the annular core arrangement, common in existing high-performance beam reactors [e.g., High Flux Isotope Reactor (HFIR) and the Institut Laue-Langevin (ILL) reactor at Grenoble]. A new geometrical arrangement using fuel elements of different diameters separated axially, offers many safety and performance advantages and forms the basis for the final preconceptual core design (see Figs. 1 and 2).

The basic concept is conventional: a heavy-water-cooled and reflected reactor. The large reflector tank places ~1.5 m of heavy water around the core and provides space for two cold sources, beam tubes and guides, rabbits, and isotope production targets (Fig. 3).

The coolant flows upward through the core, leading to a quicker, and more predictable, transition from forced-to-natural convection and also to a reduced probability of core flow blockage, because foreign objects or debris falling onto the core would be swept up and caught by the primary coolant screen when the flow is started.

There are two independent scram systems; one that is also used for control, is inside the central hole of the annular fuel elements and within the primary coolant circuit. The other scram system is outside the pressure boundary (Fig. 4). Either system alone can safely shut down the reactor, even if one rod were stuck.

III. FACILITIES

The reactor is housed in a large containment dome, with floor space for beam tube experiments. The experimenters are physically separated from the operating areas. A large guide hall provides space for cold neutron beam experiments, and office space is

Table 1

Quantitative Expression of Performance Goals

Neutron beams	
Peak thermal flux in reflector $m^{-2}.s^{-1}$	5 - 10 x 10^{19}
Thermal/fast flux ratio	≥80
Materials irradiation[a]	
Fast flux, $m^{-2}.s^{-1}$	≥1.4 x 10^{19}
Fast/thermal flux ratio	≥0.5
Transuranium production[b]	
^{252}Cf Production rate, g/year	1.5
^{254}Es Production rate, μ/year	40

[a] To match or exceed the capabilities of the irradiation positions in the HFIR flux trap.

[b] To match or exceed production capabilities at HFIR.

Table 2

Overall Requirements

Appropriate codes and regulations	User Needs
10CFR50, Appendix A (Design Criteria)	Neutron flux and spectrum
Pressure boundary integrity Two diverse scram systems Decay heat removal	Access for experiments
ASME Section III, Class 1	
DOE 6430.1a, Safety Policy	

