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ADVANCED NEUTRON SOURCE OVERVIEW AND STATUS REPORT*

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ABSTRACT

The new Advanced Neutron Source is a research facility centered around a new research reactor of unprecedented flux. Unique core and cooling system designs provide many inherent or passive safety features. The combination of a relatively high power level and a small core places special requirements on the response time of the reactor control system, and especially on the scram function. Similar requirements have been faced before on research reactors, and successfully met. The ANS design has evolved from those other reactors.

INTRODUCTION

The Advanced Neutron Source is a new facility, planned at the Oak Ridge National Laboratory, for all kinds of neutron research.

The main scientific justification for the project, expressed by the National Academy Committee on major materials facilities ("Major Facilities for Materials Research and Related Disciplines" National Academy Press, Washington, D.C., 1984), is the U.S. need for a world class neutron scattering facility. The essential requirements are for a very high flux of thermal neutrons in a region that is accessible to beam tubes and with space for one or more liquid deuterium cryostats large enough to moderate some of the thermal neutrons to much lower energies, producing so-called cold neutrons. In addition to neutron beams, capabilities for isotopes - especially for transuranium isotope production, and materials irradiation testing must be provided: the nation's current source of the transuranics is the High Flux Isotope Reactor (HFIR) at Oak Ridge, a 100 MW research reactor (currently operating at 85 MW) that also offers materials testing facilities. However, the HFIR will be nearly 35 years old when the ANS come on line early in the next century.

Table 1 lists the technical objectives of the ANS project. In addition, the project has adopted a philosophy of minimizing technical risks and safety issues by relying on known technology to meet the minimum design criteria.

CORE DESIGN CONCEPT

To meet those requirements it is clear that the reactor must produce a large number of fission neutrons, i.e., it must have enough power. In addition, the requirements dictate certain other design features (Table 2) which are very different from power reactors and have implications (many of them positive) for the safety analysis of the reactor (Table 3.)

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The reactor core design concept is outlined in Fig. 1. The annular, involute geometry of the fuel plates is copied from the HFIR and the reactor at the Institut Laue-Langevin (ILL) at Grenoble. The aluminum clad mixture of U_3Si_2 fuel particles and aluminum powder has been developed and extensively tested by the Reduced Enrichment Research and Test Reactors (RERTR) program, although more tests at higher temperatures and burnup rates are underway or planned by the ANS project. The short heated length of the HFIR core design and the long neutronic length of the ILL core have been combined. The coolant and reflector are heavy water, the usual choice for reactors that are optimized for neutron beam production. Nominal specifications of the reference conceptual design are given in Table 4 and Fig. 2.

The core cooling system design concept, which evolved through iteration among the design, safety, and R&D groups of the project, incorporates many passive safety features (Fig. 3).

CPBT

The primary coolant pressure boundary in the region of the core is called the Core Pressure Boundary Tube. It fits fairly closely around the upper fuel element (see Fig. 4), and is made from aluminum 6061, for which ASME Section 3 Code Approval is being sought. Aluminum 6061 is chosen because of its high thermal conductivity and relatively low neutron absorption, and because there is extensive experience with it as a structural material in research reactors (for example, in the HFIR and the High Flux Beam Reactor (HFBR) at Brookhaven National Laboratory).

The fracture mechanics properties of aluminum 6061 are such that one cannot, as with steel, take any credit for leak-before-break detection nor, unless unacceptably thick sections are used, can one completely rule out flaw growth during operation. Thanks to the primary coolant circuit safety features illustrated in Fig. 3 the core can survive without damage a large break in the CPBT downstream of the fuel elements, but not upstream. Therefore, a pressure vessel with an integral guard pipe concept has been adopted. In this design (Fig. 5), a continuous outer tube is the pressure boundary in normal operation. An inner guard tube is separated from it by a narrow, annular cooling channel. Holes or slots at the bottom of the lower guard tube would restrict the coolant loss rate to an acceptable value following failure of the outer tube. The flow rate in the cooling annulus is limited by a restrictor placed at the end of the upper fuel element: the flow rate must be high enough to cool the tubes, but low enough to provide a Bernoulli pressure rise that keeps the lower guard tube in compression (i.e., with no tendency for flaw growth) during normal operation.

