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Sodium Fast Reactor

Safety #2

DOE/HQ October 31, 2007

NRC/White Flint November 1, 2007

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U.S. Department of Energy

UChicago Argonne LLC

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Notice

- **This presentation has been revised to remove citations to and information from Applied Technology documents.**

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- **Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Reactor, USNRC Report NUREG-1368, February, 1994.**
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- ***Fast Reactor Database, 2006 Update,* IAEA-TECDOC-1531, IAEA, Vienna, December, 1996. See also http://www-frdb.iaea.org**
- **10CFR50 Appendix A, General Design Criteria for Nuclear Power Plants**
- **DOE O 5480.30, Nuclear Reactor Safety Design Criteria**
- **ANSI/ANS-54.1-1989, General Safety Design Criteria for a Liquid Metal Reactor Nuclear Power Plant**

Outline

- **<u>Safety Issues #1 (Last Time)</u>**
  - **Safety issues; DBA, BDBA, severe accident historically (coolant void, recriticality, FCI), passive safety**
  - **Safety analysis past results (FFTF, CRBRP, SAFR, PRISM)**
  - **Safety analysis methods; reactor, structural, coolant aerosols, containment**
- **<u>Safety Issues #2 (Today)</u>**
  - **Licensing issues in FFTF, CRBRP, SAFR, PRISM.**
  - **Domestic and international experience in perspective, high profile events in EBR-1, FERMI, BN-350, Phenix, SuperPhenix, and MONJU explained and lessons learned.**
  - **Safety testing results, EBR-II SHRT, FFTF ULOF (GEMS)**
  - **Inherent passive safety characteristics of sodium fast reactor systems (wrt loss of flow without scram, etc.). Inherent reactivity shutdown, natural circulation decay heat removal**

Licensing issues in FFTF, CRBRP, SAFR, PRISM

FFTF Chronology

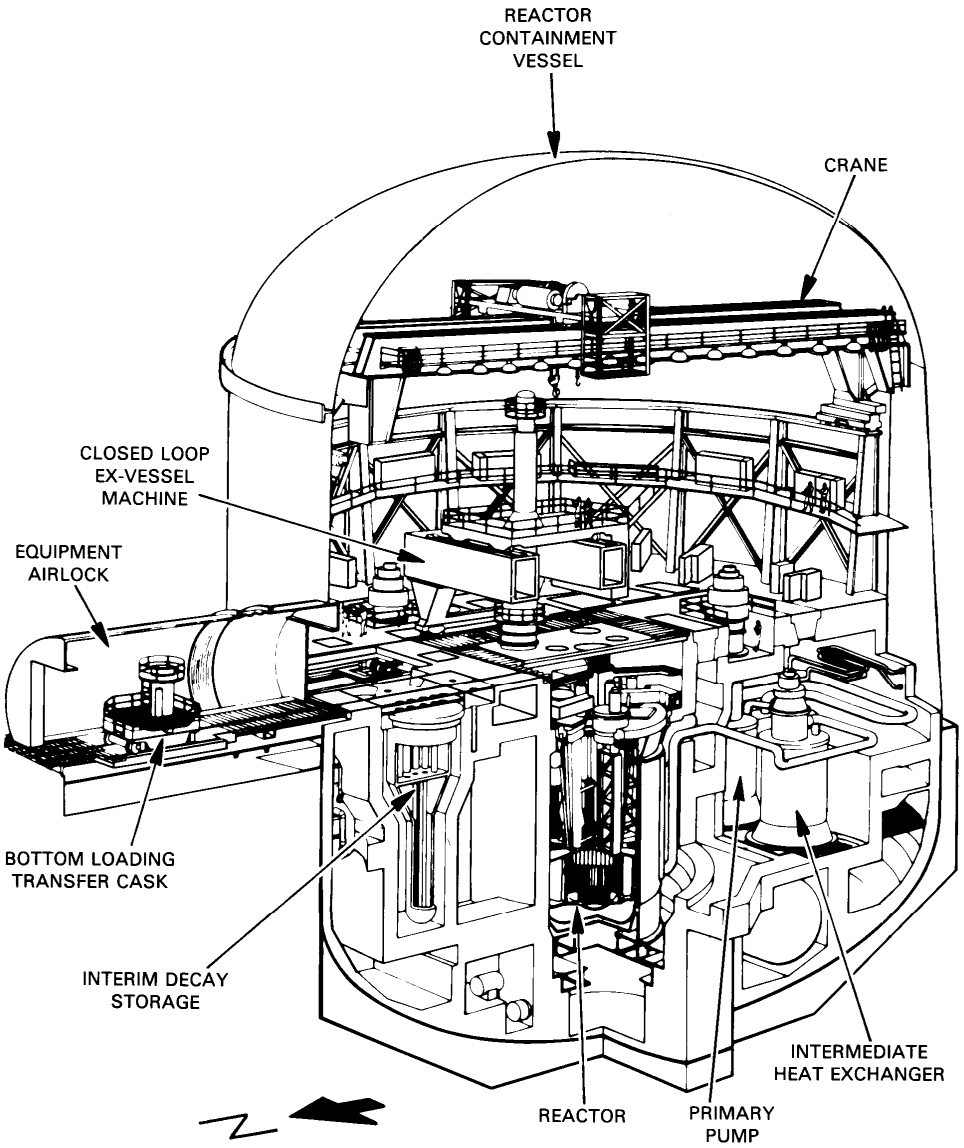
- Conceptual design studies 1966-1969
- PSAR submitted by HEDL in September 1970
- Initial construction authorization in September 1971; full construction authorization in March 1972
- ACRS letter in May 1973
- FSAR submitted by HEDL in December 1975
- Construction complete/Na fill 1978
- Criticality February 1980, full power October 1980
- Research operations April 1982 to April 1992
- DOE Shutdown order December 1993

FFTF Design

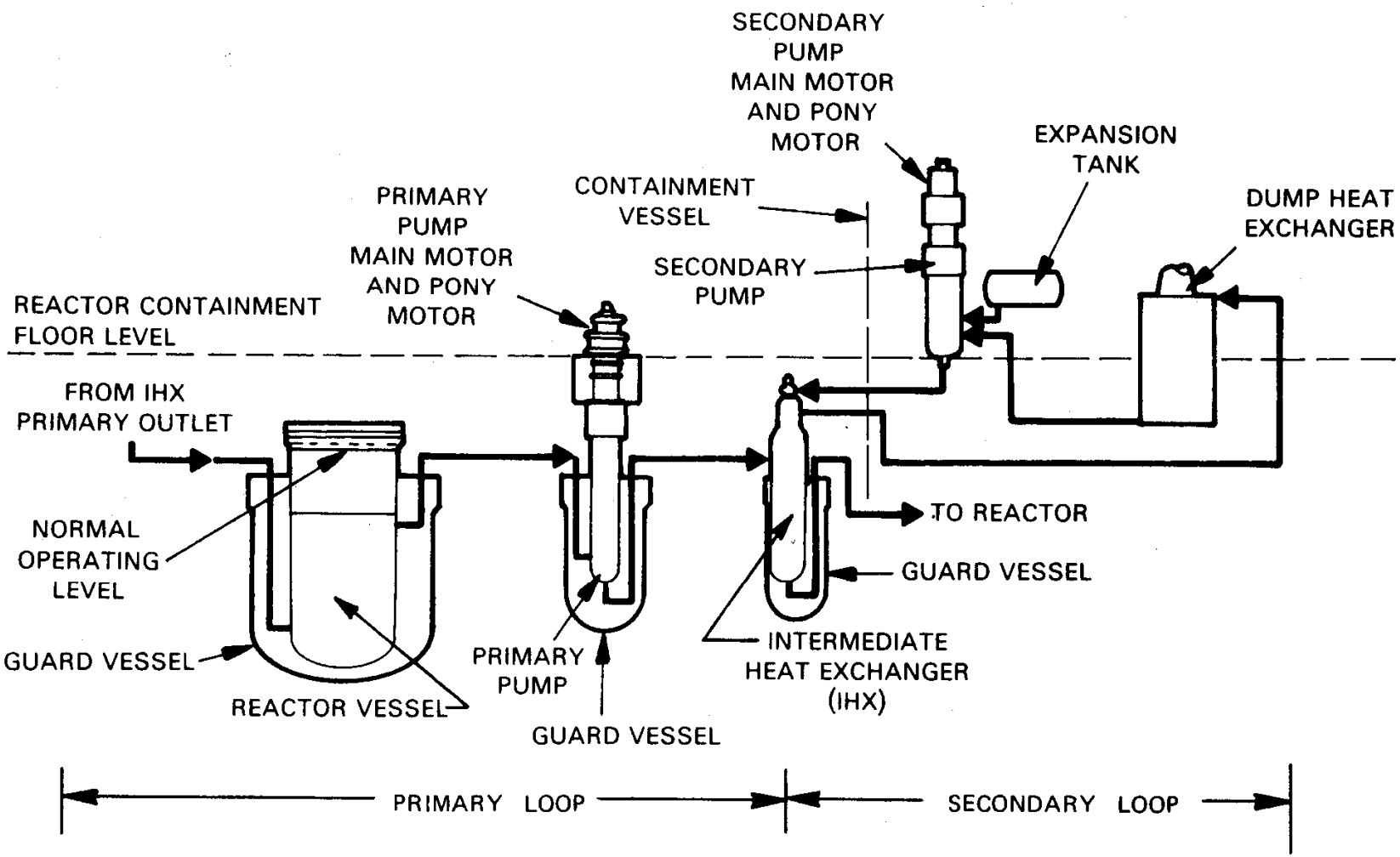
- **Mission: Provide a prototypic LMFBR operating environment for testing and development of fuels, materials, and components**
  - **Secondary: Develop design and construction experience**
- **400 MWt, MOX fuel (22% and 27% Pu), three loop primary system, three intermediate sodium loops to air dump heat exchangers**
- **Reactor core: 73 fuel assemblies, 9 control assemblies, 9 test assemblies**
- **Coolant inlet 680°F (360°C), outlet 980°F (527°C)**
- **10psi steel containment**
- **Two independent reactor shutdown systems (both by moveable rods)**
- **Forced and natural convection decay heat removal through three independent loops**
  - **Pony motors on primary and secondary pumps**
- **Core physics and structural design for inherent negative power and temperature reactivity feedbacks**

FFTF Site – Hanford, Washington

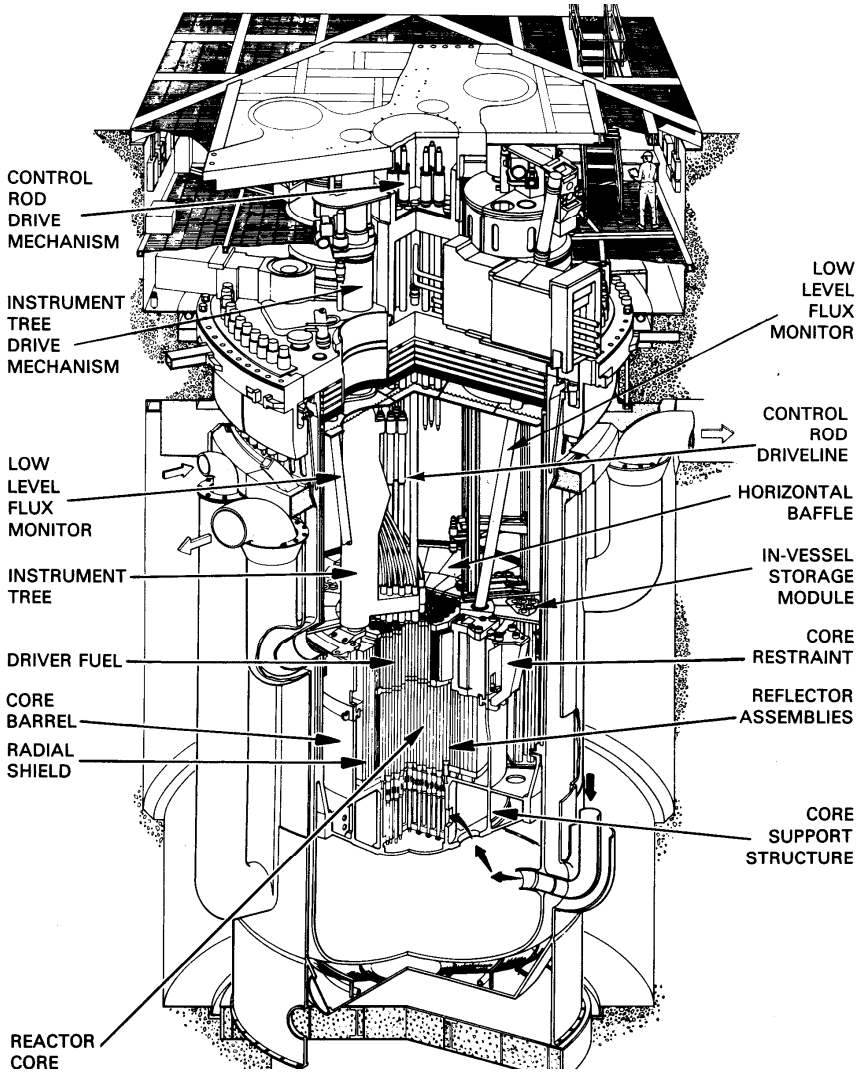
FFTF Containment Building View



FFTF Loop-Type Primary and Secondary Systems



FFTF Reactor and Vessel Design



FFTF Regulatory Review -- 1

- **As an AEC project, FFTF did not require licensing as for commercial LWR plants, but technical review by NRC was required**
  - **The depth and detail of the NRC review was similar to full licensing**
  - **Construction and operation permission; ACRS letters**
- **Site selection in 1968; site evaluation report prepared in 1969 and submitted for review in July 1970 (See "Chronology"). PSAR Sept. 1970. First meeting with NRC staff Nov. 1970, first ACRS meeting Dec. 1970**
- **Site issues: seismic (0.25g) and tornado (150 mph rotation)**
  - **Studies and analyses submitted to NRC and ACRS for review**
- **Hypothetical Core Disruptive Accidents (HCDA) received the greatest regulatory attention and review emphasis**
  - **Basis for evaluation of containment margins (10CFR100 offsite dose)**
  - **Project position: HCDA was not a design basis (150 MW-s margin evaluation basis)**
  - **NRC requested further study; response by HEDL and ANL**

FFTF Regulatory Review -- 2

- **A major part of the LMFBR safety base program was oriented to support FFTF regulatory review**
- **At HEDL**
  - **Transient Overpower (TOP) accident analysis (MELT computer code)**
  - **TOP fuel testing (TREAT)**
- **At ANL**
  - **Loss-of-Flow (LOF) accident analysis (SAS3A computer code)**
  - **LOF fuel testing (TREAT)**
  - **Post-accident Heat Removal (PAHR) analyses and experiments**
  - **Structural dynamics analysis and testing**
  - **Fuel Element Failure Propagation (FEFP) studies and experiments**
  - **Coolant dynamics analyses and experiments**
  - **Fuel dynamics analyses and experiments (OPERA)**
  - **High temperature materials properties**
  - **Fuel coolant interactions (FCI) analyses and experiments (OPERA)**

FFTF Regulatory Review -- 3

- **LMFBR safety base program activities also performed at ORNL, AI, GE, and W-ARD**
  - **Activities coordinated under HEDL technical direction**
- **ANL provided direct support to FFTF licensing**
  - **Preparation of technical reports of analyses, experiments, and tests for use as FSAR support documents**
  - **Participation in meetings with NRC staff and ACRS**
- **Regulatory review for construction nominally concluded with the May 1973 ACRS letter, but open issues continued to receive attention**
  - **HCDA energetics**
  - **Design fallbacks, including sealing the reactor head compartment and ex-vessel core melt retention**
  - **Piping integrity; provision for pipe break mediation design, and surveillance and in-service inspection**
  - **Natural convection cooling and emergency power**

FFTF Regulatory Review -- 4

- **Through 1976, HEDL and ANL continued to meet with NRC staff and ACRS**
- **ANL supplied technical support for resolution of the HCDA energetics and core melt retention issues**
  - **In 1974, NRC concurred with the ANL assessment that HCDA energetics would not exceed FFTF capability. Also, NRC concurred that sealing of the head compartment would not significantly improve containment margins**
  - **In 1975, NRC recommended that construction could be completed without addition of an ex-vessel core catcher**
- **The FFTF FSAR was issued in March, 1976, followed by an NRC staff review**
- **The NRC staff review continued, and the Final Safety Evaluation Report was issued in August 1978. The SER stated that the major unresolved issues were natural convection verification, control room habitability, piping integrity, and containment margins**
- **Natural convection verification testing was performed during start-up**

FFTF Regulatory Review -- 5

- **A safety-grade system to provide control room isolation upon detection of unacceptable levels of sodium aerosols or radioactivity was added**
- **The piping integrity and containment margin issues were resolved without design changes by additional analyses and information submittals to NRC**
- **ACRS concurred with NRC findings in a November 1978 letter**
- **Coolant filling was accomplished in 1979, and fuel loading began**
- **First criticality was in February 1980, and power operation began in October 1980**

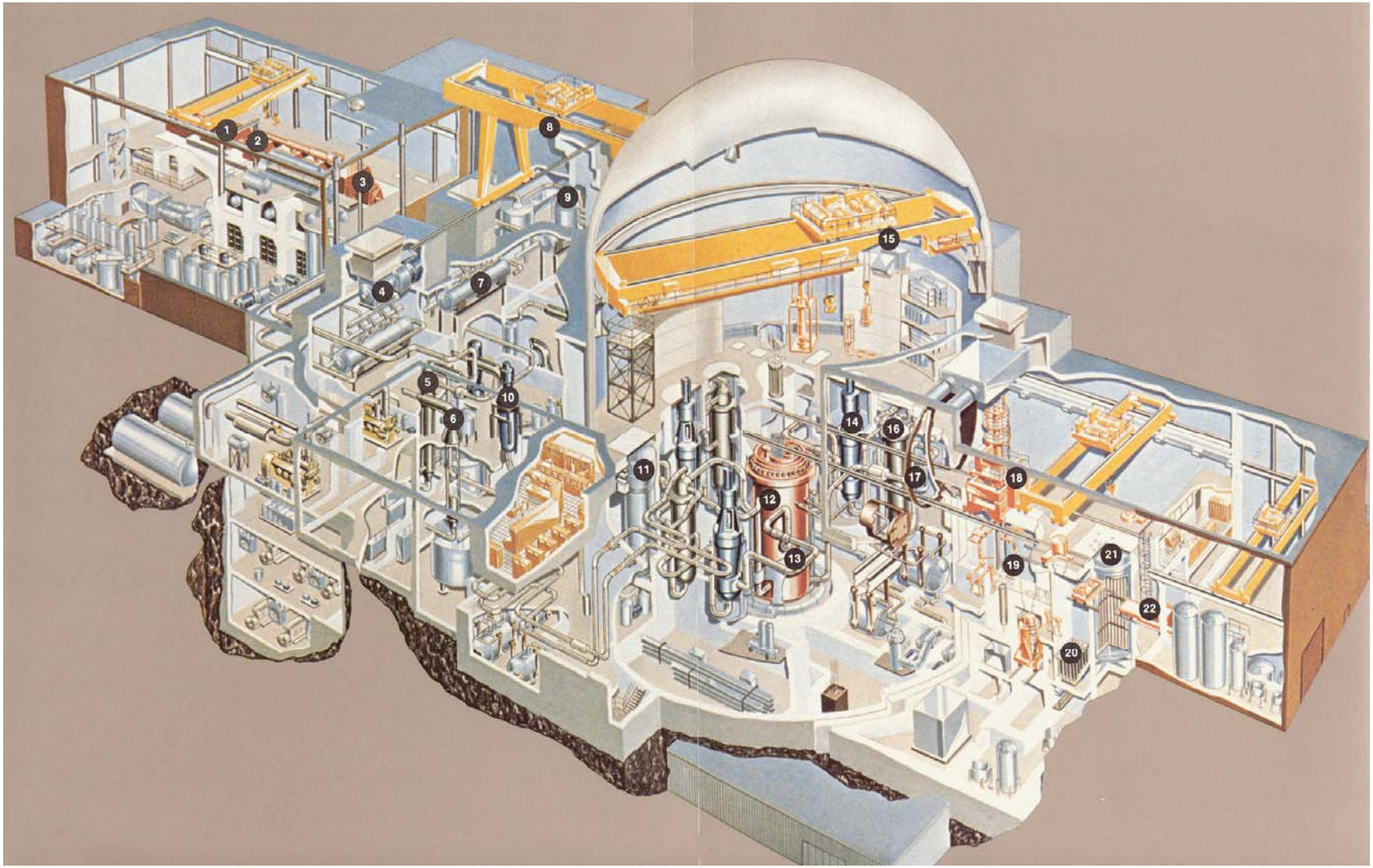
Clinch River Breeder Reactor Plant (CRBRP) Chronology

- **June 1970. U.S. Congress enacts Public Law 91-273 authorizing AEC to design, construct, and operate an LMFBR demonstration plant.**
- **January 1972. Joint proposal by Commonwealth Edison and TVA accepted.**
- **March 1972. Two not-for-profit organizations established. Project Management Corporation (PMC) for project management, and Breeder Reactor Corporation (BRC) for utility industry liaison.**
- **November 1972. Westinghouse-Advanced Reactor Division (W-ARD) selected as lead reactor manufacturer, Burns and Roe (B&R) as A/E.**
- **February 1973. Initial work authorization. AI and GE added to team.**
- **January 1974. AEC/W-ARD contract signed.**
- **PSAR submitted April 1975 (Updated through Rev. 77 May 1983)**
- **Licensing suspended: commercial reprocessing banned March 1977**
- **Licensing resumed September 1981**
- **NRC Safety Evaluation Report (NUREG-0968) March 1983, ACRS letter**
- **Funding stopped October 1983**

CRBRP Design

- **Mission: Demonstrate the safe and reliable operation of an LMFBR in a utility environment. Demonstrate LMFBR economics, and the transition from technology development to commercial operation**
- **975 MWt, 380 MWe (gross), MOX fuel (19% and 27% Pu), three loop primary system, three intermediate sodium loops to steam generators**
- **Reactor core: 198 fuel assemblies (108 inner/90 outer), 19 control assemblies (15 primary/4 secondary), 150 radial blanket assemblies**
- **Coolant inlet 730°F (388°C), outlet 995°F (535°C)**
- **10psi steel containment**
- **Two independent reactor shutdown systems (both by moveable rods)**
- **Decay heat removal through three independent loops**
  - **Pony motors on primary and secondary pumps**
  - **Auxiliary decay heat removal through water side of SG**
- **Direct decay heat removal system independent of HTS loops**
- **Core physics and structural design for inherent negative power and temperature reactivity feedbacks**

Clinch River Breeder Reactor Plant (CRBRP)



CRBRP Heat Transport and Power Conversion Systems

CRBRP Design

CRBRP Design

CRBRP Design

CRBRP Regulatory Review -- 1

- **CRBRP was licensed as a commercial power reactor by NRC**
  - **Project suspended in accord with the Presidential order in 1977**
  - **Licensing activities continued to obtain the equivalent to a construction permit in 1983**
- **Site selection in 1972, environmental report early 1975, PSAR April 1975**
- **Site issues: seismic (0.18g) and tornado (290 mph rotation)**
  - **Consistent with other TVA sites**
- **As for FFTF, HCDAs received much regulatory review attention**
  - **Early agreement (1976) between NRC and the project that HCDAs would <u>not</u> be a design basis for containment**
  - **However, the role of severe accidents and characterization of their consequences dominated the attention of the interveners, the regulators, and the project**
  - **Licensing, and treatment of severe accidents, set the critical path for construction**

CRBRP Regulatory Review -- 2

- **PSAR preparation was the responsibility of W-ARD**
  - **General design criteria**
  - **Preliminary design**
  - **Design basis event (DBE) analyses for PSAR Ch. 15**
- **The LMFBR base program, and particularly ANL provided significant resources to address design and licensing issues**
  - **TREAT fuel testing; basic phenomenological test measurements and prototypic TOP and LOF transient tests**
  - **LOF and TOP accident analyses (SAS3D computer code)**
  - **Coolant and structural dynamics tests and analyses**
  - **Post-accident heat removal analyses**
- **ANL provided direct support to CRBRP licensing**
  - **Preparation of technical reports of analyses, experiments, and tests for use as PSAR support documents**
  - **Participation in meetings with NRC staff and ACRS**

CRBRP Regulatory Review -- 3

- **HCDA issues compared to FFTF**
  - **Bigger reactor (975 MWt vs 400 MWt), more fuel**
  - **Positive coolant void reactivity worth (~3$ vs ~0$)**
  - **LOF sequence bounded energetics as in FFTF, but because of the positive coolant void worth, cladding failures and fuel melting occurred at higher power than in FFTF (~10 P_o vs ~1P_o)**
  - **Higher power LOF caused other phenomena in accident sequences that raised energy releases in analyses**
- **CRBRP structural limits (vessel head bolt strength) corresponded to an accident energy release of 661 MW-s: project structural margin beyond the design basis (SMBDB)**
  - **NRC (LANL) performed independent analyses: 1200 MW-s**

CRBRP Regulatory Review -- 4

- **Ultimately,**
  - **The ASLB ruled against intervenor's contention that HCDAs should be a design basis**
  - **NRC staff stated: "It is our current position that the probability of core melt and disruptive accidents can and must be reduced to a sufficiently low level to justify their exclusion from the design basis accident spectrum."**
  - **CRBRP project, with ANL support, built a technical case to justify exclusion**
  - **CRBRP met licensing requirements for construction without inclusion of HCDAs in the design basis**

SAFR and PRISM Background and Chronology

- **Beginning in the early/mid 1980's, DOE funded conceptual design studies for modular advanced liquid metal reactor plants**
  - **At Rockwell International, the Sodium Advanced Fast Reactor (SAFR)**
  - **At General Electric, the Power Reactor - Inherently Safe Module (PRISM)**
- **Preliminary Safety Information Documents for SAFR and PRISM were submitted to NRC in November, 1986**
- **The initial SAFR and PRISM concepts focused on innovative design approaches for economics and safety**
  - **Design simplification based on passive safety performance**
- **DOE subsequently (ca 1988?) selected the PRISM concept for continued development in the Advanced Liquid Metal Reactor (ALMR) program**
- **NRC issued pre-application SERs for SAFR (1991) and PRISM (1994)**
- **DOE support for PRISM design studies ceased with the cancellation of the ALMR program in 1994**

Rockwell International SAFR Design

- **Multiple (4) power units per site co-located with a spent fuel processing facility**
- **900 MWt reactor, U-Pu-Zr metallic fuel, pool type primary system, two intermediate loops**
- **60 year plant life**
- **Reactor core: 96 driver fuel assemblies, 46 internal and 48 radial blanket assemblies, 6 control assemblies, 3 safety assemblies**
- **Coolant outlet 950°F (510°C), inlet 675°F (357°C)**
- **Inherent response for emergency reactor shutdown**
  - **Inherent reactivity feedbacks in temperature and flow transients**
  - **Self-Actuated Shutdown System (SASS); thermally-activated magnetic latch release**
- **Two natural circulation decay heat removal systems**
  - **Direct heat removal from hot sodium pool (DRACS)**
  - **Ambient air cooling of guard vessel (RACS)**
- **Compact containment building design**

SAFR Module View

SAFR Module Elevation

SAFR Pool-Type Primary System

SAFR Decay Heat Removal Systems

General Electric PRISM Design

- **Multiple power modules (6) co-located with a spent fuel reprocessing facility (Module reactor size grew as the concept evolved, to 840 MWt in 1995. The SER looks at the 471 MWt design)**
- **840 MWt, U-Pu-Zr metallic fuel, pool-type primary system, two intermediate loops**
- **Reactor core (burner): 192 fuel assemblies, 10 control assemblies, 3 safety assemblies**
- **Coolant outlet 930°F (500°C), inlet 680°F (360°C)**
- **One safety grade automatic reactor protection system, with a manually operated safety grade ultimate shutdown system (3 safety rods)**
  - **Inherent accommodation of ATWS transients without core melt, significant reactivity addition, or large radiological release**
- **Shutdown cooling by turbine bypass with emergency removal systems**
  - **Air cooling system (ACS) on the steam generator shell**
  - **Primary sodium auxiliary cooling system (PSACS); reqs. valve action**
  - **Reactor vessel air cooling system (RVACS)**
- **Compact containment shell design**

PRISM Module Arrangement

PRISM Reactor Building

PRISM Reactor Vessel

PRISM Heat Removal Systems

SAFR/PRISM Safety Design/Licensing

- **By the time (mid 1980's) SAFR and PRISM were being designed, technical understanding of severe accident progression had advanced to a point that reactor designs could be specified to greatly reduce the consequences of accident initiators that led to severe accident conditions (coolant boiling, fuel melting, cladding failure) for FFTF and CRBRP designs**
  - **Beyond design basis, double fault accidents with failure of the automatic scram system; loss of coolant flow, reactivity addition, or loss of normal heat rejection**
- **Full scope, integral testing in EBR-II and FFTF (ca 1986) provided confirmation of design features performance that limited accident consequences to elevated temperatures short of coolant boiling or fuel melting, with margins**
  - **Inherent, passive reactor response provided by designs with negative reactivity feedbacks to reduce power and with natural circulation cooling to remove heat**
- **For SAFR and PRISM, the FFTF and CRBRP severe accident initiators cause elevated coolant temperatures that trigger inherent protection**

Safety Design and Licensing Implications

- **The double fault accident sequences that led to core disruption in FFTF and CRBRP were considered as challenges to containment margins**
  - **The same initiators in current sodium fast reactor designs do not produce conditions that approach containment margins**
  - **Fuel cladding, reactor vessel, and containment building integrity are maintained**
- **These performance characteristics were reflected in the SAFR and PRISM designs, which included simplifications of shutdown, cooling, and containment features**
- **To achieve core melt conditions in modern liquid metal reactor designs, it is necessary to assume accident initiators with very low probability**
  - **For example, “triple” fault initiators or extremely large earthquakes**
  - **Proper characterization of such initiators requires probabilistic analysis and a risk-based assessment**
- **Passive safety mechanisms effectively increase containment margins and public safety**

Lessons Learned -- 1

- **FFTF and CRBRP experiences demonstrate that a liquid sodium-cooled reactor plant can be licensed**
  - **FFTF underwent NRC review, and CRBRP construction was approved**
  - **Requires interpretation of 10CFR50 App. A in view of low pressure, chemically active liquid metal coolant, compliance with intent, and possibly additional criteria to cover physical characteristics not considered in the LWR criteria**
  - **Compliance with defense-in-depth principles in design specifications**
- **Effective management of the licensing process can be achieved**
  - **Recognition by both applicant and regulator of each other's responsibilities**
  - **Focus on top-level goals and actions necessary to fulfill both regulatory and project requirements**

Lessons Learned -- 2

- **Innovative SAFR and PRISM safety design features promise increased margins to beyond-design-basis safety limits, high reliability safety systems, and safety design simplification**
  - **Reliance on inherent mechanisms for power and reactivity control and natural circulation heat removal**
- **Early recognition of potential issues of significance and establishment of a framework for resolution**
  - **Example: Passive reactivity feedbacks and natural circulation heat removal. Proof of reliability by test?**
  - **Example: Severe accident prevention and consequence mitigation. Risk informed assessment?**
  - **Others**
- **Begin technical interactions at the earliest possible opportunity**