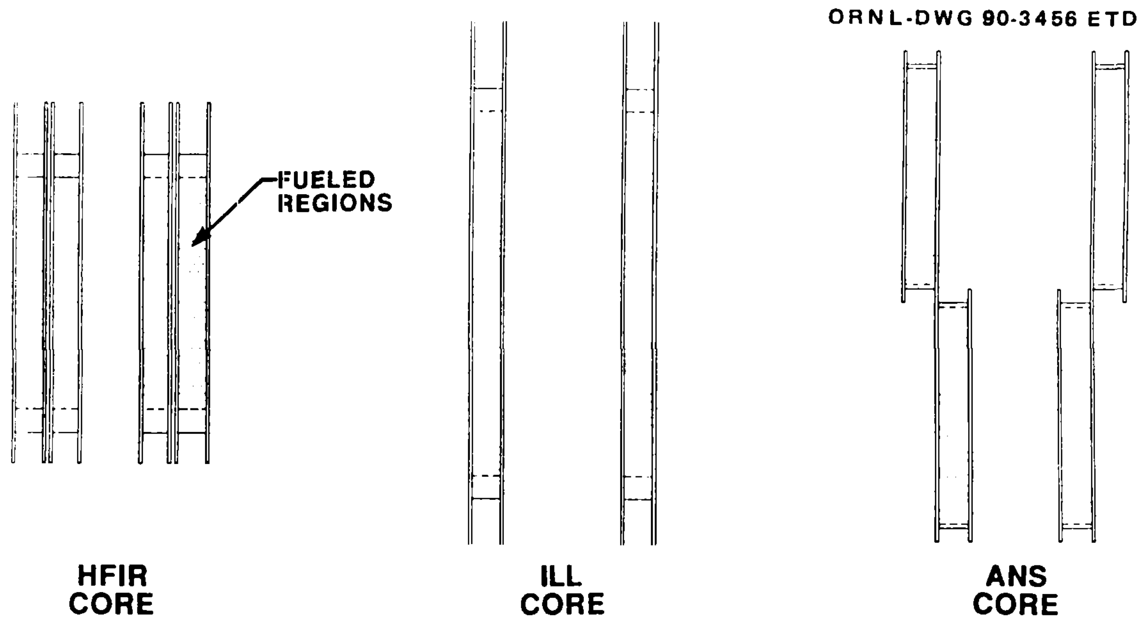


Fig. 1. Scale comparison of three involute plate, annular fuel element, research reactor cores.