REACTIVITY CONTROL

The reactivity control system includes three hafnium rods in the central hole region, driven together from below by a mechanical system based on the successful HFIR design. These three rods can also be scrammed, with high acceleration, by individual springs that are individually released by magnetically held, fail safe mechanical latches, a design also based on the HFIR. This system alone is capable of meeting the reactor shutdown criteria even if one rod fails to scram, and of shutting down the reactor (although by a smaller margin) even if two rods fail.

A second independent and diverse shutdown system is provided by eight rods in the reflector tank outside the CPBT. This set of rods is driven from above (so that, for example, a single missile will not damage both drive units). These outer rods are reset and latched hydraulically, and driven in for a scram by a combination of hydraulic and spring forces.

In addition, burnable poison (boron) in the core controls the excess reactivity in fresh cores, reducing the negative reactivity that must be provided by the moveable control rods. Because of the high power density in the core, a high-speed scram system is needed to protect the core against transient overheating. Following a scram signal, the primary, central shutdown system is required to insert $1 of negative reactivity in 100 ms, even if one rod fails to scram and with the initial positions at the maximum withdraw limit and near the end of the useful rod neutron absorber life. This response time includes the delays from electronic signal conditioning, latch release, and the actual rod motion needed to insert $1.

Similar although slightly less demanding requirements apply to the secondary (outer) shutdown system.

These design requirements were established through the project's safety analysis program, including probabilistic risk assessments.

REACTOR CONTROL AND PROTECTION INSTRUMENTATION SYSTEMS

The two shutdown systems of the ANS will be independent, probably with two-of-four trip logic in each case. There are two different scram modes for the inner rods, as there are on the HFIR control system from which the design was derived. A fast scram is accomplished by interrupting the current to magnets which, when deenergized, release latches that hold each rod against a spring. The springs give each rod an initial acceleration of about 5 g. A slow scram is accomplished by interrupting current not directly to the magnets, but to the magnet power supply.

Reactor power control is achieved by driving all three of the inner rods by a motor. Signals are derived from neutron flux measurements, but in order to account for the variations in the neutron flux/reactor power ratio arising from changes in the core power profile with burnup, movement of the ionization chambers and other factors, the gain in the neutron flux signal channel is varied after comparing the flux signal to the reactor heat balance (coolant flow rate x temperature rise).

Using a digital system, the scram rate required for the heat balance calibration is probably one-half second or longer, within the capabilities of existing commercial nuclear plant systems; remember, we wish to rely only on existing technology to meet out minimum requirements.

The protection system fast trips, however, require less than 10 ms scan time, which is not met by the digital protection systems built for commercial reactors. It would be difficult to justify, for a single and unique reactor design, the enormous effort of verifying and validating new systems and software for safety applications. Our present thinking is, therefore, that the part of the system related to trips would use analog circuitry. The analog system would be monitored and tested by a computer.

Other parts of the instrumentation and controls system, such as the primary and secondary coolant control system, heavy- and light-water processing, and radiation monitoring are not expected to have more stringent requirements than those for a commercial reactor.

STATUS

Preliminary feasibility studies for the project began in fiscal year 1984, with preconceptual design beginning in 1986 and conceptual deigns in 1989: the current plan calls for completion of the conceptual design report in July of this year (1991). The work is funded through the U.S. DOE's Office of Basic Energy Sciences. Oak Ridge National Laboratory is the lead laboratory, and is assisted by many other Federal and University laboratories, by the nuclear fuel division of Babcock & Wilcox, and by the Architect/Engineering firm of Gilbert/Commonwealth, Inc. and their associates.

There is a National Steering Committee for the Advanced Neutron Source, with over thirty members drawn from the different fields of science and the different organizations that will use the facility. Their task is to guide the project, and provide user input, ensuring that, when built, the ANS will meet the needs of the broad scientific community.