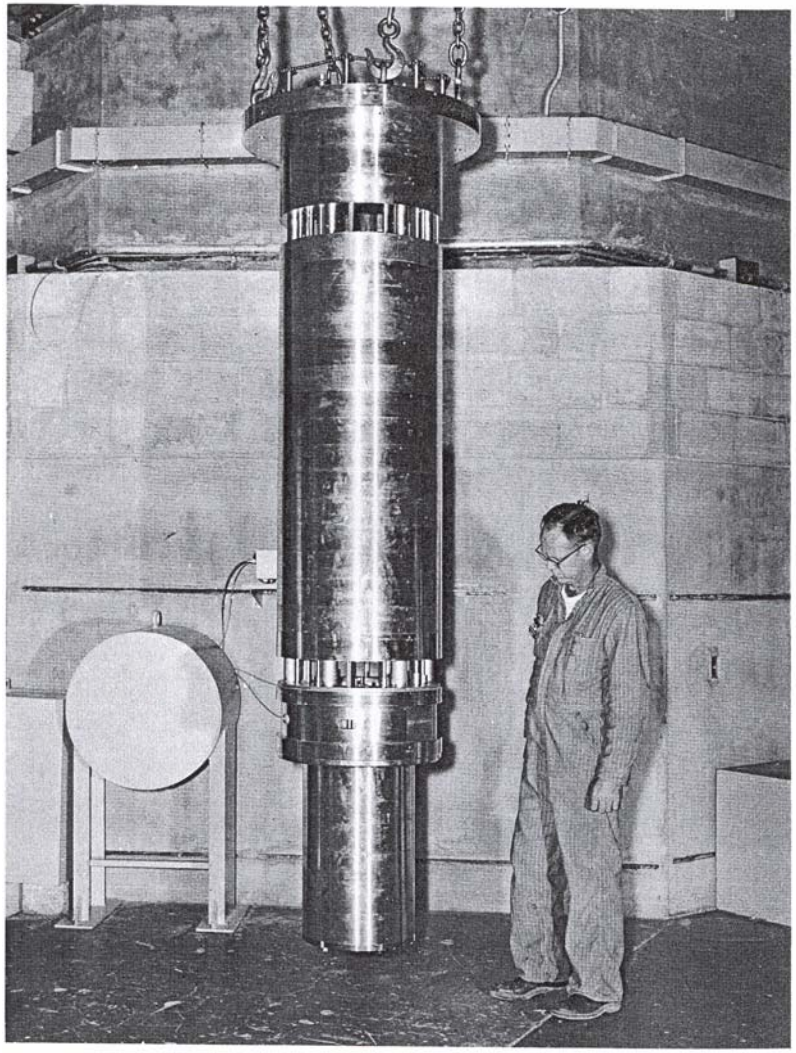
Domestic and International Accidents and Lessons Learned

- **EBR-I Power Excursion**
- **Fermi-1 Fuel Assembly Inlet Blockage**
- **BN-350 Steam Generator Leak**
- **Phenix Reactivity Anomaly**
- **SuperPhenix**
- **Monju Sodium Leak**

Experimental Breeder Reactor-1 (EBR-I)

- **Experimental Breeder Reactor-I (EBR-I) was built at the National Reactor Testing Station in Idaho to demonstrate fuel breeding**
- **From 1951 to 1963, EBR-I was operated with four core designs to demonstrate breeding and to develop an understanding of liquid metal fast reactor performance**
  - **Pu breeding demonstrated by February, 1952**
  - **Testing platform for reactor physics, fluid dynamics, and power generation**
  - **1.4 MWt, generated 200 kW of electricity for NRTS**
- **NaK cooled, U-235 (94% enriched) metal fuel in stainless steel cladding**
  - **217 pin locations on a triangular o.494 in. pitch; 0.384 in. fuel OD; 0.448 in. cladding OD; 8.5 in. core height**
  - **227°C inlet, 316°C outlet, 20 psig**
  - **Mk I and II cores used top support, above-core shield plates, and a bottom tube plate to position fuel pins**
  - **Mk III (and IV) core used hexagonal tubes and wire wraps for fuel pin positioning**
  - **Mk IV core used Pu fuel (1962)**

EBR-I Reactor



(Reactor Shield in Background)

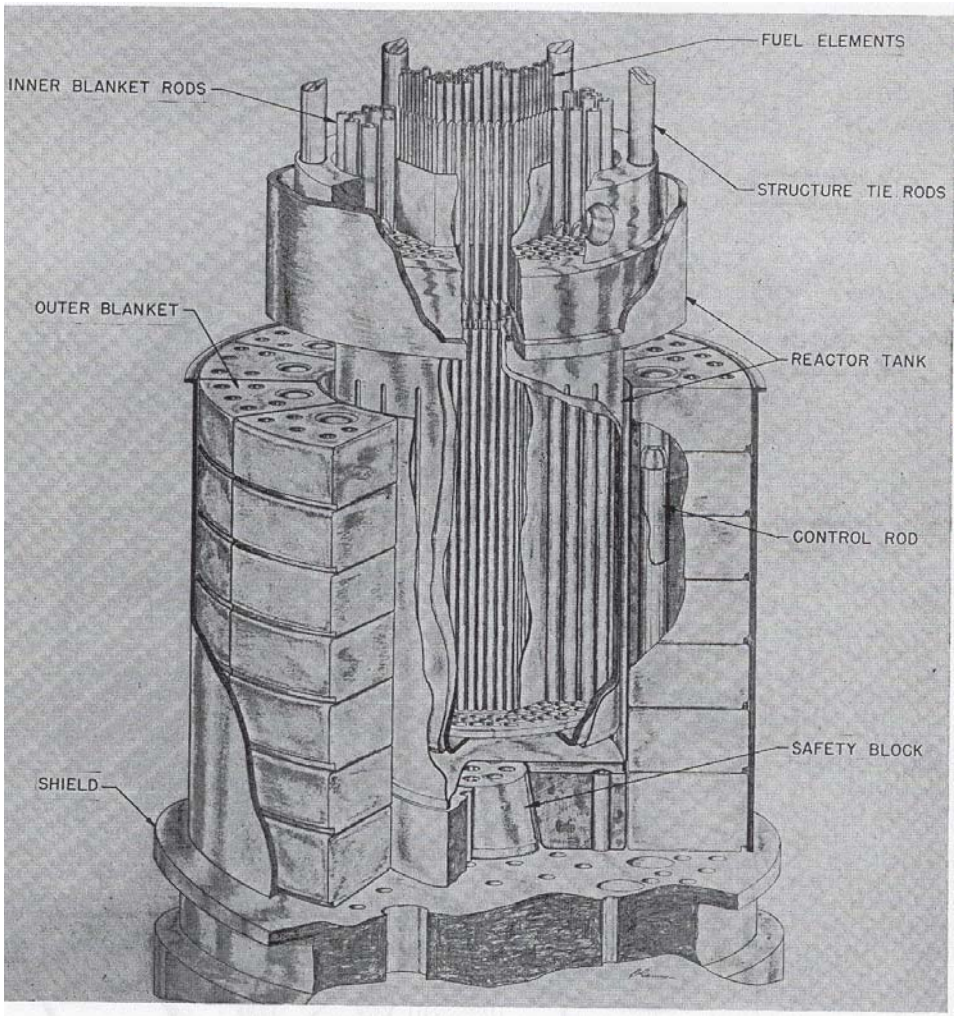


Figure 1. Cut-away showing EBR construction

EBR-I Core Mk I and II

EBR-I Mk I and II Above-Core Shield Plate

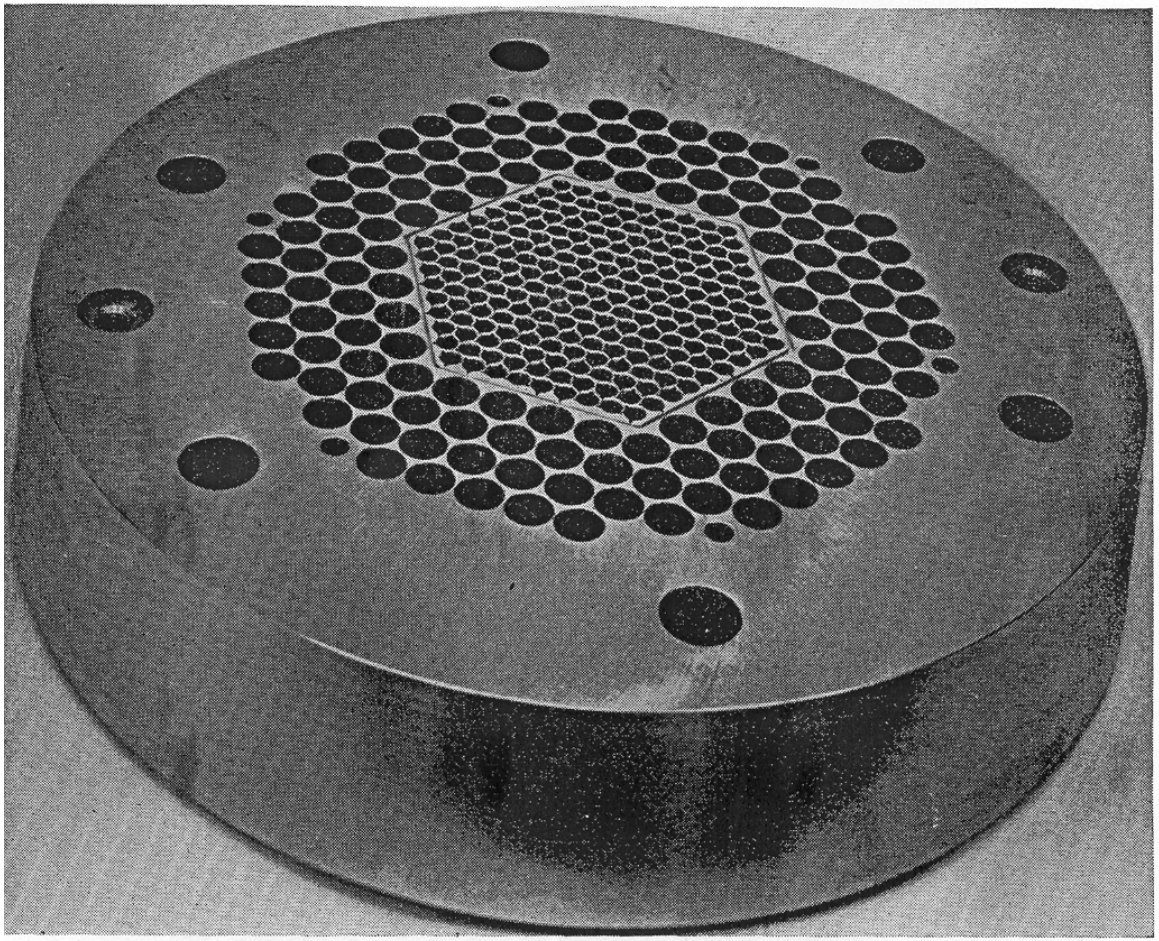


Figure 2a. Stainless steel holding plate for core fuel elements and inner blanket rods

Fuel element holes 0.460 in. OD (Cladding 0.448 in. OD)

EBR-I Accident

- **The Mk II core loading was found to have a prompt positive power coefficient, especially for very low power/very low flow conditions**
- **In November, 1955, during a test to investigate the prompt positive component of the power coefficient, an unanticipated power excursion resulted in fuel melting.**
  - **Repeat of a previous test, with additional instrumentation**
  - **60 s initial period from 50 W, no coolant flow**
  - **Excursion during 500 s to < 1 s period, ~10 MW (est.)**
  - **Scram by control rods failed to terminate the excursion; manual scram by blanket (reflector) drop**
  - **After the test, radiation alarms upon flow restart**
- **Subsequent investigations identified fuel rod bowing as the source of the positive reactivity feedback**
  - **Inward pin bending in the core due to radial power profile and insufficient radial support to prevent compaction**
- **Core was replaced and operation continued through 1963**
  - **Mk III and IV cores used wire wraps and hex tubes for radial support**

EBR-I Mk III Core Loading

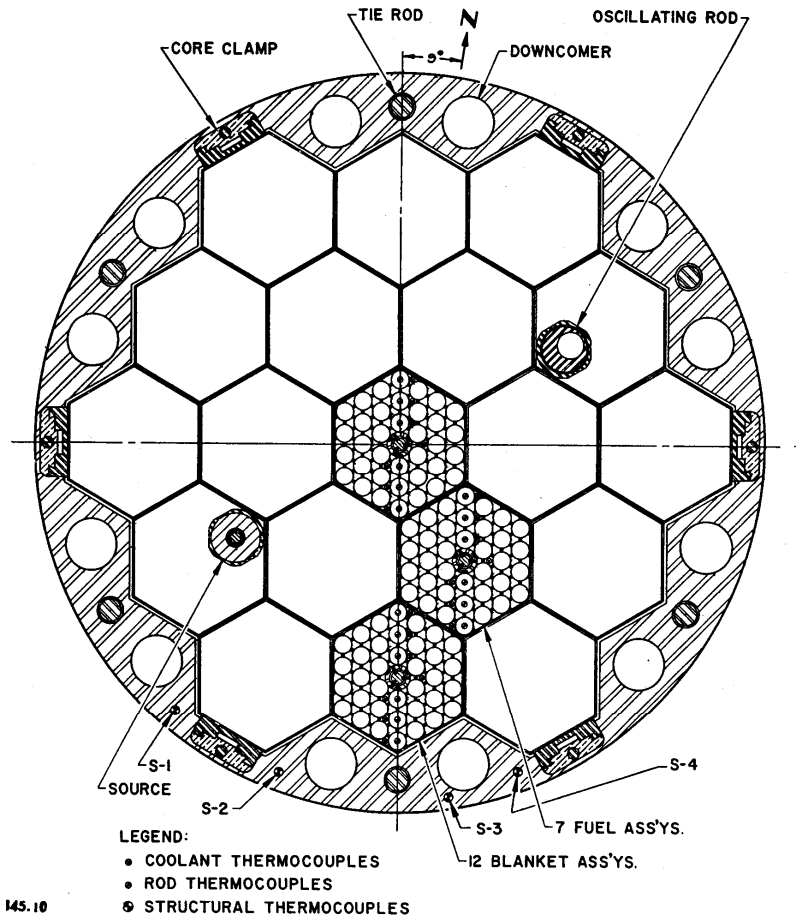


Figure 10. Plan view of EBR-I, Mark III at core centerline

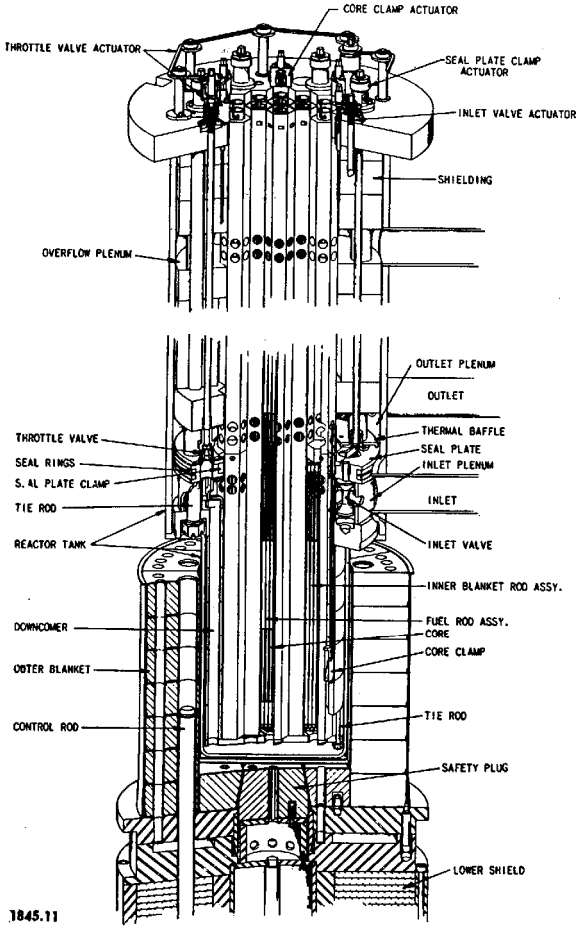


Figure 11. EBR-I, Mark III inner tank assembly

Plan View at Midplane

Elevation View

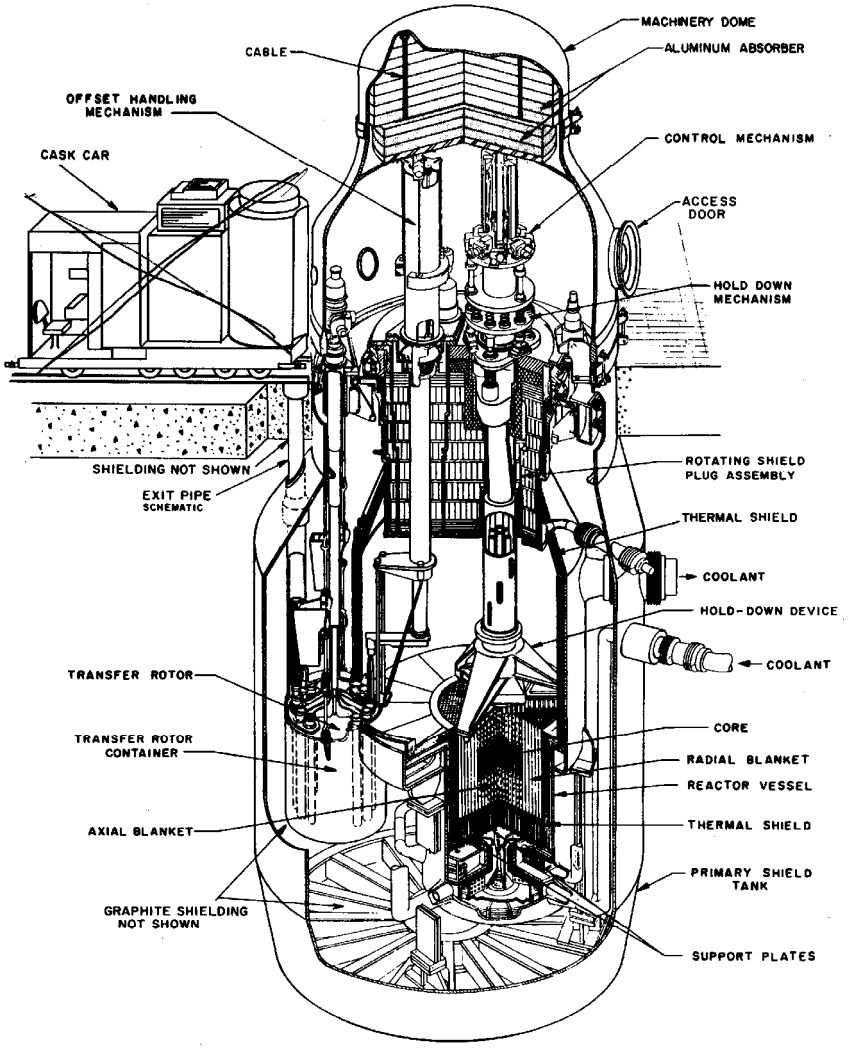
EBR-I Accident – Lesson Learned

- **The EBR-I core melt accident demonstrated the sensitivity of compact fast reactors to small changes in core fuel density**
  - **Sensitivity magnified in a very small, high enrichment core like EBR-I**
- **Subsequent fuel assembly designs have used wire wraps and hexcans in the U.S. (grids in Europe) to provide control of fuel geometry**
  - **Must be sufficiently 'loose' to accommodate thermal expansion and swelling, but 'tight' enough to provide support and positioning**
  - **Trend to bottom support of fuel pins and free axial expansion upward within the hexcan**
- **Hexcans are usually supported at the bottom in the core grid plate, and constrained radially in a manner to accommodate hexcan swelling and creep**
  - **Early designs used 'free bowing' concept (EBR-II) to give room for withdrawal in refueling; accommodation of steel swelling**
  - **Later designs (FFTF) used a 'limited free bow' design to manage hexcan bending to assure a negative power coefficient**

Fermi-1

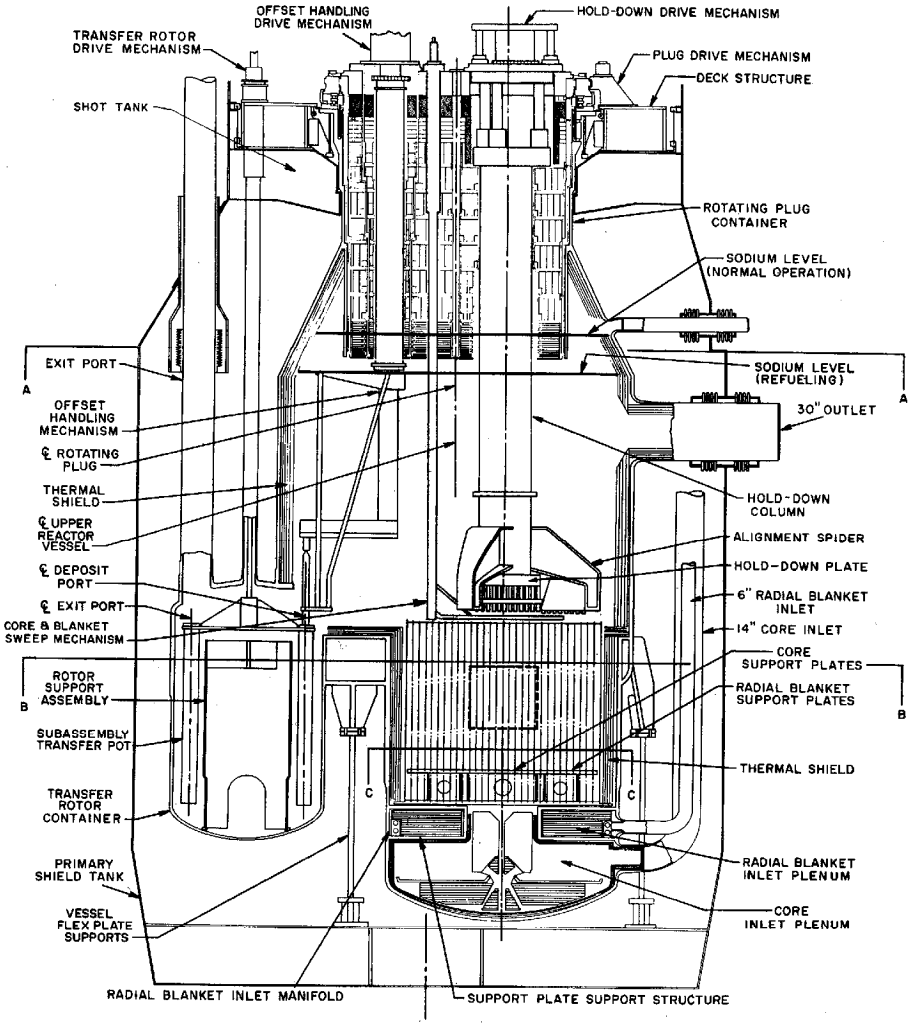
- 200 MWt power station located on the western shore of Lake Erie south of Detroit
- Designed by Atomic Power Development Associates (APDA) and constructed by Power Reactor Development Co. (PRDC) for Detroit Edison
  - Critical August 1963, first power August 1966
  - Sodium cooled, 288°C inlet, 427°C outlet, 120 psia
  - Metal fuel, Zr cladding 0.158 in. OD, 31 in. height, square pin pitch
- Subassembly flow blockage and fuel melting accident during power ascension on October 5, 1966
  - Metal fuel core removed and replaced with oxide core; full power 1969
- Operation ceased in 1972

Fermi-1 Primary System



PERSPECTIVE VIEW OF REACTOR

Fermi-1 Reactor Vessel



REACTOR VESSEL (ELEVATION)

Fermi-1 Core Inlet Plenum and Melt-Down Section

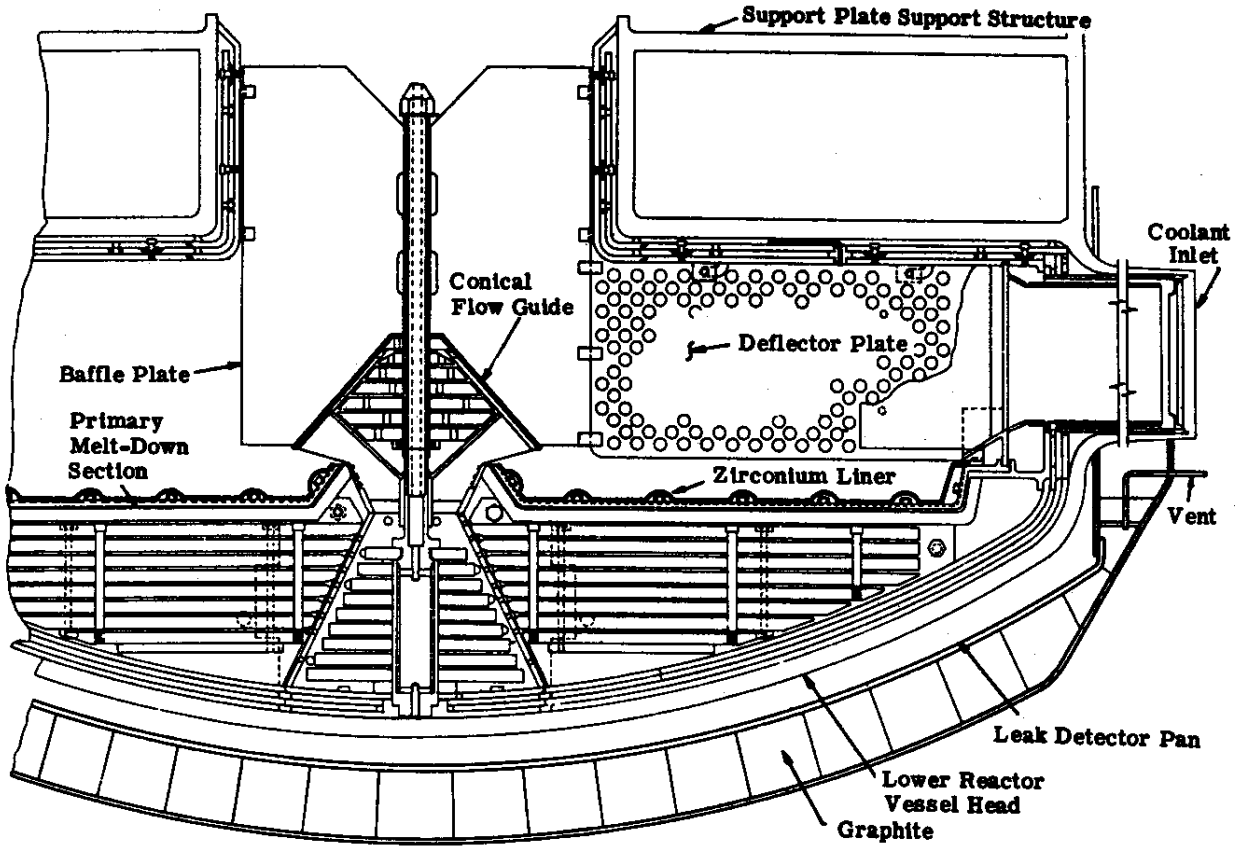


FIG. 7 MELT-DOWN SECTION IN LOWER REACTOR VESSEL

Fermi-1 Fuel Assembly

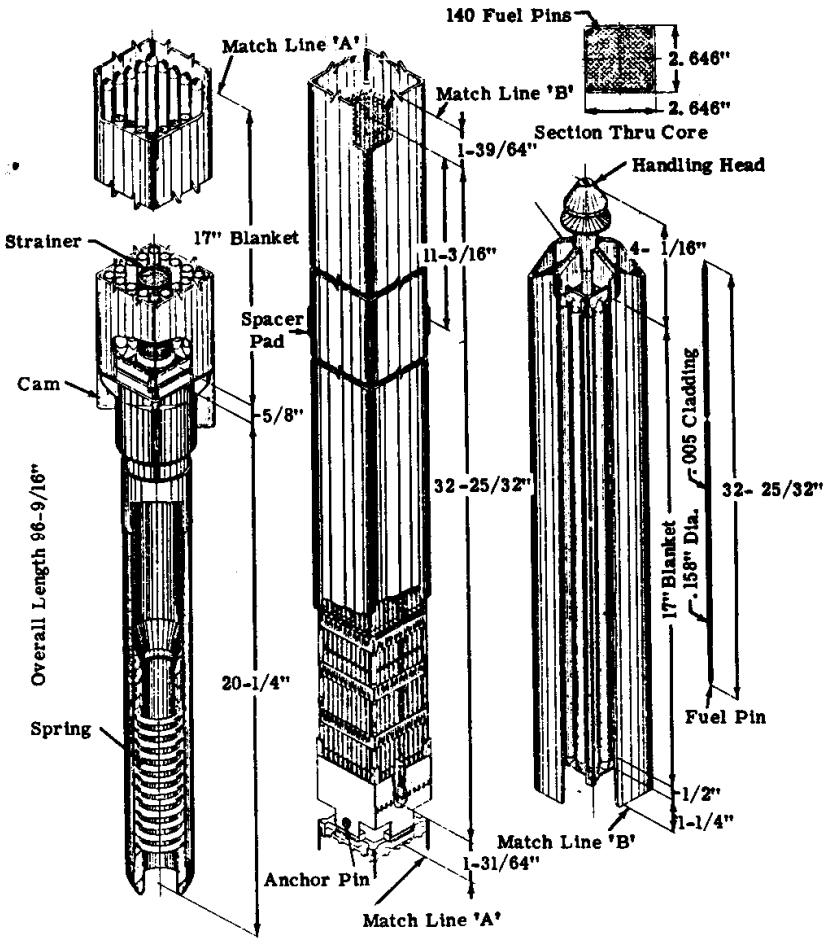


FIG. 3 CORE SUBASSEMBLY DESIGN

Fermi-1 Fuel Melting Accident

- **Prior to October 5, high fuel assembly outlet thermocouple readings had been observed during low power operations**
  - **Assemblies with abnormal temperature readings were relocated to positions under different thermocouples**
  - **The location(s) of the high temperature readings changed on each start-up, but not in correlation with the assembly movements**
- **On Oct. 5 during a power ascension at 34 MWt, building radiation alarms sounded, indicating fuel damage**
  - **The reactor had previously operated at 100 MWt without problems**
- **Subsequent investigations revealed fuel melting in two adjacent assemblies**
  - **Another adjacent assembly was bent, with no internal damage**
- **A ‘foreign object’ was found in the inlet plenum, which later proved to be a crumpled Zr plate from the melt-down section liner**
  - **The loose Zr plate had apparently been swept by flowing coolant to cover (partially or completely) the inlet nozzle of various assemblies during the multiple start-ups**

Fermi-1 Fuel Melting Accident – Lessons Learned

- **Assembly inlet nozzle designs since Fermi-1 have included multiple coolant inlet passages so that complete external blockages are 'impossible' by design**
- **Considerable research and testing of both external and internal blockages have been performed to understand and quantify the damage mechanisms and limits**
- **In the U.S., the assembly blockage scenario (external and internal) has been addressed in the assembly design (inlet flow diversity), in the inlet plenum design (coolant flow distribution and assurance of assembly supply), in the instrumentation design (detection by multiple thermocouples, delayed neutron detectors, gas tags), and in fuel handling equipment design (casks)**
- **Internationally, in some countries the fuel assembly blockage scenario has become a design basis accident**