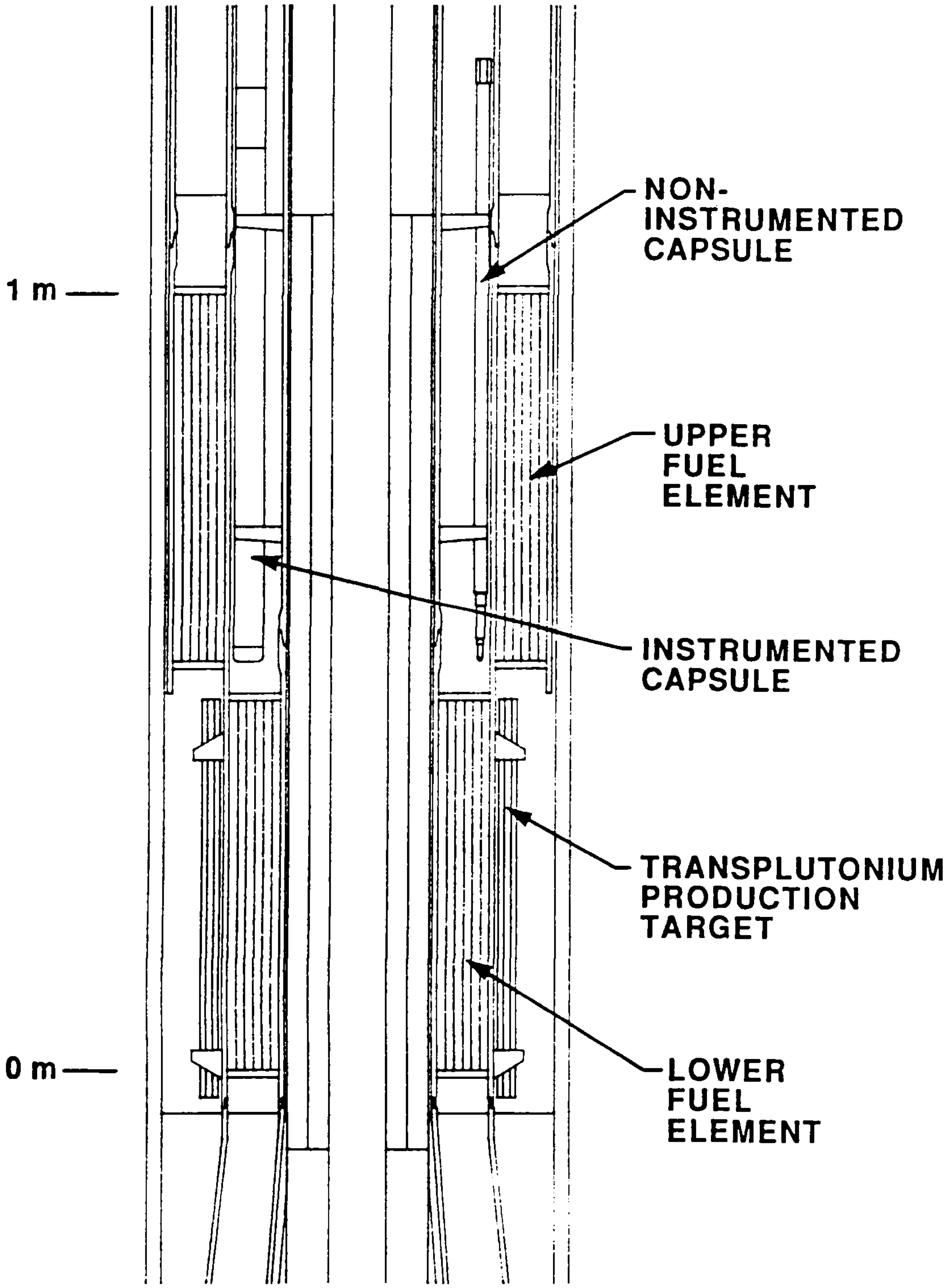


Fig. 2. Final Preconceptual Core Design

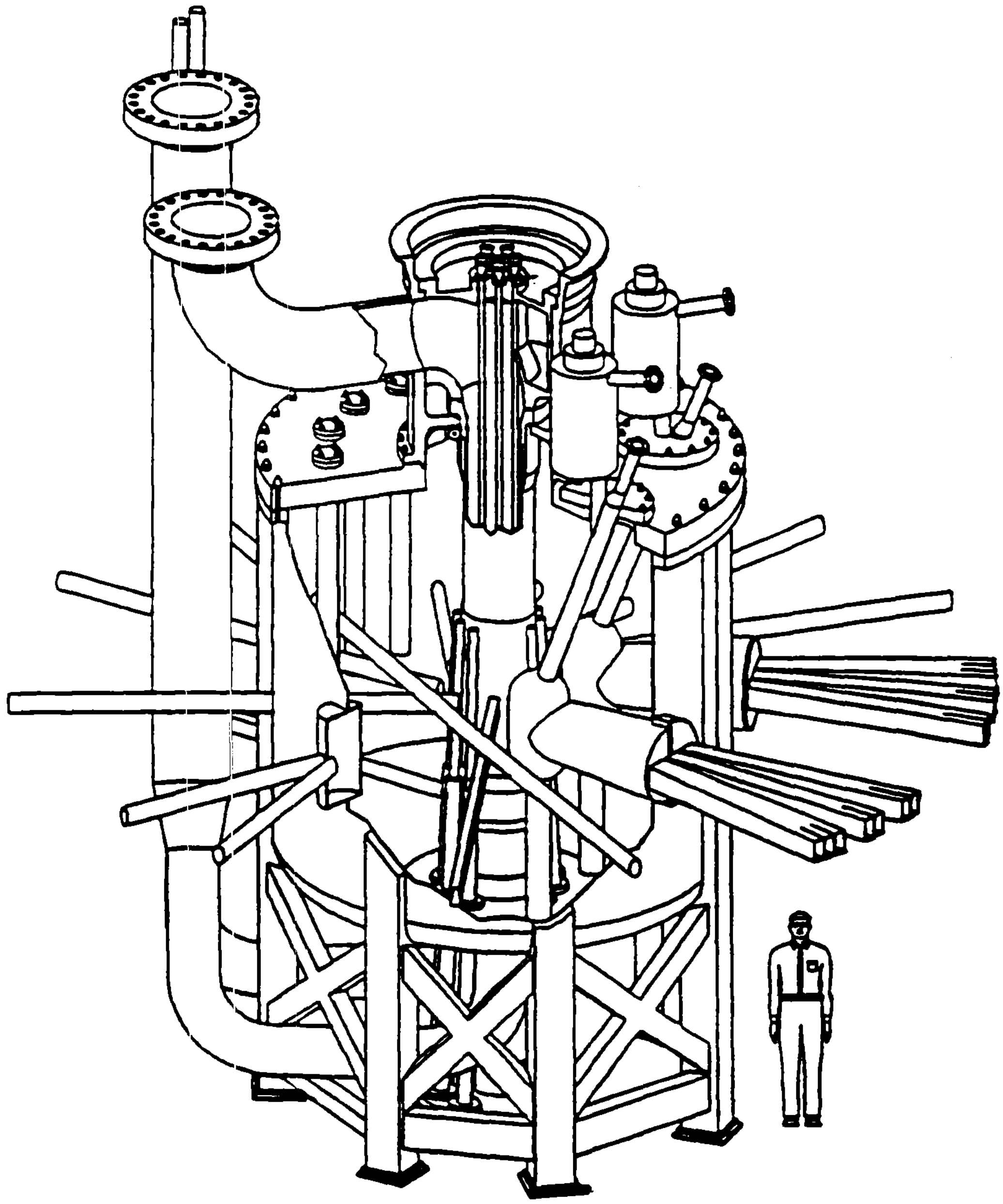
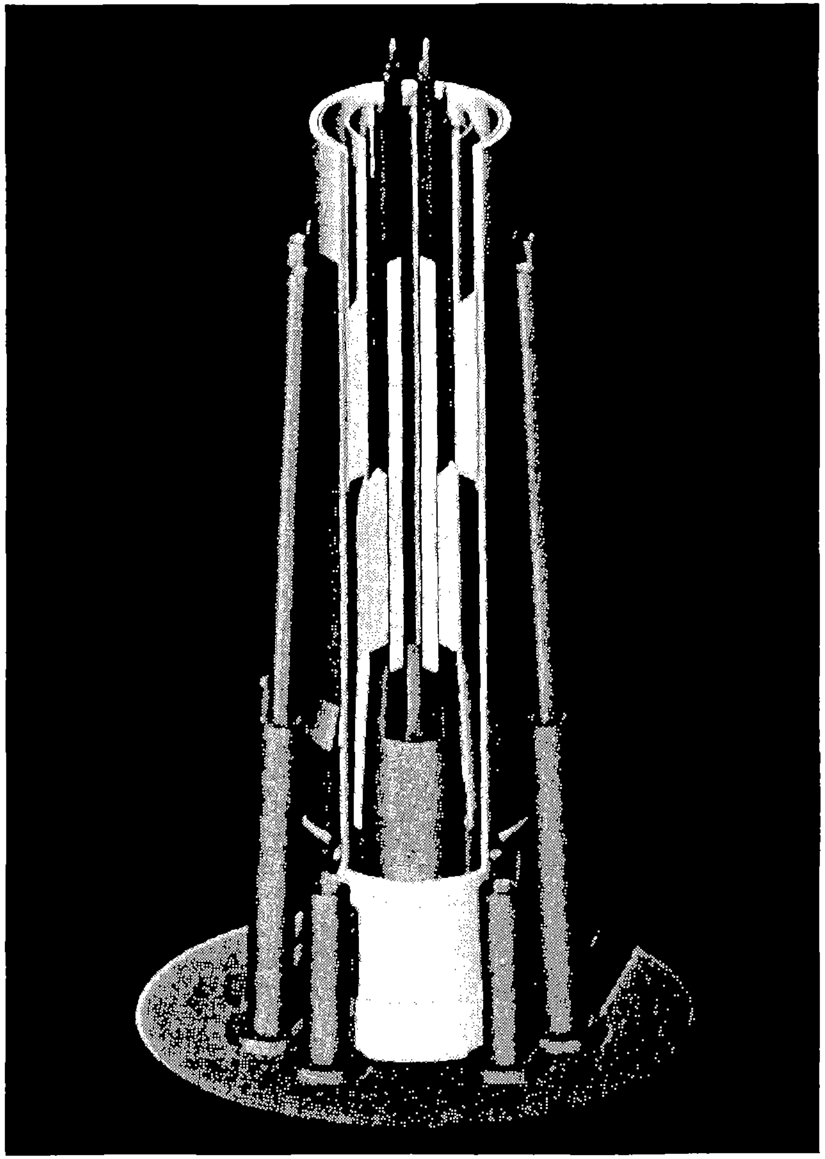


Fig. 3. ANS Reactor Systems



The ANS Reactor Core Arrangement, Using Two Separate Fuel Elements of Different Sizes, Offers Many Safety and Performance Advantages.