All aspects of the project, in fact, are extensively reviewed by outside experts: in a period of three and one-half years there were no fewer than 43 reviews and workshops covering all aspects of the project including engineering design, the R&D program, Quality Assurance, NEPA compliance, budget and accounting, and project management. The project passed them all, and was improved through the advice and guidance received.

Table 1. Advanced Neutron Source Project technical objectives

- To design and construct the world's highest flux research reactor for neutron scattering -5 to 10 times the flux of the best existing facilities
- To provide isotope production facilities that are as good as, or better than, the High Flux Isotope Reactor (HFIR)
- To provide materials irradiation facilities that are as good as, or better than, the HFIR.

Table 2. Design features and scientific requirements

Feature	Relationship to Scientific Requirements
Heavy Water Coolant	Reduces (compared with light water) the moderation of fission neutrons within the fuelled region, thus increasing the number of neutrons thermalized outside the core where they are potentially available to the beam tubes.
Small core volume	Reduces the surface area through which the reflected, thermalized neutrons must pass, thereby increasing the flux.
Heavy Water Reflector Region	Low absorption of thermalized neutrons, thus increasing the number potentially available for extraction in beam tubes or guides. Provides a large volume of high thermal neutron flux in which the cold moderators can be accommodated outside the core. Can easily accommodate complex shapes of experimental equipment, and is not subject to radiation damage.
Low temperature reflector and coolant	Thermalize neutrons at lowest practical temperature (for thermal neutron beams and as source of neutrons for the cryogenic moderators).
High power (for a research reactor, much lower than power reactors)	Produces a large number of fission neutrons for subsequent moderation to thermal energies.
Large containment building	Provides floor space for experimental stations on thermal beam tubes.

Table 3. Some features of the ANS research reactor with significant safety implications compared with typical tower reactors

Feature	Comment
Low thermal power level	~ 300 MW compared with ~ 3000 MW means less stored and circulating energy
Low fission product inventory at end of cycle (smaller core contains much less fuel)	~ 6 kg compared with ~ 100 kg means much lower source term
Small core	Small core (~ 100 kg compared with ~ 100,000 kg means less chemical energy available for release)
Lower core and reflector coolant temperature	<100°C compared with ~ 325°C, so that coolant water would not flash into steam during a pressure loss
Lower primary coolant pressure	~ 3 MPa compared with ~ 15 MPa means much less stored energy
High degree of containment	Containment as big as a typical power reactor (to provide space for neutron beam experiments), but 10 times lower power level
Heavy water moderator	Longer neutron lifetime means slower reactivity transients
High power density (from high power and small core)	~ 5 MW/Litre compared with ~ 5 kW/Litre means more rapid heat up and dryout possible.
Occupied containment	Thermal neutron beam experiments must be located and operated at positions close to the reactor
High coolant velocity	High coolant velocity to accommodate high power density leads to very large flow forces on fuel plates and reactor internals
Short core life	The high power density leads to a short core life, with more frequent opportunities for refueling accidents.