BN-350

- **750 MWt power station located on the eastern shore (~ 2 mi. inland) of the Caspian Sea near Aktau (Shevchenko), Kazakhstan**
  - **130 MWe, 150 MWt desalination**
  - **Critical 1972, Power (partial) 1973, Shutdown 1994**
- **Power level during early operation (1973-1975) limited (350-550 MWt) by leaks in the steam generator evaporator tubes**
  - **Plant designed for 5 of 6 loop operation (One loop spare)**
  - **In each loop; two evaporators with bayonet tubes and two superheaters with U-tubes**
  - **816 bayonet tubes in each evaporator, 33 mm OD x 3 mm thick (originally 32x2) with a welded end cap and a 16 mm x 1.4 mm internal downcomer tube**
  - **In start-up tests, found leaks at the tube-sheet welds and the end cap welds**
  - **In operation, eight evaporator leaks through 1975 involving all but one of the loops (Loop No. 4 never had a leak)**
    - ***Three major leaks with extensive damage***

BN-350 Plant Layout

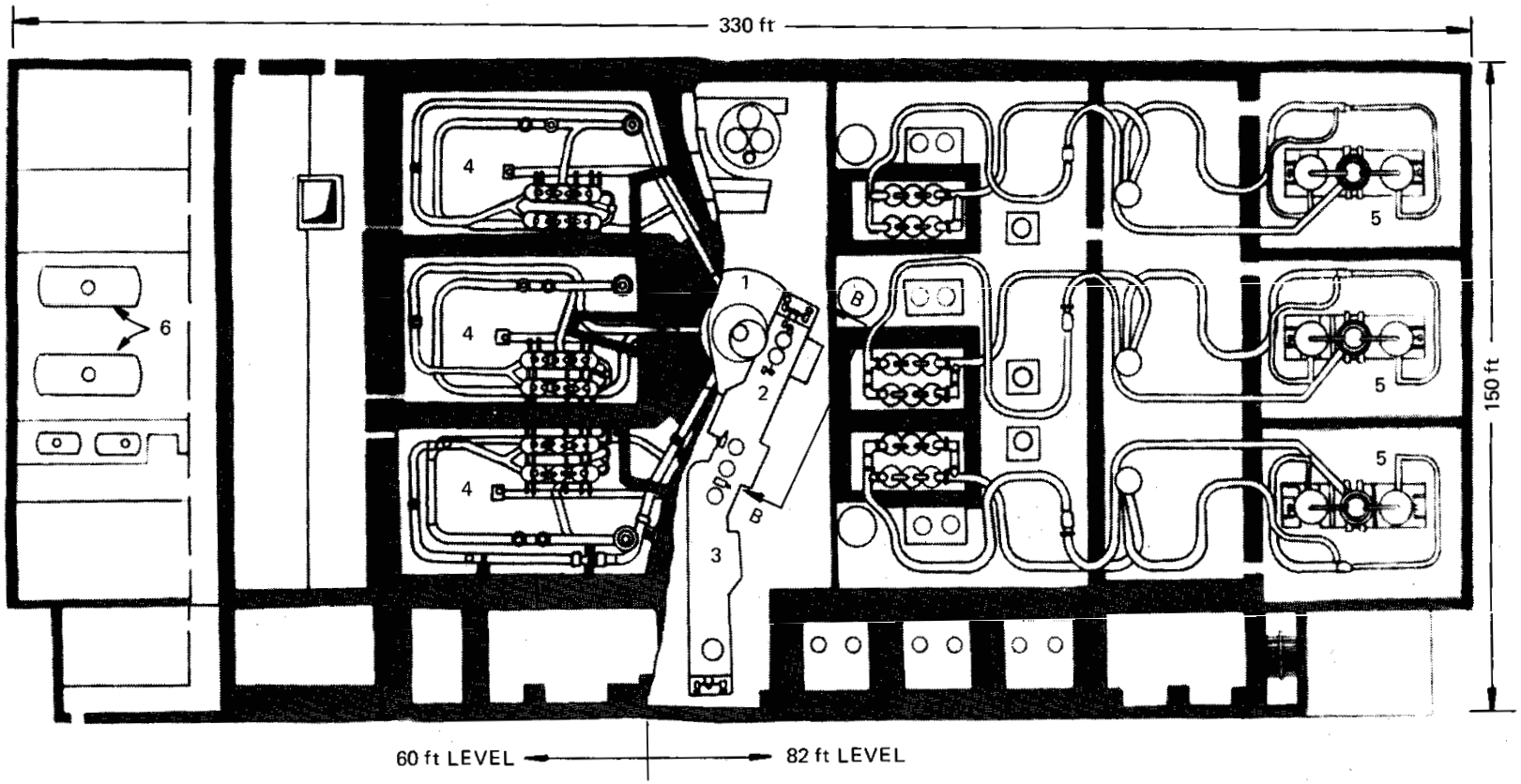
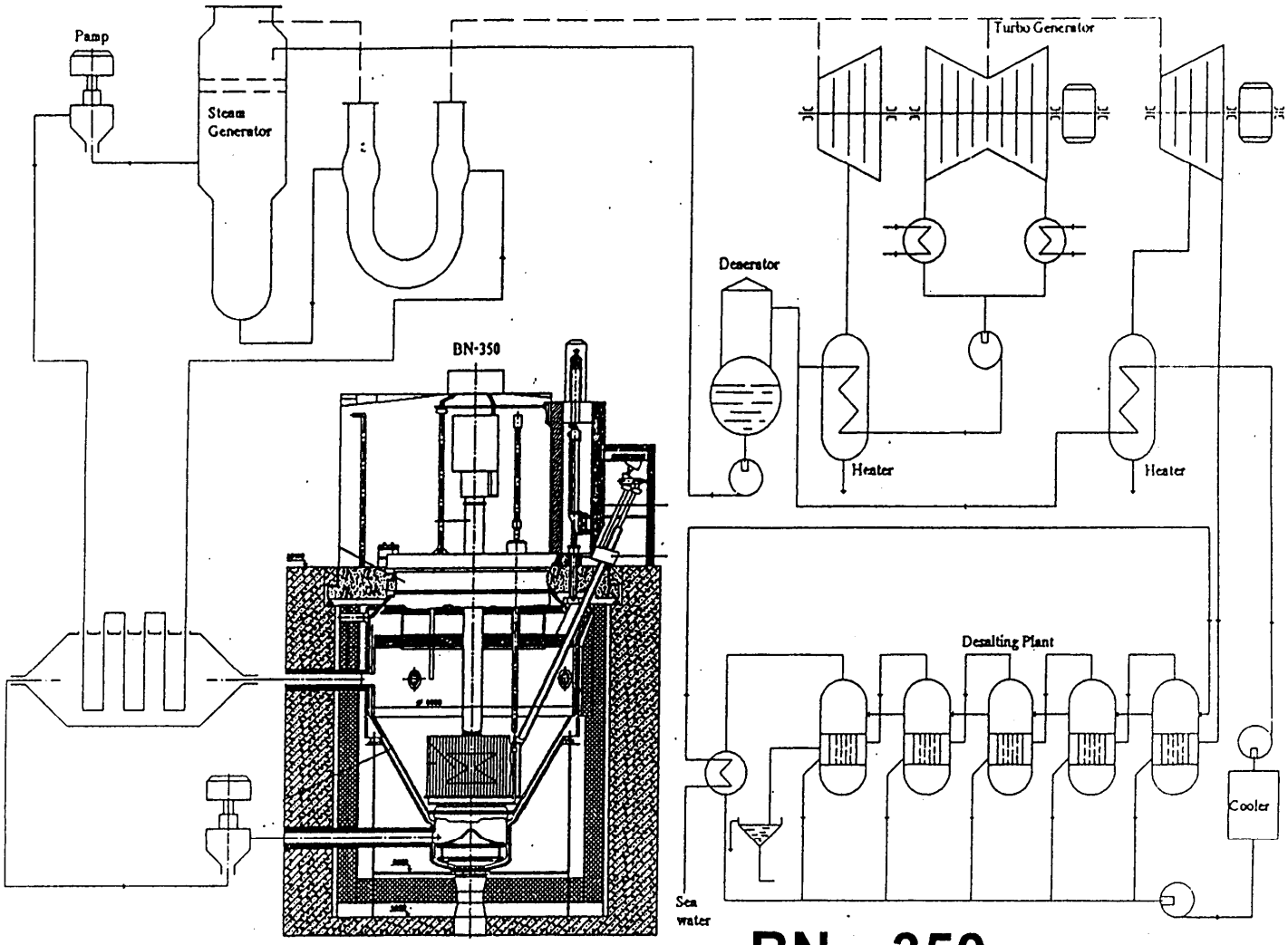


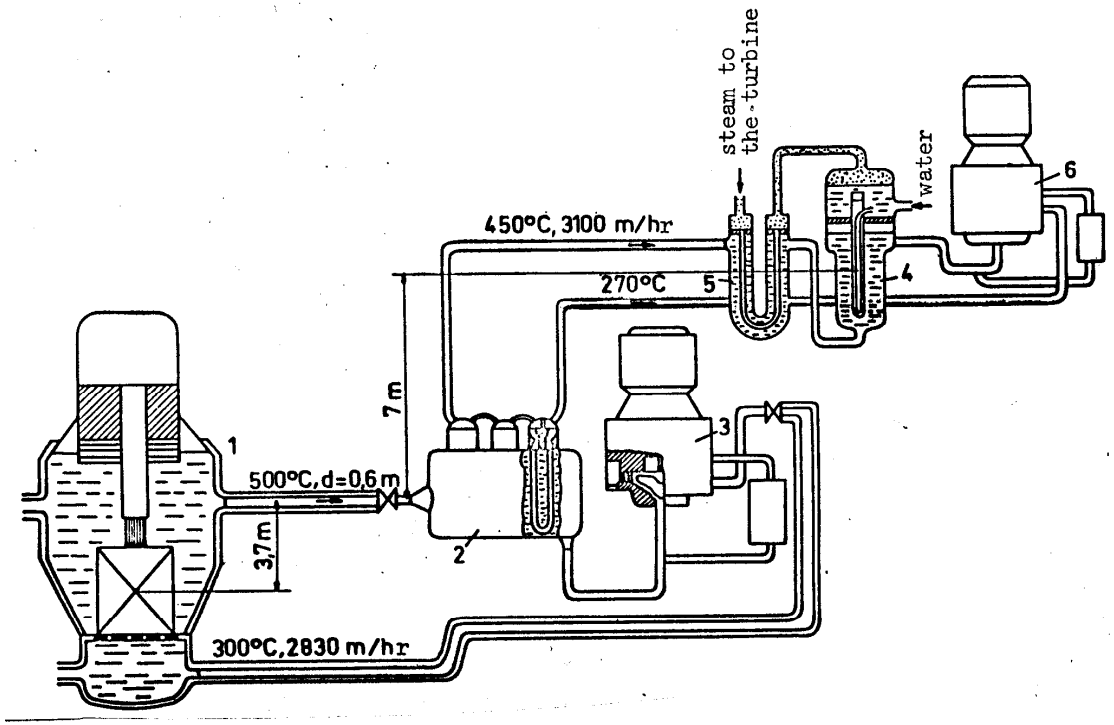
Fig. A.3.3. Plan View of BN-350, Reactor Building Only. Symmetrical layout—3 loops to the right of the reactor; 3 to the left of the reactor. 1—Reactor; 2—Refueling cell; 3—Wash cell; 4—Three of 6 IHX compartments, 2 shells in each compartment; 5—Three of 6 steam generator compartments; 6—Secondary dump tanks. For B-B see Fig. A.3.17.

BN-350 Flow Diagram



FLOW DIAGRAM OF THE NUCLEAR DESALTING COMPLEX **BN - 350**

BN-350 Coolant Loop (1 of 6)



1 - Reactor
2 - Exchanger
3 - Primary Pump
4 - Boiler
5 - Steam Superheat
6 - Secondary Pump

Fig. 1. Schematic diagram of a cooling loop of the energy-production system of BN350.

BN-350 Steam Generator and Evaporator Tube

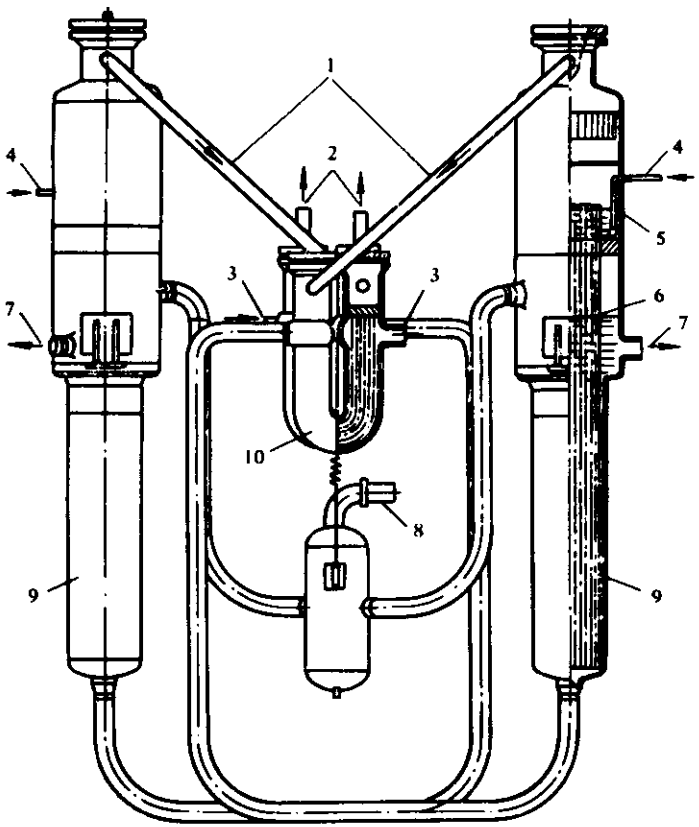
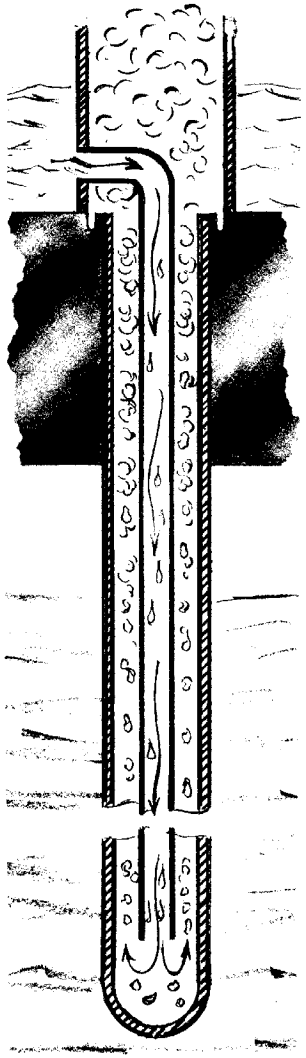


Fig. A.3.23. BN-350 Steam Generator. 1—Saturated steam inlets, 2—Superheated steam, 3,7—Sodium inlet and outlet, 4—Feedwater inlet, 5—Water level, 6—Sodium level, 8—Burst diaphragm, 9—Evaporator, 10—Superheater.



BN-350 bayonet steam generator

BN-350 Steam Generator Evaporator Experience

- **Through 1974, two major leaks and three smaller leaks**
  - **Initiated from end cap welds; micro-cracks in end cap weld seam zone, attributed to mechanical deformation at end cap manufacture**
- **In 1974, decision to re-tube all evaporators except Loop No. 4**
  - **By February 1975, three of five loops had been re-tubed**
  - **After 7 days of operation, one of the evaporators in the most recently re-tubed loop failed (No. 5), leading to the most significant leak (Balent report)**
  - **120 tubes failed, 800 kg water leak**
  - **This steam generator was dismantled and replaced with a Czechoslovakian steam generator**
- **It is claimed that safety systems (rupture disk and blowdown) prevented destruction of the evaporator vessels for the three large leaks (i.e. no sodium leaks)**
  - **1975 US delegation visit (Balent) report speculated that the reaction products stayed within the vessel shell, and this aggravated tube failure propagation**
- **After re-tubing, some leaks still occurred, but emphasis on sodium and feedwater quality control, early leak detection, and remediation (tube plugging) resulted in stable plant operation at design power levels**

BN-350 Steam Generator Failures – Lessons Learned

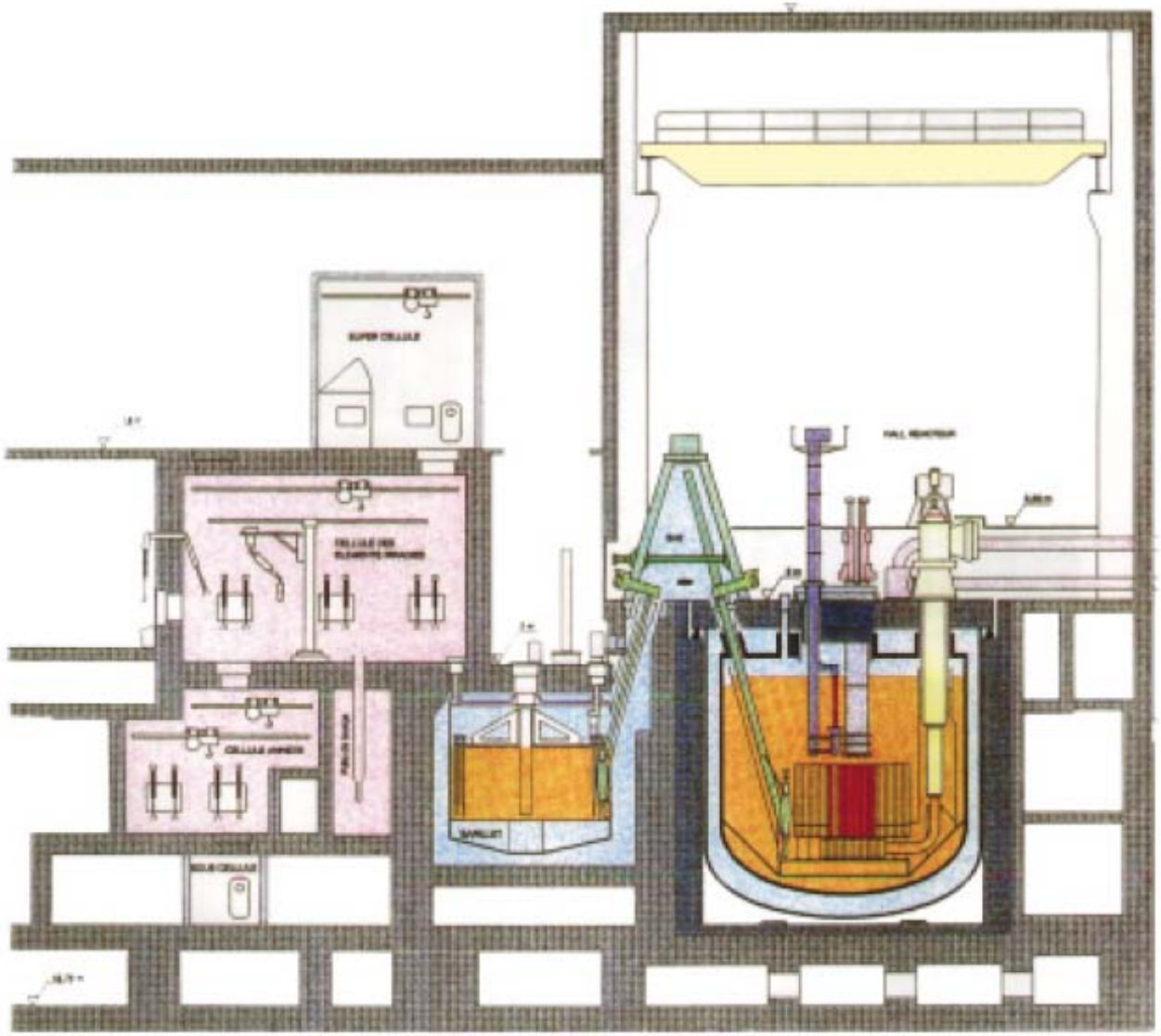
- **Soviet steam generator technology experience prompted needs for**
  - **Improved steam generator component manufacturing techniques; tube drawing/forging and welding**
  - **Design for failure prevention; sodium and feedwater quality control**
  - **Design provision for tube failure;**
    - ***Detection; quick recognition and action to prevent propagation***
    - ***Containment; blow-down relief to control intermediate sodium system pressure***
    - ***Remediation; plugging of leaking tubes***
- **U.S. and Western technology contrast**
  - **EBR-II evaporator/superheater experience**

Phenix

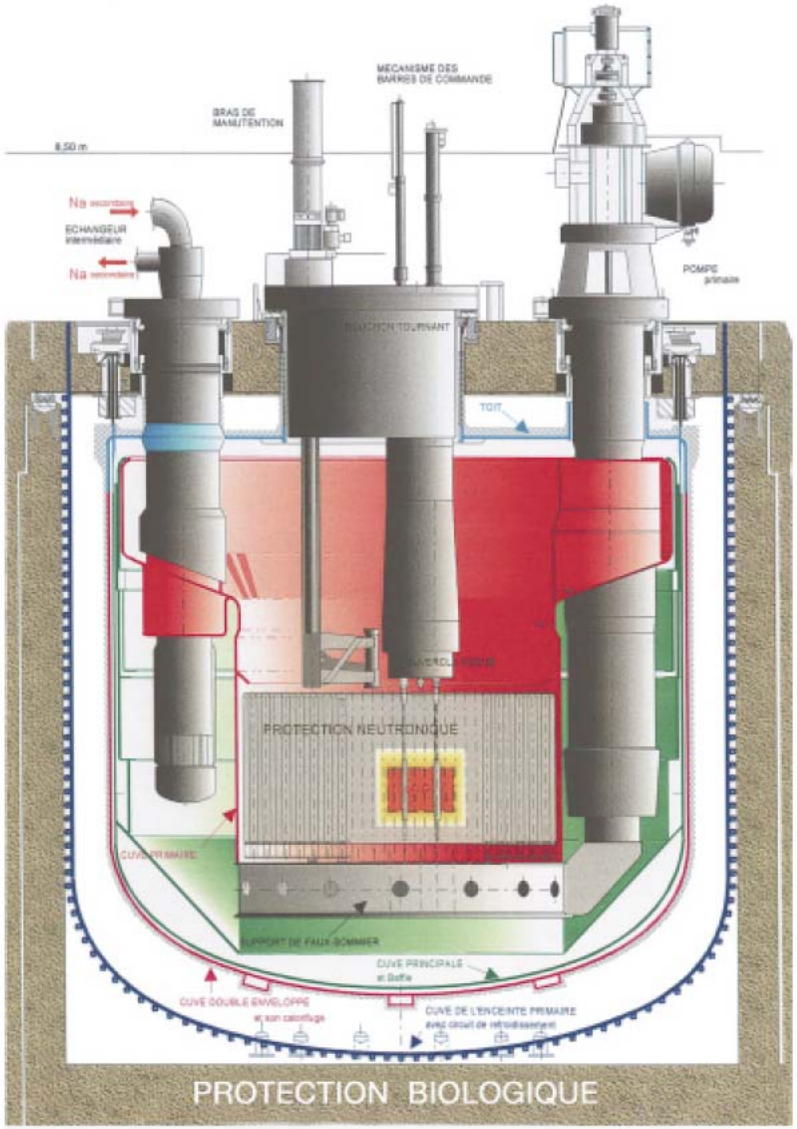
- 563 MWt, 250 MWe power station located near Marcoule, France
- Criticality 1973, full power 1974
- 85 cm core height, 430 cm subassembly height
- 217 wire-wrapped pins per subassembly
- Number of subassemblies: 55 inner core, 48 outer core, 90 radial blanket, 1317 reflector and shield
- Free-Standing (Free-Flowering) Core Restraint
  - Accommodation of thermal/mechanical/irradiation effects for refueling
- Four rapid, large, negative reactivity excursions triggered automatic scrams due to power reduction: 6 August ’89, 24 August ’89, 14 September ’89, and 9 September 1990
  - Intensive investigations failed to provide identification of the cause(s)
  - Operations twice interrupted and then resumed following testing, installation of additional instrumentation, and analyses of events

Phenix Site

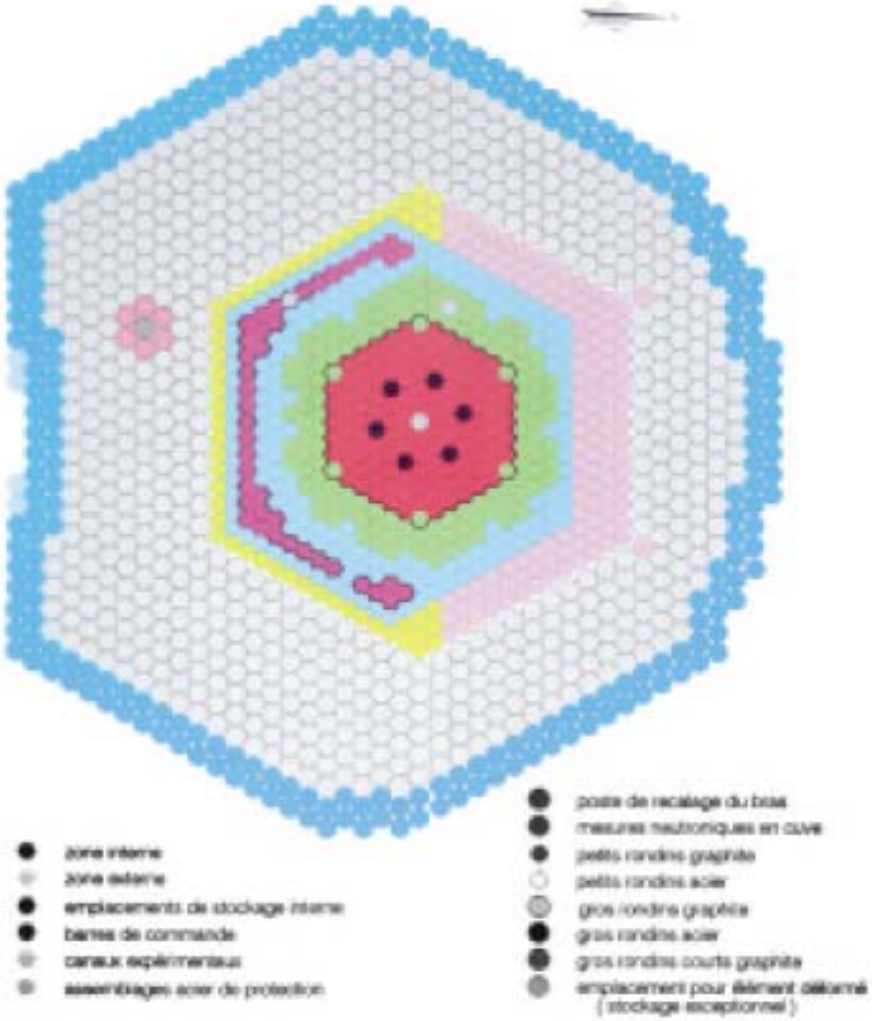
Phenix Reactor Building



Phenix Pool-Type Primary System

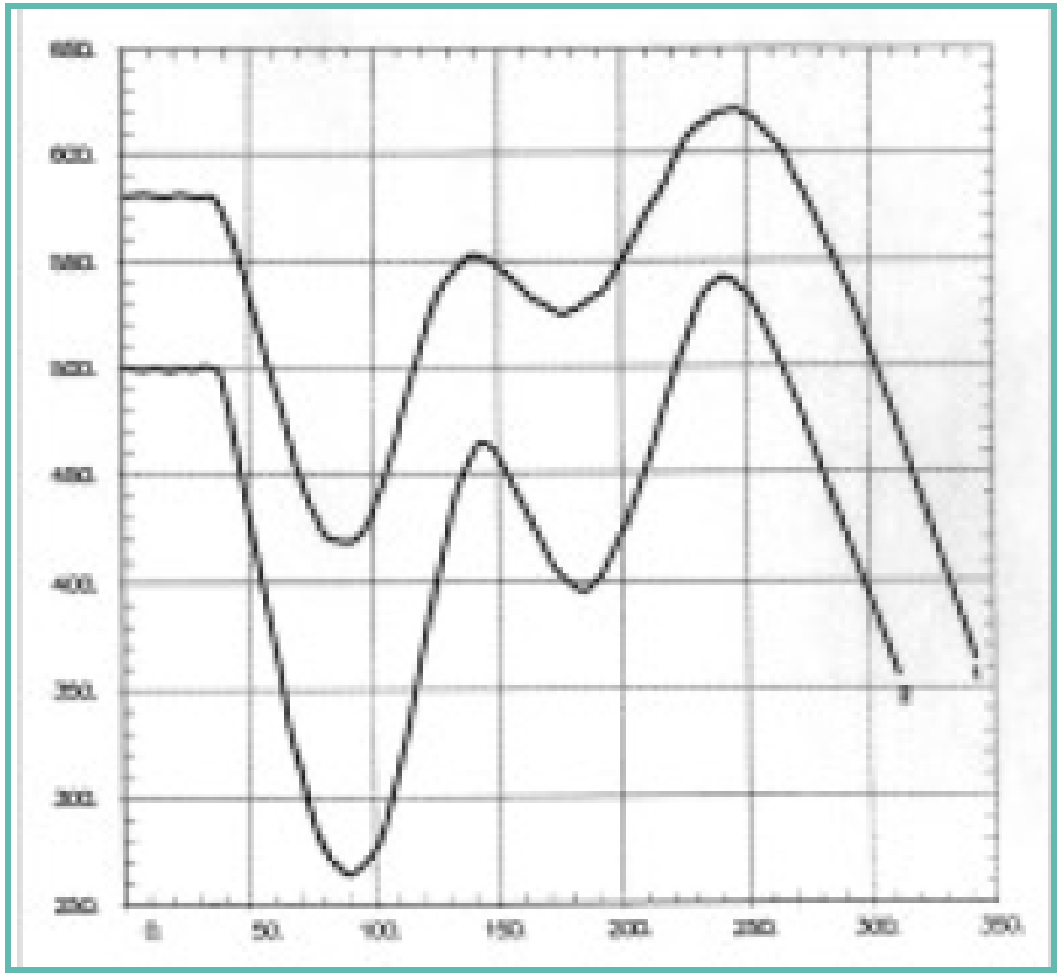


Phenix Reactor Core



IMPLANTATION GENERALE DANS LE COEUR

Phenix Negative Reactivity Transients, P (Mwt) vs t (ms)



1 – Sept. 14, 1989, 2 – Sept. 9, 1990

Phenix Reactivity Excursions - 1

- **After the first two events (August ’89), the cause was attributed to ‘interference’ in operation of the plant control system instrumentation, which had been modified just prior to 1989, but no specific fault was found**
- **After the third event (September ’89), the cause was attributed to passage of a gas bubble through the core periphery (negative reactivity) due to plugging of inlet plenum vents designed to prevent gas build-up. After a shutdown, maintenance, and analyses, the reactor was restarted in December ’89**
- **Two cycles later, the fourth event (September ’90) invalidated the gas bubble hypothesis**
  - **Reactor operation ceased and an intensive investigation was begun**
  - **A panel of experts was convened to consider all possible causes**
  - **Reactor and plant tests and repairs were performed**
- **By the end of ’91, the root cause had not been identified, but the ‘strong presumption’ was (rapid) radial expansion of the subassemblies**