Figure 4

provided to accommodate facility staff and the ~1000 users expected each year (Figs. 5 and 6).

IV. SAFETY GOALS AND RESISTANCE OF THE ANS PRECONCEPTUAL DESIGN TO ACCIDENTS

The ANS design goal for core damage probability is 10^{-5}/year, the same as that proposed in the DOE's draft safety objectives policy for the much larger production reactors. For the same core damage probability, personal risks from the ANS will naturally be lower than from the New Production Reactor (NPR), because the ANS fission product inventory (source term) is very much lower. For comparison, the HFIR and the Advanced Test Reactor (ATR) each have a core damage probability of ~10^{-4}/year.

Work began on probabilistic risk assessment (PRA) for the project very early, and the results have already led to design changes. By implementing an integrated safety, research and development (R&D), and design program from the beginning of the project, many safety issues were addressed during preconceptual design to minimize later, and more expensive, design changes or retrofits (e.g., Table 3). Continued attention to the areas identified in Table 3 for further work is expected to bring the 10^{-5} risk goal within reach.

V. R&D ISSUES RAISED BY THE DIFFERENCES BETWEEN THE ANS DESIGN AND HFIR

The ANS user requirements could not be met by a facility like the HFIR. However, many of the differences between ANS and HFIR (e.g., more beams and experimental stations, a higher heat removal capacity, and more effective scattering instruments) raise no new safety issues. The higher power density does mean that several operating parameters - including coolant velocity, fission rate in the fuel, and heat flux - are outside HFIR operating experience (Table 4). The tests and analyses necessary to verify that the new operating conditions can be safely achieved are either already under way or appropriately scheduled. Independent reviewers have endorsed the design approach, basing their judgement on available data and on the project team's plans to gather the necessary additional data.

Some of the planned research and tests are listed in Table 5. Results from this R&D program will be shared with the community.