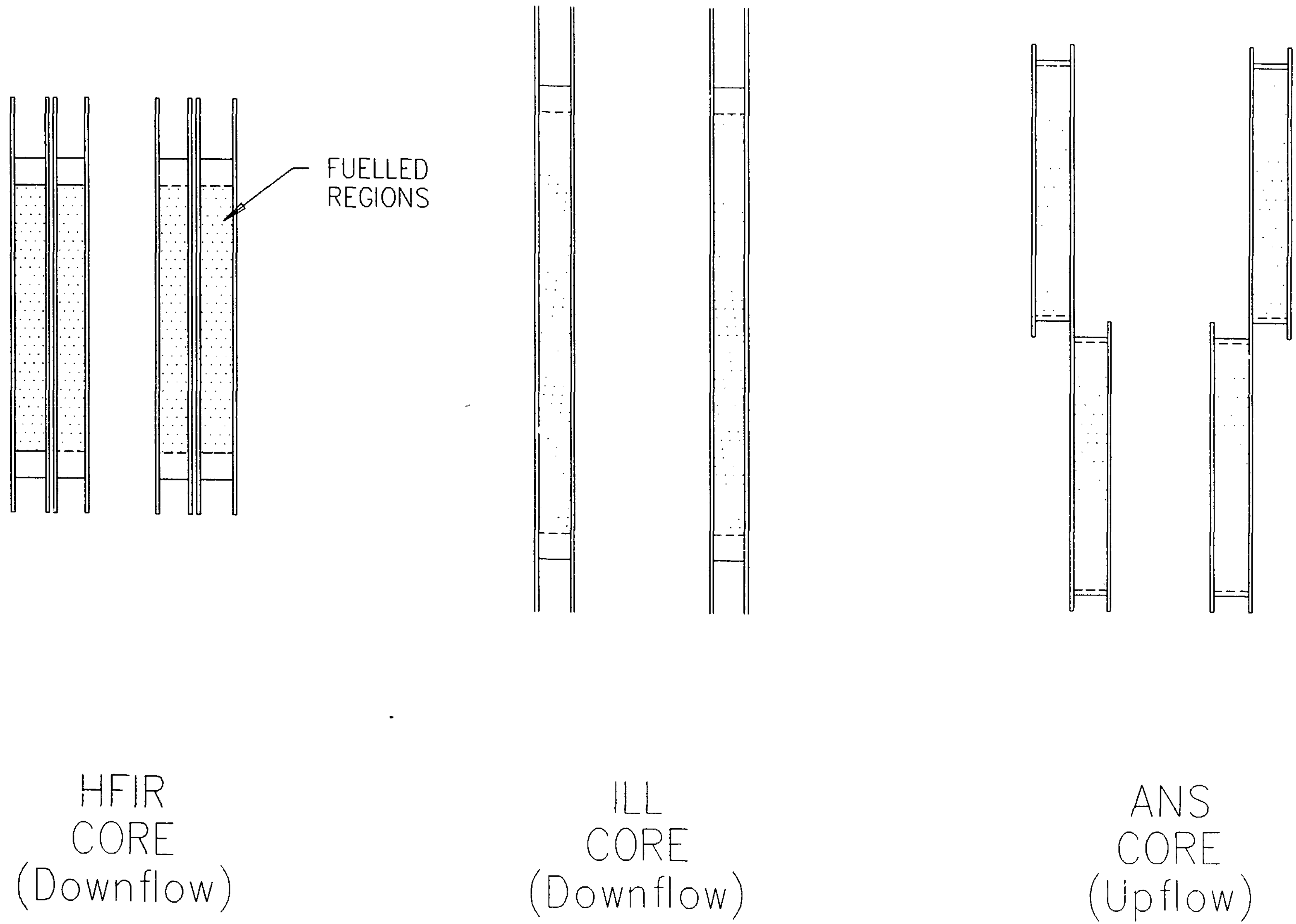


Figure 1 HFIR, ILL, and ANS Cores

Table 4. ANS reactor core nominal specifications

Quantity & Unit/Item	Reference Value/Material
Fission power, Mw(f)	330
Heat deposited in fuel (1 day into cycle), Mw	301
Heat deposited in fuel (EOC), Mw	303
Core life, d	17
Core active volume, L	67.6
Core dimensions	See figure 2
Fuel form	U_3Si_2
Fuel enrichment, %	93
Fuel matrix	Al
Vol. % of fuel in fuel meat.	11.2
No. of plates in upper element	432
No. of plates in lower element	252
Fuel plate thickness, mm	1.27
Aluminum clad thickness, mm	0.254
Coolant channel gap, mm	1.27
Coolant	D_2O
Heated length, mm	507
Coolant velocity in core, m/s	25
Inlet pressure (in plenum), MPa	3.2
Core pressure drop, MPa	1.3
Core Inlet temp, °C	45