Phenix Reactivity Excursions - 2

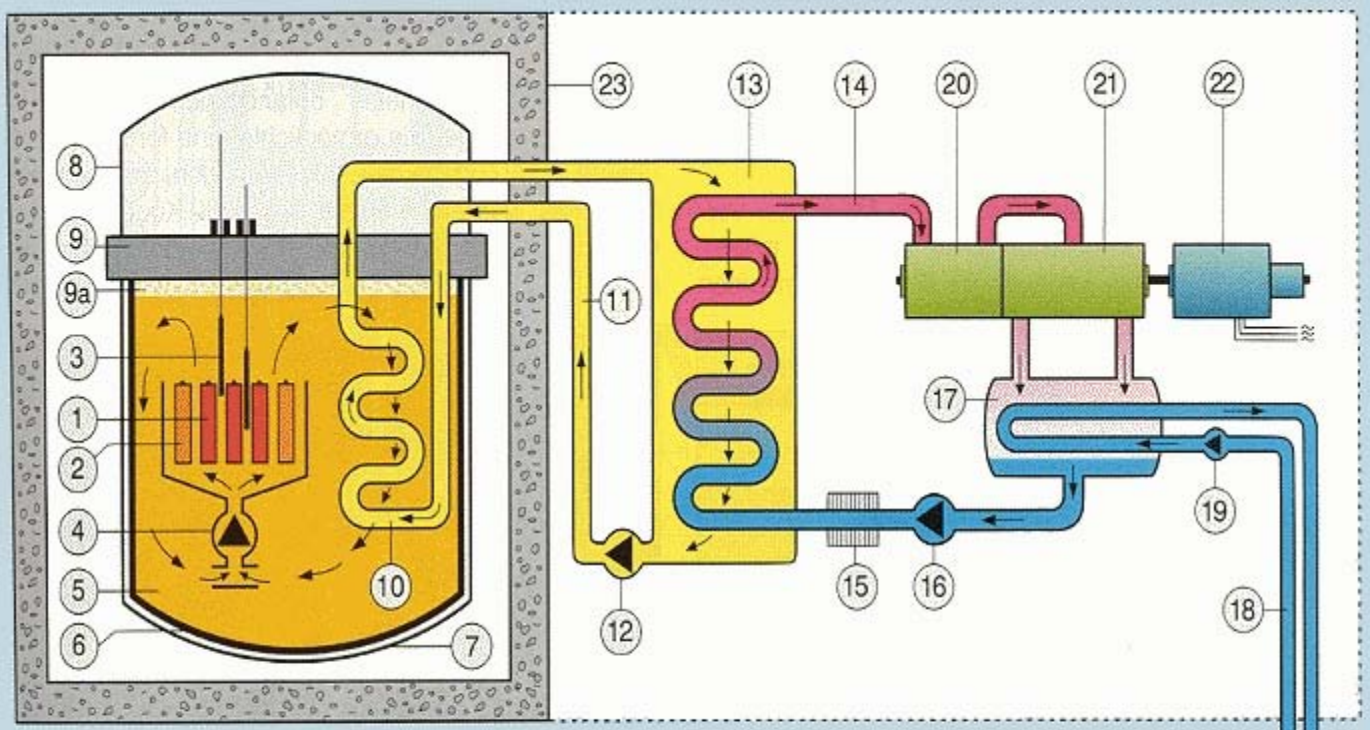
- **Through 1992, analyses and testing with installation of special reactor surveillance equipment continued**
  - **Focus on causal phenomena: coolant voiding, control rods motions, and core movements**
- **Given the amplitude and speed of the events, only core movements could cause the observed behavior**
- **Modeling and simulations were performed, but no one specific scenario or cause was identified**
  - **Final explanation: Outward (radial) expansion of the subassembly lattice, followed by a return**
- **Ultimately, consensus that operations could be safety resumed**
  - **No further events to date**
- **Lesson learned? My opinion: Avoid free flowering core restraint design.**

Super Phenix (Creys-Malville)

- **2990 MWt, 1242 MWe power station located 50 KM east of Lyon, France**
- **Criticality 1985, full power 1986**
- **1 m core height, 5.4 m subassembly height**
- **271 wire-wrapped pins per subassembly**
- **Number of subassemblies: 193 inner core, 171 outer core, 234 radial blanket, 1288 reflector and shield**
- **On March 8, 1987, a leak in the fuel storage tank was detected**
  - **Leaking sodium was contained by the storage tank guard vessel**
  - **Stored fuel (new fuel, one partly irradiated subassembly, and dummy subassemblies used for pre-startup testing) were unloaded and the tank was drained.**
  - **Investigations began to determine the location of the leak (Sept. ’87)**
- **It was ultimately concluded that the tank could not be repaired, and alternative fuel handling equipment and procedures were adopted**
- **The expense associated with this and other events led to closure of the plant in 1998 (Down 2 yrs for technical, 4 ½ yrs for ‘administrative’)**

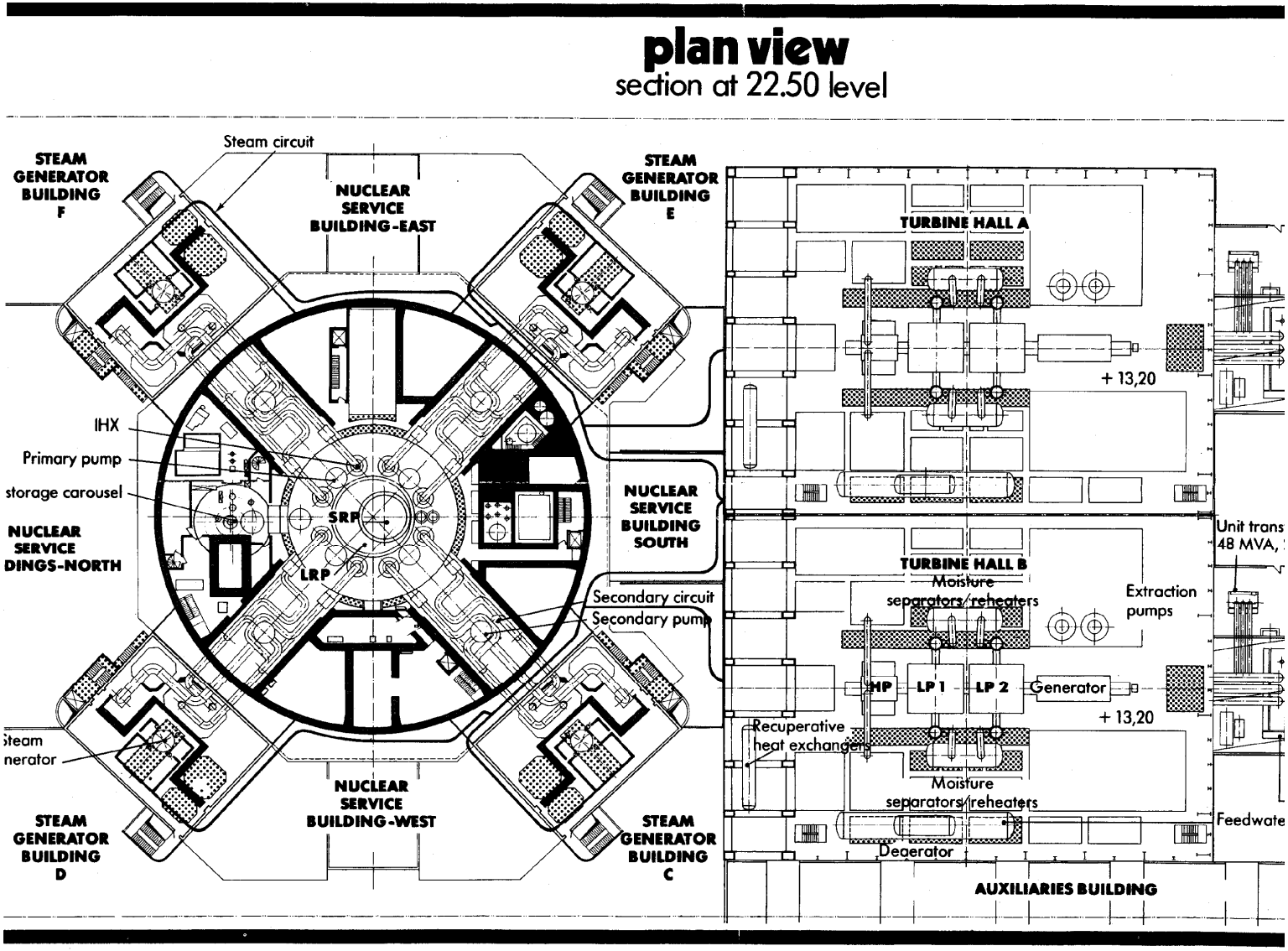
Super Phenix Plant Site

Super Phenix Plant Coolant Flow Diagram

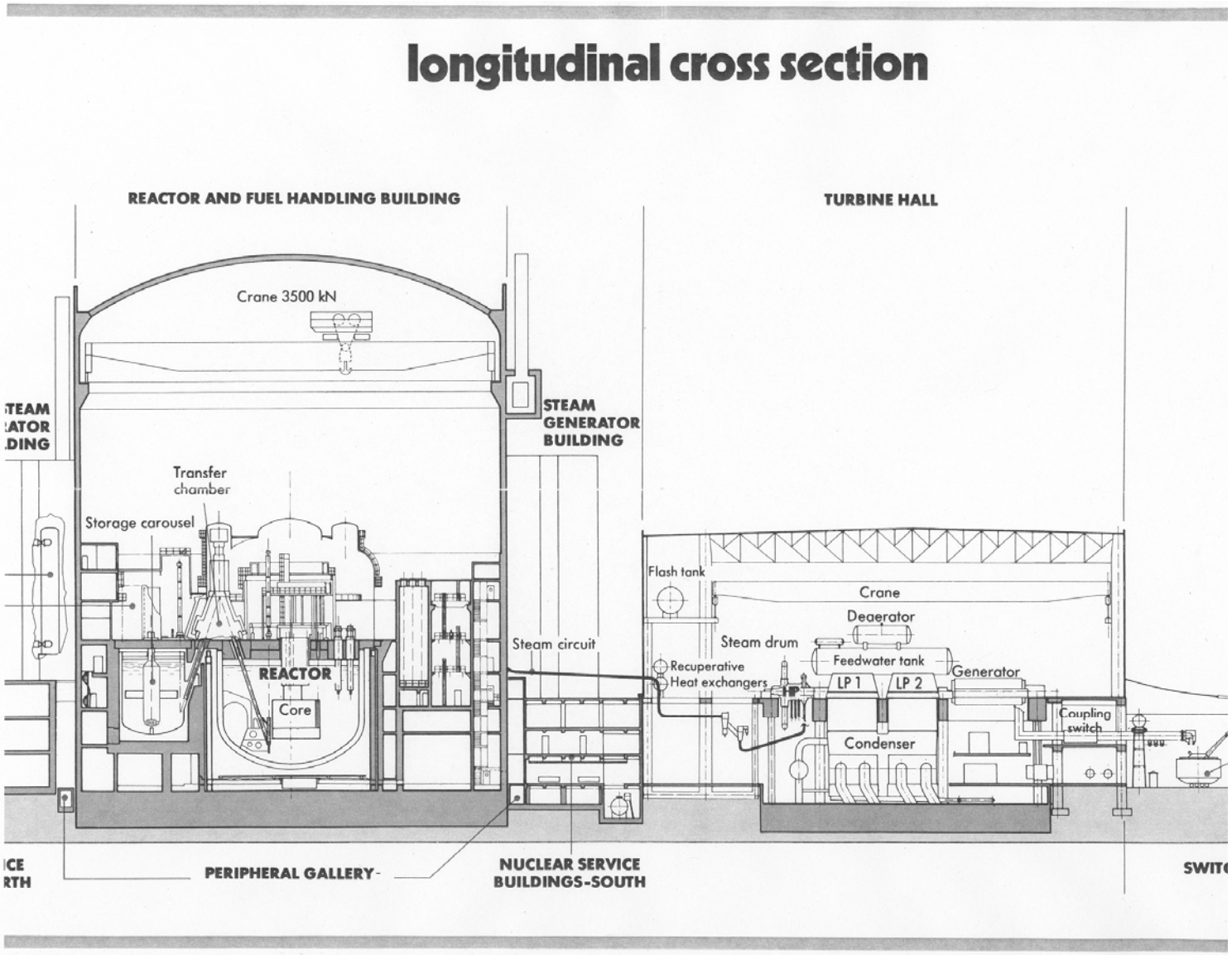


1 - élément combustible fissile
2 - élément combustible fertile
3 - barre de contrôle
4 - pompe de circulation du sodium
6 - cuve du réacteur (acier inoxydable)
7 - cuve de sécurité
8 - enceinte de confinement
9 - couvercle
9a - Atmosphère de gaz de protection (Argon)
10 - échangeur de chaleur intermédiaire (1 parmi 4)
11 - circuit de sodium secondaire
12 - pompe de ciculation du sodium secondaire
13 - Générateur de vapeur (1 parmi 4)
14 - vapeur
15 - pré-réchauffeur
16 - pompe à eau d'alimentation
17 - condenseur
18 - eau de refroidissement (fleuve)
19 - pompe à eau froide
20 - turbine haute pression
21 - turbine basse pression
22 - génératrice
23 - bâtiment réacteur

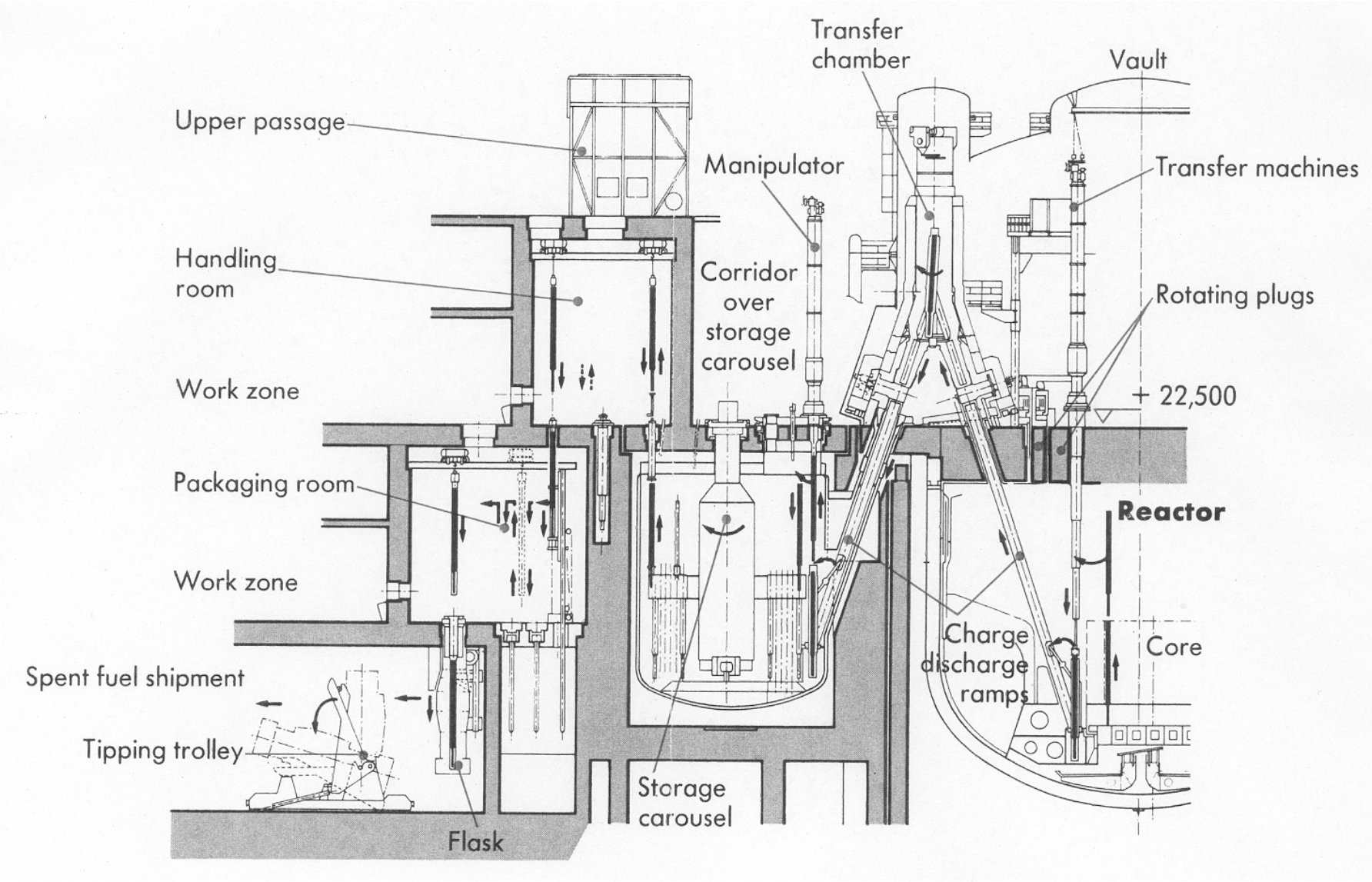
Super Phenix Plant Plan View



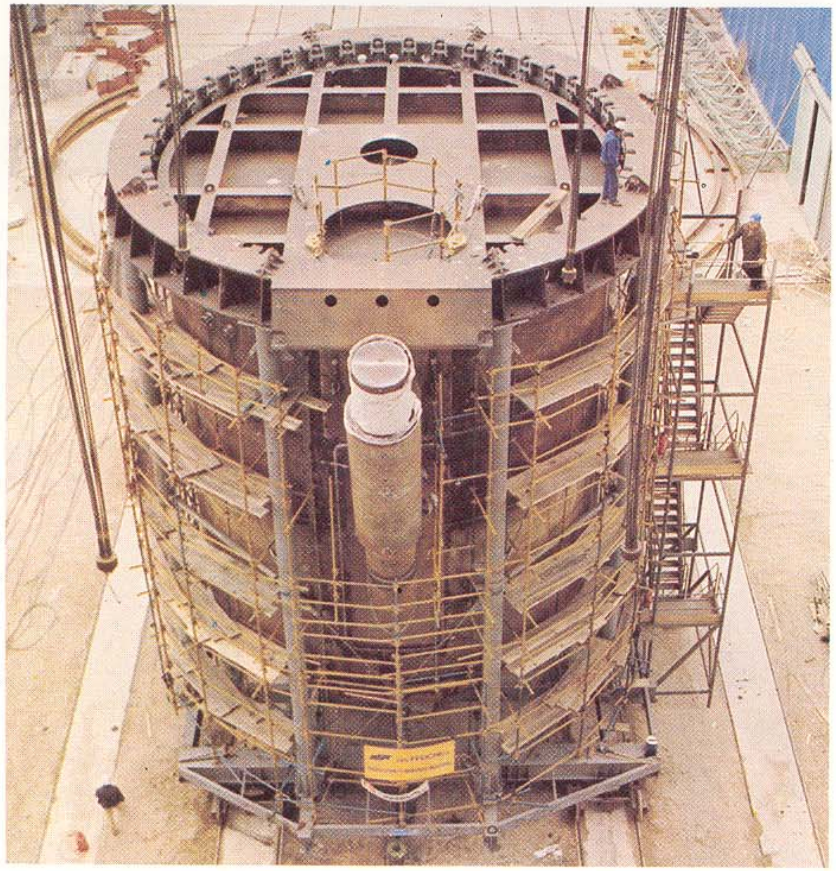
Super Phenix Plant Elevation View



Fuel Storage Tank Detail



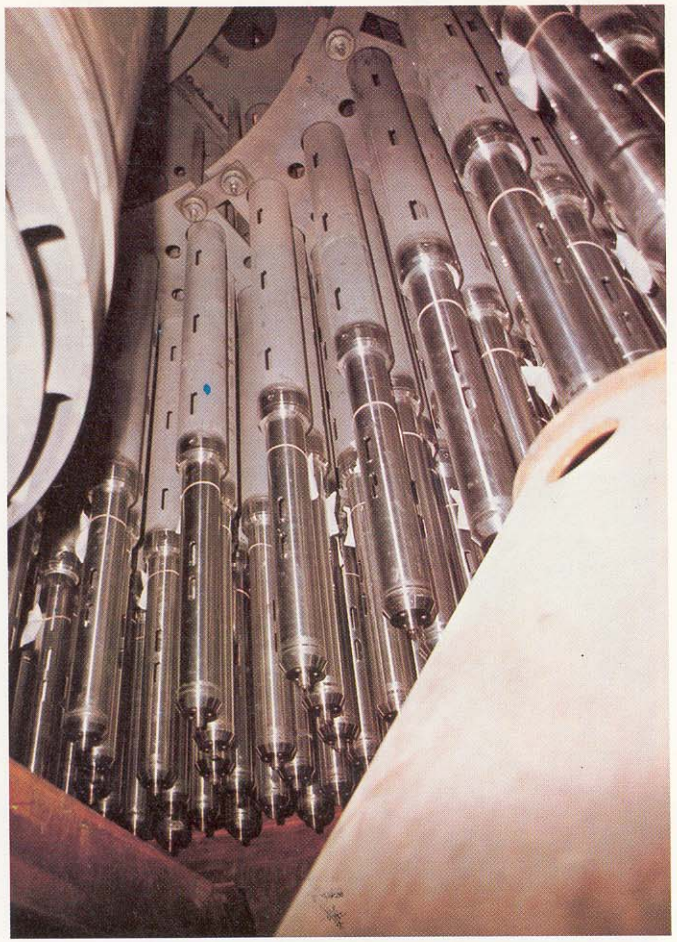
Fuel Storage Tank During Construction



April 26th, 1980: Storage drum being placed in the reactor building.

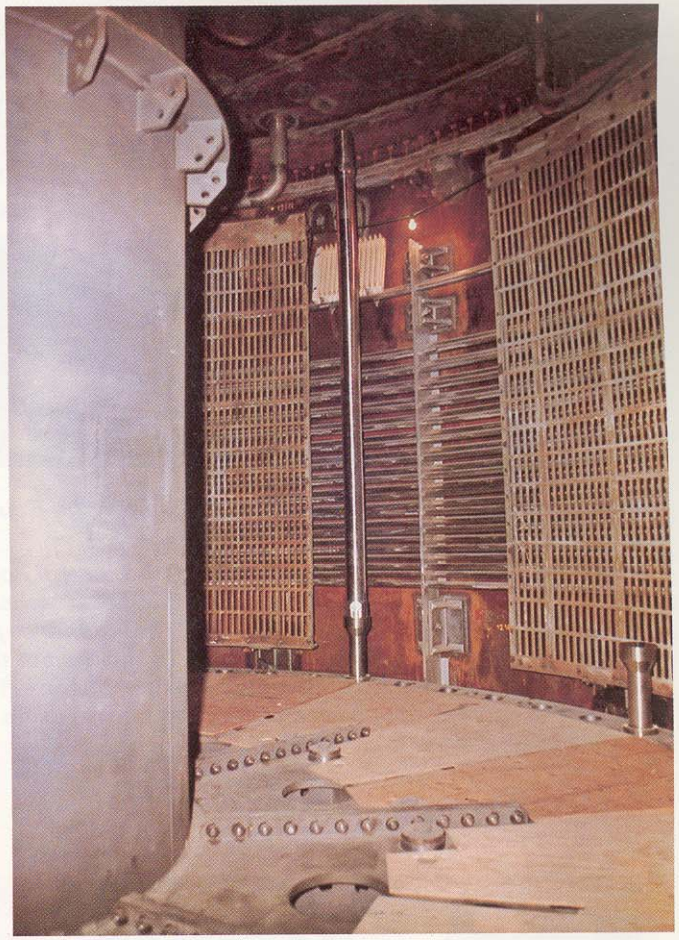
This photo has been taken from the sloping handling ramp. The criss-cross girdering of the vessel roofs can be seen and the system for connecting this roof to the hoop anchored in the main structure: 42 slings fixed to the roof flanges and the hoop supports.

Inside the Fuel Storage Tank During Fabrication



Creys-Malville: Inside the lower part of the storage drum.

Penetrating the diagrid bottom plate are the lower parts of the thimbles into which the fuel elements will be placed. Note the holes for the circulation of sodium.



Creys-Malville: Storage drum during assembly viewed from the inside.

Left: the axis of the diagrid carousel. Its upper plate is covered with hardboard sheets to facilitate movement during assembly. Right: note the cooling system tubes (sodium-sodium exchange) behind their protective grids.

Fuel Storage Tank Leak Location

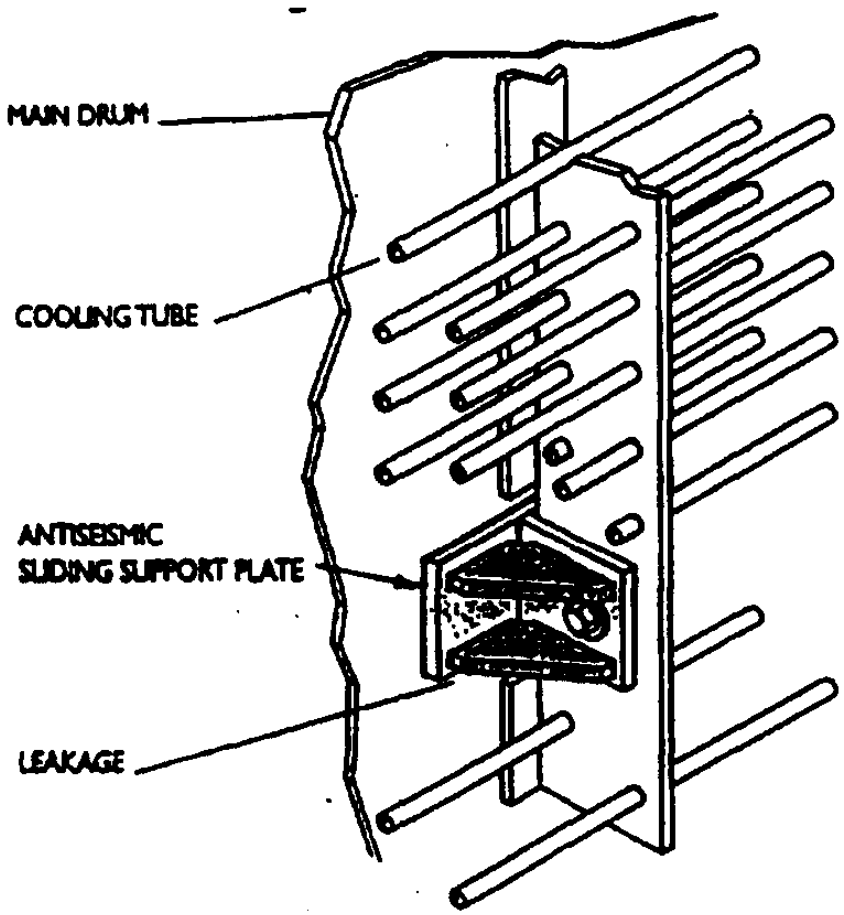


FIG. 2.21. Super-phenix fuel storage drum leak localization.