PERSPECTIVE OF ANS FACILITY

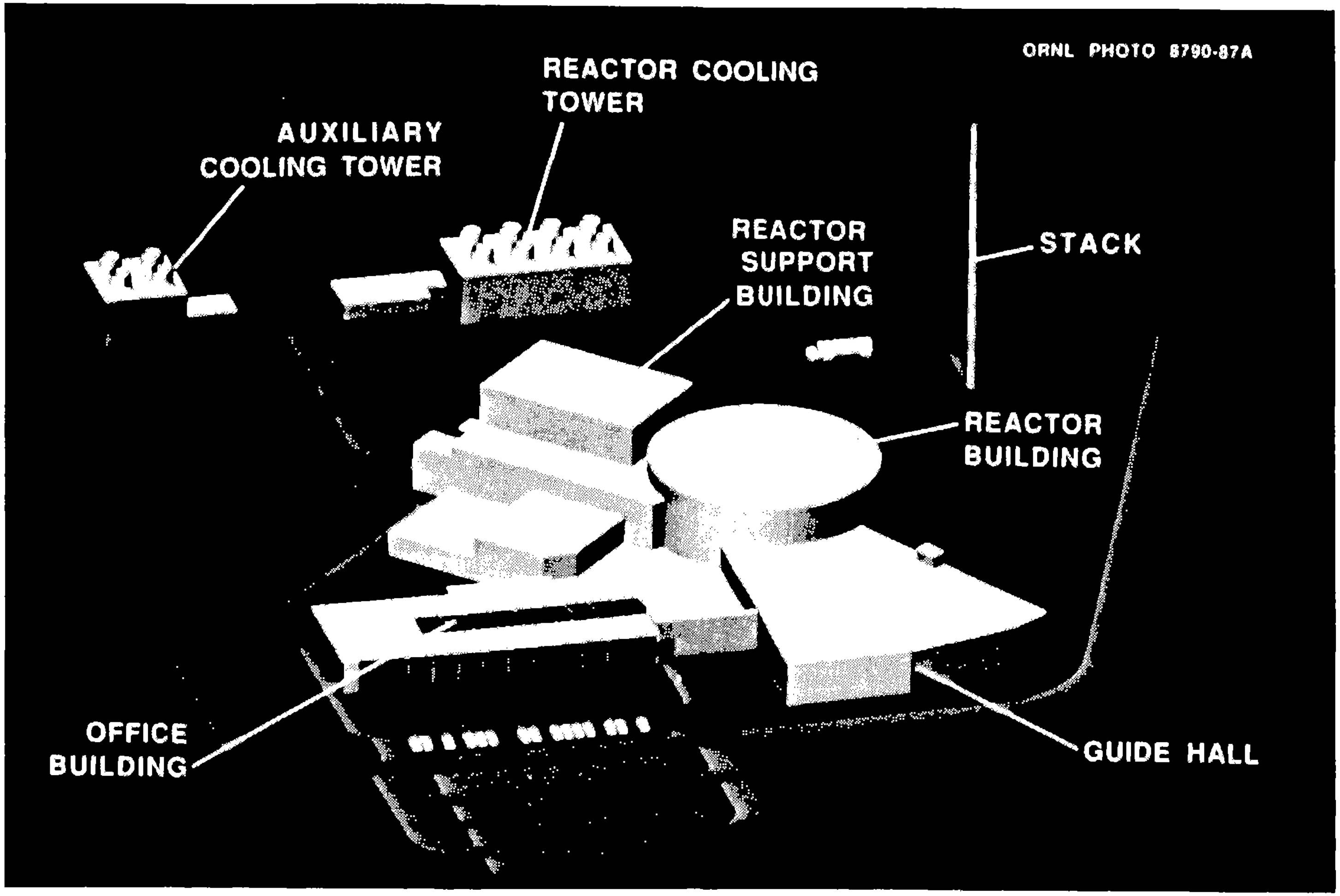


Fig. 5.

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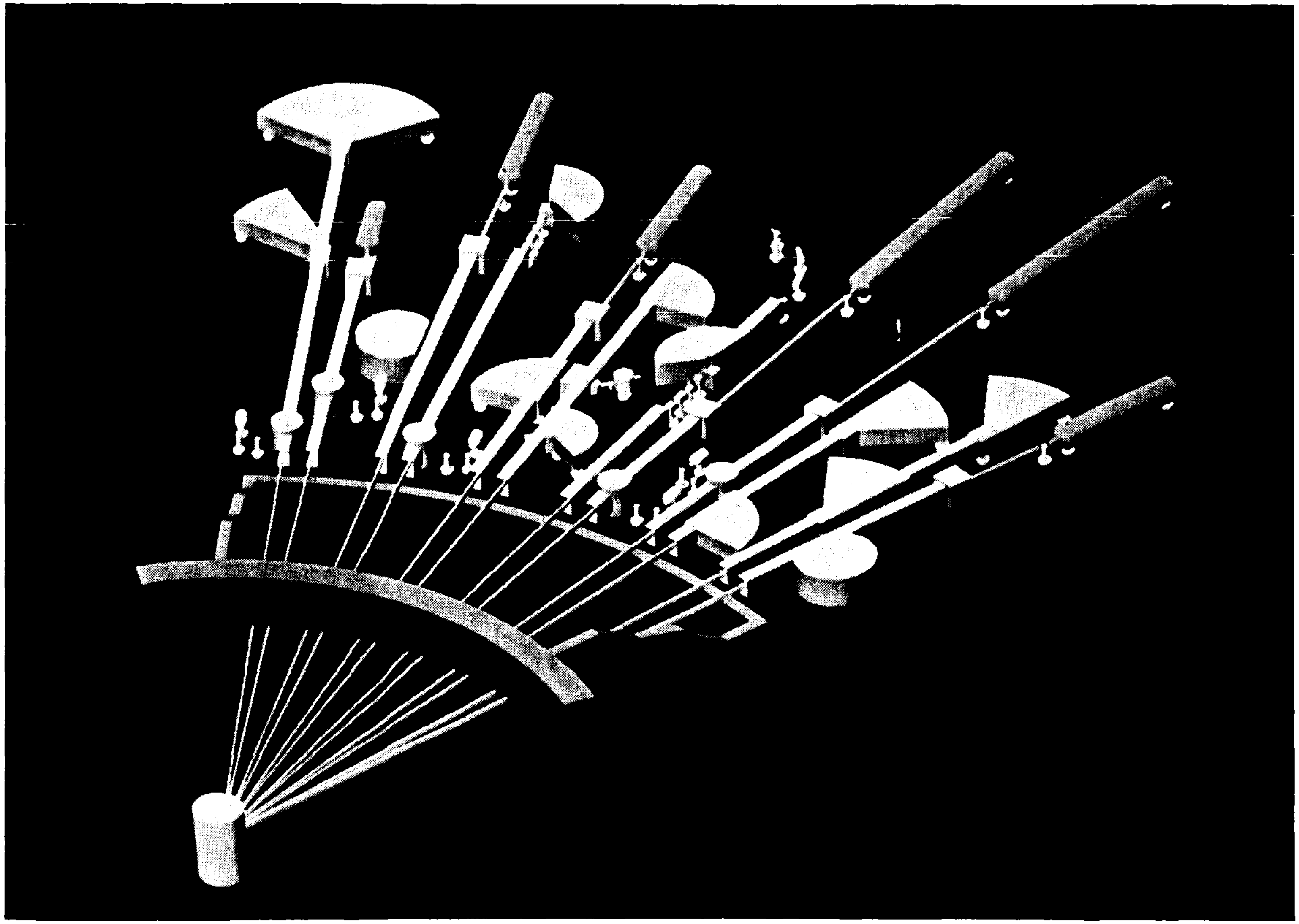