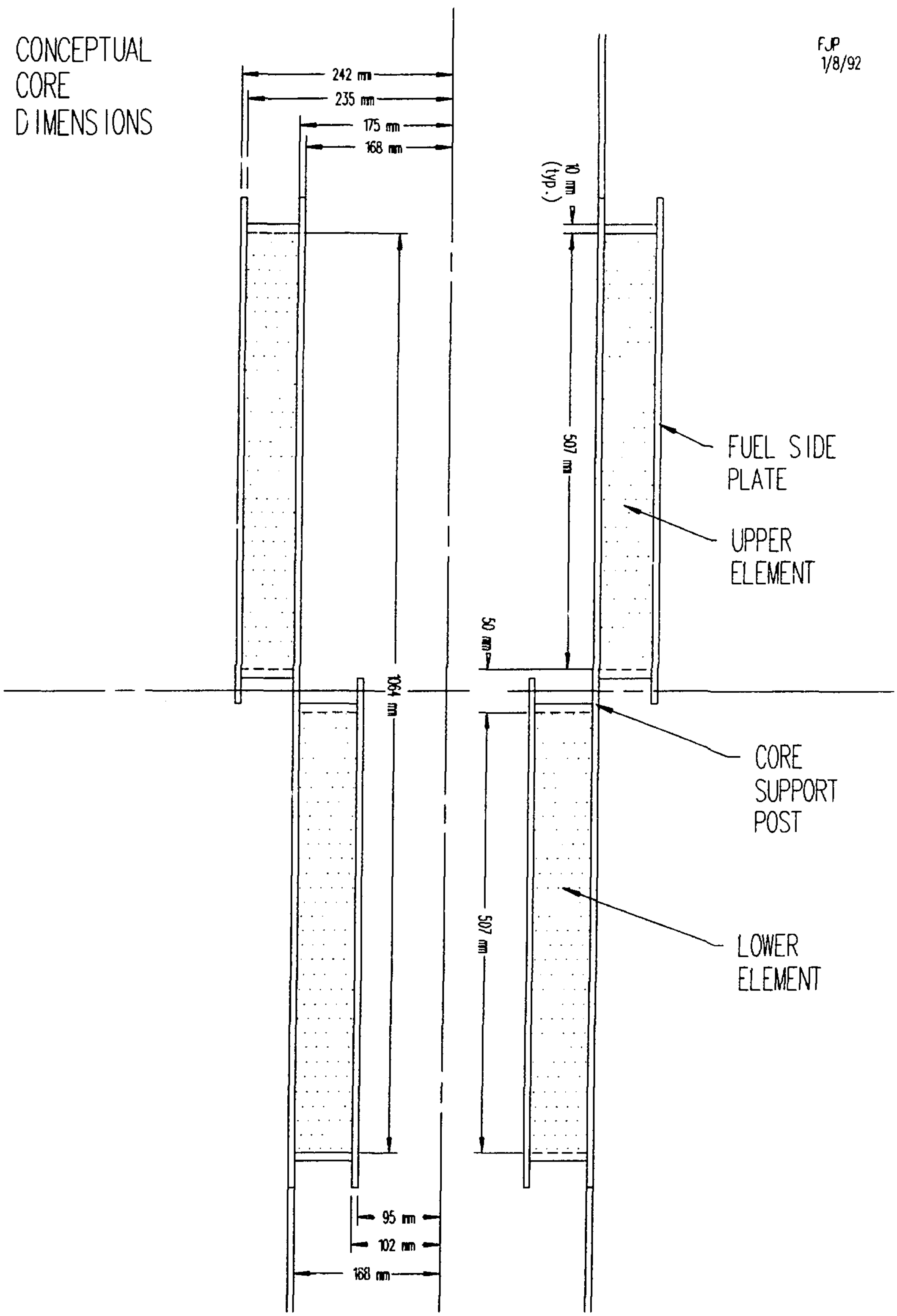

CONCEPTUAL
CORE
DIMENSIONS
FJP
1/8/92
242 mm
235 mm
175 mm
168 mm
10 mm
(typ.)
507 mm
FUEL SIDE
PLATE
UPPER
ELEMENT
50 mm
1064 mm
CORE
SUPPORT
POST
507 mm
LOWER
ELEMENT
95 mm
102 mm
168 mm