Super Phenix Fuel Storage Tank Leak

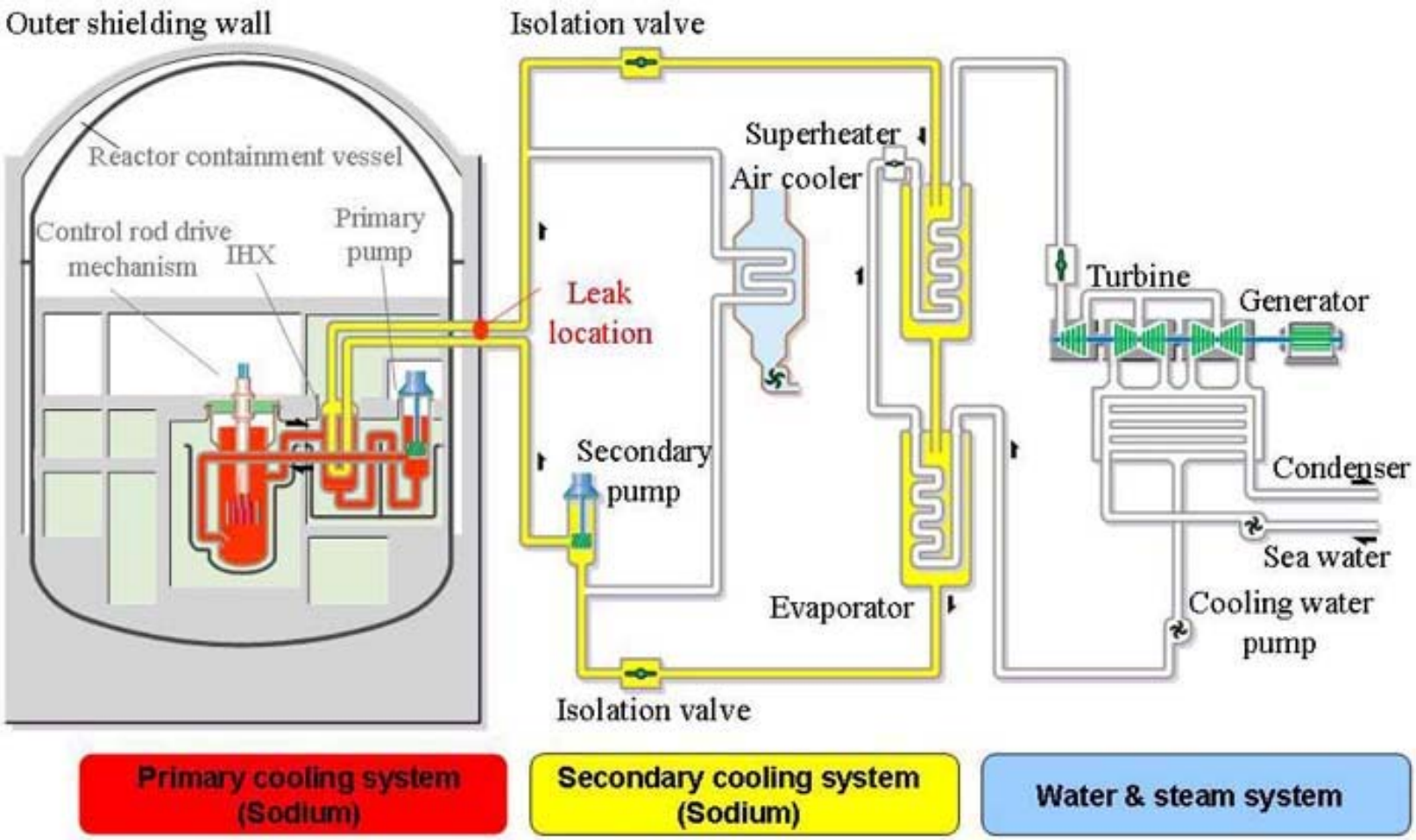
- Leak caused by a horizontal crack ~ 60 cm long on the lower welding bead of a cooling system support plate
  - Investigation showed cracks at other locations as well; other support plates and tank wall weld beads
- The “most probable scenario” causing the crack was identified as “the nature of the drum steel (ferritic 15 D3) and the simultaneous presence of three factors: the existence of start (sic) sites (micro-cracking) in zones of high hardness, residual stresses close to the elastic limit of the material, and lastly, the contributions of hydrogen which allowed the brittling phenomena to occur.”
  - It is customary to use austenitic stainless steel in contact with liquid sodium at high temperatures (reactor vessels and pipes)
- Lesson learned: Select the correct material for high temperature sodium service

MONJU

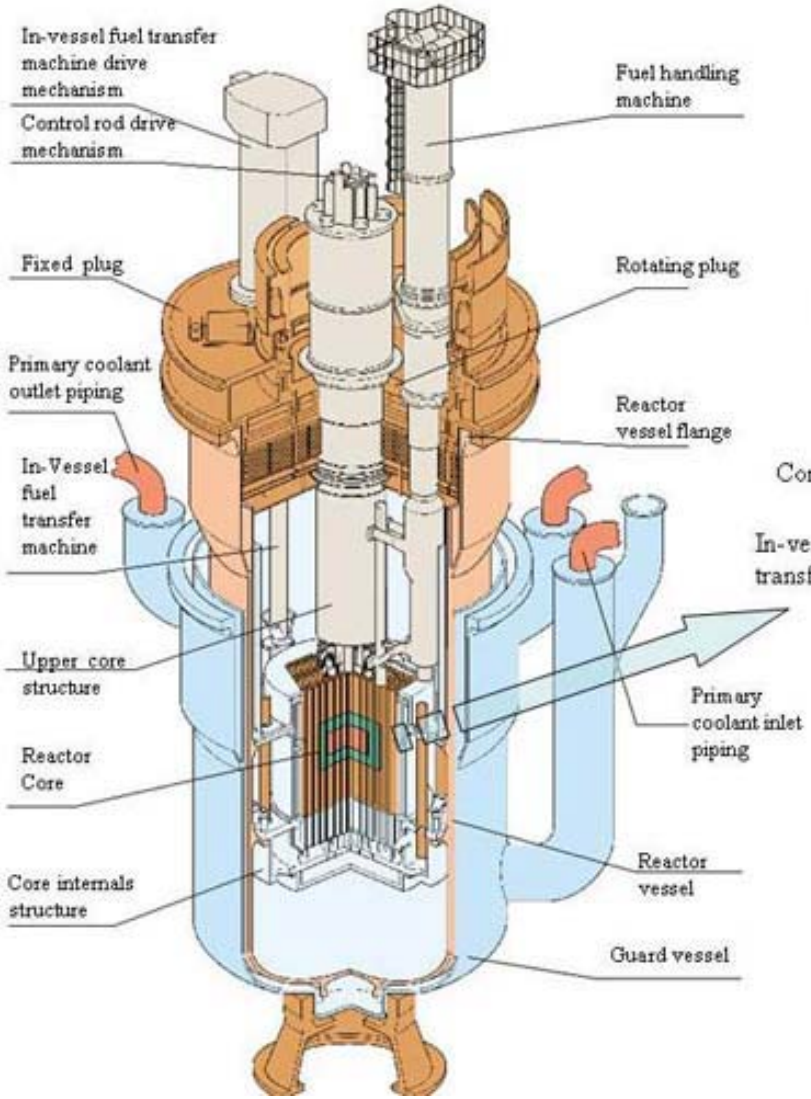
- 714 MWt, 280 MWe power station located 12 km NE of Tsuruga, Japan
- Criticality 1994, full power 1995
- 93 cm core height, 420 cm subassembly height
- 169 wire-wrapped pins per subassembly
- Number of subassemblies: 108 inner core, 90 outer core, 172 radial blanket, 324 reflector and shield
- On December 8, 1995; sodium leak detected in the room housing the "C" secondary loop (non-radioactive sodium) during operation at 43% power
  - High temperature at the IHX outlet and smoke alarm at 19:47
  - Reactor shutdown begun at 20:00, manual trip at 21:20
  - C loop draining at 22:55, complete at 00:15
  - Inspection confirmed sodium leak at thermocouple well; ~1 m^3 mound of Na_2O on the steel floor, aerosol on the walls and floor
  - Leak caused by failure of a thermocouple well
- Clean-up and repairs were made, but restart was delayed by legal actions, court decisions, appeals, safety reviews, etc.
  - Restart scheduled for 2008

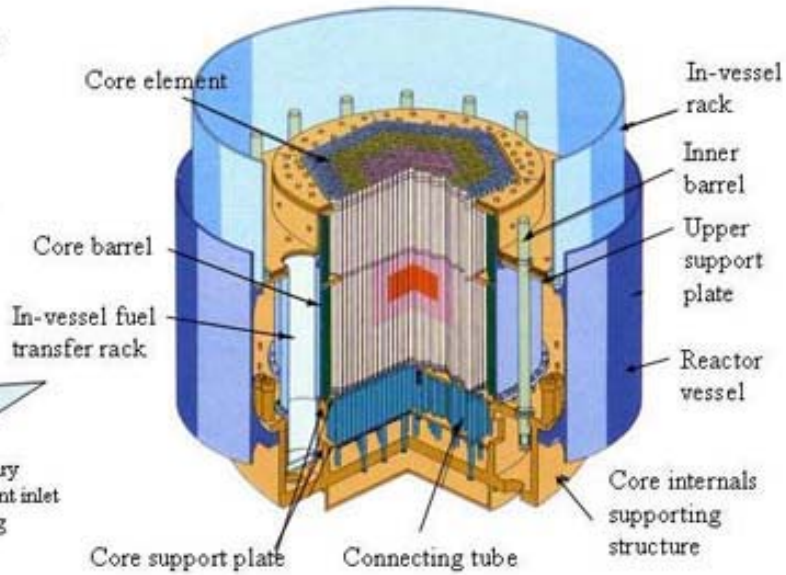
MONJU Site

MONJU Flow Diagram



MONJU Reactor Vessel and Core

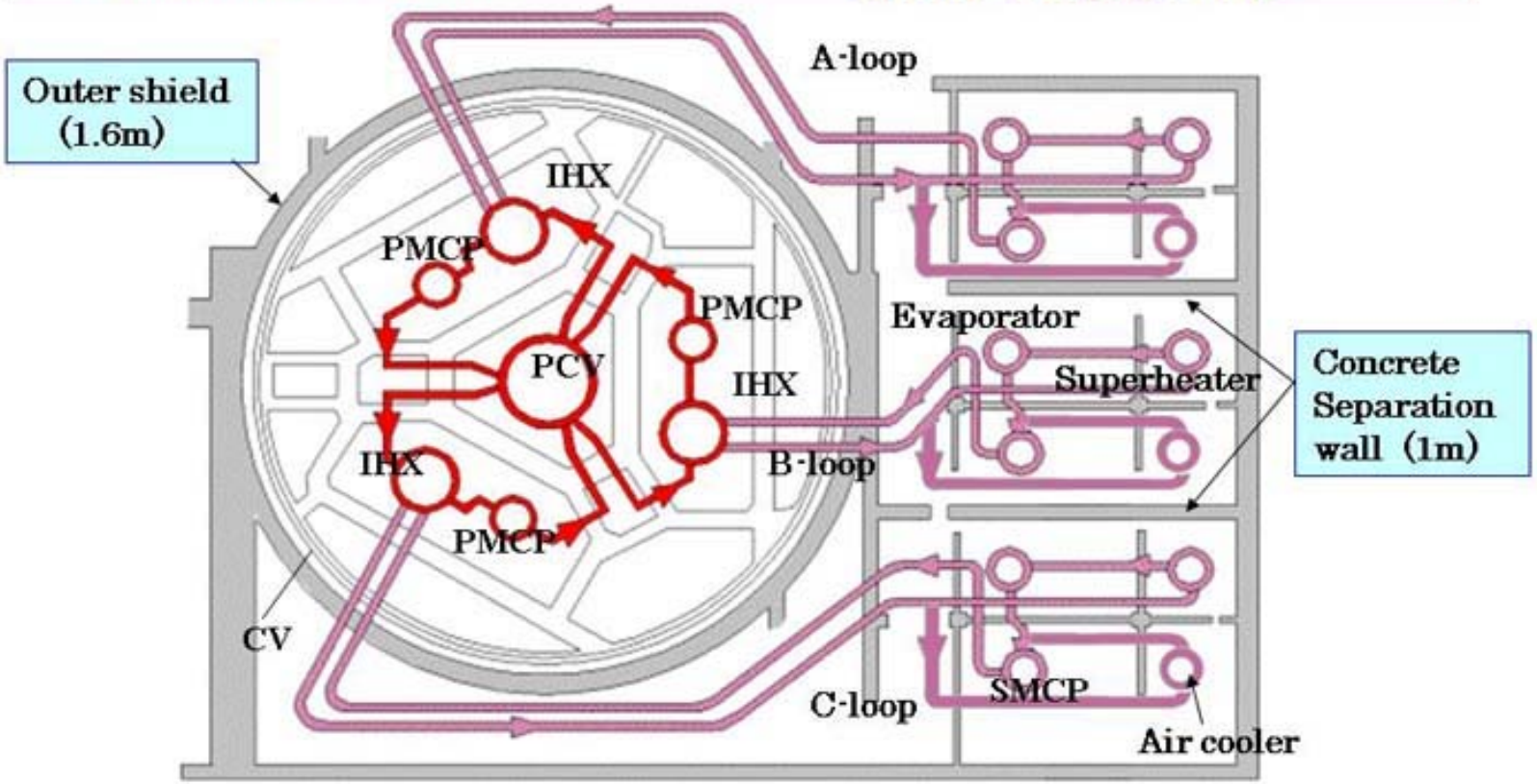




MONJU Coolant Systems

Primary systems
R/V sodium temperature
Inlet : 397 ℃
Outlet : 529 ℃
Sodium mass : 760 ton

Secondary systems
IHX sodium temperature
Inlet : 325℃
Outlet : 505℃
Sodium mass : 760 ton



MONJU Secondary Sodium Leak Location on C Loop

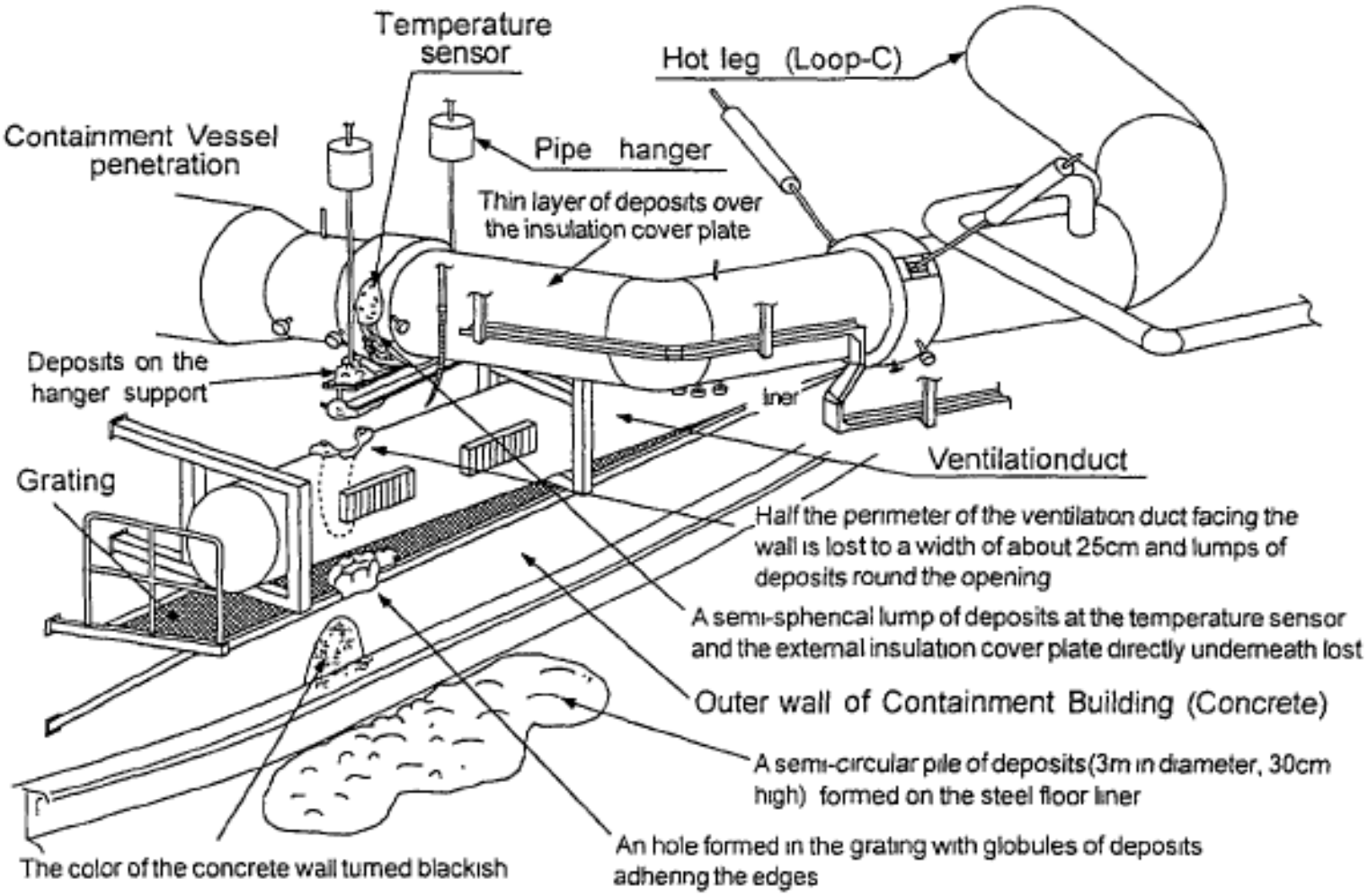


Fig. 2. Sketch of the affected area.

Original Secondary Circuit Thermocouple

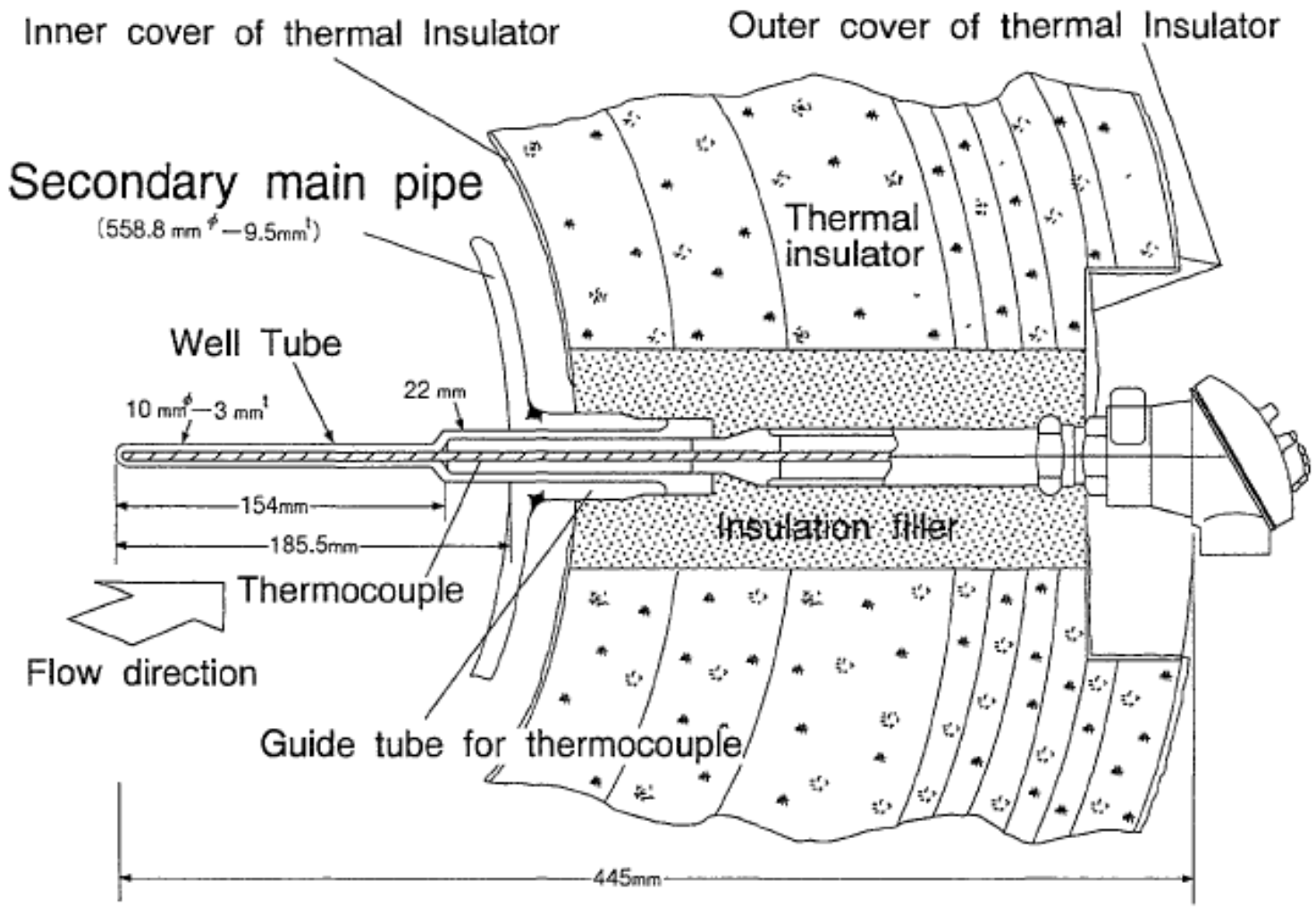


Fig. 3. The thermocouple well of the secondary circuit.

Damaged Secondary Circuit Thermocouple

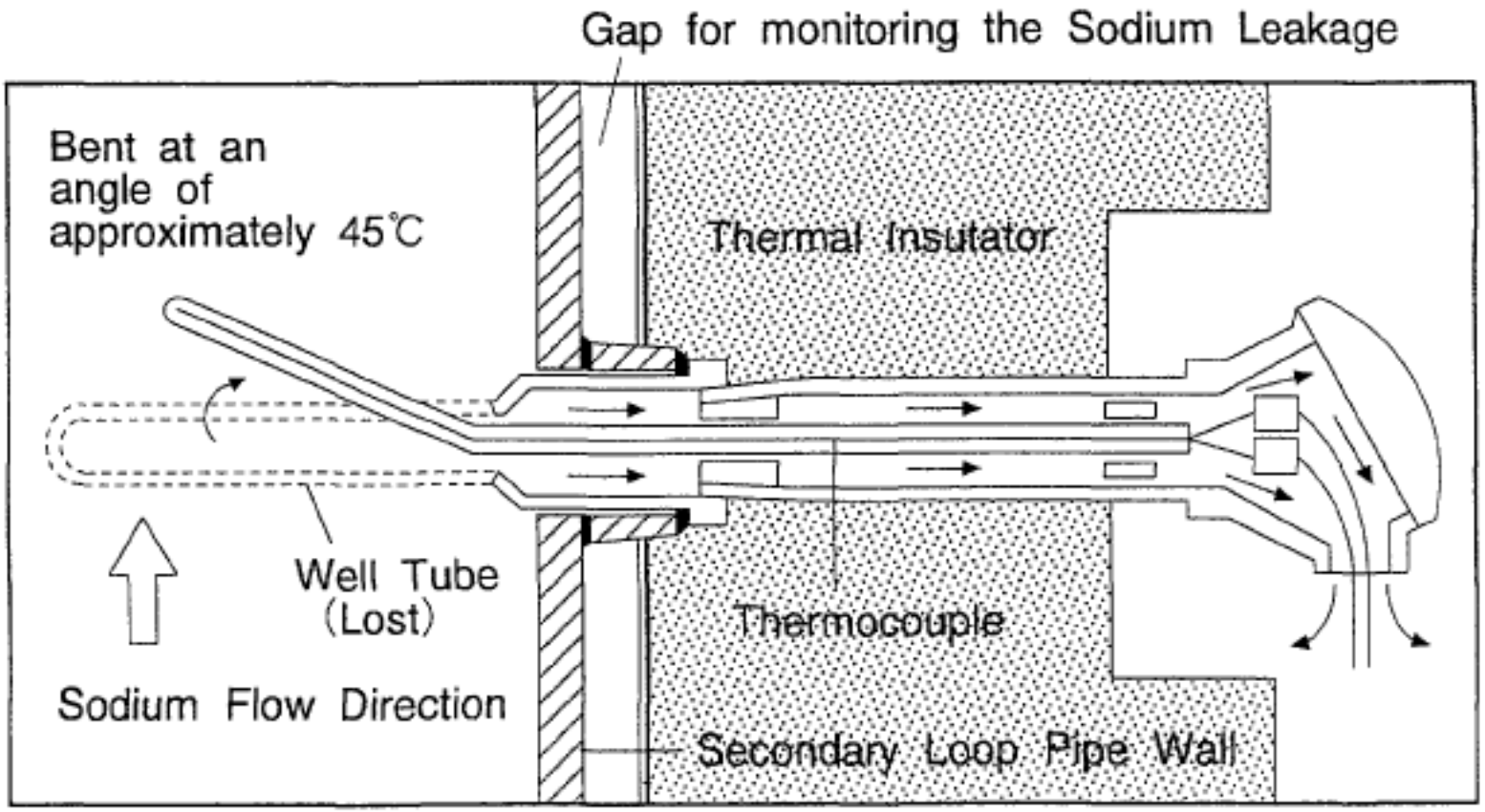


Fig. 4. The sodium leak flow path.

MONJU Thermocouple Sodium Leak

- Video: http://www.mext-monju.jp/takuhai/video/monju02.wmv
  - Video (6 min., in Japanese) shows post-leak consequences, clean up activities, repairs and remediation measures
- Thermocouple well tip failed due to flow-induced cycle fatigue
- All thermocouples in the secondary loop were replaced

Thermocouples Replaced on Secondary Circuit

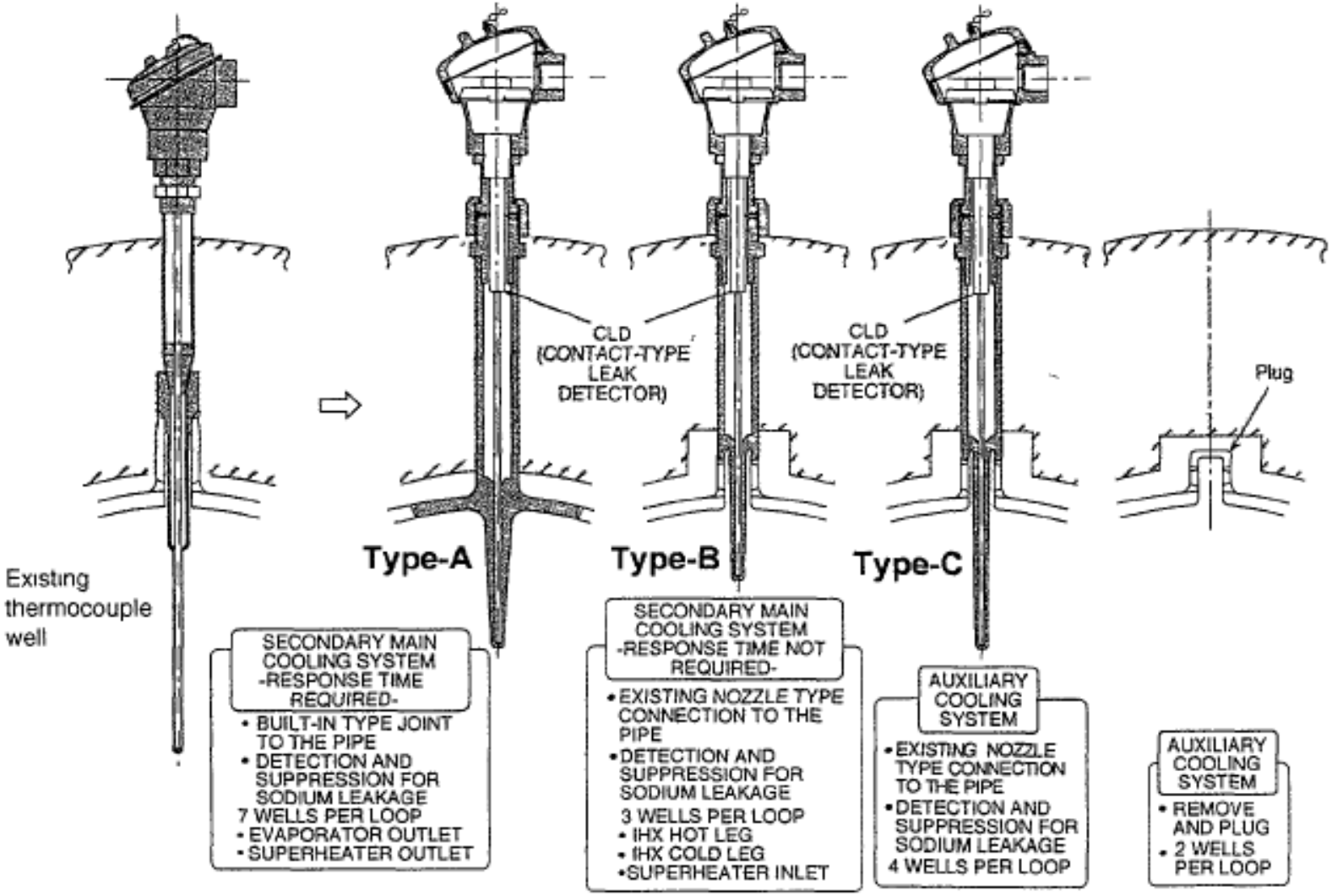
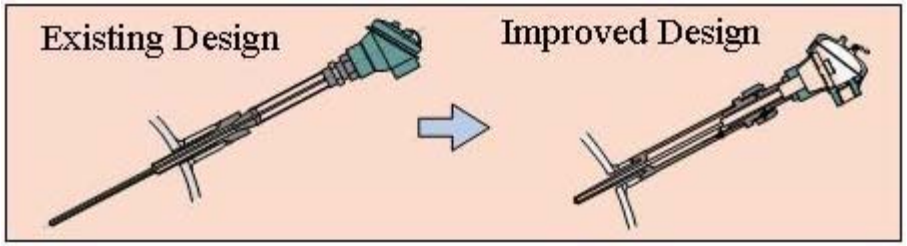


Fig. 7 Modified design concepts for the thermocouple wells on the secondary cooling system.

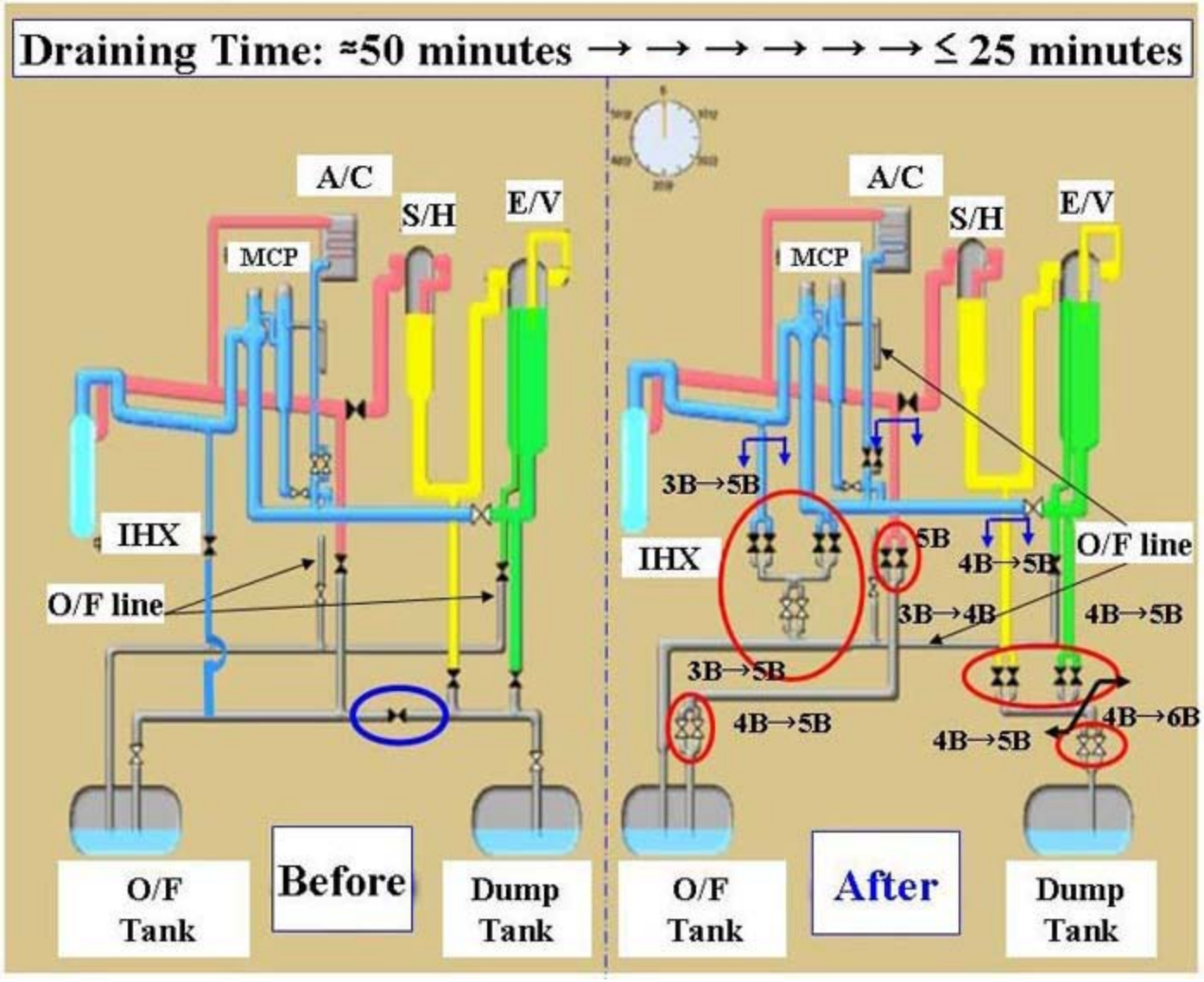
Countermeasures against Sodium Leakage

- **Replacement of Secondary Thermocouple wells**

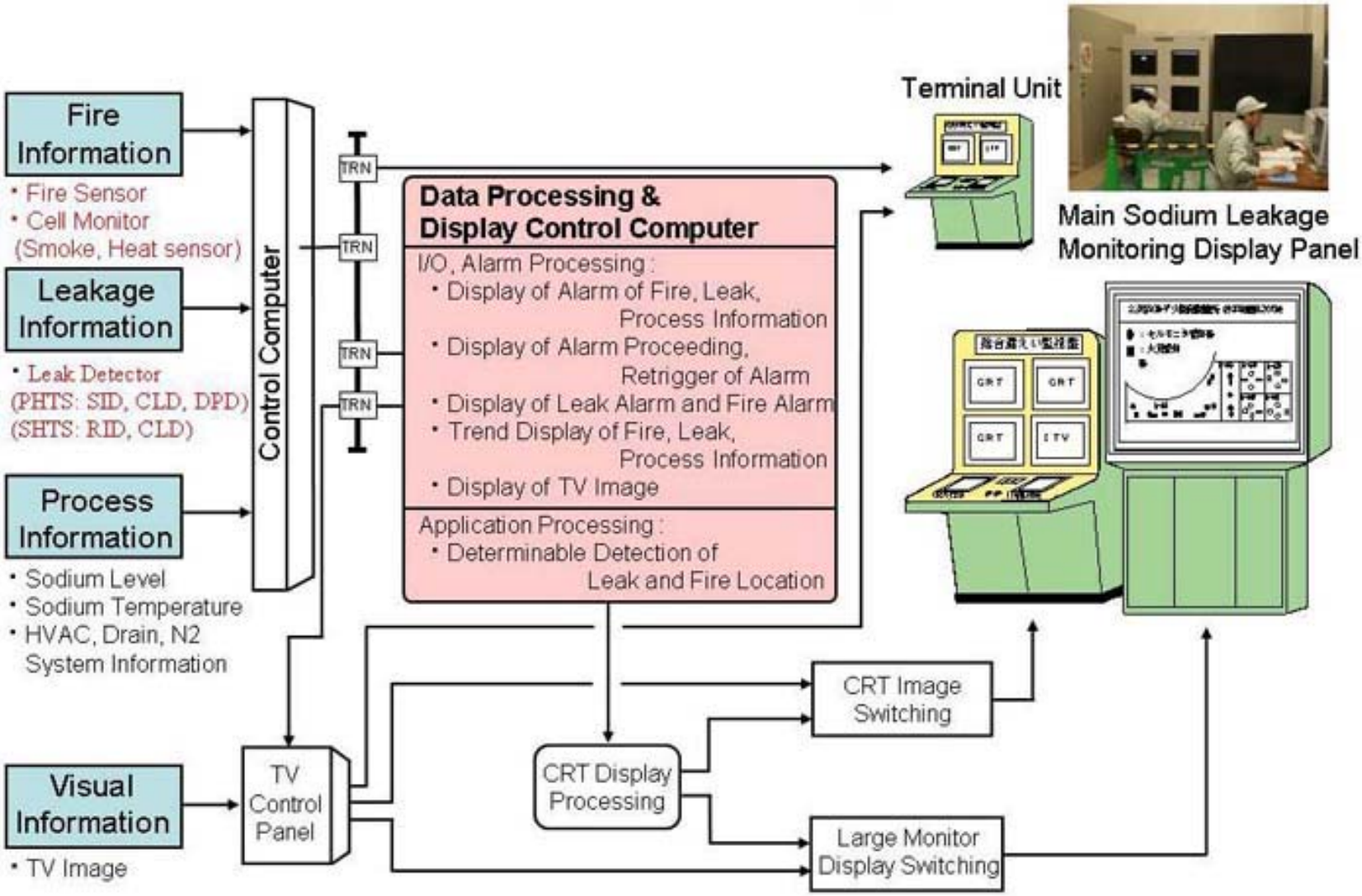


  - Leakage Control by Metallic Gasket Type Seal and Welded Seal
  - Leakage Detection
  - Prevention of Flow-induced Vibration
- **Improvement of Facilities**
  - Improvement of Drain System, Reduction of Drain Time
  - Integral Sodium Leakage Monitoring System
  - Subdivision of Building / Nitrogen Gas Extinguisher System