Fig. 6.

Table 3

HFIR AND ANS CORE DAMAGE PROBABILITIES FOR DOMINANT ACCIDENT SEQUENCES

INITIATING EVENT CATEGORY	CORE DAMAGE PROBABILITY (PER YEAR) HFIR[1]	ANS[2]	NOTES, REASONS FOR EXPECTED DIFFERENCES
FLOW BLOCKAGE	9.0×10^{-5}	7.2×10^{-6}	ADVANTAGE OF ANS REFUELING MACHINE AND UPFLOW
LARGE PIPE BREAK (COMPLETE SEVER-ANCE >51 mm PIPE)	3.3×10^{-5}	<3.0E-7	ANS USES FRACTURE MECHANICS-BASED LEAK BEFORE BREAK APPROACH IN THE DESIGN STAGE
SCRAM SCENARIOS (PRIMARILY MANUAL SCRAM FOR NORMAL SHUTDOWN)	2.3×10^{-5}	$<10^{-7}$	ANS DOES NOT HAVE AUTOMATIC[3] DEPRESSURIZATION OR CUTOFF OF SECONDARY COOLANT FLOW AFTER SCRAM
FUEL MANUFAC-TURING DEFECT	2.1×10^{-5}	ASSUMED SAME	ANS GOAL OF 10^{-5} TOTAL CORE DAMAGE PROBABILITY WILL REQUIRE ADVANCED FUEL INSPECTION TECHNOLOGY
LOSS OF INSTRUMENT AIR	1.7×10^{-5}	$<10^{-7}$	ANS DOES NOT REQUIRE AUTOMATIC DEPRESSURIZATION[3]
SMALL PIPE BREAKS (SEVERANCE OF ≤51 mm PIPE)	1.6×10^{-5}	ASSUMED SAME, IN ABSENCE OF FURTHER DESIGN WORK	ANS GOAL OF 10^{-5} TOTAL CORE DAMAGE PROBABILITY WILL REQUIRE SOME IMPROVEMENT BY DESIGN
PRESSURIZER PUMP FAILURES	7.3×10^{-6}	2.2×10^{-7}	
TOTAL	2.1×10^{-4} (92.5% OF TOTAL HFIR RISK)	4.5×10^{-5} (SHOULD BE ~90% OF TOTAL ANS RISK)	

[1]HFIR results based on Oct. 17, 1988 Update to the HFIR PRA.