Figure 2

SOME PASSIVE SAFETY FEATURES OF THE ADVANCED NEUTRON SOURCE REACTOR COOLING SYSTEM DESIGN

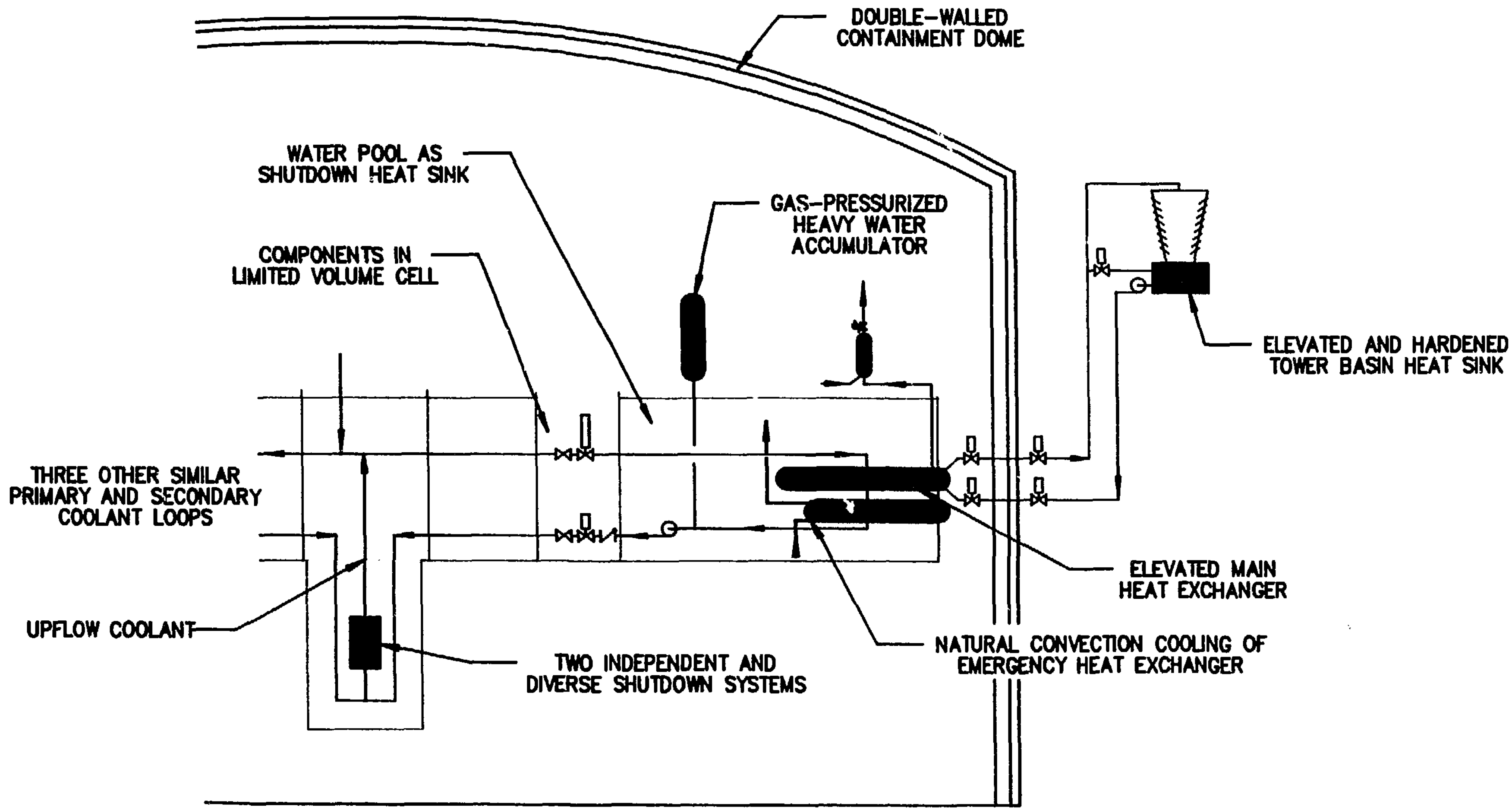


Figure 3

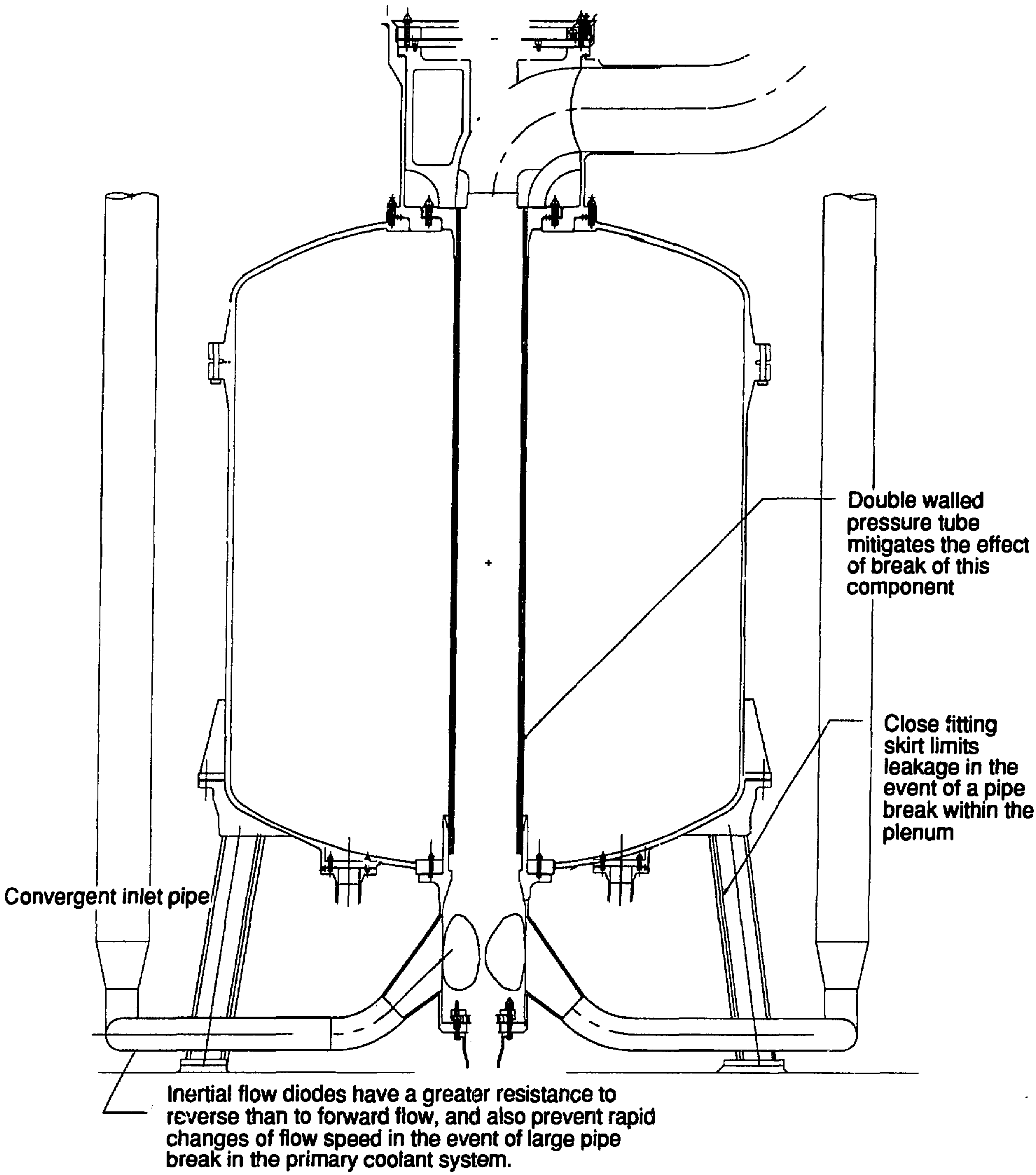