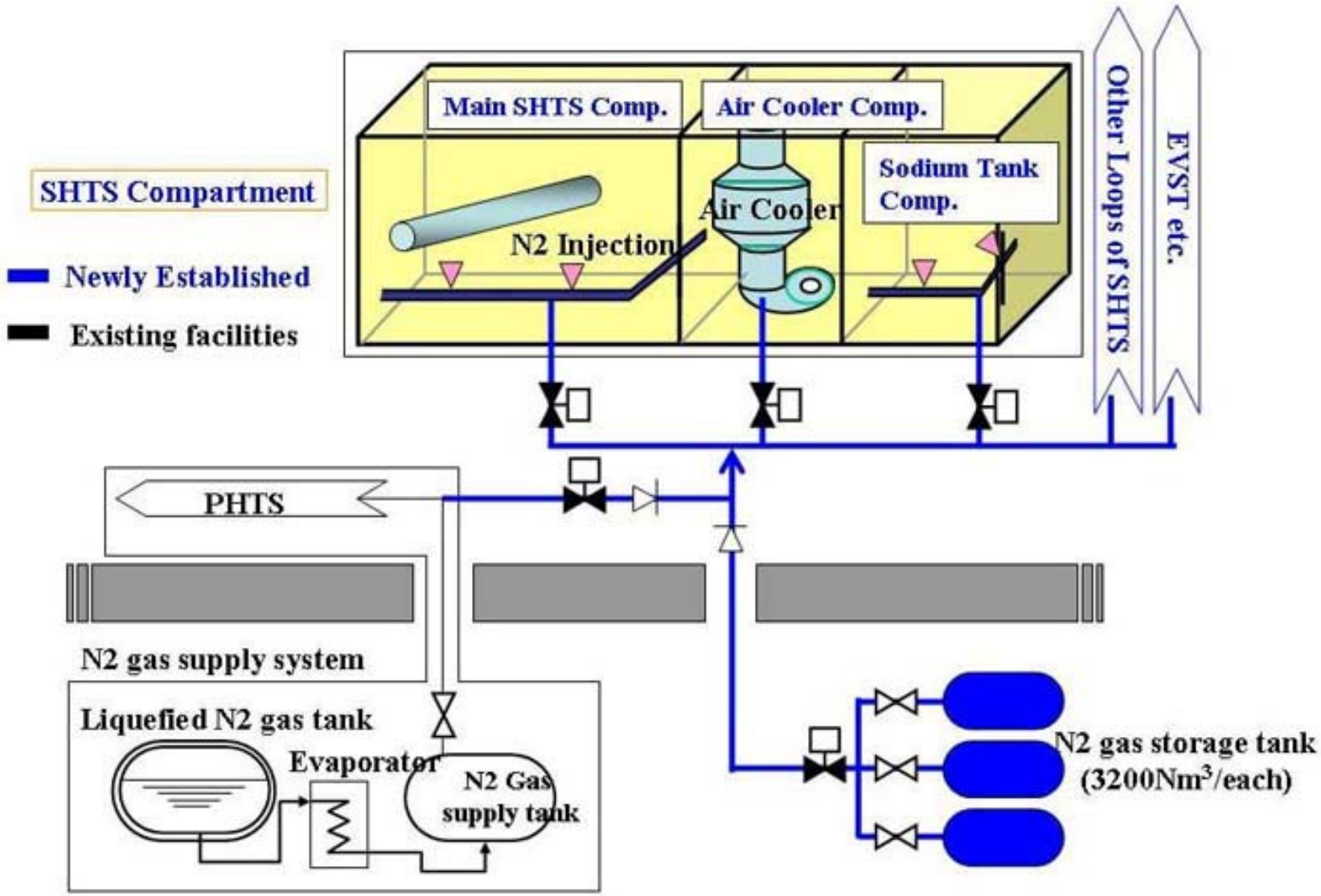
Improvement of Drain System, Reduction of Drain Time



Integrated Sodium Leakage Monitoring System



Installation of Nitrogen Gas Injection System



MONJU Sodium Leak and Lessons Learned

- **Event had no radiological consequences, no injuries, no environmental harm**
  - **Post-event information management, legal action, license review**
- **Technical lessons**
  - **Faulty thermocouple well design; inadequate prototype testing, lack of backup leak prevention**
  - **Some experts question need for pipe penetrations**

Safety Testing Results

- **EBR-II Shutdown Heat Removal Tests**
- **FFTF Inherent Safety Tests: Unprotected Loss-Of-Flow with GEMS**

EBR-II Shutdown Heat Removal Test (SHRT) Program

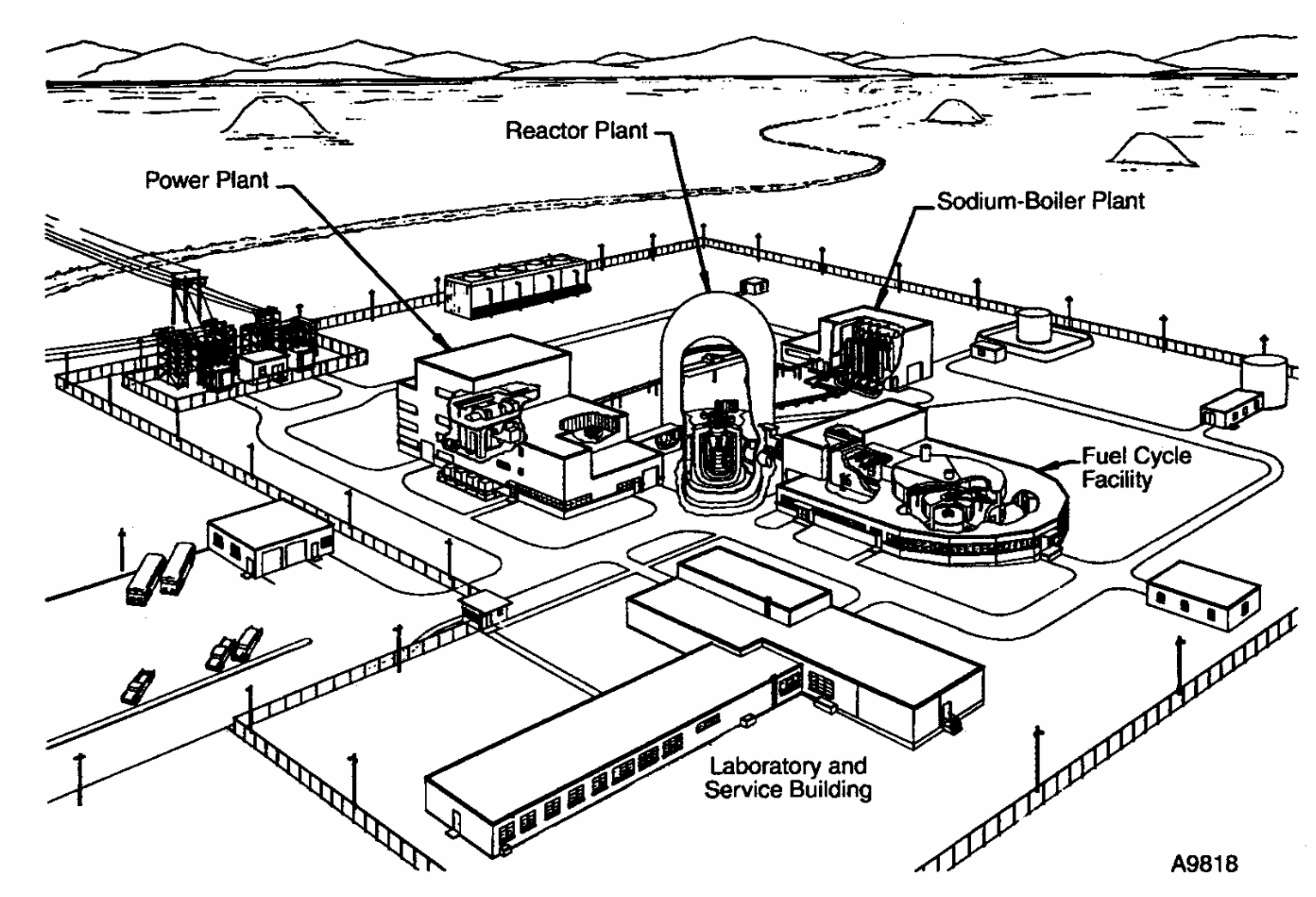
- **Thermal-hydraulic testing at EBR-II began to support safe and reliable operation of EBR-II, but evolved to become a broader program to support design and performance assessment for advanced liquid metal reactors**
- **Early testing (1974) focused on steady-state fission and decay heat removal by natural circulation, and pioneered the use of specially instrumented fuel subassembly (XX07) for flow and temperature measurements**
- **Subsequent tests examined the transition from forced to natural circulation with updated instrumentation (XX08), from a variety of initial conditions**
  - **Ex: Primary and secondary pump trips with scram from hot standby**
  - **Smooth, benign transitions to natural circulation**
- **The SHRT program employed new instrumentation (XX09) and extended the test matrix to transients from full power, with and without scram**
  - **Culmination: Full scale demonstrations of passive reactor shutdown and natural circulation shutdown heat removal for loss-of-flow and loss-of-heat-sink transients (1986)**

EBR-II Design Overview

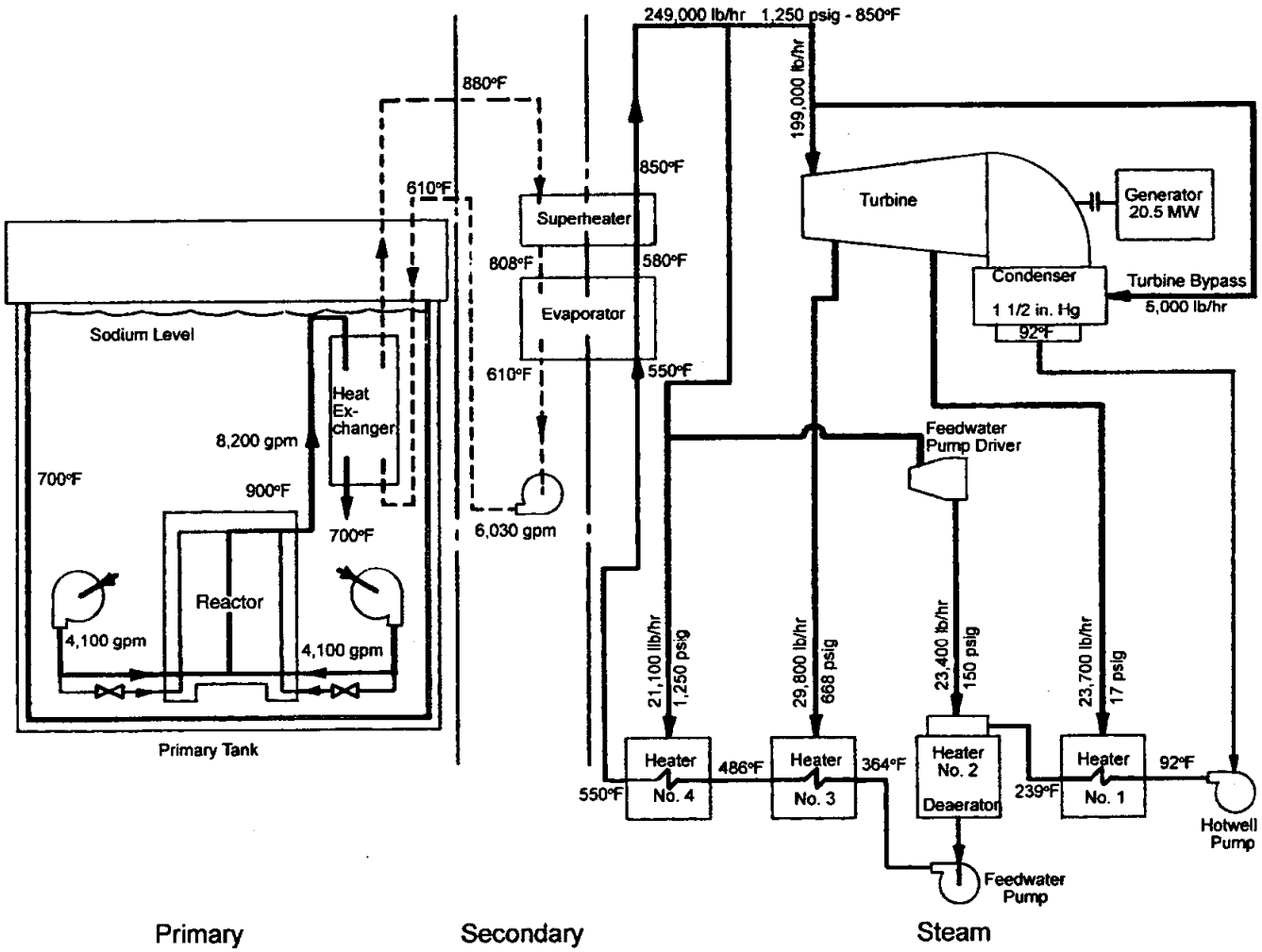
- 62.5 MWt, 20 MWe power station located at Argonne National Laboratory-West, Idaho
- Initial mission: Demonstrate fuel breeding and closed cycle operation with reprocessing of metallic fuel
  - Following mission fulfillment, shift to irradiation testing of advanced fuels
- Dry critical 1961, wet critical 1963, full power 1964, shutdown 1994
- Sodium cooled, 371°C inlet, 473°C outlet, 47 psig
- Fuel pins 0.17 in. OD, 13.5 in. core height; metal fuel in SS cladding
- First fuel processed in Fuel Cycle Facility in September 1964; recycled fuel irradiation in April 1965
- Mission oriented to irradiation testing in 1969; supporting FFTF and CRBRP oxide fuel testing and development
- Integral Fast Reactor (IFR) program began in mid 1980's
  - Testing and demonstration of high burnup metallic fuels
  - Shutdown Heat Removal Test series 1984-86; natural circulation decay heat removal and passive shutdown in ATWS events (unprotected loss-of-flow and loss-of-heat-sink)

EBR-II Site at Argonne-West

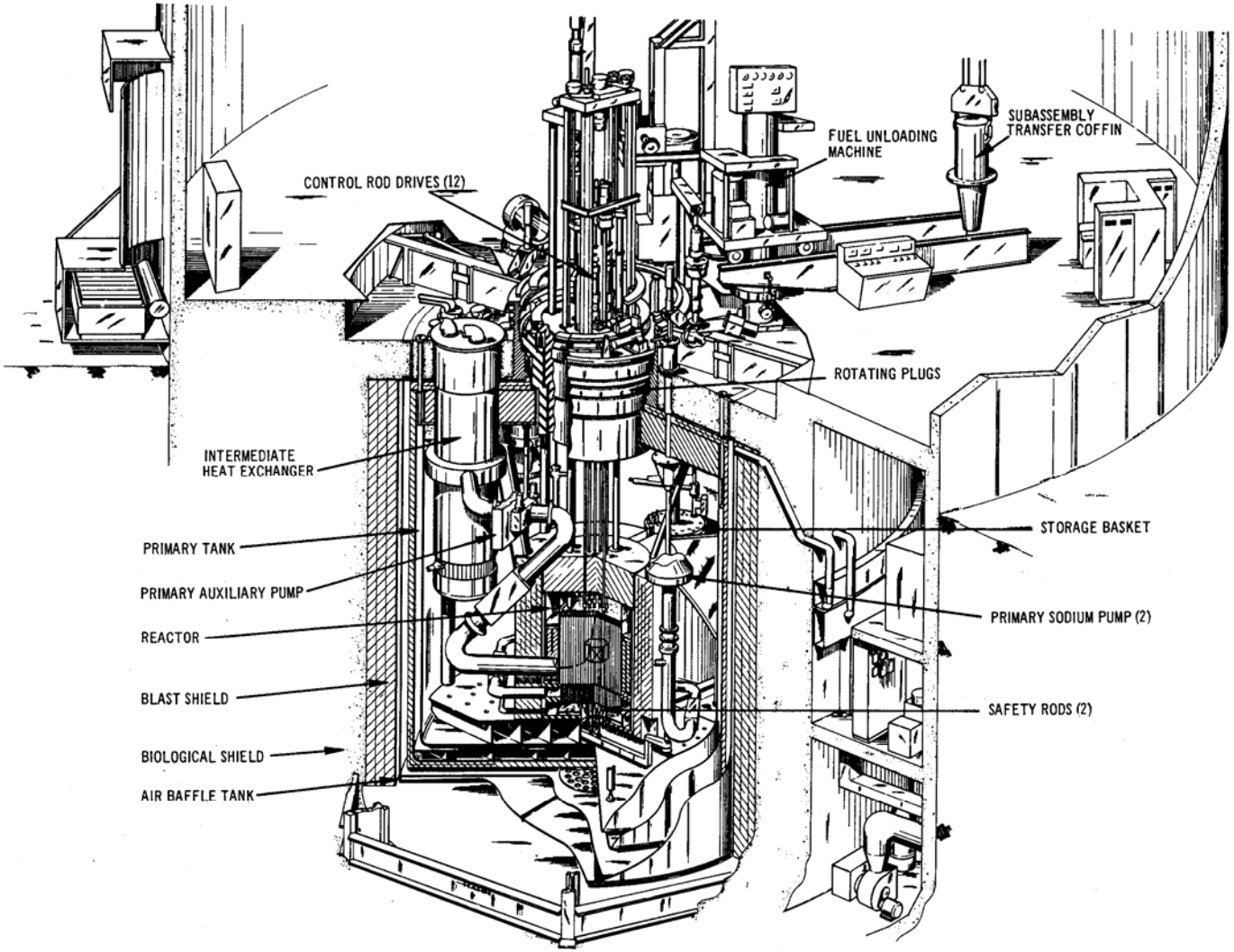
EBR-II Site at Argonne-West



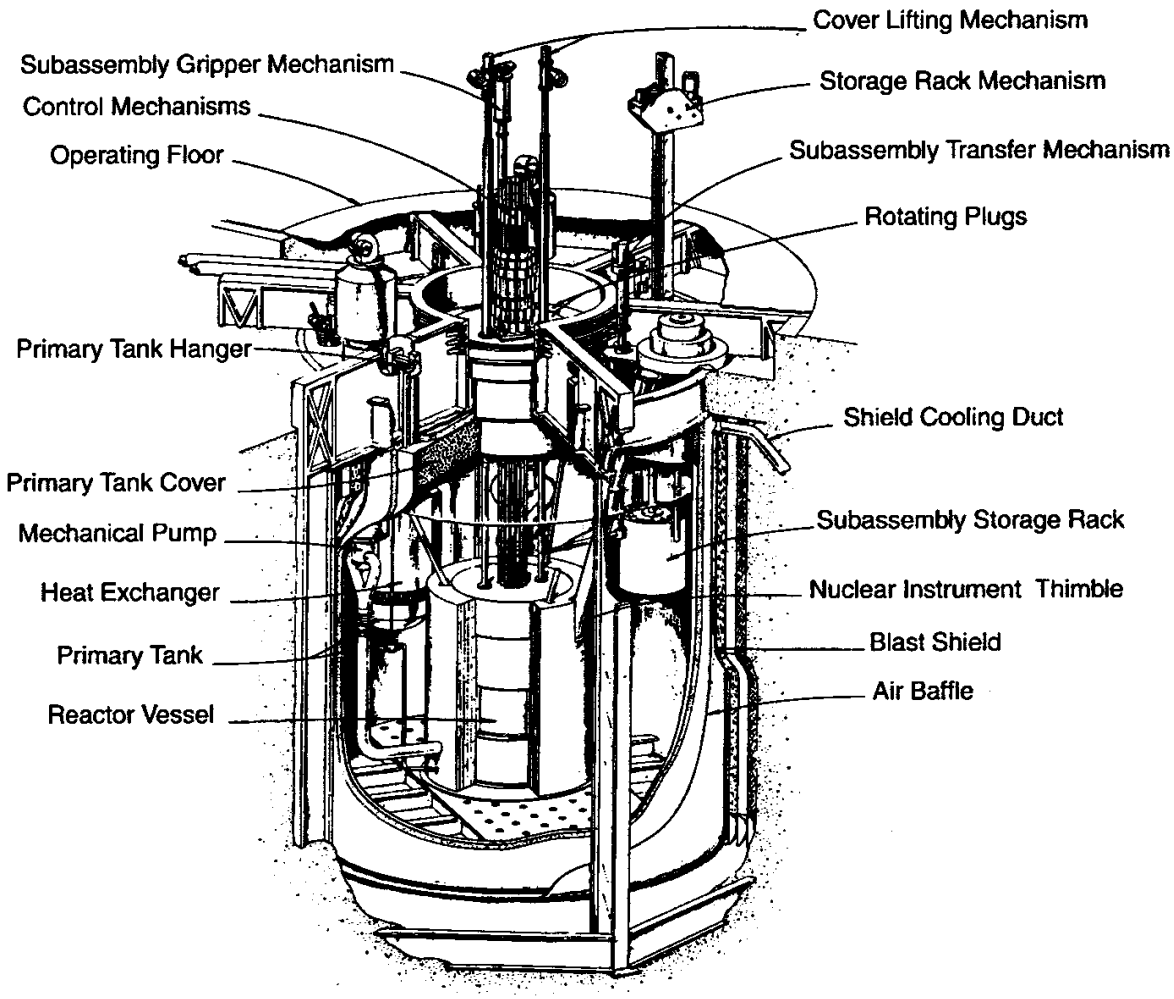
EBR-II Coolant and Power Conversion Systems