[2]ANS results provided by Brookhaven National Laboratory (Fullwood and Shier) in the July 1989 ORNL/BNL PRA review meeting. The ANS results are very much subject to change since they are based, in part, upon preconceptual design information that will be modified and defined in great detail as the design effort progresses.

[3]ANS does not need the reactor vessel NDT-avoidance fixes (automatic depressurization of the primary and automatic post scram cutoff of the secondary coolant).

[4]HFIR has recently changed the failure mode of the automatic depressurization valves from fail-open to fail-closed, giving lower probability than listed here.

Table 4

ANS Reactor - Specifications and Comparison with HFIR

Quantity & unit	ANS	ANS notes	HFIR*
Fission power level, MW(f)	350		100
Power transferred to primary coolant, MW(c)	332	Heat convected away from fuel plates	97
Average power density, MW(c)/L	4.9		1.9
Max. power density, MW(c)/L	8.3	Estimated, fuel grading not yet optimized	4.4
Core life, d	14		20
Core active volume, L	67.4	Fueled volume	50.6
Fuel form	U_3Si_2		U_3O_8
Fuel matrix	Al		Al
Vol % of fuel in meat, %	15		12.5/18.0**
Fuel loading, kg U^{235}	14.9		9.4
Fuel cladding	6061 Al		6061 Al
Fuel plate thickness, mm	1.27		1.27
Clad thickness, mm	0.254		0.254
Coolant channel gap, mm	1.27		1.27
Coolant (and reflector)	$D_2O(D_2O)$		$H_2O(Be)$
Inlet pressure, MPa	3.7		4.1
Inlet temperature, °C	49		49
Heated length, mm	474		508
Coolant velocity in core, m/s	27.4	May be reduced after detailed analysis	16
Core pressure drop, MPa	1.6		0.7
Outlet pressure, MPa	2.1		3.4
Bulk coolant outlet temp., °C	81		73
Average heat flux, $MW(c)/m^2$	6.3		2.4
Max. heat flux, $MW(c)/m^2$	10.7	Estimated; fuel grading not yet optimized	5.6
Max. fuel centerline temp., °C	400	Design groundrule	327
Peak thermal flux in reflector, 10^{19} $m^{-2}s^{-1}$	>8	Unperturbed	1.5

*At 100 MW.

**Inner element/outer element.

Table 5. Some Planned Research and Tests

Area	Tests	Dates to begin (tentative)
Fuel plate stability	Epoxy plate tests (single plate)	Feb. 1990
	Aluminum plate tests (single plate)	Oct. 1990
	Epoxy plate tests (multiplate)	Nov. 1990
	Aluminum plate tests (multiplate)	Jan. 1992
	Thermal stress tests	May 1993
	Full core test	June 1993
	Plate vibration tests	Sept. 1993
Thermal hydraulics	T.H. limits narrow channels with coolant velocity	Sept. 1990
Fuel performance	Fuel tests by accelerator irradiations	underway at ANL
	In-pile sample tests at HFIR	awaiting HFIR full power operation
	Fabrication tests of two-dimensional grading	underway at B&W
	Test fabricate fuel plates to conceptual core design	Sept. 1990
	Miniplate tests in HFIR	Jan. 1991
Oxide formation	Out-of-pile tests with high coolant velocity and heat flux	underway at ORNL
	In-pile test at HFIR	Jan. 1992
Cold source	Liquid nitrogen simulations	July 1990
	Liquid H_2 or D_2 tests	Jan. 1993

VI. SUMMARY

The ANS Project has completed the preconceptual design phase. An evolutionary process has led to a new reference core design with greatly enhanced thermal-hydraulic margins and improved performance parameters in many areas. The reactor systems design has also evolved in response to input from the integrated safety analysis program and from HFIR studies and reviews. R&D is under way or planned in those areas important to safety, design, and performance improvement.

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