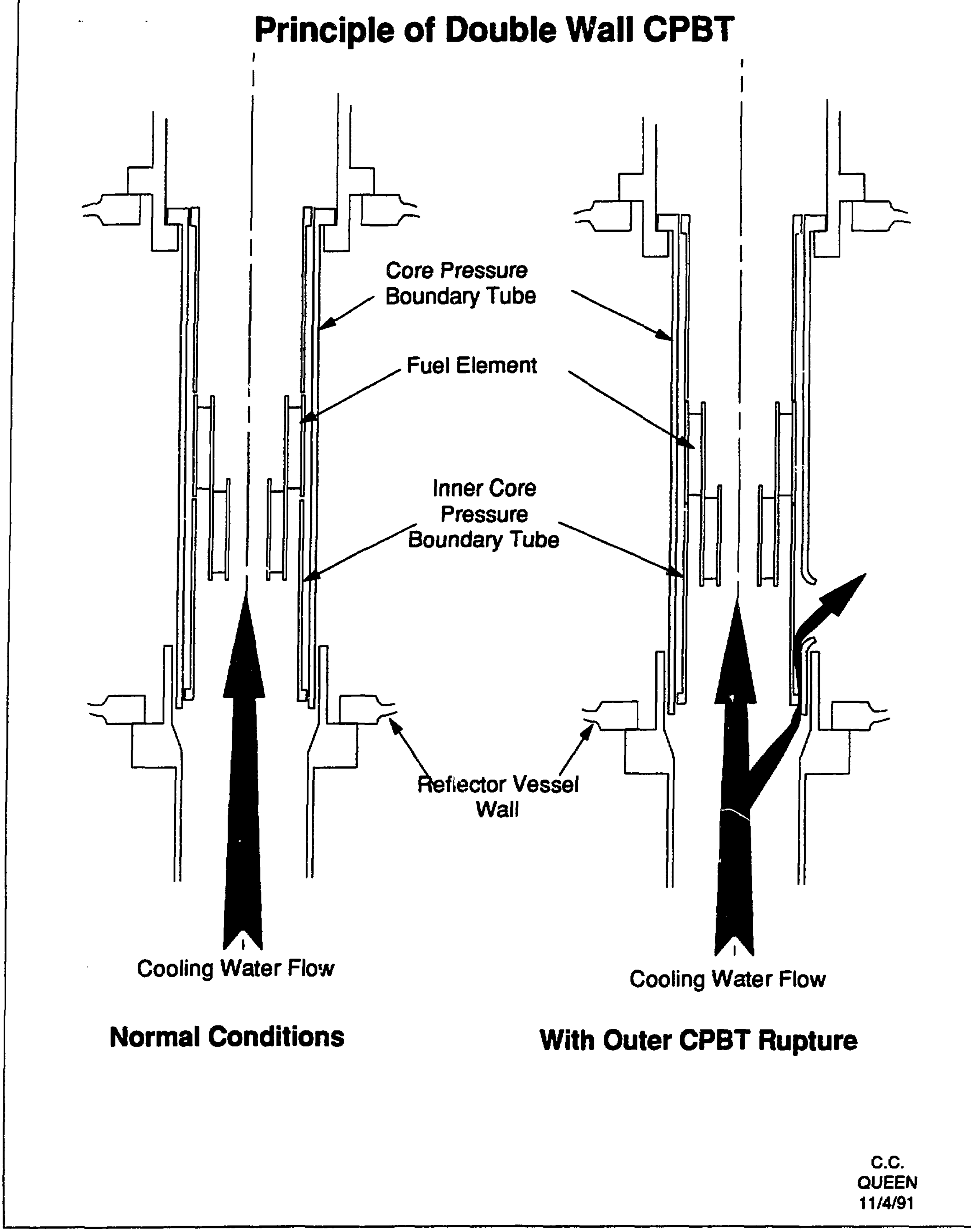


Figure 4 Safety features of the reactor system coolant components



Principle of Double Wall CPBT
Core Pressure Boundary Tube
Fuel Element
Inner Core Pressure Boundary Tube
Reflector Vessel Wall
Cooling Water Flow
Cooling Water Flow
Normal Conditions
With Outer CPBT Rupture
C.C. QUEEN 11/4/91

Figure 5