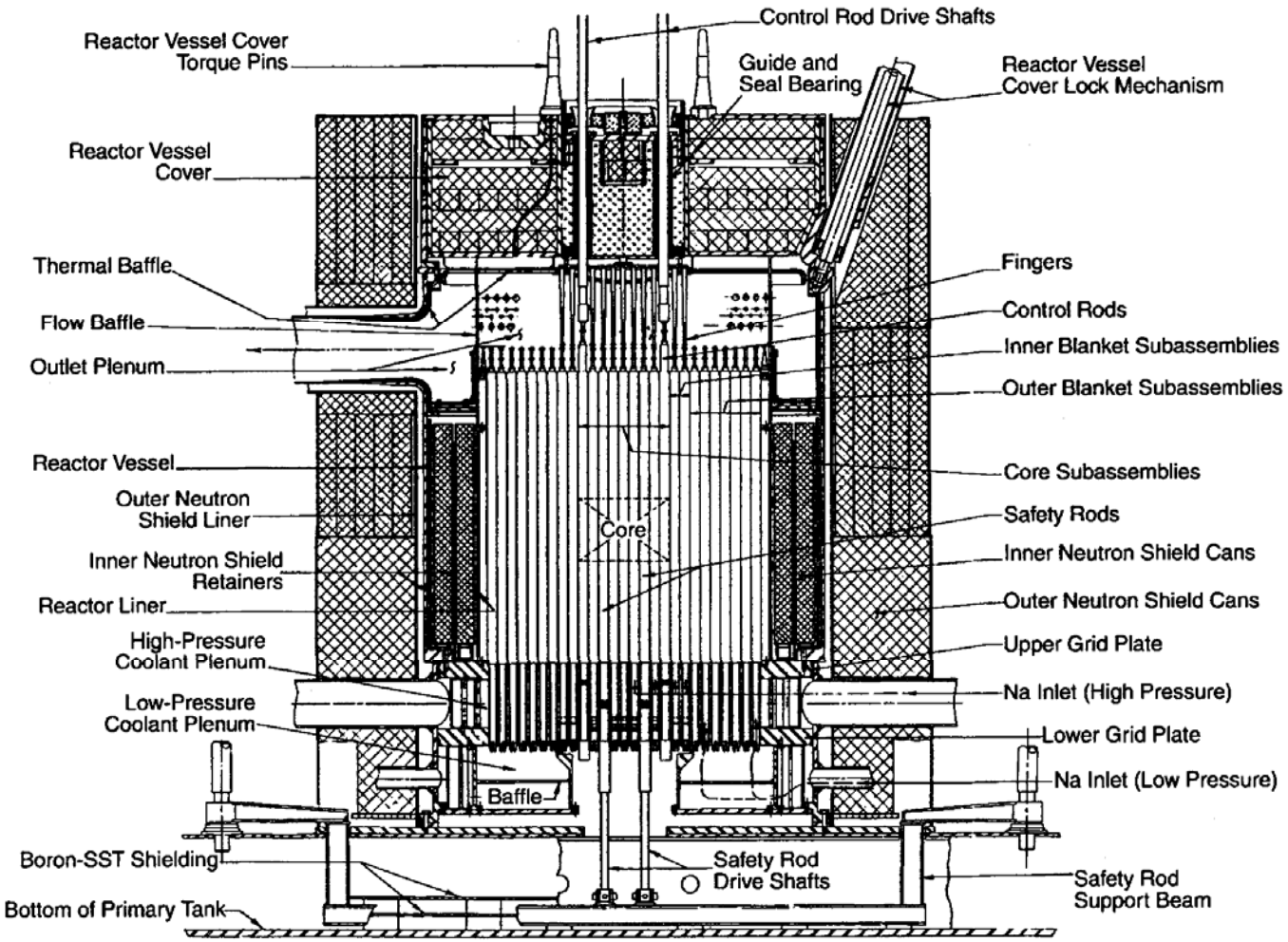
EBR-II Pool-Type Primary System



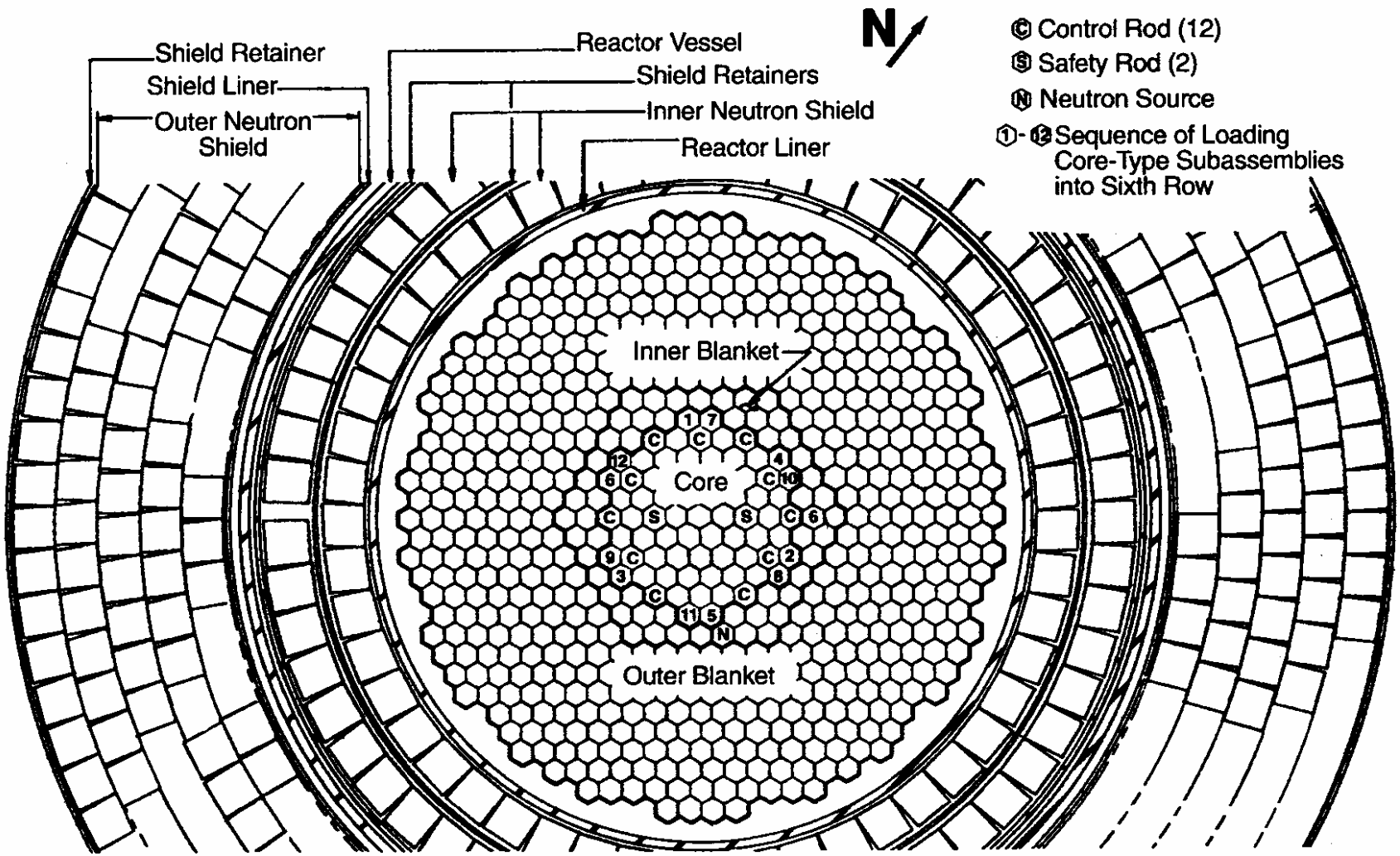
EBR-II Reactor Tank



EBR-II Reactor Vessel Assembly



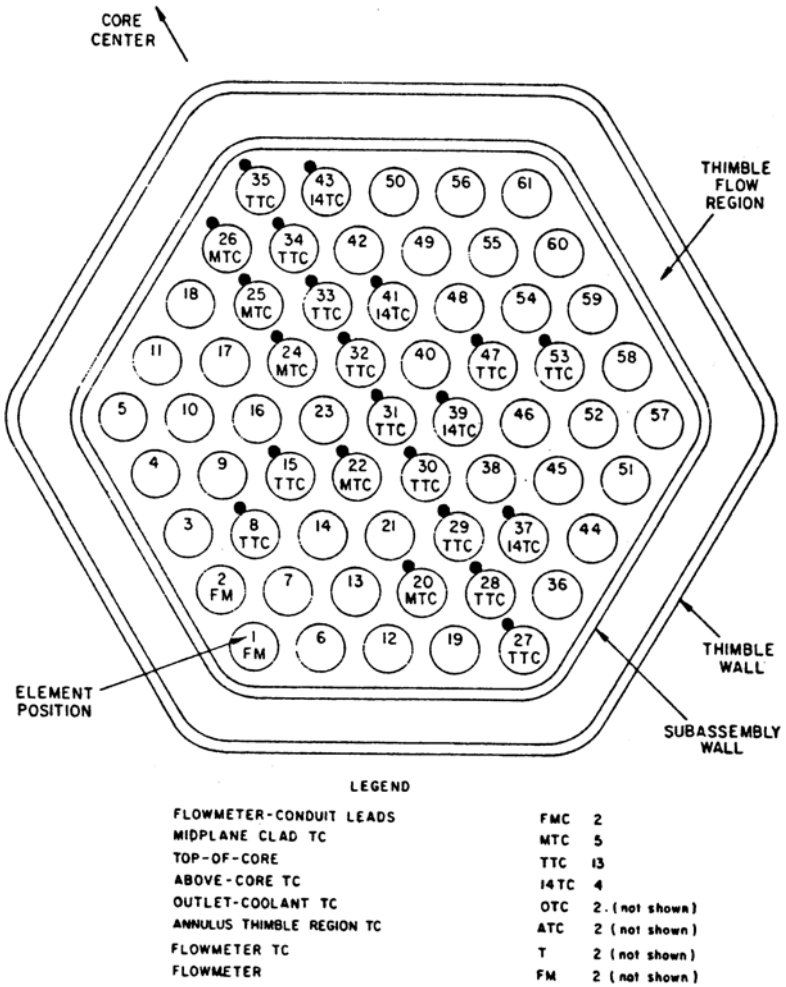
EBR-II Reactor Core Layout



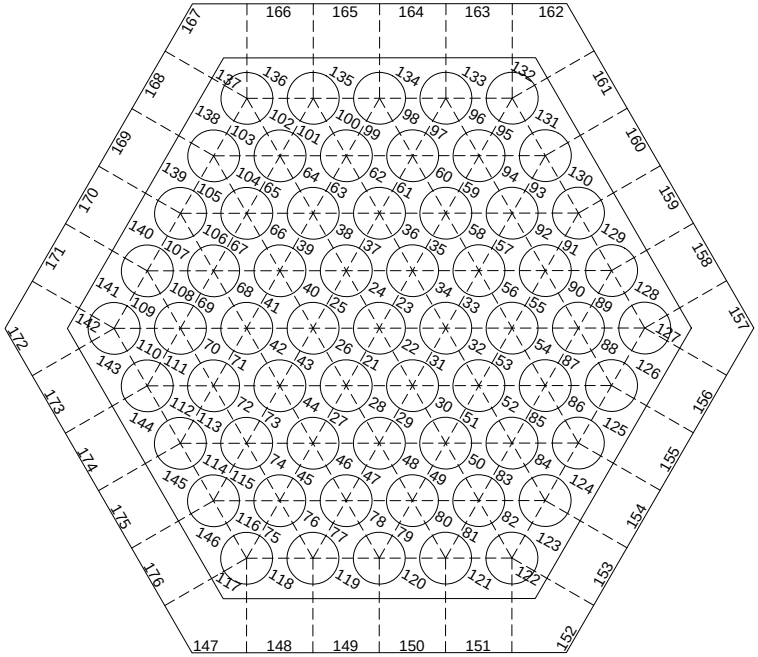
EBR-II Shutdown Heat Removal Tests (SHRT) (1984-1986)

- **A series of tests, originally intended to qualify EBR-II for continued operation, that evolved into an experimental program supporting the safety and design of advanced liquid metal reactors**
- **Testing ranged from demonstration of natural circulation decay heat removal to whole-plant simulation of unprotected (without scram) loss-of-flow (ULOF) and loss-of-heat-sink (ULOHS) accident from full power and flow**
- **Test results provided data for validation of computer codes used in design and safety analysis of advanced LMRs**
- **Instrumented subassembly XX09, equipped with calibrated thermocouples and flow meters, provided real-time measurements of coolant temperatures and flow rates during the tests**
- **The ULOF and ULOHS tests demonstrated the ability of a pool-type, metal-fueled LMR to provide self-protection in beyond-design-basis accidents**

XX09 Instrumented Subassembly and Subchannel Model

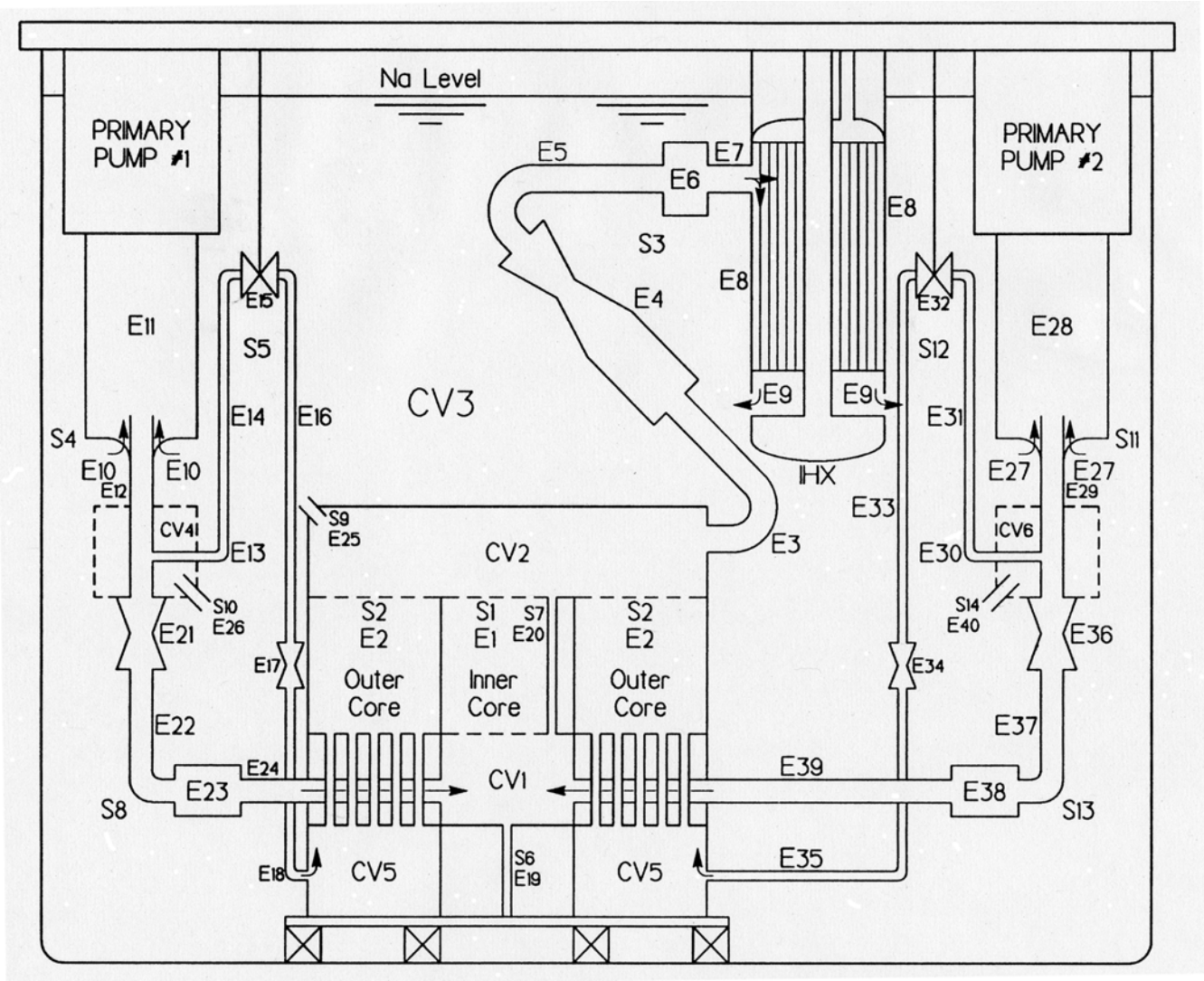


XX09 Thermocouples

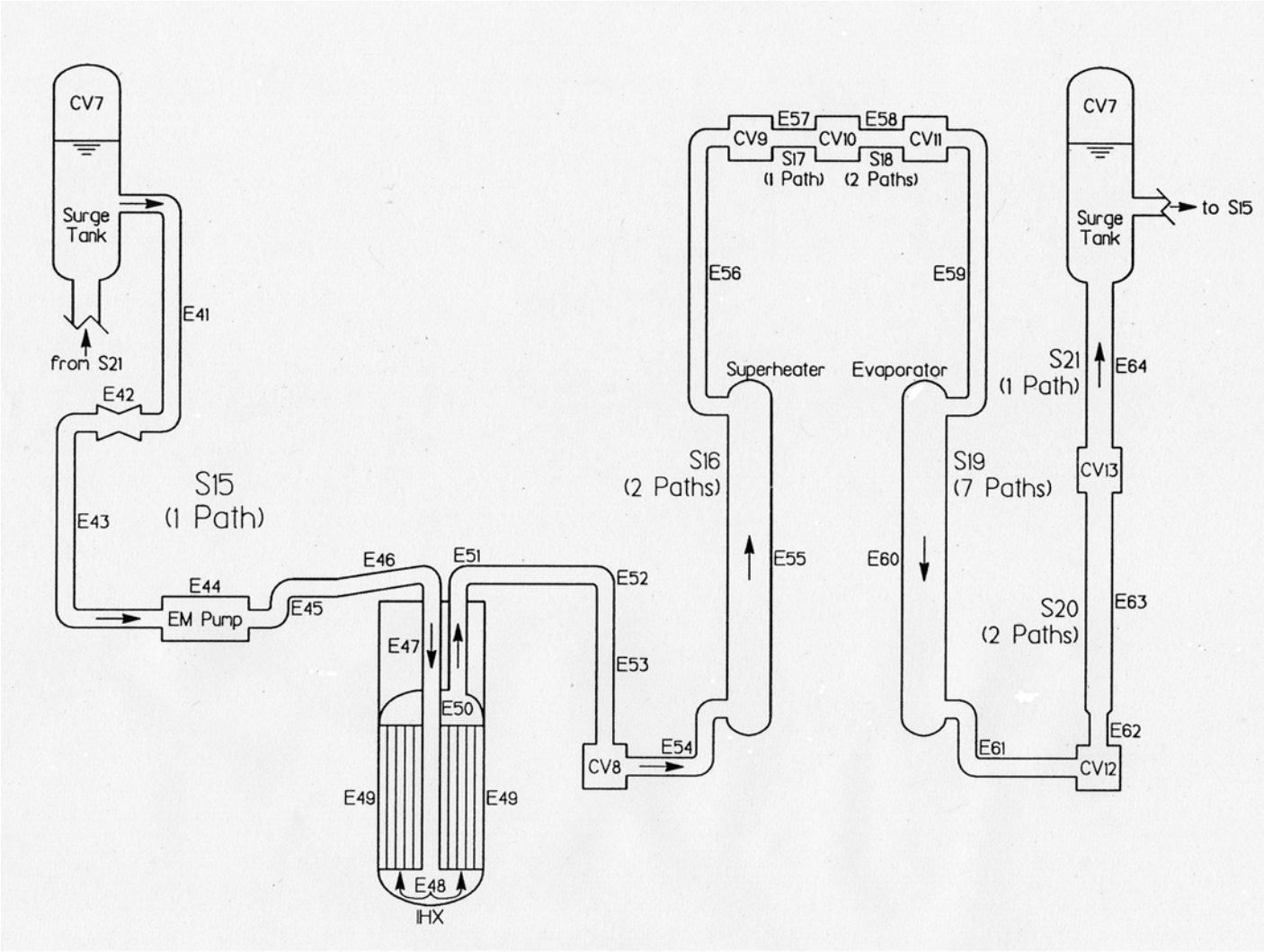


SASSYS XX09 Model

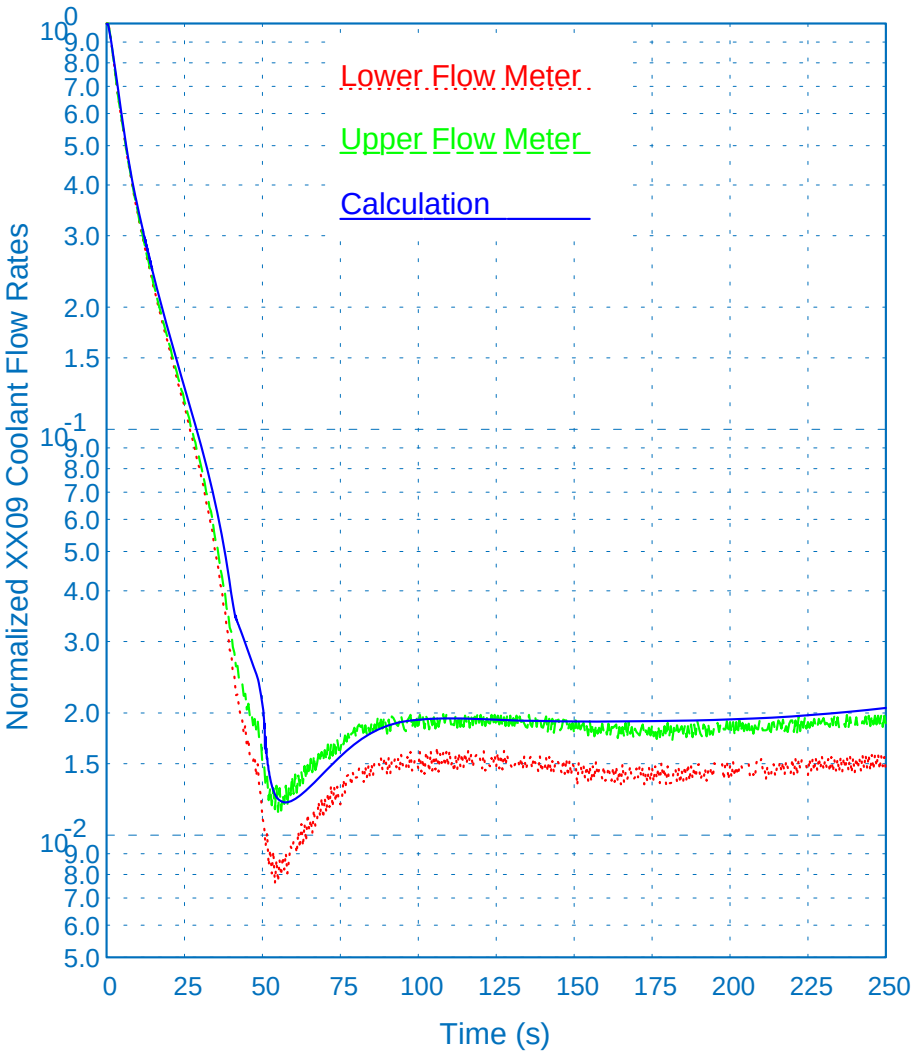
EBR-II Primary Heat Transport System SASSYS Model



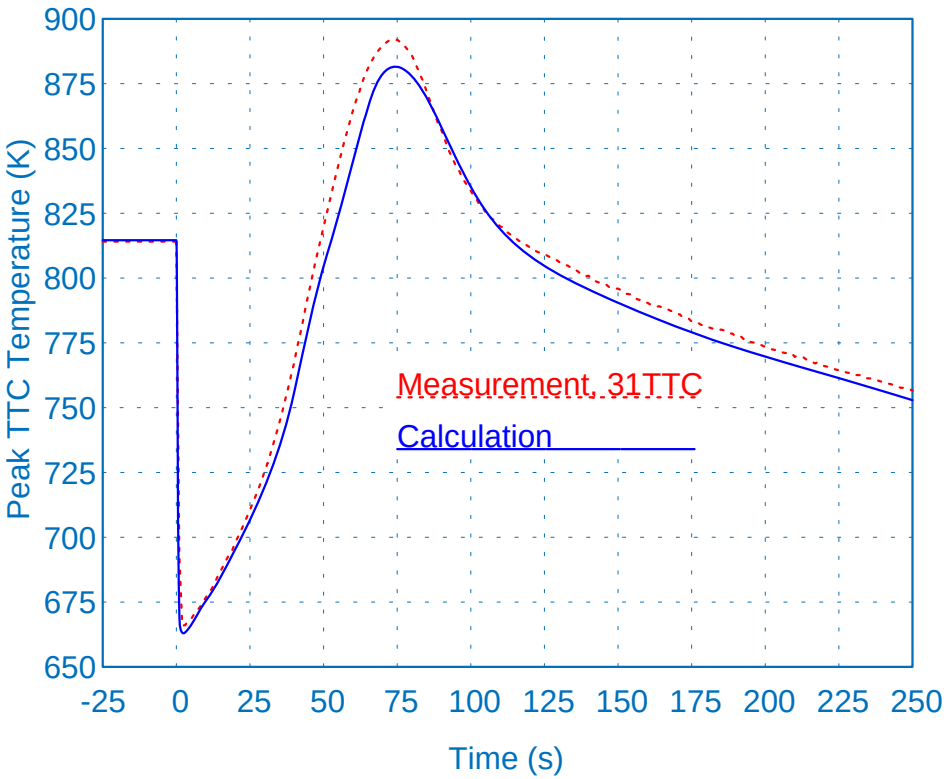
EBR-II Intermediate Heat Transport System SASSYS Model



EBR-II SHRT-17 Protected Loss-of-Flow XX09 Results SASSYS Calculation - 2006

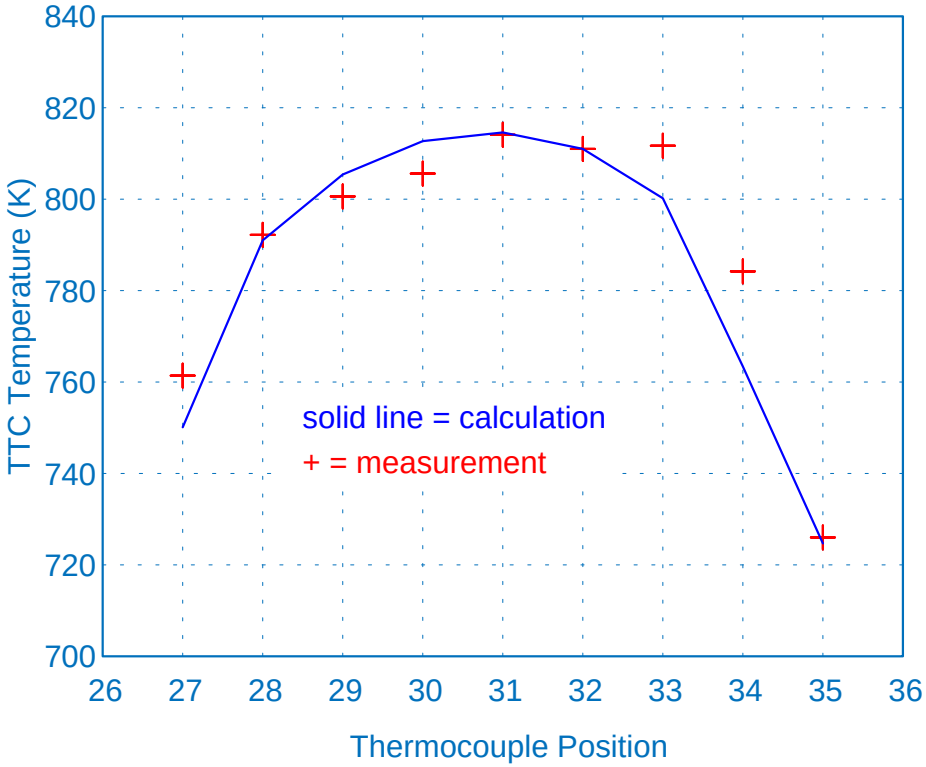


XX09 Transient Coolant Flow

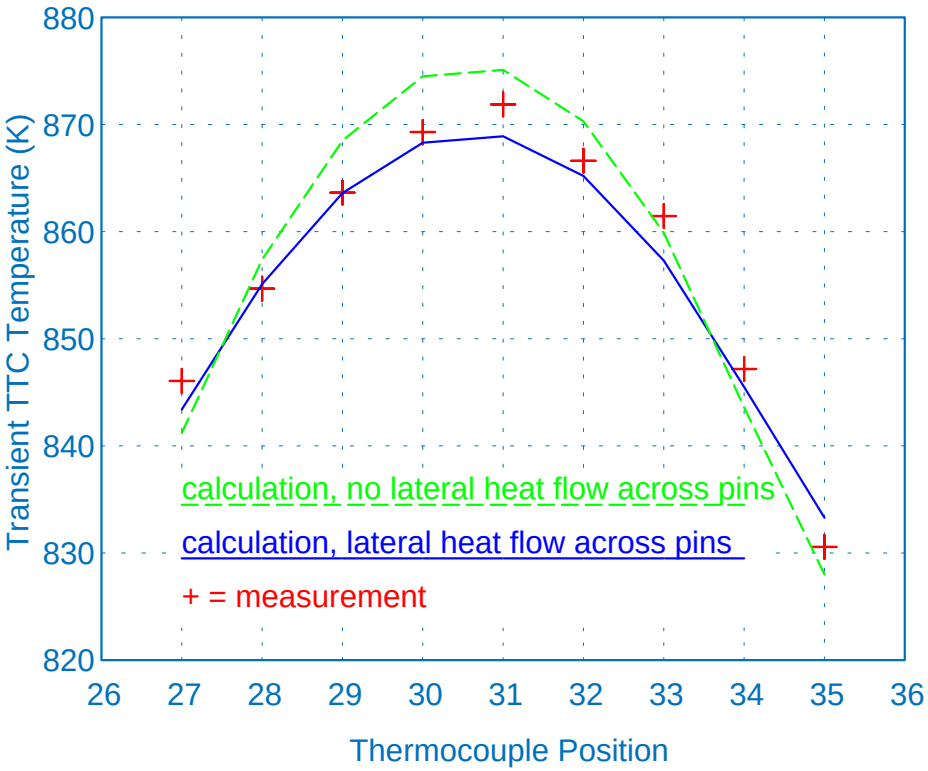


31 TTC Transient Temperatures

EBR-II SHRT-17 Protected Loss-of-Flow XX09 Temperatures SASSYS Calculation - 2006

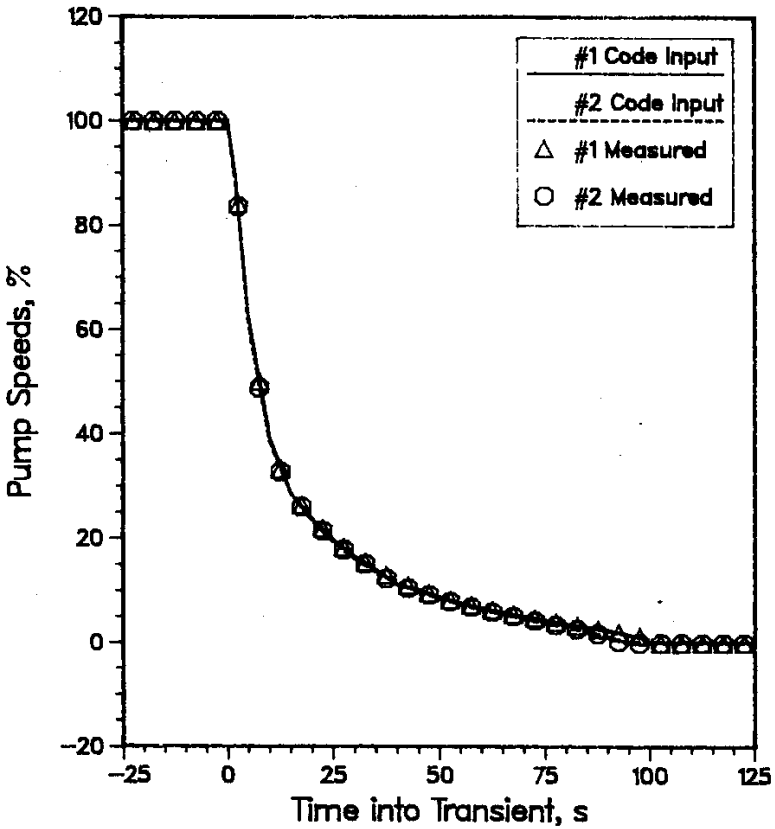


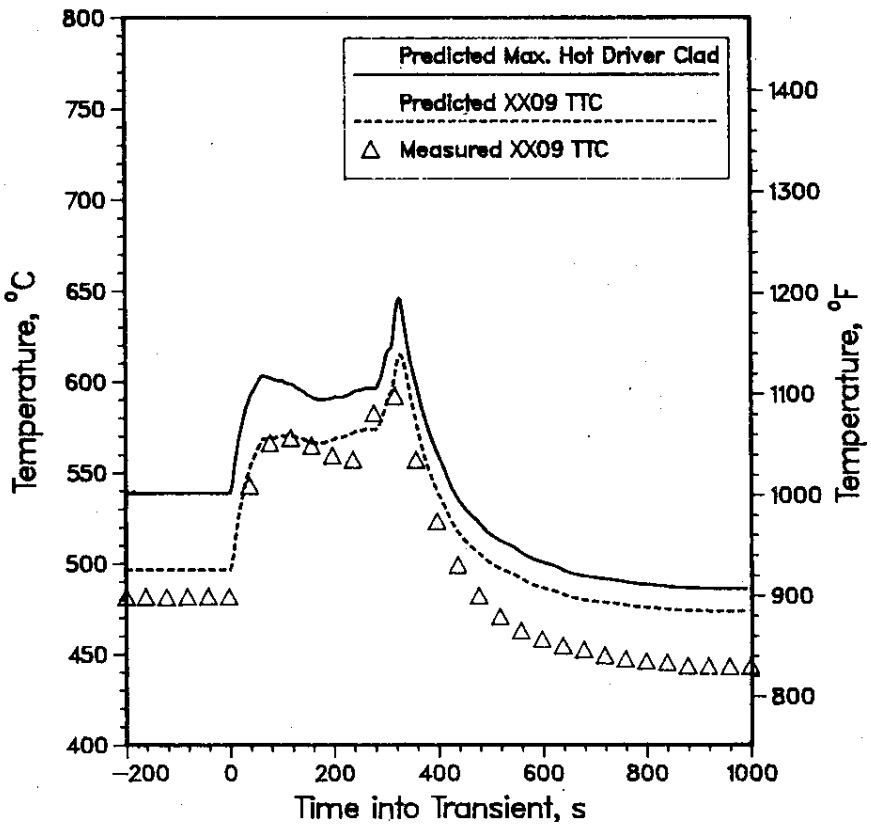
Steady State t = 0 s



Transient State t = 85 s

EBR-II SHRT 39 Unprotected Loss of Flow Sequence Results – 1
NATDEMO/HOTCHAN Calculation - 1986

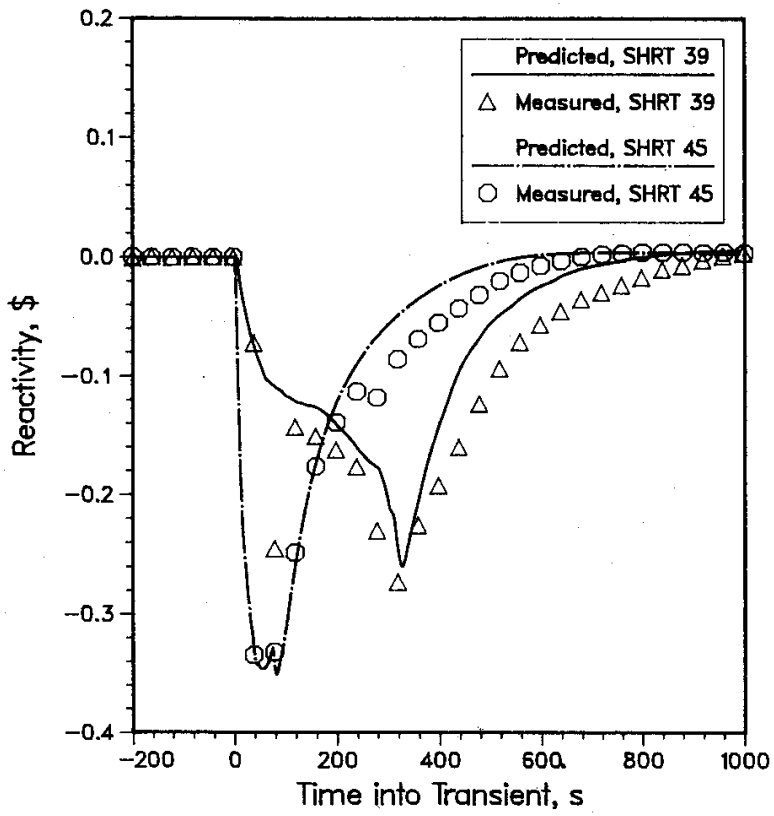


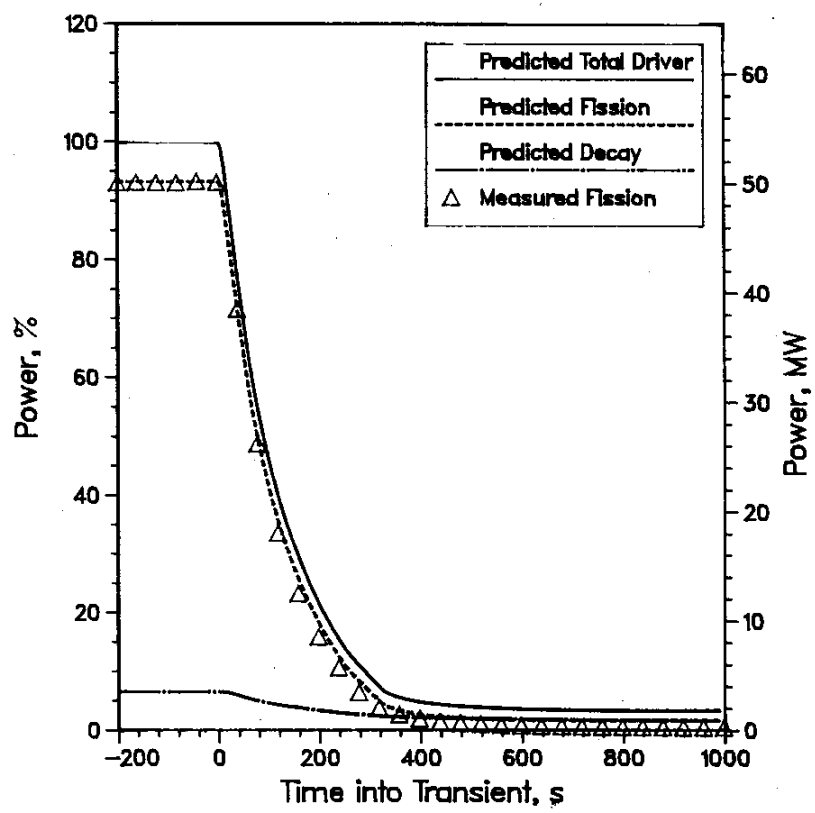


Pump coastdown without scram…

causes transient temperature rise…

EBR-II SHRT 39 Unprotected Loss of Flow Sequence Results – 2
NATDEMO/HOTCHAN Calculation - 1986

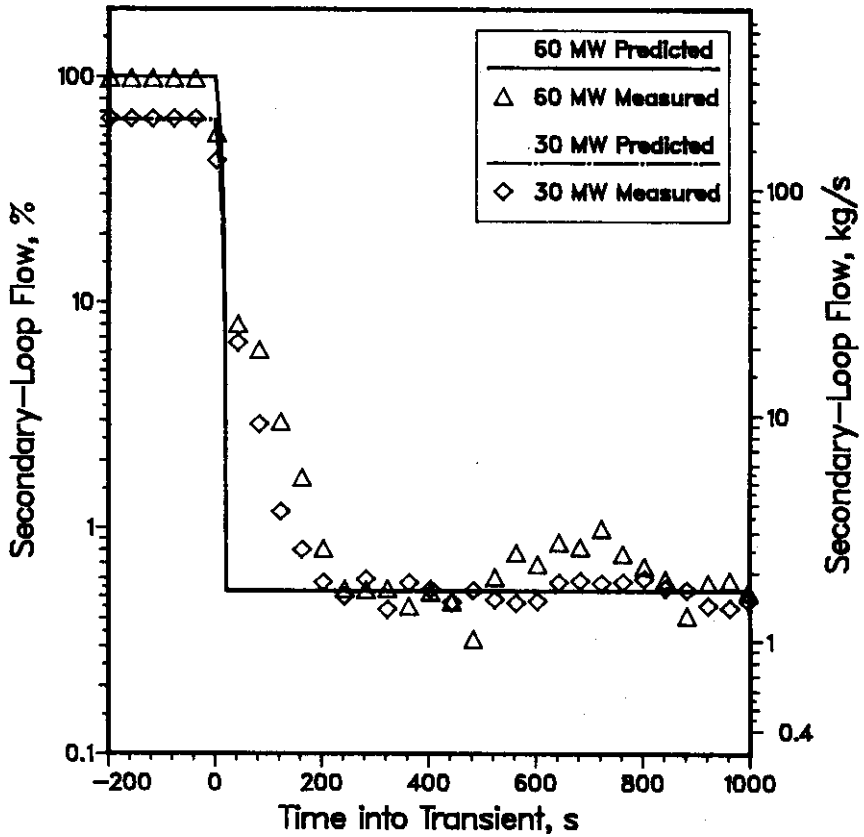


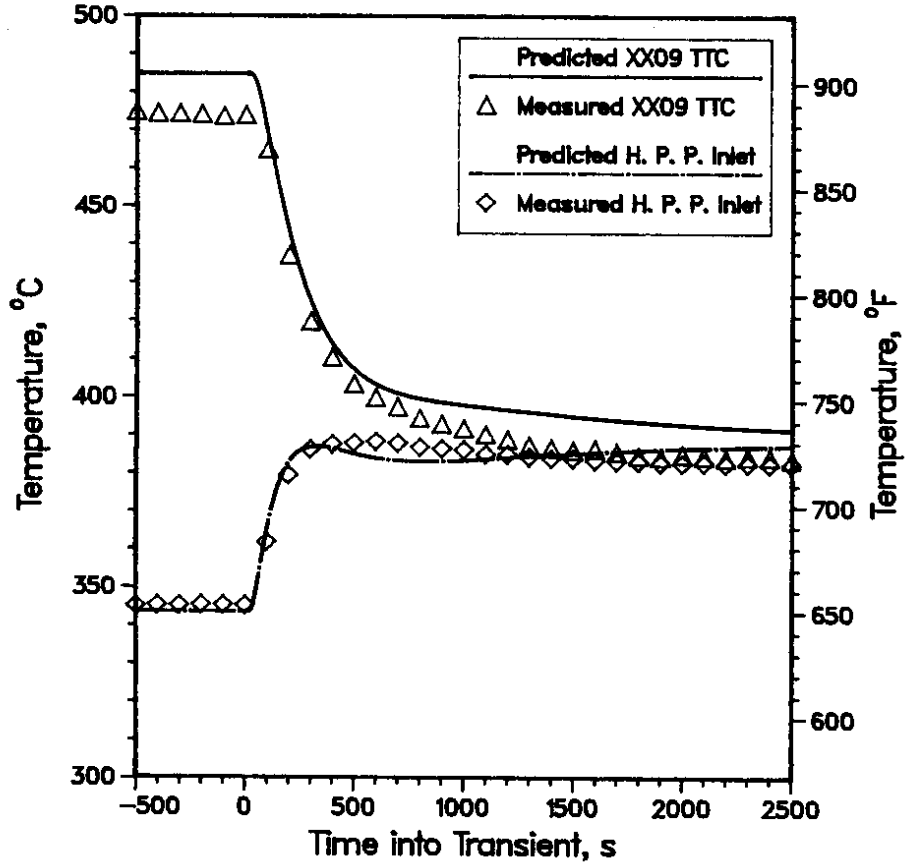


introducing negative reactivity…

reducing reactor power to decay heat.

EBR-II Unprotected Loss of Heat Sink Sequence Results – 1
NATDEMO/HOTCHAN Calculation - 1986

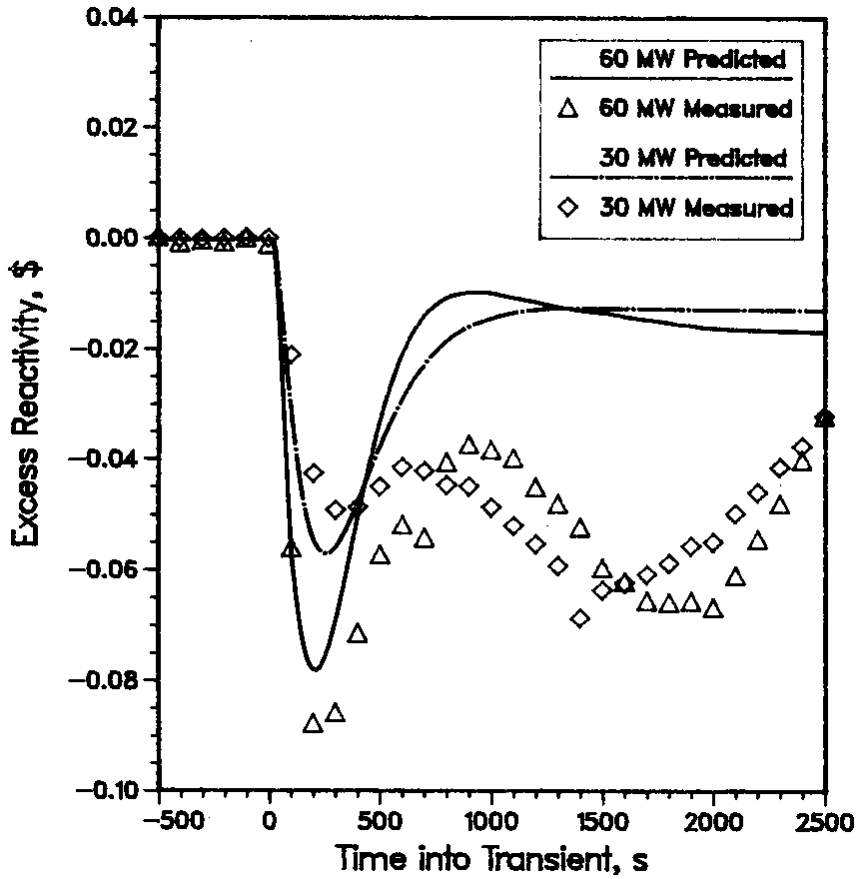


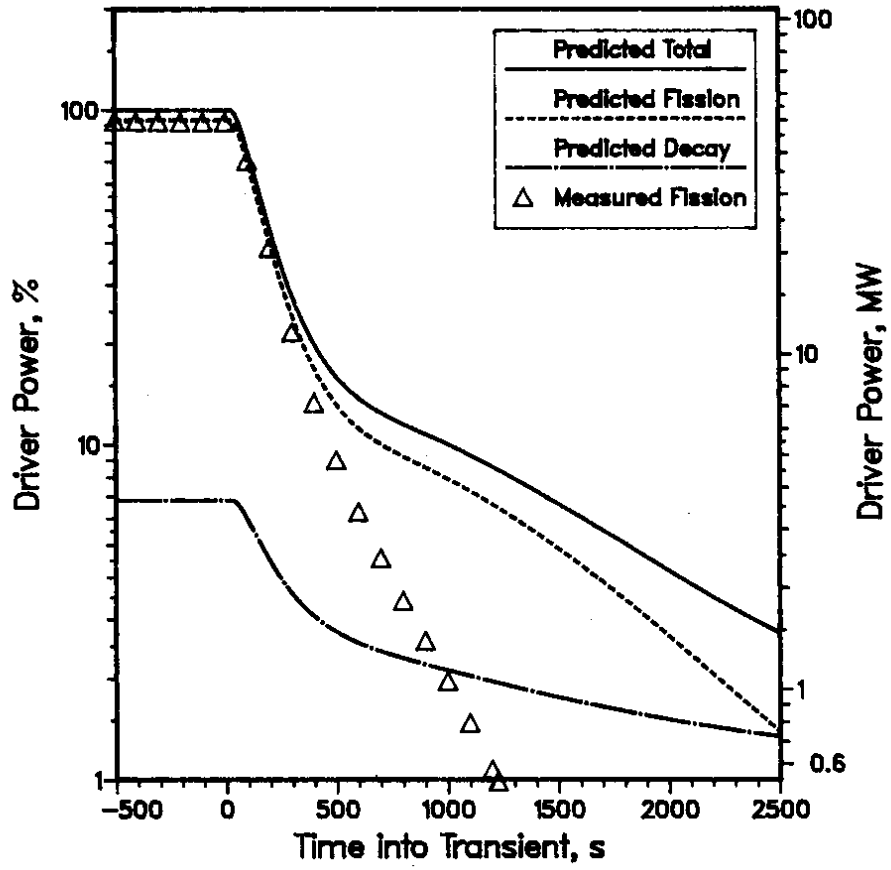


Secondary pump trip without scram…

causes transient inlet temperature rise…

EBR-II Unprotected Loss of Heat Sink Sequence Results – 1
NATDEMO/HOTCHAN Calculation - 1986





introducing negative reactivity…

reducing reactor power to decay heat.

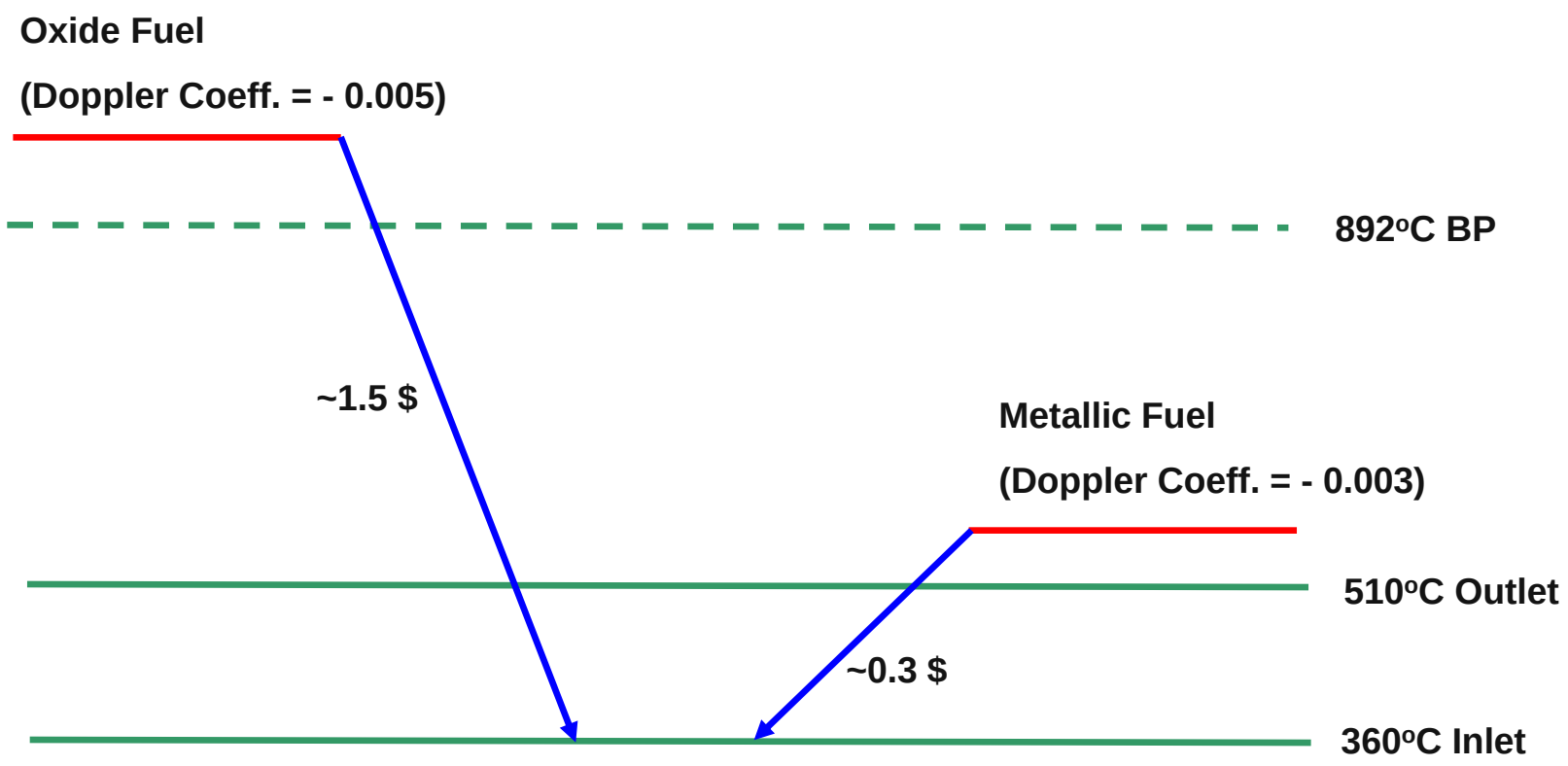
Significance of EBR-II SHRT Tests

- **The EBR-II SHRT tests demonstrate that passive safety mechanisms in sodium-cooled, metal-fueled, pool-type reactors can limit the consequences of double-fault accidents**
  - **No coolant boiling, cladding failure, or fuel melting**
  - **Reliance on inherent performance characteristics: negative reactivity feedback to reduce reactor fission power, and natural circulation decay heat removal**
- **Larger reactors can be designed to have these performance characteristics**
  - **High thermal conductivity of metallic fuel (low fuel operating temperature) reduces positive Doppler reactivity feedback upon power reduction**
  - **Pool-type design provides thermal inertia to buffer (slow down) impacts of accident initiators**
  - **Arrangement and vertical separation of heat source and sink elevations promotes natural circulation**
  - **Core restraint and support design to provide negative reactivity for increasing coolant temperature (as in FFTF and CRBRP)**

FFTF Loss-of-Flow Without Scram Tests (1986)

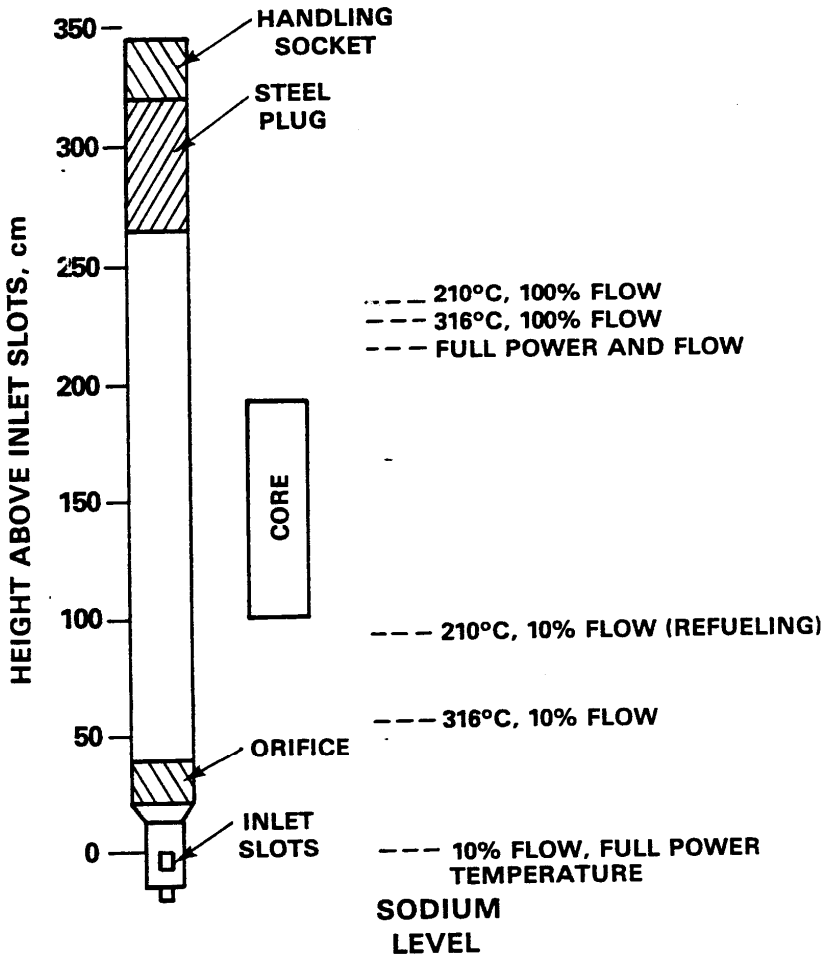
- **A series of test intended to simulate passive safety performance of an advanced LMR**
- **Unprotected (without scram) Loss-of-Flow from reduced power (range 10% to 50%) and full flow**
- **Nine Gas Expansion Modules (GEMs) were installed to provide negative reactivity during the coolant flow coastdown**
  - **Empty hexcans, sealed at the top, and installed at the core periphery. As the inlet pressure decreases, the coolant level falls, introducing void and increasing neutron leakage**
- **ULOF tests preceded by flow transients for reactivity feedbacks characterization**
  - **Pump trips with scram to natural circulation**
  - **Static measurements of GEM reactivity worth**
- **Reactor coolant flow and temperature measurements were obtained with the normal plant instrumentation (pump speed, coolant loop flowmeters, subassembly outlet thermocouples)**
- **Two fast thermocouples mounted on subassembly outlets in rows 2 and 6; (Post Irradiation Open Test Assembly – PIOTA)**

Why GEMs? Reactivity Swing for Power Reduction

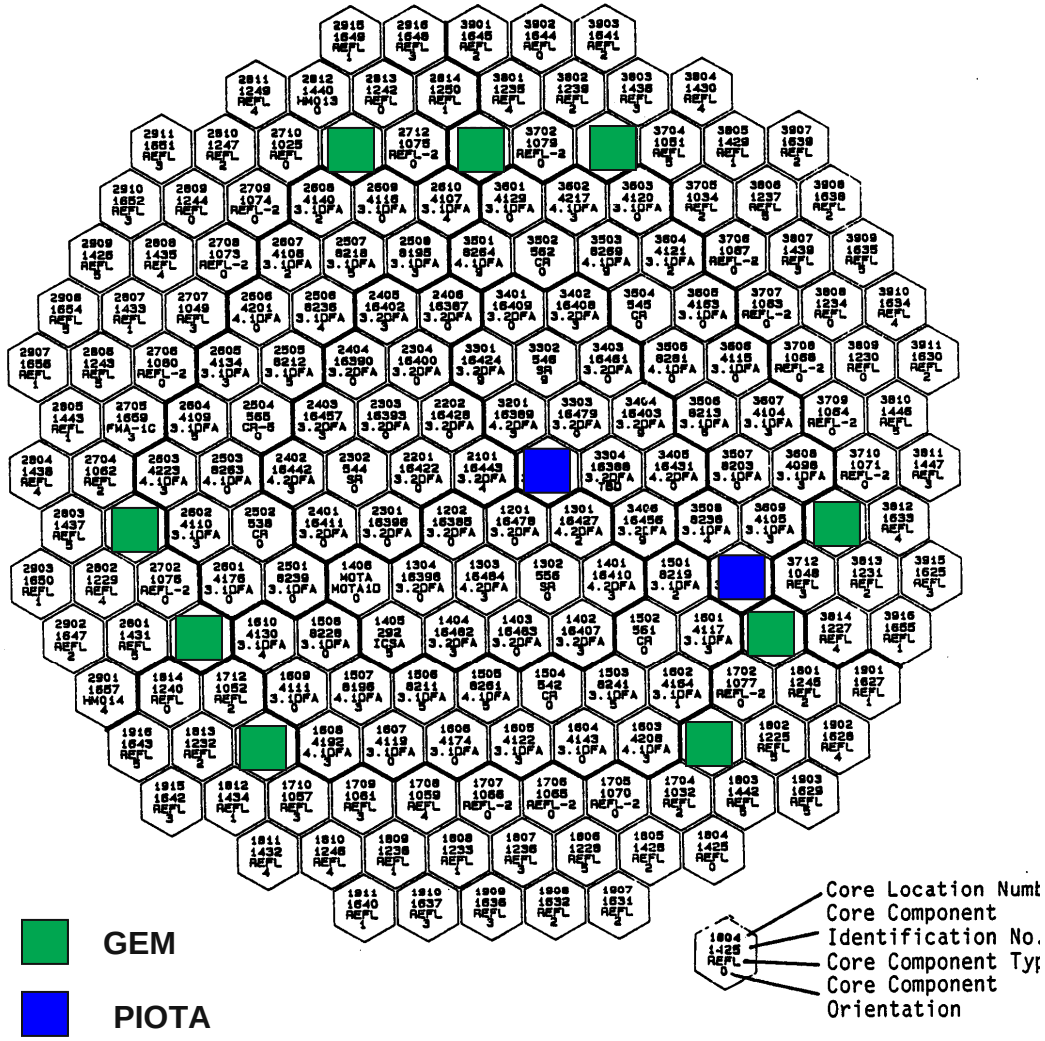


GEM Coolant Level for Power and Flow Conditions

CALCULATED GEM SODIUM LEVEL vs. PLANT CONDITIONS



FFTF Core Loading for ULOF Testing



CYCLE 8C CORE LOADING - CM.8C.1.0

FFTF PIOTA Configuration

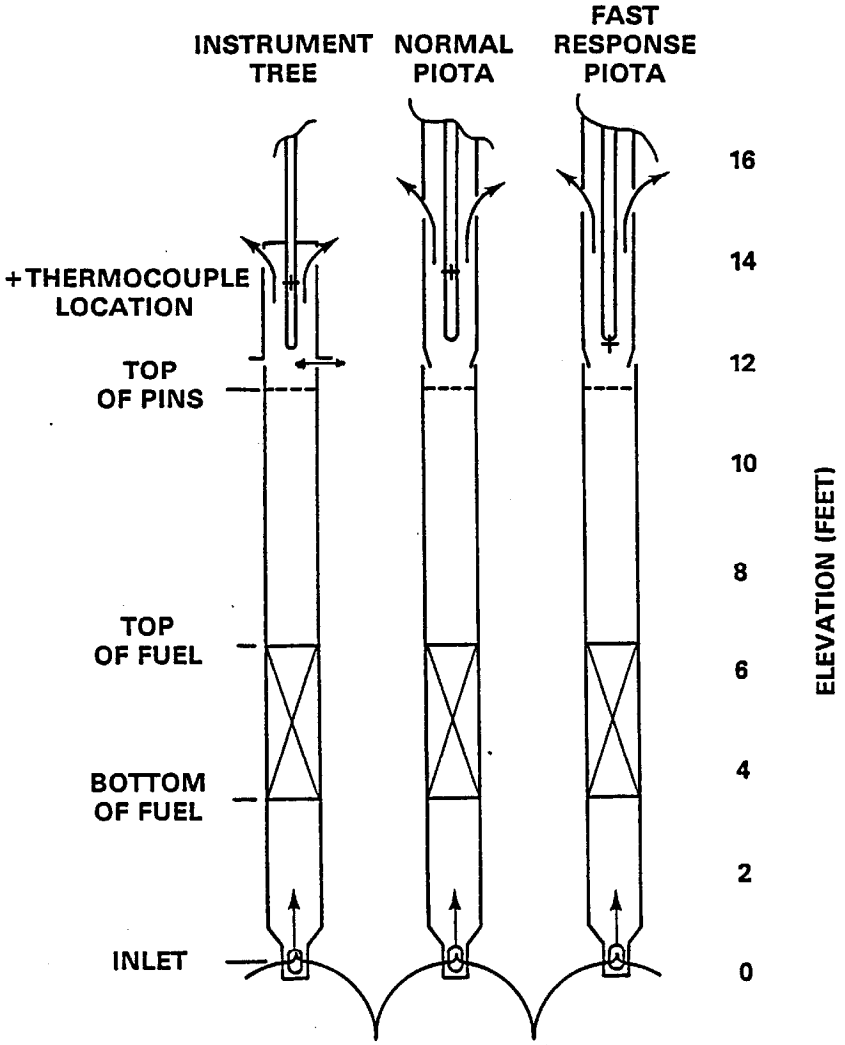


FIGURE 9

ABOVE CORE THERMOCOUPLE INSTALLATIONS

Row 2 POITA Temperatures and Calculated SASSYS Temperatures

FFTF Radial Core Expansion Reactivity

FFTF 50% Power ULOF Test Results

Comparison of Measured and Calculated Reactivity

Loss of forced reactor coolant flow without scram raises temperature, causing negative net reactivity, which reduces reactor power.

FFTF 40% & 30% Power ULOF Test Results

FFTF 20% & 10% Power ULOF Test Results

Lessons Learned from EBR-II and FFTF Testing

- **The experience of planning and executing special tests beyond the original mission of the reactor and plant can provide new insights into the capabilities of the facility**
  - **Assurance of safety performance, confirmation of margins**
- **Results of testing can contribute new knowledge that guides performance expectations and selection of design features for future reactors**
  - **Passive safety performance based on natural circulation shutdown heat removal and inherent reactivity feedbacks**
- **To maximize test value, special equipment may be necessary**
  - **Instrumentation to measure performance directly**
  - **Flow coastdown (pump power supply)**
  - **Negative reactivity feedback enhancement**

Inherent passive safety

- **Inherent passive safety characteristics of sodium fast reactor systems (wrt loss of flow without scram, etc.).**
  - **Inherent reactivity shutdown**
  - **Natural circulation decay heat removal**
- **These performance characteristics can be achieved in reactors sized and configured for commercial deployment**
- **The following examples are provided for a large, low conversion ratio conceptual design to demonstrate passive safety performance for protected and unprotected loss-of-flow accident ssequences**

SFR Safety Analysis

- **Safety analyses were carried out for a 1000 MWt pool-type concept to provide a quantitative safety assessment of reactor and plant performance.**
  - **Evaluations were made for both the metal and oxide core designs.**
- **Scope of analyses focuses on the ability of the SFR to provide inherent protection against damaging consequences following low-probability accident sequences involving multiple equipment failures.**
- **Two accident sequences were evaluated:**
  - **Protected Loss of Flow (PLOF)**
  - **Unprotected Loss of Flow (ULOF)**
- **Analyses were performed with the fast reactor safety analysis code SAS4A/SASSYS-1.**

Safety Analysis Approach

- **Protected Loss of Flow (PLOF)**
  - **Initial conditions assume full power operation at BOEC.**
  - **Total loss of normal power to the reactor cooling system at *t* = 0, with complete failure of the emergency power supply system.**
  - **Balance of plant is assumed to cease operation and provide no heat rejection capability.**
  - **Immediate reactor scram following power failure.**
- **Unprotected Loss of Flow (ULOF)**
  - **Initial and accident conditions are identical to the PLOF case.**
  - **Additionally, the reactor protection system fails to scram either the primary or secondary control rods.**
  - **Power control is exclusively through reactivity feedback mechanisms.**
- **In both cases, the only heat removal path is through the emergency heat removal system (DRACS) by natural circulation.**
- **PLOF simulations were carried out to 40,000 seconds (~11 hours)**
- **ULOF simulations sere carried out to 4,000 seconds (~1 hour)**

Metal Fuel PLOF Power and Flow History, Early Times

Metal Fuel PLOF Power and Flow History, Extended Times

Metal Fuel PLOF Temperature History, Early Times

Metal Fuel PLOF Temperature History, Extended Times

Oxide Fuel PLOF Power and Flow History, Early Times

Oxide Fuel PLOF Power and Flow History, Extended Times

Oxide Fuel PLOF Temperature History, Early Times

Oxide Fuel PLOF Temperature History, Extended Times

Metal Fuel ULOF Transient Total Power and Channel 4 Flow

Metal Fuel ULOF Transient Temperatures for Channel 4

Metal Fuel ULOF Transient Reactivity Feedback

Metal Fuel ULOF Transient Temperatures for Channel 4 (50% CDRL Feedback)

Oxide Fuel ULOF Transient Power and Channel 4 Flow

Oxide Fuel ULOF Transient Temperatures for Channel 4

Oxide Fuel ULOF Transient Reactivity Feedback

Oxide Fuel ULOF Transient Reactivity Feedback (50% CRDL Feedback, SASS Trip at 1000 K)

Oxide Fuel ULOF Transient Temperatures for Channel 4 (50% CRDL Feedback, SASS Trip at 1000 K)

Analysis Summary

- **Both metal and oxide core designs demonstrate significant safety margins to coolant boiling and fuel damage in PLOF accident sequences.**
- **The metal core design also exhibits significant safety margins in the ULOF accident sequence. This is a direct consequence of**
  - **High thermal conductivity and low operating temperature of metal fuel.**
  - **Favorable negative reactivity feedback due to thermal expansion**

However…

- **ULOF analyses for the oxide core design indicate that margins to coolant boiling may not be adequate**
  - **Inadequate margins despite significantly longer flow halving time assumed for oxide (20 seconds) compared to metal (5 seconds).**
  - **Initially high fuel temperatures result in significant positive Doppler feedback when trying to reduce temperatures.**
- **Additional enhancements, such as a self-actuating shutdown system (SASS) device, may be required for the oxide core to increase ULOF safety margins.**

End of Presentation

- **Questions?**